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John C. Brons
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IPN-85-40

Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Mr. Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Additional Information Regarding Asymmetric Steam Generator Tube Plugging Analyses and Cycle 4/5 Reload

References:
1) Letter from J. P. Bayne to S. A. Varga dated March 27, 1985 (IPN-85-15) entitled: "Asymmetric Steam Generator Tube Plugging Technical Specifications."

2) Letter from J. P. Bayne to S. A. Varga dated April 23, 1985 (IPN-85-21) entitled: "Proposed Technical Specifications Regarding the Cycle 4/5 Refueling."

Dear Sir:

Attachment A to this letter serves to respond to your June 12, 1985 letter which requested additional information concerning the transient analyses transmitted via Reference 1.

Reference 2 served to transmit proposed technical specifications in support of the Cycle 4/5 refueling, which involves a fuel design transition from the Westinghouse 15 x 15 low parasitic design to the 15 x 15 Optimized Fuel Assembly (OFA) design. As the OFA design results in an increased control rod drop time, the "fast" transients for which the reactor protection system trips the reactor within a few seconds after transient initiation, were quantitatively re-analyzed. As the locked rotor transient is a "fast" transient, it was quantitatively re-analyzed and the assumption of asymmetric steam generator tube plugging was employed. The methodology utilized to account for asymmetric steam generator tube plugging is described in WCAP-10704, "Safety Evaluation for Indian Point 3 with Asymmetric Tube Plugging Among Steam Generators (Proprietary)" dated October 1984, which was transmitted by Reference 1. The technical specifications transmitted via Reference 1 support plant operation during Cycle 4 with asymmetric steam generator tube plugging. As Cycle 4 operation has been completed, the requested additional information regarding the transient analyses transmitted via Reference 1 is germane to those analyses transmitted via Reference 2.

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During a June 14, 1985 telephone conversation with Mr. C. Kopp of the staff, additional information regarding the methodology utilized to determine the total core peaking factor, F_Q, during the fuel cycle was requested. In accordance with the approved methodology described in WCAP-9272 "Westinghouse Reload Safety Evaluation Methodology (proprietary)", the height dependent F_Q x Power values for a subset of the "18 cases" were determined. A conservative factor was subsequently applied to these values. The resulting F_Q x Power values were compared to the Technical Specification limit, thereby demonstrating that this limit will be satisfied throughout the fuel cycle.

The dropped rod transient for Indian Point 3 was reanalyzed to correct for a turbine runback setpoint error, for both the asymmetric and uniform steam generator tube plugging cases. The analysis results for the asymmetric and uniform steam generator tube plugging cases are provided in Attachments B and C, respectively. The results for the uniform steam generator tube plugging are presented in FSAR format. In order to maintain historical accuracy of the FSAR, the uniform steam generator tube plugging results will be included in the 1986 annual FSAR Update.

Should you or your staff have any questions regarding this matter, please contact Mr. P. Kokolakis of my staff.

Very truly yours,

John C. Brons
Senior Vice President
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cc: Resident Inspector's Office
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Attachment A to IPN-85-40
Additional Transient Information

New York Power Authority
Indian Point 3 Nuclear Power Plant
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1. NRC Request

The locked rotor transient should be performed assuming that all fuel which experiences a DNBR of less than 1.3 fails and the offsite dose consequences should be calculated. In the offsite dose analysis you should assume maximum Technical Specification pre-accident coolant activity and steam generator leakage. Single failures should be considered including a stuck open secondary relief valve. Loss of offsite power should be assumed per GDC-17. The effect of steam generator tube uncovery on the offsite dose consequences caused by operator action to isolate the affected steam generator should also be considered in the analysis.

Authority Response

The locked rotor analyses transmitted by References 1 and 2 did not address the offsite doses as rods in DNB were predicted to not fail based on the temperature - time criteria presented in NUREG-0562, "Fuel Rod Failure as a Consequence of Departure from Nucleate Boiling or Dryout." As the locked rotor transient is rapidly mitigated, the elapsed time at which the loss of offsite power is assumed to occur significantly impacts the resulting offsite dose. The postulated loss of offsite power has been assumed to be a result of a grid instability initiated by the resultant turbine trip. This grid instability could include a low voltage or under-frequency trip of the reactor coolant pumps. In order to assess the credibility of this loss of offsite power assumption, the Authority has performed a computer simulation of an Indian Point 3 unit trip. The model for the initial system load was conservatively assumed to exhibit characteristics of constant current for real power and constant admittance for reactive power. The demand on the grid was assumed at a maximum value.

The computer simulation results indicate that a unit trip causes a relatively minor transient disturbance at the 6.9 KV buses, which power the reactor coolant pumps. The minimum frequency and voltage at the 6.9 KV buses are 59.96 Hz and 5.9 KV, respectively. As the reactor coolant pump under-frequency and low voltage trip setpoints are nominally 57.5 Hz and 5.175 KV, respectively, flow coastdown in the unaffected reactor coolant loops will not occur.

In accordance with discussions with the NRC staff, the Authority has re-analyzed the locked rotor transient excluding the assumptions of a loss of offsite power, and a stuck open secondary relief valve, both of which are outside of the Indian Point 3 licensing basis. A description of this re-analysis follows.

