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REDUCTION IN THE MINIMUM RHRS FLOWRATE DURING MID-LOOP OPERATION FOR BYRON AND BRAIDWOOD POWER PLANT UNITS 1 AND 2

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REDUCTION IN THE MINIMUM RHR FLOWRATE DURING MID-LOOP OPERATION FOR BYRON AND BRAIDWOOD POWER PLANT UNITS 1 AND 2

1.0 Introduction

The purpose of this report is to document the evaluations and analyses performed by Westinghouse to support a reduction in the minimum Residual Heat Removal (RHR) flowrate during mid-loop operation for Byron and Braidwood Units 1 and 2.

Mid-loop operation occurs when the plant is operating with the Reactor Coolant System (RCS) partially drained in Modes 5 and 6. Currently, for Byron and Braidwood Units 1 and 2, there are no Technical Specification RHR flowrate requirements for operation in Mode 5. However, for Mode 6 operations, a minimum RHR flowrate of 2800 gallon per minute (gpm) is specified.

The Commonwealth Edison Company has requested that Westinghouse evaluate a reduction in the minimum RHR flowrate below the current Technical Specification requirement of 2800 gpm, for Byron and Braidwood Units 1 and 2. A lower RHR flow during mid-loop operation could reduce the potential for air binding of the RHR pumps.

The evaluations performed to address a reduction in the RHR specified flow requirement during mid-loop operation are consistent with the following concerns: (1) decay heat removal, (2) thermal considerations, (3) boron mixing and stratification, (4) control valve cavitation and (5) inadvertent boron dilution. In addition, a revised RHR pump thrust bearing expected life has been calculated based on reducing the minimum RHR flowrate below 2800 gpm.

Proposed revisions to the Byron/Braidwood Technical Specifications and Updated Final Safety Analysis Report (UFSAR) are provided to reflect a reduction in the minimum RHR flow requirements to below 2800 gpm.

2.0 RHRS System Description

The primary function of the Residual Heat Removal System (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 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), associated piping, valves, and instrumentation.

During RHRS operation, reactor coolant flows from the RCS to the RHR pumps, through the tube side of the residual heat exchanger, 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.

3.0 Vortex Formation and Air Entrainment

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 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.

4.0 RHR Flow Requirement Bases

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

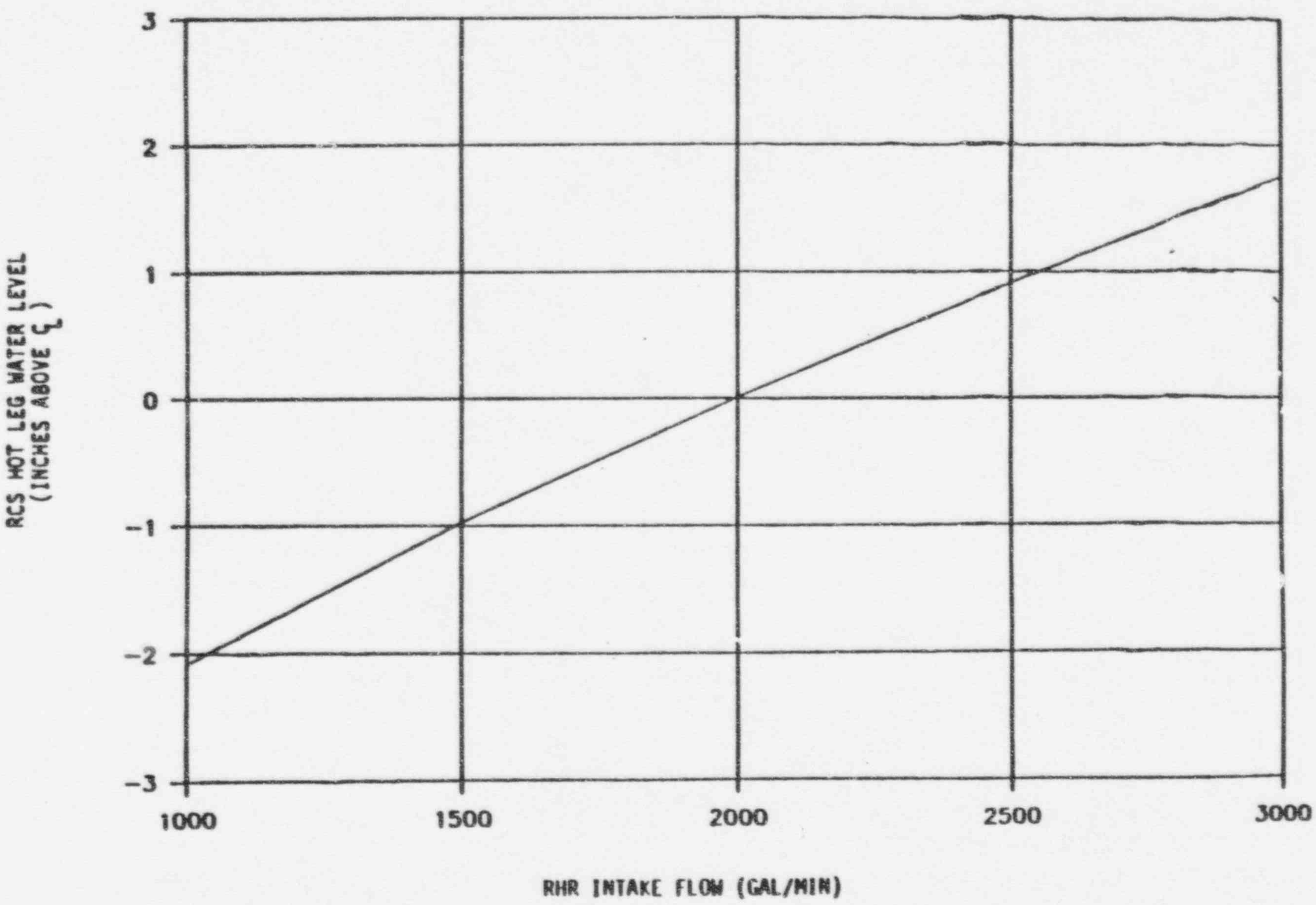
- The ability of the RHRS to remove decay heat such that RCS temperature can be controlled.

