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Initial Quantification of Human Errors Associated with Reactor Safety System Components in Licensed Nuclear Power Plants

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ABSTRACT

This report provides a methodology for an initial quantification of gross categories of human errors made in conjunction with several reactor safety system components operated, maintained, and tested in licensed nuclear power plants. The resultant human error rates provide the first real systems bases of comparison for the existing derived and/or best judgement equivalent set of such rates or probabilities.

I. INTRODUCTION AND SUMMARY

This report provides an initial quantification of gross categories of human error by applying a simple model of the number of human errors divided by the opportunities for the specific errors associated with the operation and testing of licensed nuclear power plants. This evaluation was performed in conjunction with an error event summarization developed by EG&G Idaho, Inc. (EG&G) which has been adapted by Brookhaven National Laboratory (BNL) for specific human error events. To determine the number of opportunities for human error, BNL has developed and demonstrated a reasonable and workable methodology for estimating the appropriate number of human actions or interfaces with selected components in the reactor safety systems. This methodology is based on personal licensed senior operator experience with the operations and testing of these components.

This work has resulted in the development of a broad set of human error rate (HER) categories covering operating errors. The errors occur during the human interface with remotely operated valves (ROV), manual valves (XV), and pumps in reactor safety systems of both the pressurized water reactors (PWR) and boiling water reactors (BWR) plants. These prescribed components (ROV, XV, and pumps) were selected for the quantification effort since they represent a suitable and available human error data base as provided by EG&G. A complete summary of all HERs calculated from the LER data by this work is contained in Table 5.

The BNL calculated human-component interfaces developed in this report, when used with appropriate human error data, provide the first independent, nuclear systems approach to developing HERs as a benchmark of comparison for existing derived and/or best judgement human error rates. When the appropriate derived and best judgement human error rates from the WASH-1400 study

are compared to the calculated HERs, they are in general order of magnitude agreement (Section VII and Table 5). However, more detailed analyses and comparisons are needed to draw specific conclusions since the work described in the report does not, as of yet, lend itself to the application of performance shaping factors (PSF). The HERs developed and presented in this study are also in general agreement with current literature; however, the reader is cautioned that the results are of a broad and general nature. Refinements in these areas are continuing and will be reported in future documents.

One of the principal goals of the work was to develop a systematic, repeatable and tractable method to convert reported Licensee Event Report (LER) human errors into error rates. This has been successfully accomplished by approaching the problem from a systems engineering task analysis direction. The methodology used for determining the number of chances that the human had to make the reported LER errors is discussed in later sections of this document. This approach will provide a method to estimate human interaction and develop more specific HERs where error data exists, and lends itself to use by systems and reliability engineers.

This first engineering attempt at quantifying HERs from existing LER data represents a potential for application in design and risk assessments. Further work is needed to move from the general HERs to specific ones and to improve usefulness of the results by introducing significant PSFs.

II. QUANTIFICATION OF HUMAN ERROR - DEFINITION OF HUMAN ERROR RATE (HER)

In this report, the quantification of human error is presented in terms of Human Error Rate (HER). The HER is defined as the frequency of the occurrence of human error given the number of opportunities (i.e., demands) for that particular event. In other words

$$\text{Human Error Rate (HER)} = \frac{\text{Human Errors}}{\text{Opportunities for those Errors}}$$

or,

$$\text{Human Error Rate (HER)} = \frac{\text{Human Errors}}{\text{Human Events (Errors and Successes)}}$$

III. HUMAN ERROR DATA

The human error data utilized in this study was extracted from the EG&G Idaho, Inc. (EG&G) computer-based file of one-line data summary descriptive interpretations of Licensee Event Reports (LER) (References 1&2). Three broad categories of reactor safety system components, namely, remotely operated valves (ROV), manual valves (XV), and pumps have been selected as the interfacing components of this study. Appendix A contains a listing of all the actual personnel error related LER data summaries used in this study grouped by component categories. These categories coupled with the human error related LER data summaries, provide a reported data base suitable for calculating human error rates (HERs). Since the human errors used in this study are all obtained from reported events in LER's, there is no attempt to calculate the number of times that a human error was made which did not result in an LER. The following sections of this study addresses, in a qualitative fashion, this and other uncertainties in the human error data used.

The time frame used for selecting the human error data for valves is a three-year period from January 1, 1976 through December 31, 1978. During that period a more uniform reporting system among plants was made possible by changes in the rules which govern LER reporting. In addition, all new plants licensed after January 1, 1976 have used Standardized Technical Specifications (STS). This creates more uniform reporting than for older plants with more individualized Technical Specifications. To provide as much uniformity as

possible between the human error data selection period and time frame for pumps and valves, the pump data used in this study also covers a three-year period. However, the time frame is eight months earlier, (May 1, 1975 to April 30, 1978) since this latter date marks the end of data reported for pumps in Reference 1. This eight month difference should have no appreciable effect on the HERs generated in this report.

