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DENTON,H.R. Office of Nuclear Reactor Regulation, Director

SUBJECT: Forwards responses to plant transient analysis issues identified in SER,per 830114 Amend 48 to License DPR-74. Exxon Nuclear Co critical heat flux correlation for PWR fuel designs improves critical heat flux phenomenon.

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April 15, 1983
AEP:NRC:0637M

Donald C. Cook Nuclear Plant Unit No. 2
Docket No. 50-316
License No. DPR-74
UNIT 2, CYCLE 4 SAFETY EVALUATION REPORT:
RESPONSE REGARDING PLANT TRANSIENT METHODS

Mr. Harold R. Denton
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Denton:

On January 14, 1983, the Commission issued Amendment No. 48 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant Unit No. 2, with an accompanying Safety Evaluation Report (SER). The Attachment to this letter responds to certain items raised in the SER.

Item VI of the General Conclusions (page 43 of the SER) requires the submittal of additional specific information within 90 days of the date of issuance of the Amendment. Such additional information is given in Section A of the Attachment to this letter.

Part B of the Attachment to this letter comments on some statements contained in the Accident and Transient Analysis section of the SER (Part C, page 17). These comments are relevant to the resolution of the conditions imposed in Section III of the SER (page 42) and in Section 3.(2) of the above mentioned License Amendment. We would be pleased to meet with your Staff at their convenience to define what action and information are required to remove the license conditions contained in Amendment No. 48.

A001

This document has been prepared following Corporate Procedures which incorporate a reasonable set of controls to ensure its accuracy and completeness prior to signature by the undersigned.

Very truly yours,

Vice President

/os
Attachments

cc: John E. Dolan - Columbus
M. P. Alexich
R. W. Jurgensen
W. G. Smith, Jr. - Bridgman
R. C. Callen
G. Charnoff
NRC Resident Inspector at Cook Plant - Bridgman

ATTACHMENT TO AEP:NRC:0637M

Response to the Plant Transient Analysis Issues Identified in the NRC Safety Evaluation Report (SER) for D. C. Cook Unit 2, Cycle 4

A. RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

The licensee must submit, within 90 days after receipt of this Safety Evaluation Report (SER), the specific additional information identified in this SER. This 90 days represents a reasonable period of time for preparation of the information to be submitted. The additional information needed is as follows:

Item 1.1: Improved model to represent this event, or additional justification for acceptability of the present method. [Reference: Loss of Single Reactor Coolant Pump-Locked Rotor (and Broken Pump Shaft)]

The SER identifies three points of concern on Page 23 which are answered below. A fourth point of concern, the primary coolant flow rate employed in the ENC analysis, is addressed in a January 10, 1983 letter from G.C. Cooke (ENC) to D.L. Wigginton (NRC) on the subject of D.C. Cook Unit 2 Reactor Coolant Flow Rate.

> Point One is quoted from the SER: "While initiation time for the low flow trip in the faulted loop and reactor trip time are virtually the same as in the Cycle 2 analyses, the reduction in nuclear power and thermal power (from the core) occurs approximately one and two seconds earlier, respectively, than shown in the FSAR, although the scram curve used in the PTSPWR2 model shows a reactivity insertion worth which is delayed by about 0.4 secs relevant to that used in Cycle 2."

The FSAR scram insertion curve is delayed relative to that employed by the ENC PTSPWR2 model. The ENC simulation has conservatively accounted for applicable scram delay and insertion times, as has been established during the review process (Dec. 2, 1982 meeting in Bethesda between the NRC Staff, American Electric Power Service Corporation representatives, and Exxon Nuclear Company representatives). Therefore, while the FSAR analysis is more conservative than the ENC analysis in the simulation of scram insertion, the ENC analysis is adequately conservative and consistent with the Technical Specifications.

Point Two is quoted from the SER: "The primary system pressure increase is about one third of the magnitude calculated by the earlier models accepted for D.C. Cook 2 in Cycle 2. Cycle 2 predicts a pressure increase of 280 psi to a maximum of 2633 psia over the first three seconds, compared with an increase of 100 psi using PTSPWR2."

The difference in predicted primary pressure response between the July 1982 FSAR analysis and ENC's analysis is accounted for by two factors, as follows.

All pressures cited or predicted for the primary system in the ENC analysis are pressurizer pressures. An increment of about 70 psi is employed in the July 1982 FSAR analysis to account for the pump discharge to pressurizer ΔP. Adding a 70 psi increment to ENC's calculated pressurizer pressure will result in a peak primary loop pressure of 2443 psia, still well below the relief valve setpoint.

The July 1982 FSAR analysis employs an assumption which alters the clad surface heat flux transient and thereby strongly impacts the calculated system pressure transient. At the initiation of the event, the fuel pellet-to-clad gap heat transfer coefficient is assumed to increase step-wise to a value of 10,000 Btu/hr-ft^2-^{o}F from an initial steady state value consistent with the initial fuel temperature. This results in the release to the clad of a large amount of energy initially stored in the fuel, which artificially increases clad-to-coolant heat flux. This behavior is a purposely induced artifact of the simulation method and cannot in fact occur in an operating reactor. The artificial increase in clad-to-coolant heat flux caused by this assumption results in a significant overestimate of the primary coolant heatup and volumetric expansion. In turn, this results in overestimation of the pressurizer surge flow rate, with consequent exacerbation of the system pressure transient. ENC's analysis does not include this assumption, and represents a more correct approach to the event simulation.

