APPLICATION FOR AMENDMENT

TO

FACILITY OPERATING LICENSE NUMBER NPF-3

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NO. 1

Attached are requested changes to the Davis-Besse Nuclear Power Station, Unit No. 1 Facility Operating License No. NPF-3. Also included are the Technical Description and Significant Hazards Consideration.

The proposed changes (submitted under cover letter Serial Number 1634) concern:

Section 3/4.3.1.1, Reactor Protection System Instrumentation, Table 3.3-2, Reactor Protection System Instrumentation Response Times.

By: ______________________
D. C. Shelton, Vice President, Nuclear

Sworn and subscribed before me this 21st day of February, 1989.

Laurie A. Hinkle
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Notary Public, State of Ohio

LAURIE A. HINKLE
Notary Public, State of Ohio
My Commission Expires May 15, 1991

The following information is provided to support issuance of the requested changes to the Davis-Besse Nuclear Power Station, Unit No. 1 Operating License Number NPF-3, Appendix A, Technical Specification, Section 3/4.3.1.1, Table 3.3-2.

A. Time Required to Implement: This change is to be implemented within 45 days after the NRC issuance of the License Amendment or prior to startup from the sixth refueling outage, whichever is later.

B. Reason for Change (License Amendment Request Number 89-0001):

The present response time of 451 milliseconds is close to the physical limit of the system. The revised response time is allowed by new analyses and will provide additional margin for surveillance testing. The increased response time may prevent unnecessary plant outage time.

C. Technical Description: See attached Technical Description (Attachment 1).

D. Significant Hazards Consideration: See attached Significant Hazards Consideration (Attachment 2).

TECHNICAL DESCRIPTION

Description of Proposed Technical Specification Change

The purpose of this technical description is to describe a proposed change to the Davis-Besse Nuclear Power Station Technical Specifications Table 3.3-2, Reactor Protection System Instrumentation Response Times. The change proposes an increase in the response time requirement for the High Flux/Number of Reactor Coolant Pumps On (power/pumps) trip function of the Reactor Protection System (RPS) provided in Table 3.3-2 (Item number 8) from 451 milliseconds to 631 milliseconds. This proposed change is based on an analysis performed with the NRC, topically approved, computer code: VIPRE (Reference 1).

Systems Affected

The proposed change affects the Reactor Protection System (RPS). However, increasing the response time in the Technical Specification does not affect the present logic configuration and hardware design of the RPS.

Safety Function of Systems Affected

The safety function of the RPS is to trip the reactor when an unsafe condition; e.g. high flux, high pressure or Departure from Nucleate Boiling (DNB), is approached. The Technical Specification definition of the reactor protection system response time is "that time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until power interruption at the control rod drive breakers". The Updated Safety Analysis Report (USAR) Chapter 15 safety analyses also includes the control rod drive release delay and a dedicated margin for uncertainty. The safety function of the Power/Pumps trip is to provide DNB protection for (1) multiple Reactor Coolant Pump (RCP) coastdowns, (2) single RCP coastdowns from partial pump operation or (3) RCP coastdowns resulting in the loss of both pumps in either loop.

Each RCP has four current monitors that provide running/not running status of the associated RCP to each of the four pump monitor relay cabinets. The four pump monitor relay cabinets in turn provide the interface with four RPS channels. The RPS channels utilize the pump monitor relay cabinet inputs to determine the total number of RCPs running and in which RCS loops they are running. Once it determines the total number of RCPs running and in which RCS loops they are running, it utilizes this information as a setpoint for the power to pumps reactor trip bistable. The bistable receives an analog signal from the power range channel proportional to the reactor power level. The bistable trips when the reactor power signal exceeds the setpoint signal input from the contact monitor. For a coastdown of a single RCP from a four RCP initial condition, the flux/delta flux/flow trip ensures that there is sufficient flow, i.e., heat removal capability for the reactor power level while the reactor is automatically running back in power. Therefore, the power/pumps trip is only needed to provide protection for those instances mentioned previously.