Method of Analysis

An analysis was performed assuming the locked rotor is in Loop 4 (plugging level = 8%) because a locked rotor in the least plugged loop will result in the lowest core flow. Tube plugging parameters were input to LOFTRAN to provide a 30%, 24%, 24%, 8% tube plugging distribution in Loops 1 through 4 respectively. The reactor coolant pump dynamic responses are based on the pump homologous curves developed from test data. Offsite power is assumed.

Plant responses were computed using the LOFTRAN code. Fuel and clad temperatures at the hot spot were computed using the FACTRAN code. The DNB ratios were calculated using the THINC code. A hot spot analysis was performed to demonstrate a coolable geometry.

A DNBR analysis was performed to obtain the percentage of rods in DNB for the offsite dose releases.

Results

The time sequence of events for the locked rotor transients are given in Table I. Nuclear power, core heat flux, RCS pressure, core flow and clad inner temperature are shown in Figures 1 through 3 for the most limiting case. A peak clad temperature of 1838°F and a peak reactor coolant system pressure of 2558 psia were conservatively calculated to be reached during the transient. The most limiting case, full Optimized Fuel Assembly core, yielded 1.9% rods in DNB with offsite power available.

The Authority believes that there is evidence for replacing the DNBR fuel failure criteria for "fast" transients with a time-temperature model. This evidence has been documented in NUREG-0562. However, in accordance with your request, the conservative assumption that rods in DNB fail was utilized to determine the resultant offsite doses. The offsite dose analysis includes an assumed Technical Specification primary to secondary leakage limit of 1 gpm. The condenser was assumed to be unavailable and hence the steam dump was to the secondary pressure relief valves. The Iodine partition factor was assumed to be 0.01. The initial reactor coolant activity was conservatively assumed to be greater than the Technical Specification limit. The resultant offsite doses are well within the 10 CFR 100 limits.

