OPERATIONAL DATA SUMMARY
INDIAN POINT 2

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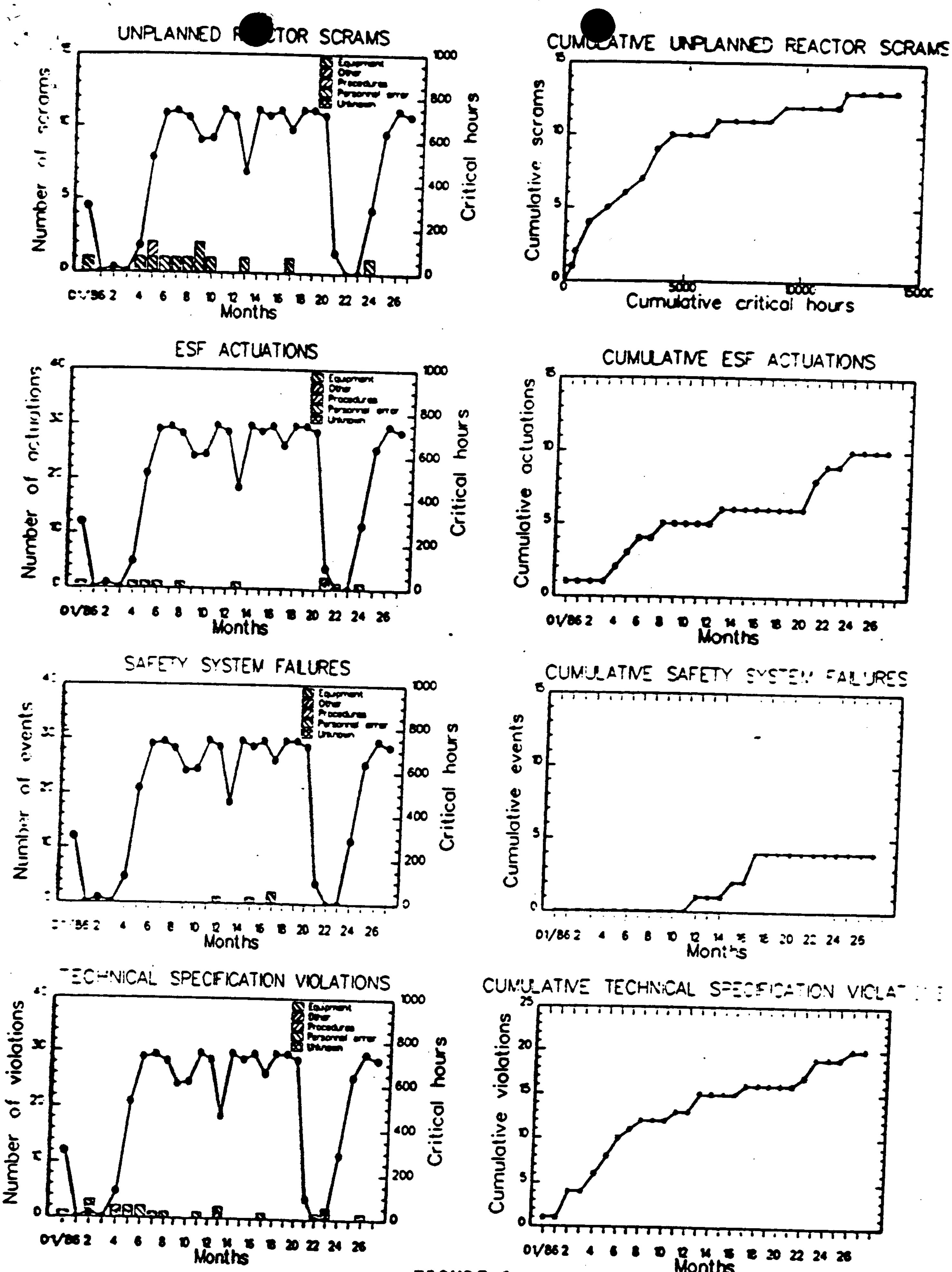
SUMMARY AND CONCLUSIONS

At the request of the Nuclear Regulatory Commission's (NRC's) Office for Analysis and Evaluation of Operational Data (AEOD), an analysis was performed of the operational data for Indian Point 2 from January 1, 1986 through April 30, 1988. The analysis was performed by the Operational Data Analysis and Evaluation Unit of EG&G Idaho, Inc. at the Idaho National Engineering Laboratory (INEL). The core of the evaluation was the operational data files maintained at the INEL for AEOD. These data files contain encoded information on Unplanned Reactor Scrams, Engineered Safety Features Actuations, Safety System Failures, and Technical Specification Violations and Shutdowns. Additional data sources were utilized as necessary to amplify and support results and conclusions.

The performance of Indian Point 2 has improved over the period of evaluation in all study areas. Figure 1 is a graphical presentation of the data in the four study areas. The most significant improvement occurred in the area of Technical Specification Violations and Shutdowns with over a 50% decrease in the number of reported events from 1986 to 1987. The overall improvement and a reduction in the unplanned scram rate is reflected in the most current Systematic Assessment of Licensee Performance (SALP) Report.

The primary cause of events in all the study areas is equipment related. Additionally, seven of the ten unplanned reactor scrams in 1986 contained equipment failures during the event. The current SALP report also documents a large maintenance backlog and a concern with operator's ability to respond to events due to the number of components out of service. The corrective actions associated with the equipment problems indicate a need for increased preventative maintenance and component surveillance. The current SALP report discusses a need for increased emphasis on preventative maintenance. However, the SALP report also states that the maintenance staff is highly skilled and performs work in a satisfactory manner.



