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L-93-319

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
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Gentlemen:

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Request for Additional Information (RAI) -
Generic Letter 93-04, Rod Control System Failure and
Withdrawal of Rod Control Cluster Assemblies

By letter L-93-186, dated August 4, 1993, Florida Power and Light Company (FPL) responded to questions regarding Generic Letter 93-04, Rod Control System Failure and Withdrawal of Rod Control Cluster Assemblies. By letter dated November 15, 1993, the NRC requested additional information to support the review of FPL's response to Generic Letter 93-04. The response to the NRC request is enclosed.

Should there be any questions, please contact us.

Very truly yours,

T F Plunkett

T. F. Plunkett
Vice President
Turkey Point Nuclear

Enclosure

TFP/CLM/cm

cc: S. D. Ebneter, Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, Turkey Point Nuclear

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STATE OF FLORIDA)
) ss.
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T. F. Plunkett being first duly sworn, deposes and says:

That he is Vice President, Turkey Point Nuclear, of Florida Power and Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this document are true and correct to the best of his knowledge, information and belief, and that he is authorized to execute the document on behalf of said Licensee.

T F Plunkett

T. F. Plunkett

Subscribed and sworn to before me this

28 day of DEC, 1993.

Cheryl A Kelly

CHERYL A. KELLY

Name of Notary Public (Type or Print)

NOTARY PUBLIC, in and for the County of Dade, State of Florida

CHERYL A. KELLY
MY COMMISSION # CC 223781
EXPIRES: September 27, 1996
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My Commission expires
Commission No.

T. F. Plunkett is personally known to me.

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION ON FPL's RESPONSE TO GENERIC LETTER 93-04 AND THE SALEM ROD CONTROL SYSTEM FAILURES

BACKGROUND

On May 27, 1993, Salem Unit 2 experienced the uncontrolled withdrawal of a single Rod Cluster Control Assembly (RCCA). The movement of this single RCCA was initially postulated to have resulted from control system logic cabinet card failures (possibly the result of a single initiating failure) coupled with failures and/or effects that had not yet been identified. If this Salem event had been the result of a single failure, the uncontrolled single rod withdrawal event of May 27th would have placed the Salem plant outside of its stated FSAR design basis, with the potential for a core power distribution not considered in their original design basis analysis.

As a result of this event, the NRC issued Generic Letter 93-04 (Reference 1), which requested a written response from Westinghouse licensees under the requirements of 10 CFR 50.54(f). Under Generic Letter 93-04, each licensee was required to provide technical/licensing information to the NRC, which addressed the design basis of the plant with regard to a single failure in the Rod Control System and specified what type of short and long term corrective actions had been taken or were planned for resolution of this issue. In response to Generic Letter 93-04, the design and licensing basis for the Turkey Point Rod Control System was examined and an FPL response was forwarded (Reference 2) based on the best information which was available from Salem at the time.

Upon review of the FPL response to Generic Letter 93-04, the NRC requested additional information as identified in their correspondence of November 15, 1993 (Reference 3). This discussion will serve to clarify FPL's initial response to NRC Generic Letter 93-04 and respond to the NRC Request for Additional Information by providing additional details of the FPL analysis that was performed for uncontrolled asymmetrical control rod withdrawal events.

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION

NRC REQUESTED ADDITIONAL INFORMATION

In response to the FPL response to Generic Letter 93-04, the NRC requested that FPL submit additional information to support its conclusions concerning the Turkey Point design basis for uncontrolled asymmetrical control rod withdrawal events. Specifically, the NRC Staff requested that "In support of your conclusions that you meet the licensing basis for asymmetrical control rod withdrawal events, please provide information and detailed discussions on the application and use of the computer codes, and a comparison of your analysis results with the UFSAR for its applicability, margin to DNBR and validity of analysis for all future cycles."

FPL RESPONSE

1.0 Method of Analysis

FPL analyses examined the minimum departure from nucleate boiling ratio (MDNBR), fuel centerline temperature, reactor coolant system (RCS) pressure and the steam generator pressure to ensure that fuel design safety limits and pressure boundary integrity are maintained during an uncontrolled asymmetric RCCA withdrawal event.

Uncontrolled bank, group, double rod and single rod withdrawals from 100%, 80%, 60%, 10% and hot zero power (HZP) were analyzed using the SIMULATE-3, RETRAN-02 and VIPRE-01 computer codes. The loss in DNBR margin was compared to the available margin to ensure that sufficient margin is available to accommodate an uncontrolled asymmetric RCCA withdrawal event. The peak fuel centerline temperature and system pressures were compared to the safety limits to ensure fuel and pressure boundary integrity.

The procedure used by FPL in the analysis of uncontrolled asymmetric RCCA withdrawal events is described below:

a) Using the SIMULATE-3 physics code, analyze bank, group, double rod and single rod withdrawals at preselected power levels identified in the FSAR (Reference 23) to calculate the rate of reactivity insertion and the peak $F\Delta H$. Table 1 lists the combinations of uncontrolled RCCA withdrawals analyzed.

b) Identify maximum post withdrawal $F\Delta H$ for each of the bank, group, double rod and single rod events. These cases are listed in Table 2.

c) Perform system thermal/hydraulics analyses for the cases identified in step (b) using the RETRAN-02 computer code. Obtain system conditions (core power and inlet temperature) at discrete time intervals up to and including the time of reactor trip.

d) For the cases identified in step (b) re-calculate the power distribution at selected time intervals using the SIMULATE-3 code with the system conditions obtained in step (c). This has the effect of crediting the mitigating effects of reactivity feedback in the calculation of the post withdrawal power distribution.

e) For each case analyzed in step (d), obtain the $F\Delta H$ augmentation factor by dividing the post withdrawal highest $F\Delta H$ by the pre-withdrawal $F\Delta H$.

f) Normalize all the power distributions using the assumption that at the time of RCCA withdrawal, the $F\Delta H$ in the hottest assembly corresponds to the Technical Specification (Reference 24) Limit. Apply the $F\Delta H$ augmentation factors obtained in step (e) to this limit to yield the values of $F\Delta H$ to be used in the DNBR calculation. In addition, a 4% calculational uncertainty is added to the $F\Delta H$ values in the at-power cases. An 8% uncertainty is used for the HZP cases.

g) For all cases, perform DNB analysis of the hottest channel using the VIPRE-01 computer code with the $F\Delta H$ obtained in step (f) and the core conditions at the corresponding time steps obtained from the transient simulations of step (c).

h) For all cases, compare the minimum DNBR calculated in step (g) for the bank withdrawal to the corresponding minimum DNBR for the group, double rod and single rod uncontrolled withdrawal cases. Calculate the percent change in minimum DNBR for all cases relative to the bank withdrawal. This change in minimum DNBR represents the DNBR penalty for an asymmetric uncontrolled rod withdrawal relative to the uncontrolled withdrawal of a bank. Tables 3, 4 and 5 list these penalties for asymmetric rod withdrawal from various power levels.

i) Compare the percent changes obtained in step (h) to the available DNB margin for the uncontrolled bank withdrawal events.

The uncontrolled RCCA bank withdrawal analyses described in sections 14.1.1 and 14.1.2 of the UFSAR have been redone by Westinghouse using their latest methodology, the Revised Thermal Design Procedure (RTDP) (References 4 and 5). The measurement and code uncertainties are statistically combined in this procedure rather than deterministically combined as in the Standard Thermal Design Procedure (STDP) for the analysis of non-LOCA transients.

The RTDP analyses have been completed and QA verified by Westinghouse. The Westinghouse RTDP methodology has been reviewed and approved by the NRC on a generic basis for application to the analysis of non-LOCA transients in nuclear power plants (Reference 6). However, plant specific analyses for Turkey Point have not been reviewed and approved by the NRC. FPL is scheduled to submit these analyses to the NRC in 1994.

The RTDP analysis results for the uncontrolled RCCA bank withdrawal event for Turkey Point were used as the basis for assessment of the available DNB margin.

j) Compare the maximum fuel centerline temperature calculated in step (g) to the fuel temperature safety limit of 4800 °F (Reference 5).

k) Compare the maximum RCS pressure to the design limit of 2750 psia (Reference 5).

l) Compare the maximum steam generator secondary pressure to the design limit of 1210 psia (Reference 5).

m) The above analyses from 100% reactor power were performed both with the beginning of cycle (BOC) minimum reactivity feedback and the end of cycle (EOC) maximum reactivity feedback conditions. It was recognized that the minimum feedback transients were more severe in terms of the peak reactor power reached during the transient (Table 3). All other power level analyses were performed with minimum reactivity feedback only.

2.0 Significant Assumptions

A number of conservative assumptions are used in the uncontrolled asymmetric RCCA withdrawal analyses. The most significant among these are the following:

i) An $F_{\Delta H}$ corresponding to the Technical Specification Limit is assumed for the fuel pin most affected by an uncontrolled rod withdrawal at the time of event initiation for both OFA and LOPAR fuel assemblies.

ii) Moderator and Doppler reactivity feedback was simulated to represent beginning-of-cycle (BOC) and end-of-cycle (EOC) conditions. The BOC conditions include +5 pcm/^{o}F for moderator temperature coefficient and -1.0 pcm/^{o}F for Doppler temperature coefficient. The corresponding values for the EOC are 0.0 pcm/^{o}F and -2.9 pcm/^{o}F. The BOC conditions provide minimum negative feedback while the EOC conditions provide maximum negative feedback. These values are conservative relative to the Technical Specification limit of 0.0 pcm/^{o}F to -35 pcm/^{o}F for moderator temperature coefficients at full reactor power.

3.0 Application and Use of Computer Codes

3.1 SIMULATE-3 Computer Code

This is a three dimensional two group (fast and thermal) physics nodal code developed by Studsvik (References 7, 8 and 9), and reviewed and approved by the NRC (References 10 and 11) for reactor physics analysis. The Turkey Point model simulates four quarters of each of its 157 fuel assemblies, each divided into 24 vertical nodes of six inch length. Power distribution in each pin of its 15x15 lattice is inferred from the power calculated for each quarter assembly. Detailed cross sections are generated by the multi-group transport computer code CASMO-3.

