ATTACHMENT 1

Con Edison requests a change in Pages 2.3-3 and 2.3-6 of the Indian Point Unit No. 2 Technical Specifications. This revision would change the setpoint for the underfrequency trip of the Reactor Coolant Pumps so that this setpoint is more consistent with the known stability of the power grid and the maximum anticipated frequency decay rate.

Presently, the underfrequency trip on the Reactor Coolant Pumps is set at 57.5 Hz. System power generating deficiencies in excess of about 25% of total load will cause the frequency of the system to drop below this setpoint. This trend, however, will be counteracted by automatic load shedding. The proper system frequency of 60 Hz will be restored in a matter of seconds for generating deficiencies down to about 40% of total load providing Indian Point Unit No. 2 does not trip off the line. Should the underfrequency trip cause the removal of Unit No. 2 from the system generating capacity, load deficiency will increase resulting in a further decrease in electrical frequency and more load shedding to restore this frequency.

The underfrequency setpoint of 57.5 Hz was chosen to protect the core against a reduced Reactor Coolant pump flow. When the Reactor Coolant Pump speed decreases as a result of lower system frequencies, coolant flow is reduced and some reactor cooling capacity is lost. Temperatures in the core could consequently increase and the minimum DNBR of 1.30 might be violated. The

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attached Westinghouse analysis, however, describes a study of this reduced cooling capacity following Reactor Coolant Pump slowdown and it was found that, for frequency decay rates as high as 12 Hz/sec, the core DNBR is maintained greater than 1.30 for a trip setting of 55 Hz. Actually, system decay rates greater than 4 Hz/sec are not considered credible and thus considerable conservatism has been introduced into the analysis.

The proposed revision to the specification that would change the underfrequency Reactor Coolant Pump trip setpoint from 57.5 Hz to 55 Hz would not compromise the safety of the reactor and instead would promote stability of the power grid. A revised basis for the proposed specification is also provided.

Revised Pages 2.3-3 and 2.3-6 of the Technical Specifications are attached.

APPENDIX A

Indian Point Unit No. 2
Underfrequency Study

Introduction

Decreasing grid frequency in an underfrequency transient applies a braking torque to the reactor coolant pump (RCP) motors. This causes RCP motor speed and primary coolant flow to decrease. An underfrequency accident is similar to a complete loss of forced flow as reported in the FSAR (Section 14.1.6).

The protection system logic for the plant calls for a trip of the Reactor Coolant Pump breakers if low frequency is sensed on 2 out of 4 pump buses, followed by a reactor trip when auxiliary contacts on the breakers sense that an RCP breaker has tripped. The RCP breaker trip allows the pumps to coast down at a rate governed by pump inertia, rather than being dragged down in speed by the decaying grid frequency. Due to single failure criterion considerations, it is assumed that at least two pump breakers work properly (i.e., one failed channel cannot prevent a reactor trip if trip signals are generated in two channels).

Analysis

a. Primary coolant flow transients for this analysis were computed using the PHOENIX (WCAP-7973)[1] computer code.

Assumptions

(1) Pump speed is directly proportional to frequency (i.e. there is no slip). This is true up to very high frequency decrease rates, e.g., rates greater than 15 Hz/sec.

(2) The delay between the time the underfrequency trip setpoint is reached and the time the pump breakers open is conservatively taken to be .6 seconds.

b. Nuclear power transients for this analysis were calculated using the LOFTRAN (WCAP-7907)[2] computer code.

Assumptions

1) Zero moderator reactivity feedback
2) -4% inserted rod reactivity upon trip
3) Large value of doppler - only power defect
4) The delay between the time the underfrequency trip setpoint is reached and the time the rods begin to move into the core is conservatively taken to be 1.0 second. Included in this delay is a 6 Hz sensor relay delay.

c. Core heat flux transients for this analysis were determined using the FACTRAN (WCAP-7908)[3] computer code and the flow and nuclear power transients calculated in PHOENIX and LOFTRAN.

d. DNBR transients were calculated using the THINC-III computer code and the flow and heat flux transients calculated in PHOENIX and FACTRAN.

Assumptions

1) Inlet temperature is 4°F above nominal T_{in}. (543.0 + 4.0 = 547.0°F).

