CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9605100199 DOC.DATE: 96/05/03 NOTARIZED: NO DOCKET #
FACIL:50-250 Turkey Point Plant, Unit 3, Florida Power and Light C 05000250
50-251 Turkey Point Plant, Unit 4, Florida Power and Light C 05000251
AUTH.NAME AUTHOR AFFILIATION
HOVEY,R.J. Florida Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION
Document Control Branch (Document Control Desk)

SUBJECT: Responds to RAI re proposed amend concerning thermal power uprate.

DISTRIBUTION CODE: A001D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 40
TITLE: OR Submittal: General Distribution

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD2-3 LA	1	1	PD2-3 PD	1	1
	CROTEAU,R	1	1			
INTERNAL:	FILE CENTER 01	1	1	NRR/DE/EMCB	1	1
	NRR/DRCH/HICB	1	1	NRR/DSSA/SPLB	1	1
	NRR/DSSA/SRXB	1	1	NUDOCS-ABSTRACT	1	1
	OGC/HDS3	1	0			
EXTERNAL:	NOAC	1	1	NRC PDR	1	1

C A T E G O R Y 1 D O C U M E N T

NOTE TO ALL "RIDS" RECIPIENTS:
PLEASE HELP US TO REDUCE WASTE! CONTACT THE DOCUMENT CONTROL DESK, ROOM OWFN 5D-5(EXT. 415-2083) TO ELIMINATE YOUR NAME FROM DISTRIBUTION LISTS FOR DOCUMENTS YOU DON'T NEED!

TOTAL NUMBER OF COPIES REQUIRED: LTTR 12 ENCL 11

FPL

MAY 03 1996

L-96-117
10 CFR §50.36
10 CFR §50.90

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

Gentlemen:

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Request for Additional Information (RAI)
Proposed License Amendments
Thermal Power Uprate

By letter L-95-245, dated December 18, 1995, Florida Power and Light Company (FPL) submitted a request to amend Turkey Point Units 3 and 4 Operating License and Technical Specifications. In a letter to T. F. Plunkett from R. P. Croteau dated March 26, 1996, the staff requested additional information to support the technical review of the Proposed License Amendments (PLA). A public meeting was held on April 4, 1996, at the White Flint offices of the NRC. FPL presented the responses to the RAI and provided the opportunity for follow-up questions. In a letter to T. F. Plunkett from R. P. Croteau dated April 24, 1996, the staff requested additional information as a result of follow-up questions. The response to both these RAI's is enclosed.

Should there be any questions, please contact us.

Very truly yours,

R. J. Hovey
Vice President
Turkey Point Plant

Enclosure

JAH

cc: S. D. Ebneter, Regional Administrator, Region II, USNRC
T. P. Johnson, Senior Resident Inspector, USNRC, Turkey Point
W. A. Passetti, Florida Department of Health and Rehabilitative Services

A001
1/1

Response to NRC Request for Additional Information Related to Thermal Power Uprating at Turkey Point Units 3 and 4

On March 26, 1996, the NRC provided a list of staff-identified questions that FPL was requested to discuss at a meeting to be held at the NRC Offices in White Flint, Maryland on April 4, 1996. Based on the discussions held at this public meeting, it was requested that FPL formally respond to all of the questions identified in the March 1996 NRC letter.

A. Emergency Preparedness and Radiation Protection Branch

1. **"Pages 3 and 14 of 17 attachment 1: Reference is made to RG 1.52 (revision 2 issued in 1978). Please address the use of more current standards or methods specified in more recent industry standards, such as ASME N510-1989 and ASTM D3803-1989, as opposed to the standards referenced in RG 1.52, revision 2 for filter testing."**

   FPL Response

   The Control Room Emergency Ventilation System is included in plant Technical Specification Section 3/4.7.5. Surveillance 4.7.5.c.2 identifies testing methods to be used and requires that a carbon sample be obtained per Regulatory Position C.6.b of RG 1.52, Revision 2, March 1978. The carbon sample is analyzed per ANSI N510-1975.

   The PLA requests change in acceptance criteria (testing results), consistent with RG 1.52, from 90% to 99%. No change in testing methods is being proposed. Because the assumed methyl iodide removal efficiency assumed in the control room dose analysis is 95%, a testing acceptance criteria of 99% is consistent with the recommendations of RG 1.52, Rev. 2, which is the licensing commitment document for Turkey Point.

2. **"Provide a detailed description of the fission product removal of the containment spray system and the extent to which credit is taken for the cleanup function in the analysis of the large break LOCA accident analysis (WCAP 14276, Rev. 1, pg 3-148). List the containment volumes not covered by the spray and the estimated forced or convective postaccident ventilation of these unsprayed volume."**

FPL Response

The containment is treated as a single well-mixed volume due to the turbulence induced by blowdown, containment sprays, emergency fan coolers, and emergency containment filter units. The containment spray system was not credited in the fission product removal in the LOCA analyses other than as a facilitator in providing a well-mixed containment volume. The only elemental iodine removal (except through filtration) from containment is due to deposition on containment surfaces up to a limit Deposition Factor (DF) of 100. In addition, the analysis assumes two units of the Emergency Containment Filtering System (ECFS) are operating after a 90 second delay. The ECFS filter removal efficiencies used in the analysis are 90% for elemental iodine, 30% for organic iodine, and 95% for particulate iodine.

3. **"For the LOCA analysis, at least ten or more computer codes were used in your evaluation. Discuss briefly why so many codes were used and discuss the accuracy and veracity of the final results."**

FPL Response

The Large and Small Break LOCA analyses which support the Turkey Point Units 3 and 4 uprating were performed with NRC-approved Evaluation Models. The only exception is the COSI/Safety Injection in the Broken Loop model for Small Break LOCA. This model is being actively reviewed by the NRC. When licensing these Evaluation Models, various comparisons were made of calculated results to single effect and integral test data and these Evaluation Models have been shown to meet all 10 CFR 50, Appendix K requirements (See NRC Safety Evaluation Reports in WCAP-10266-P-A and WCAP-10054-P-A). A more detailed discussion of the Appendix K Large and Small Break LOCA Evaluation Models is presented below:

Appendix K Large Break LOCA Evaluation Model

Historically, the Westinghouse Large Break LOCA Evaluation Model has consisted of a set of computer codes to model the three distinct phases of the Large Break LOCA transient: blowdown of the RCS, refilling of the lower plenum and reflooding the vessel. In the original Evaluation Model, which was approved by the NRC in Reference 10, the SATAN code was used for blowdown, the WREFLOOD code was used for refill of the lower plenum and reflooding of the vessel, and the LOCTA code was used for

cladding heatup calculations. In addition, the COCO code was used to calculate containment pressure response.

Westinghouse developed a code known as BART to calculate the detailed thermal hydraulic conditions in the core during the reflood period. This code replaced the FLECHT correlation used by LOCTA with a more mechanistic heat transfer coefficient Evaluation Model. As with the previous Evaluation Model, this Evaluation Model also consisted of SATAN for blowdown, WREFLOOD for refilling of the lower plenum and the NRC approved this Evaluation Model (EM) in Reference 4.

Subsequently, Westinghouse developed the BASH code, which calculated a dynamic oscillatory flooding rate during the reflood portion of the transient by combining the mechanistic core model and a more detailed loop model. In addition, BART and LOCTA, which were two separate codes in the BART EM, were combined into a single code, LOCBART. This Evaluation Model became known as the 1981 Evaluation Model with BASH. As with the previous Evaluation Model, this Evaluation Model also consisted of SATAN for blowdown, WREFLOOD for refilling of the lower plenum and COCO for containment response. The NRC approved this Evaluation Model in Reference 5.

A more detailed description of the codes that make up the Evaluation Model and references to their source WCAPs may be found in the Uprate Licensing Report, WCAP-14276, Rev. 1, Section 3.3.1.3, and graphically depicted in Figure 3.3.1-2 of WCAP-14276, Rev. 1.

Appendix K Small Break LOCA Evaluation Model

The Appendix K Small Break LOCA Evaluation Model used for the Turkey Point Units 3 and 4 uprate analysis is the NOTRUMP Evaluation Model. This Evaluation Model is documented in WCAP-10054-P-A (Reference 7). The Evaluation Model has been approved by the NRC. However, one new model used in the Turkey Point uprate analysis, COSI/Safety Injection in the Broken Loop (Reference 8), has not yet been approved by the NRC, but is currently under review. This Evaluation Model used the NOTRUMP code to calculate the thermal-hydraulic response of a hypothetical Small Break LOCA. In addition, the fuel rod cladding heatup transient based on the calculated thermal-hydraulic transient (i.e. the Peak Cladding Temperature calculation) is provided by the LOCTA code. A more detailed description of the codes that make up the Evaluation Model, and references to their source WCAPs,

may be found in the Uprate Licensing Report, WCAP-14276, Rev. 1, Section 3.3.2.3.

4. **"Discuss the impact of power uprate on radiolysis. Our experience indicates that, as a result of a power uprate, the production of oxygen by radiolysis after a LOCA will increase proportionally with the power level. Does sufficient capacity exist in the licensee combustible gas control system to accommodate this increase in oxygen production."**

   FPL Response

   As part of the original plant licensing, a hydrogen generation analysis was performed at 2300 MWt power that considered contributions from zirconium-water reaction, radiolysis, and contributions from corrodible metals. For the uprate project, this analysis was performed at 2346 MWt (102% power). Since the original FSAR work was based on a power of 2300 MWt while the work for the uprating program assumed a power of 2346 MWt, the uprating in itself leads to a 2% increase in hydrogen production through the radiolysis mechanism. This new analysis demonstrates that with no hydrogen removal mechanisms, hydrogen concentrations would remain below 4 volume percent (v/o) for the first 17 days following the accident. Operation of a 55 scfm recombiner or post-accident containment vent is capable of maintainig and reducing hydrogen concentration to below 4 v/o.

5. **"Briefly discuss how the higher power level effects the source terms, onsite and offsite doses, and control room habitability during normal and accident conditions."**

   FPL Response

   The original source term used for normal and accident analyses was based on a core thermal power of 2300 MWt. The source term was conservatively recalculated at 102% power (2346 MWt). The calculation also considered changes that have occurred since the original calculation was performed, including burnups, enrichments, fuel masses, and operating times.

   Normal source terms are affected by power level, burnup, enrichment, and fuel mass. There is also the additional effect of any changes to plant operating parameters, such as RCS temperature or mass. Normal or shielding source terms are generally proportional to the RCS coolant concentrations, which were also recalculated in the

uprating program.

A comparison between the two source terms was performed. This comparison shows essentially no change in total beta and gamma source strengths nor in the spectrum hardness.

Accordingly, the higher power level of uprate will have virtually no impact on onsite doses, offsite doses and control room doses during accident conditions and the impact on normal operating doses is expected to be small.

6. **"Discuss how the radiation levels from both accident and normal operations are affected by the uprated power level."**

<u>FPL Response</u>

Based on the review performed with respect to total source strength and spectrum hardness, it is expected that there will be no change in accident radiation levels.

The normal source.term and dose rates can be expected to increase in proportion to the increase in power level. Actual doses during normal plant operations are expected to increase by the ratio of operating power levels.

7. **"Discuss the effects of the power uprate on coolant activation products, activated corrosion products, and fissions products."**

<u>FPL Response</u>

The most significant "coolant activation product" during operation is N-16; the activity of this nuclide will be directly proportional to the increase in power levels. The situation would be similar for other coolant activation products.

The situation for activated corrosion products is similar. In this case, however, there is some degree of depletion or target burnout that may ameliorate this increase, notably for Co-60.

Based on the assessment made with respect to total source strength and spectrum hardness, there is little change in the calculated fission product activities. In the event of any fission product leakage from the fuel, fission products are expected to increase by the ratio of operating power.

Plant Technical Specification 3/4.4.8 limits the reactor coolant specific activity to certain prescribed limits that are much lower than those used in calculations (which are based on 1% failed fuel). Operation at Turkey Point will continue to remain within the Technical Specification limits at uprated power.

