NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION

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EXPIRES 06/30/2001

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Susquehanna Steam Electric Station - Unit 2	05000388	1 OF 5

TITLE (4)
Reactor SCRAM - IRM Upscale Trip

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAME	DOCKET NUMBER
										05000
7	2	98	98	-- 010 --	00	8	3	98	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): 2
POWER LEVEL (10): 001

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)	X	50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Stephen J. Ellis - Senior Engineer, Licensing	717 / 542-3537

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).	X NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On July 2, 1998, at 0417 hours, with Unit 2 in Condition 2 (Startup) at about 0.1% power, a Division I Intermediate Range Monitor (IRM) upscale trip was received, shortly followed by a Division II IRM upscale trip, resulting in a Reactor SCRAM. Both upscale trips were a result of the misoperation of the IRM range switches. Just prior to the upscale IRM trips, a licensed operator had continuously withdrawn control rod 34-23 from position 00 to position 24. This resulted in a reactor period, calculated post event, of approximately 30 seconds. Following the SCRAM, the reactor responded as expected, and there were no Emergency Core Cooling System (ECCS) initiations or injections. There were four fundamental root causes for this event: (1) the operating crew failed to exercise proper sensitivity to reactivity manipulations during the startup and did not demonstrate a profound respect for the rector core, (2) the operating crew team dynamics failed when the team members did not establish and maintain their roles, (3) management and supervisory oversight failed to recognize operator practices leading to this event, and (4) operators failed to apply the knowledge of reactor physics fundamentals and associated skills gained through training during the reactor startup beyond the initial criticality of the reactor. This event is reportable per 10CFR50.73(a)(2)(iv) as a automatic actuation of an Engineered Safety Feature (ESF), in this case, the Reactor Protection System (RPS), which resulted in a reactor SCRAM. There were no safety consequences as a result of this event and the health and safety of the public was not compromised. No safety limits were approached and the event is entirely bounded by the plant accident analysis. Corrective actions include: (1) procedure and startup sequence changes, (a) requiring proper single notch withdrawal restraints through the point of adding heat, (b) leaving the IRM recorders on fast speed until above IRM range 7, and (c) ensuring the SRMs are responsive throughout this power range; (2) operator training on this event, and management's expectations with regard to reactivity control; (3) establishment of various teams to ensure reactor reactivity changes are adequately controlled and monitored, and (4) establish a self-assessment program for reactivity management to assess in-plant performance against clearly defined standards.

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

EVENT DESCRIPTION

On July 2, 1998, at 0417 hours, with Unit 2 in Condition 2 (Startup) at 0.1% power, a Division I Intermediate Range Monitor (IRM) (EIIS Code: IG) upscale trip was received, shortly followed by a Division II IRM upscale trip, resulting in a Reactor SCRAM. All control rods inserted, the reactor responded as expected, and there were no Emergency Core Cooling System (ECCS) initiations or injections.

The reactor was being returned to service in accordance with the appropriate General Operating (GO) procedure. The operating crew in charge of the startup consisted of a Senior Reactor Operator (SRO) (utility; licensed) and two Reactor Operators (utility; licensed - RO), PCO-1, in charge of reactivity changes and PCO-2, control rod position verifier. PCO-1 had pulled control rods to achieve criticality at 0258 on July 2, 1998. Criticality had occurred when control rod 34-39 was taken to position 08. The Startup Range Monitor (SRM) / IRM overlap testing was performed, and the SRMs were fully withdrawn from the core. From 0310 to 0312, PCO-1 continued single notch withdrawal of control rod 34-39 to increase power, taking the control rod to position 22 and attaining approximately a 200 second reactor period. The reactor operators suspended control rod withdrawal to perform other startup activities per the GO procedure. The reactor coolant temperature was increasing due to decay and pump heat.

