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April 30, 1984

United States Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Mr. George W. Knighton, Chief
Licensing Branch 3
Office of Nuclear Reactor Regulation

SUBJECT: Beaver Valley Power Station - Unit No. 2
Docket No. 50-412
Open Item/Question Response

Gentlemen:

This letter forwards responses to the issues listed below. Duquesne Light Company plans to incorporate the responses to the FSAR questions into FSAR Amendment 7. The following items are attached:

Attachment 1: Response to Outstanding Issue 76 of the Beaver Valley Power Station Unit No. 2 Draft Safety Evaluation Report.

Attachment 2: Response to Outstanding Issue 3 of the Beaver Valley Power Station Unit No. 2 Draft Safety Evaluation Report.

Attachment 3: Response to Outstanding Issue 4 of the Beaver Valley Power Station Unit No. 2 Draft Safety Evaluation Report.

Attachment 4: Response to Outstanding Issue 49 of the Beaver Valley Power Station Unit No. 2 Draft Safety Evaluation Report.

Attachment 5: Response to Question 252.1 (Outstanding Issues 50 and 51 of the Beaver Valley Power Station Unit No. 2 Draft Safety Evaluation Report.

DUQUESNE LIGHT COMPANY

R. J. Washabaugh FOR

By E. J. Woolever
E. J. Woolever
Vice President

SUBSCRIBED AND SWORN TO BEFORE ME THIS
30th DAY OF April, 1984.

Anita Elaine Reiter
Notary Public

ANITA ELAINE REITER, NOTARY PUBLIC
ROBINSON TOWNSHIP, ALLEGHENY COUNTY
MY COMMISSION EXPIRES OCTOBER 20, 1986

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KAT/wjs
Attachments

cc: Mr. H. R. Denton, Director NRR (w/a)
Mr. D. Eisenhut, Director Division of Licensing (w/a)
Mr. G. Walton, NRC Resident Inspector (w/a)
Mr. M. Lacitra, Project Manager (w/a)

COMMONWEALTH OF PENNSYLVANIA)
) SS:
COUNTY OF ALLEGHENY)

On this 30th day of April, 1984, before me, a Notary Public in and for said Commonwealth and County, personally appeared R. J. Washabaugh who being duly sworn deposed and said that (1) he is duly authorized to execute and file the foregoing Submittal on behalf of E. J. Woolever, 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.

Anita Elaine Reiter
Notary Public

ANITA ELAINE REITER, NOTARY PUBLIC
ROBINSON TOWNSHIP, ALLEGHENY COUNTY
MY COMMISSION EXPIRES OCTOBER 20, 1986

ATTACHMENT 1

Response to Outstanding Issue 76 of the
Beaver Valley Power Station Unit No. 2
Draft Safety Evaluation Report

Draft SER Section 7.7.2.2: High Energy Line Breaks and Consequential Control System Failures

IE Information Notice 79-22, issued Spetember 19, 1979, raises a concern that certain nonsafety-grade or control equipment, if subjected to the adverse environment of a high-energy line break, could malfunction and cause plant conditions to be more severe than those analyzed in FSAR chapter 15. The applicant was requested to perform a review to determine what, if any, design changes or operator actions would be necessary to ensure that high-energy line breaks will not cause control system failures to complicate the event beyond the Chapter 15 analyses.

The staff has reviewed the applicant's response, contained in FSAR Amendment 4, and finds it needs further clarification in the following areas:

1. PT 444 and 445, used for the pressurizer PORV control are not qualified. The applicant's response indicates all equipment associated with this control system is Category I.

2. The intent of NRC Question 420.4 was to require the applicant to review all possible control system malfunctions due to a high-energy line break inside or outside of containment. It appears that the applicant only reviewed the four scenarios described in IE Information Notice 79-22 and limited that review to inside containment.

This item is open pending staff review of further clarification from the applicant.

Response:

The response to these issues has been incorporated into the revised response to FSAR Question 420.4. The revised response will be incorporated into the next FSAR amendment and is included below:

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 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 Final Safety Analysis Report (FSAR). 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, and
4. Automatic rod control system.

Westinghouse has not since identified any other potential interaction scenarios with similar consequences. DLC has reviewed the specific non-safety grade systems listed in IE Information Notice 79-22 for potential interactions that could constitute a substantial safety hazard. DLC has not been able to identify such an interaction. While variations from the FSAR licensing bases have been identified, the basic conclusion of the FSAR (that these events do not constitute an undue risk to the health and safety of the public) remains untouched. DLC has also not been able to identify in its review of IE Information Notice 79-22 any other nonsafety grade equipment whose performance, when subjected to an adverse environment, could impact the protective functions performed by safety grade equipment.

Implicit in the four 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 FSAR. 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 associatd uncertainty of occurring. While no quantitative analysis has been conducted concerning the improbability of overall scenarios, the following define, for two of the scenarios identified above, the conservative assumptions already contained in the design basis event analysis reported in the FSAR and the additional conservative assumptions to be made to derive the postulated interaction scenario.

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 following information 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 interaction 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.

1. Steam Generator Power-Operated Relief Valve Control System

IE Notice 79-22 is not applicable to the BVPS-2 steam generator power operated relief valves (PORV's) 2SVS*PCV101A, 101B, 101C, and 2SVS*HCV104. These valves are Category I motor-operated modulating valves (refer to Table 3.11-1). All equipment in the control system is Category I. Power to the valves is from Class IE power sources. Accident analysis for this system is provided in Section 15.1.4.