Effects on Safety

The Loss of Forced Reactor Coolant Flow transient is described in Section 15.2.5 of the USAR. This analysis was performed using the closed-channel computer code RADAR, as summarized in the NRC approved B&W topical report - BAW-10069A, Rev. 01, (Reference 2) to determine DNB as a function of time.

The most limiting transient for the Loss of Forced Reactor Coolant Flow that relies on this protective feature of the RPS is the four pump coastdown transient. This analysis summarized in Section 15.2.5 of the USAR uses a total response time of 620 milliseconds. The 620 milliseconds is comprised of 371 milliseconds for the sensor and RPS delay (including 240 milliseconds for the pump monitor), 80 milliseconds for the control rod drive (CRD) breaker delay, 125 milliseconds for the CRD release delay and 44 milliseconds for a dedicated margin for uncertainty. The resultant calculated minimum DNB Ratio (DNBR) using the Westinghouse correlation W3R is 1.49. The four pump coastdown has been reanalyzed twice since the conduct of the analysis provided in the Davis-Besse FSAR. The first time was to account for densification effects. This reanalysis, summarized in References 3 and 4, resulted in a minimum DNBR, using the Babcock & Wilcox correlation B&W-2 and a response time of 620 milliseconds, of 1.816. The second reanalysis, summarized in Reference 5, was during Cycle 1 when the Burnable Poison Rod Assemblies (BPRAs) and Orifice Rod Assemblies were removed from the core. This reanalysis resulted in a minimum DNBR of 1.85 also using the B&W-2 correlation and a response time of 620 milliseconds. The acceptance criteria for the four pump coastdown transient is a minimum DNBR greater than or equal to 1.30.

Subsequent to the completion of the above analyses, crossflow codes, i.e., open channel codes have been developed and approved by the NRC. Crossflow codes can predict the flow redistribution effects in an open lattice reactor core. The crossflow methodology provides significant DNBR improvements (by allowing the coolant to mix) over the traditional closed-channel methodology.

A reanalysis of the four pump coastdown was performed by Toledo Edison using VIPRE-01 (Reference 1). This code utilizes crossflow. First the VIPRE code was benchmarked against the LYNXT (Reference 6) crossflow code. The benchmark chosen to perform was a comparison to the locked rotor transient. A reanalysis of this particular transient had been performed by B&W for Davis-Besse Cycle 6 using LYNXT as summarized in the Cycle 6 Reload Report (Reference 7). A brief summary of the benchmark is given in Appendix A of this Technical Description. The comparison of VIPRE and LYNXT DNBR are plotted as a function of time in Figure 1 of Appendix A. As can be seen from Figure 1, the VIPRE DNBR results are conservative with respect to LYNXT at the point of interest, i.e., the minimum DNBR (MDNBR).

Once the model was benchmarked, the four pump coastdown was run using a total response time of 800 milliseconds. The minimum DNBR calculated using the B&W-2 correlation was 1.885 as shown in Figure 2 of Appendix A. This value obtained using open channel methodology and longer response time shows greater margin to DNB than the three previous analyses given above using closed channel methodology (1.49, 1.816 and 1.85). Using the same response times as are presently being used for the control rod drive release delay and dedicated margin for uncertainty (125 and 44 milliseconds respectively), the resultant response time for Technical Specification Table 3.3-2 can be increased to 631 milliseconds for sensor, reactor protection system, and control rod drive breaker delay. (The 631 milliseconds is derived from the new 800 milliseconds total response time less the dedicated margin for uncertainty of 44 milliseconds and the control rod drive release delay of 125 milliseconds.)

Since the four pump coastdown is the bounding transient for the pump monitors and has been analyzed showing greater margin to DNB, the consequences of all other accidents analyzed in the safety analysis report requiring the pump monitors are also acceptable.

In addition, an editorial change to Technical Specification Table 3.3-2 is included. The footnote designated a "***" refers to the pump contact monitor. Toledo Edison uses the nomenclature pump monitor for this device. Therefore, the word "contact" is being eliminated from the Technical Specification.