2) Primary pressure is 30 psi below nominal pressure (2250 - 30 = 2220 psia). No credit is taken for the effect of larger primary pressure on DNB ratios during the transient.

3) Initial core heat flux is 102% of nominal (1.02 x 2758 = 2813.2 MW_t).

4) $F_{\Delta H}$, the enthalpy rise hot channel factor, is 1.65 for Unit No. 2 and the axial power shape is a chopped cosine with a peak of 1.55. These peaking factors are consistent with those used in the fuel densification report.[4]

5) The power spike and the pellet eccentricity penalties on DNB, as conservatively applied in the fuel densification reports (see WCAP-8219[5]), were also used in this study.

Results

Figure 1 is a plot of minimum DNBR versus underfrequency trip setpoints at various frequency decay rates for Unit No. 2. Graphs of key system variables are shown in Figures 2a, 2b, 2c and 2d, which are, respectively, core flow, nuclear power, hot channel heat flux, and DNB ratio versus time for a 4 Hz/sec frequency

accident assuming a 55 Hz trip setpoint. Figure 3 is a plot of minimum DNBR versus frequency decay rate for Unit No. 2 with a 55 Hz trip setpoint.

Conclusions

A 55 Hz trip with a 6 Hz sensor time delay will provide core protection (e.g., minimum core DNBR 1.30) for frequency rates of up to 12 H_Z/sec for Indian Point Unit No. 2.

These conclusions also apply for three-loop operation. As shown in the FSAR, e.g., Section 14.1.6, the consequences of a complete loss of forced flow accident are less severe for three-loop operation.

References

1. WCAP-7973: "Calculation of Flow Coastdown After Loss of Reactor Coolant Pump (PHOENIX Code)", Class 3, Westinghouse Non-Proprietary.

2. WCAP-7907: "LOFTRAN Code Description", Class 3, Westinghouse Non-Proprietary.

3. WCAP-7908: "FACTRAN, A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod", Class 3, Westinghouse Non-Proprietary.

4. "Fuel Densification - Indian Point Nuclear Generating Station No. 2", January 1973, and Addendum dated March 22, 1973, Westinghouse Non-Proprietary

5. "Fuel Densification - Indian Point Nuclear Generating Unit No. 3", July 1973, WCAP-8147, Class 3, Westinghouse Non-Proprietary.

5. WCAP-8219: "Fuel Densification, Experimental Results and Model for Reactor Application", October 1973, Class 3, Westinghouse Non-Proprietary

