Bryce L. Shriver
General Manager - SSES

Susquehanna Steam Electric Station.
P.O. Box 467, Berwick, PA 18603
Tel. 570.542.3120 Fax 570.542.1477
blshriver@papl.com

ppl ™

December 22, 1999

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, DC 20555

SUSQUEHANNA STEAM ELECTRIC STATION
LICENSEE EVENT REPORT 50-387/99-008-00
PLA - 5150 FILE R41-2

Docket No. 50-387
License No. NPF-14

Attached is Licensee Event Report 50-387/99-008-00 . This report is being made pursuant to 10CFR50.73(a)(2)(ii)(B), in that Susquehanna S.E.S. Unit 1 and Unit 2 were determined to have a deficiency in the design of the Traversing Incore Probe System which created the potential for a release path during postulated Design Basis Accident Loss Of Coolant Accident conditions.

Bryce L. Shriver
General Manager - SSES

Attachment

cc: Mr. H. J. Miller
Regional Administrator
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. S. L. Hansell
Sr. Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 35
Berwick, PA 18603-0035

IE22

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

LICENSEE EVENT REPORT (LER)

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

APPROVED BY OMB NO. 3150-0104
EXPIRES 06/30/2001

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Susquehanna Steam Electric Station - Unit 1	05000387	1 OF 4

TITLE (4)
Design Deficiency of Traversing Incore Probe Equipment Could Result in a Potential Release Path During Design Basis Accident Loss Of Cooling Accident Conditions

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									Susquehanna SES Unit 2	05000388
11	23	99	99 --	008 --	00	12	22	99	FACILITY NAME	DOCKET NUMBER 05000

		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
OPERATING MODE (9)	1		20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER LEVEL (10)	100		20.2203(a)(1)		20.2203(a)(3)(i)	X	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)		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)
Duane L. Filchner - Senior Engineer, Licensing	610 / 774-7819

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)

At 1030 hours on November 23, 1999 with Unit 1 and Unit 2 in Mode 1 (Power Operation) at 100% power, engineering personnel (Utility; non-licensed) determined that a previously identified design deficiency could result in a failure of the Traversing Incore Probe (TIP) Guide Tube Isolation Ball Valve to maintain containment isolation integrity and could potentially result in a direct pathway from the Drywell atmosphere (primary containment) to the Reactor Building (secondary containment) during postulated Design Basis Accident conditions. The cause for this condition is that the design of the TIP Guide Tube Isolation Control was provided by the NSSS supplier with a design error, which was not detected or corrected by PP&L. The design is in error by not applying the appropriate standards, namely IEEE 279-1971, to the systems and components necessary to assure that the Ball Valves remain de-energized and closed when Primary Containment isolation is required. Power to the Tip Ball Valve has been disabled to assure the TIP Ball Valves do not open due to a passive failure in the control circuitry, and may be restored under administrative control, as needed, on the basis of previous risk analyses. Long term corrective actions to resolve this condition are currently being evaluated and developed. The safety significance of this condition is low. All equipment in the TIP system remains operable because the TIP Ball Valve and the TIP Shear Valve actuation circuits remain operable and capable of operating as containment boundary devices. The TIP system remains capable of performing its LPRM calibration function.

EVENT DESCRIPTION

At 1030 hours on November 23, 1999, with Unit 1 and Unit 2 in Mode 1 (Power Operation) at 100% power, engineering personnel (Utility; non-licensed) determined that a previously identified design deficiency could result in a failure at the Traversing Incore Probe (TIP, EIIS Code IG) Guide Tube Isolation Ball Valve to maintain containment isolation and could potentially result in a direct pathway from the Drywell atmosphere to the Reactor Building secondary containment under Design Basis Accident conditions. The design of the TIP guide tube primary containment isolation controls does not meet General Design Criterion 56 for lines penetrating primary containment that communicate directly with the containment atmosphere. This constitutes a condition that is reportable because recent evaluation of this pathway has determined it to be outside the design basis of the plant as described in FSAR Sections 6.2.4 and 15.6.5.

The TIP Ball Valve control is powered through the TIP Drive Mechanism. Although the TIP system controls are in the de-energized state (i.e. OFF) during normal operating conditions, the drive mechanism power remains, and is separated from the TIP Ball Valve actuation solenoid by a single unqualified relay, one for each of five TIP Guide Tube penetrations. These relays, along with the rest of the TIP Ball Valve controls, are not designed or qualified for nuclear safety-related service. As a result, the design does not meet the standards which are intended to assure that multiple failures, resulting in undesired opening of any or all of the five guide tube penetrations, would not occur under design basis accident conditions. Potential releases under design basis conditions, assuming Regulatory Guide 1.3 source terms, could exceed the values reviewed as the basis for the Susquehanna SES operating license.

Background

The TIP Guide Tube penetrates primary containment and communicates directly with the primary containment (Drywell) atmosphere via the TIP indexing mechanism. The TIP Guide Tube primary containment boundary design must, therefore, meet the provisions of General Design Criterion 56. The TIP Guide tube penetration was justified on the basis of conformance with Regulatory Guide 1.11 Instrument Lines Penetrating Primary Containment, as an alternate basis to the General Design Criterion. However, use of Reg. Guide 1.11 is not valid for the Guide Tube isolation basis because the Guide Tube is not an instrument line, as defined in the FSAR.