The primary coolant heatup calculation for this transient is adequately handled by the PTSPWR2 core thermal hydraulics model. The PTSPWR2 calculation of the core coolant temperature transient has been verified by transient XCOBRA-IIIC core thermal hydraulics calculations. Calculation of pressurizer surge flow rate is straightforward once the coolant temperature transient has been determined.

The PTSPWR2 pressurizer model is described and its predictions compared to data in XN-CC-38, Supplement 1. Since the PTSPWR2 pressurizer model employs the assumption of an isentropic compression during in-surge, the pressurizer pressure is calculated at a theoretical maximum for a given in-surge rate. Thus, pressurizer pressure is predicted by the PTSPWR2 simulation at a maximum realistic value for this event. The maximum primary loop pressure of 2443 psia noted above therefore represents a conservative maximum for the locked rotor event.

> Point Three is quoted from the SER: "The model calculates only average surface (i.e., clad) and average fuel temperatures. Information is not available on the capability of PTSBWR2 (SIC) and its adjunct thermal hydraulic models to calculate the detailed response of the fuel during these fast transients, including: stored energy, internal temperatures with possible fuel melting, gap conductances, and clad surface temperatures to ensure continuing core cooling capability and to assess zirconium/water and steam reactions."

The PTSPWR2 fuel thermal model is described in XN-CC-38. The model has transient capability and incorporates a detailed radial nodalization of the fuel pellet and clad. The model therefore explicitly accounts for stored energy and provides an adequately detailed radial temperature profile in the pellet and clad. Fuel central node temperatures predicted during the locked rotor event are below 3500°F and do not approach the 5000°F UO_2 melting temperature. Since DNB does not occur during the transient, assessment of the $Zr-H_2O$ reaction is not applicable.

Item 1.2: Provide a confirmatory analysis which demonstrates that specified acceptable fuel design limits are not violated for the case of loss of offsite power. Justification for any delays assumed between reactor/turbine trip and loss of offsite power must be provided.

The requested simulation is attached. Inputs and methodology are identical to those used in Case 1, reported in XN-NF-82-32(P), Rev. 1, except that coastdown of the three remaining primary coolant pumps is assumed to occur simultaneously with the occurrence of the locked rotor. No delay is assumed between reactor trip and loss of offsite power. The calculated MDNBR is 1.34, only slightly degraded relative to the locked rotor case 1. Attached Figures 1-6 display the calculated system response.

Item 2 : The accidental opening of one feedwater control valve with the reactor at 100% power at BOC. Provide the information necessary to justify the acceptability of the steam generator heat transfer characteristics used in these analyses.

The feedwater flow increase (FFI) event is not a DNB limiting event. The July 1982 FSAR analysis of this event clearly demonstrates that both the reactivity excursion and the thermal margin degradation induced by this event are quite minimal compared to rod withdrawal and loss of flow events. The mild thermal transient associated with this event compared to limiting events is a characteristic of the plant design and is not impacted by new fuel design and operation associated with Cycle 4. Thus, the steam generator heat transfer characteristics employed in ENC's simulation of this event do not affect plant safety.

Item 3: Reference: Excessive heat removal due to a feedwater system malfunction causing a bypass of the feedwater heating system leading to a reduction in feedwater temperature (EOC, full power and Westinghouse automatic rod control). Provide the additional information requested in the SER.

The SER (Page 37) requests supporting information to confirm the conclusion that this event is non-DNB limiting. As described in the following, this event is always bounded with respect to DNB by the rod withdrawal event. Thus, the event is non-limiting with respect to DNB.

The rod withdrawal (RWD) event and the loss of feedwater heating (LOFH) event are uncontrolled power ascension transients which induce thermal margin degradation primarily through increased clad surface heat flux and primary coolant temperature changes. The reactivity insertion rates characteristic of LOFH events with and without Automatic Rod Control (ARC) fall well within those which can occur in the RWD events. The power excursion during a particular LOFH event can therefore be matched by an equivalent RWD event, referred to hereafter as the heat flux-equivalent RWD. This pair of events will thus be characterized by equal clad surface heat flux.

In the LOFH event, the initiating primary coolant temperature decrease also mitigates thermal margin degradation relative to that which occurs during the heat flux-equivalent RWD event, which is characterized by a primary coolant temperature increase. Thus the LOFH event, with cooler coolant conditions, is always bounded with respect to DNB by a heat flux-equivalent RWD event. Thermal margin degradation resulting from RWD events in Cycle 4 is represented by the bounding RWD analyses presented in XN-NF-82-32(P), Rev. 1. Since the LOFH event is bounded by the RWD event, detailed transient simulation of these LOFH events are not necessary to establish the safety of the plant.

<u>Item 4</u>: Reference: Excessive heat removal due to a feedwater system malfunction at zero load causing feedwater temperature to be reduced to 70°F. Provide the additional information requested.

This transient where feedwater temperature is reduced to 70°F is bounded by the analysis contained in Section 14.1.10 of the FSAR (Unit 2). In the feedwater control valve malfunction event there is an accidental opening of one feedwater control valve at zero load with the feewater temperature assumed to be 70°F. The FSAR analysis shows that the corresponding reactivity insertion rate is less than the 75 pcm/sec insertion rate considered in the RCCA withdrawal from subcritical condition. In the feewater control valve accident, feedwater flow to one steam generator is assumed to increase from essentially zero to full rated flow (approximately 1000 pounds/sec.). Based on the mass of liquid in the primary system and the steam generators, the maximum cooldown rate is conservatively estimated not to exceed 1°F/sec. With a bounding end-of-cycle moderator temperature coefficient of -39 pcm/°F it is concluded that the reactivity insertion rate for the zero power feedwater control valve accident will be appreciably less than the 75 pcm/sec reactivity insertion rate noted above and hence reactivity-wise the transient is bounded by the FSAR analysis of the RCCA withdrawal event form the subcritical condition.

