Nuclear Safety Assessment

Group Project Report No. 7-84

Investigation of Unit Two

Power Transient of May 28, 1984

Report Date - 6/6/84

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Manager-Nuclear Safety Assessment Group

FILE 917-1

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1.0 Summary

At about 0100 on 3/28/84 Unit Two reactor power increased to a level of about 5.9%. This exceeded the license limit of 5.0%. The power excursion lasted for less than three minutes. It was apparently caused by a malfunction of the Number One Turbine Bypass Valve. The transient occurred while the operators were establishing plant conditions for an approved test. Operator actions were prompt and effective. The reactor did not SCRAM. No emergency core cooling systems were actuated. No nuclear safety hazard existed.

2.0 Description of Incident

At the time of the incident power was being increased by withdrawing control rods. The objective was to achieve about 60% opening of the No. 1 Turbine Bypass Valve (TBV), which occurs at a power level of about 4%. This action would provide sufficient steam flow to permit testing of the RCIC system without perturbating plant pressure.

At a reactor power of about 3.8% a Low Dilution Flow alarm was received for the Off Gas Recombiner System. The operators observed that reactor power was increasing and reactor water level was fluctuating. They inserted control rods, took manual control of the feed water system, and then took the Unit Two Off Gas Recombiner System out of service. These actions terminated the transient. Power rose to a level of 5.9% on the highest indication (APRM B) and then returned to less than 4%.

The Duty Manager was informed. At 0159 the NRC was informed that a power excursion beyond the license limit had occurred.

After the transient had settled out the Duty Manager made the decision to continue testing. The RCIC tests were completed as scheduled. At 0530 a normal shutdown began. At 0600 it was discovered that No. 1 TBV would not close. It hung up at about 18% open. The Duty Manager was informed and trouble shooting of the TBV was commenced. The reactor was shutdown at 1330.

3.0 Discussion

The incident occurred during the course of establishing plant conditions for planned tests of the RCIC system. Power was well within the specified limits when the transient began. The transient was caused by equipment malfunction. Since Plant Staff was investigating the technical aspects of the transient in detail, the Nuclear Safety Assessment Group concentrated upon the programmatic issues. NSAG attempted to determine whether the plant was being operated prudently.

3.1 Power Monitoring was Correct

Power was being monitored using the six Average Power Range Monitors (APRM's), which were displayed on a CRT on panel 2C651. The gains of the APRM's had been adjusted to the highest possible values in order to lower the actual SCRAM set points for initial testing and to improve the indication at the low end of the scale. Gain settings ranged from 1.85 to 2.37. The APRM displays showed the actual power multiplied by the gain setting. The maximum reading observed during the transient was 11% read on APRM "B". The actual power, then, was 11% divided by the gain of 1.85 or 5.9%

The Technical Specifications require that during startup the APRM SCRAM be set at a maximum of 15% and the rod block be set at a maximum of 12% (Tables 2.2.1-1 and 3.3.6-2). The values actually set were: SCRAM at 14% and rod block at 11%. When one corrects for instrument gain, the SCRAM values would range between 7.57% and 5.91% and the rod blocks between 5.95% and 4.64%.

During the transient a rod block occurred but no SCRAM signals came in. This fixes the actual power between 4.64% and 5.91% (minimum rod block setting and minimum SCRAM setting).

The Intermediate Range Monitors (IRM's) were displayed on recorders on the Standby Information Panel. It is impossible to fix an accurate correlation between the IRM readings and core thermal power.

The APRM's were calibrated on 1/15/84. The IRM's were calibrated on 4/27/84 (Both are semi-annual requirements, Tech Specs Table 4.3.1.1-1). The Weekly Channel Functional Tests were done on 5/21/84 (APRM) and on 5/22/84 (IRM). The next tests were done 5/29/84 and 5/30/84 respectively. There is every reason to believe that the APRM and the IRM SCRAMs would have functioned if required.

The operators had been specifically directed to monitor power on the APRM's. A night order entry dated 5/21/84 reads,

> "Unit 2 APRM's used to determine 5% power limit. (Rated Temp and Press 75% 1 bypass.)"

The order is somewhat vague in that it does not specify that the power limit is actual power not indicated power. That is, it does not clearly state that the limit is the APRM reading divided by the instrument gain. However, this was understood. The instrument gains were posted on panel 2C651. The operator, and the startup engineer referenced the gain settings in their log entries. There was no confusion on the part of the operating crew. Monitoring power was not an issue in the incident.