There were 64 licensed operating plants considered for human error data in the 1975 thru 1978 period. These include 41 PWR vendor plants: 9 Babcock and Wilcox (B&W), 7 Combustion Engineering (CE) and 25 Westinghouse (WX); and 23 BWR vendor plants, all General Electric (GE).

The systems used in this study relating to the gross categories of valves and pumps are the reactor safety systems, as shown in Table 1 below. This subset of systems was utilized due to their direct potential to impact public risk (Ref. 3 and 5).

TABLE 1

REACTOR SAFETY SYSTEMS EVALUATED

In a Pressurized Water Reactor (PWR)

- Auxiliary Feedwater (AFW)
- Containment Spray (CtS)
- High Pressure Safety Injection (HPSI) - including safety related charging pumps.
- Low Pressure Safety Injection (LPSI)

In a Boiling Water Reactor (BWR)

- Core Spray (CoS)
- High Pressure Coolant Injection (HPCI)
- Low Pressure Coolant Injection (LPCI) including Drywell and Suppression Chamber Spray.

LER data summaries for human error events with the selected safety system valves and pumps do not take into account all the number of events which have actually occurred. This is due to the possibility that some events may have been incorrectly reported as other than human error, or have not been reported at all. Possible reason(s) for these discrepancies include the following: (1) Original event report of the licensee - inaccurate, misleading or does not exist, (2) LER input from original report - inaccurate, misleading, or does not exist, (3) LER output to LER data summary - inaccurate or misinterpreted, and (4) LER data summary - misinterpreted.

This is the major factor for possibly making the calculated HERs low. It is anticipated that this uncertainty will be evaluated in the near future as part of a subsequent study. Some LER human error related events tabulated in the LER data summaries with a failure mechanism description of Procedural Discrepancy may actually be human errors of operations, maintenance, or testing. Thirteen total Procedural Discrepancy related LER data summaries for both valves and pumps are listed in Appendix A. The assumption that there is no operational human error data taken from these data summaries should not impact the HERs significantly.

Fifteen of the valve LER data summaries do not uniquely identify valve types so that they can be classified either as Remotely Operated Valves (ROV) or as manual valves (XV). Interpretations of the associated LER's have been performed to make the proper classification. Most of the pump LER data summaries have a failure mechanism narrative description which identifies a supporting component of the pump as the interfacing component (electrical breaker, trip/throttle, valve, etc.). Since ROV and XV are separate interfacing components in this study, they are intentionally not duplicated in the pump data used.

The following procedure was used in selecting applicable LER data:

- Human error events relating to valves in this report are obtained from LER data summaries having failure mechanism (code) descriptions of PERSONNEL (OPERATION), PERSONNEL (MAINTENANCE), AND PERSONNEL (TESTING). Those identified as DESIGN ERROR, FABRICATION/ CONSTRUCTION/QUALITY CONTROL, AND PROCEDURAL DISCREPANCY have not been used. See Appendix A for actual LER data summaries used from Reference 2.
- Human error events relating to pumps in this report are obtained from LER data summaries having a failure mechanism (code) description of PERSONNEL. Those identified as DESIGN ERROR or PROCEDURAL DISCREPANCY have not been used. See Appendix A for actual LER data summaries used from Reference 1.
- Performance Shaping Factors (PSF) have not been identified or analyzed since the LER data summaries do not lend themselves to such human behavior evaluation.

For the reactor safety systems listed in Table 1 and used in this report, the number of human error related LER data summaries are listed in Table 2 and have been categorized by interfacing component and reactor type.

TABLE 2

SUMMARY OF NUMBER OF REPORTED HUMAN ERRORS
ASSOCIATED WITH SELECTED REACTOR SAFETY SYSTEM*
INTERFACING COMPONENTS OVER A SPECIFIC THREE-YEAR PERIOD**

Reactor Safety System Interfacing Component	Reactor Type		
	PWR	BWR	Both
Remotely Operated Valves (ROV)	10	8	18
Manual Valves (XV)	11	2	13
Total Valves (ROV + XV)	21	10	31
Total Pumps	11	3	14

* see Table 1 for reactor safety systems involved.
**January 1, 1976 through December 31, 1978 for valves, May 1, 1975 through April 30, 1978 for pumps.
Note that Appendix A contains a listing of the actual LER data summaries used to develop this Table and a percentage breakdown by failure mode/mechanism.

IV. HUMAN COMPONENT INTERFACE

LER data summaries provide the number of LER's caused by personnel (human) error while interfacing with a ROV, a XV, or a pump in the three year period of interest. The number of opportunities for such human error in that same time frame must be developed based on the following.