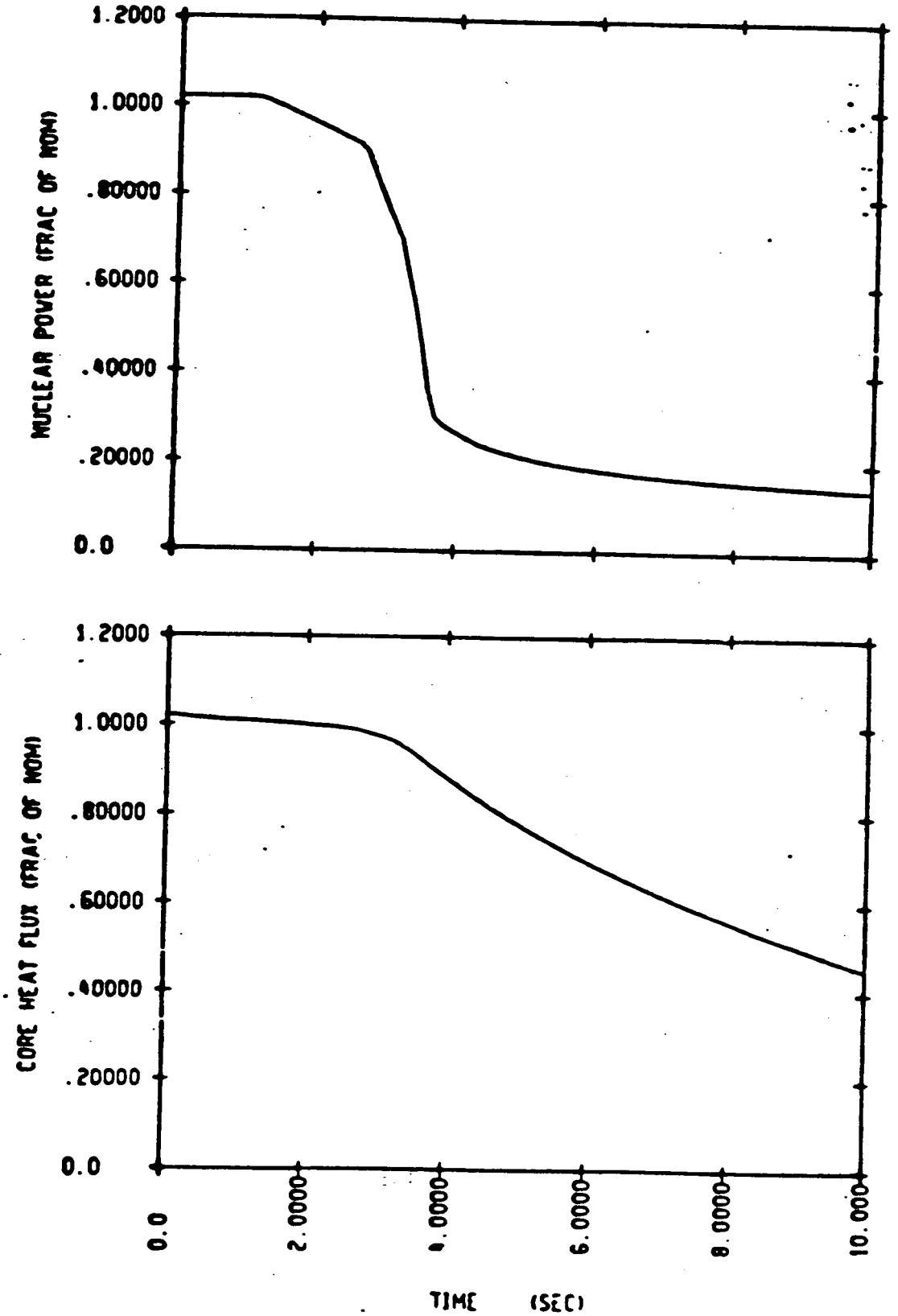
Conclusions

A locked rotor event during plant operation with an equivalent steam generator tube plugging level of up to 30% in any steam generator while the equivalent average plugging level in all steam generators is less than or equal to 24%, will not result in the exceeding of any safety criteria or dose release limits.

TABLE I

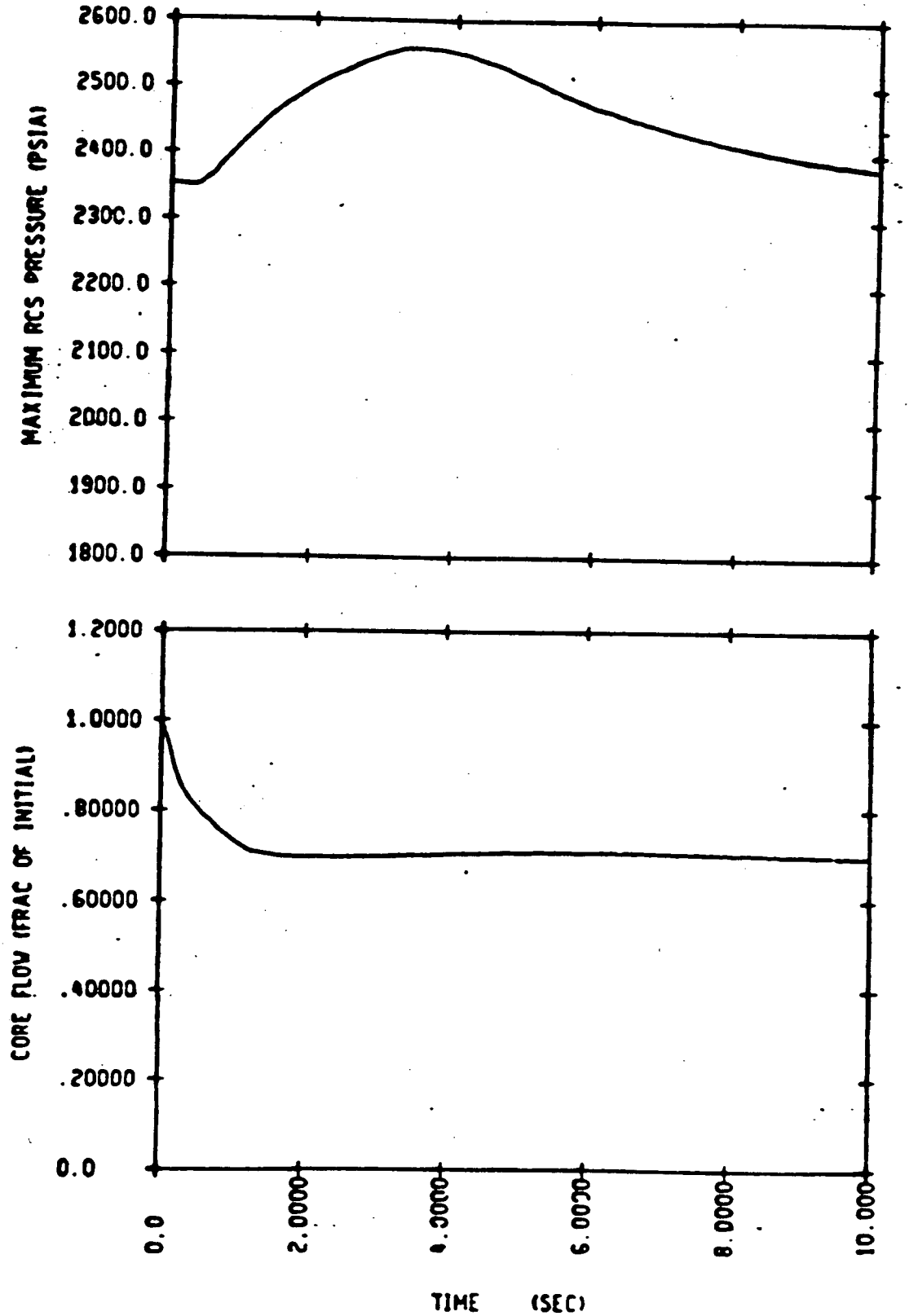
THE SEQUENCE OF EVENTS

ACCIDENT	EVENT	TIME (Sec.)
Reactor Coolant	Rotor on one pump locks	0.0
Pump Shaft Seizure	Low flow trip point reached	0.1
(Locked Rotor)	Rods begin to drop	1.1
	Maximum clad temperature occurs	3.5
	Maximum RCS pressure occurs	3.5



