PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK PROJECT

SAFETY EVALUATION

No. PSE-SE-Z-024

TITLE: DELETION OF TIP UNCERTAINTY, TEST NUMBER 16

Date: NOV 4 1985

1.0 PURPOSE

The purpose of this Safety Evaluation is to document the results of the evaluation performed to ensure that the deletion of Test Number 16, TIP Uncertainty, will not adversely affect reactor safety.

2.0 SCOPE

The scope of this Safety Evaluation is the adequacy of Hope Creek's power ascension test program as it concerns the testing of the traversing incore probe (TIP) system.

3.0 REFERENCES

1. Regulatory Guide 1.68, Revision 2, August 1978
2. Hope Creek Final Safety Analysis Report (FSAR), Chapter 14
3. General Electric Startup Test Specification, 23A4137, Revision 0

4.0 DISCUSSION

Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraph 5.y requires that the incore neutron flux instrumentation be calibrated as necessary and proper operation verified. The ability of the TIP system to obtain flux traces is demonstrated during power ascension testing of the process computer. Test Number 16, TIP Uncertainty, determines the uncertainty of TIP system readings at several reactor power/flow conditions. It is proposed that Test Number 16 be deleted.

Total TIP system uncertainty is composed of a geometric component and a random noise component. The geometric

component is due to off-center placement of the TIP tube within the LPRM instrument tube, bowing of the instrument tube, and water gap dimensional variations.
The random noise component is due to electronic noise in the TIP circuitry and boiling noise in the reactor.
Total TIP uncertainty is obtained directly in Test Number 16 by comparing TIP traces taken at symmetric core positions. The random noise component is measured by making repeated TIP runs at the common instrument tube location with each detector. The geometric component is calculated by statistically subtracting the random noise component from the total TIP uncertainty.

Measurements of TIP uncertainty during power ascension testing at previous plants have always been well below the acceptance criterion of 6.0%. TIP uncertainty data from several recent plant startups (Attachment 1) illustrate this point. This data includes results from two different types of TIP detectors. Specifically, data from Leibstadt is from a gamma TIP detector. All other data is from a thermal neutron TIP detector. The average values of geometric, random noise, and total TIP uncertainty from these plants are 1.85%, 1.02%, and 2.17%, respectively. Only one plant measured a total uncertainty of greater than 3.2% (Kuosheng, at Test Condition 3 which was subsequently reduced after correcting alignment errors in TIP axial positioning) and the highest total uncertainty measured at 100% power (Test Condition 6) was 2.65%.

Results from special tests of gamma TIP detectors at Edwin I. Hatch Nuclear plant (Attachment 2) indicate that use of the gamma TIP detectors reduced the core average nodal power asymmetry (which is TIP uncertainty plus actual core flux asymmetry) by about 33%. Hope Creek will be installing gamma TIP detectors. At other plants which have installed prototype or pilot production gamma TIP systems, the reduction was between 11% and 56%. Improvements in the minimum critical power ratio of 5% are typical following gamma TIP installation.

5.0 CONCLUSION

Because total TIP uncertainty at plants using thermal neutron TIP detectors has always been well below the acceptance criterion of 6% and because Hope Creek will use gamma TIP detectors which further reduce TIP uncertainty, Test Number 16, TIP Uncertainty, can be deleted. This will not adversely affect any safety system or the safe operation of the plant. An unreviewed safety question does not exist and no changes to the Technical Specifications are required.

6.0 DOCUMENTS GENERATED

None

7.0 RECOMMENDATIONS

Revision to Hope Creek's FSAR and startup test procedures shall be made to reflect the deletion of Test Number 16, TIP Uncertainty, as discussed above.

8.0 ATTACHMENTS

1. TIP Uncertainty Startup Data
2. EPRI Report NP-540, Special TIP Detector Measurements at Edwin I. Hatch Nuclear Plant, Unit 1, Prior to End of Cycle 1

9.0 SIGNATURES

Originator Philip O. Gopal PE 11/4/85
Date

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Date

Group Head (or SSE) SPSingh 11/4/85
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Site Engineering Manager CW Churchman /RJD 11/4/85
Date