- 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.

FIGURE 1

REQUIRED RCS WATER LEVEL



- 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.

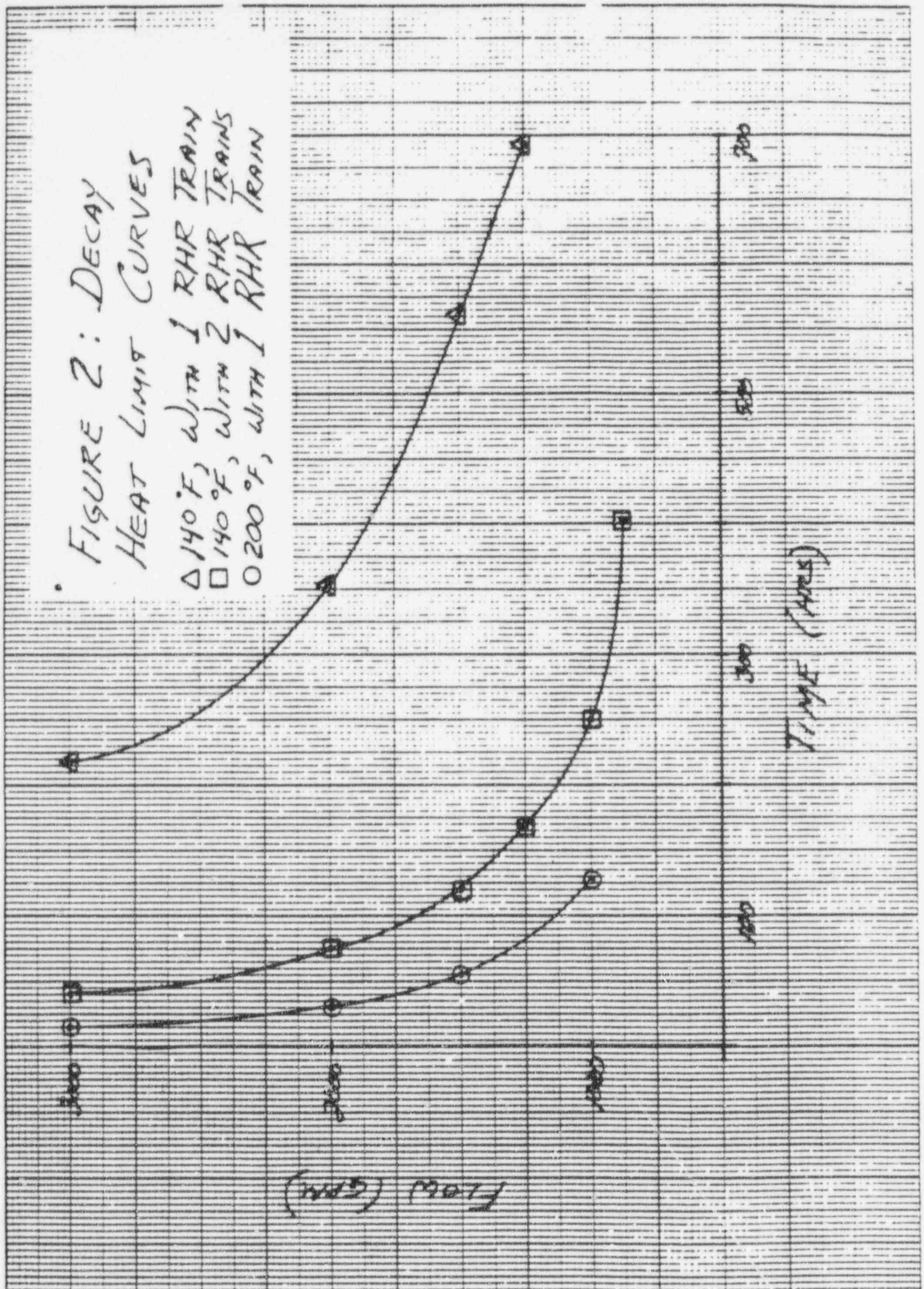


FIGURE 2: DECAY
HEAT LIMIT CURVES
△ 140 °F, WITH 1 RHR TRAIN
□ 140 °F, WITH 2 RHR TRAINS
○ 200 °F, WITH 1 RHR TRAIN
3000
2500
2000
FLOW (GPM)
TIME (HRS)

Flow enters the reactor vessel from each cold leg nozzle. The flow is forced to the bottom of the vessel and up through the reactor core. Thus, adequate mixing of the reactor vessel volume in order to minimize thermal stratification is expected at reduced RHR flows.

In addition to potential thermal stratification, a reduction in RHR flowrate will increase the reactor coolant temperature rise through the core. 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.