Assumptions for Reactor Safety System Remotely Operated Valve (ROV) Opportunities

- Those remotely Operated Valves (ROV) which are needed to move in order to perform their intended safety related functions are assumed to be cycled quarterly for valve "testing" while the reactor is critical. This assumption is based on compliance with the ASME Boiler and Pressure Vessel Code, Section XI-Inservice Testing requirements. The reactor is assumed critical nine months of each year.

- Each ROV cycled quarterly is also assumed to be additionally cycled once a year in support of valve "maintenance." This assumption provides a reasonable approximation for maintenance isolation based on experience.
- Each ROV cycled quarterly is also assumed to be additionally cycled once a year in order that the valve be able to perform its normal "operational" function. This assumption provides an operational approximation based on experience with normal annual ROV operational requirements.
- Each ROV used to lineup a reactor safety system pump for its monthly test (see Assumptions for Reactor Safety Systems Pumps Opportunities paragraphs that follow) is assumed to be cycled once per pump test, or nine times a year, and once a year (on average) for normal pump operation.
- Each ROV used to isolate a reactor safety system pump (from its system) is assumed to be cycled once a year to isolate that pump for maintenance.

In summary, specific reactor safety system ROVs are assumed to be cycled as follows on an annual basis:

a. Five times as part of safety system valve testing, maintenance, and operations

b. Ten times in support of safety system pump testing and operations

c. One time in support of safety system pump maintenance.

Assumptions for Reactor Safety System Manual Valves (XV) Opportunities

- Manual Valves (XVs) are not assumed to be cycled for quarterly valve testing since XVs normally need not be moved to perform any intended safety related functions.

- Each XV used to lineup a safety system pump for its monthly test (see Assumptions for Reactor Safety System Pump Opportunities paragraphs that follow) is assumed to be cycled once per pump test, or nine times a year and once a year (on average) for normal pump operation.
- Each XV used to isolate a reactor safety system pump (from its system) is assumed to be cycled once a year to isolate that pump for maintenance.

In summary, specific XVs are assumed to be cycled as follows on an annual basis:

a. Ten times in support of safety system pump testing and operations
b. One time in support of safety system pump maintenance.

Assumptions for Reactor Safety System Pump (SSP) Opportunities

- Safety system pumps (SSP) are assumed to be tested monthly while the reactor is critical. This assumption is based on assumed compliance with the ASME Boiler and Pressure Vessel Code, Section XI Inservice Testing requirements. The reactor is assumed critical nine months of each year.
- Each SSP is assumed to be operated once a year on the average during normal SSP operation. This assumption is based on experience with normal annual SSP operational requirements.

In summary, SSPs are assumed to be cycled as follows on an annual basis:

Ten times as a part of pump testing and operations.

The number of reactor safety system valves (both ROV and XV) moved for both valve and pump testing have been determined for the representative plants from piping and instrumentation diagrams (P&ID's) and flow diagrams for specific reactor safety systems listed in Table 1. This was augmented by

using systems operational and testing experience. It should be noted that the Commonwealth Edison's Zion Unit 1 plant's specific operating and testing procedures were available to validate the assumptions. In addition, logic diagrams were used to verify valve interlock testing and integrated system valve testing for Zion Unit 1. Even though only Zion 1's operating and testing procedures were available, the P&ID and system's experience made analysis of the other representative plants quite reasonable.

Several safety system pumps (SSP) are normally used while: (1) operating at power (charging pumps in PWRs), (2) starting up (auxiliary feed-water pumps in PWRs), and (3) in low pressure shutdown condition (residual heat removal in both BWRs and PWRs). This will have little effect on the total number of SSP opportunities since these are relatively insignificant compared with those related to testing.

The procedure for representative plant selection started with a detailed P&ID (Piping and Instrumentation Diagram) review of all safety system manual and remotely operated valves and pumps has been performed on the following number and types of plants.

Arkansas Unit 1 (AR 1)	- 1 out of 9 Babcock and Wilcox PWRs
Millstone Unit 2 (MI 2)	- 1 out of 7 Combustion Engineering PWRs
Indian Point Unit 2 (IP 2), Point Beach Unit 1 (PB 1), Zion Unit 1 (ZI 1)	- 3 out of 25 Westinghouse PWRs
Pilgrim Unit 1 (PI 1), Browns Ferry Unit 1 (BF 1)	- 2 out of 23 General Electric BWRs
	7 out of 64 Total Licensed Plants

These plants were selected to represent their respective reactor vendors by simple averaging based on the following:

The extent to which these 7 plants are representative of the total population of 64 licensed plants introduces a level of uncertainty into this analysis, however, it is assumed that this will not affect the results of this gross quantification.

Using the reactor safety systems listed in Table 1, and the P&ID's for those plants identified previously in this section, Table 3 has been developed. Table 3 shows the number of valves and pumps cycled for various activities during testing, operation, and maintenance.