UNPLANNED REACTOR SCRAMS
Number of scrams
Critical hours
Equipment
Other
Procedures
Personnel error
Unknown
Months
CUMULATIVE UNPLANNED REACTOR SCRAMS
Cumulative scrams
Cumulative critical hours
ESF ACTUATIONS
Number of actuations
CUMULATIVE ESF ACTUATIONS
Cumulative actuations
SAFETY SYSTEM FAILURES
Number of events
CUMULATIVE SAFETY SYSTEM FAILURES
Cumulative events
TECHNICAL SPECIFICATION VIOLATIONS
Number of violations
CUMULATIVE TECHNICAL SPECIFICATION VIOLATIONS
Cumulative violations

FIGURE 1

The events in other cause categories suggest the need for an improved maintenance program, and one event states that the cause of the event was maintenance oversight. Based on these findings, Indian Point 2 would be a good candidate plant for the Office of Nuclear Reactor Regulation (NRR) Performance Evaluation Branch Maintenance Inspection Program.

Two 1986 events were located in the AEOD Accident Sequence Precursor Program. These events were documented in License Event Reports (LERs) 24786017 and 24786035. Another event of potential interest is documented in LER 24787006 in that a single failure would render both motor operated Auxiliary Feedwater pumps inoperable. These are the safety related pumps and operator action must be taken to start the turbine driven pump. In all three of these events, the cause was equipment related.

INTRODUCTION

The Nuclear Regulatory Commission's (NRC's) Office for Analysis and Evaluation of Operational Data (AEOD) requested an analysis of Indian Point 2. The core of the analysis is an evaluation of the operational data for Unplanned Reactor Scrams, Engineered Safety Features Actuations, Safety System Failures and Technical Specification Violations and Shutdowns. Additional information that may provide insights would include the AEOD Accident Sequence Precursor Program, Systematic Assessment of Licensee Performance Reports, and requested technical specification changes, to name a few. The core study was performed by the Operational Data Analysis and Evaluation Unit of EG&G Idaho, Inc. with additional expert assistance from other EG&G Idaho organizations.

METHODOLOGY

The AEOD database files for Unplanned Reactor Scrams, Engineered Safety Features (ESFs) Actuations, Safety System Failures (SSFs), and Technical Specification (TS) Violations and Shutdowns maintained at the Idaho National Engineering Laboratory (INEL) were the primary data sources for the analysis. The Scram, ESF, and SSF data files are updated from both Immediate Notification Reports (50.72s) and Licensee Events Reports (LERs/50.73s). Thus, the data in these files are as current as one day after an event based on the information in the 50.72s. This information is subject to change upon receipt of an LER. The TS data file relies only on the LERs.

Comparisons of event rates and cause categories were developed for Indian Point 2, Indian Point 3, and mature plants. Indian Point 3 was included only for comparison purposes because the plant is located on the same site. An analysis of the Indian Point 3 data is not provided. Mature plants were defined as those plants that had more than 24 months of operation after operating license issuance at the beginning of the evaluation period (01/01/86). Therefore, all plants with an operating license issuance date before January 1, 1984 were considered a mature plant. Additionally, the Indian Point 2 and 3 data counts were not of sufficient magnitude to effect the mature plant information; therefore, the event counts for these plants are also included in the mature plant information.

After determining the operating profile for Indian Point 2 in the core study areas, efforts were taken to investigate the causes more thoroughly and locate supporting information for conclusions. The two most current Systematic Assessment of Licensee Performance (SALP) reports were obtained as supporting information. The AEOD Accident Sequence Precursor (ASP) data for 1986 was obtained with the LER numbers to evaluate the significance of the Indian Point 2 1986 events. A listing of LER number, event date, and title for all LERs submitted by Indian Point 2 was obtained from the LER Tracking data file maintained at INEL. The listing was checked for LERs that did not meet the criteria for the four core study areas to ensure that events of interest were not overlooked. Finally, reports of current events were investigated for events that may not have been reported as 50.72s or LERs. Sources for this information included the NRC Document Control System, Inside NRC news bulletin, and experts from other organizations within the EG&G Idaho, Inc.'s NRC Technical Assistance Group.

INDIAN POINT 2 CATEGORICAL DATA

Reactor Type	PWR
NSSS Vendor	Westinghouse
AE	United Engineers & Contractors, Inc.
Utility	Consolidated Edison Company
Licensed Thermal Power	2758 MWt
Design Electrical Rating	873 Net MWe
Number of Loops	4
Containment Type	Reinforced concrete cylinder with steel liner
Technical Specifications	Unique
Low power license	09/28/73
Initial criticality	05/22/73
Full power license	09/28/73
Commercial operation	08/01/74

ANALYSIS OF LERS AS A FUNCTION OF REPORTABILITY CODES

The reportability categories for which LERs were submitted for 1986 through early 1988 by Indian Point 2 were examined (see Appendix A) and compared to those for all Westinghouse (WE) Plants. The results are presented in Tables 1 through 3 which indicate that Indian Point 2 appears to be submitting LERs for the same reasons as other WE Plants. In general the distribution of reportability categories is nearly identical for Indian Point 2 when compared to all WE Plants. The highest most significant difference is 1986 Single Causes Affecting Multiple Trains. In 1988, Indian Point 2 has submitted four LERs. Therefore, this category has a higher percentage but represents only one LER.

Technical Specification Violations and Shutdowns and ESF Actuations make up a large percentage of the submittals. This is generally consistent with not only WE Plants but the entire industry.

