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ComEd

October 19, 1996

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

Subject: Supplemental Response to Generic Letter 87-12
Braidwood Nuclear Power Station, Units 1 and 2
NRC Docket Numbers 50-456 and 50-457

References: See Attachment A

Ladies and Gentlemen:

The purpose of this letter is to notify the United States Nuclear Regulatory Commission (NRC) that Commonwealth Edison Company (ComEd) is revising its commitments made in response to Generic Letter (GL) 87-12 (Reference 1) for Braidwood Nuclear Power Station, Units 1 and 2 (Braidwood). Specifically, Braidwood is revising Responses 8 and 9 which were provided in Reference 2, Attachment 3.

In response to the lessons learned from the April 10, 1987, loss of residual heat removal (RHR or RH) event at Diablo Canyon, the NRC issued GL 87-12 requesting information to assess safe operation of pressurized water reactors when the reactor coolant system (RCS) water level is below the top of the reactor vessel.

GL 87-12, Item 8: "Comparison of the requirements implemented while the RCS is partially filled and requirements used in other Mode 5 operations. Some requirements and procedures followed while the RCS is partially filled may not appear in other modes. An example of such differences is operation with a reduced RHR flow rate to minimize the likelihood of vortexing and air ingestion."

Braidwood responded, in part, that "RH flow will be reduced to approximately 1000 gpm through each RH loop prior to draining to minimize the likelihood of vortexing and air ingestion in the RH pumps."

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GL 87-12, Item 9: "As a result of your consideration of these issues, you may have made changes to your current program related to these issues. If changes have strengthened your ability to operate safely during a partially filled situation, describe those changes and tell when they were made or scheduled to be made."

Braidwood responded, in part, that "the changes include reduced RH flow (1500 to 1000 gpm) during draining and mid-loop operation."

Braidwood is revising those commitments to ensure that during reduced inventory conditions:

1. RH loop flow will be maintained between 1000 - 3300 gpm, inclusive,
2. the letdown portion of the chemical volume and control system (CVCS) will be utilized for draining, and
3. reactor vessel water level will be maintained greater than or equal to 2.25" above mid-loop of the hot legs.

The basis for these changes are as follows. RH flow in the shutdown cooling mode of operation is required to provide sufficient decay heat removal capability, maintain the reactor coolant temperature rise through the core within design limits for compliance with flow rates assumed in the boron dilution analysis, prevent thermal and boron stratification in the core, and ensure that inadvertent boron dilution events can be identified and terminated by operator action prior to the reactor returning critical.

As stated in ComEd's response (Reference 4) to GL 88-17 (Reference 3), Program Enhancement 4, ComEd commissioned the Westinghouse Electric Corporation (Westinghouse) to perform an analysis to allow a reduction in RH flow rate to prevent the loss of RH during mid-loop operations. That analysis (Attachment B) was used to support of a license amendment request (Reference 5) to reduce the minimum RH flow rate from 2800 gpm to 1000 gpm in Mode 6. The NRC approved that request in Reference 6.

Included in the original analysis was a plant specific curve of acceptable RCS level (as measured in inches above or below the mid-loop of the hot leg) as a function of RH flow rate. Re-analysis of that curve (Attachment C) demonstrates that with a maximum RH loop flow of 3300 gpm the reactor vessel water level could be maintained as low as 2.25" above the mid-loop of the hot leg without vortexing at the suction of the RH pump and subsequent loss of shutdown cooling. Braidwood's revised commitment will ensure that the assumptions of this analysis are maintained.

If there are any questions regarding this submittal, please contact Harry Pontious, Acting Regulatory Assurance Supervisor, at (815) 458-2801 x2980.

for

Harold Gene Stanley
Site Vice President
Braidwood Nuclear Generating Station

HGS/hdp

cc: A. B. Beach, Regional Administrator - RIII
C. J. Phillips, Senior Resident Inspector - Braidwood
R. R. Assa, Braidwood Project Manager - NRR
Illinois Department of Nuclear Safety

Attachment A

References

1. Generic Letter (GL) 87-12, "Loss of Residual Heat Removal (RHR) while the Reactor Coolant System (RCS) is Partially Filled," dated July 9, 1987

2. W. E. Morgan (ComEd) letter to Frank J. Miraglia (NRC), Commonwealth Edison Company (ComEd) Response GL 87-12, dated September 25, 1987

3. GL 88-17, "Loss of Decay Heat Removal," dated October 17, 1988

4. R. A. Chrzanowski (ComEd) letter to Thomas E. Murley (NRC), ComEd Response to GL 88-17 Program Enhancements, dated January 31, 1989

5. T. K. Schuster (ComEd) letter to Thomas E. Murley (NRC), "Application for Amendment to Facility Operating Licenses NPF-36/66 & NPF-72/77 regarding Implementation of Generic Letter 88-17," dated January 31, 1990

6. Stephen P. Sands (NRC) letter to Thomas J. Kovach (ComEd), "Issuance of Amendments (TAC Nos. 76715, 76716, 76717 and 76718)," dated August 31, 1990

Attachment B

G. P. Toth (Westinghouse) letter (CAE-89-214/CCE-89-215) to C. A. Moerke (ComEd), "Commonwealth Edison Company, Byron and Braidwood Stations, Mid-Loop (GL 88-17) Task B Report," dated July 12, 1989

128339

Westinghouse Electric Corporation **Power Systems**

Energy Systems
Service Division

Box 355
Pittsburgh Pennsylvania 15230-0355

July 12, 1989
CAE-89-214
CCE-89-215
NS-OPLS-OPL-I-89-368

Mr. C. A. Moerke
PWR Engineering
Commonwealth Edison Company
P. O. Box 767
Chicago, IL 60690

Commonwealth Edison Company
Byron and Braidwood Stations
Mid-Loop (GL 88-17) Task B Report

Dear Mr. Moerke:

Please find enclosed documents relating to the finalization of the Mid-Loop Task effort for Byron and Braidwood stations.

1. **WCAP12207, "Reduction in the Minimum RHRS Flowrate during Mid-Loop Operation for Byron and Braidwood Power Plants Units 1 and 2"**, dated June 1989 including Appendices.

2. **SECL-89-867, "Reduction in the Minimum RHR Flowrate during Mid-Loop Operation"**, revised to incorporate RHR Pump Thrust Bearing life and Commonwealth Edison comments to the WCAP.

3. A Significant Hazards Consideration Analysis for Commonwealth Edison to use in its licensing submittal.

Proposed FSAR and Technical Specification changes are included in the above WCAP report as Appendices.