B. Reactor Systems Branch

1. "Please confirm that the methodology used in the transient and accident analyses documents in WCAP-14276, Revision 1, is consistent with that used in the UFSAR. Identify any differences and discuss their acceptability."

FPL Response

All methodology used in the transient and accident analyses that are documented in WCAP-14276, Rev 1, are consistent with the proposed UFSAR sections for each of the respective analyses. Each of the areas are further discussed in this response.

The non-LOCA analyses detailed in WCAP-14276, Rev. 1, are consistent with the current UFSAR except for the Feedwater Malfunction Event (UFSAR Section 14.1.7), and the change from Standard Thermal Design Procedure (STDP) to Revised Thermal Design Procedure (RTDP). As part of the Turkey Point RTDP reanalyses, this section of the UFSAR was revised to reflect a feedwater malfunction event that resulted in an excessive feedwater flow, in contrast to the reduction in feedwater enthalpy incident (currently in the UFSAR) which is known to be less limiting and bounded by the Excessive Load Increase Event. The non-LOCA accident analyses documented in WCAP-14276, Rev. 1, are consistent with the proposed UFSAR update for the uprating which continue to reflect RTDP and a feedwater malfunction event that results in an excessive feedwater flow.

The containment integrity analysis methodology for the LOCA and MSLB transients that are documented in WCAP-14276, Rev. 1, is consistent with the proposed updated UFSAR. This methodology has been used for other Westinghouse PWRs and has been approved by the NRC for those applications.

The LOCA methodology which supports the results in the Turkey Point Units 3 and 4 Uprating Licensing Report, WCAP-14276, Rev. 1, is consistent with that used in the

UFSAR except for the following differences:

Large Break LOCA

The Turkey Point current Large Break LOCA Analysis of Record found in UFSAR Section 14.3.2.1 was performed with the Westinghouse 1981 Evaluation Model with BART (Reference 4). The 1981 Evaluation Model with BART was reviewed and approved by the NRC. The Large Break LOCA uprate analysis found in Section 3.3.1 of WCAP-14276, Rev. 1, was performed using the Westinghouse 1981 Evaluation Model with BASH (Reference 5). The BASH code of the Evaluation Model consists of the previously approved BART code with a more detailed Reactor Coolant System (RCS) loop model. The 1981 Evaluation Model with BASH was reviewed and approved by the NRC. Also, the ESHAPE methodology, which is an updated version of the methodology in Addendum 1-A of Reference 2, is used to specifically analyze skewed power shapes for Large Break LOCA. Therefore, the Turkey Point Units 3 and 4 Large Break LOCA uprate analysis has been performed with the most recent Westinghouse methodology approved by the NRC.

Small Break LOCA

The current Small Break LOCA Analysis of Record found in UFSAR Section 14.3.2.2 was performed with the Westinghouse NOTRUMP Evaluation Model (References 6 and 7). The Westinghouse NOTRUMP Evaluation Model has been reviewed and approved by the NRC. The Small Break LOCA uprate analysis found in Section 3.3.2 of WCAP-14276, Rev. 1 was also performed with the NOTRUMP Evaluation Model. However, the Small Break LOCA uprate analysis also used the COSI/Safety Injection in the Broken Loop Model as described in References 8 and 9. Since the COSI/Safety Injection in the Broken Loop Model is not yet approved by the NRC, the Small Break LOCA uprate analysis was submitted earlier than the uprate licensing report so that it could be reviewed in advance. Two other plants have also submitted Small Break LOCA analyses which used the improved condensation model.

2. **"Please confirm that only safety grade systems and components are assumed in mitigating design basis events."**

FPL Response

Safety grade systems and components are assumed to operate and perform their intended design function for accident analyses design basis events. Non-safety grade equipment may also be assumed to operate for both Non-LOCA and LOCA accidents if its operation results in more adverse calculated consequences for the accident.

3. **"Provide the results of the analyses for Locked Rotor/Shaft Break accident assuming a loss of off-site power coincident with the event. Discuss the amount of fuel failure during the event and the calculated radiological consequences. Are all fuel pins with DNBR below the MDNBR assumed to fail?"**

FPL Response

The Turkey Point plant was originally licensed assuming no Loss of Offsite Power. Consequently, the Locked Rotor analysis presented in WCAP-14276, Rev. 1, reflects this assumption. For the dose calculations associated with the locked rotor/shaft break event, all rods in DNB are assumed to fail.

4. **"Provide the results of an analysis for a postulated main feedwater line break."**

FPL Response

The non-LOCA transients analyzed for Thermal Power Uprating are consistent with the plant's original licensing basis. Because Turkey Point was licensed prior to the issuance of the Standard Review Plan, a main feedwater line break is not a part of the Turkey Point licensing basis. This is consistent with other submittals gaining NRC approval for uprating.

5. **"Provide major transient curves for the reanalysis of the postulated main steam line break."**

FPL Response

The Main Steam Line Break (MSLB) core response analysis is performed at Hot Zero Power (HZP) conditions, which did not change for the power uprating, and the HZP stuck rod coefficients remained applicable for uprated conditions since these are calculated at HZP and

parameters affecting these coefficients (mainly the moderator density coefficient) have not changed for the power uprating conditions. Therefore, the analysis performed under the RTDP project is not affected by the uprating and the DNBR design basis continues to be met. The curves associated with the proposed UFSAR write-up for the MSLB event are attached.

6. **"In your analysis of a large break LOCA, for the case of minimum ECCS case, the loss of the LHSI pump is assumed as the most limiting single failure. Please discuss the potential loss of a diesel affecting ECCS."**

<u>FPL Response</u>

Loss of a diesel generator for a Large Break LOCA would also result in the loss of one train of pressure-reducing equipment (i.e., containment spray pumps and fan coolers). This position is not consistent with 10 CFR 50, Appendix K, Section I.D.2 and Branch Technical Position CSB 6-1. That is, operation of all containment pressure-reducing equipment at maximum heat removal rates should be assumed. Through past studies, Westinghouse has determined that the loss of one Low Head Safety Injection (LHSI) pump is the limiting single failure for the Westinghouse Large Break LOCA BASH Evaluation Model (see Reference 5, Section 11.0 and Reference 10, Section 3.6). Therefore, loss of one LHSI pump is assumed to be the limiting single failure in all analyses using the current Westinghouse Large Break LOCA Evaluation Model, including the Turkey Point Units 3 and 4 Large Break LOCA uprate analysis.

7. **"Discuss the most limiting single failure assumed in the SGTR analysis in light of the maximum dose release."**

<u>FPL Response</u>

The UFSAR Steam Generator Tube Rupture (SGTR) analysis for Turkey Point Units 3 and 4 was performed to evaluate the potential offsite radiation doses due to this event. The analysis includes thermal and hydraulic calculations to determine the amount of primary to secondary break flow and steam release to the atmosphere, and radiological calculations to determine the resulting offsite dose.

The SGTR thermal and hydraulic analyses performed for many of the earlier Westinghouse plants, including Turkey Point, did not include a computer analysis to determine

the plant transient behavior following an SGTR. Rather, simplified calculations were performed, based on the expected SGTR transient response, to determine the primary to secondary break flow and the steam release to the atmosphere for use in calculating the offsite radiation doses due to the event. Because of the nature of the analysis, some simplifying assumptions were used in performing the calculations, including the assumption of no single failure. Although no single failure is explicitly modeled, the analysis is considered to provide a conservative estimate of the offsite doses following a SGTR.

The SGTR analysis performed for the Turkey Point Uprate assumed the same methodology as presented in the UFSAR, including the assumption of no single failure. The analysis was performed using uprated power and corresponding assumptions related to the uprating program. The offsite doses were calculated using the methodology defined in Standard Review Plan 15.6.3.

Plants of the same vintage as Turkey Point apply the same SGTR analysis methodology and have been approved for operation at uprated power conditions, with steam generator replacements, or changes to the Technical Specifications.

8. **"Please confirm that the new proposed loop design flow rate of 85,000 gpm is incorporated in the analyses documented in WCAP-14276, Rev. 1, including the loss of RCS flow and Locked Rotor/Shaft Break."**

<u>FPL Response</u>

The loop thermal design flow assumed for the power uprating for the Non-LOCA analyses was 85,000 gpm. The loss of RCS Flow and Locked Rotor/Shaft Break Rods-in-DNB transients are considered Revised Thermal Design Procedure (RTDP) events, i.e., uncertainties on power, pressure, temperature and flow are statistically combined into the DNBR limit-value. Consequently, for the Loss of RCS Flow and Locked Rotor/Shaft Break Rods-in-DNB, transients the Minimum Measured Loop Flow of 88,000 gpm was assumed.

The Locked Rotor/Shaft Break Peak Pressure/Peak Clad Temperature case is performed using Standard Thermal Design Procedure (STDP) i.e., uncertainties on power, pressure, temperature, and flow are explicitly assumed in

the non-LOCA analyses. Consequently, Thermal Design Flow (TDF) of 85,000 gpm was assumed for the Locked Rotor/Shaft Break Peak Pressure/Peak Clad Temperature case.

The thermal design flow rate of 85,000 gpm/loop was incorporated into the LOCA analyses which support the Turkey Point Units 3 and 4 uprating. This conservative flow rate is appropriate for use in LOCA analyses, and is called out in the LOCA input assumptions tables in WCAP-14276, Rev. 1 for both Large and Small Break LOCA analyses (Tables 3.3.1-1 and 3.3.2-1, respectively). This flow rate is also consistent with a conservatively high steam generator tube plugging level of 20% assumed in the LOCA analyses.

9. **"Please confirm that the proposed setpoints for ESFAS are incorporated in the transient and accident analyses in WCAP-14276, Rev. 1."**

<u>FPL Response</u>

The proposed Engineered Safety Features Actuation System (ESFAS) Setpoints for the power uprating have been reviewed. The appropriate safety analysis limit values for these ESFAS setpoints have been incorporated into the accident analyses for the power uprating contained in WCAP-14276, Rev 1.

10. **"Describe the steam generator tube plug level assumed in non-LOCA event analyses."**

<u>FPL Response</u>

The non-LOCA event analyses bound operation up to a maximum Steam Generator Tube Plugging (SGTP) level of 20%.

The non-LOCA event analyses assume 0% Steam Generator Tube Plugging for those events where the heat transfer from the primary to the secondary is to be conservatively high (Cooldown Events), and assume a 20% SGTP level for those events where the heat transfer from the primary to secondary is to be conservatively low (Heatup events and Loss of Flow events). Consequently, the non-LOCA event analyses bound operation up to a maximum SGTP level of 20%.

C. **Plant Systems Branch**

1. **"Please address the increase in the probability of turbine overspeed and associated turbine missile production due to plant operations at the proposed uprated power level."**

<u>FPL Response</u>

Total turbine missile probability is composed of the following components:

1. Missile generation at destructive overspeed,
2. Missile generation at running speed, and
3. Missile generation at design overspeed.

The probability of turbine missile production is dominated by the probability of a destructive overspeed. A destructive overspeed is caused by the failure to close of both high pressure turbine inlet valves on the same steam line , and is therefore independent of uprating, as outlined in WCAP-11525 "Probabilistic Evaluation of Reduction of Turbine Valve Testing Frequency, June 1989.

The missile production is also minimally dependent on the probability that a missile would occur at running speed or design overspeed. The probability of missile production at running speed is a function of capacity factor for the unit, which does not change by uprating. The probability of a turbine design overspeed is a function of control valve reliability and is independent of thermal rating.

Accordingly, the probability of a turbine missile being generated is not adversely affected by uprate.