2. Pressurizer Power-Operated Relief Valve 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 generagor 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

 1) conservative initial assumptions

  a) Appendix K decay heat model
  b) Engineered safeguards power plus calorimetric error
  c) Programmed RCS temperature plus control deadband and instrument errors
  d) initial conservative S/G inventory
  e) conservative core physics

 2) conservative accident assumptions

  a) break (all sizes) in Safety Class 2 feedline piping
  b) maximum adverse environmental errors for protective instrumentation
  c) worst single active failure (loss of any one auxiliary feed pump)
  d) operator action time

b. Additional Assumptions Required for this Scenario

 1) Break must occur inside the containment between the steam generator nozzle and the containment penetration. A break at other locations invalidates this scenario.

 2) Double ended break leads to limiting consequences. Smaller breaks permit longer operator action times.

 3) Adverse environment resulting from the break can impact the pressurizer power operated relief valve control system. The PORV's are air to open valve with the air supply being controlled by solenoid operated valves. The PORV's and SOV's are in containment. The PORV's are designed to fail closed 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 MOV's and associated control systems are qualified for post accident environmental conditions.

 4) Due to the adverse environment, the pressurizer PORV control system initiates a spurious signal to open the PORV's.

Should the control system continue to operate within specification or initate a spurious signal to close the PORV's, the scenario is invalidated.

5) 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 analysis present in WCAP-9600, ection 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 allerted to the possibility of the pressurizer PORV's failing 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.

DLC has upgraded a large portion of the Beaver Valley Unit 2 Pressurizer PORV system from the standard Westinghouse nonsafety-grade PORV. The Beaver Valley Unit 2 valves are Category I electro-solenoid-actuated valves (refer to FSAR Section 5.4.13 and Table 3.11-1, page 19). The PORV's are Class 1E with Category I power. The pressure transmitters used for normal high pressure PORV automatic actuation are not safety grade qualified components. However, no credit is taken for the high pressure automatic operation of the PORV's in the Beaver Valley Unit 2 FSAR Chapter 15 safety analyses. An analysis of a stuck-open relief valve is provided in FSAR Section 15.6.1 and it indicates that adequate protection is furnished. In addition, a safety grade interlock is supplied from three qualified pressurizer pressure bistables which will mitigate any inadvertent opening of a Pressureizer PORV. The PORV's are also used for low temperature overpressure protection (LTOP). A complete discussion on LTOP is provided in FSAR Section 5.2.2.11. The LTOP portion of the PORV system is completely safety grade.

3. 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 the FSAR. The reduced secondary liquid mass at the 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 FSAR Assumptions Concerning Feedline Break

   1) conservative initial assumptions

      a) Appendix K decay heat model
      b) Engineered safeguards power plus calorimetric error
      c) Programmed reactor coolant system (RCS) temperature plus control deadband and instrument error
      d) Initial conservative steam generator inventory
      e) conservative core physics

   2) conservative accident assumptions

      a) Break (all sizes) in Safety Class 2 feedline piping
      b) Maximum adverse environmental errors for protective instrumentation
      c) Worst single active failure (loss of any one auxiliary feed pump)
      d) Operator action time

b. Additional Assumptions Required for this Scenario

   1) Break must occur between steam generator nozzle and feedline check valve. A break at other locations invalidates this scenario.

2) Small breaks less than 0.2 square feet. Larger breaks invalidate this scenario.

3) 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 postulatd 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.

4) Due to the adverse environment, the main feedwater control system initates 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.

5) A large steam flow/feed flow mismatch will initate an immediate reactor trip.

Accident Consequences

Section 4.2 of WCAP-9600, "Report on Small Break Accidents for Westinghouse NSSS System," describes transient analyses for a postulated losss 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 assumption made in WCAP-9600, the operator continues to have at least 2,800 seconds before corrective action must be taken to inject auxiliary feedwater into the intact steam generators to prevent core uncovering. The FSAR does not 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 operations 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 decreaing 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 FSAR licensing basis is met.

4. Automatic 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 T. The potential problem is a failure in the excore neutron detectors or associated cabling, resulting in inaccurate detector output in the low direction, causing an automatic rod withdrawal accident coincident with a steamline break.

This scenario results in a lower departure from nucleate boiling ratio (DNBR) than presently presented in the FSAR.

Probability Assumptions Affecting Event Probability and Consequences

a. Standard FSAR Assumptions Concerning Steamline Break

   1) conservative initial assumptions

      a) nominal rated power plus calorimetric error
      b) programmed RCS temperature plus control deadband and instrument errors
      c) conservative end of life core physics

   2) conservative accident asumptions

      a) break (all sizes) in Safety Class 2 steamline piping
      b) maximum adverse environmental errors for protective instrumentation
      c) worst single active failure (loss of any one safety injection pump)
      d) operator action time

b. Addtional Assumptions for the Scenario

   1) Intermediate or larger steamline breaks which environmentally affect the nuclear instrumentation system. Other smaller break sizes or low power levels invalidate the scenario.

   2) Break must occur inside the containment between the steam generator nozzle and the containment penetration. A break at other locations invalidates this scenario. The physical location of the excore detectors relative to the postulated break location does not provide direct access for steam to travel to the excore detectors. The detectors are located in

an annulus around the reactor vessel that is separated by a concrete barrier from the other primary components and piping.

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

4) 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 generic bounding analysis of intermediate steamline rupture was performed for BVPS-1 to calculate the extent of fuel damage due to rod control system withdrawal prior to reactor trip (refer to the letter to Mr. Harold Denton, dated October 8, 1979, in response to IE Notice 79-22). Based upon the reduction in the radial peaking factors with burn-up an 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, which is consistent with assumptions and results stated in the FSAR. BVPS-2 has a similar reactor protection system, rod control system, and NSSS parameters. This analysis appears to also be applicable to BVPS-2.

Conclusion

Based on the low probability of the occurrnce of a consequential malfunction of the rod control system and the bounding BVPS-1 analysis, DLC does not believe that this scenario represents a significant safety questions that requires further action.