Unreviewed Safety Question Evaluation Conclusions

The following conclusions are provided as a result of Toledo Edison's review of the proposed changes to the Davis-Besse Nuclear Power Station, Unit No. 1, Operating License, Appendix A, Technical Specification and Bases.

The proposed action would not increase the probability of occurrence of an accident previously evaluated in the USAR because there have been no hardware changes or design modifications which would affect the probability of an accident. (10CFR50.59(a) (2) (i))

The proposed action would not increase the consequences of an accident previously evaluated in the USAR because the analysis performed shows the USAR DNBR remains bounding. (10CFR50.59(a) (2) (i))

The proposed action would not increase the probability of occurrence of a malfunction of equipment important to safety because there have been no hardware changes or design modifications which would affect the probability of a malfunction. (10CFR50.59(a) (2) (i))

The proposed action would not increase the consequence of a malfunction of equipment important to safety because the analysis performed shows the USAR DNBR remains bounding. (10CFR50.59(a) (2) (i))

The proposed action would not create a possibility for an accident of a different type than any evaluated previously in the USAR because there have been no hardware changes or design modifications. (10CFR50.59(a) (2) (ii))

The proposed action would not create the possibility for a malfunction of equipment of a different type than any evaluated previously in the USAR because there have been no hardware changes or design modifications. (10CFR50.59(a) (2) (ii))

The proposed action would not reduce the margin of safety as defined in the basis for the Technical Specifications because the analysis performed shows the previous analysis remains bounding. (10CFR50.59(a) (2) (iii))

Pursuant to the above, this change to Technical Specification Table 3.3-2 does not involve an unreviewed safety question.

References

1. VIPRE-01, A Thermal-Hydraulic Analysis Code for Reactor Cores (EPRI NP-2511).

2. RADAR, Reactor Thermal and Hydraulic Analysis During Reactor Flow Coastdown, Babcock & Wilcox, Nuclear Power Generation Division, Lynchburg, VA (BAW-10069A, Rev. 01, October, 1974).

3. B&W Calculation 32-10176-00, Densification Transients, DB Osborne, 3-11-75, Babcock & Wilcox, Nuclear Power Generation Division, Lynchburg, VA.

4. B&W Report BAW-1401, Davis-Besse Unit 1, Fuel Densification Report, April, 1975.

5. B&W Calculation 32-9234-00, Re-Evaluation of Accidents, DB Osborne, 6-5-78, Babcock & Wilcox, Nuclear Power Generation Division, Lynchburg, VA.

6. LYNXT, Core Transient Thermal Hydraulic Program (BAW-10156-A).

7. Cycle 6 Reload Report (BAW-2038, April 1988).

8. LYNX1, Reactor Fuel Assembly Thermal-Hydraulic Analysis Code, 7-85, Babcock & Wilcox, Nuclear Power Generation Division, Lynchburg, VA (BAW-10129-A, Rev. 00).

9. LYNX2, Subchannel Thermal Analysis Program, 7-85, Babcock & Wilcox, Nuclear Power Generation Division, Lynchburg, VA (BAW-10130-A, Rev. 00).

APPENDIX A

VIPRE-01 11-CHANNEL MODEL DESCRIPTION

A VIPRE-01 11-channel model has been developed to perform a thermal analysis of the complete loss of reactor coolant flow transient. The VIPRE-01 11-channel model was verified by comparing VIPRE-01 results of a locked rotor analysis to those computed by the fuel vendor, Babcock & Wilcox. This appendix summarizes the general characteristics of the VIPRE-01 11-channel model and also discusses the locked rotor analysis comparison.

The VIPRE-01 code predicts the three-dimensional velocity, pressure and temperature fields for single and two-phase flow in a PWR reactor core. It solves the finite-difference equations for mass, energy and momentum conservation for an interconnected array of channels assuming incompressible thermally expandable homogeneous flow. VIPRE-01 has undergone generic review by the NRC and has been found acceptable for use in licensing applications.