TABLE 1
1986 COMPARISON OF LER REPORTABILITY CATEGORIES FOR INDIAN POINT 2 AND ALL WESTINGHOUSE PLANTS

Reportability Category	Percentage of LERs* Westinghouse	Indian Point 2
Limiting Condition for Operation, 50.36(c)(2)	2%	0%
Shutdowns/Tech Spec Violations, 50.73(a)(2)(i)	41%	31%
Unanalyzed Conditions, 50.73(a)(2)(ii)	4%	5%
ESF Actuations, 50.73(a)(2)(iv)	43%	31%
Preventing Fulfillment of Safety Function, 50.73(a)(2)(v)	6%	3%
Single Cause Affecting Multiple Trains, 50.73(a)(2)(vii)	4%	39%
Other	14%	0%
No Reportability Checked	1%	3%

*NOTE: Percentages will be greater than 100 since more than one reportability category can be assigned to one LER. There were 2 "proprietary" and 10 "cancelled" Westinghouse LERs excluded from this analysis.

TABLE 2
1987 COMPARISON OF LER REPORTABILITY CATEGORIES FOR INDIAN POINT 2 AND ALL WESTINGHOUSE PLANTS

Reportability Category	Percentage of LERs*	
	Westinghouse	Indian Point 2
Limiting Condition for Operation, 50.36(c)(2)	1%	0%
Shutdowns/Tech Spec Violations, 50.73(a)(2)(i)	44%	30%
Unanalyzed Conditions, 50.73(a)(2)(ii)	4%	25%
ESF Actuations, 50.73(a)(2)(iv)	39%	15%
Preventing Fulfillment of Safety Function, 50.73(a)(2)(v)	7%	25%
Single Cause Affecting Multiple Trains, 50.73(a)(2)(vii)	3%	5%
Other	11%	5%
No Reportability Checked	1%	10%

*NOTE: Percentages will be greater than 100 since more than one reportability category can be assigned to one LER. There were 13 "proprietary" and 3 "cancelled" Westinghouse LERs excluded from this analysis.

TABLE 3
1988 COMPARISON OF LER REPORTABILITY CATEGORIES
FOR INDIAN POINT 2 AND ALL WESTINGHOUSE PLANTS

Reportability Category	Percentage of LERs* Westinghouse	Indian Point 2
Limiting Condition for Operation, 50.36(c)(2)	2%	0%
Shutdowns/Tech Spec Violations, 50.73(a)(2)(i)	43%	25%
Unanalyzed Conditions, 50.73(a)(2)(ii)	3%	0%
ESF Actuations, 50.73(a)(2)(iv)	38%	50%
Preventing Fulfillment of Safety Function, 50.73(a)(2)(v)	6%	0%
Single Cause Affecting Multiple Trains, 50.73(a)(2)(vii)	5%	25%
Other	14%	0%

*NOTE: Percentages will be greater than 100 since more than one reportability category can be assigned to one LER. There were 2 "proprietary" and 1 "cancelled" Westinghouse LERs excluded from this analysis.

UNPLANNED REACTOR SCRAMS

The Indian Point 2 scram data was evaluated to compare the number of scrams and the scram rates for Indian Point 2, Indian Point 3, the 83 mature plants and the 34 WE mature plants. The overall number of scrams showed a decrease over the evaluation period for Indian Point 2 and the mature plants as shown in Tables 4 and 5.

TABLE 4
UNPLANNED REACTOR SCRAMS

	1986	1987	1988 (1st Qtr.)
Indian Point 2	10	2	1
Indian Point 3	8	5	2
Mature Plants	328	220	49
WE Mature Plants	148	80	26

TABLE 5
UNPLANNED REACTOR SCRAM RATES PER 1000 CRITICAL HOURS

	1986	1987	1988 (1st Qtr.)
Indian Point 2	1.96	.32	.60
Indian Point 3	1.22	.91	.93
Mature Plants	.69	.45	.38
WE Mature Plants	.70	.38	.47

The causes of scrams at Indian Point 2 indicate that there were equipment problems during 1986. Many of these problems appear to have been resolved during the refueling outage in 1987. The Systematic Assessment of Licensee Performance (SALP) April 1988 Report states that there is still a backlog of maintenance to be performed but that the material condition of the plant is improving due to the amount of maintenance being performed. The April 1988 SALP Report states "The material condition of the plant is improving as evidenced by fewer reactor trips. Less time in Limiting Condition for Operation (LCO) action statements and fewer control room annunciators continuously energized."

It also stat[illegible]"Although the recent trip hist[illegible] shows a significant reduction in trip rate, the effectiveness of the Licensee's trip reduction efforts must be shown by consistent performance overtime." Table 6 provides the cause breakdown as percents of the total.

TABLE 6
UNPLANNED REACTOR SCRAM CAUSES

1986

Causes	Indian Pt. 2		Indian Pt. 3		Mature		WE	
Equipment	(8)	80%	(5)	63%	(189)	58%	(82)	55%
Personnel Error	(1)	10%	(2)	25%	(83)	25%	(40)	27%
Procedure	(1)	10%	(1)	13%	(20)	6%	(10)	7%
Others	(0)	-	(0)	-	(36)	11%	(16)	11%

1987

Causes	Indian Pt. 2		Indian Pt. 3		Mature		WE	
Equipment	(1)	50%	(4)	80%	(139)	63%	(47)	59%
Personnel Error	(1)	50%	(0)	-	(52)	24%	(22)	28%
Procedure	(0)	-	(0)	-	(10)	5%	(3)	4%
Others	(0)	-	(1)	20%	(19)	9%	(8)	10%

1988 (1st Qtr.)