If you have any questions, please do not hesitate to call.

Sincerely,

S. A. Pujadas for
G. P. Toth, Manager
Commonwealth Edison Projects
Customer Projects Department

SAP/mbs

Attachment

cc: R. Chrzanowski - 1L, 1A
R. Pleniewicz - 1L, 1A
R. E. Querio - 1L, 1A
L. Dworakowski - 1L, 1A
P. McHale - W - 1L, 1A
W. J. Feimster - W - 1L, 1A

Significant Hazards Consideration Evaluation for a Reduction in the Minimum RHR Flowrate During Mid-Loop Operation at Byron and Braidwood Units 1 and 2

Introduction

As required by 10 CFR 50.91 (a) (1) this evaluation is provided to demonstrate that a proposed license amendment to reduce the minimum required Residual Heat Removal (RHR) flowrate during mid-loop operation at the Byron and Braidwood Units 1 and 2, represents no significant hazards consideration. In accordance with the three factor test of 10 CFR 50.92 (c), implementation of the proposed license amendment was analyzed and found not to: 1) involve a significant increase in the probability or consequences for an accident previously evaluated; or 2) create the possibility of a new or different kind of accident from any accident previously evaluated; or 3) involve a significant reduction in a margin of safety.

Mid-loop operation occurs when the plant is operating with the Reactor Coolant System (RCS) partially drained in Modes 5 and 6. Currently, there is no Technical Specification RHR flowrate requirement for operation in Mode 5. For Mode 6, however, a minimum RHR flowrate of 2800 gpm is specified in Surveillance Requirements 4.9.8.1 and 4.9.8.2.

The Reactor Coolant System (RCS) water level is lowered in Modes 5 and 6 to facilitate removal and reinstallation of the reactor head during refueling outages. Operation with the RCS partially drained may also be necessary for the inspection and maintenance of RCS components such as reactor coolant pumps and steam generators. However, when the reactor coolant level in the RCS loop piping is lowered, there is a potential for air to be drawn into the RHRS suction line (air entrainment) due to RCS loop level fluctuations and/or the development of a vortex. Air entrainment into the RHRS could cause air binding of the RHR pumps and thus, result in the inadvertent loss of decay heat removal capability. The tendency for vortex formation at the RHR suction line, and subsequent air entrainment into the RHRS, is a function of the water level above the RHRS suction nozzle and the RHR flowrate. The lower the level, or the higher the RHR flowrate, the greater the potential for a vortex to develop and air to be drawn into the RHRS. Therefore, the likelihood of vortex formation due to partial draining of the RCS can be offset by reducing the RHR flowrate.

The required minimum RHR flowrate during mid-loop operations is based on the following concerns.

- The ability of the RHRS to remove decay heat such that RCS temperature can be controlled.

- Sufficient flow is provided to ensure that reactor coolant temperature rise through the core does not exceed reactor vessel internals delta T limits.

- Sufficient flow is provided to ensure that the reactor coolant is mixed such that significant boron stratification does not occur.

- Sufficient flow is provided to ensure that the pressure drop across the RHR bypass flow control valve does not result in cavitation of the reactor coolant.

- Sufficient flow is provided to ensure that inadvertent boron dilution events can be identified and terminated by operator action prior to the reactor returning critical.

Since the RHR flowrate requirements are dependent on plant conditions, it is recommended that a single flowrate requirement not be included in the Byron/Braidwood Technical Specifications. Instead, acceptable RHR flowrates that are consistent with the plant conditions would be specified in the plant procedures.

Evaluation

Conformance of the proposed amendments to the standards for a determination of no significant hazard as defined in 10 CFR 50.92 (three factor test) is shown in the following:

Operation of the Byron and Braidwood Units 1 and 2, in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of any accident previously evaluated.

A reduction in RHR flow during mid-loop operation will potentially impact those transients explicitly analyzed in Modes 5 and 6. The only event analyzed for these modes in Chapter 15 of the Byron/Braidwood UFSAR is the malfunction of the CVCS that results in a decrease in boron concentration in the reactor coolant. The CVCS malfunction event can be impacted by a reduction in RHR flow in the following two areas: 1) A reduction in explicit RHR flowrate assumptions and 2) The vessel mixing assumption during a boron dilution. The Mode 5 an 6 analyses do not assume an explicit RHR flow value, and the RHR flowrates are assumed to be sufficient to provide adequate vessel circulation to prevent boron stratification and support the boron dilution transient mixing assumptions.

In addition, since a CVCS malfunction event in Mode 6 is prevented by administrative controls which isolate the RCS from any potential source of unborated water, only the Mode 5 analysis could be impacted. However, it is determined that a reduced RHR flow of 1000 gpm or greater would not invalidate the Byron/Braidwood accident analysis assumptions. Therefore, the current Mode 5 analysis remains valid.

The proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Acceptable RHR flowrates that are consistent with the plant conditions would be specified in the plant procedures. The RHR flowrates would be such that: 1) The RHRS would be capable of decay heat removal to control the RCS temperature, 2) The reactor coolant temperature rise through the core would not exceed reactor vessel internals delta T limits, 3) The reactor coolant would be mixed to prevent significant boron stratification from occuring, 4) The pressure drop across the RHR flowrate control valve would not result in cavitation of the reactor coolant, and 5) Inadvertent boron dilution events could be identified and terminated by operator action prior to the reactor returning critical.

Thus, a reduction in RHR flow would not increase the probability of a CVCS malfunction event and the possibility of an accident which is different than any already evaluated in the FSAR would not be created.

The proposed license amendment does not involve a significant reduction in a margin of safety.

Currently, the Byron/Braidwood Technical Specifications do not specify RHR flowrate requirements for operation in Mode 5. Mode 6 operations, however, require a minimum RHR flowrate of 2800 gpm (Surveillance Requirements 4.9.8.1 and 4.9.8.2). The Technical Specifications place limitations on the RHRS during mid-loop operation by specifying a minimum flow requirement for the purpose of decay heat removal and the number of RHR trains which must be operable. They do not, however, contain restrictions based on minimizing air entrainment in the RHRS as a result of vortexing which may occur during mid-loop operation under certain conditions.

The fuel cladding (fission product barrier) is protected in Modes 5 and 6 by providing cooling and maintaining core shutdown. Adequate decay heat removal is provided to address the cooling requirements, and sufficient mixing ensures that the boron dilution analyses remain valid. Therefore, the amount of time available to identify and terminate a boron dilution event is unaffected.

Thus, a reduced RHR flowrate during mid-loop operation does not involve a significant reduction in a margin of safety.