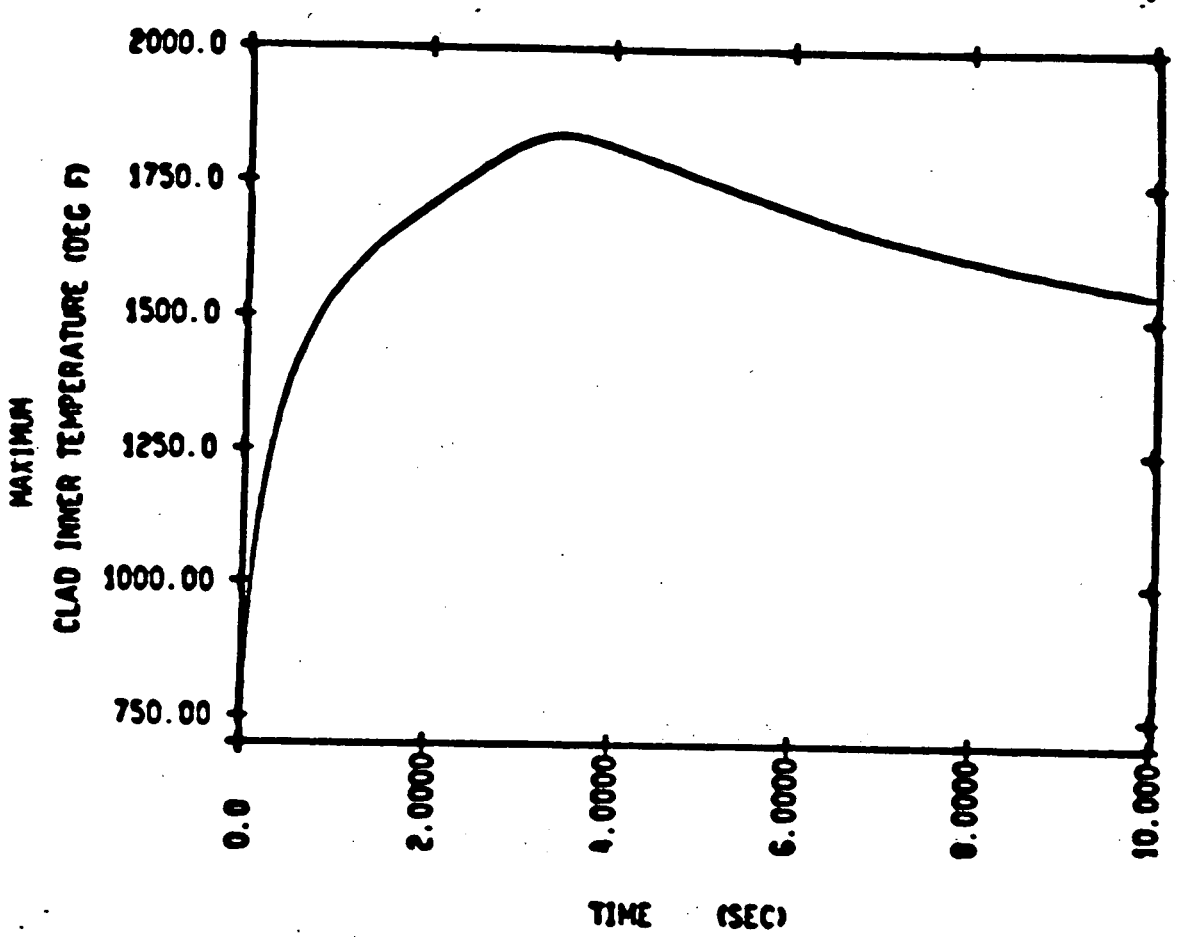
Indian Point Unit 3
Asymmetric Tube Plugging Analysis
Locked Rotor with Offsite Power

Figure 1



Indian Point Unit 3
Asymmetric Tube Plugging Analysis
Locked Rotor with Offsite Power

Figure 2



Indian Point Unit 3
Asymmetric Tube Plugging Analysis
Locked Rotor (Hot Spot)

Figure 3

2. NRC Request

Below 10% reactor power the P-7 interlock does not permit reactor trip on low reactor system flow.

(a) Justify that the consequences of loss of coolant flow while at 10% power without reactor trip would be acceptable.

(b) Justify that the consequences of a locked rotor accident while at 10% power would be acceptable.

Authority Response

The P-7 interlock is a standard design feature on Westinghouse plants and is not affected by asymmetric tube plugging or OFA transition. As pointed out in WCAP-10704, the cold leg temperatures are essentially equal to the temperature on the secondary side of the steam generator independent of tube area, and the increments in steam generator delta-P associated with tube plugging are very small when compared to the locked rotor impedance.