ATTACHMENT 1

TABLE 1 - TIP UNCERTAINTY STARTUP DATA

	TIP UNCERTAINTY (%)		
	GEOMETRIC	RANDOM	TOTAL
HANFORD-2, TC3	2.87	1.42	3.20
HANFORD-2, TC6	1.80	1.43	2.30
LASALLE-1, TC3	1.24	1.18	1.71
LASALLE-1, TC6	2.16	1.54	2.65
LEIBSTADT, TC3*	2.55	0.68	2.64
LEIBSTADT, TC6*	1.57	0.83	1.78
FUKUSHIMA-6, TC3	1.50	1.00	1.80
FUKUSHIMA-6, TC6	1.30	1.10	1.70
CHINSHAN-1, TC3	2.51	1.21	2.79
CHINSHAN-1, TC6	2.40	0.61	2.48
CHINSHAN-2, TC3	1.20	0.88	1.49
CHINSHAN-2, TC6	0.98	0.59	1.14
CAORSO, TC2 (25% POWER)	1.40	1.14	1.81
CAORSO, TC2 (43% POWER)	1.60	0.98	1.88
CAORSO, TC3 (53% POWER)	1.97	0.95	2.19
CAORSO, TC3 (49% POWER)	1.37	1.10	1.76
CAORSO, TC6 (97% POWER)	2.29	0.73	2.40
CAORSO, TC6 (97% POWER)	2.21	0.94	2.40
KUOSHENG-1, TC3**	4.80	0.78	4.86
KUOSHENG-1, TC6	2.17	0.86	2.33
SUSQUEHANNA-1, TC3	0.78	1.46	1.66
SUSQHEHANNA-1, TC6	1.08	1.08	1.53
SUSQHEHANNA-1, TC6	0.90	1.08	1.41
AVERAGE	1.85	1.02	2.17

* Leibstadt has gamma TIP detectors, all other plants have Thermal Neutron detectors.

** TIP axial positioning was incorrectly aligned.

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ATTACHMENT 2

SPECIAL TIP DETECTOR MEASUREMENTS AT EDWIN I. HATCH NUCLEAR PLANT UNIT 1 PRIOR TO END OF CYCLE 1

EPRI NP-540
(Research Project 130-3)
Final Report
September 1977

Prepared by

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FOREWORD

Power distributions in operating boiling water reactors (BWR's) are calculated by the process computer using information from traversing in-core probes (TIP's). The current TIP system design consists of ionization chambers sensitive to thermal neutron fissions. These detectors indicate power asymmetries for core locations where the actual power distribution is thought to be symmetric. The indicated asymmetries can be attributed to sensitivity of the detector response to water gap variations and detector positioning. These indicated asymmetries can result in conservative thermal-hydraulic limits which tend to reduce reactor operating flexibility.

With the encouragement and cooperation of Georgia Power Company and Southern Company Services, a cooperative research effort was developed by General Electric Company and EPRI as an extension of RP 130, Nuclear Reactor Core Benchmark Data. *To carry out this research effort, a series of measurements was performed at the Hatch 1 Nuclear Power Plant prior to the end of Cycle 1 and during the refueling outage which followed. The measurements consisted of: 1. tests with three different types of traversing in-core probes, and 2. gamma scans to determine both fuel rod and bundle power distributions at the end of Cycle 1. In the measurements with the various TIP's, experiments were conducted with both fast neutron and gamma sensitive detectors to see if either of these would be less sensitive to geometric variations in the water gaps between BWR fuel bundles. The gamma scans were performed in order to obtain detailed bundle power measurements. These were needed to benchmark calculations with the TIP data collected from the three types of detectors. The gamma scan results were also needed to enlarge the data base for qualification of nuclear analysis methods.*

Similar gamma scans were conducted at Quad Cities 1 Nuclear Power Station in January 1976. These are reported in EPRI NP-214, Gamma Scan Measurements at Quad Cities Nuclear Power Station Unit 1 Following Cycle 2. *As it approached the end of Cycle 2, the Quad Cities reactor was operated in an all-rods-out condition. The Hatch 1 core had a substantial inventory of control rods at the end of its first cycle. Hence, these data will provide a more severe test of the nuclear analysis methods.*

The measurements with the special gamma, thermal neutron, and fast neutron sensitive TIP's are reported herein. A second report (NP-511) will contain results from the fuel rod and bundle gamma scans. A third report (NP-561) will contain comparisons of bundle powers inferred from both the gamma sensitive TIP and thermal neutron sensitive TIP with those deduced from the gamma scan data. Finally, a fourth report (NP-562) will document core design and operating data for Hatch 1 during Cycle 1. This last report is intended for use by those who wish to model the Hatch 1 operating history for qualification as a test of nuclear analysis methods.