Conclusion

Based on the preceding evaluation, it is concluded that operation of Byron and Braidwood Units 1 and 2 in accordance with the proposed amendment, does not create an unreviewed safety question, increase the probability of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, nor reduce any margins to plant safety. Therefore, the license amendment does not involve a Significant Hazards Consideration as defined in 10 CFR 50.92.

SECL No. 89-867
Customer Reference No(s).

Westinghouse Reference No(s).
G. O. In22479

WESTINGHOUSE NUCLEAR SAFETY
SAFETY EVALUATION CHECK LIST

1) NUCLEAR PLANTS Byron & Braidwood Units 1 & 2

2) CHECK LIST APPLICABLE TO: Reduction in the Minimum RHR Flowrate during Mid-Loop Operation

3) The written safety evaluation of the revised procedure, design change or modification required by 10CFR50.59 (b) has been prepared to the extent required and is attached. If a safety evaluation is not required or is incomplete for any reason, explain on Page 2.

   Parts A and B of this Safety Evaluation Check List are to be completed only on the basis of the safety evaluation performed.

   CHECK LIST - PART A - 10CFR50.59 (a) (1)

   (3.1) Yes_X_ No___ A change to the plant as described in the FSAR?
   (3.2) Yes___ No_X_ A change to procedures as described in the FSAR?
   (3.3) Yes___ No_X_ A test or experiment not described in the FSAR?
   (3.4) Yes_X_ No___ A change to the plant technical specifications? (See Note on Page 2)

4) CHECK LIST - PART B - 10CFR50.59 (a) (2) (Justification for Part B answers must be included on page 2.)

   (4.1) Yes___ No_X_ Will the probability of an accident previously evaluated in the FSAR be increased?
   (4.2) Yes___ No_X_ Will the consequences of an accident previously evaluated in the FSAR be increased?
   (4.3) Yes___ No_X_ May the possibility of an accident different than any already evaluated in the FSAR be created?
   (4.4) Yes___ No_X_ Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
   (4.5) Yes___ No_X_ Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
   (4.6) Yes___ No_X_ May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?
   (4.7) Yes___ No_X_ Will the margin of safety as defined in the bases to any technical specification be reduced?

If the answers to any of the above questions are unknown, indicate under 5) REMARKS and explain below.

A Technical Specification change is required and the subject change would require an application for license amendment as required by 10CFR50.59 (c) and submitted to the NRC pursant to 10CFR50.90.

5) REMARKS:

The following summarizes the justification based upon the written safety evaluation for answers given in Part A and Part B of this Safety Evaluation Check List:

See Attached, Safety Evaluation for a Reduction in the Minimum RHR Flowrate during Mid-Loop Operation at Byron & Braidwood Units 1 & 2

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6) FOR FSAR UPDATE:

Section: 5.4.7.2.1 Page: 5.4-34

Reason for/Description of Change:

To allow a reduction in minimum RHR flowrate during mid-loop operation

7) Prepared by:

Engineer: A.J. Abels *A.J. Abels* Date: 7/11/89
Operating Plant Licensing

Manager: R.J. Sterdis *R.J. Sterdis* Date: 7/11/89.
Operating Plant Licensing

Safety Evaluation for a Reduction in the Minimum RHR Flowrate During Mid-Loop Operation at Byron and Braidwood Units 1 & 2

Background

The Commonwealth Edison Company has requested that Westinghouse evaluate the safety impact of a reduction in the minimum RHR flowrate during mid-loop operation at Byron and Braidwood Units 1 and 2. Mid-loop operation occurs when the plant is operating with the RCS partially drained in Modes 5 and 6. Currently, there are no Technical Specification RHR flowrate requirements for operation in Mode 5. However, for Mode 6 operations, a minimum RHR flowrate of 2800 gpm is specified.

The Reactor Coolant System (RCS) water level is lowered in Modes 5 and 6 to facilitate removal and reinstallation of the reactor head during refueling outages. Operation with the RCS partially drained may also be necessary for the inspection and maintenance of RCS components such as reactor coolant pumps and steam generators. However, when the reactor coolant level in the RCS loop piping is lowered, there is a potential for air to be drawn into the RHRS suction line (air entrainment) due to RCS loop level fluctuations and/or the development of a vortex. Air entrainment into the RHRS could cause air binding of the RHR pumps and thus, result in the inadvertent loss of decay heat removal capability. The tendency for vortex formation at the RHR suction line, and subsequent air entrainment into the RHRS, is a function of the water level above the RHRS suction nozzle and the RHR flowrate. The lower the level, or the higher the RHR flowrate, the greater the potential for a vortex to develop and air to be drawn into the RHRS. Therefore, the likelihood of vortex formation due to partial draining of the RCS can be offset by reducing the RHR flowrate. Figure 1 shows the RCS hot leg water level (inches above the centerline) as a function of RHR intake flow.

The primary function of the RHRS is to remove residual heat from the core and reduce the temperature of the RCS during the second phase of plant cooldown. As a secondary function, the RHRS is used to transfer refueling water between the Refueling Water Storage Tank (RWST) and the refueling cavity before and after the refueling operations. The RHRS also serves as part of the Emergency Core Cooling System (ECCS) during the injection and recirculation phases of a Loss-of-Coolant Accident (LOCA).

The RHRS consists of two parallel RHR trains. The inlet line to each train of the RHRS is connected to a reactor coolant loop hot leg, while the return lines are connected to the cold legs of each of the reactor coolant loops. Each train includes one centrifugal RHR pump, one residual heat exchanger (shell and U-tube type), and associated piping, valves, and instrumentation.

During RHRS operation, reactor coolant flows from the RCS to the RHR pumps, through the tube side of the RHR heat exchangers, and back to the RCS cold legs. Heat is transferred from the reactor coolant to the Component Cooling Water (CCW) circulating through the shell side of the residual heat exchangers.

Evaluation

The required minimum RHR flowrate during mid-loop operations is based on the following concerns:

- The ability of the RHRS to remove decay heat such that RCS temperature can be controlled.
- Sufficient flow is provided to ensure that the reactor coolant temperature rise through the core does not exceed reactor vessel internals delta T limits.
- Sufficient flow is provided to ensure that the reactor coolant is mixed such that significant boron stratification does not occur.
- Sufficient flow is provided to ensure that the pressure drop across the RHR bypass flow control valve does not result in cavitation of the reactor coolant.
- Sufficient flow is provided to ensure that inadvertent boron dilution events can be identified and terminated by operator action prior to the reactor returning critical.