SIMULATE-3 analyses have been compared with plant measured data to demonstrate accuracy of calculation (Reference 12). Attachment 1 to this report provides comparisons of such analyses. SIMULATE-3 predictions agree with the plant measured data within established acceptance criteria described in the attachment.

This code was used to simulate uncontrolled bank, group, double rod and single rod withdrawal from power and hot zero power conditions. Table 1 lists the power levels and the RCCA combinations analyzed in the uncontrolled withdrawal analysis. All of the at-power cases were simulated at the maximum rod withdrawal rate of 72 steps per minute. The manual rate of RCCA withdrawal at the plant is set at 68 steps per minute. The automatic withdrawal rate is variable depending upon the error between T_{ref} and T_{ave}. However, the plant is operated with the rod control in manual and the automatic rod withdrawal signal has been disconnected from the rod motion controller. The HZP case (uncontrolled bank withdrawal case only) was simulated with a 75 pcm/sec reactivity insertion rate to match the UFSAR analysis for bank withdrawal. The remaining HZP cases were simulated at the rod withdrawal rate of 72 steps per minute.

Results from the corresponding RETRAN-02 calculations were used in SIMULATE-3 to vary core inlet temperature and power level to simulate moderator and Doppler feedback as the transient progresses. Results of the SIMULATE-3 analyses for peak FΔH and reactivity insertion rates are presented in Table 2.

Power distributions calculated by SIMULATE-3 for the uncontrolled RCCA withdrawal cases were provided to VIPRE-01 for DNB analysis.

3.2 RETRAN-02 Computer Code

This is a one-dimensional nodal thermal/hydraulics code with a point reactor kinetics model (Reference 13). The code has been developed by EPRI and has been reviewed and approved by the NRC for application to light water reactors (Reference 14). NRC has also reviewed and approved the use of RETRAN-02 by FPL for the Turkey Point nuclear power plant (Reference 15).

Several benchmark cases were presented to the NRC in support of the above review and subsequent safety evaluation (Reference 16). Bank withdrawal from full power was simulated and compared with the UFSAR results, provided here as Attachment 2.

Reactivity insertion rates corresponding to the bank, group, double rod and single rod withdrawal cases (corresponding to the highest FΔH) were simulated using RETRAN-02. Reactor power increases until the reactor trip occurs either on high nuclear flux or Overtemperature delta T. The analyses were performed both for minimum and maximum moderator and Doppler feedback.

Reactor coolant flow, temperature and pressure calculated by RETRAN-02 were provided to VIPRE-01 for DNB analysis.

Table 3 presents results from the RETRAN-02 analyses of uncontrolled withdrawal of various RCCA combinations at different power levels. Tables 4 and 5 summarize the RETRAN-02 analysis results for the limiting case (which causes the maximum DNBR penalty) which is a single uncontrolled RCCA withdrawal from 60% reactor power with the BOC feedback conditions.

3.3 VIPRE-01 Computer Code

VIPRE-01 is a thermal/hydraulics computer code developed by EPRI for DNB analysis and approved by the NRC (References 17 and 18). An eighth core model with forty-nine axial nodes was used for Turkey Point. Westinghouse WRB-1 correlation was used for the Optimized Fuel Assemblies (OFA) and W3 L-Grid (W-3L) correlation was used for the Low Parasitic Fuel Assemblies (LOPAR). Turkey Point is in the final stage of transition from LOPAR to OFA fuel.

For the at-power asymmetric RCCA withdrawal cases, VIPRE-01 computer code was used to calculate the minimum DNBR, which is the limiting safety criterion. For the HZP cases, the VIPRE-01 computer code was used to analyze the fuel centerline temperature to ensure that the fuel safety limit was met.

To ensure confidence in the VIPRE-01 model for Turkey Point, state point values at the time of minimum DNBR in the Westinghouse RTDP analysis of an uncontrolled bank withdrawal from 60% reactor power were simulated. This analysis has been performed using a reactivity insertion rate of 1 pcm/sec, and the minimum Doppler and moderator reactivity feedback. These state point values are:

Peak reactor power	100%
RCS pressure	2350 psia
RCS T_{ave}	605 ^oF

The minimum DNBR using the W-3L correlation as predicted by VIPRE-01 was 1.80 compared to a value of 1.84 calculated by Westinghouse. This is considered to be a good agreement.

Table 3 provides results from the VIPRE-01 analysis for the minimum DNBR for various power levels and rod combinations. Tables 4 and 5 summarize the results obtained from the VIPRE-01 analyses for the limiting case (which causes maximum DNBR penalty) which is a single uncontrolled RCCA withdrawal from 60% reactor power with BOC reactivity feedback.

4.0 Uncontrolled RCCA Withdrawal Analysis Results

The FSAR uncontrolled RCCA bank withdrawal events have been reanalyzed by Westinghouse using the RTDP methodology (References 4 and 5). Available margin has been increased as a result of these RTDP analyses. These analyses show that, for the uncontrolled RCCA bank withdrawal event, a minimum of 30% DNBR margin exists for the OFA and 17% DNBR margin exists for the LOPAR fuel assemblies at-power. There are no LOPAR fuel assemblies in the present cycle of Turkey Point Unit 4 and only 5 LOPAR assemblies in the present cycle of Turkey Point Unit 3.

The uncontrolled asymmetric RCCA withdrawal analyses using SIMULATE-3, RETRAN-02 and VIPRE-01 computer codes were performed at 100%, 80%, 60%, 10% and HZP reactor power (References 19 and 20). The results of the uncontrolled asymmetric RCCA withdrawal analyses are summarized in Tables 3, 4 and 5. These results show that a single uncontrolled asymmetric RCCA withdrawal results in the maximum DNBR penalty at the BOC feedback conditions with the reactor at 60% power, where the rod movement has a significant impact on both the core power level and distribution. The results for the case of 60% power for OFA and LOPAR fuels, are provided in Tables 3 and 4, respectively. These analyses have shown that the existing DNB margin in the uncontrolled RCCA bank withdrawal analyses for Turkey Point Units 3 and 4, is sufficient to accommodate the DNBR penalty caused by an uncontrolled asymmetric RCCA withdrawal event without violating the DNBR safety limit.

At HZP, an uncontrolled withdrawal of a bank is more limiting than the withdrawal of a group, double rod or a single rod because it results in the maximum peak in the nuclear power. Results of the limiting case of an uncontrolled bank withdrawal from HZP (Table 6), show that the fuel centerline temperatures stay below the safety limit.

The peak reactor coolant system pressures as listed in Table 3, remain below their safety limit. The peak steam generator secondary pressures, though not listed in Table 3, remain below their safety limit, also.

5.0 Comparison of FPL Analysis Results With the UFSAR for Applicability

Attachment 2 provides a comparison of bank withdrawal results from full power. This transient was analyzed using the RETRAN-02 model of Turkey Point and compared with the FSAR analysis performed by Westinghouse (Reference 16). The analysis results were provided to the NRC for demonstrating FPL capability in the use of RETRAN-02 (Reference 15). This analysis showed good agreement between FPL and Westinghouse analyses in the prediction of reactor power, reactor trip, and RCS temperature and pressure during the transient.

State point comparisons corresponding to the minimum DNBR from a bank withdrawal at 60% power, and corresponding to peak nuclear power from a bank withdrawal at HZP are provided in Table 6. The 60% power case was simulated to compare FPL results from RETRAN-02 and VIPRE-01 with the RTDP analyses performed by Westinghouse. This case assumes a reactivity insertion rate of 1 pcm/sec from a bank withdrawal. There is a good agreement between FPL and Westinghouse predictions of peak reactor power, peak RCS pressure, reactor trip function, reactor trip time, RCS average temperature and minimum DNBR. For this comparison, the moderator temperature coefficient was increased from +5 pcm/sec to +7 pcm/sec to be consistent with the documented Westinghouse analysis.

The HZP case simulated a bank withdrawal with 75 pcm/sec reactivity insertion rate, to be consistent with a documented analysis (Reference 5). The transient is turned around by the Doppler feedback prior to the reactor trip on high neutron flux. The results tabulated in Table 6 show good agreement between peak nuclear power, time of trip, peak heat flux and peak centerline temperature. The system response in this transient is computed using RETRAN-02 and the fuel parameters are computed using VIPRE-01 computer code.

6.0 Validity of analysis for all future cycles

Significant parameters affecting the uncontrolled asymmetric RCCA withdrawal analyses are rod worths, core peaking factors, moderator and Doppler temperature coefficients, and axial offset. These parameters will be evaluated for their impact on the uncontrolled asymmetric RCCA withdrawal analyses during the reload safety evaluation process for future fuel cycles. Evaluation of the uncontrolled asymmetric rod withdrawal events for future fuel cycles will continue unless the plant modifications recommended by the Westinghouse Owners Group are implemented to preclude the potential for the uncontrolled rod withdrawal events of the type experienced at Salem.

7.0 Comparison of FPL and Westinghouse Owners Group Analysis Results for the Uncontrolled Asymmetric RCCA Withdrawal Event.

Turkey Point Units 3 and 4 were analyzed by the Westinghouse Owners Group (WOG) using RTDP methodology as a representative three loop, 15x15 fuel type plant (References 21 and 22). The WOG results show that the uncontrolled asymmetric rod withdrawal causes a maximum of 5.1% DNBR penalty for OFA fuel and 8.1% DNBR

penalty for LOPAR fuel. This compares well with the maximum penalties of 7.53% (OFA) and 8.63% (LOPAR) calculated by FPL. Both FPL and Westinghouse analyses show that the existing DNB margin is sufficient to accommodate uncontrolled asymmetric rod withdrawal at Turkey Point.

The Westinghouse Owners Group report (Reference 21) also states that the results are not significantly dependent on cycle-to-cycle fuel management changes.