Robert N. Whitesel
EPRI Project Manager

ABSTRACT

This report presents results, conclusions, and discussion of a special TIP (Traversing In-Core Probe) test performed at the Hatch 1 reactor. The purpose of the test was to provide power distribution data to support resolution of the apparent thermal neutron TIP asymmetry problem and to provide detailed qualification data for process computer programs and BWR core analysis methods.

Full-core power distribution data were obtained using three General Electric test detectors, i.e., thermal neutron TIP (standard production TIP), gamma TIP, and a fast neutron TIP. Apparent asymmetries measured with the gamma TIP were a factor of two lower than asymmetries indicated by the thermal neutron TIP. Although the data analysis including gamma scan evaluation is not complete, the gamma detector appears to be a suitable replacement for the thermal neutron detector.

Data obtained for the fast neutron detector were extremely noisy and of limited usefulness in the analysis of this detector's performance. However, the fast neutron data did indicate both a higher asymmetry and a higher dependency on void fraction than the gamma TIP.

1. INTRODUCTION

Power distributions calculated by the process computer in operating boiling water reactors (BWR) using information from the thermal neutron traversing in-core probe (TIP) indicate asymmetries for symmetric (about the in-core axis of symmetry) core locations. The magnitudes of the asymmetries vary with each reactor and its operating history. Analytical and experimental investigations reveal that most of the indicated asymmetries are not real power asymmetries, but rather are incorrect readings caused by thermal neutron signal sensitivity to variations in water gap* thickness and detector positioning. Because the TIP asymmetry translates directly into a power distribution uncertainty, a reduction in TIP asymmetry will result in a reduction in power distribution uncertainty used in thermal limits evaluations.

A possible solution to the TIP asymmetry problem is to replace the thermal neutron TIP with an ionization chamber which is less sensitive to water gap thickness variations and in-core placement variations. Two possible TIP replacements that are less affected by geometric considerations are the gamma detector which responds to prompt and delayed gammas and the fast neutron detector which responds to fast neutrons. It should be noted that for the purposes of this report the detector sensitive to thermal neutrons will be referred to as "thermal TIP," the detector sensitive to gamma radiation will be called "gamma TIP," and the detector which responds to fast neutrons will be called "fast TIP."

The purpose of the Hatch 1 TIP test was to provide power distribution data to support resolution of the apparent thermal TIP asymmetry problem as well as to provide data for qualification of selected process computer models through comparison with the gamma scan measurements. Specifically, objectives of the Hatch 1 TIP test included the following:

1. Determine apparent asymmetries for each of the three test TIPs and provide asymmetry data for comparison with gamma scan results.

2. Establish a void dependency difference between the thermal TIP signal and each of the gamma and fast TIP signals for the Hatch 1 core.

3. Investigate the effects of delayed gammas on the response of the gamma detector to local and full-core changes in power.

4. Determine the reproducibility characteristic of the gamma and fast detectors compared with the thermal detector.

5. Evaluate the performance of the fast TIP in a BWR environment for a period of several days.

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*The TIP traverses the core in the interstitial region between fuel bundles, referred to here as the water gap.

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2. SUMMARY

The major result of the Hatch 1 TIP test program is the conclusion that the gamma sensitive detector is a potentially viable replacement for the thermal neutron detector to solve the apparent TIP asymmetry problem. Test data indicate the gamma signal has far less apparent asymmetry than either the thermal or the fast neutron TIP. Also, the gamma signal requires less correction to nodal powers than the thermal neutron signal. Conversely, the fast neutron TIP does not appear to be a suitable replacement for the thermal neutron detector due primarily to its strong void fraction dependency and lack of reduction in apparent asymmetry (compared to a thermal neutron TIP). Another potential problem with the fast neutron TIP is signal degradation with increased exposure. Although these problems might be solved through additional detailed design work on fast detector electronics, cabling, and coating material, there is no incentive to pursue fast detector development at this time. Therefore, future efforts, especially comparison of TIP data to gamma scan results, should be concentrated on confirmation of gamma detector performance.