2. **"In page 5-33, of WCAP-14276, Rev. 1, Westinghouse indicated that for normal refueling the maximum expected SFP heat load and temperature for a ½ core offload at 150 hours after shutdown are 16.6 x 10^6 Btu/Hr and 147°F, respectively. In page 14D-16 of the FSAR Appendix 14D, Florida Power & Light Company stated that as the result of the expansion of spent fuel storage in the pool, the decay heat load for each pool increases to 16.98 x 10^6 Btu/Hr and the corresponding pool peak transient temperature after refueling increases to less than 141°F. It is not clear why the pool with a higher heat load (16.98 x 10^6 Btu/Hr vs. 16.6 x 16^6 Btu/Hr) would have a lower peak temperature (141°F vs. 147°F). Please provide detailed discussion for the above discrepancy."**

FPL Response

The spent fuel pool (SFP) cooling system analyses currently contained in Appendix 14D of the FSAR assume a Component Cooling Water (CCW) supply temperature of 100°F to the SFP heat exchanger. This CCW supply temperature is consistent with the original design of the CCW system and SFP heat exchanger component specification. However, the thermal power uprate analyses for the SFP cooling system conservatively assume a CCW supply temperature of 105°F due to the increased CCW heat exchanger fouling that has been assumed in uprate thermal-hydraulic analyses. This CCW supply temperature difference accounts for a 5°F delta in the FSAR Appendix 14D (141°F peak temperature) and thermal power uprate (147°F peak) SFP cooling analysis results. The other 1°F difference in the two analyses can be accounted for in the SFP heat exchanger modeling methodology. Note that both analyses yield essentially the same delta-T (SFP peak temperature minus CCW supply temperature) across the SFP heat exchanger (42°F for thermal uprate versus 41°F for FSAR Appendix D) for essentially the same decay heat load.

With respect to decay heat, the decay heat value of 16.98 x 10^6 Btu/Hr calculated in the FSAR Appendix 14D analysis assumes conservative burnup, batch sizes and decay times for the offloaded fuel in the SFP. The decay heat value of 16.6 x 10^6 Btu/Hr calculated for the thermal power uprate analysis assumes a core with all assemblies at a burnup of 60,000 MWD/MTU and actual burnups, batch sizes and decay times for the previously offloaded fuel. Although the methodology utilized for the uprate is still conservative, it does yield a lower overall decay heat value than the FSAR Appendix 14D analysis, even when assuming an increased core power level of 2300 MWt. For both the FSAR Appendix 14D and thermal power uprate SFP analyses, decay heat values have been calculated consistent with the methodology provided in NRC Branch Technical Position ASB 9-2.

Additional SFP heatup analyses have been performed to address the Turkey Point normal full core offload refueling practice. The analyses performed for thermal uprate are essentially identical to those performed for the Turkey Point Unit 4 Cycle 16 refueling outage. The decay heat load assumed in the analyses conservatively represents a full core offload commencing 150 hours after shutdown, with the remaining fuel rack spaces filled with previously discharged fuel. The analyses conclude that

adherence to the current administrative limit of 140°F (i.e., stopping the offload if the SFP temperature reaches 140°F) will maintain the peak pool temperature below the established acceptance criterion of 150°F.

3. "For abnormal operation without SFP cooling, the time to reach boiling is 4.5 hours and the maximum boil off rate is 76.3 gpm. Assuming a loss of SFP cooling, provide the following information:

- How long abnormal operation without SFP cooling is expected to be? What scenarios that lead to a loss of SFP cooling were considered? What actions would be required to restore SFP cooling?

- Will there be sufficient make-up water for the SFP? Provide detail description of the make-up water sources. How the pool boil-off will be collected/treated?"

FPL Response

Only credible SFP cooling system active component failures are considered as part of the original system design bases. The SFP cooling loop does not incorporate redundant active components because of the large heat capacity of the spent fuel pool and its corresponding slow heatup rate. Passive component failures are not assumed. No specific scenarios were originally considered when postulating a loss of SFP cooling during the 1984 SFP high density re-rack project, but the information below provides the more probable sequences associated with this event.

The SFP cooling system active components credited in the UFSAR are the A and B 100% capacity SFP cooling pumps. Both the A and B SFP pumps are powered from their respective unit's C vital load center. The C load centers are powered from each unit's A 4160V switchgear bus which is capable of being powered by the respective unit's A Emergency Diesel Generator (EDG). If the operating SFP cooling pump fails or if power to the C load center is lost, either with or without a loss of offsite power, forced SFP cooling will be lost until either the backup SFP cooling pump can be started or power to the load center can be restored. Plant procedures are in place to both start the second SFP cooling pump and/or repower the affected unit's C load center. It is estimated that restoration of SFP cooling can be accomplished within one hour.

In the unlikely event SFP cooling restoration is delayed, the Unit 3 and 4 spent fuel pools are designed to accommodate boiling. Existing plant procedures identify four separate make-up sources to either spent fuel pool, in the following order of preference:

1. Refueling Water Storage Tank (RWST)

2. Primary Water System

3. Demineralized Water System

4. CVCS Holdup Tanks

Each of these makeup sources has a capacity greater than the maximum calculated boil-off rate.

In the unlikely event either the Unit 3 or 4 SFP boils, steam will collect in the affected unit's SFP building and would be exhausted by the associated SFP building exhaust fan. A prefilter and high efficiency particulate (HEPA) filter are located on each SFP exhaust fan suction. A radiation detector is located downstream of each exhaust fan which monitors for any activity released to the atmosphere. However, operation of the SFP ventilation system is not assumed in the dose analysis. The offsite dose analysis for pool boiling conservatively assumes that both the Turkey Point Unit 3 and 4 spent fuel pools boil simultaneously, that each pool contains a full core offload that has decayed for 150 hours, and that 1% of the fuel rods have cladding failure. The calculated offsite doses are within specified limits.

4. **"It appears that EQ outside containment has not been addressed. Please demonstrate what impact plant operations at the proposed uprated power level will have on EQ outside containment."**

<u>FPL Response</u>

Environmental Qualification (EQ) of equipment located outside containment is essentially composed of two aspects, the radiological effects and any steam-pressure environment associated with a pipe break outside containment. As described in the response to previous questions, plant radiological and shielding analyses were originally performed at 2300 MWt and have been reassessed based on a power level of 2346 MWt. Review of this change in power level shows that since there is

essentially no change in source term, there will be no affect on radiological EQ. Refer to Section 6.3.2 of WCAP-14276, Revision 1, for additional information.

Those areas of the Turkey Point plant which are subject to pipe breaks outside containment with potentially adverse environmental consequences are located in open areas of the turbine building or steam trestle. Since these are open areas, the only environmental effect is the impact of steam impinging on safety-related equipment in the vicinity of the pipe break (i.e., no subcompartments to reanalyze). Since the pressure and temperature ratings of the piping considered to hypothetically break remain unchanged by uprating, and when steam is discharged to atmosphere, its temperature and pressure are controlled by the atmospheric conditions, the steam-pressure EQ environment associated with EQ outside containment is unaffected. Refer to Section 6.4.1 of WCAP-14276, Revision 1, for additional information.

Accordingly, since neither the radiological aspects of EQ nor the steam-temperature environment of EQ is affected, EQ outside containment is unaffected by plant uprating.

5. **"It is stated on page 9.3-7 of the FSAR that the SFP cooling loop consists of a pump, heat exchanger and associated components (i.e., filters, demineralizer, etc.) and that in the event of a failure of the SFP cooling pump, a 100% capacity spare pump is permanently piped into the SFP cooling system and is available as a standby pump. However, in Figure 9.3-11 of the FSAR, three SFP cooling pumps are shown as part of the SFP cooling system. Please provide a clarification for this discrepancy. In addition, please identify the safety class and capacity for each of these pumps."**

FPL Response

Consistent with the statement in the FSAR, the SFP cooling loop during normal system operation consists of one cooling loop, one pump, one heat exchanger and associated components. However, a total of three SFP cooling pumps are available on each Turkey Point unit. Two of the pumps are 100% capacity pumps (the A and B SFP cooling water pumps) and tie-in connections exist for a third pump. A less than 100% capacity pump is currently connected to these tie-ins on both Turkey Point Units 3 and 4 (the SFP emergency cooling water pump). In the

event of a failure of one of the 100% capacity pumps, the second 100% capacity pump is available and can be manually placed in service in accordance with current plant procedures. The emergency SFP cooling water pump is also available and can be manually placed in service in accordance with current plant procedures, if needed. The current design and configuration of the SFP cooling loop is consistent with that described by the NRC in their Safety Evaluation for the Turkey Point Units 3 and 4 SFP high density rerack project approved in 1984. Note that an FSAR change package has been recently issued to provide additional clarification on the design of the SFP cooling loop.

The original Turkey Point Units 3 and 4 SFP cooling system was designed as not safety grade. However, in accordance with the 1984 high density rerack project commitments, modifications have been made to assure that the cooling system remains functional after a safe shutdown earthquake. The seismic upgrades included the SFP cooling loop piping and major system components (A and B SFP cooling pumps, power supplies and the SFP heat exchanger). As such, passive component failures as a result of a seismic event are not considered credible. Also, as recommended by Regulatory Guide 1.26, the SFP cooling water system (excluding the emergency cooling water pump) has since been classified as a Quality Group C system. Accordingly, the A and B SFP cooling water pumps are tested in accordance with the requirements of ASME Section XI.

D. Miscellaneous

1. "Section 7.4, NON-RADIOLOGICAL EFFECTS, of WCAP-14276, Rev. 1, states that "Protection of the environment is assured by compliance with permits issued by federal, state, and local agencies." Please confirm that none of these permits are affected by the proposed thermal uprate and no changes to the permits are necessary for the uprate."

<u>FPL Response</u>

FPL has reviewed the environmental permits associated with operation of Turkey Point Units 3 and 4. No changes to any permits are required, including the National Pollutant Discharge Elimination System (NPDES) permit.

Turkey Point has no specifically prescribed protective actions associated with endangered wildlife. FPL does have a monitoring permit to tag and count American crocodiles that is issued by both the U.S. Fish and Wildlife Service and the State of Florida.

2. "Please discuss the maximum anticipated discharge temperature from the circulating water system during normal operation for the uprate condition and any limits which exist on the discharge temperature."

<u>FPL Response</u>

The circulating water system at Turkey Point discharges to a closed cooling water canal system. The National Pollutant Discharge Elimination System (NPDES) permit for Turkey Point has no discharge temperature limit. Technical Specification 3.7.4 limits intake temperature to a maximum of 100°F.

It is estimated that the increase in discharge temperatures over existing plant operation will be about 0.7°F. The impact on intake temperatures is estimated to be about 0.2°F. The temperature effect caused by plant uprating represents a small effect when compared to seasonal effects and solar radiation heat gain.

3. "Table III-3, ANTICIPATED ANNUAL RELEASE OF RADIOACTIVE MATERIALS IN LIQUID EFFLUENTS FROM TURKEY POINT PLANT UNITS 3 & 4 - RECONCENTRATION FACTORS FOR COOLING CANAL SYSTEM, of the FES indicates that the table values were calculated for a power level of 2200 MWt. Following a thermal uprate, will operation continue to be bounded by the values of FES Table III-3 and all other parameters in the FES?"

FPL Response

The Turkey Point Final Environmental Statement (FES) identifies the expected radioactive releases for gaseous effluent (Table III-2) and for liquid effluent (Table III-3) for both units operating at 2200 MWt. The evaluation performed by the staff for gaseous releases was based on 0.25% leaking fuel and a 20 gallon per day primary to secondary system leak rate. The evaluation performed by the staff for liquid releases from the Waste Disposal System assumes 0.25% leaking fuel and a decontamination factor of 10^5 for the waste evaporator-demineralizer for all isotopes except iodine and tritium (a DF of 10^4 was assumed for iodine).

Using the FES as the baseline, the following releases by category are taken from Section III of the FES and are compared to the most recent annual radioactive effluent release report for Turkey Point (FPL letter L-96-083, dated March 27, 1996):

	FES Releases Ci/yr/unit	PTN Releases in 1995 Avg. Ci/yr/unit
Noble Gasses	3650	<1
Iodines and Particulates	0.8	<0.1
SG Blowdown (liquid)	27	0
Waste Disposal System	1	<0.1

Based on the fact that the uprating for Turkey Point is less than 5%, it is expected that PTN will operate well within the limits specified by the FES.