Decay Heat Removal

As stated previously, the primary function of the RHRS is to remove decay heat during the second phase of plant cooldown. However, at reduced RHR flowrates, the decay heat removal capacity of the RHRS will be decreased. Therefore, lower flowrates require that the reactor be shutdown for a longer period of time before the RHRS can remove all of the decay heat being generated. Figure 2 shows the minimum flowrate required to maintain a constant reactor coolant temperature, as a function of time after shutdown. The three curves presented in Figure 2, correspond to the following cases.

- A reactor coolant temperature of 140°F, maintained by one RHR train operating at the indicated flowrate.
- A reactor coolant temperature of 140°F, maintained by two RHR trains, each operating at the indicated flowrate.
- A reactor coolant temperature of 200°F, maintained by one RHR train operating at the indicated flowrate.

The curves indicate that decay heat decreases as a function of time after initial reactor shutdown. Thus, as the time after plant shutdown increases, the decay heat removal requirements for RHR flow are reduced.

Thermal Considerations

Another potential concern with low RHR flowrates is thermal effects in the reactor vessel and reactor coolant loops. Thermal stratification (non-isothermal conditions) may be a concern for the following two reasons:

- Core reactivity varies as a function of coolant density and, thus, temperature.

- Coolant temperatures in the core are inferred from temperature measurements taken in the RCS/RHRS loops.

Regarding the first concern, core reactivity and adequate shutdown margin are evaluated for a range of temperatures (68°F to 200°F) during Mode 5 operation. Therefore, thermal stratification will not create temperature conditions in the core more adverse than already considered.

Since flow is entering the reactor vessel from each cold leg nozzle, adequate mixing of the reactor vessel volume is expected at reduced RHR flows so as to minimize thermal stratification.

In addition to potential thermal stratification, a reduction in RHR flowrate will increase the reactor coolant temperature rise through the core during RHR cooling. The decay heat load is removed by increasing the temperature of the coolant as it passes through the core. As the mass flowrate is decreased, the temperature rise must increase to maintain constant heat removal. Certain structural considerations of the reactor vessel internals limit acceptable core temperature rise. In particular, the most limiting components in terms of core temperature rise are the baffle-former bolts and baffle-barrel bolts.

Based on the fatigue usage factor of the baffle-former and baffle-barrel bolts, the maximum allowable steady state difference between reactor vessel inlet and outlet temperatures during mid-loop operation is 72°F.

The minimum allowable RHR flowrate as a function of time, and the decay heat limit curve for Mode 5 (temperature less than 200°F), are shown in Figure 3. The curves intersect at approximately 47 hours, after which the core delta T becomes the limiting factor. In Mode 6 (temperature less than 140°F), decay heat will always be limiting.

<u>Boron Mixing and Stratification</u>

Sufficient RHR flow must be provided to maintain a uniform boron concentration throughout the RCS. "Boron stratification" refers to the localized variations in boron concentration. Boron stratification is most likely to occur in the RCS when a controlled boration (or dilution) operation is first initiated. During this operation, the RHR flow ensures mixing within the RCS volume. Thus, as RHR flow is reduced, the mixing rate decreases, and the time required to obtain a uniform RCS boron concentration increases. Typically, however, the RCS boron concentration is stabilized at the required shutdown margin prior to reducing RHR flowrate, ensuring a uniform boron concentration.

Provided that the reactor coolant is not intentionally diluted during mid-loop operations, precipitation and local evaporation would be the most likely mechanisms for inducing a boron gradient in the reactor vessel. However, concentration would be in the range of 2000 ppm (1% concentration). Since the saturation temperature of a 1% solution is less than 32°F, boric acid precipitation would not occur. Even if mass evaporation would occur, the local boron concentrations (without mixing) would actually increase, which is in the conservative direction.

Control Valve Cavitation

The RHR flowrate is reduced during mid-loop operation by fully closing the RHR bypass flow control valve (FCV-618 or 619), and then slowly closing the associated hand control valve (HCV-606 or 607). The pressure drop across the control valve increases as flow is reduced. Eventually, cavitation of the reactor coolant could result. Cavitation that occurs in control valves under high pressure drop conditions, is due to a portion of the liquid transforming into the vapor phase during rapid acceleration of the fluid inside the valve, and the subsequent collapse of these vapor bubbles downstream of the valve. Severe cavitation could cause excessive wear and vibration in the piping downstream of the control valve.

An analysis was performed using standard equations published by valve manufacturers to predict the onset of cavitation. The results indicate that cavitation will not occur at flowrates greater than 2000 gpm. However, Westinghouse does not recommend establishing a limit on RHR flow of 2000 gpm at this time. This limit may be overly conservative, for the following reasons:

- The formulae used to predict the onset of cavitation are not exact. Thus, cavitation may not actually occur until a flowrate lower than 2000 gpm is reached.

- When cavitation occurs, it will initially be at a low level which may not be detrimental to the valve or piping.

Therefore, it may be possible to reduce the RHR flowrate significantly below 2000 gpm without harm. Any cavitation that would be severe enough to cause damage, would be evident due to the excessive noise and vibration in the piping downstream of the valve. Thus, at this time, Westinghouse recommends that the limit on RHR flowrate reduction initially be established based on the other concerns identified (decay heat removal, thermal considerations, and boron mixing and stratification). The first time that the RHR flowrate is reduced to a value less than 2000 gpm, the piping immediately downstream of the control valve should be visually monitored. If excessive vibration or audible noise is observed, it may be necessary to establish a higher minimum RHR flowrate, based on the cavitation concerns.

Inadvertent Boron Dilution

The proposed reduction in RHR flow during mid-loop operation will potentially impact those transients explicitly analyzed in Modes 5 and 6. The only non-LOCA event analyzed in these modes is the Chemical and Volume Control System (CVCS) Malfunction, which results in dilution of the primary coolant (presented in Section 15.4.6 of the UFSAR).

A reduction in RHR flow during mid-loop operation has the potential to impact the CVCS Malfunction event in the following two areas:

- A reduction in explicit RHR flowrate assumptions.

- The vessel mixing assumption during a boron dilution.

A CVCS malfunction event in Mode 6 is prevented by administrative controls which isolate the RCS from any potential source of unborated water. The appropriate CVCS valves that are required to be closed and secured are identified in the plant Technical Specifications. Thus, the proposed reduction in RHR flow during mid-loop operations has no impact on the Mode 6 analysis.

Mode 4 is the only mode analyzed in the CVCS Malfunction event that explicitly accounts for a minimum amount of RHR flow. The Mode 5 and 6 analyses do not assume an explicit RHR flow value. Since mid-loop operation is not permitted in Mode 4 and the RHR flow requirement outside of mid-loop operation will not be changed, there is no impact on the CVCS Malfunction event in Mode 4.

