UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

AEOD/E203

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JAN 21 1982

50-272

MEMORANDUM FOR: Carlyle Michelson, Director
Office for Analysis and Evaluation
of Operational Data

FROM: Wayne Lanning
Office for Analysis and Evaluation
of Operational Data

SUBJECT: INADVERTENT ISOLATION OF CONTAINMENT FAN UNITS AT SALEM GENERATING STATION - UNIT 1

An engineering evaluation has been completed for the occurrence reported in the enclosed LER. The primary purpose was to evaluate the likelihood of inadvertent isolation of the containment fan coolers during LOCA conditions.

The LER reported that the service water isolation valves were found closed to the containment fan coil units (FCUs) during cold shutdown. The cause of the occurrence was traced to the service water radiation monitors which had isolated the FCUs after detecting the radiation emitting from the residual heat removal system. The monitors for the service water return piping are located in the same pipe chase as the residual heat removal system piping. The safety significance of this occurrence is that the FCUs may be required during a LOCA and inadvertent isolation would render them inoperable. However, the containment spray provides for redundant and diverse heat removal functions when the FCUs are inoperable. The licensee identified the design deficiency in October 1978 and removed the isolation function of the radiation monitors in January 1979.

The isolation function of the radiation monitors was included in the original design to prevent radioactivity from being discharged to the river. As a result of this event, the licensee has determined that the service water pressure will exceed the containment pressure during a design basis LOCA thereby preventing releases of radioactivity through the service water system. Therefore, the automatic isolation feature is not necessary and leaking FCUs will be isolated remotely by procedure.

Isolation provisions for the service water system to the FCUs were reviewed for other operating plants to determine if the radiation monitors provided an isolation function. In response to IE Bulletin 80-24 entitled "Prevention of Damage Due to Water Leakage Inside Containment," the licensee described provisions for isolating the service water system. Based on these responses, which may not be complete since the function of the radiation monitors was not specifically addressed, no other designs included automatic isolation on radiation alarms. In general, the licensee indicated that for essential service water systems, manual actions are required to isolate the system.

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In conclusion, it appears that inadvertent isolation of the containmer' 'an cooler system due to radiation alarms is not likely since most operat plants do not have this isolation feature. It appears that Salem had incorpc.ated this feature unnecessarily as an added protection to prevent radiological releases outside containment. Since the isolation function has been removed. this occurrence does not merit further consideration.

Wayne Lanning
Office for Analysis and Evaluation
of Operational Data

Enclosure:
As stated

WAYNE 33
Your Action

LER SCREENING/DISPOSITION SHEET

Docket No. 272 LER No. 78-72 -Rev l

Engineer:	GL	TRW 4/29/81	mc
1. Add'l Info. Required? (circle one)	Yes **No**	Yes **No**	Yes **No**
If YES, describe			
2. Is this event significant? (Appendix A) (Circle one)	**Yes** No	**Yes** No	**Yes** No
If yes, why?	A-5 Could be generic?	Agree	A-6
3. Abnormal Occurrence? (Appendix B) (Circle one)	Yes **No**	Yes **No**	Yes No
If yes, why?			
4. Reportable to NEA? (Appendix C) (Circle One)	Yes **No**	Yes **No**	**Yes** No
If yes, why?			C(a)
5. Recommended Action: Category (circle one)	I II **III** IV	I II **III** IV	I **II** III IV

6. Lead Engineer (NSSS) Disposition (Category III only):

Lead Engineer: ____________ Category: I II IV Comments: ____________

7. Final Action/Disposition Category II -- Farming -- Look to see if we may have a problem of this sort where the coolers are needed during post LOCA.

[signature] 11/9/81
Director, AEOD Date

PSE&G

Public Service Electric and Gas Company 80 Park Plaza Newark, N.J. 07101 Phone 201/430-7000

March 25, 1981

Mr. Boyce H. Grier
Director of USNRC
Office of Inspection and Enforcement
Region 1
631 Park Avenue
King of Prussia, Pennsylvania 19406

Dear Mr. Grier:

LICENSE NO. DPR-70
DOCKET NO. 50-272
REPORTABLE OCCURRENCE 78-72/01X-1
SUPPLEMENTAL REPORT

Pursuant to the requirements of Salem Generating Station Unit No. 1 Technical Specifications, Section 6.9.1, we are submitting supplemental Licensee Event Report for Reportable Occurrence 78-72/01X-1.