The following responses are the result of a letter to T. F. Plunkett from R. P. Croteau dated April 24, 1996. The questions are a result of follow up inquiries during and after the FPL presentation. This response addresses only section A and B of the April 25, 1996 letter, the response to section C will be provided later.

E. Questions Regarding Compliance with 10 CFR 50.61, Part 50 Appendix G, and Part 50 Appendix H

1. **"Will the proposed thermal power uprate change your current PTS assessment? Provide the projected maximum end-of-license (EOL) fluences at the inner diameter (ID) of the vessels and the RT_{PTS} values for the Turkey Point reactor vessel beltline materials."**

FPL Response

The uprate will change Turkey Point's current PTS assessment very slightly. Turkey Point will remain below the 300°F screening criteria through end-of-license (EOL). The revised EOL RT_{PTS} values for the limiting intermediate to lower shell girth weld (SA 1101) are 295°F @ 2.74E19 n/cm^2 and 293°F @ 2.68E19 n/cm^2 for Turkey Point Units 3 and 4, respectively.

2. **"Regarding Pressure Temperature (P/T) Limit Curves, provide the 1/4T and 3/4T fluence levels estimated for 19 EFPY."**

FPL Response

The pressure-temperature (P/T) curves were not re-calculated as part of the uprating analyses; only the term of applicability changed based on the projected change in fluences.

The limiting inside surface fluence is 2.022E19 n/cm^2 at 19 EFPY for both units (Reference 11). 1/4T and 3/4T fluences can be calculated using the Regulatory Guide 1.99, Revision 2, expression:

$$f(x)=f(\text{surface})\ e^{-0.24x}$$

where vessel thickness; (x)= 7.75" +.156" (clad)
(x)= 7.906"

then: f(1/4T)= 1.26E19 n/cm^2
f(3/4T)= 4.87E18 n/cm^2

3. "Provide an assessment of how the proposed thermal uprate will affect the EOL upper shelf energies and FPL's equivalent margins analyses for the limiting upper shelf energy materials in the Turkey Point reactor vessels. Include appropriate calculations and figures based on the guidelines of Regulatory Guide 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Material," dated May 1988."

FPL Response

The equivalent margins analysis (Reference 12) was performed using an inner wall EOL fluence of 2.7E19 n/cm^2 which approximates the value for EOL fluence after uprate. Therefore, the results of the equivalent margins analysis would be unchanged as would the EOL upper shelf energy estimates included in Tables 2-11 and 2-12 of the analysis. An assessment was performed which concluded that the existing analysis addresses the uprated condition and a revision to the equivalent margins analysis was not required. In addition, since the limiting material (SA 1101) at both Turkey Point units is a circumferential weld seam, the applied loading is significantly lower than vessels with long seams. The significance of this is reflected in Fig. 5-6 of Reference 12 where the large margin in the applied versus plant specific material properties can be compared.

F. Questions Regarding Steam Generator Tube Integrity

1. "Page 4-21 of WCAP-14276. FPL should assess the effect of the power uprate on (1) the minimum wall thickness of steam generator tubes, (2) the number of steam generator tubes susceptible to anti-vibration bar wear, and (3) susceptibility of the steam generator tubing to various forms of degradation mechanisms."

FPL Response

(1) The effect of the power uprate on the minimum wall thickness of the steam generator tubes was assessed and it was concluded that the current Technical Specification plugging limit of 40% by NDE is acceptable. Using conservative allowances for NDE uncertainty and continued growth, a minimum wall thickness of 0.020 inch is established for the 40% plugging limit. Since the Technical Specification plugging limit includes NDE uncertainty, an eddy current indication of 40% depth inherently provides

margin to the structural limit. Therefore, pluggable indications are not expected to exceed the minimum wall thickness. FPL plans to continue to comply with the existing Technical Specification tube plugging requirements.

(2) The effect of the power uprate on the number of steam generator tubes susceptible to anti-vibration bar (AVB) wear was assessed and is summarized as follows. The total bundle flow rate remains essentially unchanged with uprating. The increase in steam flow and concurrent increase in void fraction does result in an increased vibration potential in the U-bend region. This circumstance, however, does not contribute to any significant decrease in long term bundle integrity for the Model 44F steam generators. Small radius U-bend vibration does not lead to significant fatigue. A necessary condition for significant fatigue is denting at the top tube support plate. This does not occur in the Model 44F steam generators, which have stainless steel support plates. The larger radius U-bends, which are in contact with the AVBs, have been evaluated for increased wear. Evaluation for three steam generator models showed that the additional tubes subject to wear at the AVB intersections as a result of uprating constitute less than 0.3% of the total tube count over the life of the steam generator.

(3) The effect of the power uprate on the susceptibility of the steam generator tubing to various forms of degradation mechanisms was assessed. The increase in average heat flux resulting from the uprating will cause some increased potential for corrosion and long term fouling, though it is not the dominant factor. Operating history, more than changes which result from uprating, is the best indicator of whether the Turkey Point units are susceptible to significant corrosion or performance loss due to fouling. Turkey Point steam generators are not experiencing any significant corrosion or performance loss due to fouling and no significant change is expected.

2. **"After reviewing Section 3.4, Steam Generator Tube Rupture, and Section 4.9, Steam Generator in WCAP-14276, the staff is not clear whether FPL has addressed structural integrity of the steam generator tubing under uprate conditions based on Regulatory Guide 1.121. FPL should perform a steam generator tube assessment in accordance with Regulatory Guide 1.121."**

   FPL Response

   Paragraph IWB-3521.1 in Section XI of the ASME Code defines the plugging limit for tubing with an r/t ratio less than 8.7 to be 40%. The r/t ratio for 0.875 OD x 0.050 inch wall thickness tubing is 8.25, therefore, IWB-3521.1 applies. In the past, Regulatory Guide 1.121 has been used to assess the operability of specific tube degradation modes. Since no active corrosion phenomena are occurring within the Turkey Point Model 44F steam generators, the current Technical Specification plugging limit is considered acceptable, and a plant specific Regulatory Guide 1.121 analysis is not necessary.

3. **"Is FPL considering any additional surveillances to monitor for changes in steam generator degradation (wear, cracking, etc.) as a result of the proposed thermal power uprate for the Turkey Point Units."**

   FPL Response

   FPL has conducted extensive steam generator inspections which exceeded Technical Specification requirements in each of the past 3 refueling outages (prior to 1996) at Turkey Point Units 3 & 4[1]. These inspections included full length bobbin coil inspection for 100% of active tubes and sampling of manufacturing anomalies which affect a limited number of tubes in each steam generator with motorized rotating pancake coil (MRPC) techniques. No corrosion-related damage has been reported in any of these inspections.

   Manufacturing anomalies include minor denting at support intersections and minor overexpansion of the tube expansion transition at the top of the tubesheet. The basis for MRPC sampling of manufacturing anomalies is that anomalous conditions make the affected locations more susceptible to inter-granular attack/stress corrosion cracking (IGA/SCC) than tubes without the

[1] As a result of favorable past inspection results, no inspection was performed during the 3/96 refueling outage at Turkey Point Unit 4.

conditions. The extent of sampling in the last three full inspections was provided in letter L-95-175 dated June 22, 1995 as part of our response to NRC Generic Letter 95-03, "Circumferential Cracking of Steam Generator Tubes".

Future inspections at Turkey Point Unit 3 & 4, including methods, equipment, personnel training and qualification will be conducted in accordance with the examination protocol established in EPRI "PWR Steam Generator Tube Examination Guidelines". The requirements of this EPRI document are expected to be incorporated as part of the "Steam Generator Rule" which is expected to be published for comment later in 1996.

References

1. JPN-PTN-95-5009, Turkey Point Units 3 & 4 Thermal Power Uprate Project Non-LOCA Transient Input File No. PTP 425-18, January 10, 1995.

2. Friedland, A. J. and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A (Proprietary), WCAP-11397-A (Non-Proprietary), April 1989.

3. WCAP-11394-P-A, Methodology for the Analysis of the Dropped Rod Event, January 1990.

4. J. Kabadi, et al., "BART-A1: A Computer Code for the Best Estimate Analysis of Reflood Transients", WCAP-9561-P-A, Revision 1 (proprietary) and WCAP-9695-NP-A (non-proprietary), January 1980.

5. J. Kabadi, et al., "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code", WCAP-10266-P-A , Revision 2 (proprietary) and WCAP-11524-NP-A, Revision 2 (Non-Proprietary), March 1987; including Addendum 1-A "power Shape Sensitivity Studies" December 1987 and Addendum 2-A, "BASH Methodology Improvement and Reliability Enhancement", May 1988.

6. Meyer, P.E., "NOTRUMP - A Nodal Transient Small Break and General Network Code,"WCAP-10079-P-A, (proprietary) and WCAP-10080-NP-A (non-proprietary), August 1985.

7. Lee, N. et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (proprietary) and WCAP-10081-NP-A (non-proprietary), August 1985.

8. Thompson, C. M, et al., "Addendum to the Westinghouse Small Break LOCA Evaluation Model Using the NOTRUMP Code: Safety Injection Into the Broken Loop and the COSI Condensation Model", WCAP-10054-P, Addendum 2, Revision 1 (proprietary) and WCAP-10081-NP, Addendum 2, Revision 1 (non-proprietary), October 1995.

9. Shimeck, D. J., "COSI SI/Steam Condensation Experiment Analysis", WCAP-11767-P (proprietary), March 1988.

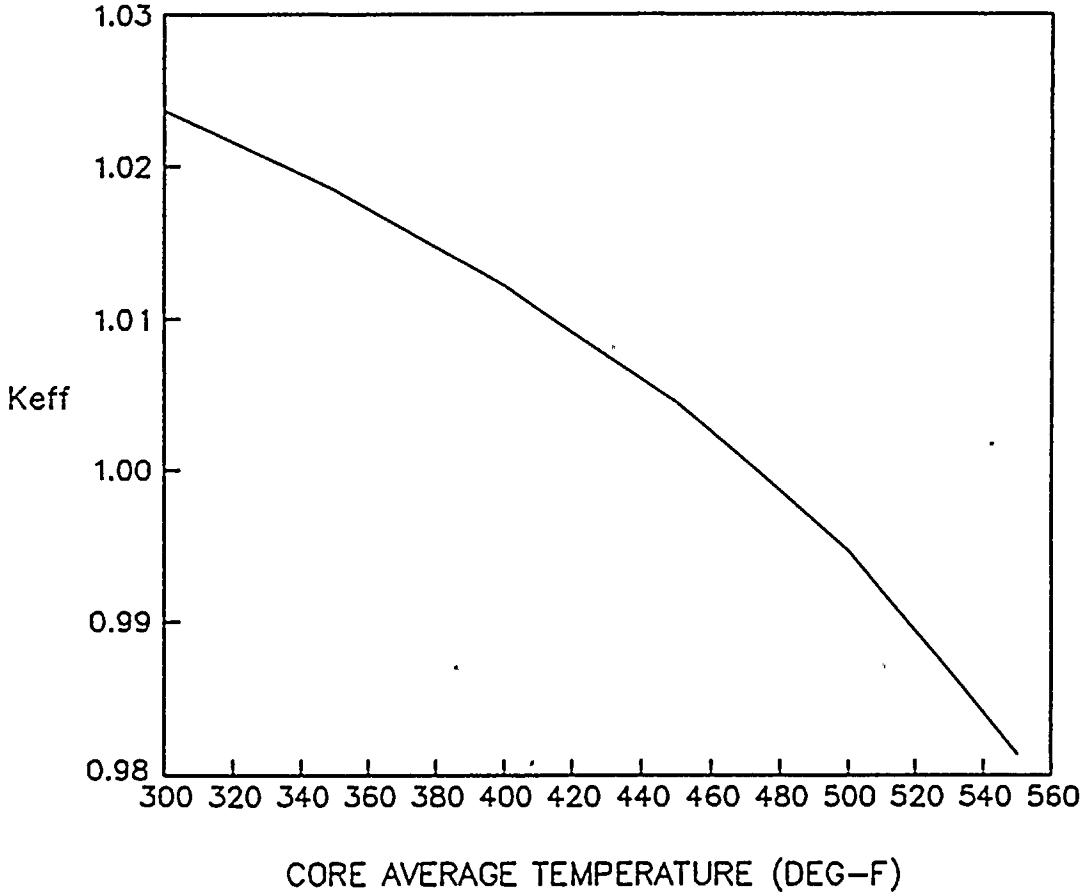
10. Bordelon, F. M., et al., "The Westinghouse ECCS Evaluation Model: Supplementary Information", WCAP-8471-P-A (proprietary) and WCAP-8472 (non-proprietary), January 1975.