The Modes 4 through 6 analyses that account for RHR performance (either explicitly or implicitly), assume that the RHR flowrates provide adequate vessel circulation to prevent boron stratification and support the boron dilution transient mixing assumptions. The proposed reduction in the minimum RHR flow requirement during mid-loop operation impacts only the Mode 5 analysis. Therefore, it was necessary to confirm that the current Mode 5 boron dilution analysis remains valid for the reduced RHR flowrate. The proposed reduction in RHR flow to 1000 gpm during mid-loop operation will not invalidate these assumptions, thus, the current Mode 5 analysis is still valid.

It has been demonstrated above that a reduction in the minimum RHR flowrate requirement during mid-loop operation to 1000 gpm or greater will have no adverse effects on the non-LOCA accident analyses. The results and conclusions presented in the Byron/Braidwood UFSAR remain valid.

RHR Pump Thrust Bearing

Westinghouse evaluated the Byron/Braidwood RHR pump motor thrust bearing life for the hydraulic thrust, deadweight and seismic loads which act on the thrust bearing. The hydraulic thrust loads were based on results recorded during Ingersoll-Rand testing of the same pump model. This testing recorded only hydraulic thrust loads developed by the pump internals and is independent of the pump support conditions.

The seismic evaluation is a calculation of the bearing load due to the rotor deadweight loads, vendor test hydraulic loads and the seismic load of 2.1 g[1]. The total bearing load is based only on those loads reacting on the pump/motor rotor assembly and is independent of pump support conditions. The seismic evaluation of the thrust bearing demonstrated that the bearing could withstand the magnitude of the combined seismic, hydraulic and deadweight loads and operate through the duration of a

[1] identified in Byron/Braidwood RHR pump equipment specification

seismic event. Furthermore it demonstrated that the thrust bearing capacity is adequate to withstand seismic event loadings incurred and that the duration of five OBE and one SSE events is so short that the seismic conditions have an insignificant effect on the overall bearing life.

Bearing life was predicted based on the normal hydraulic and deadweight loads. The B10[2] bearing life was calculated for worst case operation and for realistic normal pump operation with both single pump service duty and split service duty between the two pumps per plant. The B10 bearing life is the minimum expected bearing life as defined by AFBMA Standard 9. The B10 thrust bearing life was first calculated assuming that the RHR pump operates continuously under the worst thrust load conditions. The thrust bearing life while operating at the flowrate corresponding to the peak hydraulic thrust (~2000 GPM) is 7937 hours. This value is very conservative since the RHR pumps do not operate at this flowrate during any of the defined plant modes of operation. The B10 bearing life was then calculated as a cumulative value based on pump operation at the various operating modes identified in Table 1. This bearing life was converted into a replacement interval which must be followed in order to ensure that the bearings are capable of operating for the entire post-accident requirement. The replacement intervals were calculated for both one year and 100 day post-accident operation. The intervals were calculated both assuming that one RHR pump performs all service duty and also assuming that the pumps each see 50 percent service duty for the refueling, shutdown, midloop and plant cooldown operating modes.[3] The resulting thrust bearing replacement intervals are tabulated below.

2 The formula for bearing life B10 is available in engineering handbooks and bearing manufacturers' catalogs, but may appear in different forms and use different factors depending upon the source. This is because it is a statistical life based on empirical data. Results, however, using the various forms are not expected to be significantly different. This evaluation is based on the TRW Engineer's Handbook (2nd Edition, 1982) formula:

$$\text{Life} = 1500\,(\text{service factor})^3$$
$$\text{Service Factor} = (\text{rated capacity}) / (\text{equivalent load})$$
$$\text{Rated Capacity} - \text{defined by Handbook based on bearing model}$$
$$\text{Equivalent Load} = 0.62\,(\text{radial load}) + (\text{thrust load})$$

3 The selection of single pump service and 50 percent split duty service is intended to predict a minimum and a realistic bearing life expectation. The results are presented in terms of a recommended bearing replacement interval as a more practical means for developing a maintenance program than is predicted bearing life expressed in total hours of operation. A modification to the pump impeller wear ring can be performed to reduce the effective downthrust on the motor thrust bearing. An evaluation of the specific operating modes for another plant application with the modification showed a significant increase (~4x) in thrust bearing interval replacement. A plant specific evaluation is required to determine the anticipated increase in bearing replacement interval (or if the recommended replacement interval would exceed the life of the plant) with the modification incorporated.

Post-Accident Period	Single Pump Service Duty	Split Service Duty
100 days	6.9 years	13.7 years
1 year	5.9 years	11.8 years

There is a precaution regarding the use of the thrust bearing replacement intervals as identified above. The vendor testing of hydraulic thrust loads identified that the thrust load drops significantly between flow rates of 2800 and 3000 GPM. Since the Byron/Braidwood RHR pumps operate for nearly 70 percent of their life at 3000 GPM, the calculated bearing replacement intervals are predominantly controlled by the thrust load at 3000 GPM. Due to the shape of the hydraulic thrust curve, any slight change in flowrate or a minor variation in pump hydraulic characteristics while operating at a nominal 3000 GPM can result in a significant reduction in the thrust bearing life. For this reason, it is advisable to increase the flowrate to at least 3300 GPM for all operating modes which currently have 3000 GPM.

Technical Specification Changes

During mid-loop operations, the RHRS operates at reduced flowrates to avoid air binding of the RHR pumps. The acceptability of the reduced RHR flowrate could be based on accompanying administrative actions such as a presrcibed minimum time after shutdown or temperature rise monitoring. Since RHR requirements are dependent on plant conditions, it is recommended that a single flowrate requirement not be included in the Technical Specifications. Instead, acceptable RHR flowrates (minimum and maximum) that are consistent with the plant conditions will be specified in the plant procedures.

Since the safety concerns for Modes 5 and 6 are similar, it is also recommended that the Mode 5 and 6 specifications be consistent in addressing the above concerns.

The Technical Specifications place limitations on the RHRS during mid-loop operation by specifying a minimum flow requirement for the purpose of decay heat removal and the number of RHR trains which must be operable. The Technical Specifications do not, however, contain restrictions based on minimizing air entrainment in the RHRS as a result of vortexing which may occur during mid-loop operation under certain conditions.