Indian Point 3 Technical Specification 3.1 states that reactor heat equivalent to 10% rated power can be removed with natural circulation. A calculation, which is bounding with respect to (a) and (b) has been performed. Natural circulation conditions at 10% rated power were calculated assuming conservative flow resistances including steam generator tube plugging and a lock rotor in each loop. This assessment indicated that even at 10% power, natural circulation is adequate to transport heat from the reactor core to the steam generators. The calculated flow was 6.8% of thermal-hydraulic design, (323,600 gpm), and hot leg temperatures were below saturation. Calculations of DNBR at these conditions were made using the W-3 correlation. Although the flow rate for this evaluation is outside the range of applicability for the W-3 correlation, the DNBR predicted at these low flow conditions is considered to be conservative. Results of the DNB evaluation show a minimum DNBR of $\sim$7, which is significantly above the applicable limit.

3. NRC Request

The analyses of the reactor response to postulated accidents did not include the consequences of the rupture of a main feedwater line. Provide the analysis and predicted consequences of this accident.

Authority Response

This transient is not analyzed for Indian Point 3 as it is outside of the plant's licensing basis.

Attachment B to IPN-85-40
Dropped Rod Reanalysis Results
Asymmetric Steam Generator Tube Plugging Case

New York Power Authority
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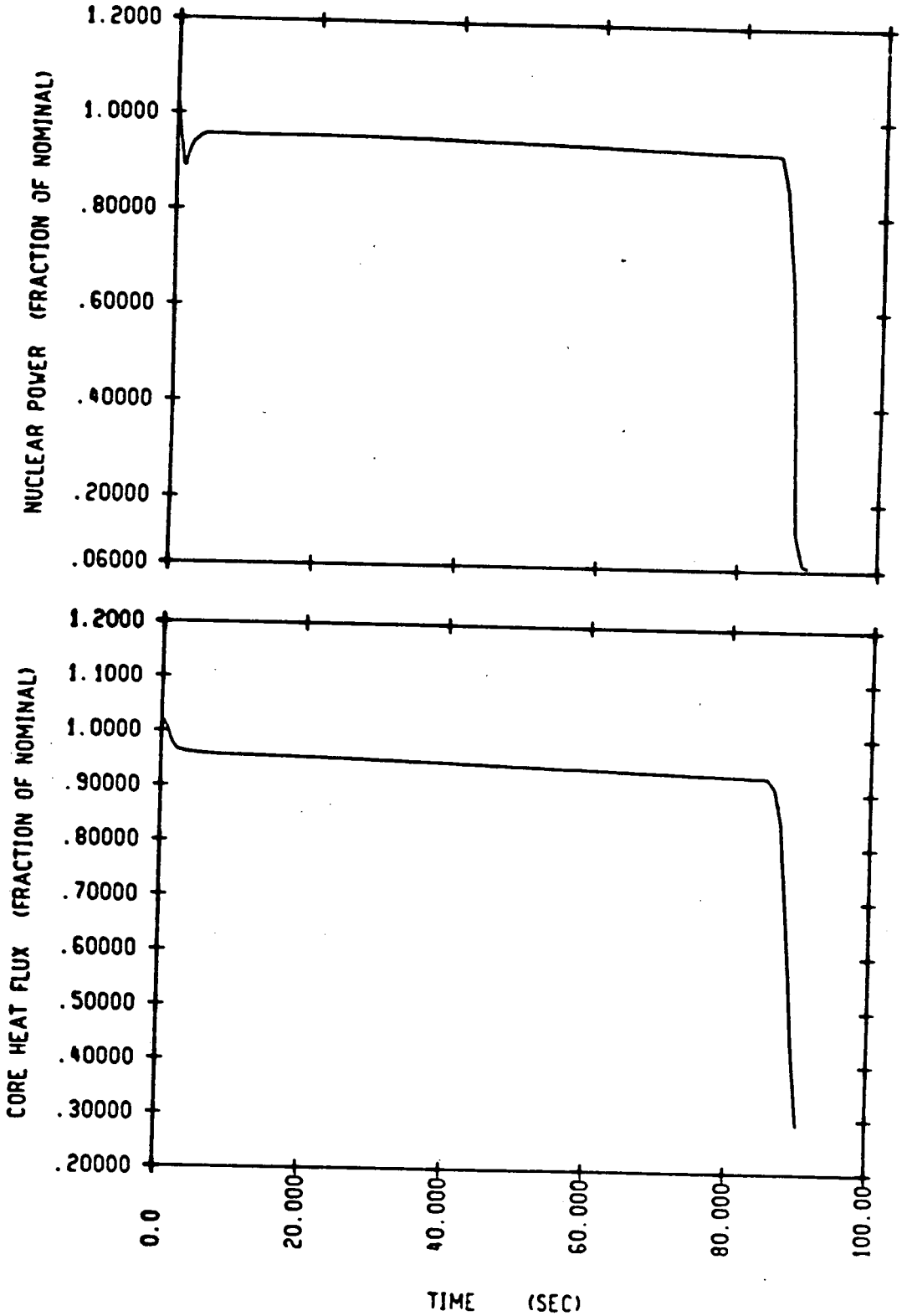
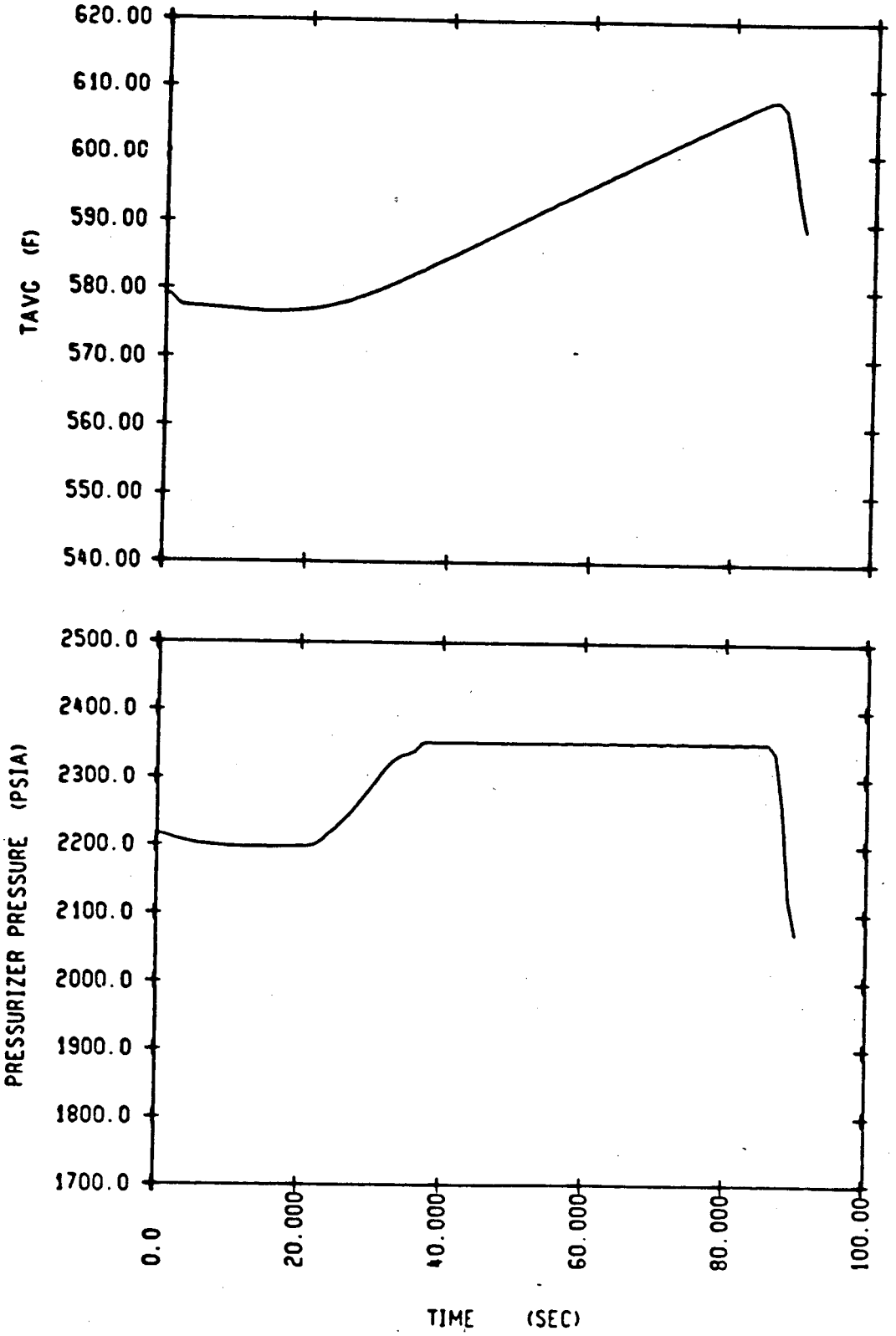
THE SEQUENCE OF EVENTS