11. Westinghouse Letter report MT/SMART/116(88), Reactor Vessel Heatup and Cooldown Limit Curves For Normal Operation, Nihar Ray,August 1988

12. BAW-2118P, Low Upper Shelf Toughness Fracture Analysis of Reactor Vessels of Turkey Point Units 3 and 4, K.K. Yoon, November,1991.

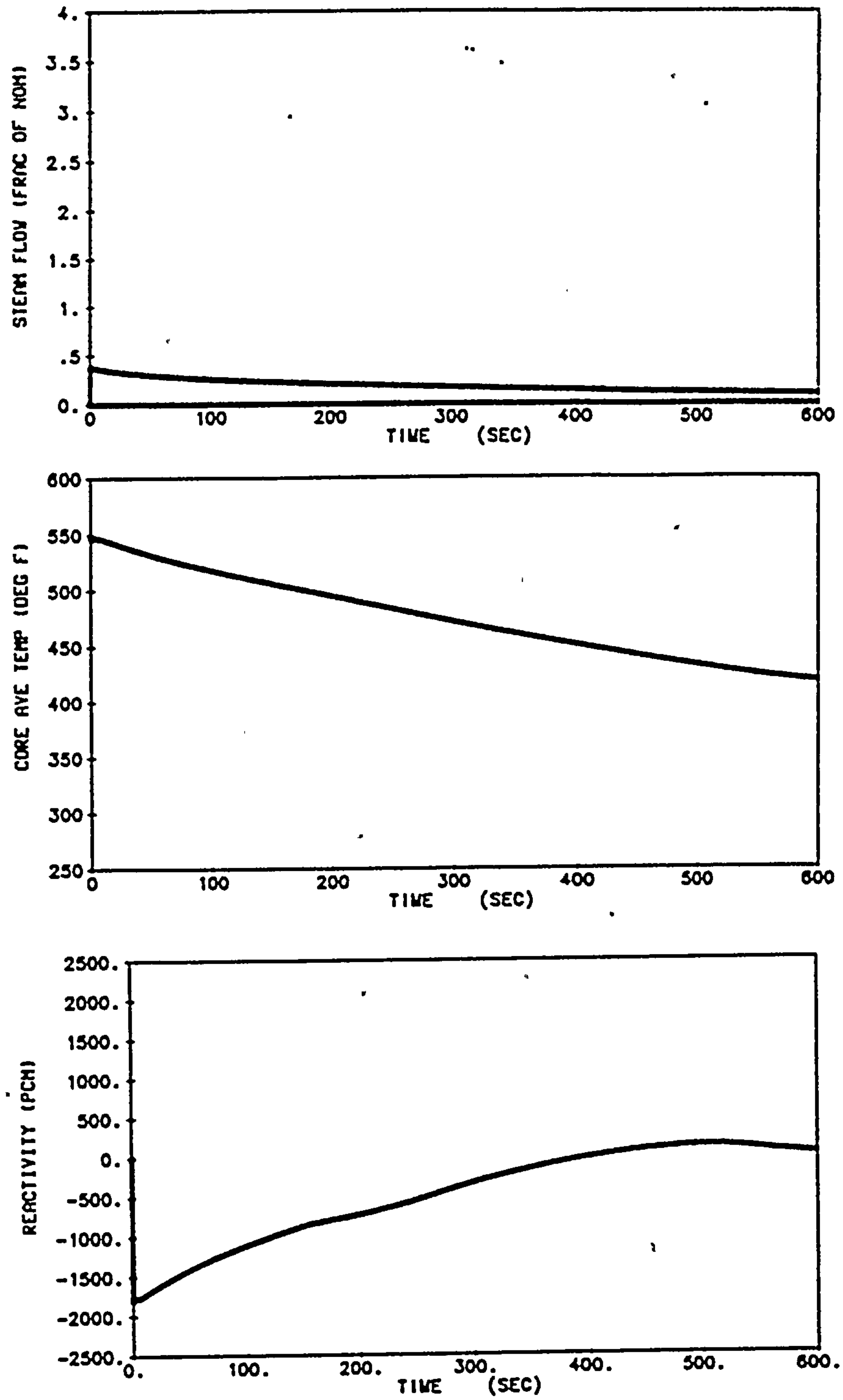
MAIN STEAM LINE BREAK

CORE RESPONSE CURVES



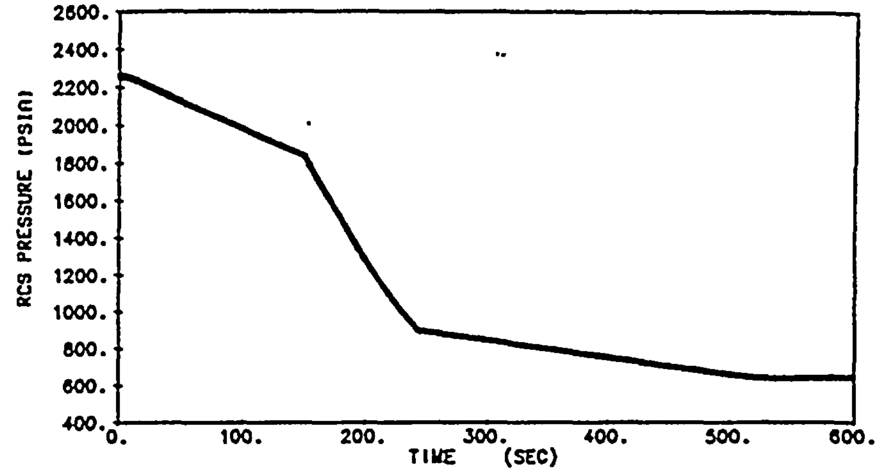
TURKEY POINT
UNITS 3 & 4

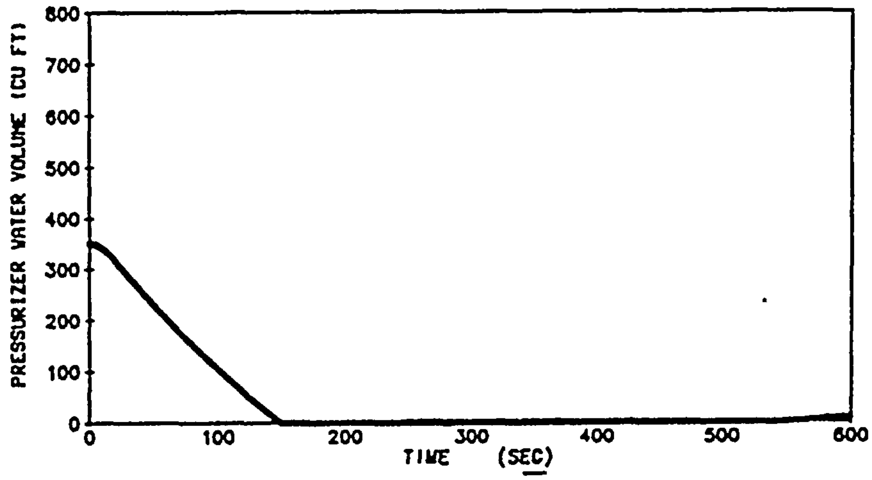
K_{eff} VS TEMPERATURE

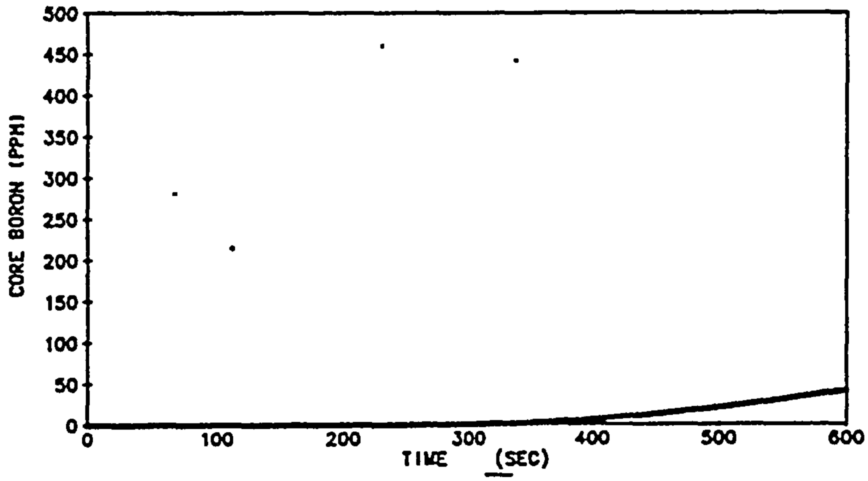


TURKEY POINT
UNITS 3 & 4

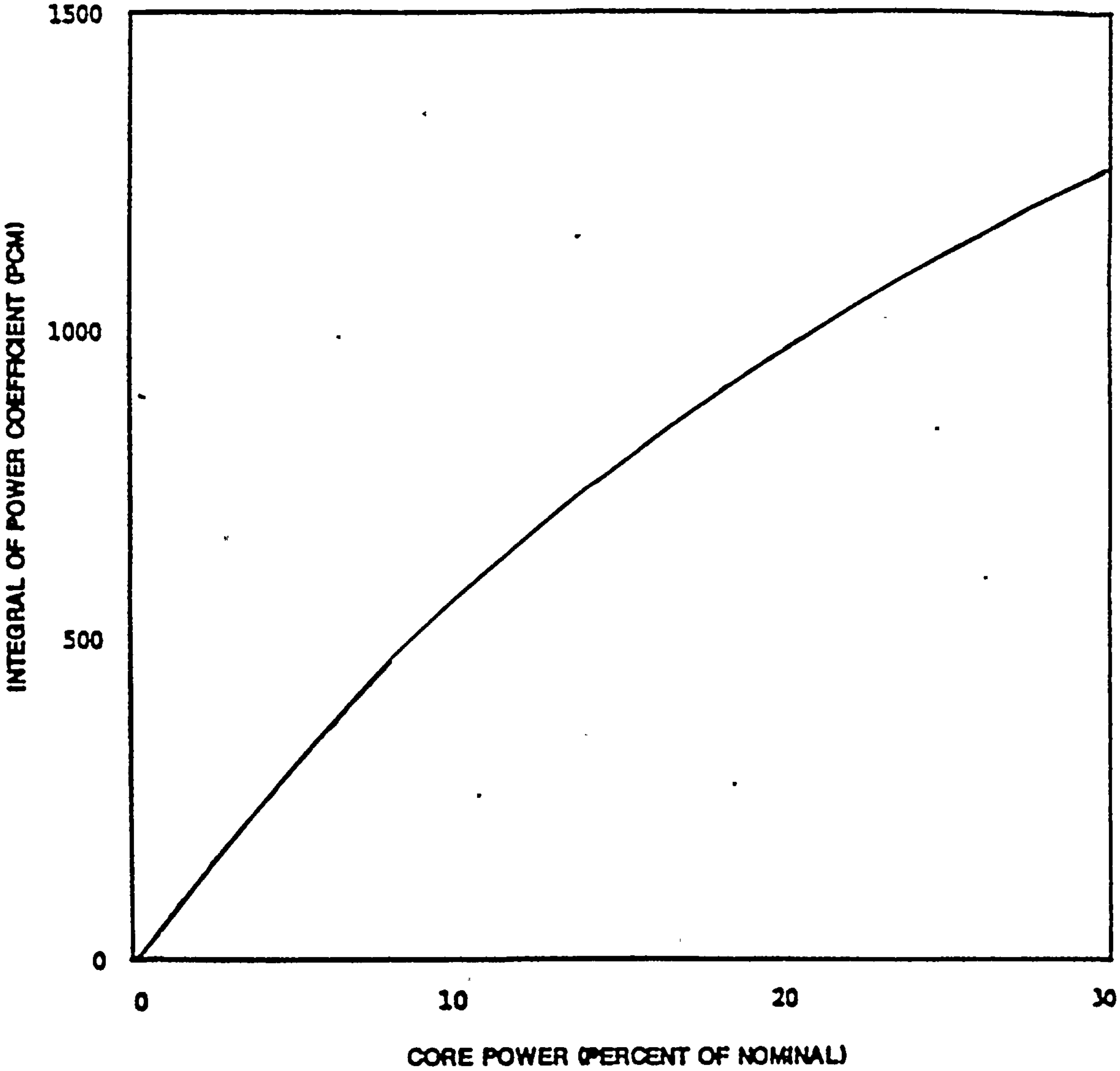
FAILURE OF STEAM GENERATOR
SAFETY OR RELIEF VALVE