Causes	Indian Pt. 2		Indian Pt. 3		Mature		WE	
Equipment	(0)	-	(2)	100%	(33)	67%	(17)	65%
Personnel Error	(1)	100%	(0)	-	(9)	18%	(4)	15%
Procedure	(0)	-	(0)	-	(3)	6%	(2)	8%
Others	(0)	-	(0)	-	(4)	8%	(3)	12%

The number of associated failures of a component immediately after a scram in 1986 was significant for Indian Point 2. Seven of the ten (70%) scrams in 1986 had an associated failure after the scram.

Electrical system failure	(1)
Safety injection failure to start	(1)
Steam dumps opening	(1)
Chemical and volume control	(2)
Auxiliary feed system failure	(2)

Descriptions of the events and associated failures can be found in Appendix A. The LERs that report associated failures with the scram are 24786001, 24786017, 24786019, 24786021, 24786024, 24786035 and 24786036. Table 7 provides an associated failure breakdown as percents of the total.

TABLE 7
UNPLANNED REACTOR SCRAM ASSOCIATED FAILURES

	Indian Pt. 2		Indian Pt. 3		Mature		WE	
1986	(7)	70%	(1)	13%	(58)	18%	(28)	19%
1987	(1)	50%	(1)	20%	(47)	21%	(22)	28%
1988	(0)	-	(0)	-	(10)	20%	(5)	19%

Indian Point 2 also had an electrical failure in 1987 similar to the electrical failure in 1986 after a scram. LER 24787004 documents the electrical failure and a pinhole leak on a charging pump. This event was encoded as one associated failure although more than one component failed.

UNPLANNED ENGINEERED SAFETY FEATURES ACTUATIONS

The Indian Point 2 Engineered Safety Features (ESF) Actuation data was evaluated in relation to Indian Point 3, the 83 mature plants and the 34 WE mature plants. Table 8 shows that Indian Point 2 has about 50% less ESFs than the mature plant average and is an average WE plant.

TABLE 8
UNPLANNED ENGINEERED SAFETY FEATURES ACTUATIONS

	1986	1987	1988
Indian Point 2	5	4	1
Indian Point 3	2	4	0
Mature Plant Average	9.69	9.16	1.43
WE Mature Plant Average	5.15	5.18	1.00

The 1986 ESFs all occurred during a scram, three events were due to Emergency Core Cooling System (ECCS) actuations and the other two were due to emergency diesel generator starts that occurred because of low voltage conditions after or during the scram. The LERs in Appendix A are 24786001, 24786017, 24786019, 24786024 and 24786031. The 1987 ESF events included two emergency diesel generator starts. LER 24787004 reported a diesel generator start after the 6.9 KV bus did not auto transfer after a scram. LER 24787013 reported that a diesel did not start on loss of offsite power due to being out of service for maintenance. Two Weld Channel Penetration Pressurization System isolations were reported in LERs 24787010 and 24787012. The event reported in LER 24787010 was the release of a small amount of gaseous radioactivity from the reactor vessel head during maintenance. The 1988 ESF LER 24788001 reported a main steam safety relief lifting during a relief valve stroke test causing an ECCS actuation and an RPS actuation with no rod movement. Table 9 provides the cause breakdown as percents of the total.

TABLE 9
UNPLANNED ENGINEERED SAFETY FEATURES ACTUATION CAUSES

1986

Causes	Indian Pt. 2		Indian Pt. 3		Mature		WE	
Equipment	(4)	80%	(1)	50%	(363)	45%	(72)	41%
Personnel Error	(1)	20%	(1)	50%	(230)	29%	(54)	31%
Procedure	(0)	-	(0)	-	(67)	8%	(20)	11%
Others	(0)	-	(0)	-	(144)	18%	(29)	17%

1987

Causes	Indian Pt. 2		Indian Pt. 3		Mature		WE	
Equipment	(1)	25%	(2)	50%	(343)	45%	(88)	50%
Personnel Error	(1)	25%	(2)	50%	(242)	32%	(55)	31%
Procedure	(1)	25%	(0)	-	(83)	11%	(19)	11%
Others	(1)	25%	(0)	-	(92)	12%	(14)	8%

1988

Causes	Indian Pt. 2		Indian Pt. 3		Mature		WE	
Equipment	(1)	100%	(0)	-	(53)	45%	(15)	44%
Personnel Error	(0)	-	(0)	-	(34)	29%	(7)	21%
Procedure	(0)	-	(0)	-	(20)	17%	(8)	24%
Others	(0)	-	(0)	-	(12)	10%	(4)	12%

SAFETY SYSTEM FAILURES

The Indian Point 2 Safety System Failure (SSF) data was evaluated in relation to Indian Point 3, the 83 mature plants and the 34 mature plants. Table 10 shows that Indian Point 2 reports approximately the same number of SSF events as mature plants and is an average Westinghouse plant.

TABLE 10
SAFETY SYSTEM FAILURES

	1987	1988 (1st Qtr.)
Indian Point 2	4	0
Indian Point 3	2	0
Mature Plant Average	3.95	0.81
Westinghouse Mature Plant Average	4.08	0.67

Indian Point 2 reported four SSF events during the first half of 1987 and has not reported any since. The 1987 SSF events are:

24787002 (Mode 3, 0% Power, 01/30/87)

Two of four Main Steam Isolation Valves failed to close on demand during a shutdown. The valves were manually closed. All indications were that the valve problems were related to excessive friction. The Indian Point 2 Final Safety Analysis Report (FSAR) assumes the single failure of one of the four MSIVs to close following a postulated main steam line break. The failure of more than one MSIV to close would result in the blowdown of more than one steam generator which is beyond the assumptions of the steam line break safety analysis. Past similar events throughout the industry prompted a modification in the packing design that was completed in 1984 at Indian Point 2. Since the new packing design was installed, lubrication of the MSIV packing was not performed due to an oversight in the preventative maintenance schedule.