Sincerely yours,

R. A. Uderitz
General Manager -
Nuclear Production

CC: Director, Office of Inspection
and Enforcement (30 copies)
Director, Office of Management
Information and Program Control
(3 copies)

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Report Number: 78-72/01X-1

Report Date: March 25, 1981

Occurrence Date: 11/7/78

Facility: Salem Generating Station - Unit 1
Public Service Electric & Gas Company
Hancocks Bridge, New Jersey 08038

IDENTIFICATION OF OCCURRENCE:

Inoperable Fan Coil Units

CONDITIONS PRIOR TO OCCURRENCE:

Operational Mode 5

DESCRIPTION OF OCCURRENCE:

During the week of October 19, 1978, while the plant was shutdown for a maintenance outage, all five Containment Fan Coil Unit's service water isolation valves were found to be tripped closed. The problem was traced to the CFCU service water radiation monitors alarming due to radiation emitting from the residual heat removal piping passing through the area where the detectors are located. PSE&G Engineering Department was notified of this occurrence on October 19, 1978, and a resolution was requested. On November 6, 1978, the Engineering Department notified the station that this was a potential unreviewed safety question and the Resident NRC Inspector was immediately notified.

DESIGNATION OF APPARENT CAUSE OF OCCURRENCE:

The cause of this occurrence is an apparent design deficiency for the RMS detector installation. The purpose of these detectors is to monitor radioactivity in the service water discharge from the Fan Coil Units during normal operation. Since the detectors are sensitive to changes in background radiation levels, they will cause service water isolation to the Fan Coil Units when the radiation levels are sufficient to cause the RMS detectors to initiate an alarm condition.

ANALYSIS OF OCCURRENCE:

The five Fan Coil Units use service water for cooling with a portion of the outlet flow diverted to a radiation monitor designed to initiate automatic isolation of the service water flow if there is radioactivity in the water, thus preventing radioactive water from being discharged to the river. This possibility existed, if the service water system failed during a LOCA with the containment at peak pressure, since the peak pressure was thought to be greater than service water pressure. However, as documented in the Mechanical Division Safety Evaluation SE-004, the service water pressure in the containment would be greater than the peak containment pressure during a LOCA.

The radiation levels in the area of the monitors will be extremely high during a LOCA due to the activity present in the RHR lines or from streaming through the containment penetrations. These radiation levels would trip the monitors, thus isolating service water flow, resulting in inoperative Fan Coil Units. The isolation function is not applicable or necessary for the LOCA condition since no leakage of activity into the service water lines will occur.

Shielding necessary to effectively alternate radiation levels during a LOCA would be in the range of tons per unit (approximately 12 to 16 inches of lead) and would require structural steel supports. The immediate solution is to remove the isolation function circuitry and leave remote manually operated isolation valves in these service water lines. By maintaining only the alarm function, administrative action could be taken to manually isolate service water flow to these components, if an alarm is received and determined to be valid. How? This also eliminates the possibility of no service water flow to the Fan Coil Units during a LOCA due to high background radiation from the RHR lines or other sources.

CORRECTIVE ACTION:

Design Change 1EC-0448 was completed on January 11, 1979 which removed the control function of the Fan Coil Units radiation monitors. The detectors have been shielded with lead blankets to reduce their sensitivity to general area radiation. No further corrective action is planned.

FAILURE DATA:

Not Applicable

Prepared By W. J. Steele

SORC Meeting No. 81-19

H J Midura
Manager - Salem Generating Station

NRC FORM 366
(7-77)

U. S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

CONTROL BLOCK: [| | | | |] (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

0 1 | N J S G S 1 (2) | 0 0 - 0 0 0 0 0 - 0 0 (3) | 4 1 1 1 1 (4) | (5)
LICENSEE CODE | LICENSE NUMBER | LICENSE TYPE | CAT

CON'T
0 1 | REPORT SOURCE: L (6) | DOCKET NUMBER: 0 5 0 0 0 2 7 2 (7) | EVENT DATE: 1 1 0 7 7 8 (8) | REPORT DATE: 0 3 2 5 8 1 (9)