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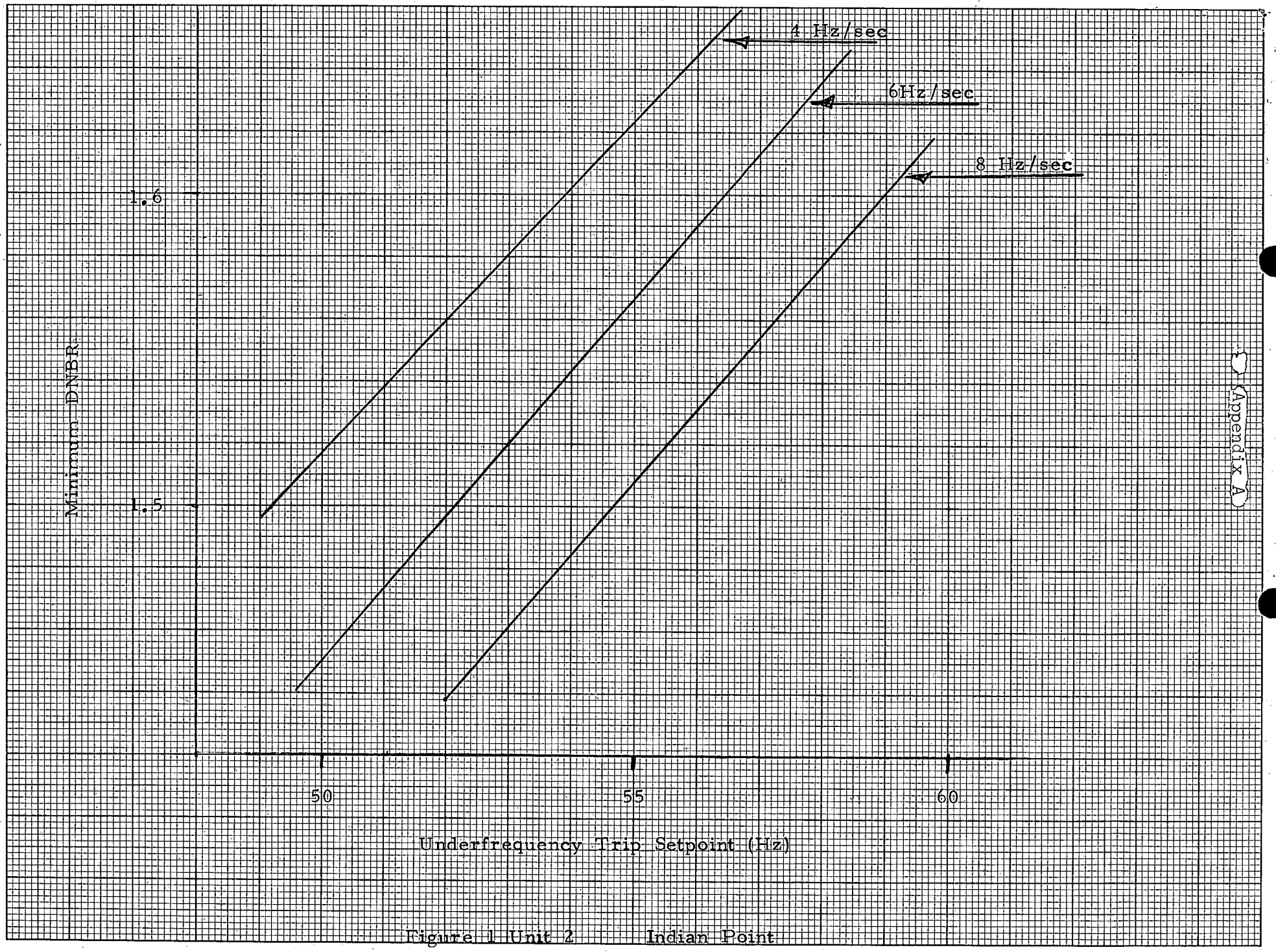

Figure 1 Unit 2 Indian Point

(Appendix A)

