CATEGORY 1

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FACIL:50-400 Shearon Harris Nuclear Power Plant, Unit 1, Carolina 05000400
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VERRILLI,M. Carolina Power & Light Co.
DONAHUE,J.W. Carolina Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 97-003-00:on 970227,steam generator low level protection circuitry outside design basis occurred.Caused by inadequate failure modes & effects analysis performed as-built piping configuration for S/G level.Review performed.W/970331 ltr.

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NOTES:Application for permit renewal filed. 05000400

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Harris Nuclear Plant
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Serial: HNP-97-071
10CFR50.73

SHEARON HARRIS NUCLEAR POWER PLANT UNIT 1
DOCKET NO. 50-400
LICENSE NO. NPF-63
LICENSEE EVENT REPORT 97-003-00

Sir or Madam:

In accordance with Title 10 to the Code of Federal Regulations, the enclosed Licensee Event Report is submitted. This report describes a condition determined to be outside the design basis of the plant, related to Steam Generator low level protection circuitry.

Sincerely,

J. W. Donahue
Director of Site Operations
Harris Plant

MV

Enclosure

c: Mr. J. B. Brady (HNP Senior NRC Resident)
Mr. L. A. Reyes (NRC Regional Administrator, Region II)
Mr. N. B. Le (NRC - NRR Project Manager)

IE22 1/1

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Mr. R. L. Warden (RNP)
HNP Real Time Training
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NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 04/30/98
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Harris Nuclear Plant Unit-1	50-400	1 OF 4

TITLE (4)

Steam Generator low level protection circuitry outside design basis.

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
2	27	97	97	-- 003 --	00	3	31	97	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	1
POWER LEVEL (10)	100%

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)	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)
Michael Verrilli Sr. Analyst - Licensing	(919) 362-2303

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

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 February 27, 1997, with the plant operating in mode 1 at 100% power, engineering review concluded that the Steam Generator (S/G) upper instrument line tap arrangement for S/G flow and narrow range level channels do not completely satisfy the design requirements of IEEE Standard 279-1971. This conclusion was reached following a review of Westinghouse Nuclear Safety Advisory Letter (NSAL) 96-004, "Control and Protection Interaction", which requires redundant protection channels to be capable of providing protective action even when degraded by a second random failure.

Due to the arrangement of the S/G upper instrument line tap connection for S/G steam flow instruments, which share a common tap connection with a S/G narrow range level channel, combined with the logic utilized in the reactor protection system circuitry logic, a potential scenario exists that results in the S/G low-low level reactor trip being unavailable. This constitutes operation outside the design basis of the plant and was reported to the NRC via the emergency notification system at approximately 1513 hours on February 27, 1997.

This condition was caused by an inadequate failure modes and effects analysis performed on the as-built piping configuration for S/G level instrumentation during initial plant construction.

Immediate corrective actions included eliminating the potential failure scenario by placing the Steam Flow - Feed Flow selector switch on the main control board to the "Channel III" position and placing caution tags on the switch. A review was also performed of other protection channels to ensure that protection and control interaction requirements were met. No additional discrepancies were identified during this review. Planned corrective actions will include a permanent resolution to the identified design deficiency.

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

EVENT DESCRIPTION:

On February 27, 1997, with the plant operating in mode 1 at 100% power, a review performed by Harris Nuclear Plant (HNP) Engineering personnel concluded that the Steam Generator (S/G, EIIS Code: JB) upper instrument line tap arrangement for S/G steam flow and narrow range level channels (EIIS Code JB-FT,LT) does not completely satisfy the design requirements of IEEE Standard 279-1971.