Boron Mixing and Stratification

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

RHR flow 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

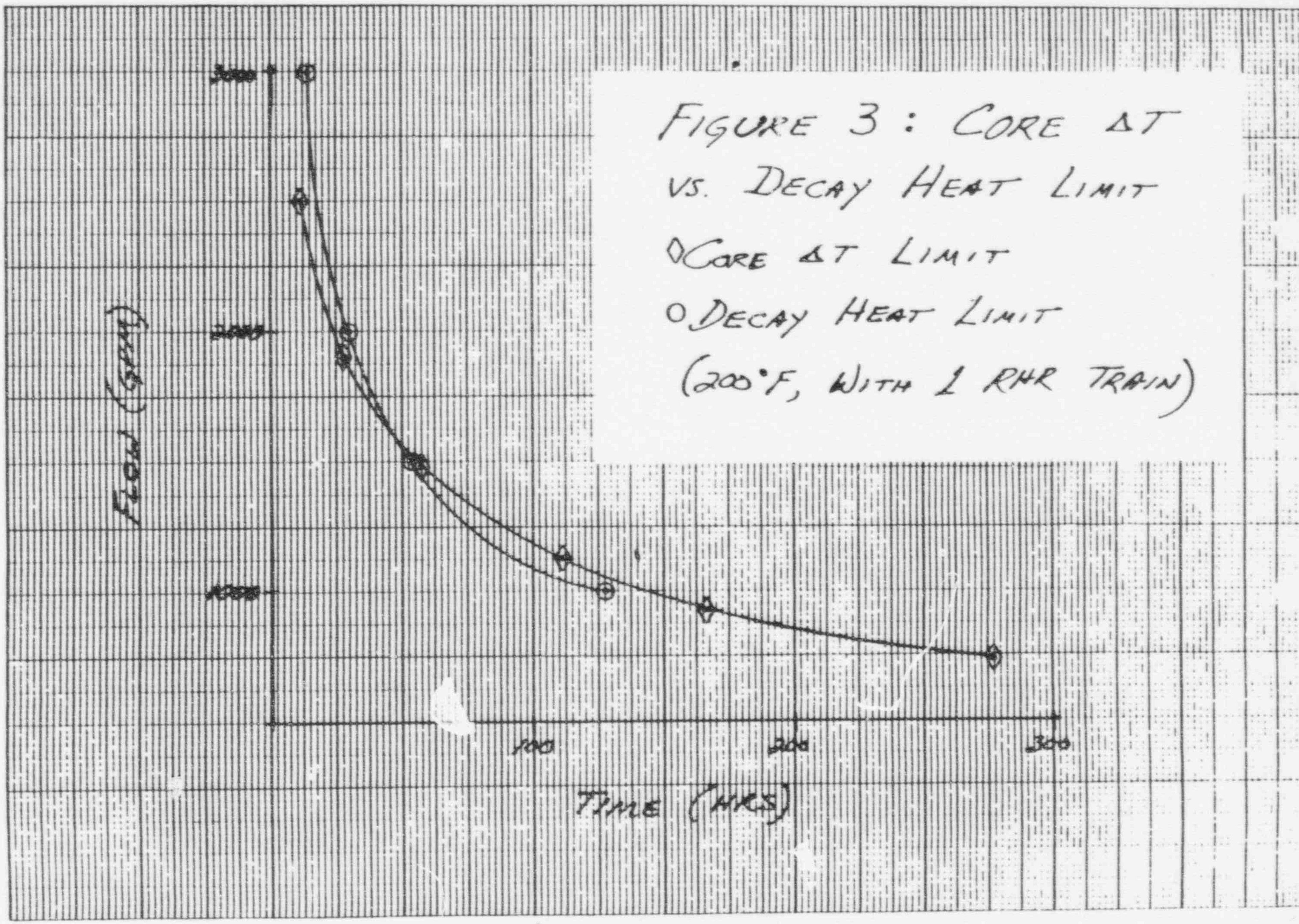


Figure 3: Core ΔT vs. Decay Heat Limit

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 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 an RHR flow limit 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 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 flowrate to significantly below 2000 gpm without harm. Any cavitation that is severe enough to cause damage would be evident from excessive noise and vibration in the piping downstream of the valve. At this time, Westinghouse recommends that the limit on RHR flowrate initially be established based on the other concerns identified (decay heat removal, thermal considerations, and boron mixing and stratification). The first time that RHR flow 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 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 Byron/Braidwood 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 Byron and Braidwood 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. Therefore, the results and conclusions presented in the Byron/Braidwood UFSAR remain valid.

5.0 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 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.

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

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 (approximately 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 [illegible] 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:

Life = 1500 (service factor)^3
Service Factor = (rated capacity) / (equivalent load)
Rated Capacity - defined by Handbook based on bearing model
Equivalent Load = 0.62 (radial load) + (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 the 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 incorporating 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.

6.0 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 prescribed minimum time after shutdown, elimination of dilution sources, avoidance of boration/dilution operations, or core temperature rise monitoring. Since the 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 would 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.

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

Based on the above discussion, 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. The affected pages are provided in Appendix A of this report.

7.0 FSAR Changes

The impact of a reduction in the minimum RHR flow requirements on the Byron/Braidwood Updated Final Safety Analysis Report (UFSAR) has been reviewed. It is recommended that Section 5.4.7.2, Residual Heat Removal System Design, be revised. The following additional information is provided with regard to mid-loop operation: (1) the concerns addressed in determining the required minimum RHR flowrate, (2) the specification of the RHR flow requirements consistent with plant conditions, and (3) the potential for vortex formation and air binding of the RHR pumps.

The Byron/Braidwood UFSAR has been marked-up to reflect the above changes. The mark-up is provided in Appendix A of this report.

8.0 Conclusions

Based on the evaluations and analyses performed by Westinghouse to support a reduction in the minimum RHR flowrate during mid-loop operation for Byron and Braidwood Units 1 and 2, the following is concluded:

- The minimum RHR flowrate dictated by decay heat and core delta T concerns is a function of time after shutdown, as is shown in Figure 2 and Figure 3.