ATTACHMENT 2

Response to Outstanding Issue 3 of the
Beaver Valley Power Station Unit No. 2
Draft Safety Evaluation Report

Draft SER Section 2.4.11.2: Emergency Water Supply (excerpt)

The basis for selecting a limiting condition for operation of 654 ft. msl. has not been discussed in the FSAR. Therefore, the staff cannot conclude that this level will permit safe shutdown of the station during low flow periods in the Ohio River. The staff will thus require assurances that if the river is at this level and falling, that the plant will be in a cold shutdown condition before the river level drops below the suction level of the service water pumps.

Response:

FSAR 2.4.11.1 is being revised as shown on the following pages.

2.4.11 Low Water Considerations

2.4.11.1 Low Flow in Streams

The New Cumberland Lock and Dam maintains the New Cumberland Pool at el 664.5 feet. Records indicate that this elevation can be maintained at flows up to 20,000 cfs as shown on Figure 2.4-15.

A low-flow frequency curve for the Ohio River at Shippingport is shown on Figure 2.4-16. This curve represents the lowest continuous 7-day mean flows that would occur. It is based on a statistical analysis of historical flows for the past 44 years (1929-1973) modified by the present reservoir system (U.S. Army Corps of Engineers, Pittsburgh District 1970). An instantaneous flow could be lower, but with the large impoundments behind the storage dams, the 7-day flow could be provided continuously by temporarily drawing on the river storage when needed.

Computerized models developed by the U.S. Army Corps of Engineers were used to simulate regulated stream flows in the Ohio River. Results of the analysis show that a minimum flow of 4,000 cfs would have occurred at the site during the record drought of 1930 with the contemporary reservoir system. A complete failure of the nearest downstream dam (the New Cumberland Dam) during minimum flow would result in a minimum water surface elevation at the site of 648.6 feet msl (U.S. Army Corps of Engineers, Pittsburgh District 1969, 1973). This is discussed in more detail in Section 2.3.4 of the BVPS-2 PSAR.

INSERT A

The USNRC, in its review of the BVPS-2 PSAR, indicated that, by extrapolating an unregulated low-flow frequency for drought conditions which may be characterized as the most severe reasonably possible at the plant site, an instantaneous low flow of 800 cfs could occur.

A Technical Specification, as described in Section 2.4.14, will be established for low-flow/low level in the Ohio River so that safe shutdown will be accomplished while an adequate water supply is available.

2.4.11.2 Low Water Resulting from Surges, Seiches, or Tsunami

Because the site is not located on the coast or on a lakeshore, this section is not applicable to BVPS-2.

2.4.11.3 Historical Low Water

The lowest flow of record occurred during the extreme drought of 1930. A minimum of 1,250 cfs flowed past Shippingport in August of that year. Since that time, eight reservoirs with low flow augmentation capabilities have been constructed. The lowest flow that would have occurred in 1930 with the contemporary reservoir system is 4,000 cfs.

Insert A

A limiting condition for operation of 654 ft. msl. is the minimum operating level of the Auxiliary Intake Structure. The backup service water pumps located in the auxiliary intake structure will not have sufficient water available at the pump suctions at river levels below 654 ft. msl. Below elevation 654' msl. the service water pumps located in the main intake structure will be the only source available to supply service water to the station. These service water pumps are designed to supply water to the station at river levels down to 648.6 ft. msl. The minimum design water level of 648.6 ft. msl. is the river level which would occur only if the nearest downstream dam failed concurrent with a river flow of 4,000 cfs. equal to draught of record. However, the Corps of Engineers has maintained that the nearest downstream dam is considered safe against earthquakes (U.S. Army Corps of Engineers 1968, 1969). Therefore, the river level will not drop below the suction level of the service water pumps.

References: U.S. Army Corp of Engineers 1968, letter from Wayne S. Nichols, Colonel, Pittsburgh District Engineer, to R. McAllister, Duquesne Light Company, dated December 16, 1968.

U.S. Army Corps of Engineers 1969, letter from Wayne S. Nichols, Colonel, Pittsburgh District Engineer, to R. Kitchell, Stone and Webster Engineering Corporation, dated August 26, 1969.

ATTACHMENT 3

Response to Outstanding Issue 4 of the
Beaver Valley Power Station Unit No. 2
Draft Safety Evaluation Report

Draft SER Section 2.4.11.2: Emergency Water Supply (excerpt)

Because the intake structure is located on the bank of the Ohio River, it is expected that suspended sediment in the river will accumulate in the lower part of the structure. The applicant states that silt accumulation in the intake structure will be monitored semiannually and that silt exceeding a 15-inch allowable limit will be removed by a pumping operation. The applicant has not provided the design basis for an allowable silt accumulation level of 15 inches nor assurances that a semiannual inspection interval is frequent enough to preclude sediment from accumulating to a level that could affect the capability of the service water pumps in the intake structure to obtain an adequate amount of cooling water for safety-related purposes. The staff has submitted questions to the applicant and will determine the adequacy of the silt monitoring program when responses are received from the applicant.

Response:

The requested additional information on silt accumulation and removal is contained in the response to FSAR Question 240.09 (Amendment 6).

ATTACHMENT 4

Response to Outstanding Issue 49 of the
Beaver Valley Power Station Unit No. 2
Draft Safety Evaluation Report

Draft SER Sections 5.3.1 and 5.3.3: Reactor Vessel Materials and Reactor Vessel Integrity (excerpt)

The applicant has not reported the Charpy V-notch energy and mils lateral expansion data versus temperature for each reactor vessel beltline material.

Response:

The Charpy V-notch energy and mils lateral expansion data versus temperature for each reactor beltline material is provided in the following tables.