The failure to satisfy these requirements was identified during a review of Westinghouse Nuclear Safety Advisory Letter (NSAL) 96-004, "Control and Protection Interaction" dated August 14, 1996. NSAL 96-004 identified a design configuration involving the failure of a common tap for S/G steam flow and narrow range level which could cause an undesired control/protection interaction scenario. The specific condition described in NSAL 96-004 was not applicable at HNP, however, during the review of this document, the following design deficiency was identified:

At HNP, each S/G has four narrow range level channels which provide input to a 2 out of 3 logic for the S/G low-low level reactor trip signal. (reference attached drawing) Each S/G also has two steam flow transmitters that share a common tap with two of the narrow range level channels. These two steam flow transmitters provide input to the steam flow-feedwater flow mismatch logic and can be selected to provide input to the S/G Water Level Control System. The Technical Evaluation contained in NSAL 96-004 specifically states that "two steam flow channels are provided for each loop, only one of which shares a tap connection with a narrow range steam generator level channel in a three level channel system." IEEE Standard 279-1971, section 4.7.3, requires that "where a single random failure can cause a control system action that results in a generating station condition requiring protective action and can also prevent proper action of a protection system channel designed to protect against the condition, the remaining redundant protection channels shall be capable of providing the protective action even when degraded by a second random failure"

Investigation concluded that, due to the arrangement of the S/G upper instrument line tap connection for the two S/G steam flow channels, which share a common tap connection with a S/G narrow range level channel, if the steam flow selector switch is placed in channel IV, the coincidence logic will not cause a S/G low-low level reactor trip during the following scenario:

- Steam Generator upper tap or impulse line used by Channel I Steam Generator Level and the Channel IV Steam Flow channel ruptures causing the level signal to fail high.

- A second random failure occurs on Channel II or Channel III.

In this scenario only one S/G low-low level reactor trip channel is available, which is not sufficient to actuate the 2 of 3 reactor trip logic. During this postulated scenario, the rupture causes a false high S/G level signal and a false low steam flow signal. If this steam flow signal is selected as input to the S/G water level control system on the steam flow selector switch, the feed regulating valve will start to shut to match feedwater flow to the apparent steam flow signal. The actual level in the affected SG will start to decrease and the reactor trip will not be received as required when the SG level drops.

This scenario requires a specific level instrument on the SG to sustain the pipe failure. The postulated second random failure must occur specifically on at least one of two remaining level instruments. Finally, the control input to the SG water level control system must be selected to a specific input (Channel IV). All three of these must occur for the scenario to occur.

This condition constitutes operation outside the design basis of the plant and was reported to the NRC via the emergency notification system at approximately 1513 hours on February 27, 1997.

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

CAUSE:
This condition was caused by an inadequate failure modes and effects analysis performed on the as-built piping configuration for S/G level instrumentation by the S/G vendor and HNP Engineering. This analysis was performed in August 1985 during initial plant construction.

SAFETY SIGNIFICANCE:
There were no actual consequences associated with this condition. The probability of this scenario occurring and not being identified and corrected by operator action was determined to be very unlikely (7.4E-7 per year) using probabilistic safety analysis estimates. In addition, numerous automatic plant protective features (ie., OT-Delta T Reactor Trip, High Power (Neutron Flux) Reactor Trip, Pressurizer High Pressure Reactor Trip, Pressurizer PORVs, etc..) would remain available to mitigate the consequences of this failure scenario with no operator action. This provides confidence that the Reactor Coolant System would remain intact and that no core damage would occur.

This is being reported per 10CFR50.73.a.2.ii as a condition outside the design basis of the plant.

PREVIOUS SIMILAR EVENTS:
Westinghouse Nuclear Safety Advisory Letter (NSAL) 96-004 was issued on August 14, 1996. This document alerted industry personnel of a control and protection interaction design deficiency associated with Steam Generator instrumentation tap/impulse line configuration. Though the specific condition identified in NSAL 96-004 was not applicable to HNP, the design deficiency identified in this LER was found during NSAL 96-004 review. There have been no previous similar HNP LERs related to deficient S/G instrumentation design/configuration.

CORRECTIVE ACTIONS COMPLETED:
1. The Steam Flow Selector Switch on the main control board was taken to Channel III, which eliminates the potential failure scenario. Caution tags have been placed on the Steam Flow Selector switch as an interim administrative control to ensure that Operations personnel maintain the switch in channel III or control feedwater flow in manual.

2. A review was performed during the investigation of this condition for other protection channels, to ensure that protection and control interaction requirements were met. No additional discrepancies were identified during this review.

CORRECTIVE ACTIONS PLANNED:
1. A permanent resolution for the identified design deficiency will be implemented by July 15, 1997.