24787006 (Mode 1, 100% Power, 04/30/87)

As assessment of the Indian Point 2 Auxiliary Feed Water System (AFW) identified a potential single relay failure that could disable the auto-start function of the two motor driven AFW pumps for several actuation signals: steam generator low-low water level, trip of any main feedwater pump, or loss of offsite power concurrent with a unit trip. Safety injection of the electric AFW pumps would remain. The auto-start feature of the steam operable driven AFW pump on steam generator low-low water level would not be affected and manual actuation of the electric AFW pumps would also be possible.

24787007 (Mode 1, 100% Power, 06/03/87)

Both control room booster fans were unable to achieve design flow during a routine surveillance test. Inspection of both fans indicated that the drive belts had stretched. Surveillance testing will be performed at more frequent intervals as a result of this event.

24787008 (Mode 1, 100% Power, 06/23/87)

Each of the two charcoal filters in the Central Control Room (CCR) Filtration System were replaced and a sample of the used charcoal was analyzed for methyl-iodine absorption efficiency. Results of the sample showed that the CCR charcoal filters were unable to meet the required absorption efficiency. Regulatory Guide 1.52, Rev. 2 criterion is based upon at least a 2 inch bed, whereas the Indian Point 2 charcoal bed is 1 inch. The cause of this event was the failure to recognize the differences between the Indian Point 2 design and the standard test configuration.

TECHNICAL SPECIFICATIONS

The Indian Point 2 Technical Specification (TS) data was evaluated in relation to the 83 mature plants and Indian Point 3. Indian Point 2 experienced 13 TS violations and two shutdowns required by TSs during the evaluation period. The overall number of TS violations show a marked decrease over the evaluation period. The opposite is true for the mature plants as shown in Table 11.

TABLE 11
TECHNICAL SPECIFICATION VIOLATIONS

	1986	1987	1988
Indian Point 2	13	6	1
Indian Point 3	3	10	0
Mature Plant Average	8.6	8.8	2.2

Only the data through March of 1988 was used in the TS evaluation. The TS data relies completely on License Event Reports (LERs) which are not required for 30 days after the event date or discovery date. When mailing and processing time is added, the latest complete month for LERs would be March of 1988.

The cause of the TS violations differ greatly from that of the mature plants. Table 12 provides the cause breakdown as a percent of the total.

TABLE 12
TECHNICAL SPECIFICATION VIOLATION CAUSES

1986

	Indian Point 2	Indian Point 3	Mature
Equipment	46%	50%	21%
Personnel Error	46%	50%	45%
Procedure	8%	--	27%
Others	--	--	7%

TABLE 12 (continued)

1987

	Indian Point 2	Indian Point 3	Mature
Equipment	51%	20%	22%
Personnel Error	16%	20%	48%
Procedure	16%	60%	27%
Others	16%	--	3%

1987

	Indian Point 2	Indian Point 3	Mature
Equipment	100%	--	22%
Personnel Error	--	--	42%
Procedure	--	--	30%
Others	--	--	6%

At Indian Point 2, equipment problems are the primary cause of TS violations. At Indian Point 3 and the mature plants, the primary cause is personnel error. A summary of the Indian Point 2 TS LERs is contained in Appendix A. The LERs associated with the equipment causes are: 24788003, 24787007, 24787015, 24787016, 24786013, 24786009, 24786011, 24786022, 24786004, 24786029, 24786038(shutdown), 24786018(shutdown). These LER numbers include the two shutdown events as these events were also the result of equipment problems.

Two of the 1986 events document the same problem. LERs 24786011 and 24786022 document a problem with the level transmitters on the Refueling Water Storage Tanks. The level instruments were not suitable for the application and were scheduled for replacement. The surveillance schedule for the instruments was reduced to monthly until replacement. No further documentation of the problem was located in the plant's LERs.

Three of the events involved wear or erosion of valves. Erosion of the valve discs and nozzles, thermal cycling, and setpoint drift are possible causes documented for pressurizer relief valve setpoint problems in 1986 and 1987 (LERs 24786004 and 24787016). As of March 1988, the problem has not been documented in 1988. The third event involved normal wear of a Nitrogen System check valve. The condition was not discovered earlier because a test procedure did not exist. Corrective action included writing a test procedure and requiring a period test (LER 24787015).

The event documented in LER 24787007 also involves a Safety System Failure. The cause was stretched fan belts on Control Room Ventilation fans. The corrective action included reduction of the surveillance interval.

A final event that warrants discussion is documented in LER 24786013. Control and power cables to a Safety Injection motor operated valve were discovered to be routed through a penetration that was not electrically qualified. In a letter in September 1981, the plant stated that no safety related electrical equipment was subject to submergence inside or outside containment. However, the subject penetration could become submerged during a LOCA. The cables were rerouted as required.

The remaining equipment problems are not recurring or share a common problem. Several of the remaining events document primary equipment failures.

POTENTIAL PRECURSORS

The AEOD Accident Sequence Precursor contained two events in the 1986 data as potential precursors to core melt. These events are documented in LERs 24786017 and 24786035. Both of these events involved equipment problems as the cause and resulted in an unplanned reactor scram. Additionally, equipment failures occurred after the scram. During the event in LER 24786017, the train 'B' safety injection did not automatically actuate and had to be manually initiated. In LER 24786035, Auxiliary Feedwater tripped after auto start and had to be manually started while the steam relief valve to the turbine driven pump lifted.