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

0 2 During Mode 5 operation, all containment Fan Coil Units were found to be
0 3 inoperable due to tripped service water valves. The problem was traced to the RMS
0 4 detectors and Engineering Department was notified on 10/19/78. The station received
0 5 a reply on 11/6/78 that a safety evaluation would be made and corrective action
0 6 initiated. The Resident NRC Inspector was immediately notified.
0 7
0 8

	SYSTEM CODE	CAUSE CODE	CAUSE SUBCODE	COMPONENT CODE	COMP. SUBCODE	VALVE SUBCODE
0 9	S B (11)	B (12)	C (13)	Z Z Z Z Z Z (14)	Z (15)	Z (16)

(17) LER/RO REPORT NUMBER	EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
	7 8	—	0 7 2	/	0 1	X	—	1

ACTION TAKEN	FUTURE ACTION	EFFECT ON PLANT	SHUTDOWN METHOD	HOURS (22)	ATTACHMENT SUBMITTED	NPRD-4 FORM SUB.	PRIME COMP. SUPPLIER	COMPONENT MANUFACTURER
F (18)	Z (19)	Z (20)	Z (21)	0 0 0 0	Y (23)	N (24)	Z (25)	Z 9 9 9 (26)

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

1 0 A design deficiency in the installation of the RMS detectors for the CFCU service
1 1 water did not sufficiently shield the detectors from background radiation coming from
1 2 RHR piping in the area. The isolation function of the detectors was removed as per
1 3 DCR 1EC-0448. The detectors were shielded with lead blankets to reduce their
1 4 sensitivity to general area radiation. No further corrective action is planned.

	FACILITY STATUS	% POWER	OTHER STATUS (30)	METHOD OF DISCOVERY	DISCOVERY DESCRIPTION (32)
1 5	G (28)	0 0 0 (29)	N/A	C (31)	Maintenance Test

	ACTIVITY RELEASED	CONTENT OF RELEASE	AMOUNT OF ACTIVITY (35)	LOCATION OF RELEASE (36)
1 6	Z (33)	Z (34)	N/A	N/A

PERSONNEL EXPOSURES

	NUMBER	TYPE	DESCRIPTION (39)
1 7	0 0 0 (37)	Z (38)	N/A

PERSONNEL INJURIES

	NUMBER	DESCRIPTION (41)
1 8	0 0 0 (40)	N/A

LOSS OF OR DAMAGE TO FACILITY (43)

	TYPE	DESCRIPTION
1 9	Z (42)	N/A

PUBLICITY

	ISSUED	DESCRIPTION (45)	NRC USE ONLY
2 0	N (44)	N/A	

PDR

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NAME OF PREPARER W. J. Steele PHONE: 609-935-0998

ROY

Frederick W. Schneider
Vice President
Production

Public Service Electric and Gas Company 80 Park Place Newark, N.J. 07101 201/430-7373

January 2, 1981

Mr. Boyce H. Grier, Director
Office Of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
Region 1
631 Park Avenue
King of Prussia, Pennsylvania 19406

Dear Mr. Grier:

NRC IE BULLETIN NO. 80-24
PREVENTION OF DAMAGE DUE TO WATER LEAKAGE
INSIDE CONTAINMENT (OCTOBER 17, 1980 INDIAN POINT 2 EVENT)
SALEM GENERATING STATION
UNITS NO. 1 AND 2
DOCKET NOS. 50-272 AND 50-311

In response to your letter of November 21, 1980, transmitting NRC IE Bulletin 80-24, the attached response is hereby submitted for your review.

Approximately ninety-eight (98) manhours were expended on this bulletin's review.

If you have any further questions, we will be pleased to discuss them with you.

Sincerely,

F W Schneider

CC Director
Nuclear Regulatory Commission
Office of Inspection and Enforcement
Washington, D. C. 20555

PDR

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The following response corresponds to the Item Nos. of NRC Bulletin 80-24.

1. The only open cooling water system present inside of Salem Units 1 and 2 containment is the containment fan cooling system. The following information provides a summary description of the system.

   a) The mode of operation of the fan cooler is different during normal reactor operation and during a response to a LOCA. During normal reactor operation, two to four of the five fan coil units are running depending on seasonal conditions and reactor power. The remaining units are in a standby condition and are ready for service. During a LOCA, the following three different operating modes would be possible:

      1. All five containment fan coil units and no containment spray,
      2. Two containment spray trains and no fan coil units, or
      3. Three fan coil units and one containment spray train.