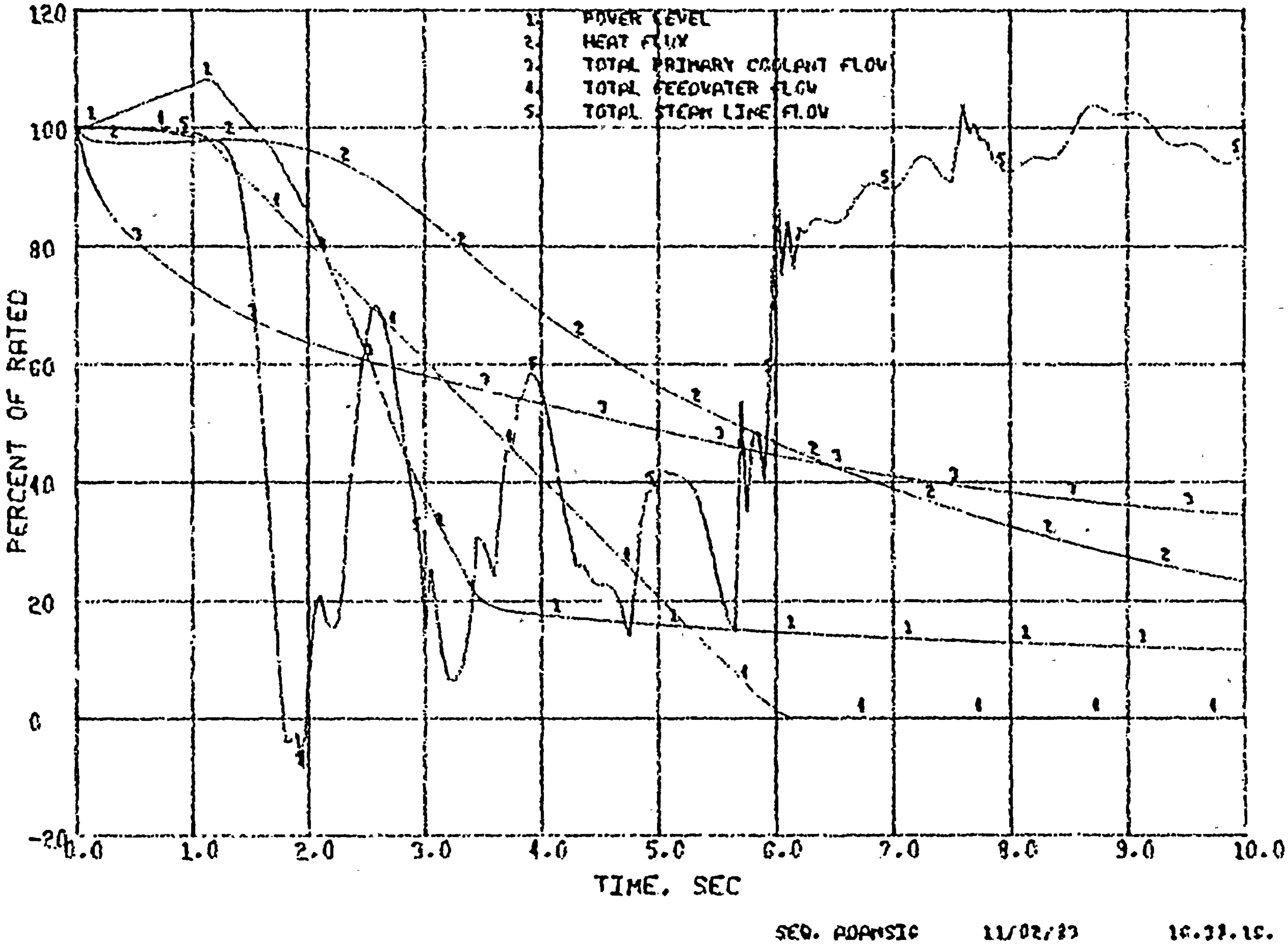

Figure 1: Power, Heat Flux, and System Flows for Locked Rotor with Concurrent Loss of Offsite Power (Minimum DNBR Assumptions)