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3. SUMMARY OF RESULTS

1. The gamma TIP indicated integral asymmetries which are about 1/2 the asymmetries indicated by the thermal or fast TIP. Average apparent integral asymmetries indicated by the gamma, thermal, and fast TIP's were 2.5, 6.8, and 7.3%, respectively. Average apparent nodal asymmetries indicated by the three test TIP's were 5.5, 8.2, and 12.2% respectively.

2. Fast TIP data were exceedingly noisy, i.e., signal-to-noise ratios of approximately 5 to 1. Fast TIP data set 3 was too noisy to be considered reliable for data analysis. Most of the noise is thought to have been caused by the unshielded detector cable being pulled through a rough TIP tube.

3. Signal-to-signal ratio analysis for the Hatch 1 core indicates that the thermal TIP signal can be related to the gamma TIP signal by the following correlation:

$$(TIP)_{thermal} = (TIP)_{gamma}\,(a + b\bar{v})$$

where $\bar{v}$ represents nodal four-bundle average void fraction

Coefficients for data sets 1, 2, and 3 are as follows.

Data Set	a	b
1	1.13	−0.368
2	1.11	−0.313
3	1.11	−0.325

4. The thermal TIP signal can also be related to the fast TIP signal as a function of void fraction. However, the thermal to fast relationship is much more dependent on void fraction than the thermal to gamma dependency. For example, the RMS difference between the thermal TIP and gamma TIP signals was about 9.6% while the RMS difference between the thermal TIP and fast TIP signals was about 31%.

5. The gamma TIP signal requires less correction to indicate nodal powers than either the thermal TIP signal or the fast TIP signal.

6. The gamma TIP detector is sensitive to both photons from fission products and prompt gammas from the fission reaction. As a consequence, the gamma TIP is not as responsive to instantaneous power variations as the thermal TIP. Gamma TIP measurements made during and following control rod movement indicate that after a local power change a steady-state signal is achieved within ~1-2 minutes. This response time, although slower than the thermal TIP, is sufficiently rapid to avoid any additional monitoring uncertainty.

7. The gamma TIP signal does not respond to neutron flux depressions caused by fuel rod spacers or LPRM's as well as the thermal TIP. Because these signal depressions are used for TIP axial alignment, a new axial alignment procedure would have to be developed before a gamma TIP system could be implemented at a BWR.

8. The gamma TIP signal has the same degree of reproducibility (measure of the TIP's ability to accurately duplicate an axial power distribution) as the thermal TIP signal and is more reproducible than the fast TIP signal. Reproducibility analysis indicates the standard deviations in the means for the gamma, thermal, and fast TIP's are 0.76, 0.66, and 6.2%, respectively. The large deviation for the fast TIP is due primarily to noise pickup by the unshielded TIP cable.

9. The fast TIP sensitivity decreased by about 15% during the 2-day period that the detector was left in the core.

8. CORE SYMMETRY ANALYSIS

Figures 13, 14, 15, and 16 present results of symmetry analyses performed on nodes 3 to 22. The bottom and top two nodes were excluded from analysis because of unreasonably accentuated errors due to steep signal gradients at core bottom and top. These nodes were omitted to differentiate between true asymmetries due to water gap variations and detector positioning, and slight shifts in axial alignment. Apparent integral TIP asymmetry for three data sets is as follows:

INTEGRAL

Detector	Data Set 1 (%)	Data Set 2 (%)	Data Set 3 (%)	Average (%)
Thermal	6.57	6.79	6.93	6.76
Gamma	2.77	2.64	2.33	2.64
Fast	9.92	4.65	—	7.28

Apparent nodal TIP asymmetry for three data sets is as follows:

NODAL

Detector	Data Set 1 (%)	Data Set 2 (%)	Data Set 3 (%)	Average (%)
Thermal	8.07	8.20	8.35	8.21
Gamma	6.48	6.78	6.32	6.53
Fast	12.30	12.12	—	12.21

Table 1 contains detailed results of core-average symmetry analysis.

As can be seen above, the gamma sigal indicated about 1/2 of the asymmetry indicated by the thermal neutron TIP. This is expected since the gamma detector output is less sensitive to water gap geometric variations than the thermal neutron detector. Actual Hatch 1 core asymmetries are being determined by fuel bundle gamma scan measurements; the results will be subsequently compared to the above analysis.