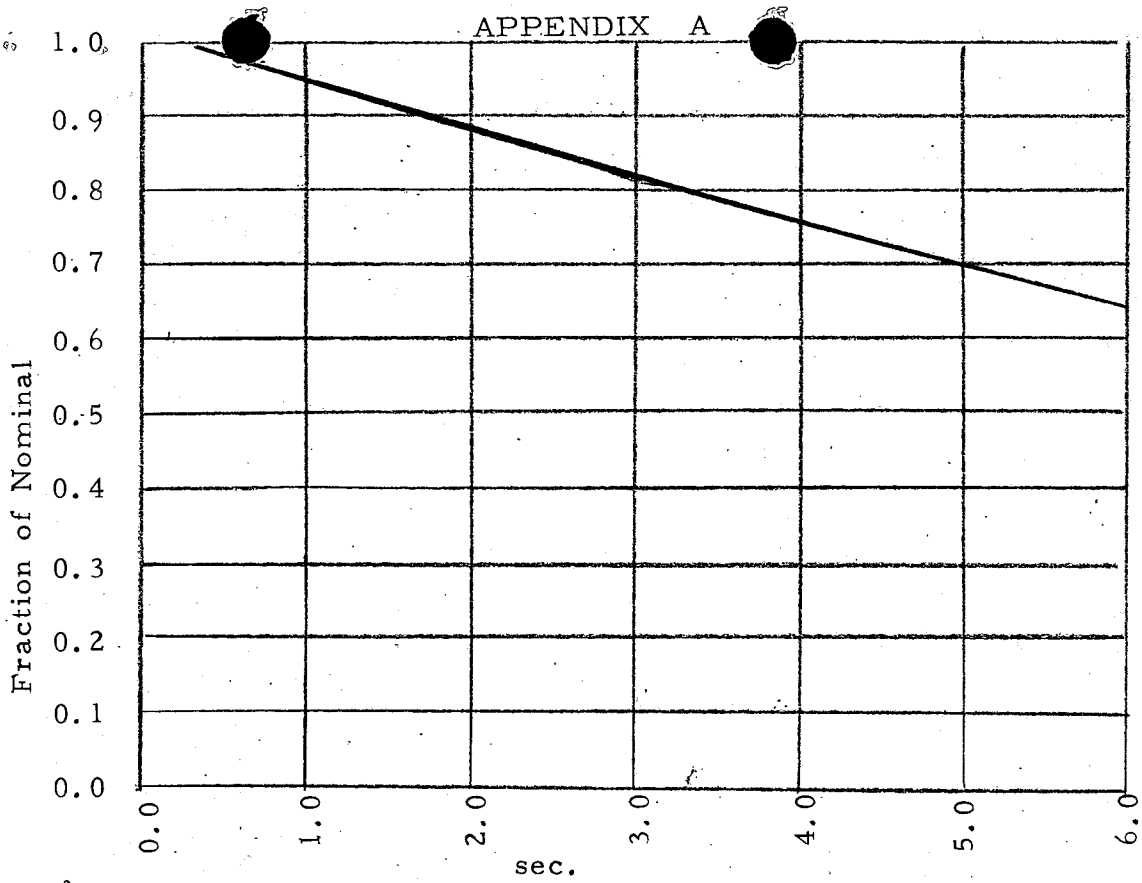

Figure 2a Unit 2 Core Flow - 4 Hz/sec , 55 Hz Trip

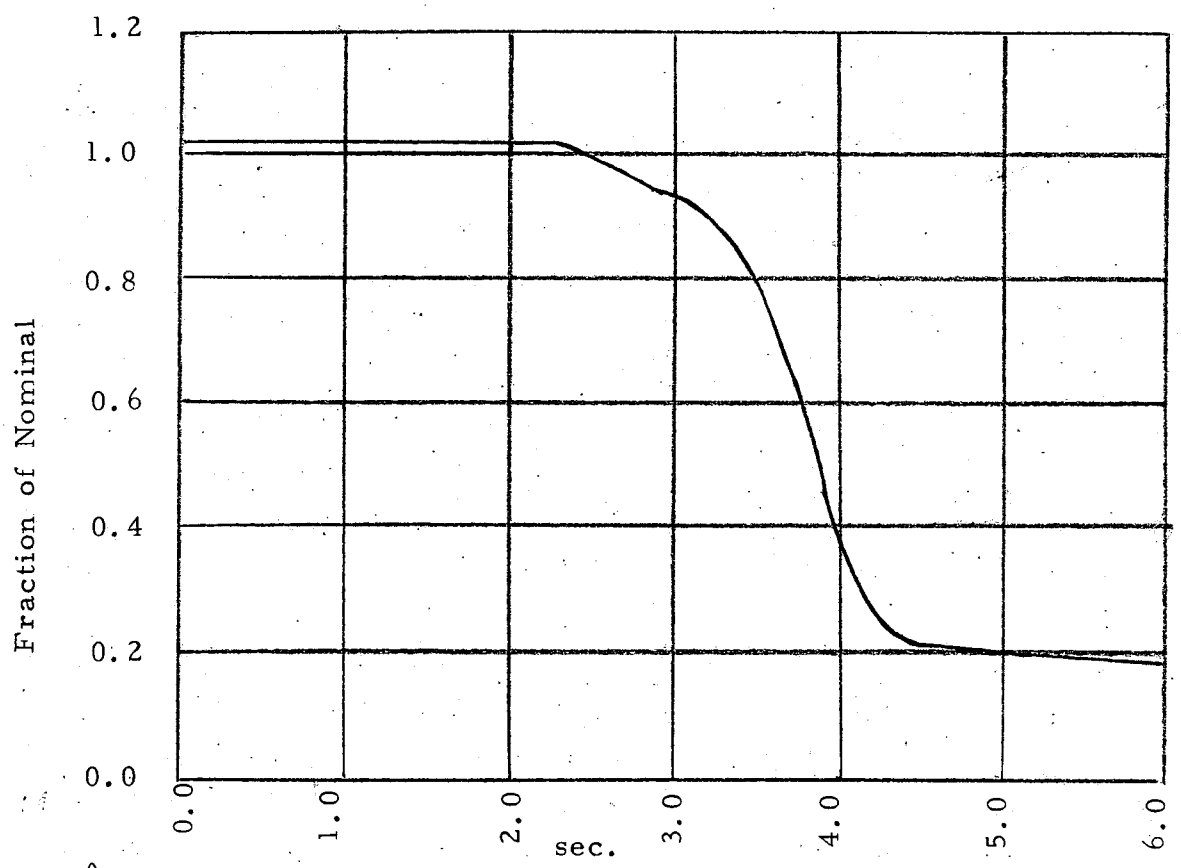