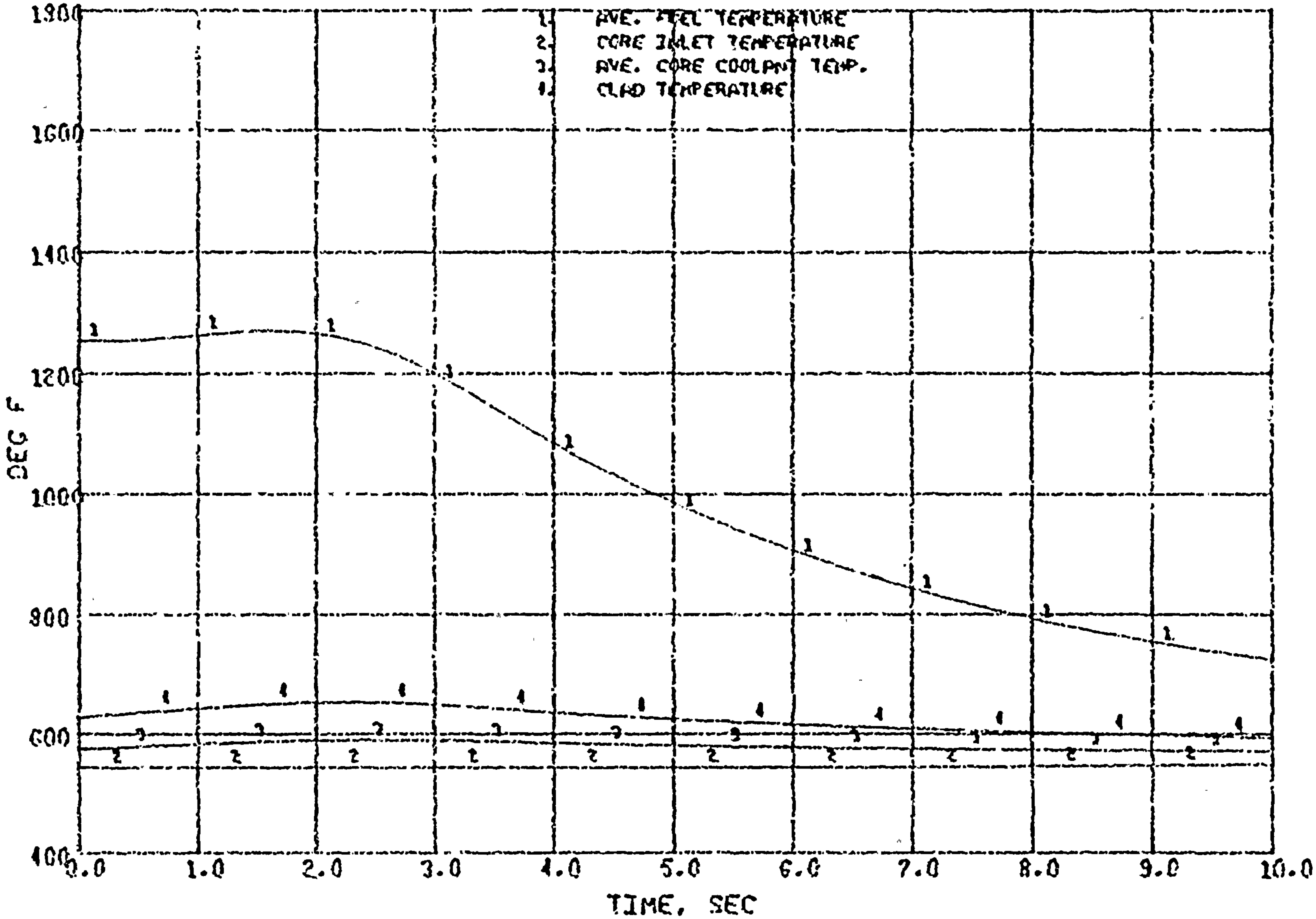

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Figure 2: Primary Temperatures for Locked Rotor with Concurrent Loss of Offsite Power (Minimum DNBR Assumptions)