At approximately 0415, the operators were ready to resume power ascension. Due to the increasing coolant temperature, the reactor had become sub-critical. This was not recognized by the operating crew. PCO-1 and PCO-2 discussed resumption of control rod withdrawal with the SRO assigned to oversee the startup. The crew decided to perform continuous rod withdrawal until a 100 second period was seen on the SRMs. At 0415, control rod 34-39 was fully withdrawn, with only slight increases in IRM indication observed, which then settled out. The next control rod in the startup sequence, 34-23, was selected and continuously withdrawn from 00 to position 24. As the control rod was being withdrawn, reactor power began increasing, with an average period, calculated post-event, of approximately 30 seconds. As power was increasing, the IRMs had to be ranged-up. PCO-2 was directed by the SRO in charge of reactivity to range-up the 'E' IRM. The PCO-2 inadvertently moved the IRM range switch from position 5 to position 7 (i.e., two positions vice one). PCO-2 then immediately moved the switch to the 6 position. Eleven seconds later IRM 'C' needed to be ranged-up, from range 5 to range 6. PCO-2 inadvertently ranged IRM 'C' down to position 4, instead, causing a half SCRAM from the Division I IRM upscale trip. Three seconds later, PCO-1 inadvertently ranged IRM 'D' from position 6 to position 5, instead of to position 7. This caused a Division II IRM upscale trip and the second half SCRAM. The reactor scrammed at 0417.

CAUSE OF EVENT

A formal root cause analysis was performed for this event. The fundamental causes of this event have been identified as:

1. The operating crew failed to exercise proper sensitivity to reactivity manipulations during the startup and did not demonstrate a profound respect for the reactor core.

The operators did not deliberately, and in a carefully controlled manner, move control rods while monitoring reactor response. There are a number of conditions that contributed to this event, which are related to this root cause. The procedural guidance, with regard to control rod movement, was less than adequate for the period from initial criticality to the point of adding heat (POAH). Also, the review of industry events with respect to reactivity control, was less than adequate.

During this event, the IRM recorders had been placed in slow speed (per procedural guidance). While in slow speed, the recorders do not provide useful trend information for changes in Reactor Power. Also, SRMs were fully withdrawn, and as a result, were decoupled from the core, causing their indication to be delayed. Finally, the incore nuclear instrumentation being displayed was also delayed due to data processing time associated within the plant computer system. These shortcomings in data presented to the operator were not addressed in the operator's procedural guidance.

2. The operating crew team dynamics failed when the team members did not establish and maintain their roles.

   This was not in keeping with PP&L's expectations. PCO-2 was the verifier, and as such, should not be performing any manipulations. The SRO directed PCO-2 to uprange the 'E' IRM. The expectation is that one operator manipulates the controls at the reactor benchboard. The dilution of responsibility contributed to the human error on both divisions of IRM range switches. The role and responsibility of the team in charge of reactivity changes was not adequately defined in procedures.

3. Management and Supervisory oversight failed to recognize operator practices leading to the event.

   PCO-2 was the verifier, and as such, should not have been performing manipulations. However, the SRO requested PCO-2 to uprange the 'E' IRM. The expectation is that one operator manipulates the controls. Management oversight did not recognize and correct this deficiency.

4. Operators failed to apply the knowledge of reactor physics fundamentals and associated skills gained through training during the reactor startup beyond the initial criticality of the reactor.

   All operating shifts received simulator startup training during a training cycle earlier this year (ended 6/19/98). During this training, the operating shifts practiced a unit startup from just prior to criticality to 50°F beyond the POAH. However, during this Unit 2 startup, the operators did not apply fundamental concepts of reactor theory to this practical situation of an actual reactor startup in the intermediate range.

REPORTABILITY/ANALYSIS

This event was determined to be reportable per 10CFR50.73(a)(2)(iv) as a automatic actuation of a Engineered Safety Feature (ESF). In this case, the two IRM upscale trips initiated the Reactor Protection System (RPS) (EIIS Code: JC), resulting in a reactor SCRAM from less than 1% power.