APPENDIX A

INDIAN POINT 2 DATA LISTINGS

1986 INDIAN POINT 2 LERs

LER Number	Event Date	Title
24786001	01/13/86	Main boiler feedwater pump trip/reactor trip.
24786002	01/14/86	Steam generator relief valve setpoints exceed specification.
24786003	01/17/86	Defective relay caused test failure of manual safety injection.
24786004	01/31/86	Pressurizer safety valves outside range for setpoint pressure.
24786005	02/07/86	Replacement of components with incomplete EQ documentation.
24786006	02/26/86	EQ not documented for thermocouple cold reference junction in H_2 recombiners.
24786007	03/07/86	Environmental qualification of main steam flow transmitters.
24786008	02/05/86	Degraded bus voltage relay setpoint drift.
24786009	03/15/86	Low pressure safety injection setpoint out of specification.
24786010	03/20/86	Inadequate alarm response procedure for high energy line break.

24786011	03/22/86	Refueling water storage tank level transmitters set points below specified level.
24786012	03/03/86	Plant vent gas releases without completion of surveillance test for Monitor R-14.
24786013	05/01/86	Electrically unqualified penetration during postulated submergence.
24786014	05/18/86	Disconnect seismic restraint above cold shutdown due to pump maintenance procedure inadequacy.
24786015	05/21/86	Excessive leakage of main steam safety relief valves.
24786016	05/23/86	Actuation of reactor protection system.
24786017	05/28/86	Reactor trip due to steam dump valves opening.
24786018	06/06/86	Inoperable battery charger and battery.
24786019	06/09/86	Manual reactor trip with safety injection actuation.
24786020	06/24/86	Two containment pressure instruments became inoperable due to static inverter trip.

24786021	06/25/86	Reactor trip due to turbine trip circuit de-energization.
24786022	06/30/86	Refueling water storage tank level transmitters set points outside specified levels.
24786023	06/30/86	Rod exercise surveillance interval exceeded when procedure not followed.
24786024	07/18/86	Reactor trip.
24786025	07/18/86	Loss of instrument air to auxiliary feedwater system.
24786026	07/24/86	Hydrogen-oxygen monitor malfunction.
24786027	08/02/86	Unit trip due to loss of main boiler pump.
24786028	08/04/86	Refueling water storage tank level instruments.-
24786029	08/06/86	Diesel generator breaker failure.
24786030	08/12/86	Violation of containment integrity at personnel air lock.
24786031	09/16/86	Control rod drop - reactor scram/SI actuation.
24786032	09/22/86	Missed surveillance test.
24786033	10/15/86	Emergency diesel generator 22 breaker malfunction.

24786034	10/16/86	Two inoperable overcompensated/ overpower channels due to controller failure.
24786035	10/20/86	Reactor trip due to reactor trip relay de-energizing.
24786036	10/23/86	Manual reactor trip due to loss of main boiler feed pump.
24786037	11/06/86	Reactor trip due to malfunction of relay.
24786038	11/14/86	Failure of safety injection pump due to seizing completion of hot shutdown.
24786039	12/17/86	Misalignment causes inoperable control rod.

1987 INDIAN POINT 2 LERs

LER Number	Event Date	Title
24787001	01/21/87	Erroneous declaration of total diesel generator unavailability.
24787002	01/30/87	Main steam isolation valves failure to close.
24787003	02/02/87	Excessive closing time of condensate storage tank isolation valve with two motor driven auxiliary feedwater pumps inoperable.
24787004	02/10/87	Reactor trip due to operator error.
24787005	03/18/87	Inoperable diesel generators.
24787006	04/30/87	Single failure would render both motor driven AFWS inoperable.
24787007	06/03/87	Failure of two booster fans render control room ventilation inoperable.
24787008	06/23/87	Central control room charcoal filters - low methyl-iodine adsorption efficiency.
24787009	06/27/87	Reactor trip due to steam generator level relay malfunction.
24787010	10/08/87	High radioactivity in containment causes operation of ESF.

24787011	10/09/87	Failure of service water pumps during surveillance test.
24787012	10/19/87	Electrical power supply spike in containment causes operation of ESF.
24787013	11/05/87	Inadvertent actuation of SIS causes ESF operation.
24787014	11/06/87	Procedural deficiency prevents obtaining 'as found' test data.
24787015	11/18/87	Inoperability of backup nitrogen supply to PORVs.
24787016	12/11/87	Pressurizer safety valves actuation setpoint tolerance.
24787017	12/01/87	Environmental qualification of electrical splices.
24787018	12/04/87	Common RHR recirculation line can lead to pump failure.
24787019	12/08/87	Accumulator tank level instrument calibration level error.
24787020	12/31/87	Environmental qualification of resistance temperature detectors.

1988 INDIAN POINT 2 LERs

LER Number	Event Date	Title
24788001	01/17/88	Reactor trip during valve stroke test.
24788002	01/25/88	Reactor trip on intermediate range high flux.
24788003	03/23/88	Technical specification limit exceeded IVSWS.
24788004	04/01/88	Instrument maintenance recirculation pumps inoperable.

24786001 (Mode 1, 81% power, 01/13/86)
Internal high pressure hose to MFP governor control oil system ruptured causing loss of MFW and reactor trip on Low SG Level. 6.9 KV Bus 2 failed to auto transfer causing diesel generator to start and supply 480V Bus.

24786016 (Mode 3, 0% power, 05/23/86)
Technician opened wrong reactor trip breaker during reactor protection logic channel functional testing. This caused RPS actuation and the rods which were withdrawn to drop.

24786017 (Mode 1, 30% power, 05/28/86)
Faulty steam dump controller opened all 12 steam dump valves causing high steam flow coincident with low reactor coolant temperature reactor trip. Train 'B' of the safety injection did not automatically actuate and a second safety injection signal was initiated starting Safety Injection 'B'.