BEAVER VALLEY UNIT 2 REACTOR VESSEL CORE BELTLINE REGION

TOUGHNESS PROPERTIES

Inter. and Lower Shell Longitudinal Weld Seams

and Girth Seam

Weld Code No. G1.42

Temperature (°F)	Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
-80	7	1	0
-80	5	1	0
-80	7	2	0
-40	21	12	5
-40	17	8	0
-40	28	19	10
0	28	22	10
0	46	32	20
0	30	22	10
30	93	62	50
30	82	49	40
30	110	74	60
100	146	87	100
100	117	78	80
100	138	84	90
160	145	85	100
160	134	83	100
160	127	81	95
212	140	83	100
212	148	82	100
212	146	83	100

T_{NDT} = -30°F

RT_{NDT} = -30°F

BEAVER VALLEY UNIT 2 REACTOR VESSEL CORE BELTLINE REGION TOUGHNESS PROPERTIES

Lower Shell Course

Plate B9005-1

Temp. (°F)	Energy (ft-lb)	Lat. Exp. (mils)	Shear (%)
-40	10	6	0
-40	11	7	0
-40	15	10	0
10	18	14	10
10	28	21	15
10	21	17	10
40	29	23	15
40	38	28	20
40	30	26	15
74	54	42	40
74	41	32	30
74	49	37	35
100	65	51	60
100	76	57	70
100	57	45	60
160	85	65	100
160	82	63	100
160	80	60	100

T_{NDT} = -50°F

RT_{NDT} = 28°F

Plate B9005-2

Temp. (°F)	Energy (ft-lb)	Lat. Exp. (mils)	Shear (%)
-40	8	4	0
-40	8	4	0
-40	7	3	0
10	22	18	10
10	24	20	10
10	28	23	15
40	34	26	20
40	33	27	20
40	37	29	20
74	40	33	30
74	48	39	35
74	52	41	40
100	60	49	60
100	59	47	60
100	55	45	60
160	77	61	100
160	75	62	100
160	81	65	100

T_{NDT} = -40°F

RT_{NDT} = 33°F

BEAVER VALLEY UNIT 2 REACTOR VESSEL CORE BELTLINE REGION TOUGHNESS PROPERTIES

Intermediate Shell Course

Plate B9004-1

Temp. (°F)	Energy (ft-lb)	Lat. Exp. (mils)	Shear (%)
-40	5	5	0
-40	6	8	0
-40	7	7	0
10	24	19	10
10	20	18	10
10	19	16	10
40	22	21	20
40	26	23	20
40	24	20	20
100	43	37	30
100	51	42	30
100	45	40	30
110	48	39	25
110	50	40	30
110	48	38	25
120	59	46	40
120	53	40	30
120	62	50	40
160	73	61	80
160	70	58	80
160	71	59	80
212	82	64	100
212	85	67	100
212	83	65	100

T_{NDT} 0°F

Plate B9004-2

Temp. (°F)	Energy (ft-lb)	Lat. Exp. (mils)	Shear (%)
-40	8	5	0
-40	10	7	0
-40	10	6	0
10	15	13	5
10	16	15	5
10	18	16	5
40	24	19	10
40	26	20	10
40	23	17	10
75	39	29	25
75	38	30	25
75	40	32	25
100	50	40	35
100	51	40	35
100	54	43	35
160	69	57	90
160	68	55	90
160	70	60	95
212	70	55	100
212	78	59	100
212	79	61	100

T_{NDT} = -10°F

ATTACHMENT 5

Response to Question 252.1 and
Outstanding Issues 50 and 51 of the
Beaver Valley Power Station Unit No. 2
Draft Safety Evaluation Report

Draft SER Sections 5.3.1 and 5.3.3 (OI 50): Reactor Vessel Materials and Reactor Vessel Integrity (excerpt)

The applicant has not identified the azimuthal location and lead factors for each surveillance capsule.

Response:

The azimuthal location and lead factors for each surveillance capsule is provided in the response to Question 252.1, Table 1, "Surveillance Capsule Removal Schedule."

Draft SER Section 5.3.2 (OI 51): Pressure Temperature Limits (excerpt)

The applicant has not supplied pressure-temperature limit curves for the reactor pressure vessel, which comply with the beltline and closure flange requirements of Appendix G, 10CFR50.

The applicant has not reported the amount of nickel for each beltline material.

Response:

The pressure temperature limit curves for hydrostatic pressure and leak tests, heatup, cooldown, and core operations is provided in the response to Question 252.1, Item C-1.

The amount of nickel for each beltline material is provided in the response to Question 252.1, Table 3, "Fracture Toughness Properties of the Reactor Vessel."

Request for Additional Information
Beaver Valley Unit 2

Materials Application Section
Materials Engineering Branch

252.1 Appendices G and H, 10 CFR Part 50 were revised in the Federal Register on May 27, 1983 and became effective on July 26, 1983.

a. Identify ferritic reactor coolant pressure boundary materials that do not comply with the fracture toughness requirements of Section 50.55a and Appendices G and H of 10 CFR Part 50.

b. For materials which cannot meet the fracture toughness requirements of Section 50.55a and Appendices G and H of 10 CFR Part 50, provide alternative fracture toughness data and analyses to demonstrate their equivalence to the requirements of 10 CFR Part 50.

c. To demonstrate conformance to Appendices G and H, 10 CFR Part 50:

(1) Provide pressure temperature limit curves for hydrostatic pressure and leak tests, heat-up, cooldown and core operations.

(2) Identify the withdrawal schedule, lead factor, test samples and materials in the Reactor Vessel Materials Surveillance Program.

(3) Indicate the reference temperature, RT_{NDT}, for materials in the reactor vessel closure flange region and the beltline regions.