It is recommended that Surveillance Requirements 4.9.8.1 and 4.9.8.2 be revised to delete reference to a specified flowrate of greater than or equal to 2800 gpm. In addition, revision to the Bases of Technical Specifications 3/4.4.1 and 3/4.9.8 are proposed to identify: (1) the concerns that are to be addressed in determining the minimum RHR flow requirements during mid-loop operation, (2) the dependency of the required minimum RHR flowrate on plant conditions, and (3) the potential for vortexing to cause air binding of the RHR pumps and subsequent loss of decay heat removal due to partial draining of the RCS.

The Byron/Braidwood Technical Specifications have been marked-up to reflect these proposed changes.

Conclusion

The potential impact of a reduction in the minimum RHR flowrate during mid-loop operation has been addressed by this safety evaluation in accordance with the requirements of 10 CFR 50.59. Based on the decay heat removal, reactor vessel internals delta T limits, boron mixing and stratification, control valve cavitation, inadvertent boron dilution, and RHR pump bearing evaluations, the following is concluded.

The probability of an accident previously evaluated in the FSAR will not be increased.

A reduction in RHR flow during mid-loop operation will potentially impact those transients explicitly analyzed in Modes 5 and 6. The only event that is analyzed for Modes 5 and 6 in Chapter 15 of the UFSAR is the malfunction of the CVCS that results in a decrease in reactor coolant boron concentration. The CVCS malfunction event can be impacted by a reduction in RHR flow by a reduction in explicit RHR flowrate assumptions and the vessel mixing assumption during a boron dilution. The Mode 5 and 6 analyses do not assume an explicit RHR flow value and the RHR flowrates are assumed to be sufficientto provide adequate vessel circulation to prevent boron stratification and support the boron dilution transient mixing assumptions. Thus, the probability of that accident evaluated in the FSAR would not be increas by a reduction in the RHR flowrate during mid-loop operation.

The consequences of an accident previously evaluated in the FSAR will not be increased.

A reduction in the minimum RHR flowrate during mid-loop operation potentially impacts those transients explicitly analyzed in Modes 5 and 6. The only non-LOCA event analyzed in these modes is CVCS malfunction (UFSAR Section 15.4.6). A CVCS malfunction event in Mode 6 is prevented by administrative controls which isolate the RCS from any potential source of unborated water. Thus, only the Mode 5 analysis could be impacted. As stated previously, since a reduced RHR of 1000 gpm or greater will not invalidate the Byron/Braidwood accident analysis assumptions, the current Mode 5 analysis remains valid, and the consequences of the accident evaluated in the FSAR would not be increased.

The possibility of an accident which is different than any already evaluated in the FSAR may not be created.

Acceptable RHR flowrates that are consistent with plant conditions will be specified in the plant procedures. The RHR flowrates would be such that: 1) the RHRS would be capable of decay heat removal to control the RCS temperature, 2) the reactor coolant temperature rise through the core would not exceed reactor vessel internals delta T limits, 3) the reactor coolant would be mixed to prevent significant boron stratification from occuring, 4) the pressure drop across the RHR bypass flow control valve would not result in cavitation of the reactor coolant, and 5) inadvertent boron dilution events could be identified and terminated by operator action prior to the reactor returning critical.

Thus, a reduction in RHR flow would not create the possibility of an accident which is different than any already evaluated in the FSAR.

The probability of a malfunction of equipment important to safety previously evaluated in the FSAR will not be increased.

When the reactor coolant level in the RCS loop piping is lowered, there is a potential for air to be drawn into the RHRS suction line (air entrainment) due to RCS loop level fluctuations and/or the development of a vortex. Air entrainment into the RHRS could cause air binding of the RHR pumps and thus, result in the inadvertent loss of decay heat removal capability. The tendency for vortex formation at the RHR suction line and, subsequent air entrainment into the RHRS, is a function of the water level above the RHRS suction nozzle and the RHR flowrate. The lower the level, or the higher the RHR flowrate, the greater the potential for a vortex to develop and air to be drawn into the RHRS. Thus, the likelihood of vortex formation due to partial draining of the RCS can be offset by reducing the RHR flowrate.

Therefore, the probability of a malfunction of equipment important to safety previously evaluated in the FSAR would not be increased due to a reduction in RHR flowrate during mid-loop operation.

The consequences of a malfunction of equipment important to safety previously evaluated in the FSAR will not be increased.

A reduction in RHR flow during mid-loop operation will potentially impact those transients explicitly analyzed in Modes 5 and 6. The only non-LOCA event analyzed in these modes is the CVCS Malfunction. A CVCS Malfunction event in Mode 6 is prevented by administrative controls which isolate the RCS from any potential source of unborated water. Thus, a reduction in the minimum RHR flow requirement during mid-loop operation, only impacts the Mode 5 analysis. Since the proposed reduction in RHR flow to 1000 gpm during mid-loop operation would not invalidate these assumptions, the current Mode 5 analysis remains valid.

The required minimum RHR flowrate during mid-loop operations would be sufficient to ensure that inadvertent boron dilution events could be identified and terminated by operator action prior to the reactor returning critical. Thus, the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR would not be increased.

The possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR will not be created.

When the reactor coolant level in the RCS loop piping is lowered, there is a potential for air to be drawn into the RHRS suction line (air entrainment) due to RCS loop level fluctuations and/or the development of a vortex. Air entrainment into the RHRS could cause air binding of the RHR pumps and thus, result in the inadvertent loss of decay heat removal capability. The tendency for vortex formation at the RHR suction line and, subsequent air entrainment into the RHRS, is a function of the water level above the RHRS suction nozzle and the RHR flowrate. The lower the level, or the higher the RHR flowrate, the greater the potential for a vortex to develop and air to be drawn into the RHRS. Thus, the likelihood of vortex formation due to partial draining of the RCS can be offset by reducing the RHR flowrate.

The RHRS consists of two parallel RHR trains. The redundancy in the RHRS design provides the system with the capability to maintain its cooling function even with major single failures, such as a failure of an RHR pump, valve, or heat exchanger.

The possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR would not be created.

<u>The margin of safety as defined in the bases to any technical specification will not be reduced</u>.

Currently, the Byron/Braidwood Technical Specifications do not specify RHR flowrate requirements for operation in Mode 5. Mode 6 operations, however, require a minimum RHR flowrate of 2800 gpm. The Technical Specifications place limitations on the RHRS during mid-loop operation by specifying a minimum flow requirement for the purpose of decay heat removal and the number of RHR trains which must be operable. They do not, however, contain restrictions based on minimizing air entrainment in the RHRS as a result of vortexing which may occur during mid-loop operation under certain conditions.