Asymmetries indicated by the fast TIP were about the same or even larger than asymmetries indicated by the thermal neutron TIP. The fast TIP was expected to produce lower asymmetries since it is less affected by water gap geometric considerations than the thermal TIP. The major cause of fast TIP asymmetries probably is variation in the detector cable noise levels. Reproducibility analysis shows relatively high uncertainty due to excessive noise. See Figure B-12 for a good example of an excessively noisy fast TIP signal. Fast neutron signal for location 36-21, data set 1, had an unexplained electronic shift during TIP traverse. Thus, symmetric analysis for this location is not included for this particular string.

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9. TIP SIGNAL REPRODUCIBILITY ANALYSIS

Three sets of data were taken to determine signal trace reproducibility. These data are presented in Figures B-10 through B-18. The reproducibility test involved repeated traversing of the same in-core location with the same probe to check the consistency of the TIP output. The first reproducibility data set (presented in Figures B-10 to B-12) was recorded on one graph per detector. Thus, the individual traces were not discernible for analysis. Note the high noise level and signal spikes on the fast TIP traces. Results of analysis for the second and third data sets (nodes 3 to 22) are as follows:

Detector	Reproducibility Test Set 2 (%)	Reproducibility Test Set 3 (%)	Average (%)
Thermal	0.68	0.64	0.66
Gamma	0.96	0.56	0.76
Fast	5.95	6.40	6.18

The numbers in the above table are the standard deviation in the means of the nodal differences between the input 6 traces and the average nodal value for the 6 traces.

GENERAL ELECTRIC COMPANY TECHNICAL ANALYSIS
HOPE CREEK GENERATING STATION

TEST NUMBER 16 - TIP UNCERTAINTY
JUSTIFY TEST DELETION

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2; August 1978), Appendix A, paragraph 5.y requires that the incore neutron flux instrumentation be calibrated as necessary and proper operation verified. The Traversing Incore Probe (TIP) system is one of several incore neutron/gamma flux instrumentation systems. It provides gross core power distribution information for several applications. TIP system operability is demonstrated during preoperational testing and during power ascension testing of the process computer. Test Number 16, TIP Uncertainty, determines the uncertainty of the TIP system readings. It is proposed to delete Test Number 16.

DISCUSSION:

TIP system operability is demonstrated during preoperational testing of the TIP hardware and electronics and during power ascension testing when the process computer undergoes the dynamic system test case. During the latter testing, the process computer program OD-1 is used in conjunction with the TIP system to provide information on the gross core power distribution. Test Number 16 is a separate test performed later in the power ascension test program. It provides a measure of the uncertainty in TIP system data.

Uncertainty in TIP indication effects the accuracy of LPRM calibrations, thermal limits calculations, operating recommendations, etc. For Test Number 16, the acceptance criterion states that total TIP uncertainty shall be less than 6.0%.

Total TIP uncertainty is comprised of geometric and random noise components. Geometric uncertainty results from the off-center placement of the TIP tube within the LPRM instrument tube, bowing of the instrument tube, and water gap dimensional variations. These geometric differences cause the thermal neutron TIP detectors to indicate flux levels different from the values ideally obtained by an axial scan down the center of the water gap. A measure of this uncertainty is obtained by comparing data from symmetric TIP locations and correcting for random noise uncertainty.

Random noise uncertainty is caused by neutron, electronic and boiling noise in the reactor. This uncertainty is determined by comparing data from repetitive scans in the common instrument tube by each TIP detector.

Measurement of these uncertainties at the beginning-of-life of an initial core, during power ascension testing, provides the best measure of TIP uncertainty caused by these effects because the fuel bundle power asymmetry is at a minimum. Results from previous plant startups show that measured total TIP asymmetry has always been well below the acceptance criterion, 6%. Detailed analysis of 45 TIP sets from eight plants for power levels ranging from 18% to 100% and core flow from 33% to 105% showed that the average total TIP uncertainty was 3.8%. Results from more recent power ascension testing of 8 plants, summarized in Table 1, show that the average values of the geometric uncertainty, random noise uncertainty and total TIP uncertainty were 1.85%, 1.02%, and 2.17% respectively.

Geometric uncertainty has been reduced at plants which have replaced the thermal neutron TIP detector with a gamma flux sensitive detector (the type which will be used at Hope Creek Generating Station). Gamma TIP detectors sense the gamma flux in the water gap between fuel bundles and because of the relative insensitivity of gamma flux to water content between the fuel bundles and detector, the variation of TIP signals resulting from TIP tube orientation and water gap geometry differences is minimal.