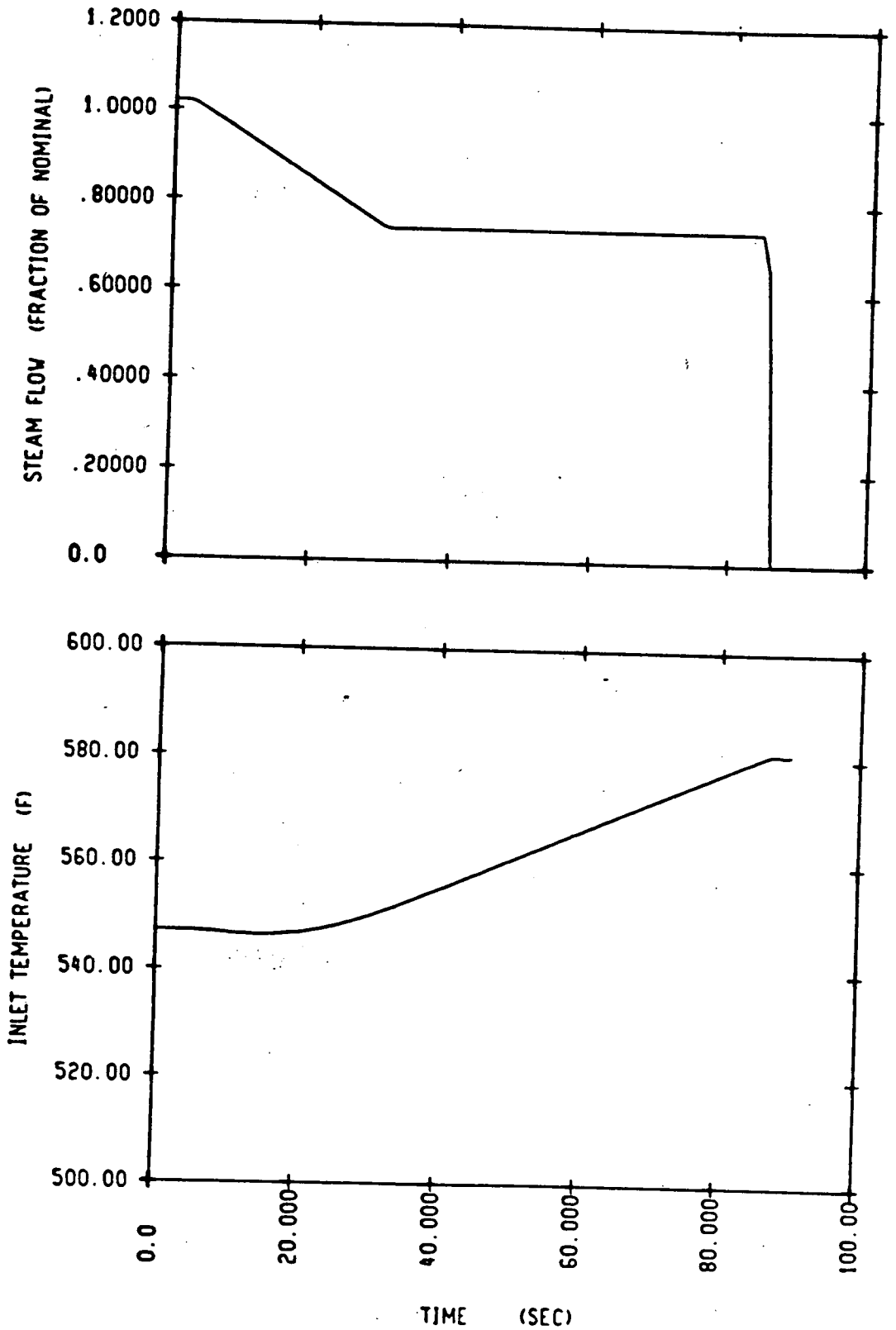
ACCIDENT	EVENT	TIME (Sec.)
Dropped Rod	Initiation of a rod drop (100 pcm)	0.0
	Overtemperature delta-T Reactor trip signal initiated	83.1
	Rods begin to drop	85.1