Figure 2b Unit 2 Nuclear Power - 4 Hz/sec , 55Hz Trip

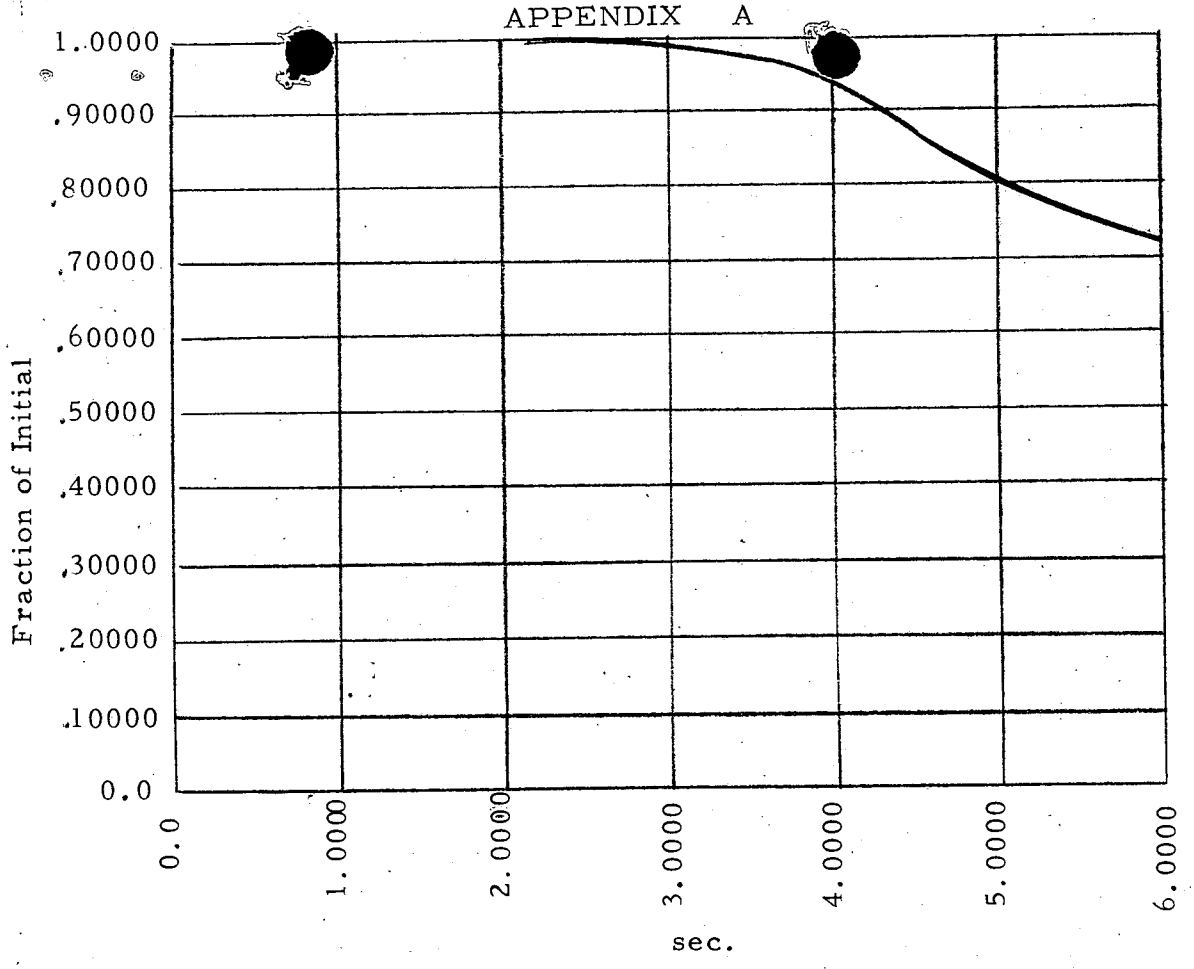

Figure 2c. Unit 2 Hot