      The service water system flows to the operating and standby fan coil units.

   b) The source of water to the fan coil units is service water. Its typical chemical content is:

Item	Minimum	Average	Maximum
Conductivity, micro-ohms	250.0	8,500.00	18,000.0
pH	6.2	7.1	9.0
Total dissolved solids, ppm	139.0	5,890.0	13,689.0
Suspended matter, ppm	5.0	135.0	694.0
Sulfides, ppm as H_2S	0.0	0.04	.48
Chlorides, ppm as NaCl	20.6	5,300.0	11,080.0
Dissolved oxygen, ppm	3.28	7.9	17.39
Chemical oxygen demand, ppm	0.0	84.5	594.0
Total ammonia, ppm	0.025	.32	2.33
Sulfates, ppm as SO_4	5.0	474.0	1,050.0
Free carbon dioxide, ppm	0.0	3.6	26.2

c) The piping and cooler tubes used in the fan cooling system are made of corrosion resistant materials. The piping to the coolers is cement lined carbon steel. The high erosion piping sections are currently being replaced with 316 stainless steel piping. The cooler tubes are currently 90/10 CuNi, and these will be replaced with AL6X tubing.

d) Experience with system leakage is documented in the response to item 1(e).

e) The following list is a history of the type of repairs done on the fan cooler system:

Work Order No.	Description	Date
MD-2162	11 FCU Motor Cooler Replacement	10/24/77
OD-6092	14 FCU Motor Cooler Repair	10/24/77
MD-2319	13 FCU Motor Cooler Repair	10/31/77
OP-0146	15 FCU Motor Cooler Repair	11/03/77
MD-2315	13 FCU Motor Cooler Repair	11/03/77
MD-2895	11 FCU Motor Cooler Repair	8/04/78
MD-2948 & OD-10060 & MD-2949 & OP-90396	11 through 15 FCU Replace Spool	8/23/78
OD-10158	14 FCU Motor Cooler Replace/Repair	8/30/78
MD-0353 & MD-903366	12 FCU Motor Cooler Replace/Repair	3/31/79
MD-905965	13 FCU Repaired Flange Leak	79
OD-916898	11 FCU Motor Cooler Replace/Repair	9/05/79
OD-915491	13 FCU Motor Cooler Replace/Repair	10/23/79
OD-932346	15 FCU Motor Cooler Replace/Repair	80
	11 FCU Motor Cooler Replace/Repair	6/11/80
MD-912685	11 FCU Primary Cooler Coil Replace	7/08/80
	11 FCU Motor Cooler Replace/Repair	9/03/80
	11 FCU Primary Cooler Coil Replace	9/11/80
MD-910204	12 FCU Primary Cooler Coil Replace	9/11/80
	12 FCU Motor Cooler Replace/Repair	8/02/80
MD-936319	14 FCU Primary Cooler Coil Replace	8/08/80
MD-936324	14 FCU Secondary Cooler Coil Replace	8/11/80
	15 FCU Secondary Cooler Coil Replace	9/09/80

All repairs were done by welding of copper nickel to carbon steel.

f) The service water system to the fan coolers is provided with isolation valves. Each fan cooler unit has an inlet and outlet isolation valve located outside containment. Isolation of the individual cooler can be accomplished by remote air operated valves from the control room or local manual operation. The pilot solenoid valves are the same power channel as the power feed to the respective fan cooler. Loss of power or air will cause the isolation valves to fail open (due to safety function design of the fan coolers). Redundant air is supplied to each valve to minimize probability of failure to close when needed. The fail safe condition for the valves must be open due to safety function conditions. This arrangement negates the vulnerability of this system to single failure.

g) There are no provisions for testing the isolation valves in accordance with Appendix J to 10CFR50. This procedure is not required because the isolation valves do not meet the criteria of II. H1 through 4 of Appendix J to 10CFR50.

h) The following instrumentation is in place to detect leakage:

<u>Service Water Flow</u> - Each containment fan coil has an individual flow indicator on the control console. In addition, a differential service water flow inlet to outlet will cause a bezel alarm on the control console.