24786019 (Mode 1, 56% power, 06/09/86)
Incorrect low bearing oil trip setpoint caused MFP trip when starting a second MFP supplied by the same oil pump. Manual scram due to loss of MFW. Steam dumps opened after trip due to a failed bistable causing a high steam flow condition safety injection actuation.

24786021 (Mode 1, 43% power, 06/25/86)
Inadequate circuit elementary diagram used by technician during relay replacement did not identify trip circuit which tripped main turbine and reactor scram. During recovery pressure relief line for charging pump 23 parted causing a chemical and volume control leak.

24786024 (Mode 1, 100% power, 07/18/86)
When restoring electrical system to normal, 6.9 KV to 480 V breaker failed to close causing an undervoltage condition. Diesel generators supplied bus. Rod control MG Set 22 tripped. Operator attempted to restart MG set and return it to service. The operator set output voltage too low when paralleling the MG set to control rod drive system, voltage went too low causing control rods to drop. Seal injection filter in chemical and volume control system developed a leak on the scram recovery.

24786027 (Mode 1, 100% power, 08/02/86)
Worn shaft seal on MFP caused low oil pressure MFP trip and subsequent scram on low SG level.

24786031 (Mode 1, 100% power, 09/16/86)
Temporary jumper installed during biweekly rod exercise test had higher resistance than expected. This caused three rods to drop necessitating a manual scram. Safety injection actuated due to sensed high steam flow condition when main turbine first stage pressure decreased rapidly due to turbine stop valve closure.

24786035 (Mode 1, 100% power, 10/20/86)
Loose connections in relay rack of reactor trip breaker 'B' caused breaker to open during testing. This caused a reactor scram. One motor driven auxiliary feed pump (AFP) tripped after auto starting - restarted manually. Relief valve in steam line to turbine driven AFP lifted due to setpoint being incorrect.

24786036 (Mode 1, 38% power, 10/23/86)
Failed MFP discharge check valve caused backflow through idle MFP. When check valve isolated, MFP tripped on high discharge pressure necessitating a manual scram. Turbine driven AFW pump tripped after auto start signal.

24786037 (Mode 1, 97% power, 11/06/86)
Three faulty relays in the reactor protection logic caused reactor trip breaker 'B' to open during testing.

24787004 (Mode 1, 100% power, 02/10/87)
Operator opened wrong breaker when returning diesel DC control power to normal. Operator took switch for wrong breaker - 'on' to 'off' versus right breaker 'off' to 'on' causing loss of power to reactor protection Train 'B' and reactor scram. 6.9 KV Bus 2 failed to transfer from station power to offsite power de-energizing RCP #24. Diesel generators started. Later when trying to match switch positions, 6.9 KV Bus 3 de-energized and RCP #23 de-energized. Pinhole leak developed on charging pump #23.

24787009 (Mode 1, 100% power, 06/27/87)
Two faulty relays in SG level circuit failed caused false low SG level signal when resetting SG low level setpoints. This caused a reactor scram.

24788001 (Mode 3, 0% power, 01/17/88)
During stroke test of atmospheric relief valve for #23 steam line, steam line safety relief lifted causing high steam flow signal safety injection and RPS actuation with no rod motion.

24788002 (Mode 1, 10% power, 01/25/88)
Conservative IRM setpoint 15% vs 25%. The rapid loading of the main generator caused an IRM high flux reactor scram.

24786001 (Mode 1, 80% power, 01/13/86)
Internal high pressure hose to MFP governor control oil system ruptured causing loss of MFW and reactor trip on Low SG Level. 6.9 KV Bus 2 failed to auto transfer causing diesel generator to start and supply 480V Bus.

24786017 (Mode 1, 30% power, 05/28/86)
Faulty steam dump controller opened all 12 steam dump valves causing high steam flow coincident with low reactor coolant temperature reactor trip. Train 'B' of the safety injection did not automatically actuate and a second safety injection signal was initiated starting Safety Injection 'B'.

24786019 (Mode 1, 56% power, 06/09/86)
Incorrect low bearing oil trip setpoint caused MFP trip when starting a second MFP supplied by the same oil pump. Manual scram due to loss of MFW. Steam dumps opened after trip due to a failed bistable causing a high steam flow condition safety injection actuation.

24786024 (Mode 1, 100% power, 07/18/86)
When restoring electrical system to normal, 6.9 KV to 480V breaker failed to close causing an undervoltage condition. Diesel generators supplied bus. Rod control MG Set 22 tripped. Operator attempted to restart MG set and return it to service. The operator set output voltage too low when paralleling the MG set to rod drive system, voltage went too low causing control rods to drop. Seal injection filter in chemical and volume control system developed a leak on the scram recovery.

24786031 (Mode 1, 100% power, 09/16/86)
Temporary jumper installed during biweekly rod exercise test had higher resistance than expected. This caused three rods to drop necessitating a manual scram. Safety injection actuated due to sensed high steam flow condition when main turbine first stage pressure decreased rapidly due to turbine stop valve closure.

24787004 (Mode 1, 100% power, 02/10/87)
Operator opened wrong breaker when returning diesel DC control power to normal. Operator took switch for wrong breaker - 'on' to 'off' versus right breaker 'off' to 'on' causing loss of power to reactor protection Train 'B' and reactor scram. 6.9 KV Bus 2 failed to transfer from station power to offsite power de-energizing RCP #24. Diesel generators started. Later when trying to match switch positions, 6.9 KV Bus 3 de-energized and RCP #23 de-energized. Pinhole leak developed on charging pump #23.