In Table 3, the column labeled "#PUMP/PUMP TEST" gives the number of pumps tested in each selected safety system. The columns labeled "#ROV/VALVE TEST", "#ROV/PUMP TEST" and "#ROV/PUMP MAINT" give the number of ROV's cycled in each selected safety system for each of the stated activities, namely valve test, pump test, and pump maintenance, respectively. In like manner, "#XV/PUMP TEST" and "#XV/PUMP MAINT" yields the quantity of XV's cycled in each selected safety system for each pump test and pump maintenance.

To determine the "SYSTEM ANNUAL" cycles for ROV, XV, and pumps as shown in the last three columns in Table 3, the previously discussed assumptions lead to the following annual demands.

- Safety system pumps, counted in the column labeled "#PUMP/ PUMP TEST", are assumed to be cycled ten times annually. This is based on nine times for monthly testing (reactor assumed critical nine months) and one time for all such pumps being operated once on the average during normal pump operations.
- Safety system valves, counted as requiring quarterly valve testing (column labeled "#ROV/VALVE TEST") are assumed to be cycled five times annually. This is based on three times for quarterly testing (reactor assumed critical nine months), one time in support of valve

TABLE 3

NUMBER OF SELECTED REACTOR SAFETY SYSTEM INTERFACING COMPONENTS
(namely Remotely Operated Valves(ROV), Manual Valves (XV) and Pumps)
CYCLED FOR VARIOUS ACTIVITIES
(Valve Test, Pump Test, or Pump Maintenance including Normal Operations)

PWR VENDOR*/ REACTOR SAFETY SYSTEM/PLANT	#PUMP/ PUMP TEST	#ROV/ VALVE TEST	#ROV/ PUMP TEST	#ROV/ PUMP MAINT	#XV/ PUMP TEST	#XV/ PUMP MAINT	#ROV/ SYSTEM ANNUAL	#XV/ SYSTEM ANNUAL	#PUMP/** SYSTEM ANNUAL
B&W/AFW/AR 1	2	4	0	8	0	2	28	2	20
B&W/CtS/AR 1	2	10	2	0	4	4	70	44	20
B&W/HPSI/AR 1	2	9	0	0	0	9	45	9	20
B&W/LPSI/AR 1	3	10	2	0	3	4	70	34	30
	AR 1 (Arkansas Unit 1) TOTALS B&W						213	89	90
CE/AFW/MI 2	3	5	6	6	0	3	91	3	30
CE/CtS/MI 2	2	8	4	2	0	4	82	4	20
CE/HPSI/MI 2	3	12	4	4	0	8	104	8	30
CE/LPSI/MI 2	2	8	4	0	0	8	80	8	20
	MI 2 (Millstone Unit 2) TOTALS CE						357	23	100
WX/AFW/IP 2	3	0	0	0	0	14	0	14	30
WX/CtS/IP 2	2	4	0	0	4	4	20	44	20
WX/HPSI/IP 2	2	18	0	4	0	5	94	5	20
WX/LPSI/IP 2	2	6	1	0	0	4	40	4	20
	IP 2 (Indian Point Unit 2) Totals						154	67	90
WX/AFW/PB 1	2	0	0	6	0	6	6	6	20
WX/CtS/PB 1	2	4	0	2	0	2	22	2	20
WX/HPSI/PB 1	2	5	2	2	0	4	47	4	20
WX/LPSI/PB 1	2	2	0	2	0	6	12	6	20
	PB 1 (Point Beach Unit 1) Totals						87	18	80
WX/AFW/ZI 1	3	0	12	3	0	6	123	6	30
WX/CtS/ZI 1	2	8	9	0	12	6	130	126	20
WX/HPSI/ZI 1	4	33	3	2	0	10	197	10	40
WX/LPSI/ZI 1	2	21	0	4	0	2	109	2	20
	ZI 1 (Zion Unit 1) Totals						559	144	110
	IP 2, PB 1, and ZI 1 AVERAGE TOTALS - WX						267	76	93

* PWR Vendors - (B&W) Babcock & Wilcox, (CE) Combustion Engineering, (WX) Westinghouse.
PWR Reactor Safety Systems - see Table 1.