- Cavitation of the reactor coolant may occur when RHR flow is reduced below approximately 2000 gpm, due to the increase in pressure drop across the flow control valve.

- A reduction in the minimum RHR flowrate requirement to 1000 gpm or greater during mid-loop operation, will not adversely affect the non-LOCA accident analyses.

In addition, 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 would be specified in the plant procedures for Byron and Braidwood Units 1 and 2.

APPENDIX A

TECHNICAL SPECIFICATION CHANGES

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is greater than or equal to 23 feet.

ACTION:

With no RHR loop OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.1 At least one RHR loop shall be verified in operation and circulating reactor coolant ~~at a flow rate of greater than or equal to [illegible] gpm~~ at least once per 12 hours.

*The RHR loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

REFUELING OPERATIONS

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two residual heat removal (RHR) loops shall be OPERABLE, and at least one RHR loop shall be in operation.

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status, or establish greater than or equal to 23 feet of water above the reactor vessel flange, as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2 At least one RHR loop shall be verified in operation and circulating reactor coolant ~~at a flow rate of greater than or equal to 2800 gpm~~ at least once per 12 hours.

*Prior to initial criticality, the RHR loop may be removed from operation for up to 1 hour per 2 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above the applicable design bases DNBR during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers. Single failure considerations require that two loops be OPERABLE at all times.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

INSERT

The restrictions on starting a reactor coolant pump with one or more RCS cold legs less than or equal to 350°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

The requirement to maintain the boron concentration of an isolated loop greater than or equal to the boron concentration of the operating loops ensures that no reactivity addition to the core could occur during startup of an isolated loop. Verification of the boron concentration in an idle loop prior to opening the stop valves provides a reassurance of the adequacy of the boron concentration in the isolated loop.

Startup of an idle loop will inject cool water from the loop into the core. The reactivity transient resulting from this cool water injection is minimized by delaying isolated loop startup until its temperature is within 20°F of the operating loops.

3/4.9.6 REFUELING MACHINE

The OPERABILITY requirements for the refueling machine and auxiliary hoist ensure that: (1) refueling machines will be used for movement of drive rods and fuel assemblies, (2) each refueling machine has sufficient load capacity to lift a drive rod or fuel assembly, and (3) the core internals and reactor vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE FACILITY

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool areas ensures that in the event this load is dropped: (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that: (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the core to minimize the effect of a boron dilution incident and prevent boron stratification.

INSERT

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of RHR capability. With the reactor vessel head removed and at least 23 feet of water above the reactor vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT PURGE ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

Byron/Braidwood Technical Specification Bases

Insert

The surveillance requirement verifies that the RHR loop is operating and circulating reactor coolant to ensure the capability of the RHR system to maintain compliance with plant design limits. The required RHR loop reactor coolant flowrate is determined by the flowrate necessary to: (1) provide sufficient decay heat removal capability, (2) maintain the reactor coolant temperature rise through the core within design limits, for compliance with flowrates assumed in the boron dilution analysis, (3) prevent thermal and boron stratification in the core, (4) preclude cavitation of the reactor coolant downstream of the RHR flow control valve, and (5) ensure that inadvertent boron dilution events can be identified and terminated by operator action prior to the reactor returning critical.

In addition, during operation of the RHR loop with the water level in the vicinity of the reactor vessel nozzles, the RHR loop flowrate determination must also consider the RHR pump suction requirements. At this water level, the RHR pump can experience vortexing or cavitation conditions which would cause the loss of RHR pump operation, if the flowrate demand is too high. Operation with reactor coolant water at this level is often called mid-loop operation. Care must be taken in determining the RHR loop flowrate, when operating with water level in this region, to prevent loss of the RHR pump and subsequent loss of the RHR loop for decay heat removal.

APPENDIX B

FSAR CHANGES

If one RHRS pump is out of service and the alternate train becomes unavailable due to a passive failure during cooldown, the auxiliary feedwater system, along with the steam generator safety valves and steam generator power-operated relief valves, provides a completely separate, independent, and diverse means of performing the safety function of removing residual heat, which is normally performed by the RHRS when the RCS temperature is less than 350°F. The auxiliary feedwater system is capable of performing this function for an extended period of time following plant shutdown until the RHRS is made available.

When the steam generators are down for maintenance, the RCS is depressurized and the RHRS operates at steady-state pressure and temperature conditions significantly below the RHRS design values. Passive failures of magnitude that could affect RHRS operation are not expected to develop at these conditions.

However, if one RHRS pump is out of service and the steam generators are down for maintenance, the development of a passive failure in the remaining RHRS train would not make that train unavailable for residual heat removal since in-service inspections are conducted periodically and ASME "code-allowable" defects are not expected to grow appreciably during the life of the plant. A passive failure of the RHRS piping is not expected to produce a rapidly propagating crack that could result in a major rupture of a system pipe. Therefore, a detectable leakage crack is not expected to produce the effect of rendering an RHRS train inoperable. The operator would continue to use the RHRS train in conjunction with the chemical and volume control system. The centrifugal charging pump(s) will provide the makeup supply to compensate for the system inventory leakage.

Loss of RHRS cooling during maintenance activities and especially due to air entrainment in the system has been considered. INSERT ~~If it is required that the water level be lowered to drain the steam generator tubes, the residual heat removal flow rate is throttled to about 1500 gpm through each of the residual heat removal loops.~~ Draining is to the point where the indicated level is stable and at the elevation of the center of the reactor vessel nozzles. At this point, the reactor coolant level is monitored continuously to assure that the RHRS inlet lines do not become uncovered. Inventory makeup, if required, can be accomplished via the chemical and volume control system (CVCS)/centrifugal charging pumps(s).