24787010 (Mode 5, 0% power, 10/08/87)
Slight pressure buildup in reactor vessel (1.5 psi) caused release of noble gas which exceeded setpoint on weld channel penetration pressurization system (part of containment purge system).

24787012 (Mode 5, 0% power, 10/19/87)
Electrical spike in the electrical supply to containment particulate radiation monitor caused containment ventilation valves to isolate.

24787013 (Mode 5, 0% power, 11/05/87)
Inadequate procedure did not pre-establish parameter conditions causing loss of 480 VAC vital busses during preventative maintenance. Diesels tagged out for maintenance.

24788001 (Mode 3, 0% power, 01/17/88)

During stroke test of atmospheric relief valve for #23 steam line, steam line safety relief lifted causing high steam flow signal safety injection and RPS actuation with no rod motion.

24787002	Two MSIV's failed to fully close. Valves manually shut and packing lubricated, subsequently operating correctly. Valve packing lubrication was not performed since 1984 due to an error in the preventative maintenance schedule.
24787006	Potential single failure could render both motor driven Auxiliary Feedwater (AFW) pumps inoperable. Plant entered LCO and corrected deficiency in starting relays.
24787007	Control Room Ventilation System booster fans unable to achieve design flow rendering Control Room Ventilation inoperable. Drive belts adjusted, fans met design limits.
24787008	"Adsorption efficiency" of control room charcoal filter less than Technical Specification limit. Caused by change in regulatory requirements after system designed and installed.

24788003 03/23/88 Equipment
Leakage from the isolation valve seal water system exceeded the T.S. limits.

24787007 06/03/87 Equipment
Both control room ventilation fans could not achieve design flow in accordance with T.S. Fan belts had stretched and slippage was taking place. System balance damper was also out of adjustment.

24787015 11/18/87 Equipment
The nitrogen system was found to be inoperable due to normal check valve wear. The nitrogen system provides backup protection for PORV's.

24787016 12/11/87 Equipment
A pressurizer safety relief valve was found to have setpoint valves outside T.S. 3.1.A.3.C limits due to erosion of the valve disc, thermal cycling, and setpoint drift. Valve adjusted.

24787002 02/02/87 Personnel Error
Due to a preventive maintenance oversight, the packing on two MSIV's were not lubricated. When valves were closed during a shutdown, the valves did not close all of the way. T.S. 3.4.A.5 and 4.7.

24787003 02/03/87 Other
Due to an isolation valve failure, both AFW pumps being inoperable and an extended period at hot shutdown, the condensate storage tank level dropped below T.S. limit. T.S. 3.4.A(3).

24787019 2/08/87 Unknown

The plant has been operating with accumulator water volume range in excess of T.S. 3.3.A.1.C. This displaces some of the N2 that is required as a driving head for accumulator water injection following a LOCA.

24786013 05/01/88 Equipment

Control and power cables to a safety injection motor operated valve were routed through a penetration that was not electrically qualified and could become submerged during a LOCA. T.S. 3.3.A.2.D was violated.

24786009 03/15/86 Equipment

The low pressure safety injection setpoints for three (3 loops) pressure transmitters were found to be less than T.S. requirements and all were declared inoperable.

24786011 03/22/86 Equipment

Refueling water storage tank level transmitters did not meet the trip requirements of T.S. 3.3.A.1.K. Instrument is not suitable for the tank design.

24786012 03/03/86 Personnel Error

Monthly source check of vent noble gas monitor was not performed as required by T.S. 4.10-4. Additionally 15 releases were performed without meeting the requirements of T.S. 3.9-2. Technician signed-off satisfaction with incomplete test procedure.

24786022 06/30/86 Equipment

The low level alarm setpoint for the refueling water storage tank was lower than T.S. limits. Injection spray, continued after switchover to recirculation during a LOCA, would have terminated slightly early.

24786014 05/18/86 Procedure

A seismic restraint was disconnected from a RCP seal leak-off line and not re-connected before entering hot shutdown mode per T.S. 3.12.1 and 3.12.4.

24786004 01/31/86 Equipment

Three pressurizer relief valves were outside T.S. 3.1A.3.C limits due to erosion of the valve discs and nozzles, thermal cycling and setpoint drift.

24786029 08/06/86 Equipment

Diesel generator #23 output breaker failed when attempting to close. Subsequently, the breaker burned up when being racked-in. Plant entered T.S. 3.0.1. Single failure disabled multiple components in independent systems.

24786039 12/17/86 Personnel Error

Two control rods were misaligned from the demand position greater than allowed by T.S. and LCO time limit was exceeded. Condition not detected sooner because the rod position deviation monitor was inadvertently inoperable.

24786038 11/14/86 Equipment

Safety injection pump seized due to mechanical failure and repair time exceeded that allowed by T.S. Plant shutdown per Technical Specifications.

24786032 09/22/86 Personnel Error

Daily grab sample of the containment atmosphere was not taken per T.S. 3.1.F.1.B.6 interval. Sample was two hours and 40 minutes late. Communication error among chemist personnel.

24786026 07/24/86 Personnel Error

Waste gas decay system hydrogen-oxygen monitor was inoperable due to recorder problem and being in manual, but 24 hour grab sample and readings were not required per T.S. 3.9.B.2.

24786026 07/24/86 Personnel Error

Waste gas decay hydrogen-oxygen monitor channel functional test did not meet T.S. 4.10.B.3. Erroneous interpretation of chemistry staff caused utilization of procedure that did not check the alarm function as required.

24786018 06/06/86 Equipment

A battery charger was found to have zero output current and voltage. Subsequently, it was determined that cell voltage was determined to be below minimum acceptable value for battery 21. Plant shutdown per Technical Specifications.