(4) Indicate the chemical composition (copper, nickel and phosphorus), unirradiated upper-shelf energy, and projected end-of-life RT_{NDT} and upper-shelf energy for all beltline materials. RT_{NDT} projections are to be estimated using the "Guthrie Formula" in Commission Report SECY-82-465. Upper-shelf energy projects are to be estimated using Regulatory Guide 1.99, Rev. 1. These projects are to be for the end-of-life neutron fluence at the 1/4T and ID reactor vessel locations.

Response

252.1(a) All the Beaver Valley Unit 2 ferritic reactor coolant pressure boundary materials meet the July 26, 1983 effective revision of 10 CFR 50 Appendices G and H. Specifically, the 10 CF 50 ruling states that all reactor vessel beltline materials must have an initial Charpy upper shelf energy of 75 ft-lbs and must maintain upper shelf energy throughout the life of the vessel of no less than 50 ft-lbs. Since the ferritic material of the reactor vessel pressure boundaries lowest initial upper shelf energy is 75.5 ft-lbs, in the intermediate shell plate B9004-2, and the lowest end-of-life upper shelf energy of the beltline is predicted to be 54 ft-lbs (Table 4), the fracture toughness requirements of 10 CFR 50 are met.

252.1(b) All ferritic reactor coolant pressure boundary materials meet the fracture toughness requirements of Section 50.55a and Appendices G and H of 10 CFR 50.

252.1(C-1) Beaver Valley Unit 2 heatup and cooldown curves are not impacted by the new 10 CFR 50 rule; which addresses the metal temperature of the closure head flange. Specifically, the 10 CFR 50 rule states the minimum metal temperature of the closure head flange should be RT_{NDT} +120°F for pressure above 621 psig which is 20 percent of the preservice hydrotest pressure of 3106 psig. This minimum temperature for the closure head is 110°F since the RT_{NDT} is -10°F. As a result, the closure head flange region is less limiting than the heatup and cooldown curves which are based on the beltline region. The original heatup limitations (Figure 1) and cooldown limitations (Figure 2) curves still apply for the Beaver Valley Unit 2 reactor.

252.1(C-2) The Beaver Valley Unit 2 surveillance capsule removal schedule is listed in Table 1. The lead factors have changed due to an updated analysis of lead factors that is more advanced and true to life.

TABLE 1

SURVEILLANCE CAPSULE REMOVAL SCHEDULE

Identification	Orientation of Capsules(A)	Lead Factor(B)	Removal Time	Expected Capsule Fluence (n/cm^2)
U	343°	3.12	1st Refueling	6.83×10^{18}
V	107°	3.12	6 EFPYs	3.16×10^{19}(c,e)
W	110°	2.7	12 EFPYs	5.47×10^{19}(d)
X	287°	3.12	18 EFPYs	9.48×10^{19}
Y	290	2.7	---	Standby
Z	340	2.7	---	Standby

(a) Reference Irradiation Capsule Assembly Drawing, Figure 2-5. (From WCAP-9615.)

(b) The factor by which the capsule fluences leads the vessels maximum inner wall fluence.

(c) Approximate Fluence at 1/4 wall thickness at end-of-life.

(d) Approximate fluence at vessel inner wall at end-of-life.

(e) Not required by 10 CFR Part 50 Appendix H or ASTM E185-82 but recommended by Westinghouse.

The plate material used for surveillance was the intermediate shell plate B9004-2. The weld material was taken from the joining of the surveillance plate and the adjoining lower shell plate B9005-2. All heat affected zone specimens were taken from the weld heat affected zone of the intermediate shell plate B9004-2. The test specimens in each capsule are listed in Table 2.

TABLE 2
TYPE AND NUMBER OF SPECIMENS IN THE
BEAVER VALLEY UNIT 2 SURVEILLANCE
TEST CAPSULES

Material	Number of Specimens of Indicated Type			
	Charpy	Tensile	CT	Bend Bar
Plate B9004-2				
Longitudinal	15	3	4	-
Transverse	15	3	4	1
Weld Metal	15	3	4	-
HAZ	15	-	-	-

252.1(C-3) The reference temperature RT_{NDT}, for materials in the reactor vessel closure flange region and beltline region are listed in Table 3.

252.1(C-4) Chemical compositions for all beltline material is listed in Table 3. Unirradiated upper shelf energy (U.S.E.), projected end-of-life USE's and projected end-of-life. RT_{NDT} are indicated in Tables 4 and 5. RT_{NDT} projections were estimated using the "Guthrie Formula" from Commission Report SECY-82-465. EOL upper shelf energy predictions were estimated using Regulatory Guide 1.99, Revision 1 (Figure 3). Although projections for both 1/4T (Table 5) and ID reactor vessel locations (Table 4) were calculated, the ID location is more limiting due to the higher fluence level.