Should a RHRS inlet line become uncovered, air may be drawn into the suction piping and entrained in the fluid. Factors which minimize the effects of air entrainment on pump performance are as follows:

1. the location of the pumps provides positive head on the pump inlet, and

Insert

During mid-loop operations, the RHRS operates at reduced flowrates to prevent air binding of the RHR pumps. Since the minimum RHR flow requirement is dependent on plant conditions, a single flowrate requirement is not specified in the Technical Specifications. Instead, acceptable RHR flowrates (minimum and maximum) that are consistent with the plant conditions are specified in the plant procedures. The acceptability of the reduced minimum RHR flow requirement 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.

Attachment C

B. S. Humphries (Westinghouse) letter (CCE-96-207) to J. R. Meister (ComEd), "Commonwealth Edison Company, Braidwood Units 1 & 2,Safety Evaluation SECL-96-193, RHR Operation During Reduced Inventory Conditions," dated October 18, 1996

Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

CCE-96-207
October 18, 1996

Mr. J. R. Meister
Commonwealth Edison Company
Braidwood Nuclear Station
Rural Route #1, Box 84
Braceville, IL 60407

Commonwealth Edison Company
Braidwood Units 1 & 2
Safety Evaluation SECL-96-193
RHR Operation During Reduced Inventory Conditions

Dear Mr. Meister:

Enclosed for formal transmittal to Commonwealth Edison, Braidwood Station is Safety Evaluation SECL-96-193, "RHR Operation During Reduced Inventory Conditions," for Braidwood Units 1 and 2.

Please contact us if you have any questions.

Very truly yours,

WESTINGHOUSE ELECTRIC CORPORATION

Janet M Bunecicky for

B. S. Humphries, Manager
Commonwealth Edison Project
Operating Plant Programs

Enclosure

cc: R. Krbec Braidwood
J. Sanchez Braidwood
T. Tulon Braidwood

0985jb

SECL No. 96-193

Customer Reference No(s).

Westinghouse Reference No(s).

WESTINGHOUSE
SAFETY EVALUATION CHECK LIST

1.) NUCLEAR PLANT(S): **Braidwood Units 1& 2**

2.) SUBJECT (TITLE): **RHR Operation During Reduced Inventory Conditions**

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 ___ No X 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 ___ No X 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 which is 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 described in the bases to any technical specification be reduced?

NOTES:

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

If the answer to any of the above questions in Part A (3.4) or Part B cannot be answered in the negative, based on written safety evaluation, the change review would require an application for license amendment as required by 10CFR50.59(c) and submitted to the NRC pursuant to 10CFR50.90.

5.) REMARKS:

The answers given in Section 3, Part A, and Section 4, Part B, of the Safety Evaluation Checklist, are based on the attached Safety Evaluation.

FOR FSAR UPDATE

Section: N/A Pages: N/A Tables: N/A Figures: N/A

SAFETY EVALUATION APPROVAL LADDER:

Nuclear Safety Preparer: B A Maurer Date: 10/18/96
B. F. Maurer

Nuclear Safety Verifier: G.W. White Date: 10/18/96
G. W. Whiteman

Licensing Engineer Review: RJ Morrison Date: 10/18/96
R. J. Morrison

SAFETY EVALUATION
RHR OPERATION DURING REDUCED INVENTORY CONDITIONS
BRAIDWOOD UNITS 1 & 2

1.0 BACKGROUND

In response to Generic Letter 87-12 (Reference 1), Commonwealth Edison (ComEd) made the commitment that RHR flow would be reduced to approximately 1000 gpm through each RHR train prior to RCS draindown and while the RCS is in a drained down condition (Reference 2). At reduced inventory conditions, normally one RHR train is in operation, while the other RHR train is operabl but not operating. If a condition should occur that would affect the operating RHR train, the other train would be available for continued decay heat removal. ComEd maintains the necessary emergency procedures for recovery from such an event.

The Reactor Coolant System (RCS) water level is lowered in Modes 5 and 6 during outages to facilitate miscellaneous maintenance activities. Operation with the RCS partially drained may 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, and/or the higher the RHR flowrate, the greater the potential for a vortex to develop and air to be drawn into the RHRS.

In order to facilitate RCS cooldown while maintaining adequate margin to prevent air entrainment in the RHR system, resulting in cavitation of the operating RHR pump, the effect of implementing an increased RHR flow rate of 3300 gpm at reduced inventory conditions has been evaluated. This evaluation demonstrates that extending the approved range of operation of an RHR train from 1000 gpm to 3300 gpm at or above prescribed minimum RCS water levels, as shown in Figure 1 to this safety evaluation, does not represent an unreviewed safety question per the criteria of 10CFR50.59 and will not require a change to the Technical Specifications.

2.0 LICENSING BASIS

Title 10 of the Code of Federal Regulations, Part 50, Section 59 (10 CFR 50.59) allows the holder of a license, authorizing operation of a nuclear power facility, the capacity to investigate and disposition a change to the normal plant configuration. The increase in RHR flow rate at reduced inventory

conditions represents a change to the normal plant configuration. Prior Nuclear Regulatory Commission (NRC) approval is not required to implement a change provided that the proposed change does not involve an unreviewed safety question or result in a change to the plant technical specifications. However, it is the obligation of the licensee to maintain a record of the changes or modifications to the facility, to the extent that such changes impact the Updated Final Safety Analysis Report (UFSAR). The code further stipulates that these records shall include a written safety evaluation that provides the basis for the determination that the change does not involve an unreviewed safety question. It is the purpose of this document to support the requirement for a written safety evaluation.