The fuel cladding (fission product barrier) is protected in Modes 5 and 6 by providing cooling and maintaining core shutdown. Adequate decay heat removal is provided to address the cooling requirements, and sufficient mixing ensures that the boron dilution analyses remain valid. Therefore, the amount of time available to identify and terminate a boron dilution event is unaffected.

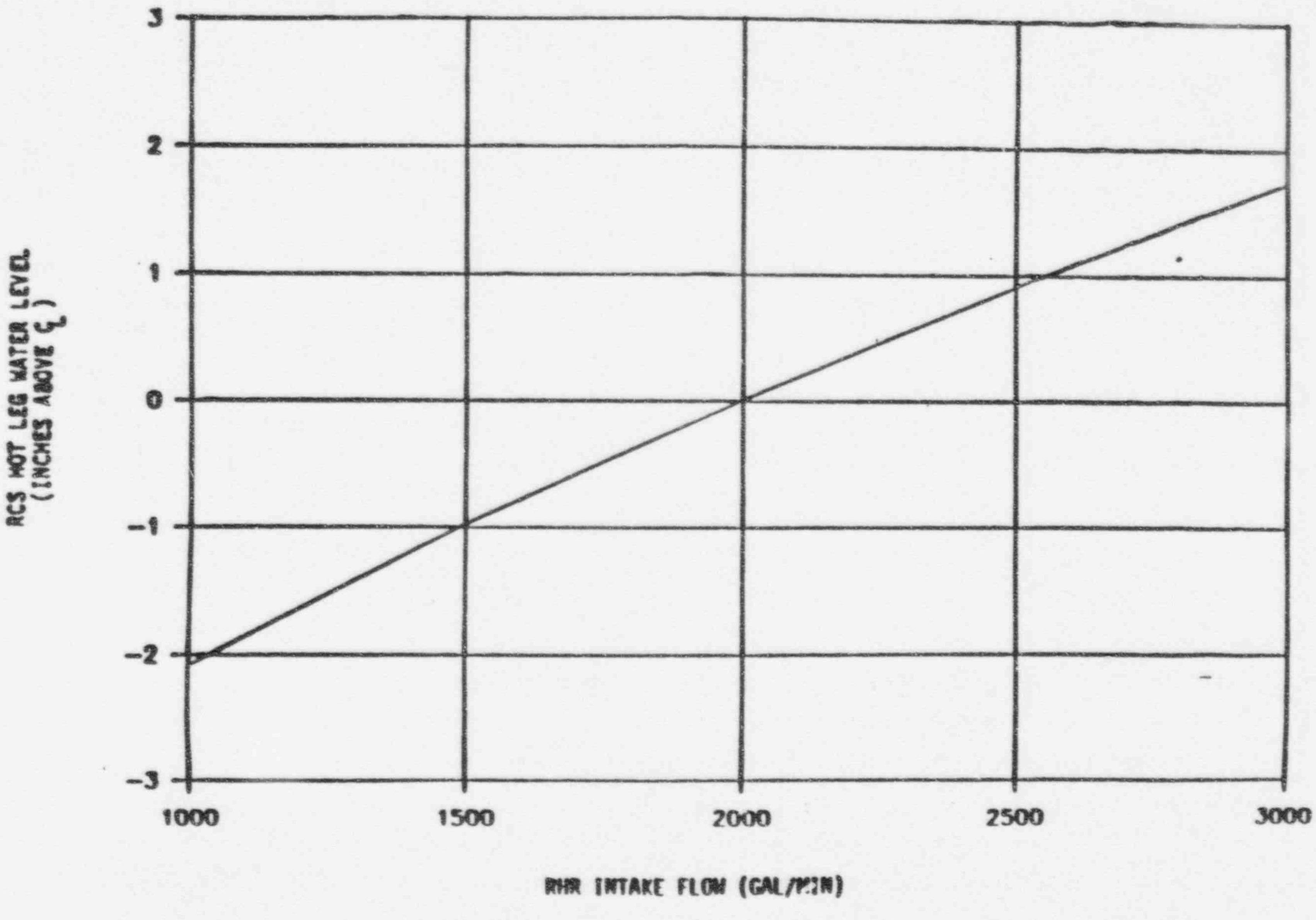
Thus, the margin of safety provided by the Technical Specification shutdown margin limits would not be reduced.

TABLE 1 - PUMP OPERATING MODES

MODE	FREQUENCY	DURATION	FLOW (gpm)	SUCTION PRESSURE (psia)
Surveillance Testing	4 per year	30 minutes	500	59
RHR Initiation	2 per year 2 per year	5 minutes 30 minutes	500 575	400 400
Plant Cooldown	2 per year	30.3 hours	3000	391
Refueling & Shutdown	2 per year	1 month	3000	38
Mid-Loop Operation	2 per year	2 weeks	1000 to 1300	26
Spurious SI	2 per year	30 minutes	500	48
Large LOCA	1 per 40 yr.	30 minutes	3800	41
Post-Accident Recirculation	1 per 40 yr.	1 year	3950	26

FIGURE 1

REQUIRED RCS WATER LEVEL.