INDIAN POINT UNIT 3
ASYMMETRIC TUBE PLUGGING ANALYSIS
DROPPED RCCA OF WORTH 100 PCM



INDIAN POINT UNIT 3
ASYMMETRIC TUBE PLUGGING ANALYSIS
DROPPED RCCA OF WORTH 100 PCM



INDIAN POINT UNIT 3
ASYMMETRIC TUBE PLUGGING ANALYSIS
DROPPED RCCA OF WORTH 100 PCM

Attachement C to IPN-85-40
Dropped Rod Reanalysis Results
Uniform Steam Generator Tube Plugging Case

New York Power Authority
Indian Point 3 Nuclear Power Plant
Docket No. 50-286

14.1.4 Rod Cluster Control Assembly (RCCA) Drop

Analyses and results for nonplugging situation are included in Section 14.1.3, Rod Assembly Misalignment.

Dropped Rod/Dropped Bank (with 24% tube plugging)

These transients were reanalyzed to assure that a return to power would not occur that would result in violating the DNBR criteria. After a rod or bank drops, the peaking factor increases and, when coupled with a high return to power, may result in a DNBR violation. This type of transient will be aggravated by the 24% tube plugging because of the reduced reactor coolant flow rate and increased average temperature.

Method of Analysis

The return to power was investigated assuming the following.

1. The following initial operating conditions, most adverse with respect to the margin to DNBR, were used.

   a. Power = 1.02% of nominal

   b. Pressure = 2216 psia

   c. RC Temperature = 579.3°F

2. Feedback - maximum MTC for dropped bank
   minimum MTC for droopped rod

3. Rod Stop - manual control assumed

The dropped rod/bank accidents were simulated using the LOFTRAN code. Analyses were performed for the following rod or bank worths.

a. 50 pcm - minimum rod worth
b. 300 pcm - maximum rod worth
c. 370 pcm - minimum bank worth
d. 480 pcm - minimum bank worth for highest peaking factor
e. 1200 pcm - maximum bank worth

Results

The return to power for the five cases evaluated are the following.

Case	Worth/Type	Return to Power (fraction of nominal)	Comment
1	50/rod	0.969	min
2	300/rod	0.788	max
3	370/bank	0.8967	min
4	480/bank	0.8573	min (highest peaking factor)
5	1200/bank	0.6413	max

DNB calculations were performed for all cases at the state point power level. Power vs time are shown in Figures 14.1-12A and 14.1-12B for Cases 1 and 5 respectively.