The RHR system is described in UFSAR Section 5.4.7, "Residual Heat Removal System," and refueling operations and operations during Mode 6 are discussed in UFSAR section 9.1 (Reference 3). RHR operational criteria are further discussed in Reference 2. Technical Specification 3/4.9.8.2, "Refueling Operations, Low Water Level" (Reference 4) presents the minimum requirements for RHR operation during Mode 6 with a water level less than 23 feet above the vessel flange.

3.0 EVALUATION

The primary function of the RHR system 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 and the refueling cavity before and after the refueling operations. The RHR system 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 RHR system 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 RHR system 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 circulating through the shell side of the residual heat exchangers.

The required minimum RHR flow rate during reduced inventory operations is based on the following functional considerations: 1) provide sufficient decay heat removal capability, 2) maintain the reactor coolant temperature rise through the core within design limits for compliance with flow rates assumed in the boron dilution analysis, 3) prevent thermal and boron stratification in the core, 4) preclude cavitation of the reactor coolant downstream of the RHR flow control valve, and 5) ensure that inadvertent boron dilution events can be identified and terminated by operator action prior to the reactor returning critical. These issues have been evaluated and presented in Safety Evaluation SECL-89-867 (Reference 5) and WCAP-12207 (Reference 6). However, since the desired change is

to increase the RHR flow rate at reduced inventory conditions, the possibility of vortexing and air entrainment becomes the dominant concern.

The sensitivity of air entrainment on RHR pumps to RCS level and pump flow rate is of increased importance at reduced inventory conditions. Based on operating experience and various test programs, guidelines have been developed regarding required water level for various RHR flow rates. Correlations between RCS hot leg water level and RHR intake flow rate have been developed for RHR operation with a partially filled system. Correlations applicable to the Braidwood units are presented in References 5 and 6. The relationship of maximum allowable RHR flow rate versus minimum reduced inventory water level (relative to the centerline of the hot leg pipe) is provided for the range of RHR flows from 1000 gpm to 3000 gpm.

The supporting calculations that provided the basis for the flow versus water level correlations in References 5 and 6, and WCAP-11916 (Reference 7), have been reviewed. The calculations have been determined to remain applicable for the reduced inventory conditions at Braidwood. As a result of this review, it is concluded that the range of RHR flows of 1000 gpm to 3000 gpm can be reasonably extended to a flow rate of 3300 gpm in one RHR train. The minimum allowable water level in the RCS with an RHR flow rate of 3300 gpm has been calculated to be 2.25 inches above the centerline of the hot leg pipe. The calculational methodology used in this calculation is supported by the test data for calculated Froude numbers between 1 and 3. The 3300 gpm flow rate in this application results in a calculated Froude number of 2.3, which is well within the bounds or applicability of the methodology. Maintaining the water level at or above this level empirically limits the air entrainment to acceptable levels for RHR pump operation.

The concerns previously addressed in References 5 and 6 are not exacerbated by the increase in RHR flow to 3300 gpm. Decay heat removal capacity will be increased, with a subsequent decrease in the time required to remove the decay heat and maintain the RCS at the desired temperature. The increased flow rate will also result in improved thermal mixing in the reactor vessel and will minimize the temperature rise across the core. It will also result in increased boron mixing throughout the RCS. With increased RHR flow, the likelihood of cavitation of reactor coolant across the RHR control valves is further reduced.

The effects on non-LOCA transients in the previous evaluations (References 5 and 6) were related to minimum RHR flow rate assumptions and the ability of the RHR to provide adequate circulation to prevent boron stratification in the RCS. The increase in RHR flow rate to 3300 gpm will be beneficial to both of these concerns, and thus, there will be no adverse effects on the non-LOCA transients.

4.0 DETERMINATION OF UNREVIEWED SAFETY QUESTION

Increasing the RHR flow rate to a maximum of 3300 gpm during reduced inventory conditions at Braidwood Units 1 and 2 with the attendant limitation on water level has been evaluated using the guidance of NSAC-125 and does not involve an unreviewed safety question, per the criteria of 10CFR50.59, on the basis of the following justification.

4.1 Will the probability of an accident previously evaluated in the UFSAR be increased?

No. A change in RHR flow rate during reduced inventory operation will potentially impact transients for which a minimum RHR flow rate assumptions are specified. Since the change represents an increase in the RHR flow rate, no transients will be adversely affected. Also, it is necessary to assure adequate circulation of coolant in the RCS to prevent boron stratification and support the boron dilution transient mixing assumptions. The increase in flow rate is beneficial to maintaining the appropriate mixing assumptions. The evaluation further provides the acceptable limiting relationship between the maximum allowable RHR flow rate with the coincident allowable minimum RCS water level to preclude vortexing and subsequent air entrainment in the operating RHR train, thus maintaining RHR operability under reduced inventory conditions.

4.2 Will the consequences of an accident previously evaluated in the UFSAR be increased?

No. A change in RHR flow rate during reduced inventory operation will potentially impact transients for which a minimum RHR flow rate assumptions are specified, as well as boron mixing assumptions. As discussed in the response to Question 4.1, the increase in RHR flow rate to 3300 gpm will not invalidate the Braidwood accident assumptions, and thus, the consequences of the accidents evaluated in the UFSAR would not be increased. The acceptable limiting relationship for maximum allowable RHR flow rate versus allowable minimum RCS water level is provided to preclude vortexing and subsequent air entrainment in the operating RHR train. Thus, RHR operability under reduced inventory conditions will be maintained.