TABLE 3 (Cont'd)

NUMBER OF SELECTED REACTOR SAFETY SYSTEM INTERFACING COMPONENTS
(namely Remotely Operated Valves(ROV), Manual Valves (XV) and Pumps)
CYCLED FOR VARIOUS ACTIVITIES
(Valve Test, Pump Test, or Pump Maintenance including Normal Operations)

BWR VENDOR*/ REACTOR SAFETY SYSTEM/PLANT	#PUMP/ PUMP TEST	#ROV/ VALVE TEST	#ROV/ PUMP TEST	#ROV/ PUMP MAINT	#XV/ PUMP TEST	#XV/ PUMP MAINT	#ROV/ SYSTEM ANNUAL	#XV/ SYSTEM ANNUAL	#PUMP/** SYSTEM ANNUAL
GE/CoS/BF 1	4	8	4	12	0	4	92	4	40
GE/HPCI/BF 1	1	3	3	4	0	4	49	4	10
GE/LPCI/BF 1	4	12	4	12	0	16	112	16	40
	BF 1 (Browns Ferry Unit 1) Totals						253	24	90
GE/CoS/PI 1	2	6	2	2	0	4	52	4	20
GE/HPCI/PI 1	1	3	3	4	0	4	49	4	10
GE/LPCI/PI 1	4	9	4	4	0	12	89	12	40
	PI 1 (Pilgrim Unit 1) Totals						190	20	70
	BF 1 & PI 1 AVERAGE TOTALS - GE						221	22	80

* BWR Vendor - (GE) General Electric
BWR Reactor Safety Systems - see Table 1.

maintenance, and one time for normal valve operation. Therefore, to obtain the number of ROV's cycled annually for valve testing and normal operation, multiply the number in the column labeled "#ROV/VALVE TEST" by five.

- Safety system valves which are cycled for monthly pump testing (columns labeled "#ROV/PUMP TEST" and "#XV/PUMP TEST") are assumed to be cycled ten times annually. This is based on of nine times for monthly testing (reactor assumed critical nine months) and one time for normal safety system pump operations. Therefore, to obtain the number of ROV's and XV's cycled annually for pump testing and normal operations, multiply the numbers in the columns labeled "#ROV/PUMP TEST" and "#XV/PUMP TEST" by ten.
- Safety system valves which are cycled for pump maintenance (columns labeled "#ROV/PUMP MAINT" and "#XV/PUMP MAINT") are assumed to be cycled one time annually. Therefore, the number of ROV's and XV's cycled annually for pump maintenance are the same numbers as in these two columns.

For each representative plant's safety systems, the total annual human (safety system) component interface estimates are calculated for ROV, XV and pumps by summing the appropriate annual demands above. Therefore, the estimate calculations in relation to Table 3's SYSTEM ANNUAL columns are as follows for each system:

$$\begin{aligned}
\text{\#ROV/SYSTEM ANNUAL} &= 5 \times \text{\#ROV/VALVE TEST} + 10 \times \text{\#ROV/PUMP TEST} + 1 \times \text{\#ROV/PUMP MAINT} \\
\text{\#XV/SYSTEM ANNUAL} &= 10 \times \text{\#XV/PUMP TEST} + 1 \times \text{\#XV/PUMP MAINT} \\
\text{\#PUMP/SYSTEM ANNUAL} &= 10 \times \text{\#PUMP/PUMP TEST}
\end{aligned}$$

By way of an example following Table 3 for the "PWR Vendor/Reactor Safety System/Plant" of "B&W/CtS/AR 1" namely Babcock & Wilcox - Containment Spray System of Arkansas Nuclear One Unit 1,

$$\begin{array}{l}\#ROV/\\SYSTEM\\ANNUAL\end{array} = 5 \times \begin{array}{l}\#ROV/\\VALVE\\TEST\end{array} + 10 \times \begin{array}{l}\#ROV/\\PUMP\\TEST\end{array} + 1 \times \begin{array}{l}\#ROV/\\PUMP\\MAINT\end{array}$$

$$70 = (5 \times 10) + (10 \times 2) + (1 \times 0)$$

which means that a total of 70 remotely operated valve (ROV) cycles are estimated to occur annually in the Containment Spray System of Arkansas Nuclear One Unit 1 (AR 1).

$$\begin{array}{l}\#XV/\\SYSTEM\\ANNUAL\end{array} = 10 \times \begin{array}{l}\#XV/\\PUMP\\TEST\end{array} + 1 \times \begin{array}{l}\#XV/\\PUMP\\MAINT\end{array}$$

$$44 = (10 \times 4) + (1 \times 4)$$

which means that a total of 44 manual valve (XV) cycles are estimated to occur annually in the Containment Spray System of AR 1.

$$\begin{array}{l}\#PUMP\\SYSTEM\\ANNUAL\end{array} = 10 \times \begin{array}{l}\#PUMP/\\PUMP\\TEST\end{array}$$

$$20 = 10 \times 2$$

which means that a total of 20 pump cycles (start and stop) are estimated to occur annually in the Containment Spray System of AR 1.

By adding the annual totals for all reactor safety systems provided, the estimated annual number of cycles for all ROV, XV, and pumps for all reactor safety systems in each of the seven representative plants is shown in the Table 3 rows identified as "TOTALS". The average totals are listed in the Table 3 rows identified as "AVERAGE TOTALS" for both the three Westinghouse (WX) and the two General Electric (GE) plants by using arithmetic averages.