<u>Containment Fan Coil Leak Detector</u> - The condensate from the fan coil drain pans is collected and funneled into a standpipe which has a high alarm and a high-high alarm which are located on the overhead annunciator. Also a selectable level indicator is located on the control console.

<u>Dewpoint</u> - A dewpoint measuring system is installed to continuously monitor inlet dew temperature of each fan coil unit and recorded on panel 1RP1.

<u>Radiation Detection</u> - A radiation detector is installed in the service water outlet piping of each containment fan coil. Upon initiation of a high radiation level a bezel alarm is actuated and the coil is isolated by procedure.

<u>Containment Sump Level Indication</u>- On the control console two channels of analog level indication are installed on Unit 1 and are now being installed on Unit 2 as per post TMI requirements. Also included is a containment sump overflow alarm on the overhead annunciator.

Procedures are in place to detect leakage in the containment building utilizing both the reactor coolant leak detection procedure OI II-1.3.5 and reactor coolant leak rate computation procedure SP(O)4.4.6.2(d).

i) Radiation monitors are provided to minotor fan cooler service water discharge and provide alarms and indication to operators. Grab sample analysis capability is also provided. No automatic isolation of fan cooler service water is initiated upon radiation alarms. The fan coolers are a safeguards system and perform an accident mitigation function.

2. The following actions and verifications were accomplished at our Salem plant.

a) A redundant means exists to detect and alert control room operators of a significant accumulation of water in the containment sump. There are two channels of level indications on the control console. Also included is a containment sump overflow alarm on the overhead annunciator. The reactor sump has sump pump start-stop times and the sump high level alarm indicated on the auxiliary alarm printer.

b) A positive means exists to determine flow from the containment sump. Observation of containment sump pump start and stop times are indicated on the auxiliary alarm printer. Utilizing the standard plant operating instruction OI-1.3.5, the operator can thus determine leak rate into the sump. A similar arrangement is provided for the reactor sump pump.

c) Whenever a containment fan coil leak detection high alarm is received, shift routine requires that the total fan coil unit leak rate be determined in accordance with OI II-1.3.5 (Reactor Coolant Leak Detection). In addition, it is required that an auxiliary annunciator alarm summary be initiated and evaluated at least once per shift, indicating sump pump operation and unusual alarms. These two shift procedures assure that the plant operators have at least two methods of determining water level in each location and at least one pump available to remove water from each sump location.

d) A review of the present leak detection systems and procedures indicates they provide adequate means and measures to promptly detect, verify and isolate leaking components or systems within the containment building.

e) All measures described in a) through d) above are implemented; consequently, no interim surveillance measures have been undertaken.

f) Procedures, as per the station's standard administrative procedures, have been established to notify the NRC of any service water system leaks within containment via a special license event report as a degradation of a containment boundary.

3. Portions of the component cooling system, a closed cooling system, are inside the containment. To date the units have not experienced any significant amount of component cooling water leakage into the containment.

4. This letter serves as a written report in response to your items listed in IE Bulletin 80-24. The attached letter of affirmation is provided.

State of New Jersey)
) SS:
County of Essex)

Frederick W. Schneider, being duly sworn according to law deposes and says:

I am a Vice President of Public Service Electric and Gas Company and as such, I find the matters set forth in our response dated January 2, 1981, to all items of Bulletin No. 80-24 "Prevention Of Damage Due To Water Leakage Inside Containment" are true to the best of my knowledge, information, and belief.

Frederick W. Schneider
Frederick W. Schneider

Subscribed and sworn to before me
this 2nd day of January, 1981

Jeanne C. Maccioccа
Notary Public of New Jersey

My commission expires on October 1, 1983.