4.3 May the possibility of an accident which is different than any already evaluated in the UFSAR be created?

No. Acceptable RHR flow rates versus specific reduced inventory water level conditions have been provided. This relationship assures that vortexing and air entrainment of the RHR system will be avoided. Also, normally only one RHR train will be in operation at any given time. Thus, should RCS conditions result in the impairment of the operating RHR train, the other train which is maintained in an operable condition per the Technical Specifications can be used to maintain the decay heat removal function. The other considerations which can be impacted by a change in RHR flow rate (decay heat removal, thermal and boron mixing in the vessel and RCS, and control valve cavitation) have also been evaluated. The increase in RHR flow rate does not adversely affect these considerations. No new accident is created and no new single failures have been identified. Safety related systems and equipment required to mitigate the consequences of

postulated accidents are unaffected and will operate as required. For those cases where two trains are in service, no new accident will be created.

4.4 Will the probability of a malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

No. When the reactor coolant level in the RCS loop piping is lowered, there is a potential for air to be drawn into the RHR suction line due to RCS loop level fluctuations and/or the development of a vortex. Air entrainment into the RHR system 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 is a function of the RCS water level and the RHR flow rate. An acceptable limiting relationship for maximum allowable RHR flow rate versus allowable minimum RCS water level is provided to preclude vortexing and subsequent air entrainment in the operating RHR train, assuring the operability of the RHR system under reduced inventory conditions.

4.5 Will the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

No. The maximum allowable RHR flow rate as a function of RCS water level is sufficient to preclude vortex formation which could lead to pump cavitation and loss of the operating RHR train. Maintaining the prescribed flow rate/water level will preclude air entrainment due to vortexing and subsequent loss of RHR operation. Plant systems assumed operable to mitigate the consequences of an accident are will remain operable. There is no change to any analysis assumptions due to the malfunction of safety related equipment resulting from the increase in RHR flow rate.

4.6 May the possibility of a malfunction of equipment important to safety different than any already evaluated in the UFSAR be created?

No. Acceptable RHR flow rates versus specific reduced inventory water level conditions have been provided. This relationship assures that vortexing and air entrainment of the RHR system will be avoided. Also, normally only one RHR train will be in operation at any given time. Thus, should RCS conditions result in the impairment of the operating RHR train, the other train which is maintained in an operable condition per the Technical Specifications can be used to maintain the decay heat removal function. The increase in RHR flow rate does not create any new failure modes that could adversely impact safety related equipment. No new equipment malfunctions have been introduced. For those cases where two trains are in service, no new equipment malfunction will be created.

4.7 Will the margin of safety as defined in the bases to any technical specifications be reduced?

No. The increase in RHR flow rate does not violate the RHR flow rate requirements for Mode 6 in the Byron/Braidwood Technical Specifications. The Technical Specifications place limitations on the RHR system by specifying a minimum flow requirement for the purposes of decay heat

removal,maintain the reactor coolant temperature rise through the core within design limits, 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. They do not, however, contain restrictions based on minimizing air entrainment in the RHRS as a result of vortexing which may occur during reduced inventory operation under certain conditions. As stated in the Technical Specification Bases, operation of the RHR loop with the water level in the vicinity of the reactor vessel nozzles can result in vortexing or cavitation conditions which could cause the loss of RHR pump operation. Care must be taken in determining the RHR flow rate to prevent the loss of the RHR pump and subsequent loss of the RHR loop for decay heat removal. The analysis presented in this safety evaluation supports the basis for Technical Specification 3/4.9.8 to assure the operability of the RHR system. Thus, the margin of safety provided by the Technical Specification shutdown margin limits is not reduced.

5.0 CONCLUSIONS

An increase in RHR flow rate during Modes 5 and 6, specifically when the RCS is in a reduced inventory condition has been evaluated for Braidwood Units 1 and 2. The effect on potential vortexing at the RHR suction has been evaluated and acceptable RHR flow versus RCS water level limits have been provided to preclude loss of the RHR heat removal capability. It has demonstrated the acceptability of extending the approved range of operation of an RHR train to a range of 1000 gpm to 3300 gpm. Also, previous considerations for RHR operation have been reviewed and determined to be unaffected by the increase in RHR flow rate. Therefore, it is concluded that the increase in RHR flow rate in accordance with the recommended limits on RCS water level does not represent an unreviewed safety question as defined in 10CFR50.59, and does not require a change to the plant technical specifications.

6.0 REFERENCES

1. U.S. NRC Generic Letter 87-12, "Loss of Residual Heat Removal (RHR) While the Reactor Coolant System (RCS) is Partially Filled."