8.0 References

1. NRC Generic Letter 93-04,"Rod Control System Failure and Withdrawal of Rod Cluster Control Assemblies, 10 CFR 50.54(f)," dated June 21, 1993.

2. FPL letter to the NRC L-93-186,"NRC Generic Letter 93-04 - Rod Control System Failure and Withdrawal of Rod Cluster Control Assemblies," dated August 4, 1993.

3. NRC letter to FPL,"Turkey Point Units 3 and 4 - Generic Letter 93-04 - 'Rod Control System Failure and Withdrawal of Rod Cluster Control Assemblies' - Request for Additional Information (TAC No.s M86873 and M86874)," dated November 15, 1993.

4. Turkey Point Units 3 and 4,"Non-LOCA Reanalyses for 1990 Fuel Contract," Westinghouse 1993.

5. Turkey Point Units 3 and 4,"Accident Analysis Design Basis Document," Westinghouse 1993.

6. WCAP-11397-P-A, "Revised Thermal Design Procedure," Westinghouse, dated April 1989.

7. SIMULATE-3: Advanced Three-Dimensional Two-Group Reactor Analysis Code, Version 3.03, Studsvik, dated April 6, 1990.

8. TABLES-3: Library Preparation Code for SIMULATE-3, Version 3.03, Studsvik, dated April 6, 1990.

9. CASMO-3: Fuel Assembly Burnup Program, Version 4.4, Studsvik, dated November 26, 1990.

10. Ashok C. Thadani (USNRC) letter to Mr. G. Papanic, Jr., Yankee Atomic Electric Company, Acceptance of Referencing of Topical Report YAEC-1659, "SIMULATE-3, Validation and Verification", dated February 20, 1990.

11. Ashok C. Thadani (USNRC) letter to Mr. G. Papanic, Jr., Yankee Atomic Electric Company, Acceptance for Referencing of Topical Report YAEC-1363, "CASMO-3G Validation".

12. FPL Calculation JPN-PSL-OFJF-92-071,"SIMULATE-3 Validation Analysis," Revision-1, dated March 9, 1993.

13. RETRAN-02: A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-1850-CCMA, Vol. 1, Rev. 2, Computer Code Manual, dated November 1984.

14. Cecil O. Thomas (USNRC) letter to Dr. Thomas W. Schnatz,"Acceptance for Referencing of Licensing Topical Reports EPRI CCM5, RETRAN-A Program for One-Dimensional Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems, and EPRI NP1850-CCM, RETRAN-02-A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," dated September 4, 1984.

15. Gus C. Lainas (USNRC) letter to Mr. W. F. Conway, "Florida Power and Light Company - Topical Report on RETRAN (TAC No. 60550) and Topical Report on PWR Physics Methodology (TAC No. 60549)," dated April 19, 1988.

16. NTH-G-6,"Topical Report, RETRAN Code, Transient Analysis Model Qualification," Florida Power and Light Company, dated July 1985.

17. Charles E. Rossi (USNRC) letter to Mr. J. A. Blaisdell,"Acceptance for Referencing of Licensing Topical Report, EPRI NP-2511-CCM,"VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores", Volumes 1, 2, 3 and 4, dated May 1, 1986.

18. VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, Volumes 1, 2, 3 and 4, EPRI NP-2511-CCM-A, Rev 3, dated August 1989.

19. FPL Calculation JPN-PTN-BFJF-93-041/044,"PTN Physics Data for Rod Withdrawal Analysis at Power and at Hot Zero Power," Rev. 0, dated July 1993.

20. FPL Calculation JPN-PTN-BFJF-93-043,"Impact of Asymmetric RCCA Withdrawal on Available DNB Margin for Turkey Point Units 3 and 4," Rev. 0, dated July 1993.

21. WCAP-13803 Rev. 1,"Generic Assessment of Asymmetric Rod Cluster Control Assembly Withdrawal," Westinghouse, dated August 1993.

22. Westinghouse Owners Group, Analysis Subcommittee letter OG-93-80,"Revisions to the Results of Asymmetric Rod Withdrawal Analysis Program," dated September 10, 1993.

23. Turkey Point Units 3 and 4 Updated FSAR, Revision 11, dated November 1993.

24. Turkey Point Units 3 and 4 Technical Specifications, Operating License Amendments 157 and 151, effective date November 18, 1993.

TABLE 1

Description of SIMULATE-3 Cases Evaluated		
Power Level; Rod Insertion Limit (RIL) Steps	Control Rod Cases	Full Core Coordinates
100%; RIL: D @171 80%; RIL: D @132 60%; RIL: D @ 93	Bank D	Figure 1
	Bank D Group 1 Bank D Group 2	D-8;M-8 H-4;H-8;H-12
	Two Rods	H-8;D-8 D-8;H-4
	Single	H-8 D-8
10% RIL: D @0 C @ 127	Bank D Bank C Banks C & D	Figure 1
	Bank D Group 1(D/1) Bank D Group 2(D/2) Bank C Group 1(C/1) Bank C Group 2(C/2)	D-8;M-8 H-4;H-8;H-12 F-4;D-10;K-12;M-6 K-4;D-6;F-12;M-10
	Group + Single	D/1 + D-6 D/2 + D-6 D/2 + F-4 C/1 + D-8 C/1 + H-8
	Two Rods	D-10;D-6 D-8;D-6 D-6;H-8 F-4;D-6
	Single Rod	D-8 D-6

TABLE 1
(Continued)

Description of SIMULATE-3 Cases Evaluated		
Power Level; Rod Insertion Limit (RIL) Steps	Control Rod Cases	Full Core Coordinates
HZP; ARI	Bank D Bank C Bank B Bank A Bank SA Bank SB	Figure 1
	Bank D Group 1 Bank D Group 2 Bank C Group Bank B Group Bank A Group Bank SA Group Bank SB Group	D-8;M-8 H-4;H-8;H-12 assumed 1/4 core symmetry and only evaluated one group for Banks A-C and SA, SB
	Two Rods	F-2;G-3 H-4;G-3 F-4;G-3 J-3;G-3 J-5;J-3 H-4;J-3 H-4;K-4 J-3;K-4
	Single Rod	H-4 H-8 F-2 G-3 D-6 J-3 J-5

TABLE 2

Summary of SIMULATE-3 Results			
Power	Case	Peak $F_{\Delta H}$	Avg pcm/sec
100 %	Bank Group 2 Rods Single	1.70 1.73 1.76 1.76	5.3 2.7 2.7 1.5
80 %	Bank Group 2 Rods Single	1.79 1.88 1.96 1.98	6.6 2.8 2.6 1.4
60 %	Bank Group 2 Rods Single	1.82 1.88 1.93 2.03	6.2 2.7 2.6 1.4
10 %	Bank Group 2 Rods Single	2.02 2.16 2.39 2.42	11.3 1.7 1.7 1.5
HZP	Bank Group 2 Rods Single	2.70 2.99 4.10 4.76	75 8.7 10.2 5.0

** The average pcm/sec was calculated from the total reactivity insertion and the time required to withdraw the rods from the rod insertion limits to ARO using 72 steps/min. In RETRAN-02, however, the differential rod worths vs. time were actually used.

TABLE 3

RESULTS OF RETRAN-02 AND VIPRE-01 ANALYSES
FOR VARIOUS POWER LEVELS AND RCCA WITHDRAWAL EVENTS

CASE DESCRIPTION	PEAK POWER (MW_{th})	PEAK RCS PRESSURE (PSIA)	MDNBR OFA (WRB-1)	MDNBR LOPAR (W-3L)	DNB PENALTY (1) OFA (% DNBR)	DNB PENALTY (1) LOPAR (% DNBR)	NEW MARGIN OFA (% DNBR)	NEW MARGIN LOPAR (% DNBR)
100 % Power :								
Min. Feedback -								
Bank	2500.6	2306.9	2.06	1.72	---	---	---	---
Group	2436.9	2308.9	2.09	1.76	-1.56	-2.27	30.63	19.67
Two Rods	-----	-----	2.04	1.75	0.68	-1.63	28.39	19.03
Single Rod	2398.3	2309.4	2.08	1.76	-1.12	-2.56	30.19	19.96
Max. Feedback(3) -								
Bank	2397.0	2309.3	---	---	---	---	---	---
Group	2305.5	2278.0	---	---	---	---	---	---
Two Rods	-----	-----	---	---	---	---	---	---
Single Rod	2255.7	2252.0	---	---	---	---	---	---
80 % Power :								
Min. Feedback -								
Bank	2461.8	2335.4	1.95	1.45	---	---	---	---
Group	2321.1	2326.8	1.92	1.48	1.49	-2.28	27.58	19.68
Two Rods	-----	-----	1.80	1.44	7.44	0.21	21.63	17.19
Single Rod	2253.5	2343.6	1.84	1.47	5.39	-1.80	23.68	19.20
60 % Power :								
Min. Feedback -								
Bank	2312.4	2352.3	1.98	1.80	---	---	---	---
Group	2174.7	2338.9	1.95	1.75	1.21	2.39	27.86	15.01
Two Rods	-----	-----	1.89	1.70	4.60	5.46	24.47	11.94
Single Rod	2076.0	2341.2	1.83	1.64	7.53	8.63	21.54	8.77
10 % Power :								
Min. Feedback -								
Bank	2215.1	2352.5	1.72	1.56	---	---	---	---
Group	1571.5	2347.0	1.93	1.68	-12.40	-7.56	41.47	24.96
Two Rods	1693.3	2334.8	1.61	1.43	6.46	8.59	22.61	8.81
Single Rod	1509.1	2338.9	1.75	1.53	-1.92	1.99	30.99	15.41
HZP Power :								
Min. Feedback -								
Bank	0.66(2)	2229.1	1.45	1.36	---	---	---	---
Group	0.21(2)	2258.3	5.88	5.50	---	---	---	---
Two Rods	0.43(2)	2247.2	4.13	3.78	---	---	---	---
Single Rod	0.21(2)	2268.3	3.57	3.20	---	---	---	---

NOTES :

1. For all cases initiated from hot zero power (HZP), the MDNBR calculated for the bank withdrawal is limiting. Therefore, asymmetrical withdrawal cases from hot zero power are bounded by the reference bank withdrawal from hot zero power for DNB considerations.
2. Maximum core average heat flux as a fraction of nominal.
3. Since minimum feedback resulted in higher power peaks at 100% power, all other cases were analyzed with minimum feedback only.