AEOD/E204

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JAN 28 1982

MEMORANDUM FOR: R. Vollmer, Director
Division of Engineering, NRR

E. Jordan, Director
Division of Engineering and
Quality Assurance, IE NRRE

FROM: Carlyle Michelson, Director
Office for Analysis and Evaluation
of Operational Data

SUBJECT: EFFECTS OF FIRE PROTECTION SYSTEM ACTUATION
ON SAFETY-RELATED EQUIPMENT

At the Operating Reactor Event meeting held on January 7, 1982, the subject of recent fire protection system actuations at operating nuclear plants was discussed.[1/] The events showed that safety-related equipment subjected to water spray from fire protection system could be rendered inoperable. The events also indicated that spurious actuation of fire protection system can be initiated by operator error, by steam, high humidity or maintenance activities in the vicinity of fire protection system detectors. Other events also exemplify that interactions of the fire protection system with other systems (e.g., ventilation and diesel fuel oil) have not been adequately considered. At the meeting, IE was assigned the responsibility to review the recent fire system actuations and consider development of an Information Notice and the Division of Engineering, NRR was to review the events and consider the need for modifications to requirements or review procedures for fire protection systems.

We have reviewed some of the recent operating reactor events involving fire protection system actuation. Brief descriptions of these events are enclosed. Based on a review of the events, the following information is provided for your consideration in the efforts that are underway.

We share your conclusion that the adequacy of design and qualifications of safety-related equipment and systems located in areas where fire protection is provided should be re-evaluated. Potential interactions between fire protection systems and other systems that affect the operation of safety-related systems need to be thoroughly understood. Safety-related equipment, not damaged by a fire itself, should be designed and qualified to perform its intended function during and following a fire protection system activation.

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1/ Memorandum for D. Eisenhut from G. Lainas dated January 13, 1982 on "Summary of Operating Reactor Events Meeting on January 7, 1982."

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JAN 28 1982

These considerations should include all types of fire protection systems, e.g., water, halides, carbon dioxide and other chemicals. In addition, consideration could be given to incorporate diverse design considerations in the fire protection system to minimize inadvertent spray, e.g., smoke detectors and heat detectors. The diverse detectors should also minimize the likelihood of inadvertent fire protection activation during a seismic event which can induce smoke detector alarms due to airborne dust.

In summary, the NRC should have confidence that all safety-related and essential support equipment located in areas where fire protection spray systems are provided will perform the intended function during and following the activation of the fire protection system.

If you should desire additional information or assistance, the AEOD contact is Matthew Chiramal.

Carlyle Michelson

Carlyle Michelson, Director
Office for Analysis and Evaluation
of Operational Data

Enclosure:
As state

cc: H. Denton, NRR
D. Eisenhut, NRR
G. Lainas, NRR
R. Ferguson, NRR
Z. Rosztoczy, NRR
V. Benaroya, NRR
W. Lanning, AEOD
C.J. Heltemes, AEOD
S. Rubin, AEOD

Events Related to Fire Protection System Actuation

Plant	Date of Occurrence	Description of Event
Surry-2	May 28, 1981	LER 81-033 - An open valve from the fire main to the fire protection foam system allowed approximately 4,000 gallons of water to enter the above ground fuel oil storage tank. Water was subsequently found in the underground and wall tanks for the emergency diesel generators. This occurrence represents a potential common mode failure for both trains of the onsite emergency power system. The fire protection sparger is located inside the fuel oil storage tank. The potential for water to leak from the fire protection system into the fuel oil had not been considered during the design or before installation of the fire protection system. Inadequate procedures and sampling techniques contributed to this event. (AEOD is performing an engineering evaluation of this event.)
Trojan	July 26, 1981	LER 81-16 - During steady state operation with the plant at 80% power, the control room operator noticed that the control power had been lost to the "B" train hydrogen recombiner... The loss of control power was due to inadvertent activation of the fire protection deluge system while welding in the electrical penetrating area. The spray caused a short circuit and loss of control power to the hydrogen recombiner...
Trojan	Sep. 10, 1981	LER 81-021 - During normal operations, high ambient temperatures in the room housing the "A" trains

Plant	Date of Occurrence	Description of Event
		of the preferred instrument and control power buses for the ESF equipment exceeded the Technical Specifications. The occurrence resulted from the installation of a three-hour rated fire barrier between the two trains of equipment which lead to inadequate ventilation in the new room created by the wall. Inadequate interdisciplinary review resulted in an incomplete safety evaluation for the plant design change that created the fire barrier, i.e., cooling requirements for the installed heat loads versus the cooling capability of the installed ventilation system were not analyzed.
Ginna	Nov. 14, 1981	Daily Report - During start-up testing of the new fire suppression system, failure to follow test procedures caused activation of several portions of the system... Various power cabinets and electrical equipment in the turbine and intermediate buildings were sprayed. A manual reactor trip was initiated at 10:26 am following indication of two dropped rods and numerous control room annunciator alarms. The dropped rods were attributed to a trip of the "A" RPS MG set which may have reduced voltage enough to drop two rods. All sytems functioned properly following the trip and plant was maintained in "hot-shutdown" status while operability of equipment affected by the suppression system was assured...