At the time of the incident reactor power was being increased in preparation for testing the RCIC system. The test procedures called for power level greater than 2% with sufficient steam flow to

prevent reactor pressure decay during RCIC operation. The Unit Supervisor's goal was 60% opening on No. 1 TBV which corresponds to about 4% actual power. (APRM reading corrected for gain.) When the transient occurred APRM "B" reading was about 7% and No. 1 TBV was about 55% open. A 7% reading equates to 3.8% actual power. The plant was being operated conservatively. Three-point-eight percent is comfortably below the limit of 5%.

In summary:

- o The instruments were set in a conservative manner.
- o The instruments were in calibration and the required functional tests had been done.
- o SCRAM protection existed from the APRM's and the IRM's.
- o The operators were monitoring power in accordance with management's instructions.
- o The plant was operating at a conservative power level.

3.2 The Bypass System had been Properly Tested

The following tests were performed prior to the incident. They required proper response by the pressure regulator:

Pressure	Test	Description
135#	HF-293-030	Verify proper response of BPV's to pressure regulator setpoint changes.
150#	SO-250-003	RCIC Full Flow Test Steam flow to RCIC requires Pressure Regulator (P.R.) to close BPV slightly to maintain pressure. Initially No. 1 BPV about 0.5 open.
150#	SO-252-003	HPCI Full Flow Test Steam flow to HPCI requires P.R. to close BPV by ~30% to maintain pressure. Initially, #1 BPV-3/4 open.
150#	ST26.1	SRV Low Pressure Test Steam flow to S/RV requires P.R. to close ~1 BPV to maintain pressure. Initially, ~2 BPV's open.
920#	HF-250-010 ST14.1 SO-250-002	RCIC Functional Checks RCIC CST to CST RCIC Full Flow Test Steam Flow to RCIC requires P.R. to close ~5% of one BPV to maintain pressure.

920#	SO-252-002	HPCI Full Flow Test Steam flow to HPCI requires P.R. to close ~50% of one BPV to maintain pressure.
128-250# 920-950#	CRD Movement	Movement of CRD's to increase power to perform above test requires P.R. to open BPV's to maintain pressure.

In each of the above tests the performance of the system was monitored using the GETARS. The turbine bypass valves responded properly in every one of these tests.

On the night of 5/28/84 there was no reason to expect problems with the pressure control system or the bypass valves.

3.3 Incident Caused by Equipment Malfunction

At about 0100:30 the TBV's began to oscillate. At 0101 a Low Dilution Flow alarm was received on the Off Gas Recombiner Panel. The maximum power occurred at about 0101:45, and the transient was over by 0104.

Over the course of the previous twenty-four hours several flow oscillations had occurred in the Off Gas Recombiner System. It appeared at the time that the oscillations in the TBV's had been caused by the perturbation on the Off Gas System. The Off Gas System was taken out of service and vacuum was maintained by the mechanical vacuum pump. No further oscillations were observed.

However, during the subsequent reactor shut down TBV No. 1 could not be closed fully. Subsequent trouble shooting indicated that at least one TBV was operating sluggishly. Debris was found in the number one bypass valve.

NSAG did not attempt to determine the cause of the equipment malfunction. That is being done by the plant staff Technical Section. We are satisfied, however, that the transient was not caused by operator error. Clearly there was a malfunction of some kind which caused reactor pressure and feed water flow transients that resulted in minor power excursions.

3.4 Operator Response was Prompt and Effective

The first indication of a problem was the Low Dilution Flow alarm which occurred at 0101. At this time APRM "B" indicated that reactor power was about 7% (3.8% corrected for gain). NSAG could not determine precisely whether or not a control rod was being moved at the time the alarm appeared. At any rate, the operators observed that power was increasing, that the TBV's were oscillating and that reactor water level was fluctuating. The operators inserted control rods into the core and took manual control of feed water flow. Power increased to 11% on APRM B. Power turned at about 0101:45 and

the TBV oscillation died out by 0104. The operators then isolated the Off Gas Recombiner System and started the mechanical vacuum pump.

Within a period of less than four minutes the transient was over. The operators then proceeded to correct the apparent cause of the problem by securing steam to the air ejector system. There were no further symptoms until about 0600 when TBV No. 1 would not close during the plant shut down.

The operators recognized that a limit may have been exceeded. After conditions had stabilized they notified the Duty Manager and subsequently notified the NRC.

In the opinion of NSAG, the operators responded effectively. They recognized the problem, took steps to terminate the transient, corrected what they believed to be the cause, and informed the proper authorities.