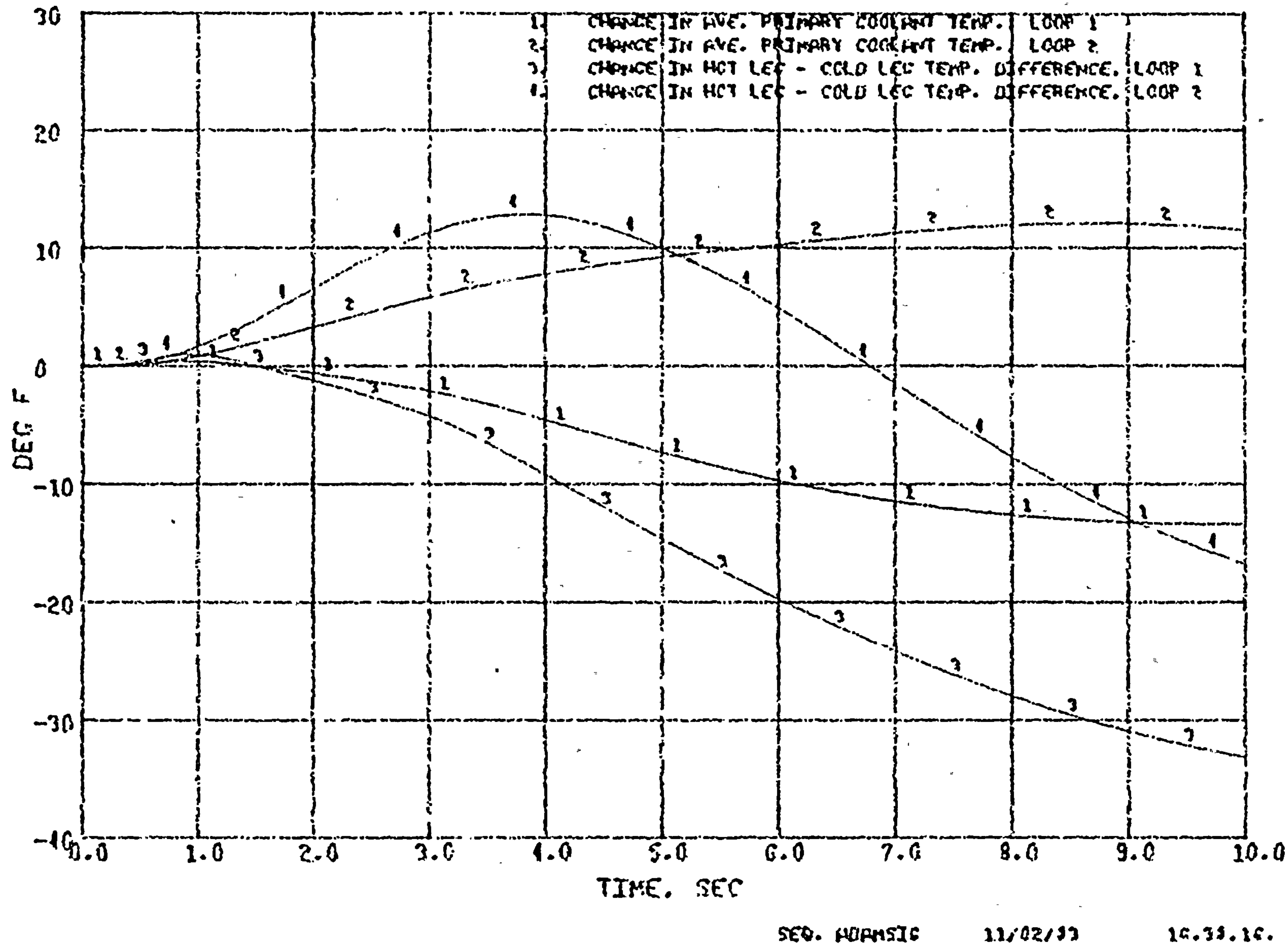

Figure 3: Primary Temperature Changes for Locked Rotor with Concurrent Loss of Offsite Power (Minimum DNBR Assumptions)

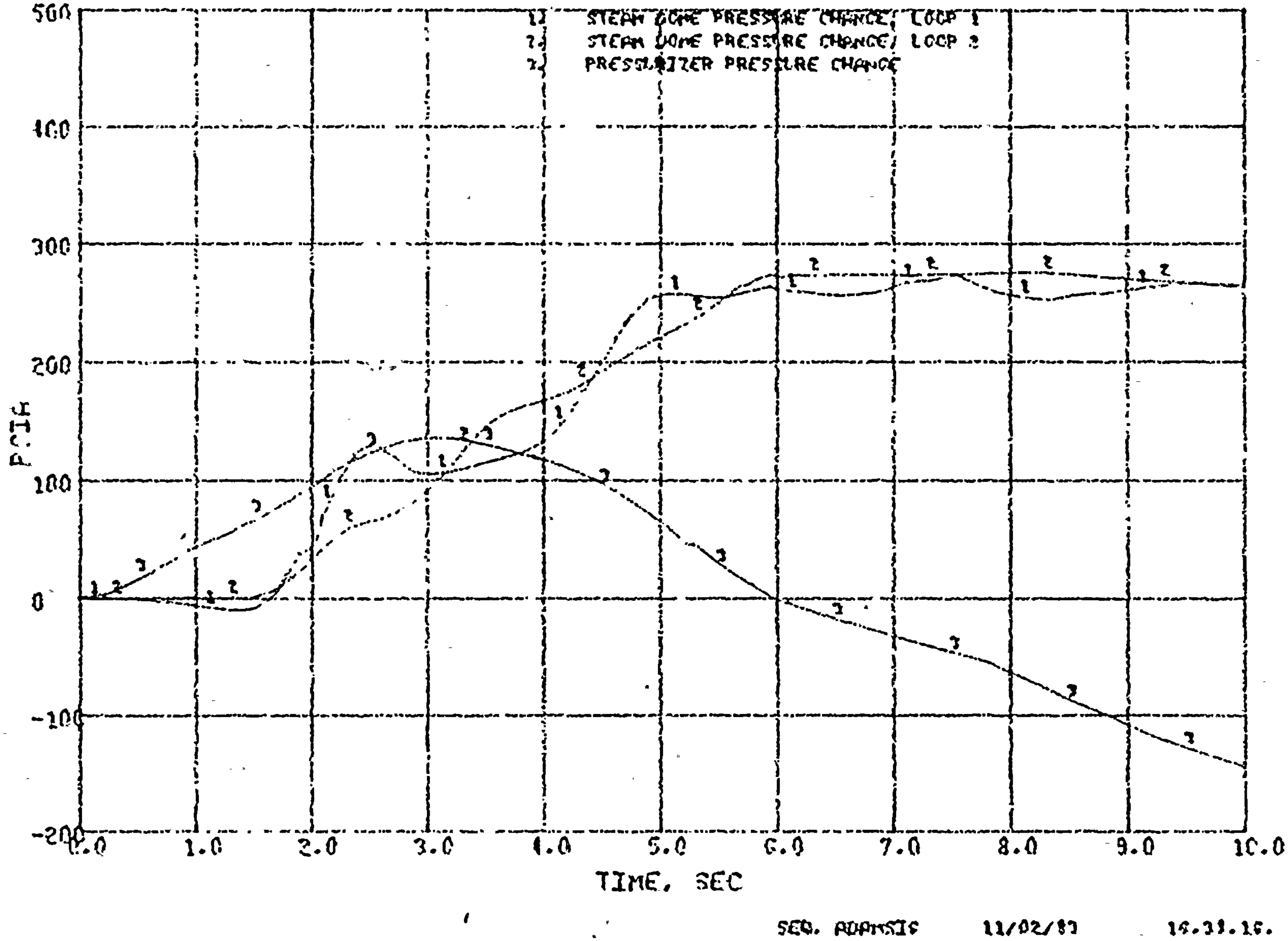

Figure 4: System Pressures for Locked Rotor with Concurrent Loss of Offsite Power (Minimum DNBR Assumptions)