Plant	Date of Occurrence	Description of Event
Dresden 1	Nov. 30, 1981	LER 81-39/01T-0 - Unit start-up was in progress when the control room received a HPCI Room Fire System Initiation alarm from the south ionization smoke detector. The HPCI system was declared inoperable and the HPCI steam line isolated. An Unusual Event was declared and a normal unit shutdown initiated. The health and safety of the general public was not endangered since all safety systems performed as designed and this was the first event of this type at Dresden Station. The cause of the fire system initiation is believed to have been a buildup of humidity/steam vapor in the HPCI room. The smoke detector operates on the ionization principle and is usually activated by the presence of combustion products. Discussions with the manufacturer of the smoke detector indicated that the detector may actuate if exposed to a high concentration of water vapor. The HPCI room has had a history of high humidity/steam because of steam leaks and the leakoff/drain system which runs to the sump in the HPCI room. Temporary ventilation was not operating prior to the occurrence which would have reduced the water vapor concentration. The smoke detector continued to intermittently alarm until the ventilation was restored. The station Fire Marshall will be reviewing the entire fire protection system for this HPCI room. In addition,

Plant	Date of Occurrence	Description of Event
		our Station Nuclear Engineering Department has been requested to review both the fire protection detectors and the ventilation system for possible modifications to improve reliability.
Dresden 2	Dec. 24, 1981	PNO-III-81-120A - The Unit 2 reactor was brought from full power to a cold shutdown condition on December 24, 1981, after a failure of both required high pressure ECCS systems; HPCI and ADS... The HPCI system was declared inoperable on December 23, 1981, following activation of the HPCI room fire protection water deluge system. The deluge system was activated by smoke from welding operations near a HPCI room smoke detector. The water spray caused water intrusion into the HPCI turbine oil system (which did not affect HPCI operability)...
Oyster Creek	Jan. 9, 1982	Daily Report - With the plant in cold shutdown at about 9:50 am, the auxiliary pump on the reactor water cleanup system seized. Its motor overhead. Smoke from the motor activated the fire suppression system on the south side of the reactor building at the 51-foot elevation... The fire suppression system was secured at 10:25 am. Water spray from the suppression system shorted out the position indication on one torus vent valve, damaged one reactor lo-lo- water level sensor and one reactor high pressure sensor.
Trojan	Jan. 9, 1982	Daily Report - Shortly after 1:00 am the control room

Plant	Date of Occurrence	Description of Event
		operators received signals indicating a fire in the turbine building and actuation of several deluge systems located in that area. Fire brigade personnel responding to the alarm reported that the turbine bldg was filled with steam. The control room operator there upon manually tripped the reactor and brought it to hot shutdown. Further investigation revealed that the source of the steam was a failure of a 90 degree elbow in a low pressure (150 psi) steam line from the high pressure turbine to the No. 5 feedwater heater. In addition, the heat from the steam is credited with tripping the fire alarms and deluge systems...
McGuire 1	Jan. 6, 1982	Daily Report - On January 6 licensee identified an interaction of non-safety related to safety-related equipment that could compromise Diesel Generator IA operation following a seismic event. During a check of equipment installation the licensee identified fire protection piping routed over one of the safety-related cable and an instrument panel used for HVAC inside the diesel generator room.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JAN 12 1982

MEMORANDUM FOR: Robert F. Burnett, Director
Division of Safeguards, NMSS

FROM: Carlyle Michelson, Director
Office for Analysis and Evaluation
of Operational Data

SUBJECT: METHODOLOGY FOR VITAL AREA DETERMINATION

In our meeting of July 23, 1981, we indicated that we would provide our thoughts on the vital area identification process. Based on review of selected reports, contractor meetings, and discussions between members of our staff, the following comments are provided for your consideration.