TABLE 4

ROD WITHDRAWAL FROM 60% POWER (BOC)
OFA FUEL

PARAMETER	BANK W/DRAW	GROUP W/DRAW	DOUBLE ROD W/DRAW	SINGLE ROD W/DRAW
REACTIVITY INSERTION RATE, AVERAGE (pcm/sec)	6.2	2.7	2.6	1.4
TIME OF REACTOR TRIP (sec)	41.5	74.1	74.1	106.3
TRIP SIGNAL	OTΔT	OTΔT	OTΔT	OTΔT *
FΔH REACTOR TRIP	1.82	1.88	1.93	2.03
MINIMUM DNBR	1.98	1.95	1.89	1.83
DNBR PENALTY RELATIVE TO BANK WITHDRAWAL (%)	0.	1.21	4.60	7.53
REMAINING MARGIN TO DNBR LIMIT (%)	29.	27.79	24.40	21.47

NOTES * OTΔT = Over Temperature delta T Trip

TABLE 5

ROD WITHDRAWAL FROM 60% POWER (BOC)
LOPAR FUEL

PARAMETER	BANK W/DRAW	GROUP W/DRAW	DOUBLE ROD W/DRAW	SINGLE ROD W/DRAW
REACTIVITY INSERTION RATE, AVERAGE (pcm/sec)	6.2	2.7	2.6	1.4
TIME OF REACTOR TRIP (sec)	41.5	74.1	74.1	106.3
TRIP SIGNAL	OTΔT	OTΔT	OTΔT	OTΔT
FΔH REACTOR TRIP	1.82	1.88	1.93	2.03
MINIMUM DNBR	1.80	1.75	1.70	1.64
DNBR PENALTY RELATIVE TO BANK WITHDRAWAL (%)	0.	2.39	5.46	8.63
REMAINING MARGIN TO DNBR LIMIT (%)	17.	14.61	11.54	8.37 *

Notes: * Additional margin is present in the analyses because the latest RTDP analyses (relative to which the above margins are quoted) assumed the following conservatism:

1) 20% steam generator tube plugging (presently less than 5% tubes are plugged),

2) the LOPAR fuel assemblies are at least once-burned and thus run well below the allowable FΔH limit.

There are no LOPAR fuel assemblies in Unit 4 and only 5 assemblies in Unit 3 in the present cores.

TABLE 6

FPL VERSUS WESTINGHOUSE (RTDP) BANK WITHDRAWAL COMPARISON

RCCA BANK WITHDRAWAL FROM 60% POWER (1 PCM/SEC)

PARAMETER	WESTINGHOUSE	FPL
Peak Power (% Nominal)	100%	93%
Peak RCS Pressure	2350 psia	2318 psia
Core Tave @ Reactor Trip	605 °F	594 °F
Trip Function	OTΔT	OTΔT
Time of Trip	104.1 sec	100.7 sec
Minimum DNBR (W-3L)	1.84	1.97

RCCA BANK WITHDRAWAL FROM HOT ZERO POWER (75 PCM/SEC)

PARAMETER	WESTINGHOUSE	FPL
Peak Nuclear Power (% Nominal)	200%	263%
Peak Heat Flux (% Nominal)	55%	74%
Trip Function	Hi Flux (35%)	Hi Flux (35%)
Time of Trip	10.3 sec	11.4 sec
Peak Center-line Temperature	2538 °F	2486 °F
Peak Fuel Average Temperature	2148 °F	2106 °F
Peak Clad Temperature	725 °F	757 °F

FIGURE 1

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2						B				B					
3							S_A		S_A						
4						C		D		C					
5					S_B		A		A		S_B				
6		B		C				S_B				C		B	
7			S_A		A						A		S_A		
8				D		S_B		D		S_B		D			
9			S_A		A						A		S_A		
10		B		C				S_B				C		B	
11					S_B		A		A		S_B				
12						C		D		C					
13							S_A		S_A						
14						B				B					
15															

ABSORBER MATERIAL:
Ag-In-Cd

CONTROL ROD DESIGNATION

FUNCTION	NUMBER OF CLUSTERS
CONTROL BANK D	5
CONTROL BANK C	8
CONTROL BANK B	8
CONTROL BANK A	8
SHUTDOWN BANK S_B	8
SHUTDOWN BANK S_A	8

CONTROL AND SHUTDOWN ROD LOCATIONS

Validation of FPL SIMULATE-3 Core Physics Models

Reference: JPN Calculation PSL-0FJF-92-071, Revision 0, "SIMULATE-3 Validation Analysis," Approved 3/9/93

The SIMULATE-3 core physics models have been validated against measured data. Included here is part of the validation performed consisting of axial power shapes, control rod worth comparisons, and radial power distributions for various units and cycles. The good comparisons between measured and calculated radial power distributions, axial power shapes, and control rod worths provides the justification for the use of SIMULATE-3 to calculate control rod insertion rates and peaking factors used in the uncontrolled rod withdrawal analyses.

The following acceptance criteria were used during these benchmarks. These criteria were obtained from ANSI 19.6, Technical Specifications, Operating Procedures and Industry experience.

Radial power distribution	±0.100 for each measured assembly power & RMS ≤5%
Axial power distribution	±0.03 axial offset units
Control rod worths	Individual banks ±15% or ±100 pcm whichever is greater

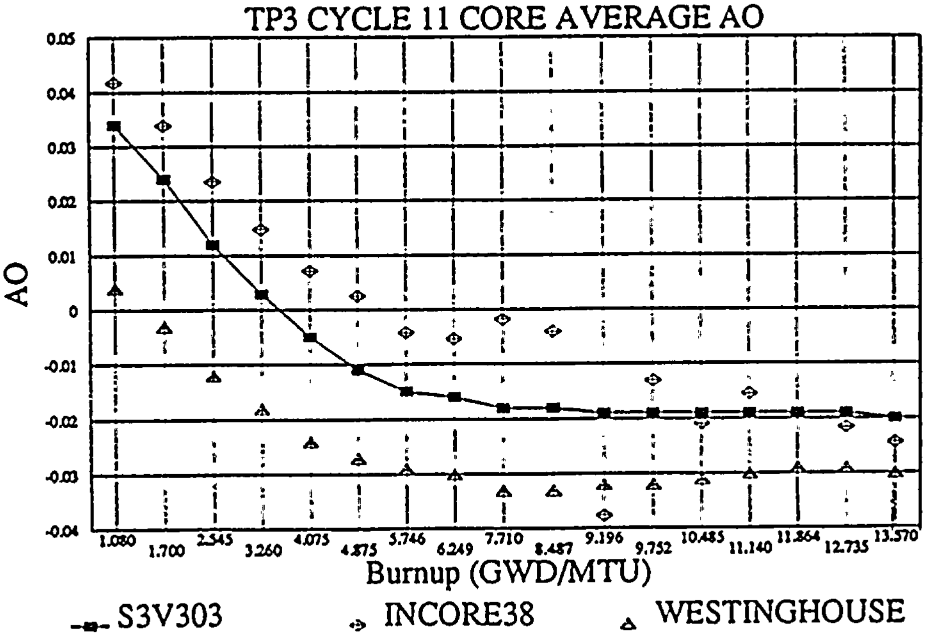
Summary of Attached Comparisons

	Description	Page
Axial Power Shape:	Turkey Point 3 cycle 11	2
	Turkey Point 3 cycle 12	3
	Turkey Point 4 Cycle 13	4
Control Rod Worth:	Turkey Point 3 cycle 10	5
	Turkey Point 3 cycle 11	
	Turkey Point 3 Cycle 12	
	Turkey Point 4 cycle 11	
	Turkey Point 4 cycle 12	
	Turkey Point 4 Cycle 13	
Radial Power Distribution:	Turkey Point 3 BOC 12	6
	Turkey Point 3 MOC 12	7
	Turkey Point 3 EOC 12	8
	Turkey Point 4 MOC 12	9
	Turkey Point 4 EOC 12	10
	Turkey Point 4 BOC 13	11
	Turkey Point 4 MOC 13	12

Table 5.3e
Axial Shape Index Comparisons
Turkey Point Unit 3/Cycle 11

BURNUP (GWD/MTU)	[1] SIM-3 AO	MEAS AO	[2] FLUXMAP ID	FLUXMAP DATE	DIFF. (S3-M)	[3] VENDOR AO	DIFF (S3-VEND)
1.080	0.033	0.042	FM3XI6	03/15/88	-0.009	0.004	0.029
1.700	0.023	0.034	FM3XI8	04/19/88	-0.011	-0.003	0.026
2.545	0.011	0.024	FM3XI9	05/17/88	-0.013	-0.012	0.023
3.260	0.003	0.015	FM3XI10	06/09/88	-0.012	-0.018	0.021
4.075	-0.005	0.007	FM3XI11	07/07/88	-0.012	-0.024	0.019
4.875	-0.011	0.003	FM3XI12	08/03/88	-0.014	-0.027	0.016
5.746	-0.015	-0.004	FM3XI13B	09/06/88	-0.011	-0.029	0.014
6.249	-0.014	-0.005	FM3XI14	09/16/88	-0.009	-0.030	0.016
7.710	-0.018	-0.002	FM3XI16	03/16/89	-0.016	-0.033	0.015
8.487	-0.018	-0.004	FM3XI17	07/10/89	-0.014	-0.033	0.015
9.196	-0.019	-0.038	FM3XI18	08/03/89	0.019	-0.032	0.013
9.752	-0.019	-0.013	FM3XI19R	08/23/89	-0.006	-0.032	0.013
10.485	-0.019	-0.021	FM3XI20	09/19/89	0.002	-0.031	0.012
11.140	-0.019	-0.016	FM3XI21	10/10/89	-0.004	-0.030	0.011
11.864	-0.019	-0.019	FM3XI22	11/06/89	0.000	-0.029	0.010
12.735	-0.019	-0.022	FM3XI23	12/05/89	0.003	-0.029	0.010
13.570	-0.019	-0.024	FM3XI24	01/03/90	0.005	-0.030	0.011
			AVERAGE DIFFERENCE:		-0.015		0.016
			STANDARD DEVIATION:		0.0135		0.0055

Notes:
1. SIM-3 reference j9387, 1/14/93.
2. Measured data from INCORE-3D, ver.3.8 fluxmaps.
3. Vendor data from Westinghouse WCAP-11454, 4/87, PCNDR.