NRC FORM 366A
(4-95)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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

EXISTING STEAM GENERATOR REACTOR TRIP LOGIC

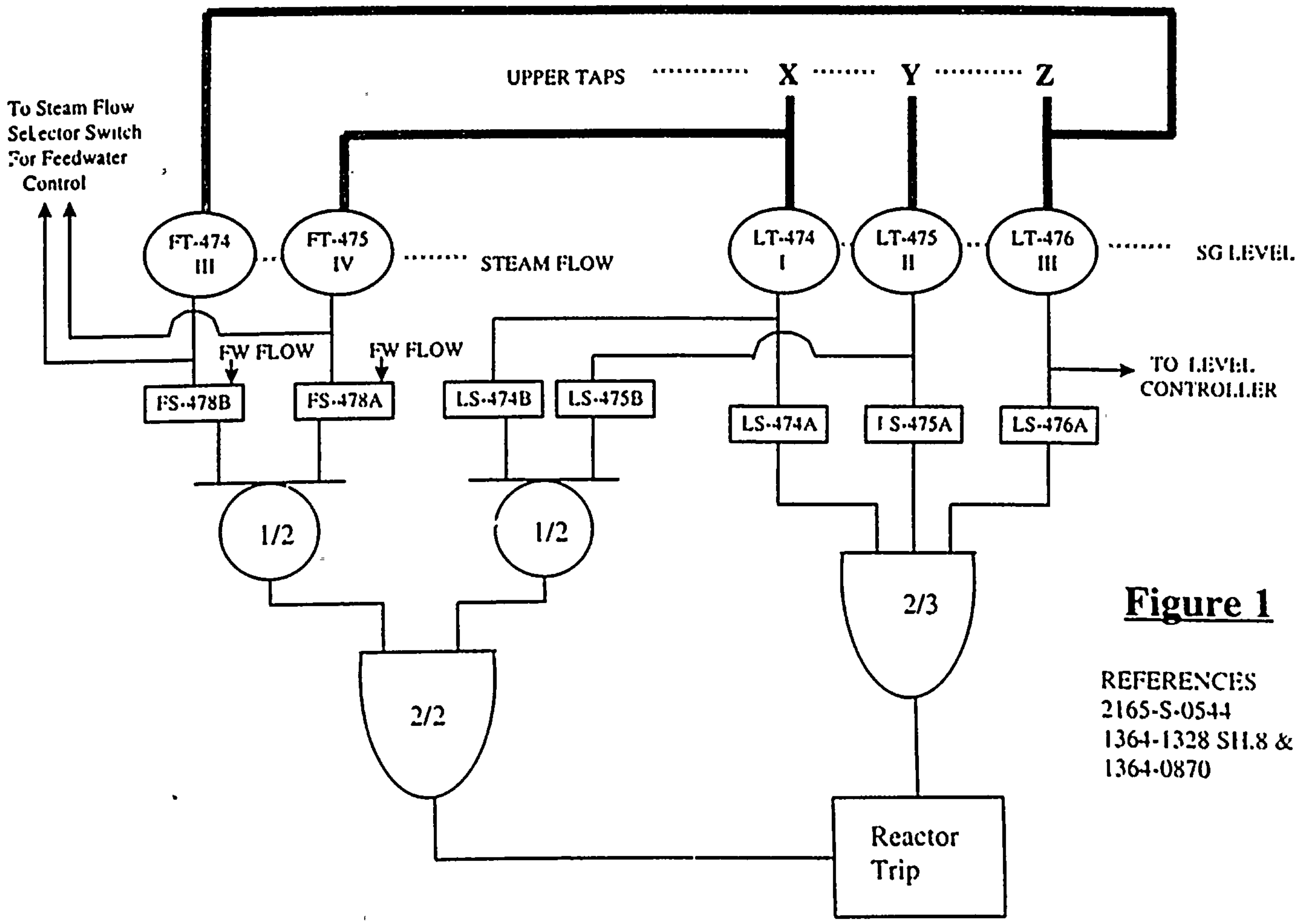

Figure 1

REFERENCES
2165-S-0544
1364-1328 SH.8 & 9
1364-0870