1. Generic sabotage fault trees are used for the analysis of nuclear power plants to identify vital areas and provide the basis for the proposed rule on vital area definition. Application of this technique for developing sabotage scenarios is an important part of a systematic approach for identifying vital equipment. Significant efforts have been directed toward the development and application of fault trees as exemplified by the major expenditures of resources within the safeguards research program for this purpose. However, as discussed below we believe that it is practical and necessary to identify the vulnerabilities of reactor systems and components before the application of these fault trees is undertaken.

   Although the need for vulnerability studies have been recognized, the only documented vulnerability study that we are aware of is the SAI component vulnerability study. This was a commendable effort and we believe that additional studies of this general type and approach are needed. For example, vulnerability studies of safety systems, considering system interactions and common mode failures resulting from an act of sabotage, should be used to help identify fault trees which may not otherwise have been considered. In addition, transient and accident initiators may be identified which should be further analyzed through detailed fault trees, such as air systems which have not yet been properly analyzed in sabotage scenarios. Finally, we believe that additional vulnerability studies of reactor systems are needed to help define "key vital areas" as used in the proposed rule.

   With regard to the generic fault trees developed by Sandia, some tests for completeness and accuracy may be beneficial. This would complement the review by RES's Division of Risk Analysis, with regard to the methodology and its application. For example, a working group of senior reactor

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operators could provide a valuable perspective and review of the sabotage sequences including the vulnerability of systems. A second test might be to compare the fault trees to reactor operational experiences, such as events which have resulted from manual valve manipulations and system misalignment. In this regard, it is our understanding that the fault trees do not explicitly include the manipulation of manual valves. If true, this would be a significant omission in the usefulness of generic sabotage fault trees. Further, based on our review of the Beaver Valley vital area analyses, it appears that review teams consider manual valves only on an ad hoc basis during site visits.

2. We believe the major threat of sabotage to a nuclear power plant is associated with the insider or an employee of the plant who has access to the vital areas of the plant. As previously discussed, the identification of the vital areas is an important first step in the physical protection process. The second, and equally important consideration, is how should the vital area be protected against the insider threat.

The prevalent method employed to date is access control utilizing locks. Yet, access to equipment during an emergency may be critical for particular systems of certain plants to prevent damage to equipment and degradation of safety systems functions. The impact on operational safety due to physical protection measures need to be carefully evaluated as an integral step before implementing protective measures which restrict access. Since a large number of plant personnel are authorized access to all vital areas, a specific analysis should address the reduction in risk due to an insider compared to the reduction in operational safety resulting from the physical protection measures employed. This is particularly important where "compartmentalization" of equipment is involved. The impact on operational safety due to physical protection requirements continues to be a concern to the licensees and others and requires further and timely consideration.

Protecting nuclear power plants from insider threats is an extremely difficult and necessary undertaking. Based on our review of licensee reports, it appears that the number of "employee problems" has increased in recent years suggesting that the insider threat is increasing. The problem is finding a practical and effective method of safeguards. As you know, access control measures were never intended to be effective against the insider and were to be replaced or supplemented with other assurances of personnel integrity, e.g., clearances, psychological evaluations, profile identification and recognition, special application of access control measures, and design changes to protect against sabotage. Furthermore, a majority of Security Incident Reports are related to improperly secured vital area doors and improper key controls which indicates a real concern regarding the effectiveness of access control measures. In summary, we recommend that additional resources be allocated for developing and evaluating practical methods to minimize insider threats and that this activity receive budgetary priority.

3. The Beaver Valley study does not clearly define the criteria for identifying the type of situations to be prevented from postulated sabotage actions. For example, there are a number of other accident scenarios which could produce radiological releases. The assumptions are not provided in order to analyze the identified events with regard to such items as operator actions and credit for nonsafety-related equipment. The scope seemed incomplete in that protection of vital equipment to prevent station blackout was not considered, and randomly occurring transients in combination with a covert act of sabotage were not considered. While the events analyzed include a number of other events as subsets during power operation, events occurring during shutdown and refueling did not receive proper emphasis. Vulnerability during these conditions is increased due to the increased number of personnel onsite and reduced system operability requirements.

If you desire additional information or if we can provide additional assistance, please contact me or Wayne Lanning in my office.

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