Relative performance measurements at one plant near the end of cycle 1 show that the gamma TIP detector reduced the core average total power asymmetry (total TIP uncertainty plus bundle power asymmetry) by approximatley 33% (Reference 1). At other plants which have installed prototype or pilot production gamma TIP systems, the reduction was 11% to 56%. Five percent improvements in the minimum critical power ratio (MCPR) are typical following gamma TIP installation (Reference 2). It is expected that Hope Creek Generating Station will have similar results.

CONCLUSION:

Based on the test results from previous plant startups, TIP uncertainty for the Hope Creek Generating Station is expected to be much less than the limiting value of 6%. In addition, the geometric component of TIP uncertainty will be reduced by the use of gamma sensitive TIP detectors. TIP system operability will be demonstrated during preoperational testing of the TIP hardware and electronics and during power ascension testing of the process computer. In view of these considerations, it is concluded that deletion of Test Number 16, TIP Uncertainty, does not adversely affect any safety related systems or the safe operation of the plant and as such does not involve an unreviewed safety question.

REFERENCES:

1. K. W. Burke, "Special TIP Detector Measurements at Edwin I. Hatch Nuclear Plant Unit 1 Prior to End of Cycle 1," Electric Power Research Institute (EPRI), September 1977 (EPRI NP-540).

2. Station Nuclear Engineering Manual," General Electric Company, September 1983 (NEDO-24810B).

ATTACHMENT 1

TABLE 1 - TIP UNCERTAINTY STARTUP DATA

	TIP UNCERTAINTY (%)		
	GEOMETRIC	RANDOM	TOTAL
HANFORD-2, TC3	2.87	1.42	3.20
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LASALLE-1, TC6	2.16	1.54	2.65
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FUKUSHIMA-6, TC3	1.50	1.00	1.80
FUKUSHIMA-6, TC6	1.30	1.10	1.70
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CHINSHAN-1, TC6	2.40	0.61	2.48
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CHINSHAN-2, TC6	0.98	0.59	1.14
CAORSO, TC2 (25% POWER)	1.40	1.14	1.81
CAORSO, TC2 (43% POWER)	1.60	0.98	1.88
CAORSO, TC3 (53% POWER)	1.97	0.95	2.19
CAORSO, TC3 (49% POWER)	1.37	1.10	1.76
CAORSO, TC6 (97% POWER)	2.29	0.73	2.40
CAORSO, TC6 (97% POWER)	2.21	0.94	2.40
KUOSHENG-1, TC3**	4.80	0.78	4.86
KUOSHENG-1, TC6	2.17	0.86	2.33
SUSQUEHANNA-1, TC3	0.78	1.46	1.66
SUSQHEHANNA-1, TC6	1.08	1.03	1.53
SUSQHEHANNA-1, TC6	0.90	1.08	1.41
AVERAGE	1.85	1.02	2.17

* Leibstadt has gamma TIP detectors, all other plants have Thermal Neutron detectors.

** TIP axial positioning was incorrectly aligned.

ATTACHMENT 2

TEST NO. (22)	TEST NAME	OPEN VESSEL	HEAT UP	1	2	3	4	5	6	WARRANTY
1	Chemical and Radiochemical	X	X	X		X		X	X	
2	Radiation Measurement	X	X	X		X			X	
3	Fuel Loading	X								
4	Full Core Shutdown Margin	X								
5	Control Rod Drive	X	X	X[2]	X[2]	X[2]			X[2]	
6	SRM Performance	X								
8	IRM Performance		X	X						
9	LPRM Calibration		X	X		X			X	
10	APRM Calibration		X	X	X	X		X	X	
11	Process Computer	X	X	X[3]		X		X		
12	RCIC		X	X						
13	HPCI		X			X				
14	Selected Process Temp		X			X	X[4]		X[4]	
14	Water Level Ref Leg Temp		X			X			X	
15	System Expansion	X	X	X		X			X	
~~16~~	~~TIP Uncertainty~~					~~X~~			~~X~~	
17	Core Performance			X	X	X	X	X	X	X
18	Steam Production									X
19	Core Pwr-Void Mode Response						X	X		
20	Pressure Regulator			X	X	X	X	X	X	
21	Feed Sys-Setpoint Changes		X	X	X	X	X	X	X	
21	Feed Sys-Loss FW Heating								X[5]	
21	Feedwater Pump Trip								X[6]	
21	Max FW Runout Capability								X[7]	
22	Turbine Valve Surveillance					X[8]		X[9]	X[10]	
23	MSIV Functional Test		X	X[11]	X[12]			X[13]		
23	MSIV Full Isolation								X	
24	Relief Valves		X	X[20]	X	X[20]			X[20]	
25	Turbine Trip & Load Rejection				X[15]	X[16]			X[17]	
26	Shutdown Outside CRC				X					
27	Recirculation Flow Control				X[14]			X[18]		
28	Recirc-One Pump Trip					X			X	
28	RPT Trip-Two Pumps					X[19]				
28	Recirc System Performance				X	X	X		X	
28	Recirc Pump Runback					X				
28	Recirc Sys Cavitation					X				
30	Loss of Offsite Pwr			X						
31	Pipe Vibration		X	X	X	X			X	
29	Recirc Flow Calibration					X			X	
32	RWCU		X							
33	RHR				X				X[21]	
34	Drywell & Steam Tunnel Cooling		X	X		X			X	
35	Gaseous Radwaste			X		X			X	
38	SACS Performance					X			X	
40	Confirmatory In-Plant Test				X					