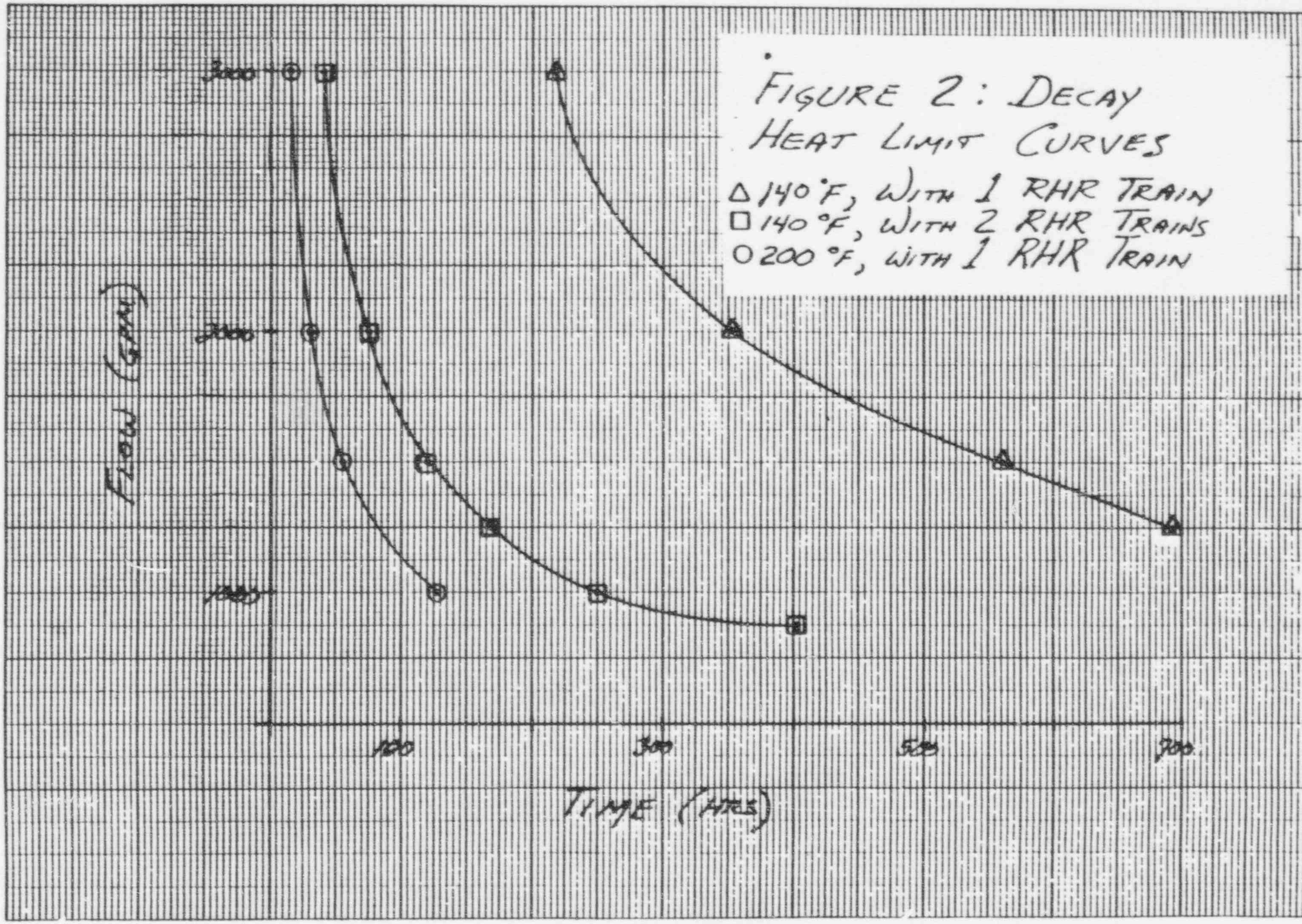


FIGURE 2: DECAY
HEAT LIMIT CURVES
△ 140°F, WITH 1 RHR TRAIN
□ 140°F, WITH 2 RHR TRAINS
○ 200°F, WITH 1 RHR TRAIN
3000
2000
1000
100
300
500
700
FLOW (GPM)
TIME (HRS)

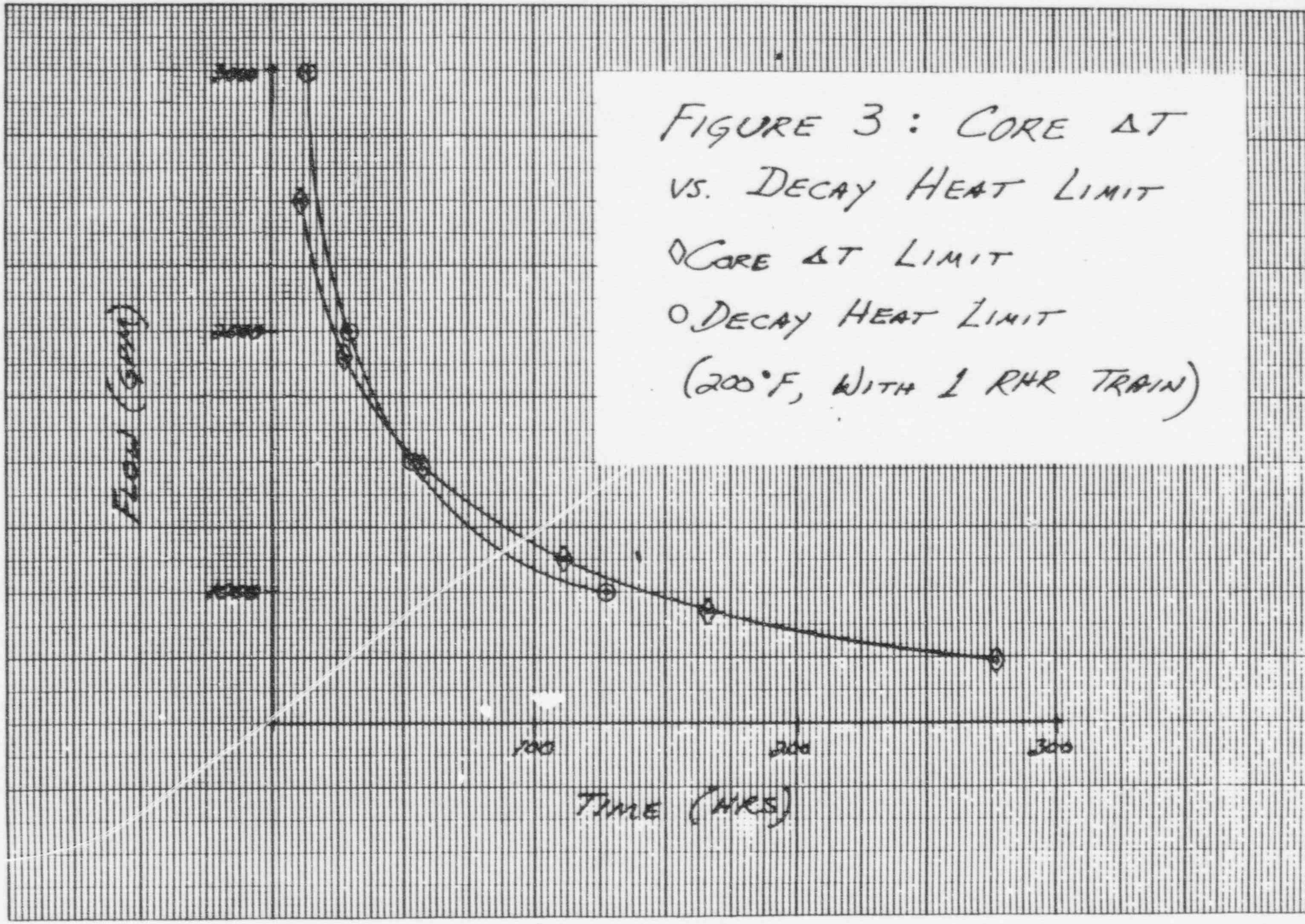


FIGURE 3: CORE ΔT VS. DECAY HEAT LIMIT