2. Commonwealth Edison letter to the NRC, Morgan to Miraglia, dated September 25, 1987.

3. Updated Final Safety Evaluation Report (UFSAR) for Byron/Braidwood Stations, Commonwealth Edison Company.

4. NUREG-1276, Technical Specifications for Braidwood Station Units 1 and 2, Docket Nos. 50-456 and 50-457.

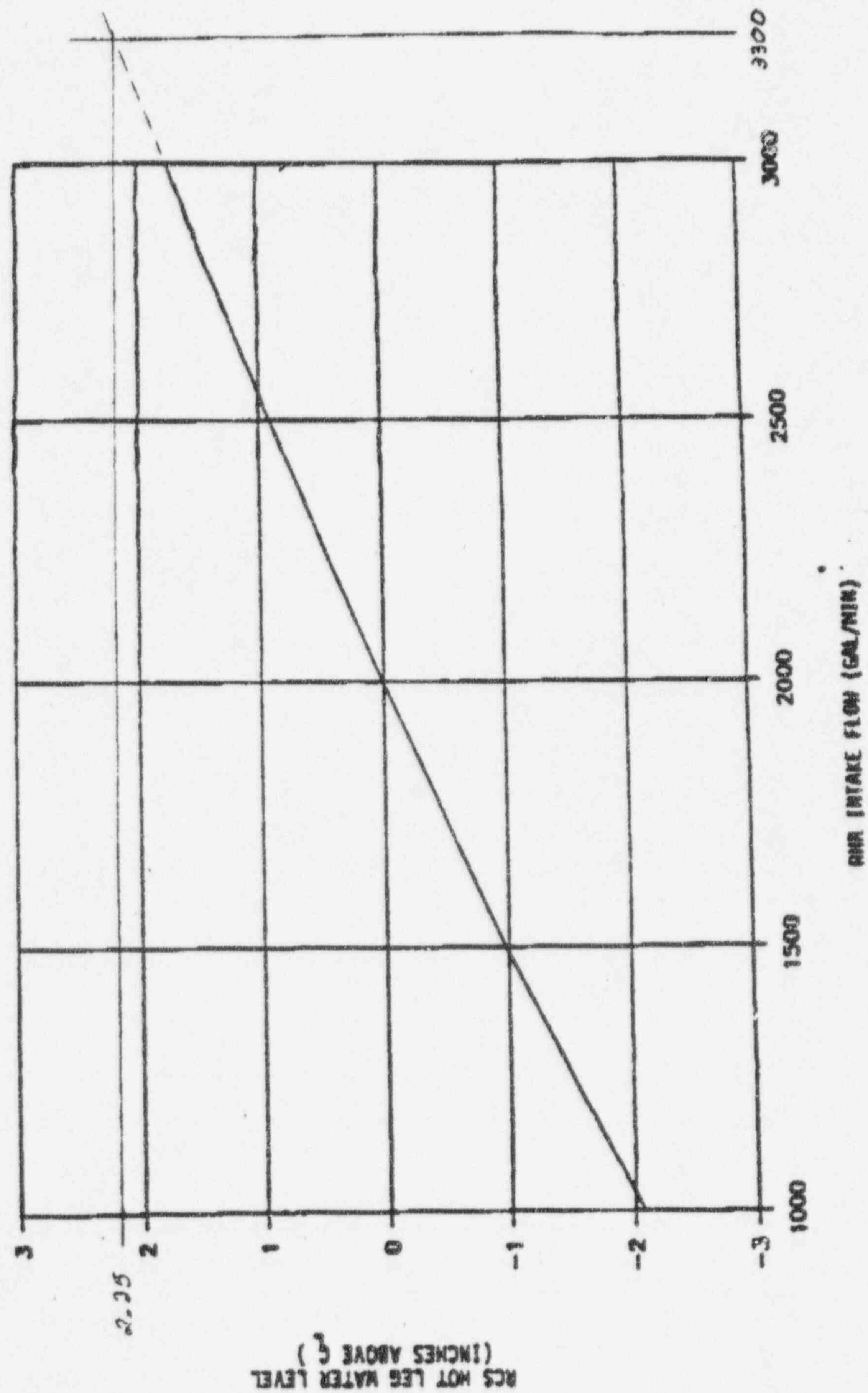
5. Westinghouse Safety Evaluation SECL-89-867, "Reduction in the Minimum RHR Flowrate During Mid-Loop Operation," Byron & Braidwood Units 1 & 2, July 1989.

6. WCAP-12207, "Reduction in the Minimum RHRS Flowrate During Mid-Loop Operation for Byron and Braidwood Power Plant Units 1 & 2," June 1989.

7. WCAP-11916, "Loss of RHRS Cooling While the RCS is Partially Filled," Revision 0, July 1989, prepared for use by the members of the Westinghouse Owners Group.

FIGURE 1

REQUIRED RCS WATER LEVEL



10CFR50.59 SAFETY EVALUATION VALIDATION FORM

1. LIST the documents implementing the proposed activity (eg: Modification Number, etc.) and include revision data (where appropriate).
   Westinghouse SECL No. 96-193; RHR Flow During Reduced Inventory Conditions; BwOP RC-4, Revision 6 (1000-3300 GPM FLOW REFERENCES ONLY)

2. IDENTIFY the activity addressed by the Safety Evaluation being validated.

   ☒ Same activity as listed above in Step 1. Proceed to Step 3.
   ☐ Different activity:
   a. List the documents implementing that activity (eg: Modification Number) and include revision date (where appropriate).
      ____
      ____
   b. Describe and justify any differences (eg: different train, unit, station, etc.) between the two activities.
      ____
      ____
      ____

3. REVIEW the Safety Evaluation being validated and ensure the following:
   a. The Safety Evaluation was performed by a procedure implementing NOD-TS.11; such as:
      - BwAP 1205-6T1, Revision 1 or later
      - BAP 1210-5T1, Revision 1 or later
      - ENC-QE-06.1, Revision 3 or later
   b. The proposed activity does not extend beyond the plant mode bounds assumed in the Safety Evaluation.
   c. There are no know facility changes (eg: Modifications, etc.) in place since the Safety Evaluation was written which would invalidate it.
   d. The Safety Evaluation conclusions are reasonable, well supported and documented, and have determined that prior NRC review/approval is NOT required.

4. PROCESS this Safety Evaluation Validation as follows:

   a. Attach a copy of the Safety Evaluation.
   b. Complete the Preparer & Reviewer verifications below and sign.
   c. Initiate an OSR to address the proposed activity and this Validation.

PREPARER of this Validation (must be qualified per BwAP 1205-6T7):

(Print) J. Tolar (Sign) J. Tolar (Date) 10/18/96

REVIEWER of this Validation (must be qualified per BwAP 1205-6T7):

(Print) R. Krbec (Sign) R. Krbec (Date) 10/18/96

(Final)