(SIMULATE-3)

Table 5.3f
Axial Shape Index Comparisons
Turkey Point Unit 3/Cycle 12

BURNUP (GWD/MTU)	[1] SIM-3 AO	MEAS AO	[2] FLUXMAP ID	FLUXMAP DATE	DIFF. (S3-M)	[3] VENDOR AO	DIFF (S3-VEND)
782	0.012	0.010	FM3XII04	07/11/90	0.0024	0.006	0.0056
1552	0.009	0.001	FM3XII05	08/06/90	0.0080	-0.001	0.0103
3,140	-0.002	0.013	FM3XII07	09/28/90	-0.0146	-0.013	0.0113
4,740	-0.010	0.005	FM3XII09	11/20/90	-0.0150	-0.021	0.0105
5,646	-0.014	0.001	FM3XII12	10/17/91	-0.0154	-0.021	0.0072
7,222	-0.016	-0.004	FM3XII14	12/09/91	-0.0118	-0.022	0.0061
8,945	-0.017	-0.010	FM3XII16	02/05/92	-0.0073	-0.022	0.0047
9,743	-0.015	-0.016	FM3XII17	03/06/92	0.0005	-0.021	0.0062
11,026	-0.015	-0.012	FM3XII19	04/23/92	-0.0032	-0.021	0.0058
12,222	-0.016	-0.009	FM3XII21	06/22/92	-0.0072	-0.021	0.0053
13,050	-0.016	-0.004	FM3XII22	07/23/92	-0.0123	-0.021	0.0055
			AVERAGE DIFFERENCE:		-0.0069		0.0071
			STANDARD DEVIATION:		0.0076		0.0023

Notes:
1. SIM-3 data from j9244, 10/29/92.
2. Measured data for fluxmaps prior to FM3XII19 are from INCORE-3D, ver. 3.8 calculations. Data for FM3XII19 and later were calculated using INCORE-3D, ver. 7.2.
3. Vendor data from Westinghouse WCAP-12538, 4/90, Fig. 3-16.

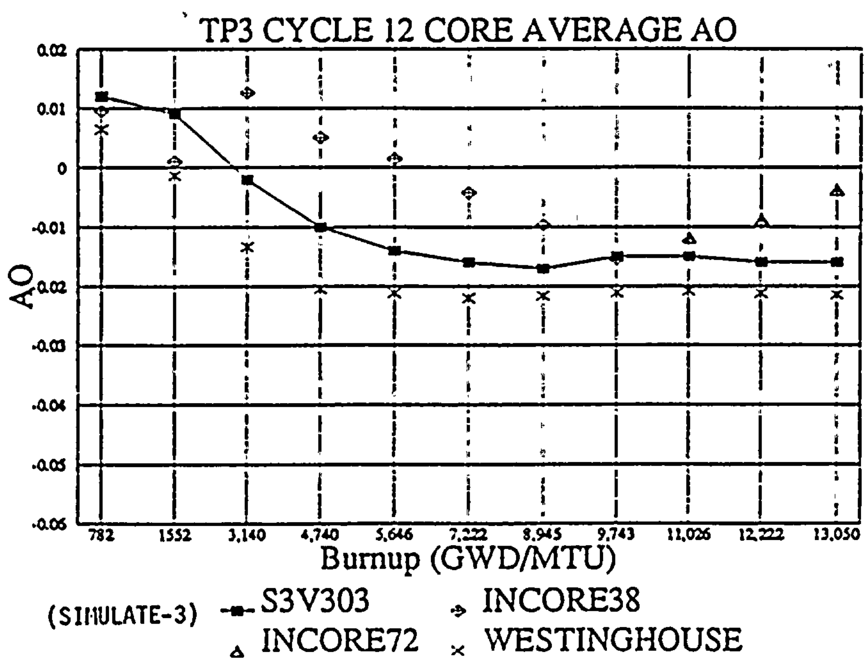


Table 5.3g
Axial Shape Index Comparisons
Turkey Point Unit 4/Cycle 13

BURNUP (GWD/MTU)	[1] SIM-3 ASI	MEAS ASI	[2] FLUXMAP ID	FLUXMAP DATE	DIFF. (S3-M)	[3] VENDOR ASI	DIFF (S3-VEND)
1536	0.022	0.024	FM4135	12/31/91	-0.0018	0.017	0.0048
3224	0.003	0.009	FM4137	03/04/92	-0.0061	-0.006	0.0092
4031	-0.003	0.001	FM4138	03/30/92	-0.0041	-0.012	0.0093
4888	-0.009	-0.004	FM4139	04/27/92	-0.0045	-0.0205	0.0119
5788	-0.013	-0.004	FM41310	05/26/92	-0.0094	-0.0208	0.0078
6678	-0.016	-0.007	FM41311	06/25/92	-0.0094	-0.023	0.0070
7,461	-0.018	-0.012	FM41312	07/21/92	-0.0062	-0.0245	0.0062
8,265	-0.019	-0.015	FM41313	08/19/92	-0.0042	-0.0251	0.0058
			AVERAGE DIFFERENCE:		-0.0057		0.0078
			STANDARD DEVIATION:		0.0025		0.0022

Notes:
1. SIM-3 reference j8357, 9/28/92.
2. Vendor data from Westinghouse WCAP-13021, 8/91, Fig. 3-18.

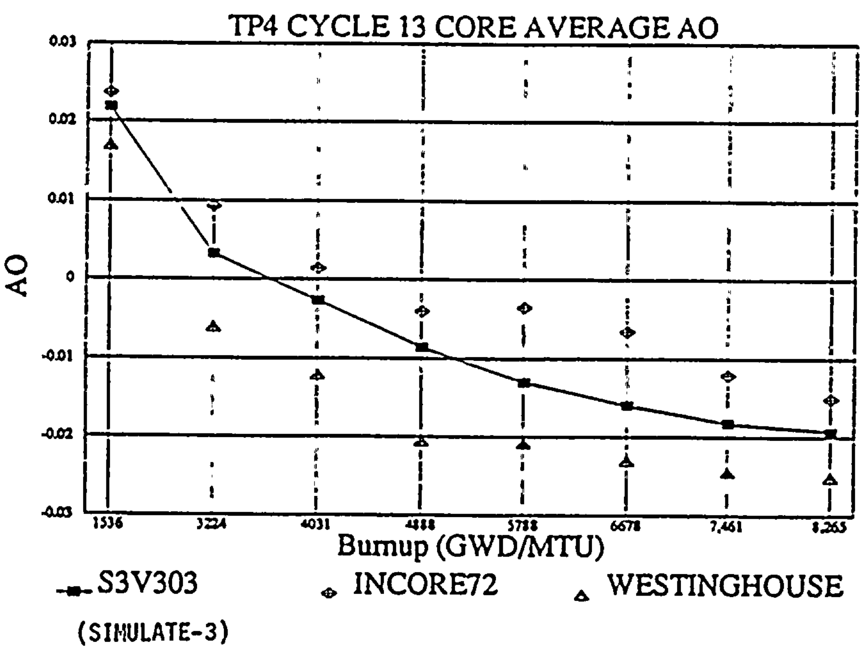


Table 5.4
Control Rod Worth Comparisons

TURKEY POINT 3/CYCLE 10

BANK ID	ADJ. MEAS (PCM)	SIM-3 (PCM)	DIFFERENCE (PCM)	DIFFERENCE (%)
D	654	663	9	1.44%
C	1414	1419	5	0.36%
B	515	553	38	7.32%
A	1170	1171	1	0.09%
SB	1091	1057	-34	-3.09%
SA	1083	1125	42	3.84%
TOTAL	5927	5988	61	1.03%
AVE			10	1.66%
STD DEV			25	3.26%

Ref: L-85-388, 10/14/85
SIM-3 j3111, 10/02/92
BETA ADJ.= 1.056

TURKEY POINT 4/CYCLE 11

BANK ID	ADJ. MEAS (PCM)	SIM-3 (PCM)	DIFFERENCE (PCM)	DIFFERENCE (%)
D	771	753	-18	-2.31%
C	1532	1456	-76	-4.97%
B	607	631	24	3.94%
A	1161	1084	-77	-6.64%
SB	1164	1110	-54	-4.66%
SA	1249	1154	-95	-7.63%
TOTAL	6485	6188	-297	-4.57%
AVE			-49	-3.71%
STD DEV			41	3.80%

Ref: L-86-472, 11/17/86
SIM-3 j1962, 1/14/93
BETA ADJ.= 1.063

TURKEY POINT 3/CYCLE 11

BANK ID	ADJ. MEAS (PCM)	SIM-3 (PCM)	DIFFERENCE (PCM)	DIFFERENCE (%)
D	694	635	-59	-8.54%
C	1349	1310	-39	-2.88%
B	632	661	29	4.56%
A	1109	969	-140	-12.59%
SB	1127	1017	-110	-9.75%
SA	1070	1056	-14	-1.30%
TOTAL	5981	5648	-333	-5.56%
AVE			-55	-5.08%
STD DEV			57	5.81%