The VIPRE-01 11-channel model was developed using the subchannel analysis technique described in reference 1 in which the core is divided into parallel computational cells or "channels" of varying sizes. Each channel is uniquely identified by number, cross-sectional area, wetted perimeter and heated perimeter and can communicate laterally through gaps by diversion crossflow. The 11-channel model consists of six channels which model individual subchannels of the hot bundle. The remaining channels model varying numbers of subchannels and bundles. The conservation equations for mass, energy and momentum are solved for each channel. General characteristics of the 11-channel model are listed in Table 1.

The thermal-hydraulic (T-H) codes used by the fuel vendor (starting with Davis-Besse Cycle 6) are LYNX1 (Reference 8), LYNX2 (Reference 9) and LYNXT (Reference 6). LYNX1 is designed to compute steady-state bundle average T-H parameters including mass flow rates, bulk fluid temperatures and pressure drops. The bundle boundary conditions from LYNX1 are input to LYNX2 which computes T-H parameters on a subchannel basis for a single assembly. The LYNXT code is used for transient analyses and is capable of modeling computational cells of varying sizes.

A LYNXT model is currently used by the fuel vendor for thermal analyses of loss-of-flow transients. This model has been benchmarked to the more detailed LYNX1/LYNX2 analysis.

The VIPRE-01 and LYNXT models use the same design parameters which describe the fuel assembly geometries currently in use at Davis-Besse. These data include form loss coefficients, bypass flow and hot channel factors. The modeling technique previously described was used to construct a VIPRE model consisting of 11 channels and was derived independently of the LYNXT model.

A VIPRE-01 analysis of the locked rotor transient using the 11-channel model has been performed and compared to results obtained using the LYNXT model. The locked rotor transient is initiated by an instantaneous seizure of a reactor coolant pump rotor. The large flow resistance caused by the stationary impeller results in a rapid decrease in reactor coolant flow and minimum DNBR. The locked rotor transient was chosen in order to compare the VIPRE-01 and LYNXT models during a large change in minimum DNBR. Results are plotted in Figure 1 and show that VIPRE-01 and LYNXT minimum DNBR's are within 4 DNBR points (1 DNBR point = 0.01 DNBR) during the transient. This variation is small and the comparison demonstrates that the VIPRE-01 11-channel model produces minimum DNBR's that are nearly identical to those predicted by LYNXT.

The lock rotor comparison demonstrates that the VIPRE 11-channel model is capable of accurately predicting T-H parameters during a loss-of-flow transient. Therefore, the 11-channel model may be used to analyze the complete loss-of-flow transient.