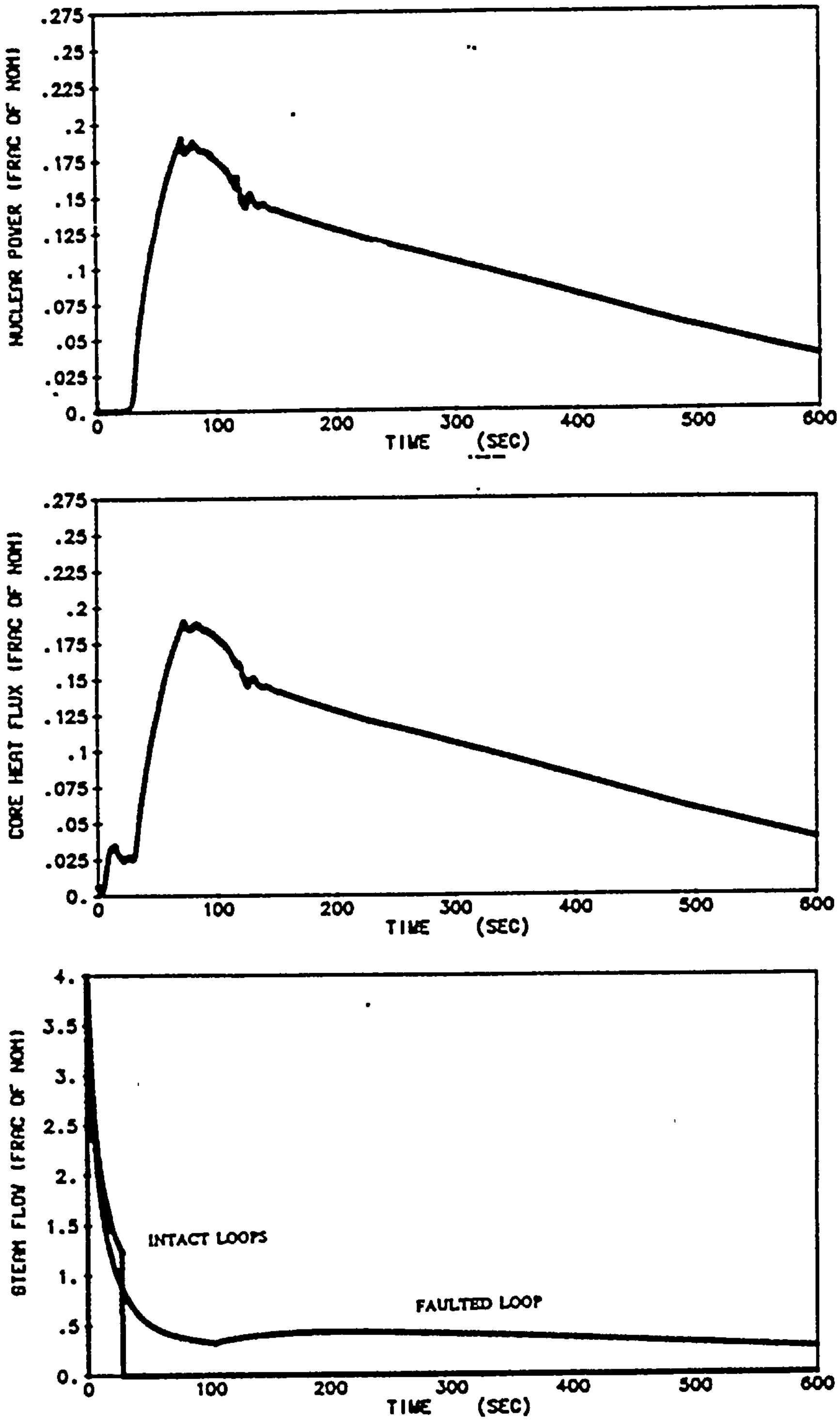


TURKEY POINT UNITS 3 & 4
FAILURE OF STEAM GENERATOR SAFETY OR RELIEF VALVE



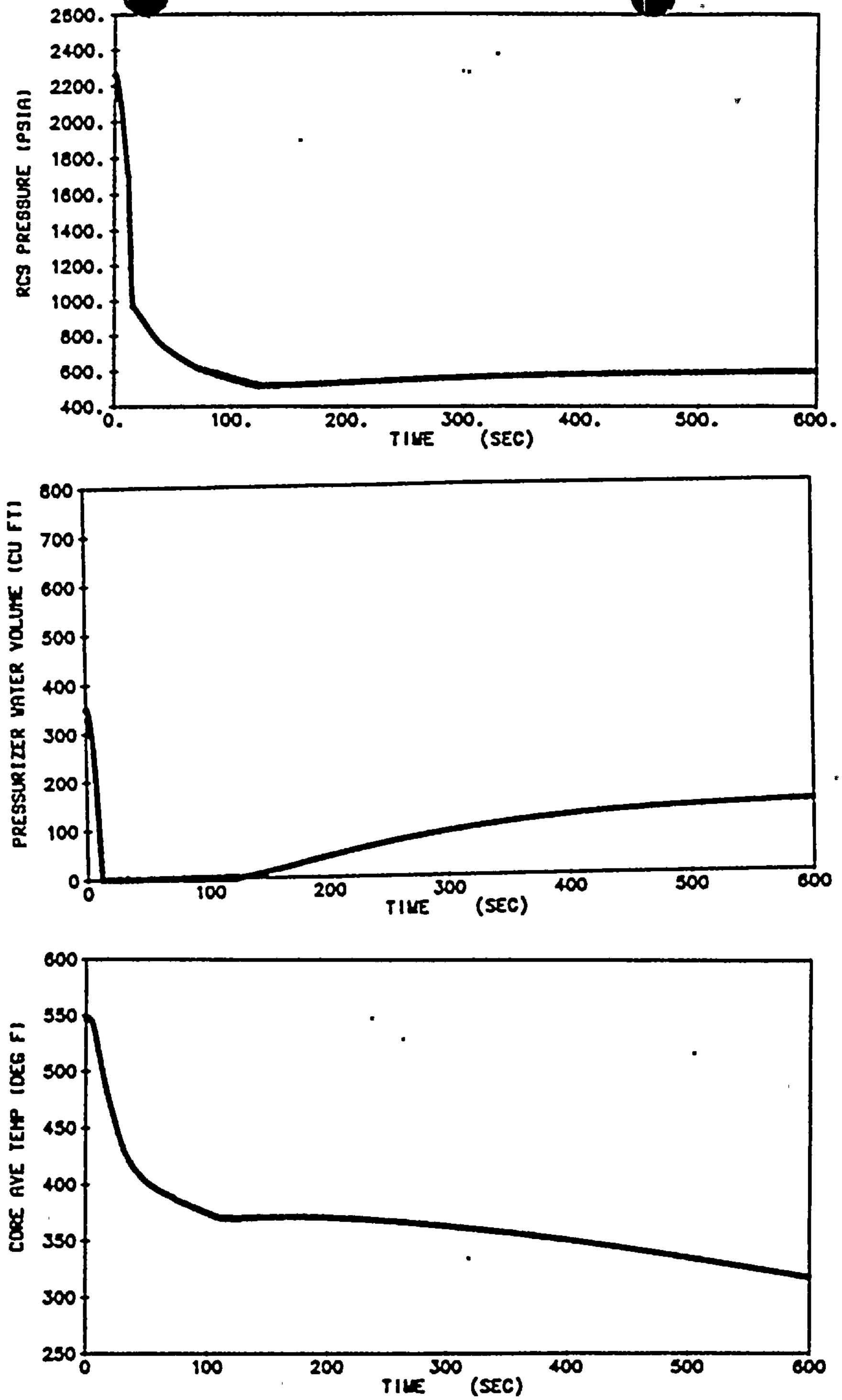
TURKEY POINT
UNITS 3 & 4

DOPPLER POWER FEEDBACK



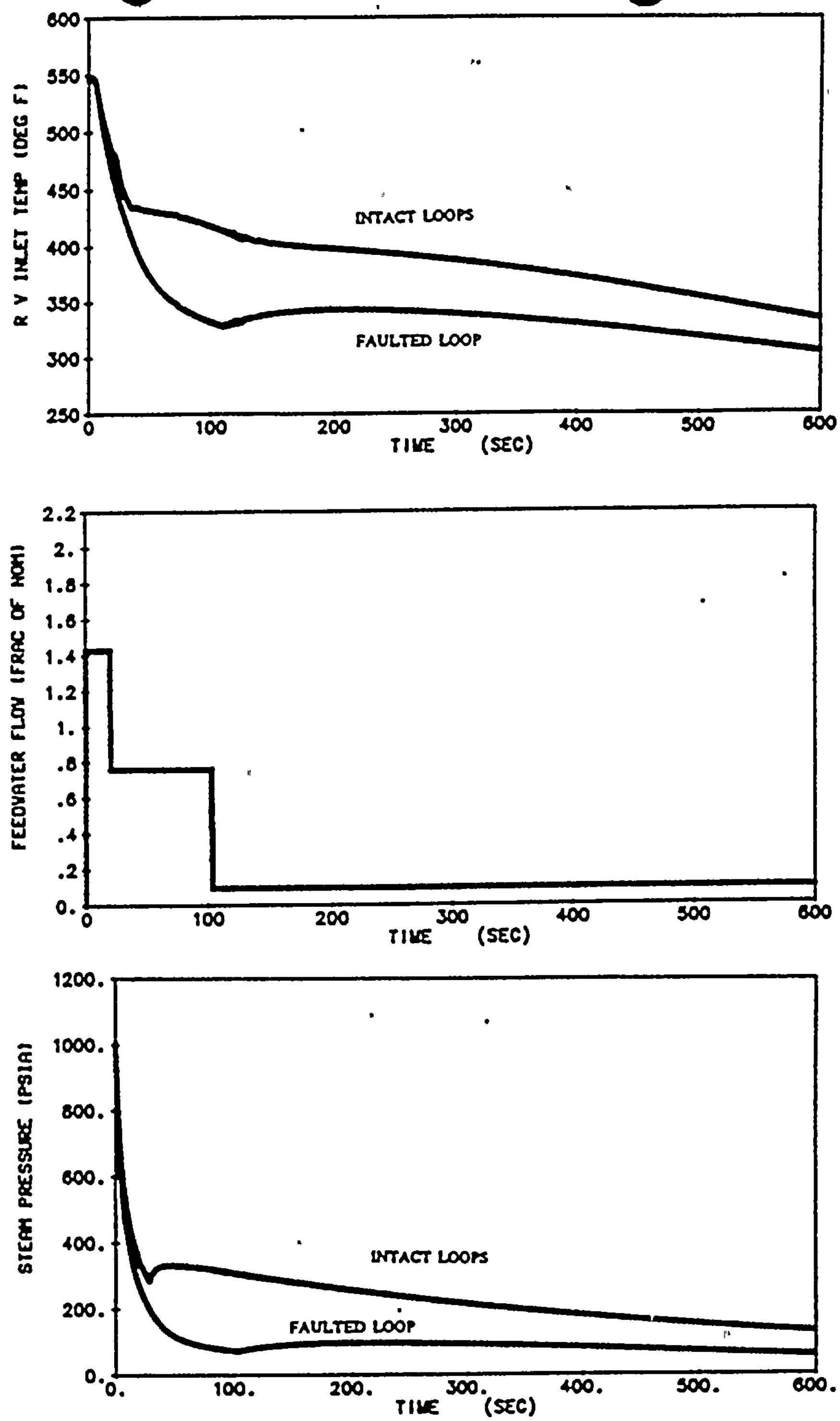
TURKEY POINT
UNITS 3 & 4

1.4 FT² STEAM LINE RUPTURE
OFFSITE POWER AVAILABLE



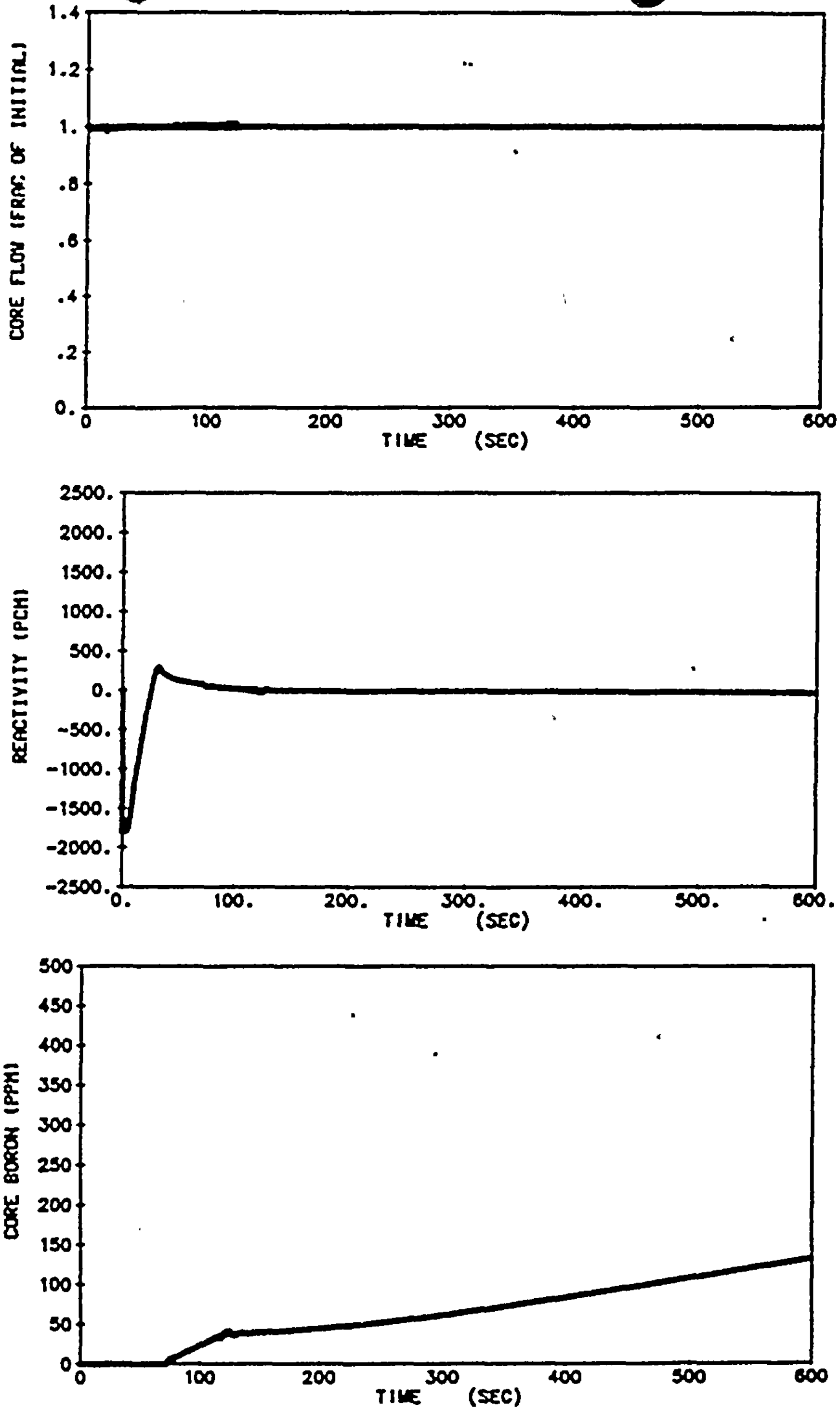
TURKEY POINT
UNITS 3 & 4

1.4 FT^2 STEAM LINE RUPTURE