TABLE 3

FRACTURE TOUGHNESS PROPERTIES OF THE REACTOR VESSEL

Component	Code Number	Mat'l Spec. No.	Cu %	P %	Ni%	T_{NDT} °F	RT_{NDT} °F	USE ft-lbs
Closure Head Dome	B9008-1	A533B, Cl. 1	0.13	0.013	0.51	-20	-10	137
Closure Head Flange	B9002-1	A508, Cl. 2	--	0.012	0.74	-10	-10	136
Vessel Flange	B9001-1	A508, Cl. 2	--	0.010	0.73	0	0	132.5
Inlet Nozzle	B9011-1	A508, Cl. 2	--	0.006	0.88	0	0	104
Inlet Nozzle	B9011-2	A508, Cl. 2	--	0.010	0.88	10	10	115
Inlet Nozzle	B9011-3	A508, Cl. 2	--	0.009	0.84	20	20	122
Outlet Nozzle	B9012-1	A508, Cl. 2	--	0.007	0.71	-10	-10	137
Outlet Nozzle	B9012-2	A508, Cl. 2	--	0.006	0.74	-10	-10	121
Outlet Nozzle	B9012-3	A508, Cl. 2	--	0.008	0.68	-10	-10	112
Nozzle Shell	B9003-1	A533B, Cl.1	0.13	0.008	0.61	-10	-10	98
Nozzle Shell	B9003-2	A533B, Cl. 1	0.12	0.009	0.58	0	60	79.5
Nozzle Shell	B9003-3	A533B, Cl. 1	0.13	0.008	0.61	-10	50	97.5
Inter. Shell	B9004-1	A533B, Cl. 1	0.07	0.010	0.53	0	60	83
Inter. Shell	B9004-2	A533B, Cl. 1	0.07	0.007	0.59	-10	40	75.5
Lower Shell	B9005-1	A533B, Cl. 1	0.08	0.009	0.59	-50	28	82
Lower Shell	B9005-2	A533B, Cl. 1	0.07	0.009	0.58	-40	33	77.5
Bottom Head Torus	B9010-1	A533B, Cl. 1	0.15	0.007	0.49	-30	-4	97
Bottom Head Dome	B9009-1	A533B, Cl. 1	0.14	0.007	0.53	-30	-25	116
Weld (Inter. & Lower Shell Long Seams & Girth Seam)*			0.08	0.008	0.09	-30	-30	144.5
HAZ (Plate B9004-2) rwn			--	--		-80	-20	76

(Intermediate and Lower Shell Longitudinal Weld Seams and Girth Seam)

NOTE: *Same heat of wire and lot of flux used in all seams including surveillance weldment.

TABLE 4

PREDICTED EOL RT_{NDT} AND UPPER SHELF ENERGY

OF BELTLINE MATERIAL FOR ID LOCATION

Component	Code No.	Fluence n/cm^2	°F RT_{NDT}	$*\Delta RT_{NDT}$	RT_{NDT} EOL	Avg. USE ft-lb	Reg.** Guide 1.99 %ΔUSE	EOL USE ft-lb
Inter. Shell	B9004-1	$5.4x10^{19}$	60	57	117	83	-28%	60
	B9004-2	$5.4x10^{19}$	40	59	99	75.5	-28%	54
Lower Shell	B9005-1	$5.4x10^{19}$	28	70	98	82	-28%	59
	B9005-2	$5.4x10^{19}$	33	59	92	77.5	-28%	56
Inter. + Lower Shell Long Weld Seam. Inter. to Lower Shell Girth Weld***		$5.4x10^{19}$	-30	47	17	144.5	-36%	92

* - ΔRT_{NDT} was calculated using the Guthrie method.

$$\Delta RT_{NDT}(°F) = (-10 + 470(Cu) + 350\ (\%\ Cu \times \%\ Ni))\ \left(\frac{F}{10^{19}}\right)^{.27}$$

** - Δ Upper shelf energy was determined from the Regulatory Guide 1.99 method (Attached Figure 3.)

*** - Same heat of wire and lot of flux used in all seams including the surveillance weld.

TABLE 5

PREDICTED EOL RT_{NDT} AND UPPER SHELF ENERGY

OF BELTLINE MATERIAL FOR ¼T LOCATION

Component	Code No.	Fluence n/cm^2	°F RT_{NDT} Initial	$*\Delta RT_{NDT}$	RT_{NDT}-EOL	Avg. USE ft-lb	Reg.** Guide 1.99 %ΔUSE	EOL USE ft-lb
Inter. Shell	B9004-1	3.2×10^{19}	60	49	109	83	-25	62
	B9004-2	3.2×10^{19}	40	51	91	75.5	-25	57
Lower Shell	B9005-1	3.2×10^{19}	28	60	88	82	-25	62
	B9005-2	3.2×10^{19}	33	51	84	77.5	-25	58
Inter. & Lower Shell Long Weld Seams. Inter. to Lower Shell Girth Weld***			-30	41	11	144.5	-32	98

* - ΔRT_{NDT} was calculated using the Guthrie method.

$$\Delta RT_{NDT}(°F) = (-10 + 470\ (Cu) + 350\ (\%\ Cu \times \%\ Ni))\ \left(\frac{F}{10^{19}}\right)^{.27}$$

** - Δ Upper shelf energy was determined from the Regulatory Guide 1.99 method. (Attached Figure 3.)

*** - Same heat of wire and lot of flux used in all seams including the surveillance weld.

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL : PLATE METAL
COPPER CONTENT : CONSERVATIVELY ASSUMED TO BE 0.10 WT%
PHOSPHORUS CONTENT : 0.010 WT%
RT_{NDT} INITIAL : 60°F
RT_{NDT} AFTER 10 EFPY : 1/4T, 139°F
3/4T, 114°F

CURVE APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 10 EFPY AND CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS

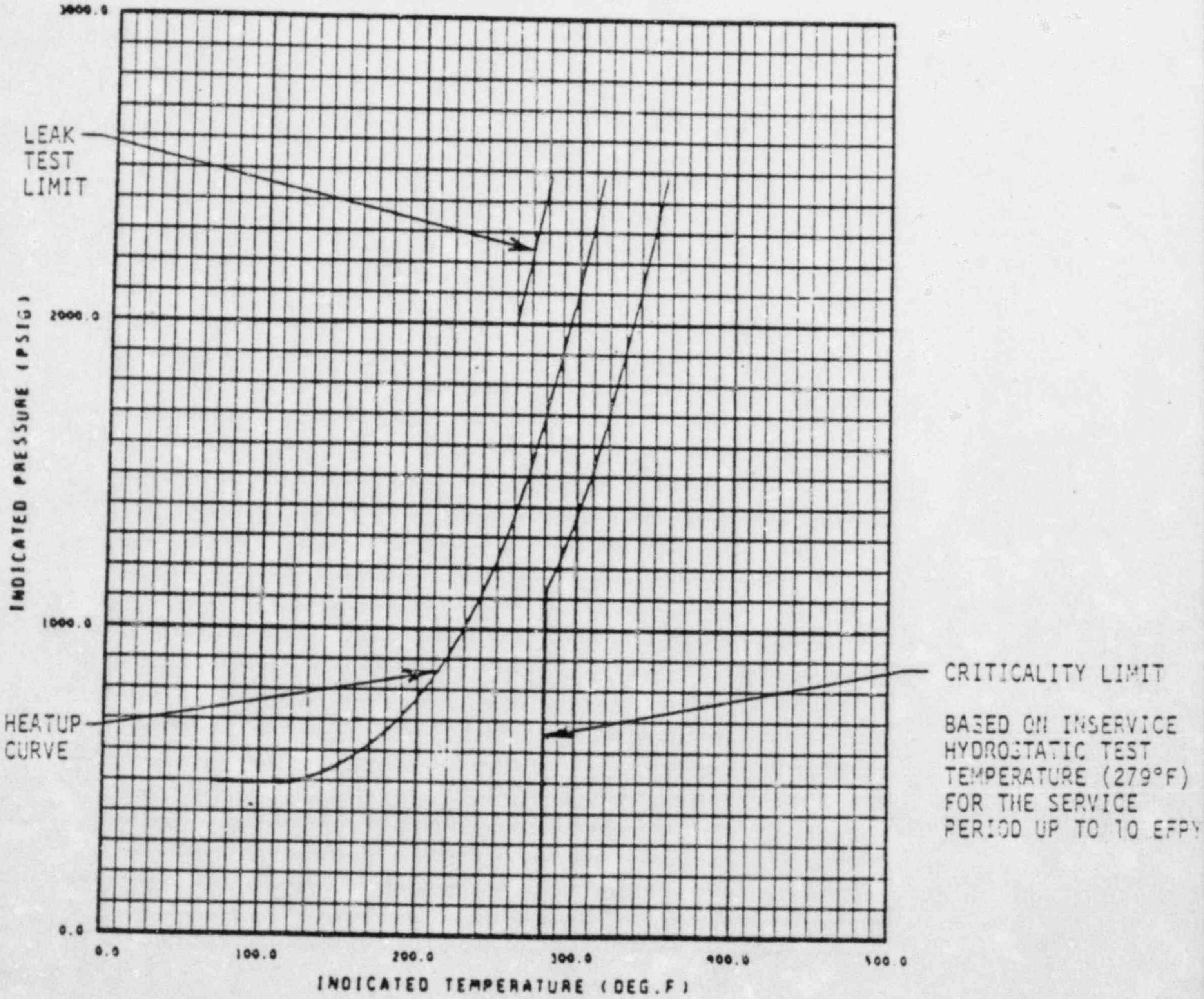

FIGURE 1 BEAVER VALLEY UNIT 2 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE UP TO 10 EFPY

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL : PLATE METAL
COPPER CONTENT : CONSERVATIVELY ASSUMED TO BE 0.10 WT%
PHOSPHORUS CONTENT : 0.010 WT%
RT_{NDT} INITIAL : 60°F
RT_{NDT} AFTER 10 EFPY : 1/4T, 139°F
3/4T, 114°F

CURVE APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 10 EFPY AND CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS

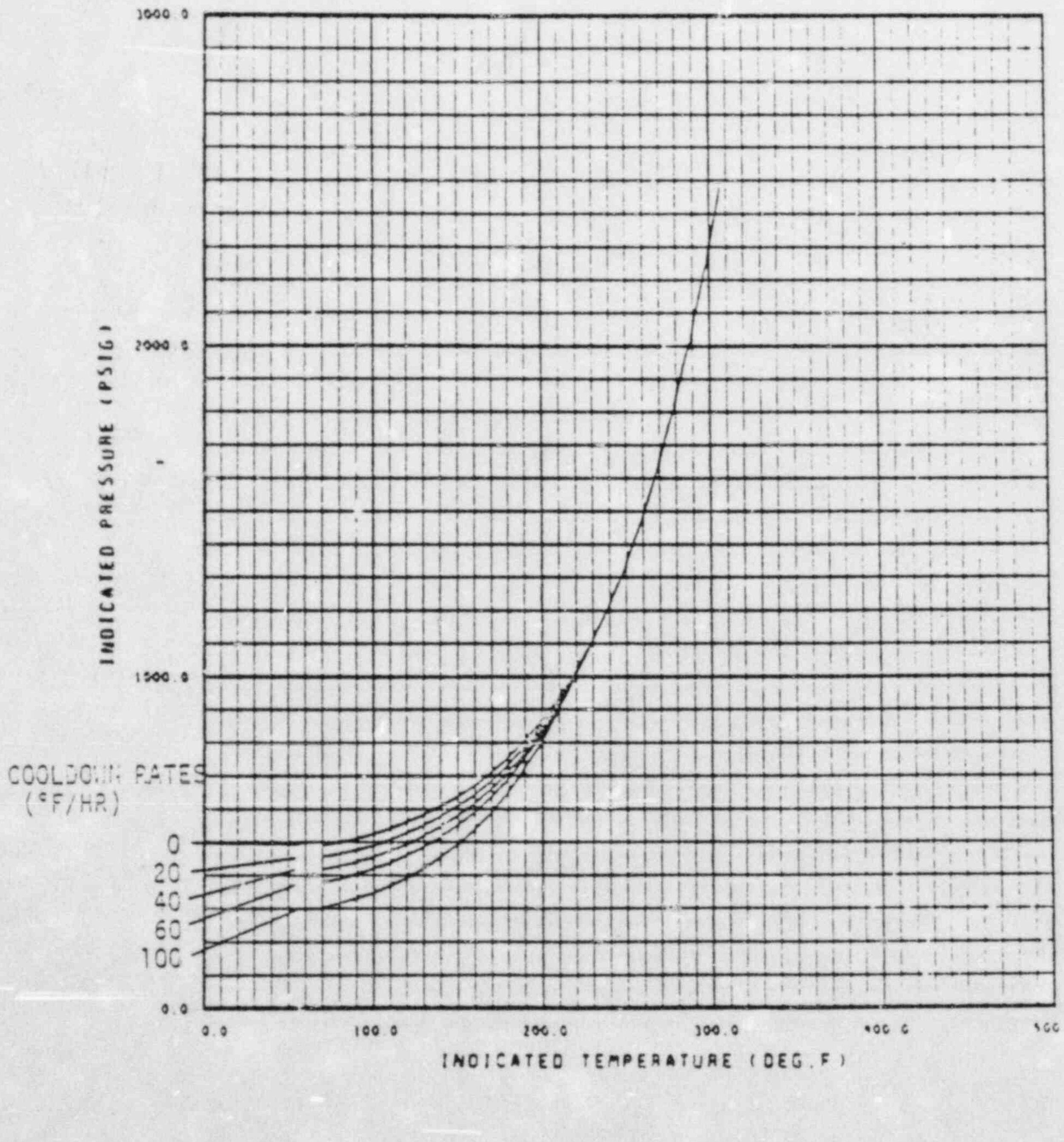

FIGURE 2 BEAVER VALLEY UNIT 2 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS APPLICABLE UP TO 10 EFPY

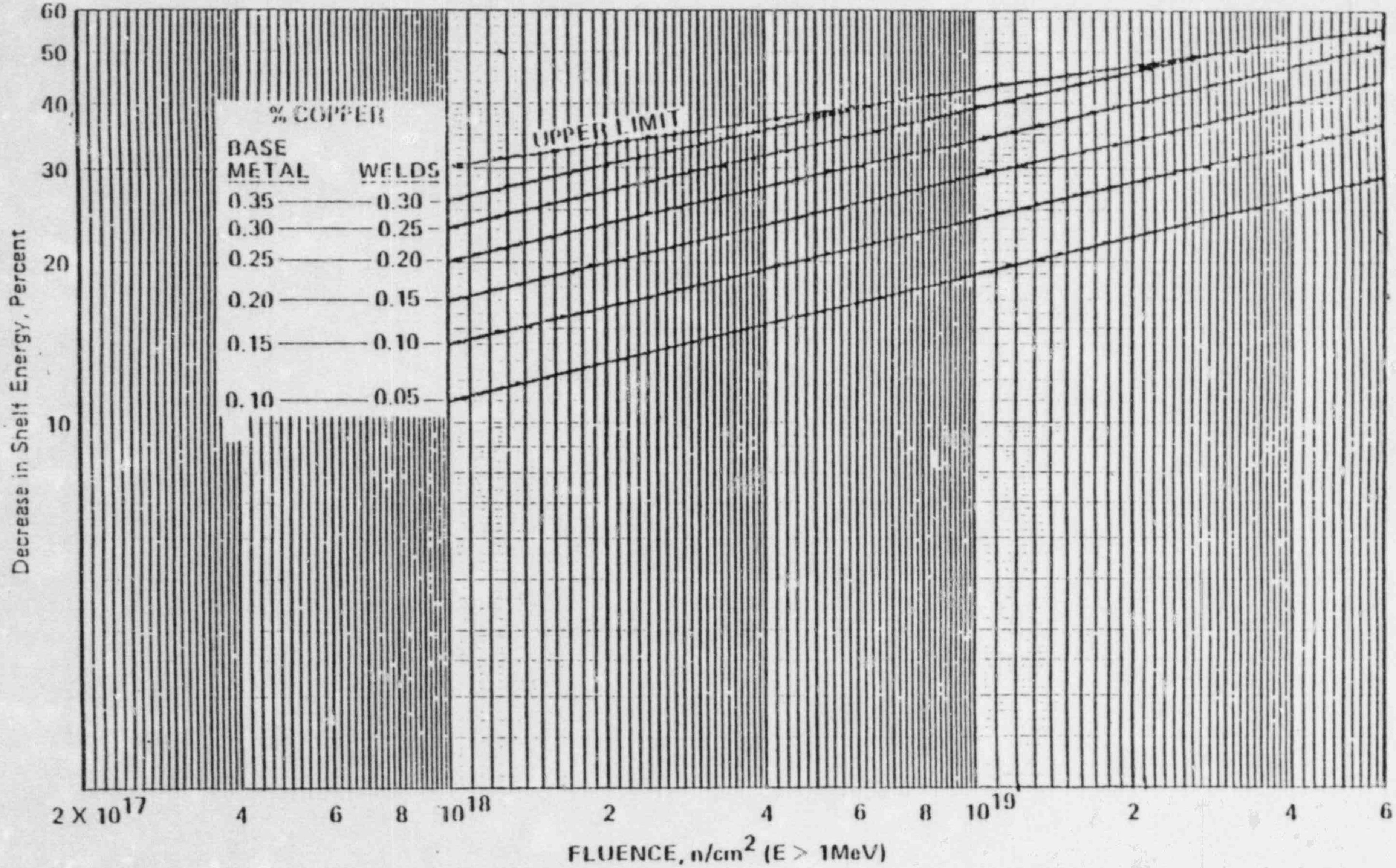

Figure 3: Predicted Decrease in Shelf Energy as a Function of Copper Content and Fluence.