Channel Heat Flux - 4 Hz/sec , 55 Hz Trip

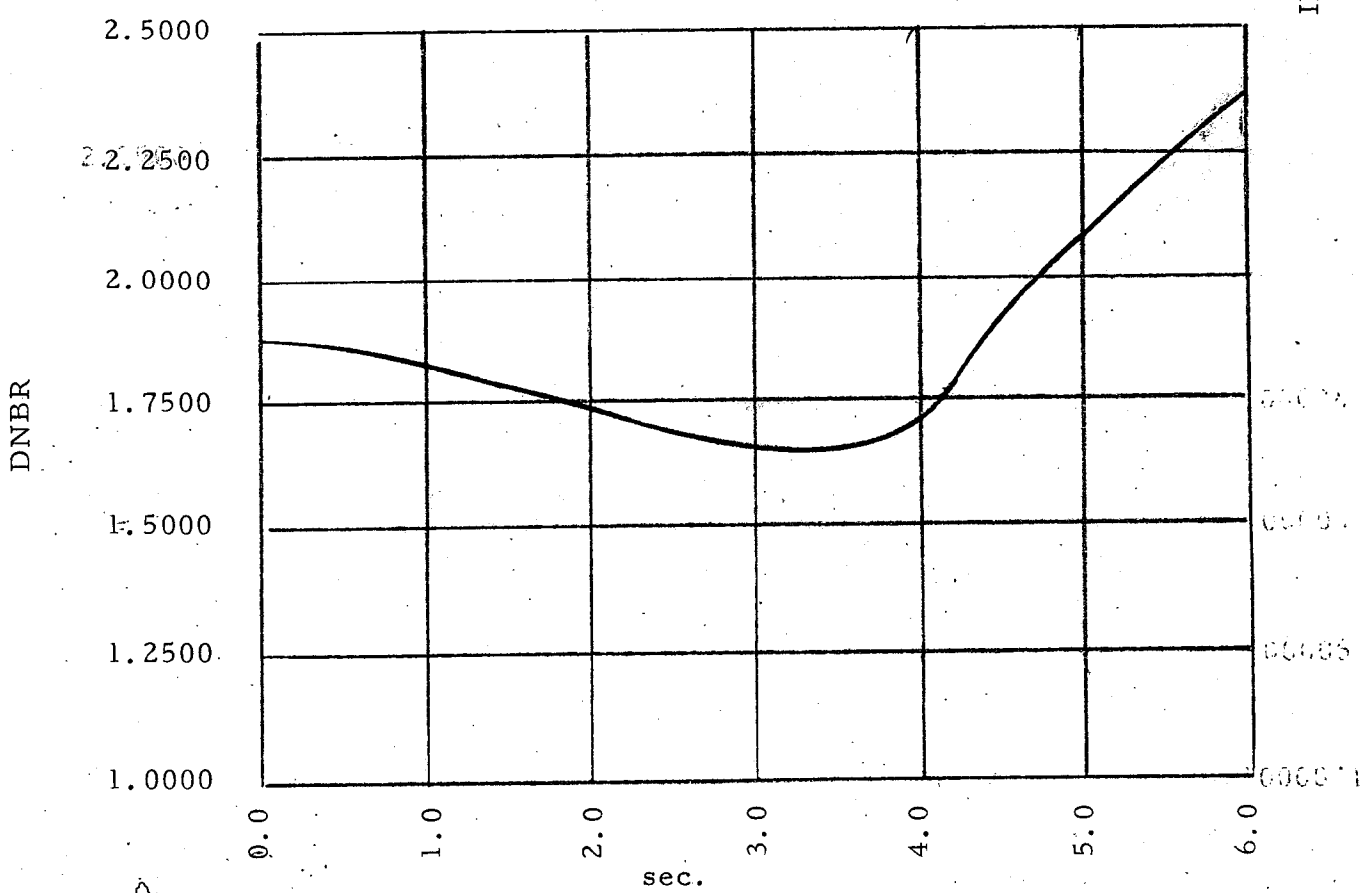

Figure 2d Unit 2 DNBR - 4Hz/sec, 55Hz Trip