The majority of the time, the TIP Guide Tube penetration is closed by virtue of the TIP Ball Valve, whose solenoid operator is de-energized when the TIP Probe is withdrawn into the Chamber Shield. When withdrawn, the Ball Valve actuating circuit remains de-energized because of the open contacts of the TIP probe withdrawal sensor proximity switch output relay. With power available at the TIP Drive Mechanism, the relay becomes the sole device assuring the Ball Valve solenoid coil remains de-energized and the Primary Containment penetration remains closed.

Passive failures of electrical devices, under accident conditions, are within the design basis of the plant when analyzing the ability of the plant to satisfy the design criteria. This is explicitly stated in the definition of Single Failure in 10CFR50 Appendix A. The assumption of a common cause failure of all unqualified (non-safety grade) systems, structures, and components is clearly endorsed as stated in IEEE 379-1972 and SECY 77-439.

The ability to maintain the TIP Ball Valve in the closed position is considered to be a safety function, to which the requirements of IEEE 279-1971 apply. Credible failure of non-safety grade components is an essential consideration in the analysis of the plant response to accident conditions. The existing design of the TIP Guide Tube isolation actuation provisions do not meet this standard, and its failure can result in the breach of primary containment. The Guide Tube isolation design is, therefore, deficient and not in conformance with the plant design and licensing basis.

CAUSE OF EVENT

The TIP Guide Tube Isolation Control is part of the Standard Design for BWRs of the type and vintage that were supplied under the NSSS scope of supply. This design was deficient, as it was received from the Vendor, in that it failed to apply the proper design standards that qualify this system for its specific safety function.

PP&L evaluated this deficiency several times since the original design was installed at SSES. However, it was not until recent deterministic calculations of the potential release path leak rates and postulated dose consequences, that it was determined that this condition is reportable.

REPORTABILITY/ANALYSIS

This condition was determined to be reportable per 10CFR50.73 (a)(2)(ii)(B) in that a design deficiency could result in a failure at the TIP Guide Tube Isolation Ball Valve to maintain containment isolation, and could potentially result in a direct pathway from the Drywell atmosphere to the Reactor Building secondary containment. At Susquehanna S.E.S. Unit 1 and Unit 2 TIP Ball Valve controls are not designed or qualified for nuclear safety-related service and do not meet the requirements of IEEE 279-1971, Criteria for Protection Systems. As a result, there can be no assurance that multiple failures, resulting in undesired opening of any or all of the five guide tube penetrations would not occur under design basis accident conditions.

Preliminary determination of the Primary Containment leak rate consequences of the postulated event, using design basis LOCA assumptions for containment temperature and pressure, predicts that leakage rates of 8% of Primary Containment volume per day could occur. This would be a factor of eight higher than the SSES Tech Spec limit, and would exceed that limit for the purpose of determining offsite dose. Preliminary estimates of dose consequences, using Regulatory Guide 1.3 source terms, are that (a) all evaluation cases result in higher dose values than those shown in the SSES FSAR Chapter 15.6.5, and (b) the NRC acceptance criterion of 75 Rem to the skin of the control room operators may be exceeded. All other doses are expected to be below the NRC criteria.

There were no actual safety consequences as a result of this condition. There were no plant transients, reactor shutdowns, changes to reactivity or loss of the system's ability to perform any of its functions. The safety significance of this condition is low. All equipment in the TIP system remains operable because the TIP Ball Valve and the TIP Shear Valve actuation circuits remain operable and capable of operating as containment boundary devices. The TIP system remains capable of performing its LPRM calibration function. There remains a high degree of confidence that the TIP Ball Valves will remain isolated, when required, and that the Shear Valves will be capable of isolating when necessary.

Power to the TIP isolation control will be restored, as needed, for the LPRM calibration function, as required by Plant Technical Specifications and for system maintenance and testing, on the basis of previous risk analyses.

In accordance with the guidance provided in NUREG 1022, Rev. 1, Item 5.1.1, the required submission date for this report was determined to be December 23, 1999.

CORRECTIVE ACTIONS

Corrective actions that have been completed:

Power to the Tip Ball Valves has been disabled at the non-class 1E power distribution panels to assure the TIP Ball Valves do not open due to a passive failure in the control circuitry. This was accomplished by opening the circuit breakers to remove power from the drive mechanism when the TIP system is not actively in use per Technical Specification Surveillance Requirement (SR) 3.3.1.1.8. The Drive Mechanism power will be restored, under administrative control, when the TIP system is required for use per SR 3.3.1.1.8 and as necessary for system maintenance and testing.

Corrective actions that are to be completed:

Evaluate and implement the long term solution to this condition from the following options:

- Retain the current system configuration of open power supply circuit breakers with the associated administrative controls for performing Tech Spec Surveillance SR 3.3.1.1.8.
- Perform a safety significance analysis based upon risk informed methodology and return the system to the original power supply configuration of closed circuit breakers. Administrative controls would still be required in this configuration while performing SR 3.3.1.1.8.
- Modify the controls for the TIP system to provide qualified equipment in the control circuitry.
- Other long term solutions as may be developed by PP&L.

ADDITIONAL INFORMATION

Failed Component Identification: None

Previous Similar Events: None