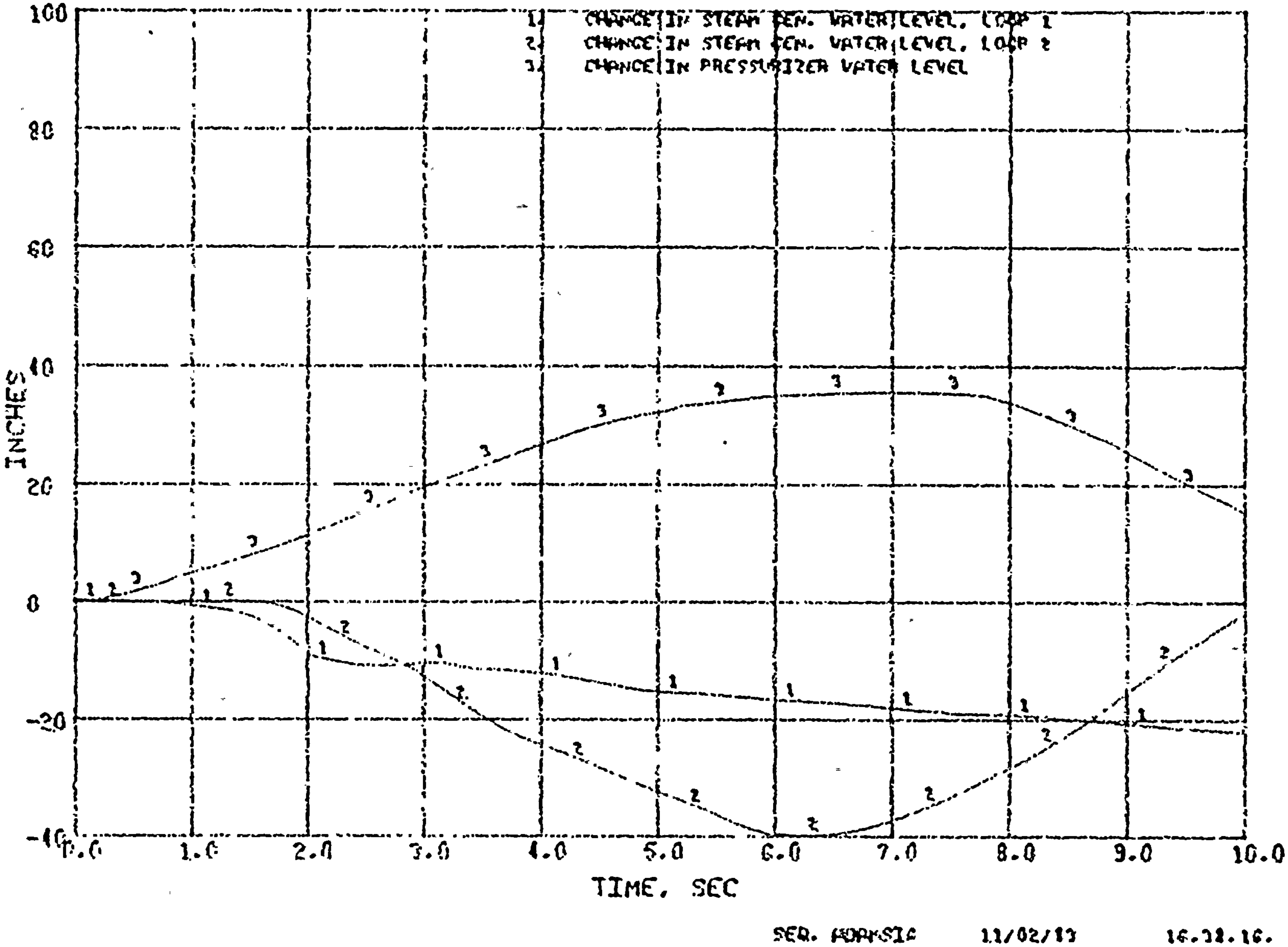

Figure 5: System Level Changes for Locked Rotor with Concurrent Loss of Offsite Power (Minimum DNBR Assumptions)