Ref: L-88-190, 5/09/88
SIM-3 j9420, 1/14/93
BETA ADJ.= 1.018

TURKEY POINT 4/CYCLE 12

BANK ID	ADJ. MEAS (PCM)	SIM-3 (PCM)	DIFFERENCE (PCM)	DIFFERENCE (%)
D	708	712	4	0.52%
C	1347	1339	-8	-0.59%
B	384	413	29	7.44%
A	1206	1157	-49	-4.10%
SB	1210	1198	-12	-0.95%
SA	1025	1043	18	1.75%
TOTAL	5881	5862	-19	-0.32%
AVE			-3	0.68%
STD DEV			25	3.51%

Ref: L-89-309, 8/21/89
SIM-3 j2233, 1/14/93
BETA ADJ.= 1.025

TURKEY POINT 3/CYCLE 12

BANK ID	ADJ. MEAS (PCM)	SIM-3 (PCM)	DIFFERENCE (PCM)	DIFFERENCE (%)
D	856	821	-35	-4.11%
C	1389	1435	46	3.30%
B	457	479	22	4.82%
A	1143	1153	10	0.88%
SB	1038	1084	46	4.47%
SA	1217	1206	-11	-0.90%
TOTAL	6100	6178	78	1.28%
AVE			13	1.41%
STD DEV			29	3.18%

Ref: L-90-299, 8/21/90
SIM-3 j3721, 10/02/92
BETA ADJ.= 1.013

TURKEY POINT 4/CYCLE 13

BANK ID	ADJ. MEAS (PCM)	SIM-3 (PCM)	DIFFERENCE (PCM)	DIFFERENCE (%)
D	661	664	3	0.52%
C	1052	1046	-6	-0.58%
B	448	466	18	3.95%
A	1270	1236	-34	-2.65%
SB	1219	1175	-44	-3.62%
SA	851	871	20	2.33%
TOTAL	5501	5458	-43	-0.78%
AVE			-7	-0.01%
STD DEV			24	2.64%

Ref: L-92-006, 1/23/92
SIM-3 j5564, 9/23/92
BETA ADJ.= 1.031

TURKEY POINT UNIT 3, CYCLE 12

RELATIVE POWER DENSITY

1.552 GWD/MT (1225 EFPH)

	1	2	3	4	5	6	7	8
	1.024	1.303	1.281	1.003	1.315	1.309	0.773	0.245
1	1.029	1.308	1.274	0.990	1.287	1.277	0.780	0.258
	-0.005	-0.005	0.007	0.013	0.028	0.032	-0.007	-0.013
	1.302	1.102	1.223	1.352	1.241	1.279	1.062	0.228
2	1.309	1.120	1.213	1.315	1.211	1.247	1.062	0.242
	-0.007	-0.018	0.010	0.037	0.030	0.032	0.000	-0.014
	1.281	1.221	1.358	1.214	1.225	1.251	0.769	
3	1.293	1.239	1.348	1.196	1.200	1.210	0.792	
	-0.012	-0.018	0.010	0.018	0.025	0.041	-0.023	
	0.998	1.343	1.212	1.029	1.276	0.981	0.341	
4	1.021	1.368	1.221	1.042	1.246	0.969	0.341	
	-0.023	-0.025	-0.009	-0.013	0.030	0.012	0.000	
	1.303	1.223	1.218	1.267	0.837	0.380		
5	1.322	1.262	1.244	1.264	0.839	0.377		
	-0.019	-0.039	-0.026	0.003	-0.002	0.003		
	1.302	1.270	1.242	0.971	0.373			
6	1.321	1.297	1.252	0.973	0.381			
	-0.019	-0.027	-0.010	-0.002	-0.008			
	0.765	1.055	0.764	0.339				
7	0.780	1.068	0.775	0.341				
	-0.015	-0.013	-0.011	-0.002				
	0.242	0.226						
8	0.250	0.236						
	-0.008	-0.010						

MIN. = -0.039
MAX. = 0.041
R.M.S. = 0.020

X.XXX	SIMV303 928 PPM
X.XXX	FM3XII5 0 PPM
X.XXX	DIFFERENCE

CM2.SVVR.TP312.J2222.OUTPUT
JOB = UFRXTJCS - J02222 - 24 SEP 92

TURKEY POINT UNIT 3, CYCLE 12

RELATIVE POWER DENSITY

7.222 GWD/MT (5700 EFPH)

	1	2	3	4	5	6	7	8
1	1.062 1.070 -0.008	1.392 1.410 -0.018	1.215 1.243 -0.028	0.963 0.976 -0.013	1.212 1.224 -0.012	1.327 1.335 -0.008	0.766 0.761 0.005	0.260 0.255 0.005
2	1.393 1.403 -0.010	1.084 1.113 -0.029	1.165 1.176 -0.011	1.373 1.337 0.036	1.153 1.151 0.002	1.198 1.201 -0.003	1.048 1.034 0.014	0.242 0.235 0.007
3	1.219 1.243 -0.024	1.165 1.188 -0.023	1.373 1.351 0.022	1.148 1.139 0.009	1.161 1.151 0.010	1.290 1.265 0.025	0.783 0.762 0.021	
4	0.964 0.980 -0.016	1.371 1.373 -0.002	1.148 1.148 0.000	0.993 0.981 0.012	1.299 1.257 0.042	1.117 1.068 0.049	0.387 0.370 0.017	
5	1.211 1.231 -0.020	1.145 1.169 -0.024	1.159 1.164 -0.005	1.294 1.269 0.025	0.885 0.865 0.020	0.433 0.421 0.012		
6	1.330 1.325 0.005	1.199 1.210 -0.011	1.289 1.272 0.017	1.112 1.089 0.023	0.428 0.425 0.003			
7	0.765 0.771 -0.006	1.049 1.054 -0.005	0.782 0.770 0.012	0.387 0.381 0.006				
8	0.260 0.266 -0.006	0.242 0.248 -0.006						

MIN. = -0.029
MAX. = 0.049
R.M.S. = 0.019

X.XXX	SIMV303 553 PPM
X.XXX	TP3XII14 0 PPM
X.XXX	DIFFERENCE

CM2.SVVR.TP312.J2222.OUTPUT
JOB = UFRXTJCS - J02222 - 24 SEP 92

TURKEY POINT UNIT 3, CYCLE 12

RELATIVE POWER DENSITY

12.222 GWD/MT (9647 EFPH)

	1	2	3	4	5	6	7	8
1	1.032 1.007 0.025	1.333 1.356 -0.023	1.170 1.185 -0.015	0.953 0.951 0.002	1.176 1.120 0.056	1.302 1.329 -0.027	0.794 0.797 -0.003	0.294 0.293 0.001
2	1.334 1.343 -0.009	1.052 1.083 -0.031	1.125 1.143 -0.018	1.326 1.327 -0.001	1.124 1.120 0.004	1.176 1.184 -0.008	1.065 1.082 -0.017	0.272 0.267 0.005
3	1.173 1.174 -0.001	1.125 1.140 -0.015	1.336 1.335 0.001	1.123 1.121 0.002	1.142 1.126 0.016	1.291 1.298 -0.007	0.819 0.816 0.003	
4	0.955 0.935 0.020	1.326 1.317 0.009	1.124 1.132 -0.008	0.999 1.004 -0.005	1.300 1.309 -0.009	1.169 1.163 0.006	0.434 0.426 0.008	
5	1.176 1.122 0.054	1.118 1.086 0.032	1.142 1.150 -0.008	1.297 1.342 -0.045	0.937 0.942 -0.005	0.486 0.482 0.004		
6	1.306 1.303 0.003	1.178 1.182 -0.004	1.292 1.317 -0.025	1.166 1.187 -0.021	0.482 0.475 0.007			
7	0.794 0.783 0.011	1.066 1.086 -0.020	0.820 0.821 -0.001	0.434 0.424 0.010				
8	0.295 0.292 0.003	0.273 0.264 0.009						

MIN. = -0.045
MAX. = 0.056
R.M.S. = 0.017

X.XXX	SIMV303 88 PPM
X.XXX	FM3XII21 0 PPM
X.XXX	DIFFERENCE

CM2.SVVR.TP312.J2222.OUTPUT
JOB = UFRXTJCS - J02222 - 24 SEP 92

TURKEY POINT UNIT 4, CYCLE 12

RELATIVE POWER DENSITY

7.620 GWD/MT (5999 EFPH)

1.117				
1.152				
-0.035				
1.071	1.364			
1.095	1.393			
-0.024	-0.029			
1.168	1.165	1.059		
1.181	1.182	1.074		
-0.013	-0.017	-0.015		
1.165	1.118	1.376	1.125	
1.171	1.116	1.357	1.122	
-0.006	0.002	0.019	0.003	
1.044	1.373	1.135	1.170	1.156
1.072	1.372	1.136	1.164	1.130
-0.028	0.001	-0.001	0.006	0.026
1.338	1.080	1.291	1.135	0.471
1.358	1.111	1.272	1.109	0.453
-0.020	-0.031	0.019	0.026	0.018
0.820	1.048	0.679	0.388	
0.849	1.051	0.660	0.373	
-0.029	-0.003	0.019	0.015	
0.296	0.258			
0.301	0.249			
-0.005	0.009			

MIN. = -0.035
MAX. = 0.026
R.M.S. = 0.018

X.XXX	SIM3V303 587 PPM
X.XXX	FM4XII12 0 PPM
X.XXX	DIFFERENCE

JOB = UFRXTJCL - J08126 - 15 JAN 93

TURKEY POINT UNIT 4, CYCLE 12

RELATIVE POWER DENSITY

11.812 GWD/MT (9299 EFPH)

1.108 1.147 -0.039				
1.065 1.091 -0.026	1.356 1.390 -0.034			
1.149 1.158 -0.009	1.148 1.174 -0.026	1.056 1.075 -0.019		
1.148 1.160 -0.012	1.106 1.122 -0.016	1.372 1.359 0.013	1.113 1.104 0.009	
1.040 1.073 -0.033	1.367 1.361 0.006	1.121 1.113 0.008	1.155 1.145 0.010	1.164 1.146 0.018
1.315 1.342 -0.027	1.070 1.055 0.015	1.274 1.252 0.022	1.143 1.119 0.024	0.499 0.475 0.024
0.836 0.853 -0.017	1.054 1.044 0.010	0.701 0.674 0.027	0.415 0.394 0.021	
0.323 0.327 -0.004	0.281 0.269 0.012			