Conclusions

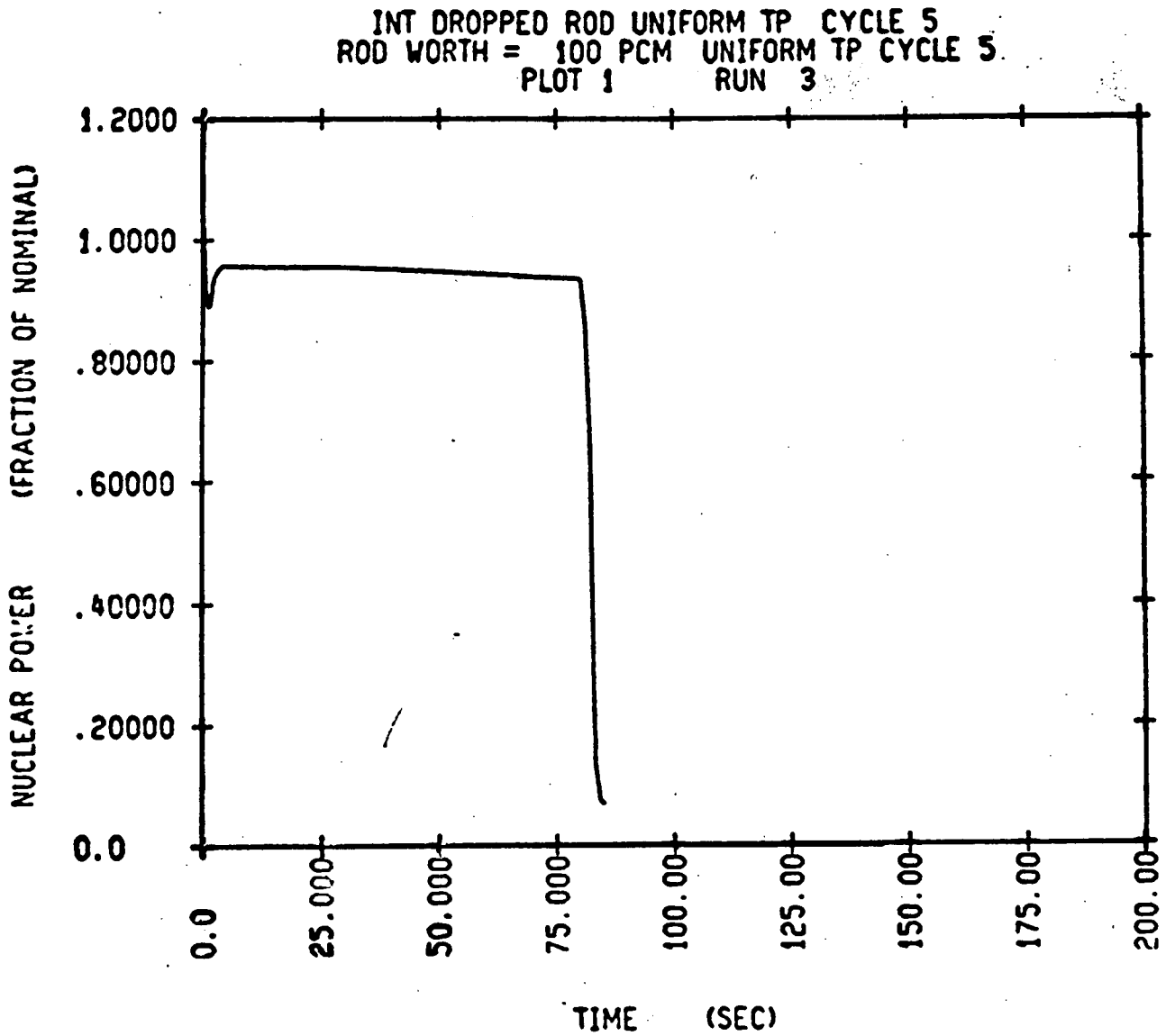
Steam generator tube plugging has a small adverse affect on the Dropped Rod or Dropped Bank accident. It has been verified that the DNBR criteria will not be violated for the return to power levels computed for the dropped rod or dropped bank accident.

14.1.5 Chemical and Volume Control System Malfunction

Reactivity can be added to the core by feeding primary grade water into the Reactor Coolant System via the reactor makeup portion of the Chemical and Volume Control System. The normal dilution procedures call for a limit on the rate and magnitude for any individual dilution, under strict administrative controls. Boron dilution is a manual operation. A boric acid blend system is provided to permit the operator to match the boron concentration of reactor coolant makeup water during the normal charging to that in the Reactor Coolant System. The Chemical and Volume Control System is designed to limit, even under various postulated failure modes, the potential rate of dilution to a valve which, after indication through alarms and instrumentation, provides the operator sufficient time to correct the situation in a safe and orderly manner.

The opening of the primary water makeup control valve provides the only supply of makeup water to the Reactor Coolant System which can dilute the reactor coolant. Inadvertant dilution can be readily terminated by closing this valve. In order for makup water to be added to the Reactor Coolant System, at least one charging pump must also be running in addition to the primary makeup water pumps.

The rate of addition of unborated water makeup to the Reactor Coolant System when it is not at pressure is limited by the capacity of the primary water supply pumps. The maximum addition rate in this case is 300 gpm with both



FSAR FIGURE 14.1 - 12A