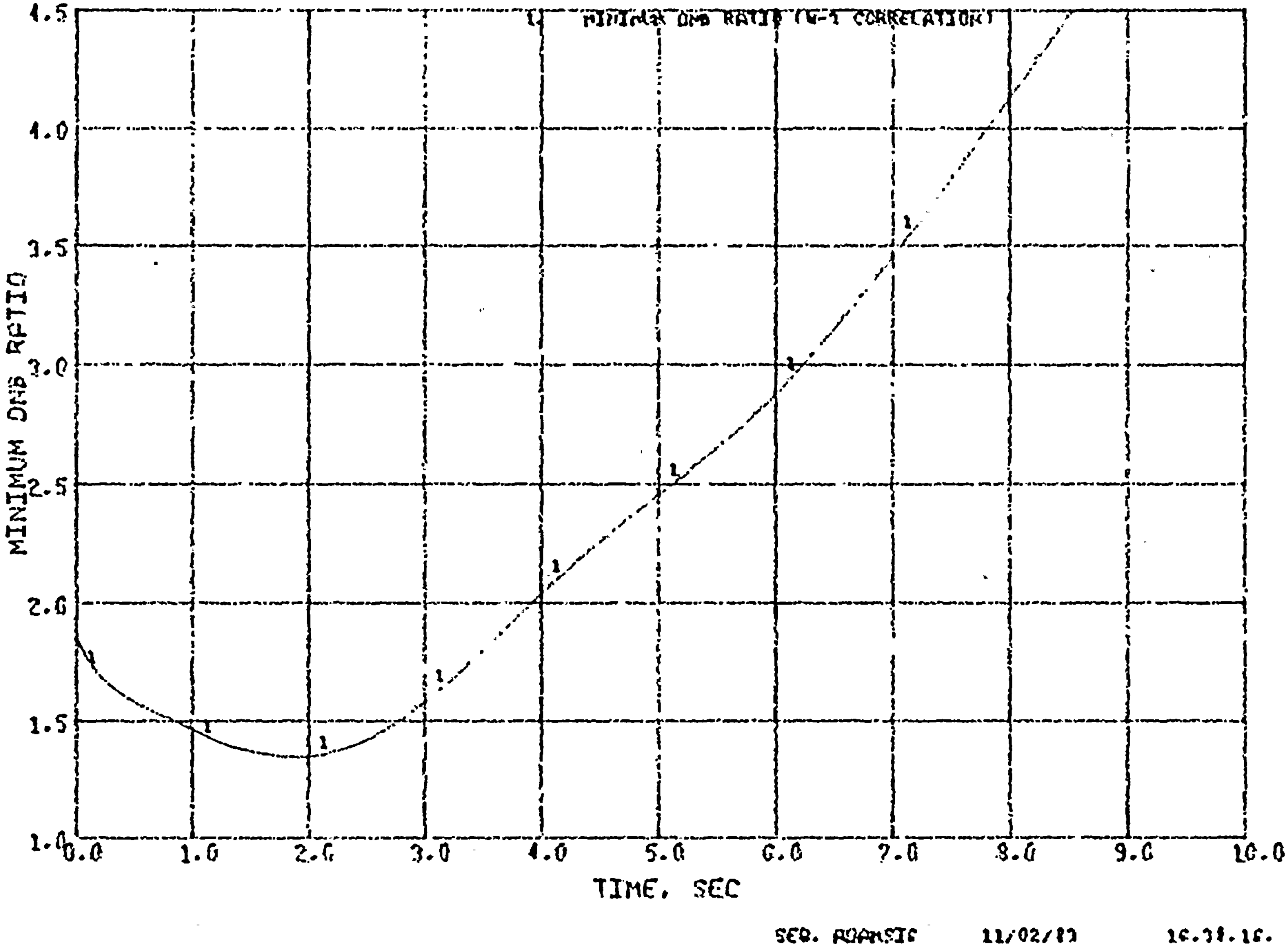

Figure 6: MDNBR for Locked Rotor with Concurrent Loss of Offsite Power (Minimum DNBR Assumptions)

B. RESPONSES TO THE NRC SER FOR D.C. COOK UNIT 2 CYCLE 4

This section provides a commentary on information presented in the Accident and Transient Analysis section (Part C) of the NRC SER for D.C. Cook Unit 2, Cycle 4.

Item 1: The introduction to Part C provides an incomplete list of transient events submitted for NRC Staff review. The correct list of submitted transient event simulations is:

a. Uncontrolled rod withdrawal

b. Loss of coolant flow

c. Locked pump rotor accident

d. Loss of external load

e. Loss of feedwater heating

f. Increase in feedwater flow

g. Excessive load increase

h. Main steamline break accident

i. Rod ejection accident

Item 2: The following statement appears in Section 2 of Part C:

> "The PTSPWR2 model was originally used in a less developed form for a very limited number of anticipated operational occurrences (AOOs). This particular application for the D.C. Cook 2 Cycle 4 reload uses a substantially updated version of the original model and extends the application to an increased number of AOOs and postulated accidents. This updated model and its application to the broader range of accidents has not been subject to generic review to verify and validate its methodology, nor has it received approval on any plant specific application. This generic model has only recently been received (October 1982) on the docket for the D.C. Cook 2 Cycle 4 reload."

The current status of the PTSPWR2 simulation code with respect to application and NRC Staff review follows.

The PTSPWR2 plant transient simulator code is described in XN-74-5, Rev. 1, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTSPWR)," dated May 15, 1975. This report was submitted in 1975 for Staff review and approval in support of the initial insertion of ENC fuel into the H.B. Robinson nuclear power plant. Since that time, PTSPWR2 has been accepted as licensing basis for a number of PWR license amendments involving the simulation of a wide range of anticipated operational occurrences.

In the original application of PTSPWR to H.B. Robinson Unit 2, the following transients were considered:

a. Rod withdrawal events

b. Rod drop

c. Inactive loop startup

d. Reduction in feedwater enthalpy

e. Excessive load

f. Loss of flow

g. Locked rotor

h. Loss of load

i. Steam line break

The ENC D.C. Cook Unit 2 Cycle 4 analysis includes effectively the same transients as for H.B. Robinson Unit 2.

Generic review of PTSPWR2 proceeded to the generation of a draft Safety Evaluation Report in 1977. According to NUREG-0390, Topical Report Review Status, this SER has not been issued because acceptance review criteria are being established for review of accident analysis programs.

During the D.C. Cook Unit 2 Cycle 4 review, ENC was asked for additional information describing PTSPWR2. In response ENC provided XN-CC-38, "User's Manual for PTSPWR2: A FORTRAN Program for Simulation of Pressurized Water Reactor Plant Transients," and XN-CC-38, Supplement 1. These reports expand the initial content of XN-74-5 with user-oriented information, and describe an improved pressurizer model which was first implemented in support of the initial insertion of ENC fuel into the Ft. Calhoun nuclear power plant.

As a result of the Staff's request, Exxon Nuclear Company has requested that the generic review of PTSPWR2 be reactivated [letter, J.C. Chandler (ENC) to Dr. C.O. Thomas (NRC), dated February 3, 1983].

Item 3: The following statement appears in Section 2 of Part C:

> "...however pin diameters have been reduced and the pressure drop of the assembly increased, requiring new XN-DNBR correlations to be validated."

Exxon Nuclear Company's new critical heat flux correlation for PWR fuel designs, XNB, improves the correlation of the critical heat flux phenomenon; its development was not contingent on assembly pressure drop. The ENC reload fuel design has essentially the same pressure drop as the co-resident fuel.