MIN. = -0.039
MAX. = 0.027
R.M.S. = 0.019

X.XXX	SIM3V303 198 PPM
X.XXX	FM4XII18 0 PPM
X.XXX	DIFFERENCE

JOB = UFRXTJCL - J08126 - 15 JAN 93

TURKEY POINT UNIT 4, CYCLE 13

RELATIVE POWER DENSITY

0.919 GWD/MT (724 EFPH)

	1	2	3	4	5
1	1.044 1.044 0.000				
2	1.368 1.330 0.038	1.082 1.063 0.019			
3	1.089 1.089 0.000	1.363 1.343 0.020	1.290 1.283 0.007		
4	1.308 1.278 0.030	1.257 1.257 0.000	1.246 1.255 -0.009	1.304 1.320 -0.016	
5	0.979 1.017 -0.038	1.281 1.283 -0.002	0.973 1.009 -0.036	1.110 1.129 -0.019	1.053 1.039 0.014
6	1.315 1.280 0.035	0.968 0.991 -0.023	1.250 1.230 0.020	1.087 1.023 0.064	0.435 0.431 0.004
7	0.907 0.917 -0.010	1.120 1.120 0.000	0.788 0.770 0.018	0.397 0.393 0.004	
8	0.263 0.273 -0.010	0.236 0.251 -0.015			

MIN. = -0.038
MAX. = 0.064
R.M.S. = 0.023

X.XXX	S3V303 1055 PPM
X.XXX	TP4XIII4 0 PPM
X.XXX	DIFFERENCE

CM2.SVVR.TP413.J9343.OUTPUT
JOB = UFRNRJR4 - J09343 - 24 SEP 92

TURKEY POINT UNIT 4, CYCLE 13

RELATIVE POWER DENSITY

7.461 GWD/MT (5875 EFPH)

	1	2	3	4	5
	1.018				
1	1.023				
	-0.005				
	1.364	1.016			
2	1.346	1.000			
	0.018	0.016			
	1.037	1.238	1.148		
3	1.033	1.218	1.138		
	0.004	0.020	0.010		
	1.387	1.196	1.143	1.207	
4	1.392	1.197	1.135	1.179	
	-0.005	-0.001	0.008	0.028	
	1.016	1.361	0.970	1.098	1.173
5	1.059	1.399	0.998	1.081	1.147
	-0.043	-0.038	-0.028	0.017	0.026
	1.343	0.984	1.304	1.151	0.494
6	1.387	1.015	1.316	1.138	0.494
	-0.044	-0.031	-0.012	0.013	0.000
	0.905	1.109	0.813	0.440	
7	0.902	1.124	0.833	0.456	
	0.003	-0.015	-0.020	-0.016	
	0.285	0.255			
8	0.289	0.267			
	-0.004	-0.012			

MIN. = -0.044
MAX. = 0.028
R.M.S. = 0.021

X.XXX	S3V303 534 PPM
X.XXX	TP4XIII12 0 PPM
X.XXX	DIFFERENCE

CM2.SVVR.TP413.J9343.OUTPUT
JOB = UFRNRJR4 - J09343 - 24 SEP 92

6.0 REACTIVITY INSERTION

Events in this category involve localized reactivity additions which cause anomalies in the core power distribution. Important modeling considerations are the reactor protection system, reactor kinetics and reactivity feedback coefficients. Analyses presented in this category are the Turkey Point Uncontrolled RCCA Withdrawal transient benchmarked to FSAR results (Section 6.1), and the St. Lucie Unit 2 CEA Drop transient benchmarked to FSAR results (Section 6.2).

6.1 Turkey Point Uncontrolled RCCA Withdrawal

6.1.1 Transient Description

A slow, uncontrolled rod cluster control assembly (RCCA) withdrawal transient from 100% power was simulated with the RETRAN02 computer code and benchmarked to the analogous transient documented in the Turkey Point FSAR. (Ref. 13). In this transient the rod withdrawal causes an increase in core power and heat flux which result in increases in RCS temperature and pressure. Reactor trip can occur on high RCS pressure, high pressurizer level or on exceeding the high power, overpower ΔT or overtemperature ΔT setpoints. This transient assesses, the adequacy of the RETRAN reactor kinetics modeling and the modeling of the reactor protection system.

6.1.2 RETRAN Analysis Description

The initial conditions of the benchmark and RETRAN02 analysis, presented in Table 6.1.1, were incorporated into the Turkey Point RETRAN base model (see Appendix B). These initial conditions represent beginning of cycle conditions as listed in the Turkey Point FSAR. The analysis was performed for a rod withdrawel rate of $2.5\times10^{-5}\Delta k$/sec. For this case the reactor trips on overtemperature ΔT. Presented in Table 6.1.2 is the status of safety systems included in the RETRAN simulation of this transient.

6.1.3 Results

Results of the RETRAN02 calculation and the FSAR are presented in Figures 6.1.1, 6.1.2 and 6.1.3. A sequence of events for both the RETRAN calculation and the FSAR results is shown in Table 6.1.3. As the core power increases the sensed temperature difference between the hot leg and cold leg reaches the dynamic overtemperature ΔT setpoint, when the scram signal is generated and the reactor trips. The turbine trips on the reactor trip. RETRAN predicts the reactor trip about 3 seconds later than the FSAR analysis but maximum core power, maximum RCS pressure and temperature calculated for the RETRAN and FSAR analyses are essentially identical. Overall, the RETRAN simulation shows good agreement with the FSAR analysis results.

TABLE 6.1.1

INITIAL CONDITIONS AND KEY PARAMETERS

UNCONTROLLED RCCA WITHDRAWAL

PARAMETER	VALUE
Core Power, MW (Thermal)	2244
Core Inlet Coolant Temperature, °F	550.2
Core Mass Flow Rate, 10^6 lbm/hr	101.5
Pressurizer Pressure, psia	2220
Doppler Coefficient, $10^{-4} \Delta k/°F$	-.12
Moderator Temperature Coefficient, $10^{-4} \Delta k/°F$	-.4
Over-Temperature ΔT Above Nominal ΔT Trip setpoint (%)	4
Rod Withdrawal Rate Δk/sec	2.5×10^{-5}

TABLE 6.1.2

SAFETY SYSTEMS STATUS ASSUMED IN MODEL

UNCONTROLLED RCCA WITHDRAWAL

SYSTEM	ACTUATED	AVAILABLE BUT NOT ACTUATED	NOT SIMULATED
Reactor Protection System (SCRAM)	X		
Pressurizer Pressure Control System	X		
Main Steam Safety Valves		X	
Pressurizer Safety Valves		X	
Main Steam Isolation Valves		X	
Main Feedwater Isolation Valves	X		
Auxiliary Feedwater System		X	
Safety Injection System			
HPSI		X	
Accumulators			X
LPSI		X	
Atmospheric Dump Valve Systems			X
Steam Dump and Bypass System			X
Pressurizer Level Control System			X
Pressurizer Power Operated Relief Valves (PORV)		X	
Chemical and Volume Control System			X
S. G. Level Control System		X	
Automatic Rod Motion			X

TABLE 6.1.3

SEQUENCE OF EVENTS

UNCONTROLLED RCCA WITHDRAWAL

EVENT	TIME(S)		PARAMETER	
	FSAR	RETRAN	FSAR	RETRAN
Rod Withdrawn	0	0	--	--
Reactor Tripped on Over-Temp. ΔT	50.5	55.6	--	--
Turbine Tripped on Reactor Trip	--	56.6	--	--
Maximum Core Power	51	55	113.6%	113.4%
Maximum Pressurizer Pressure	51	55	2332 psia	2322 psia
Maximum Core Average Temperature	52	57	585.8°F	585.6°F