All other plants of a given vendor not involved in this review were assumed to use the same number of valves and pumps as the vendor representative plant(s) over a given period of time.

The total interface estimates for all human-safety system components of interest are provided in Table 4. Note that each reactor vendor's contribution to the total plant interface is weighted based on its total number of plant years (since initial criticality) occurring during the three year period of interest. This is indicated in Table 4 under the column entitled "#PLANT-YEARS" and has been calculated based on data from Reference 2.

V. QUANTIFICATION OF HUMAN ERROR - CALCULATION OF HER's

The quantification of human error is presented in this report in terms of Human Error Rate (HER). This measure has been defined in Section II, and is the frequency of the occurrence of human error given the total number of opportunities (i.e. demands) for that particular event. HER's are calculated by dividing the number of human errors summarized for interfacing safety system valves and pumps from Table 2 by the appropriate total human reactor safety system component interface estimates for those valves and pumps from Table 4. The results of these calculations are contained in Table 5 with human error rates ranging from a low of 0.3×10^{-3} for interfacing with remotely operated valves (ROVs) in PWR safety systems to a high of 1.5×10^{-3} for interfacing with manual valves (XVs) in PWR safety systems.

VI. SENSITIVITY OF HER CALCULATIONS TO MODIFICATIONS OF HUMAN-COMPONENT INTERFACE ASSUMPTIONS

In order to test the sensitivity of the results to the basic assumption, several of the more dominant human-component interface assumptions were modified. This resulted in a new set of calculated values for selected reactor

TABLE 4

HUMAN - SELECTED REACTOR SAFETY SYSTEM* COMPONENT

TOTAL INTERFACE ESTIMATES OVER A

SPECIFIC THREE-YEAR PERIOD**

REACTOR VENDOR	# PLANTS OPERATIONAL 1976-1978 (5/75-4/78)	# PLANT YEARS 1976-1978 (5/75-4/78)	# PLANTS SURVEYED	AVERAGE # ROV CYCLED PER YEAR PER PLANT	TOTAL # ROV CYCLED 1976-1978	AVERAGE # XV CYCLED PER YEAR PER PLANT	TOTAL # XV CYCLED 1976-1978	AVERAGE # PUMP CYCLED PER YEAR PER PLANT	TOTAL # PUMP CYCLED (5/75-4/78)
B&W	9 (9)	22.0 (20.0)	1	213	4,686	89	1,958	90	1,800
CE	7 (7)	19.7 (18.0)	1	357	7,033	23	453	100	1,800
WX	25 (24)	67.2 (62.7)	3	267	17,942	76	5,107	93	5,831
All PWR	41 (40)	108.9 (100.7)	5	274	29,661	68	7,518	94	9,431
GE BWR	23 (22)	65.0 (63.2)	2	221	14,365	22	1,430	80	5,056
ALL	64 (62)	173.9 (163.9)	7	TOTALS	44,026		8,948		14,487

ROV - Remotely Operated Valve
XV - Manual Valve

*see Table 1 for reactor safety systems involved
**January 1, 1976 through December 31, 1978 for valves.
May 1, 1975 through April 30, 1978 for pumps.

TABLE 5

HUMAN ERROR RATE (HER)* CALCULATION

SUMMARY

Reactor Safety System** Interfacing Component	Reactor Type PWR	BWR	Both
Remotely Operated Valves (ROV)	0.3×10^{-3} (0.5×10^{-3})***	0.6×10^{-3} (1.8×10^{-3})	0.4×10^{-3} (0.6×10^{-3})
Manual Valves (XV)	1.5×10^{-3} (1.8×10^{-3})	1.4×10^{-3} (1.4×10^{-3})	1.5×10^{-3} (1.8×10^{-3})
Pumps	1.2×10^{-3} (1.7×10^{-3})	0.6×10^{-3} (0.8×10^{-3})	1.0×10^{-3} (1.4×10^{-3})

*see Section II for HER definition
**see Table 1 for reactor safety systems involved.
***all numbers in parenthesis are the result of a sensitivity study from Section VI.

The assumptions of Section IV that were modified are as follows:

- Those safety system valves which are needed to move in order to perform their intended functions are assumed to be cycled annually (instead of quarterly) for valve testing. This modification addresses the uncertainty that the valve tests may have actually been performed annually instead of as quarterly.
- Safety system pumps are assumed to be cycled once a month for pump testing when the reactor is critical. Assume that the reactor is critical only six months each year (instead of nine). This modification addresses the uncertainty associated with less pump testing due to less reactor operating time.