3.5 Evolutions were Authorized

Tests HF-250-010, RCIC Turbine Control System Tune Up and ST14.2, Reactor Vessel Injection were scheduled on the Startup and Test Three-Day Schedule dated 5/25/84 and signed by the Day Shift Supervisor. The cover sheet indicating that the schedule applied from 1600 5/25/84 through 1600 5/29/84 was signed by the Unit Coordinator. The test procedures had been approved by the PORC and had been signed by the Plant Superintendent. The initial conditions were in accordance with the procedures and had been successfully achieved on several previous occasions. After conditions had stabilized the Duty Manager was informed. He concurred in the decision to notify the NRC and he granted permission to complete the scheduled testing.

The evolutions were properly authorized by cognizant line management. There was no improvisation by the operating crew.

3.6 One Hour Report was Required

There has been some discussion as to whether the event should have been reported to the NRC at all. The reactor was being operated within the limits of the license and a brief transient was caused by an equipment malfunction. No safety limits were violated. In the opinion of NSAG, the situation is analogous to operating at 100% power and experiencing a casualty which causes an excursion above the steady state limit but below the SCRAM setting.

The 100% power level excursion is covered by NRC memorandum SSINS 0200 E. L. Jordan to Distribution, "Discussion of Licensing Power Level (AITS F14580H2)", August 22, 1980. The basic guidance is that average power over an 8 hour interval may not exceed the license limit and that the instantaneous power may not exceed 102% of the license limit.

102% of 5% is 5.1%. An excursion of 5.9% violates the 102% guideline for instantaneous power.

Paragraph 6.6 of the Technical Specifications defines Reportable Event Action. It states,

"a) The commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10CFR50,..."

Specific instructions to the operators are found in Administrative Procedure AD-QA-424, Significant Operating Occurrence Reports, Rev. 4 effective 1/1/84. Table A of Attachment C Operational Events One Hour ENS Notification lists, item 8, "Violations of Operating License." Page 27 of 53 discusses item 8 and specifically states "Any violation of License Conditions 2C(1)...". Condition 2C(1) of the Unit Two operating license states "...Pending Commission approval, this license is restricted to power limits not to exceed five percent of full power (164.6 megawatts thermal)."

It is clear that the decision to make a one-hour report to the NRC was consistent with the Technical Specifications, the NRC interpretation of the power limits, and the station instructions.

3.7 No Hazard Existed

The Nuclear Plant Engineering Engineering Analysis Group analyzed the transient and determined that it is within the bounds of transients analyzed in the FSAR.

A copy of the NPE Evaluation (File 247-01 of 5/30/84) is attached.

4.0 Conclusions

1. The license power limit of 5% was exceeded for less than 3 minutes. Maximum power was 5.9% per APRM "B" after gain adjustment.
2. No nuclear safety hazard existed to the plant or to the public.
3. The transient was caused by an equipment malfunction.
4. Operator actions to control the casualty were prompt and effective.
5. All evolutions were authorized by responsible line management.

jrm/rpe1431/rla

PP&L

MEMORANDUM

TO: Rick Nobles DATE: May 30, 1984

FROM: A. J. Roscioli

JOB: ER 100450 NUMBER: EA-096 COPIES TO: T. M. Crimmins, A6-2
G. D. Miller, A2-5
J. S. Stefanko, A2-5
SRMS Corres. File, A6-2
SRMS Letter File, A6-2

FILE: 247-01 REPLY: NO

SUBJECT: EVALUATION OF MAY 28, 1984 SSES UNIT 2 TRANSIENT EVENT

During the 5/28/84 SSES plant event, the bypass valves closed as demanded by the pressure regulation controller. The bypass valves remained closed until pressure and power exceeded their initial values of the transient. As a result power increased above the 5% licensed power level before the bypass valves reopened to control pressure and mitigate the power rise.

This event is less limiting than the Turbine Trip without Bypass transient from ≤ 30% power which is discussed in Section 15.2.3 of the SSES FSAR. The turbine trip event results in a faster reduction in steam flow and a higher initial power level. The higher initial power level results in a larger void collapse in the core causing a higher power spike. Section 15.2.3.3.3.3 states that the turbine trip without bypass event results in a high vessel pressure scram. Therefore, the peak power remains below the flow biased simulated thermal power upscale trip setpoint and the MCPR remains well above the GETAB safety limit.

Since the initial power is lower, the steam flow reduction and subsequent pressurization is slower, the magnitude of the pressurization is mitigated by reopening of the bypass valves, and the void collapse is less severe due to the lower initial power, the event that occurred at SSES on 5/28/84 is much less severe than the Turbine Trip without bypass event from low power which is analyzed in Section 15.2.3 of the FSAR.

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