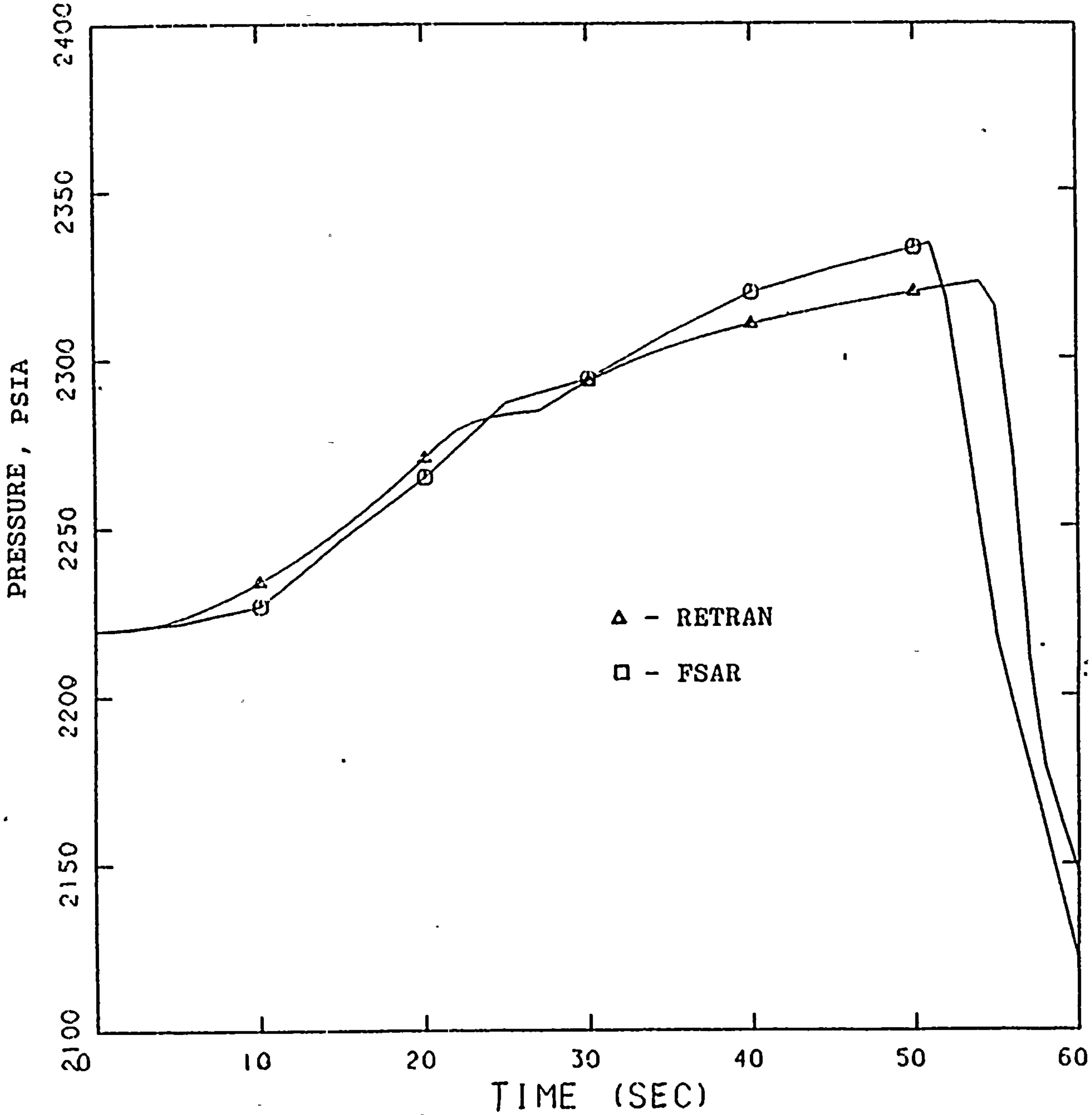


FIGURE 6.1.1 PRESSURIZER PRESSURE
UNCONTROLLED RCCA WITHDRAWAL

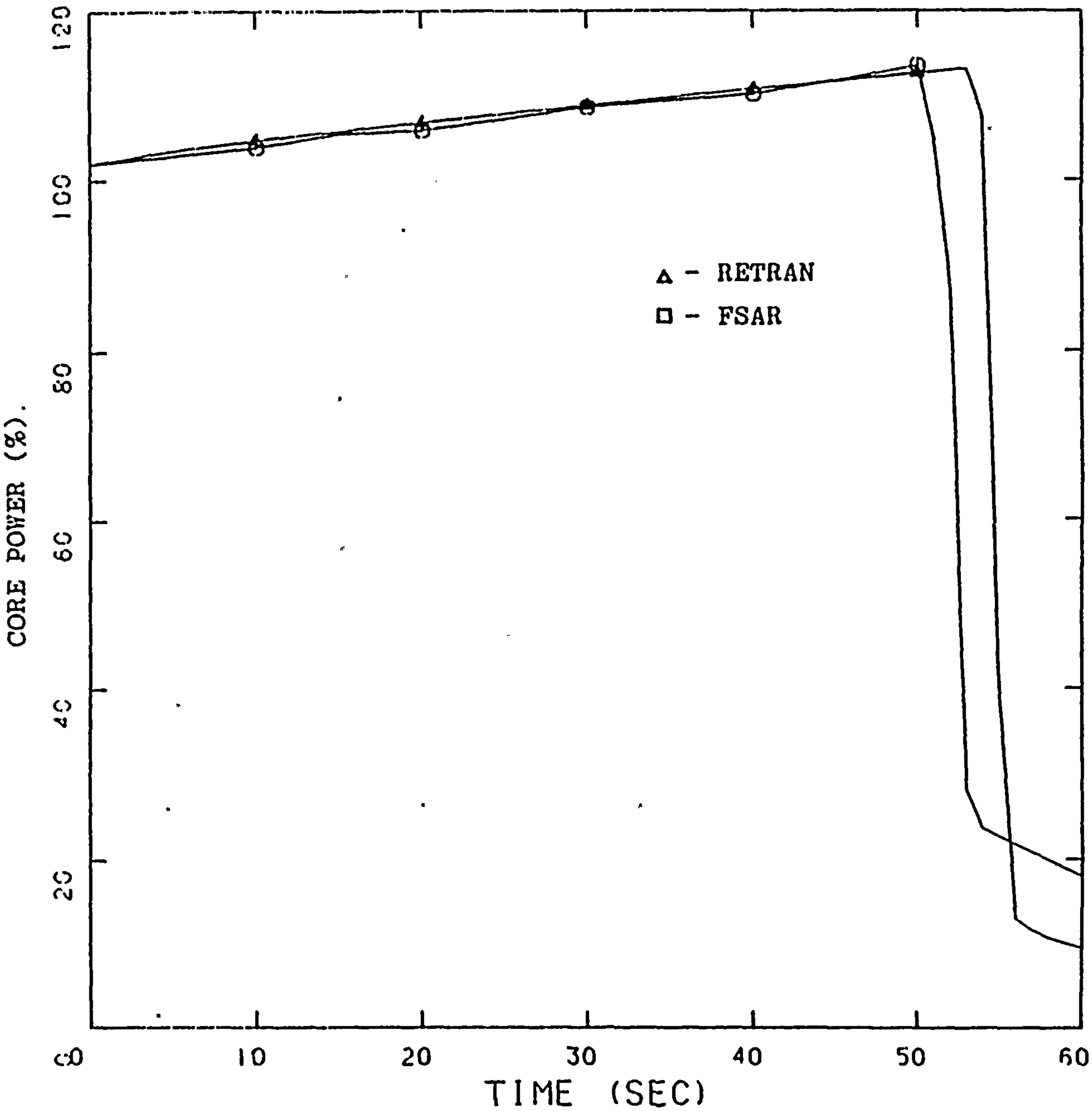


FIGURE 6.1.2 PERCENT CORE POWER UNCONTROLLED RCCA WITHDRAWAL

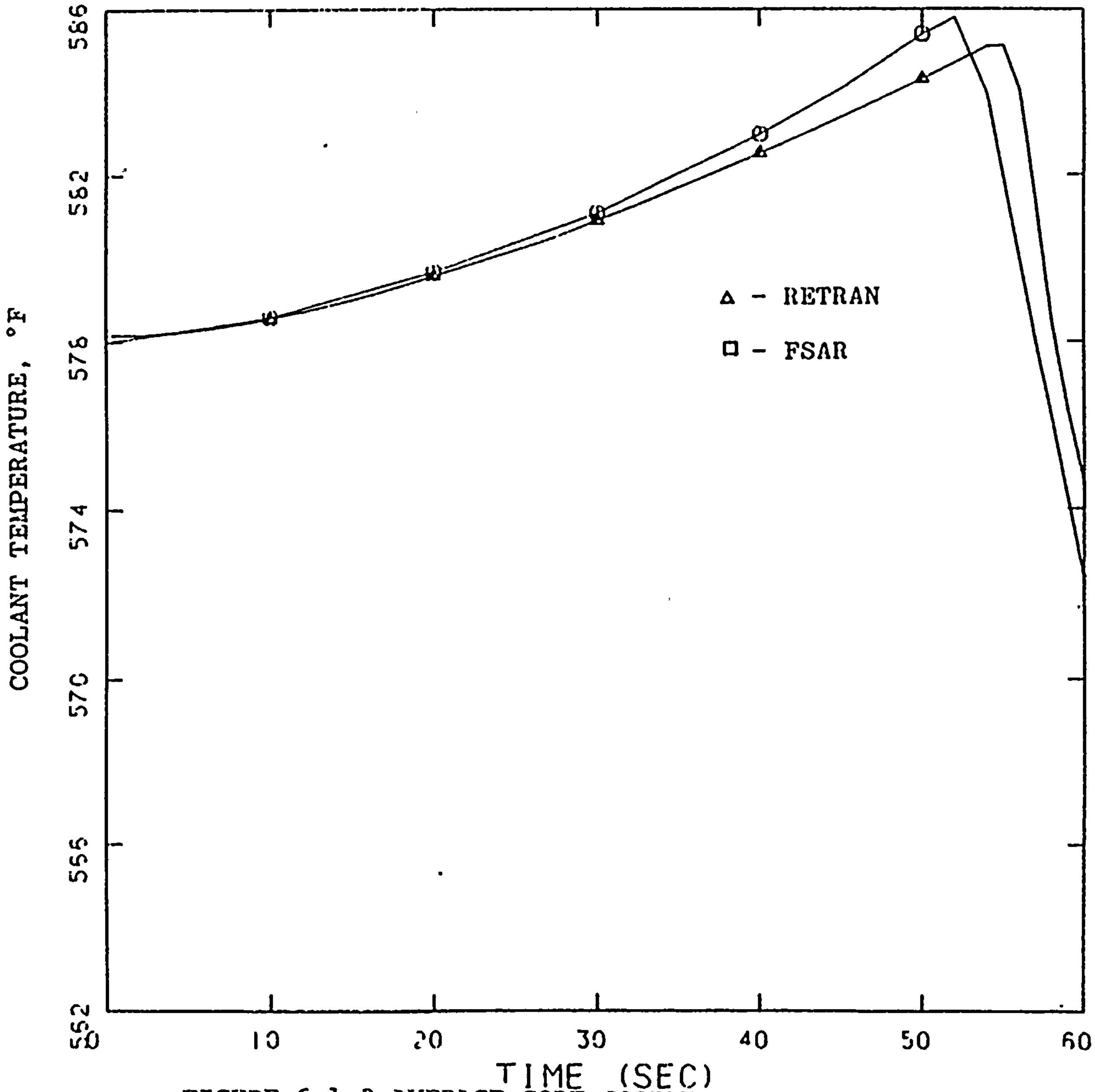


FIGURE 6.1.3 AVERAGE CORE COOLANT TEMPERATURE
UNCONTROLLED RCCA WITHDRAWAL