Item 4: Subsection b) of Section 3 of Part C discusses the locked rotor transient. Paragraph 3 of this discussion contains the following statement:

> "For DNBR calculations, reactor core inlet temperature is increased by $4^{o}F$, RCS pressure reduced by 30 psi, and RCS flow reduced by 3-1/2%, compared to rated values."

The RCS flow rate of 142.7×10^{6} lb/hr was reduced by 4-1/2% in the DNBR calculations to obtain a core flow rate effective for heat transfer.

Item 5: Subsection b) of Section 3 of Part C discusses the locked rotor transient. Paragraph 3 of this discussion contains the following statement:

> "Reference 18, Page 21, Figure 23 shows a scram curve which is significantly delayed by approximately 0.4 seconds over that of Figure 14.1-3 of Reference (1) (FSAR)."

The scram time assumed in the calculation is approximately 0.4 seconds sooner than that shown in the FSAR.

Item 6: Subsection b) of Section 3 of Part C discusses the locked rotor transient, and questions the PTSPWR2 simulation model as applied to this event. Validity of the model and its application to this event is contained in Part B of this attachment. In particular, the SER questions the PTSPWR2 pressure calculation for this event. In Part B of this attachment, the validity of the PTSPWR2 system pressure calculation results for the locked rotor event is demonstrated. It is not necessary to rely on the Cycle 2 pressure calculations for Cycle 4 conditions.

Item 7: Subsection b) of Section 3 of Part C offers the following conclusion to the Cycle 4 locked rotor event review:

> "There is a substantive uncertainty in the validity of the Cycle 4 predictions."

There is a 1.5% additional penalty in RCS flow rate not accounted for in the Cycle 4 calculations. The impact on calculated MDNBR of an additional 1.5% flow penalty is a less than 2% reduction in calculated MDNBR. The system pressure impact of this flow penalty will be insignificant.

The differences between Cycle 2 and Cycle 4 locked rotor analyses are explained in the response to General Conclusion 6, Item 1.1. It is considered that the Cycle 4 simulation represents a more realistic and accurate assessment of the actual consequences of this event.

Thus, ENC considers that the Cycle 4 predictions of the locked rotor event are a conservative evaluation of the consequences of the event, and provide a sufficient basis for establishing the acceptability for Cycle 4 operation.

Item 8: Subsection d) of Section 3 of Part C discusses the excessive load increase incident. The following statement appears in subsection d):

> Considering the transients in Reference (30), the results need correction for automated crossflow methodology, an adjustment to the W-3 correlation to allow for mixed flow methodology effects, and a recognition of the further correction required for -1-1/2% RCS flow. Our estimate is 0.95 x 1.52 x 0.98 = 1.415 to be compared with a W-3 value of 1.3 x 1.02 = 1.33.

All MDNBR calculations during the course of this analysis were performed with the XNB critical heat flux correlation. The applicable safety limit for this correlation is 1.17.

Item 9: Subsection d) of Section 3 of Part C offers the following conclusion for the excessive load increase event:

> "The hot spot in the core will be restricted by Technical Specifications (LOCA limit FQ) to approximately 80% of the peak power assumed in the transient analyses; this represents a considerable conservatism in the predicted DNBR and it is on this basis that operation during Cycle 4 is acceptable.

The core thermal conditions calculated in the purposely conservative Cycle 4 PTSPWR2 simulations of this event are much more limiting with respect to MDNBR than any predicted by the July 1982 FSAR simulations. The Cycle 4 result is thus considered by ENC an acceptably conservative evaluation of the consequences of the excessive load increase event.

Item 10: Subsection e) of Section 3 of Part C discusses the excessive heat removal events resulting from feedwater system malfunctions. Subsection e) contains the following statement:

"This event (Accidental Opening of One Feedwater Control Valve with the Reactor at 100% Power at BOC) was proposed as a limiting event by the licensee in References 8 and 18 because of the assumption of a positive moderator coefficient of +5pcm/^{O}F/at BOC over that of Cycle 2, which was taken as 0.0 pcm/^{O}F. We question this because Cycle 3 was recalculated on the bases of +5 pcm, and this particular event was not included for consideration. (Reference FSAR Reference 1, Appendix 14 B, I.C.)"

The event has not been described as limiting due to the assumption of a positive moderator coefficient. The Cycle 2 FSAR has established that the event is not limiting with respect to fuel or vessel design limits. Further comments regarding this event have been included in the response to Items 2, 3 and 4 in Part B of this Attachment.

Item 11: Subsection e) offers the following conclusion to the NRC review of the excessive heat removal events:

"Transient calculations for Cycle 4 were performed using unapproved analytic models. The licensee asserts that the margin to DNBR limits have been demonstrated using these models. Based upon the limited staff review to date, the staff concludes that these predicted margins to DNBR can be over-estimated and that a detailed review could substantively erode these margins. However, the hot spot in the Cycle 4 core will be restricted by Technical Specifications (LOCA limit FQ) to approximately 80% of the peak power assumed in the transient analysis; this represents a considerable conservatism in the predicted DNBR and it is on this basis that operation during Cycle 4 will be acceptable."

The review status of the PTSPWR2 simulation code models is given in the response to Item 2 of Section C.

The margin to DNBR limits has been conservatively evaluated in the Cycle 4 calculations, and appropriate justification to establish such conservatism has been provided in this Response and other information during the course of the Cycle 4 review. Thus, the Cycle 4 calculations provide a sufficient basis for establishing the acceptability of Cycle 4 operation.