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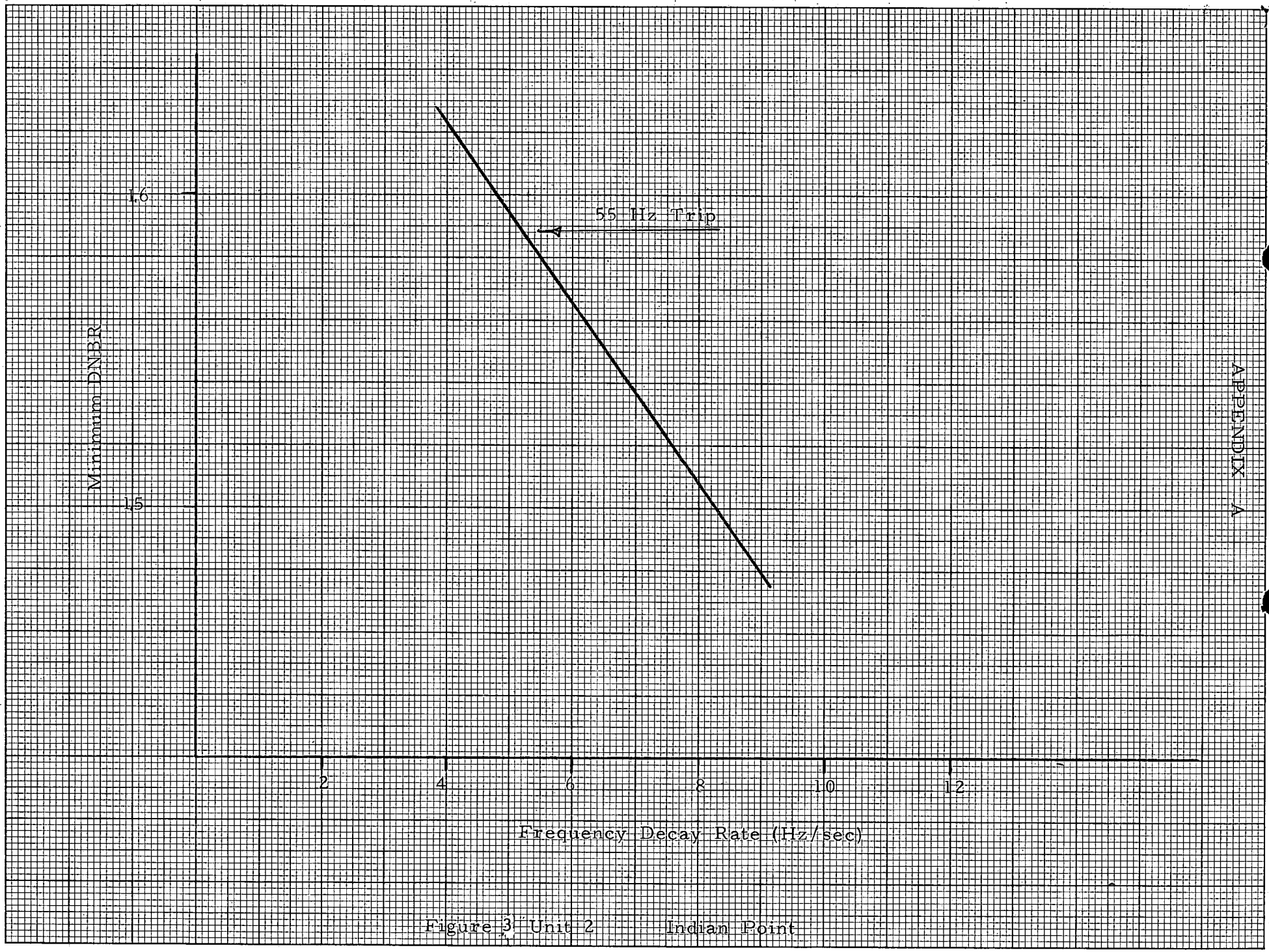

Figure 3 Unit 2 Indian Point