These perturbations result in the following modified estimate calculations which are applied in Table 3 - MODIFIED:

$$\#\text{ROV/SYSTEM ANNUAL} = 3 \times \#\text{ROV/VALVE TEST} + 7 \times \#\text{ROV/PUMP TEST} + 1 \times \#\text{ROV/PUMP MAINT.}$$

$$\#\text{XV/SYSTEM ANNUAL} = 7 \times \#\text{XV/PUMP TEST} + 1 \times \#\text{XV/PUMP MAINT.}$$

$$\#\text{PUMP SYSTEM ANNUAL} = 7 \times \#\text{PUMP/PUMP TEST}$$

Even though the assumption modifications may appear to be quite extreme, it can be concluded that the results are fairly insensitive, in an order of magnitude fashion, to the perturbation of the initial estimates. Future efforts will expand this test and also address the errors introduced by the actual LER data.

TABLE 3 - MODIFIED

NUMBER OF SELECTED REACTOR SAFETY SYSTEM INTERFACING COMPONENTS
(namely Remotely Operated Valves(ROV), Manual Valves (XV) and Pumps)
CYCLED FOR VARIOUS ACTIVITIES
(Valve Test, Pump Test, or Pump Maintenance including Normal Operations)

PWR VENDOR*/ REACTOR SAFETY SYSTEM/PLANT	#PUMP/ PUMP TEST	#ROV/ VALVE TEST	#ROV/ PUMP TEST	#ROV/ PUMP MAINT	#XV/ PUMP TEST	#XV/ PUMP MAINT	#ROV/ SYSTEM ANNUAL	#XV/ SYSTEM ANNUAL	#PUMP/** SYSTEM ANNUAL
B&W/AFW/AR 1	2	4	0	8	0	2	20	2	14
B&W/CtS/AR 1	2	10	2	0	4	4	44	32	14
B&W/HPSI/AR 1	2	9	0	0	0	9	27	9	14
B&W/LPSI/AR 1	3	10	2	0	3	4	44	25	21
	AR 1 (Arkansas Unit 1) TOTALS B&W						135	68	63
CE/AFW/MI 2	3	5	6	6	0	3	63	3	21
CE/CtS/MI 2	2	8	4	2	0	4	54	4	14
CE/HPSI/MI 2	3	12	4	4	0	8	68	8	21
CE/LPSI/MI 2	2	8	4	0	0	8	52	8	14
	MI 2 (Millstone Unit 2) TOTALS CE						237	23	70
WX/AFW/IP 2	3	0	0	0	0	14	0	14	21
WX/CtS/IP 2	2	4	0	0	4	4	12	32	14
WX/HPSI/IP 2	2	18	0	4	0	5	58	5	14
WX/LPSI/IP 2	2	6	1	0	0	4	25	4	14
	IP 2 (Indian Point Unit 2) Totals						95	55	63
WX/AFW/PB 1	2	0	0	6	0	6	6	6	14
WX/CtS/PB 1	2	4	0	2	0	2	14	2	14
WX/HPSI/PB 1	2	5	2	2	0	4	31	4	14
WX/LPSI/PB 1	2	2	0	2	0	6	8	6	14
	PB 1 (Point Beach Unit 1) Totals						87	18	56
WX/AFW/ZI 1	3	0	12	3	0	6	87	6	21
WX/CtS/ZI 1	2	8	9	0	12	6	87	90	14
WX/HPSI/ZI 1	4	33	3	2	0	10	122	10	28
WX/LPSI/ZI 1	2	21	0	4	0	2	67	2	14
	ZI 1 (Zion Unit 1) Totals						363	104	77
	IP 2, PB 1, and ZI 1 AVERAGE TOTALS - WX						172	60	65

* PWR Vendors - (B&W) Babcock & Wilcox, (CE) Combustion Engineering, (WX) Westinghouse.
PWR Reactor Safety Systems - see Table 1.

TABLE 3 - MODIFIED (Cont'd)

NUMBER OF SELECTED REACTOR SAFETY SYSTEM INTERFACING COMPONENTS
(namely Remotely Operated Valves(ROV), Manual Valves (XV) and Pumps)
CYCLED FOR VARIOUS ACTIVITIES
(Valve Test, Pump Test, or Pump Maintenance including Normal Operations)

BWR VENDOR*/ REACTOR SAFETY SYSTEM/PLANT	#PUMP/ PUMP TEST	#ROV/ VALVE TEST	#ROV/ PUMP TEST	#ROV/ PUMP MAINT	#XV/ PUMP TEST	#XV/ PUMP MAINT	#ROV/ SYSTEM ANNUAL	#XV/ SYSTEM ANNUAL	#PUMP/ SYSTEM ANNUAL
GE/CoS/BF 1	4	8	4	12	0	4	64	4	28
GE/HPCI/BF 1	1	3	3	4	0	4	34	4	7
GE/LPCI/BF 1	4	12	4	12	0	16	76	16	28
	BF 1 (Browns Ferry Unit 1) Totals						174	24	63
GE/CoS/PI 1	2	6	2	2	0	4	34	4	14
GE/HPCI/PI 1	1	3	3	4	0	4	34	4	7
GE/LPCI/PI 1	4	9	4	4	0	12	59	12	28
	PI 1 (Pilgrim Unit 1) Totals						127	20	49
	BF 1 and PI 1 AVERAGE TOTALS - GE						151	22	56

* BWR Vendor - (GE) General Electric
BWR Reactor Safety Systems - see Table 1.