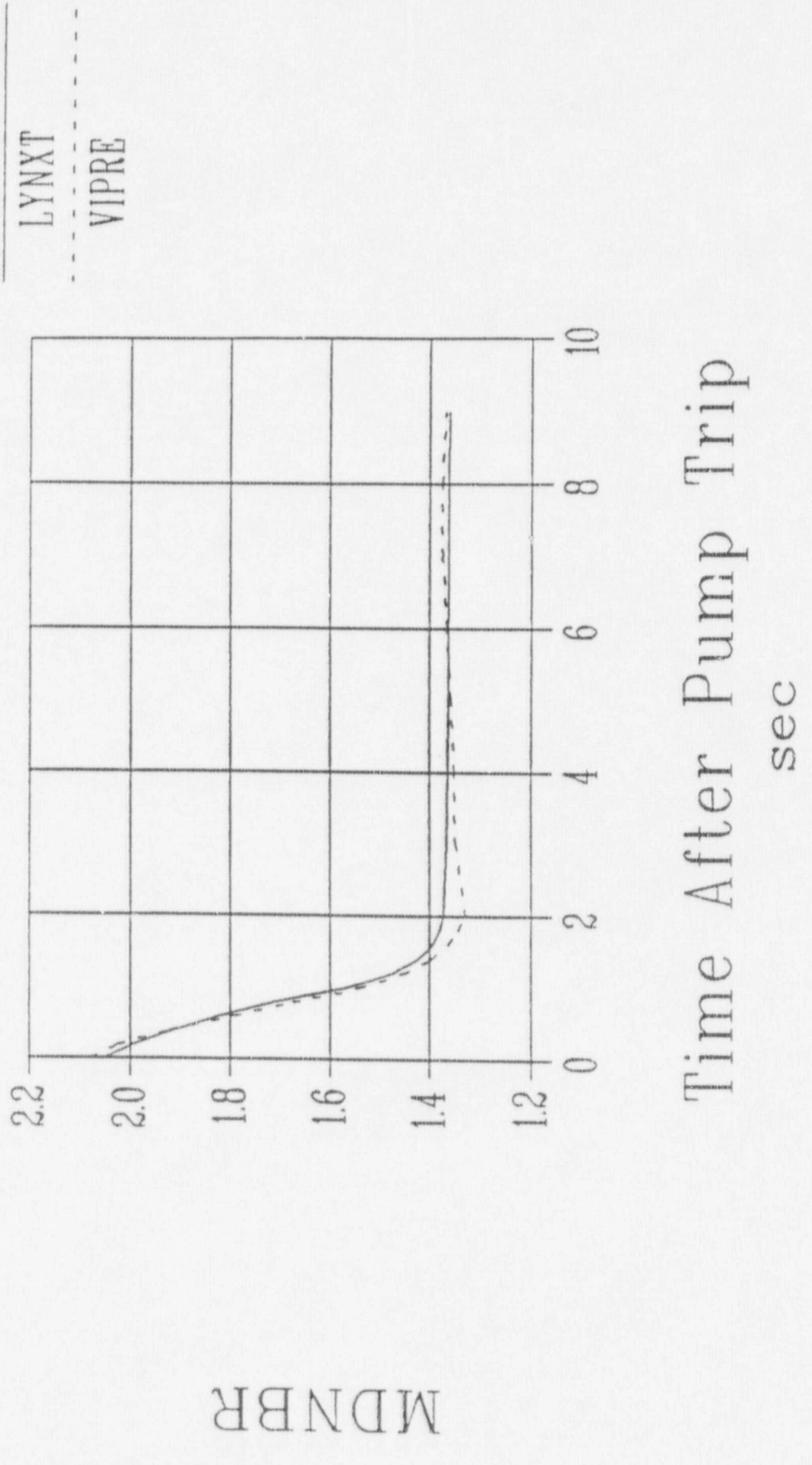
APPENDIX A

TABLE 1

VIPRE-01 11-CHANNEL MODEL CHARACTERISTICS

Geometry	
Symmetry	eighth core
Number of Axial Nodes	48
Total Number of Channels Modeled	11
Number of Individual Subchannels in Hot Bundle	6
Number of Lumped Channels in Hot Bundle	2
Number of Lumped Channels Modeled in Remainder of Eighth Core	3
Total Number of Fuel Rods Modeled	13
Number of Individual Fuel Rods	8
Number of Lumped Fuel Rods	5
Number of Conduction Rods	4
Number of Dummy Rods	9
Power Distribution	
Radial x Local Peaking Factor	1.71
Axial Flux Shape	1.65 chopped cosine located at midplane
Flow Correlations	
Single Phase Friction	VIPRE-01 Default
Two Phase Friction Multiplier	EPRI
Subcooled Void	EPRI
Bulk Void	EPRI
Heat Transfer Correlations	
Single Phase Forced Convection	Identical to LYNXT
Subcooled Nucleate Boiling	Thom
Saturated Nucleate Boiling	Thom
Film Boiling	None
Mixing Model	
Turbulent Momentum Factor	0.8
Turbulent Mixing Coefficient	0.02
Lateral Resistance Factor	0.5 x Length/Pitch
Critical Heat Flux	
CHF Correlation	B&W-2