OFFSITE POWER AVAILABLE



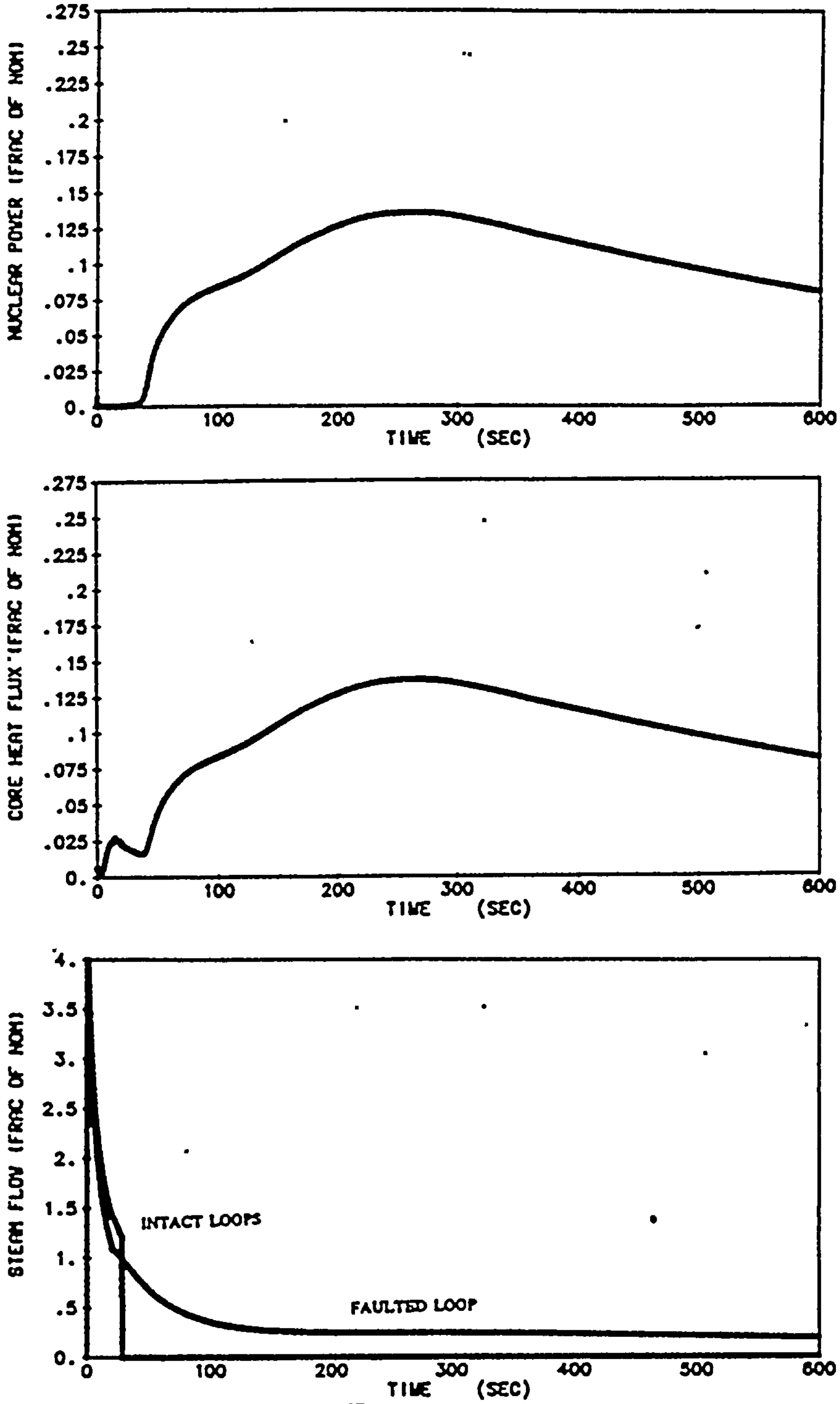
TURKEY POINT
UNITS 3 & 4

1.4 FT^2 STEAM LINE RUPTURE
OFFSITE POWER AVAILABLE



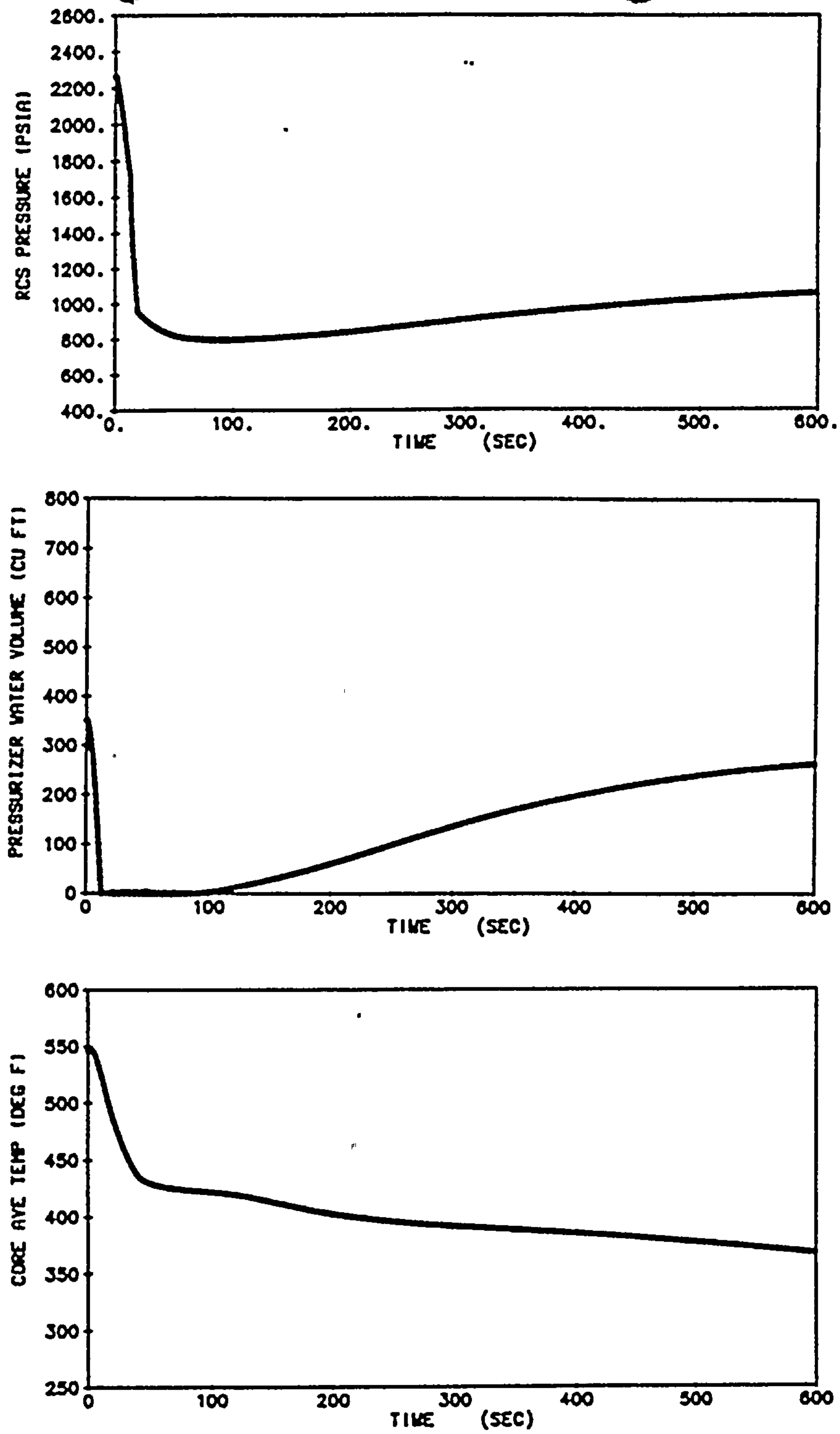
TURKEY POINT
UNITS 3 & 4

1.4 FT² STEAM LINE RUPTURE
OFFSITE POWER AVAILABLE



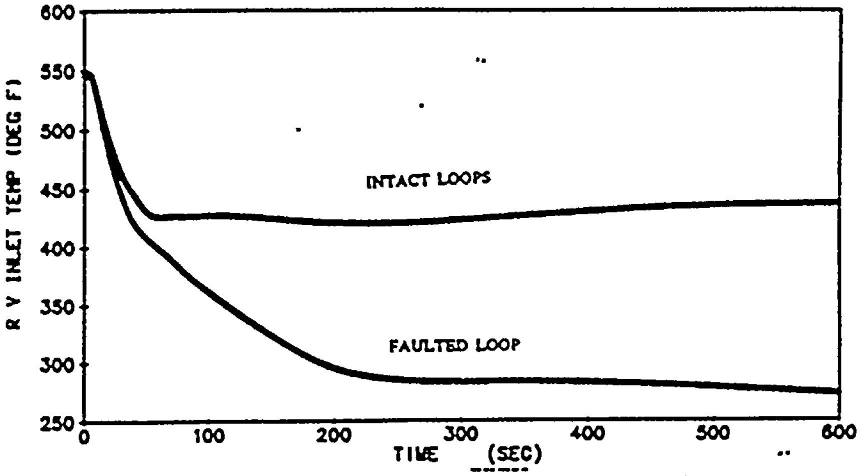
TURKEY POINT
UNITS 3 & 4

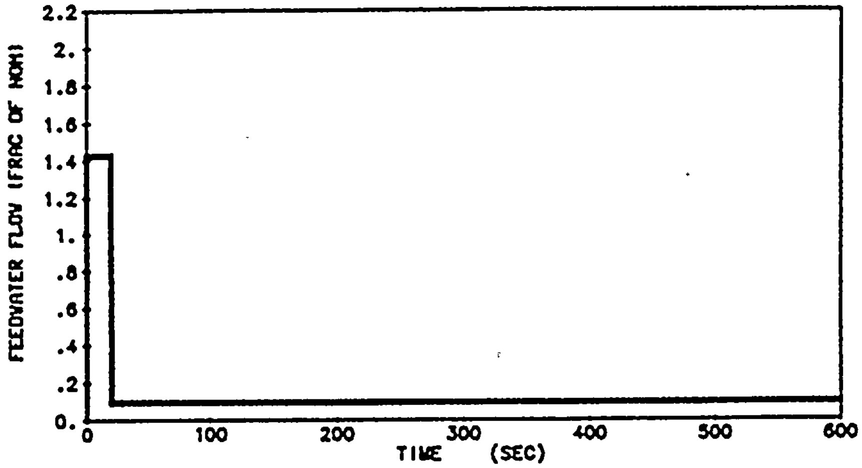
1.4 FT² STEAM LINE RUPTURE
OFFSITE POWER NOT AVAILABLE