FSAR 3/7

(1) Test conditions refer to plant conditions on Figure 14.2-4

(2) Perform Test 5, timing of 4 slowest control rods, in conjunction with expected scrams

(3) Dynamic System Test Case to be completed between test conditions 1 and 3

(4) After recirculation pump trips (natural circulation)

(5) Between 80 and 90 percent thermal power, and near 100 percent core flow

(6) Max FW Runout Capability & Recirc Pump Runback must have already been completed

(7) Reactor power between 80 and 90 percent

(8) Reactor power between 45 and 65 percent

(9) Reactor power between 75 and 90 percent

(10) At maximum power that will not cause scram

(11) Perform between test conditions 1 and 3

(12) Reactor power between 40 and 55 percent

(13) Reactor power between 60 and 85 percent

(14) Between test conditions 2 and 3

(15) Generator load rejection, within bypass valve capacity

(16) Reactor power between 60 and 80 percent at core flow $\geq$ 95 percent - turbine trip

(17) Load rejection

(18) Between test conditions 5 and 6

(19) >50% power and $\geq$95 core flow, and performed before Turbine Trip & Load Rejection

(20) Check SRV set points during major scram tests

(21) Performed during cooldown from test condition 6

(22) The test number correlates to FSAR Section 14.2.12.3.x where x is the indicated test number.

HOPE CREEK GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

TEST SCHEDULE AND CONDITIONS

FIGURE 14.2-5

d. Acceptance Criteria

<u>Level 1</u>:

1. There shall be no evidence of blocking of the displacement of any system component caused by thermal expansion of the system.

2. Inspected hangers shall not be bottomed out or have the spring fully stretched.

3. The position of the shock suppressors shall be such as to allow adequate movement at operating temperature.

4. The piping displacements at the established transducer locations shall not exceed the limits specified by the piping designer, which are based on not exceeding ASME Section III Code stress values. These specified displacements will be used as acceptance criteria in the appropriate startup test procedures.

14.2.12.3.16 TIP Uncertainty

a. Objective

The test objective is to demonstrate the reproducibility of the TIP system readings.

b. Prerequisites

The core is at steady-state power level with equilibrium xenon, so as to require no rod motion or change in core flow to maintain power level during data acquisition by the TIP system.

c. Test Method

1. Core power distribution data are obtained during the power ascension test program. Axial power distribution data are obtained at each TIP location. At intermediate and higher power levels, several sets of TIP data are obtained to determine the overall TIP uncertainty.

2. TIP data are obtained with the reactor operating with a symmetric rod pattern and at steady-state conditions. The total TIP uncertainty for the test is calculated by averaging the total TIP uncertainty determined from each set of TIP data. The TIP uncertainty is made up of random noise and geometric components.

3. Core power symmetry is also calculated using the TIP data. Any asymmetry, as determined from the analysis, will be accounted for in the calculations for MCPR.

d. Acceptance Criteria

Level 2:

The total TIP uncertainty shall be within the specified limits required in the GE startup test specification.

14.2.12.3.17 Core Performance

a. Objective

The test objective is to evaluate the principal thermal and hydraulic parameters associated with core behavior.

b. Prerequisites

The plant is operating at a steady-state power level.