Figure 1 MDNBR VS Time
Locked Rotor Benchmark



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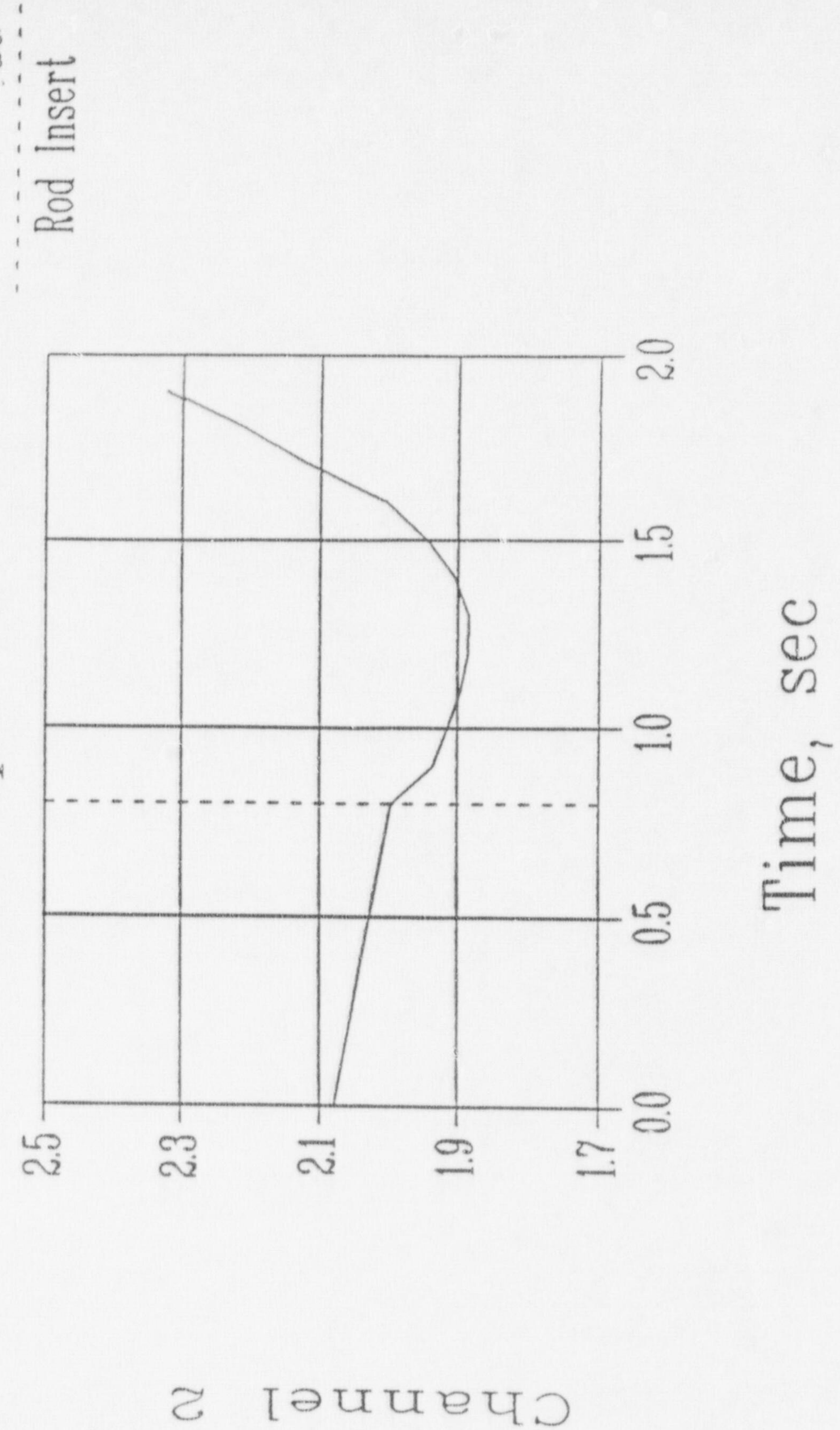

Fig 2 MDNBR
Four Pump Coastdown