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There were no safety consequences as a result of this condition. The energy added by the control rod withdrawn was only a small fraction of that needed to cause fuel damage. This event is bounded by the Control Rod Drop Accident analysis, and did not approach any of the limiting parameters of that analysis. All control rods were in sequence, in accordance with the Banked Position Withdrawal Sequence (BPWS) rules, as was being enforced by both the Rod Sequence Control System (RSCS) and the Rod Worth Minimizer (RWM) (both EIIS Code: JD). The reactor power increase associated with the continuous withdrawal was approximately 0.1%. There were no thermal limits challenged as a result of this event. Considering the relatively small amount of energy added due to the control rod continuous withdrawal, there is no chance of preconditioning fuel failure. The plant responded properly to the event, and subsequent reactor SCRAM. All control rods inserted, and there were no ECCS initiations or injections. Based on the analysis above, there were no consequences or compromises to public health and safety as a result of this event.

From a pure safety standpoint, this is a very low significance event. However, it does represent a breakdown of several barriers (procedures and people) to our defense in depth. As such, this event is worthy of serious scrutiny and comprehensive actions to prevent recurrence.

The circumstances of this event have generic implications in that a profound respect for the reactor core is required at all times, not just during startups. As such, the actions to prevent recurrence must address the reactivity management program as a whole.

Further review of previous startups at Susquehanna has determined that during at least one other startup, continuous rod withdrawal was used soon after initial criticality, resulting in a fast period which the operating crew corrected by insertion of the control rod several notches.

A review was performed to see if core design changes over the past several years to support power uprate and 24 month cycles may have had any adverse affect on the reactor response during startup. The review indicated that the control rod worths associated with this core are comparable or less than past cores of 9x9 fuel and 18 month cycles. This potential had been adequately reviewed in the past and minor changes associated with the current fuel (ATRIUM 10) have been adequately addressed in procedures and licensed operator training. This event was not related to those changes.

CORRECTIVE ACTIONS

The following corrective actions have been taken:

- The control rod pull sheets were changed to impose notch withdrawal restraint to all control rods through the POAH.
- The GO's for Plant Startup, Heatup and Power Operations were revised to leave IRM recorders on fast speed until above range 7.

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- Training was conducted for the operating shift responsible for the subsequent Unit 2 startup.
- The Vice President - Nuclear Site Operations presented to Operations supervision his impression of the event and his expectations for the operation of the plant, especially with respect to reactivity controls and a profound respect for the reactor core. These performance expectations were reinforced by the General Manager - Susquehanna, Manager - Nuclear Operations, and the Supervisor - Nuclear Operations.
- All Operations personnel reviewed this event, the causes known at the time, and the immediate corrective actions.

The following actions are scheduled for future completion:

- Assemble a team to develop a reactivity management enhancement plan, which will review all aspects of reactivity control and recommend changes. The charter for this team will be established by the Manager - Nuclear Operations,
- Establish a Reactivity Management Oversight committee to ensure thorough review of reactivity related industry and Susquehanna SES events.
- Establish a self-assessment program for reactivity management to assess in-plant performance against clearly defined standards.
- Review Operations standards and expectations. Ensure professional operating practices are clearly defined and include the elements to establish a professional demeanor, with clear focus on the operation of the power plant at all times. Ensure the standards promote a profound respect for the reactor core, a philosophy of preventing events through proactive actions, an environment of conservative decision making, and definition of specific roles and responsibilities of all members of the operating team.
- Establish a Station Policy on Nuclear Safety.
- Re-evaluate PP&L's response to SOERs 96-01, Control Room Supervision, Operational Decision Making and Teamwork, and 96-02, Design and Operating Considerations for Reactor Cores.
- Provide additional training to all licensed personnel on reactor behavior between initial criticality and POAH.
- Change GO procedures to ensure SRM detector response during startup.

<u>ADDITIONAL INFORMATION</u>

Past Similar Events: None

Failed Component: None