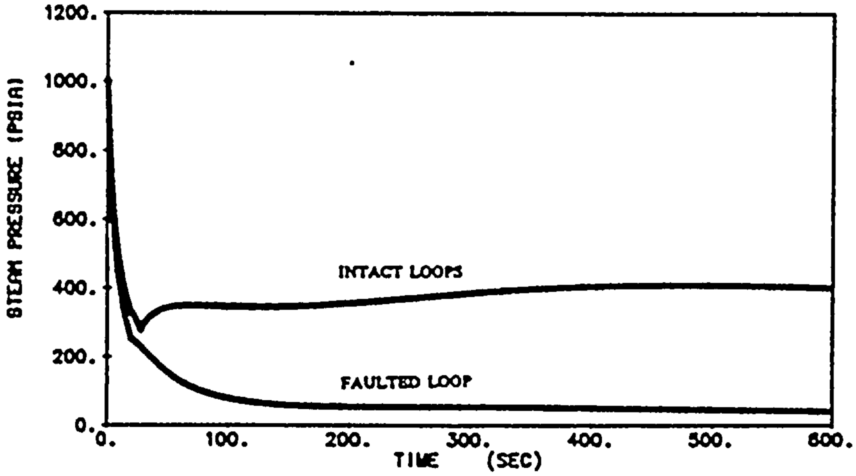


TURKEY POINT
UNITS 3 & 4

1.4 FT^2 STEAM LINE RUPTURE
OFFSITE POWER NOT AVAILABLE

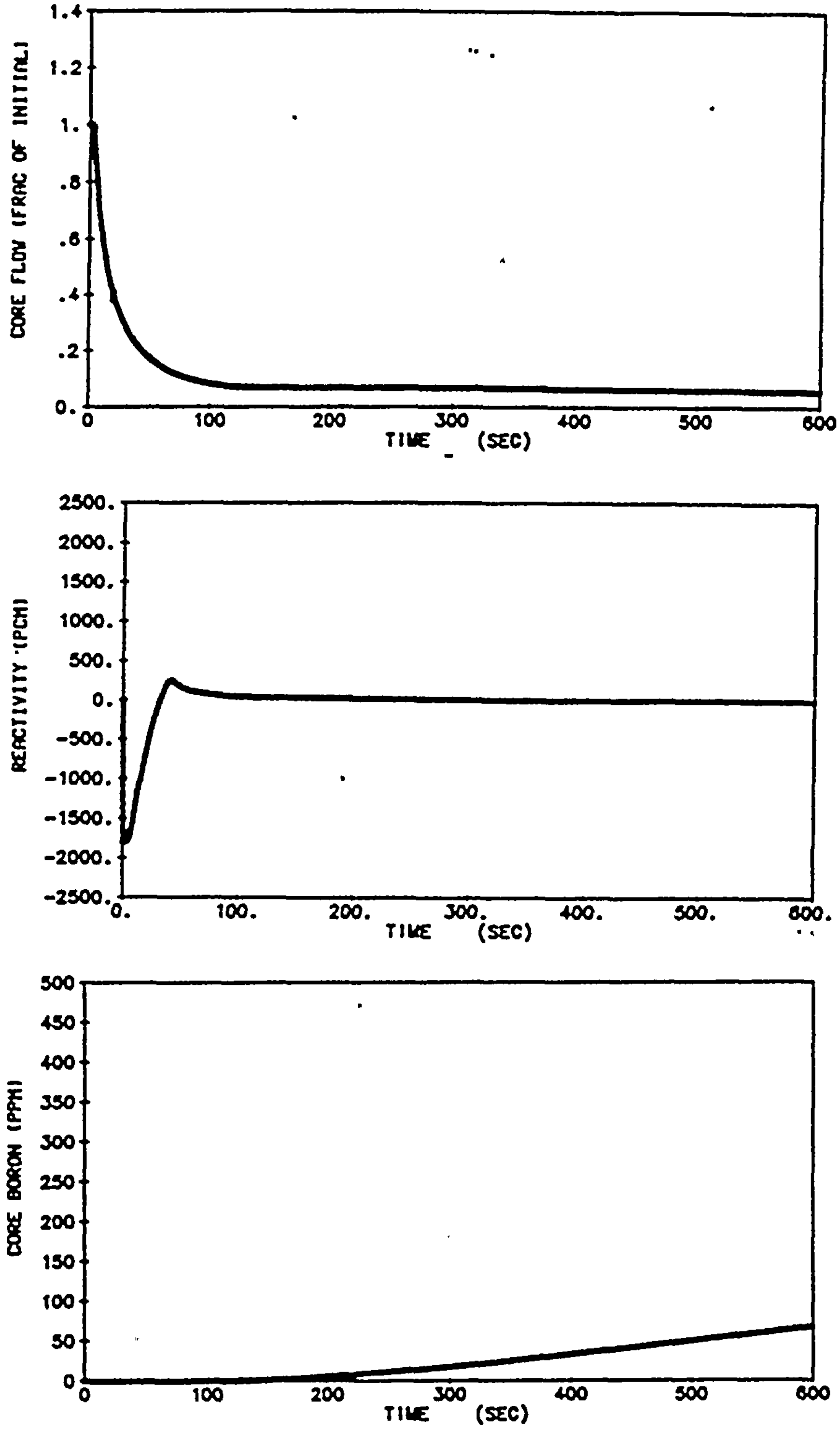






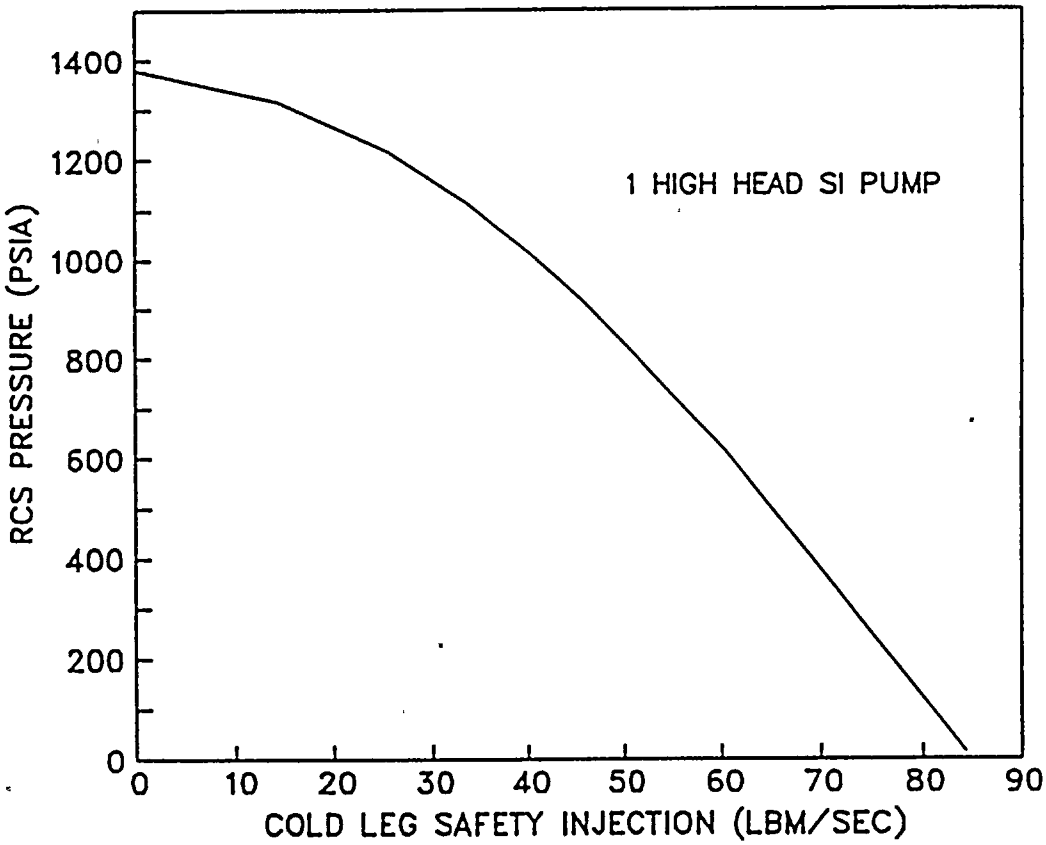
TURKEY POINT
UNITS 3 & 4

1.4 FT^2 STEAM LINE RUPTURE
OFFSITE POWER NOT AVAILABLE



TURKEY POINT
UNITS 3 & 4

1.4 FT^2 STEAM LINE RUPTURE
OFFSITE POWER NOT AVAILABLE



TURKEY POINT UNITS 3 & 4
STEAM LINE BREAK SAFETY INJECTION FLOW