TABLE 4 - MODIFIED

HUMAN - SELECTED REACTOR SAFETY SYSTEM* COMPONENT

TOTAL INTERFACE ESTIMATES OVER A

SPECIFIC THREE-YEAR PERIOD**

REACTOR VENDOR	# PLANTS OPERATIONAL 1976-1978 (5/75-4/78)	# PLANT YEARS 1976-1978	# PLANTS SURVEYED	AVERAGE # ROV CYCLED PER YEAR PER PLANT	TOTAL # ROV CYCLED 1976-1978	AVERAGE # XV CYCLED PER YEAR PER PLANT	TOTAL # XV CYCLED 1976-1978	AVERAGE # PUMP CYCLED PER YEAR PER PLANT	TOTAL # PUMP CYCLED
B&W	9 (9)	22.0 (20.0)	1	135	2,970	68	1,496	63	1,260
CE	7 (7)	19.7 (18.0)	1	237	4,669	23	453	70	1,260
WX	25 (24)	67.2 (62.7)	3	172	11,558	60	4,032	65	4,076
All PWR	41 (40)	108.9 (100.7)	5	178	19,197	66	5,981	66	6,596
GE BWR	23 (22)	65.0 (63.2)	2	151	9,815	22	1,430	56	3,539
ALL	64 (62)	173.9 (163.9)	7	TOTALS	29,012		7,411		10,135

ROV - Remotely Operated Valve
XV - Manual Valve

*see Table 1 for reactor safety systems involved
**January 1, 1976 through December 31, 1978 for valves.
May 1, 1975 through April 30, 1978 for pumps.

VII. RESULTS - COMPARISON WITH EXISTING DERIVED DATA

The general human error rate estimates from WASH-1400 (Reference 3, Table III 6-1) which follow have, in general, been derived from existing non nuclear data, as modified by the independent judgements of two human reliability analysts after extensive review of many factors. From WASH-1400 come the following activities:

- General human error of commission, e.g. misreading label and therefore selecting wrong switch (3×10^{-3}).
- General human error of omission where there is no display in the control room of the status of the item omitted e.g. failure to return manually operated test valve to proper configuration after maintenance (1×10^{-2}).

Note that these WASH-1400 activities listed above most closely correspond to the human safety system component interfaces developed in this report.

The first WASH-1400 activity above is generally compared to the remotely operated valve (ROV) interface HER calculation of 0.4×10^{-3} (0.6×10^{-3}) from Table 5, assuming the HERs with ROV are mostly acts of commissions and the switches are for ROVs.

The latter WASH-1400 activity is generally compared to the manual valve (XV) interface HER calculation of 1.5×10^{-3} (1.8×10^{-3}) from Table 5, assuming the HERs with XV are mostly acts of omission.

The more specific errors of omission and commission as presented by the derived human error probabilities in the Handbook of Human Reliability Analysis (Reference 4) do not lend themselves, at this time, to direct comparison to the broad categories of human-component interface in this report.

VIII. CONCLUSIONS AND FUTURE REFINEMENTS

The approaches and conclusions described in this study represent the first attempts at quantifying human error rates based on nuclear data and task analysis. Thus, the method is more directly linked to the actual operations of a nuclear power station. As work continues in the area, efforts will be made to refine the results and techniques so that specific HER estimates can be made with a known confidence. These estimates will, as much as practical, provide benchmarks for existing HERs that have been developed by other means. The refinements that are necessary can be divided into three categories.

First, there is a need to further validate the developed number of human interfaces on a plant specific basis. This review of the denominator of the HER equation should highlight only order of magnitude type differences, if any, since the HER is relatively insensitive to small perturbations.

Second, since this effort did not attempt to utilize the systems approach to evaluate the number of human errors utilized in the numerator of the HERs, there is a need to do so now. The LERs should be further investigated to correctly classify those events that could not be defined as human error by the initial review. Additionally, human errors that might be marked by incorrect description should be extracted to gain further confidence in the results. With this expanded data available for analysis, the gross categories of remotely operated valves, manual valves, and pumps could be subdivided. The more specific categories, which would lend themselves more readily to Obenchmarking and application, will also require a more detailed calculation of the denominator to result in HERs. Initial suggested categories are by systems and subsystems, NSSS supplier, specific plant, training level of individuals, point in life cycle of plant, and specific actions (test, maintenance, omission, commission). PSFs that are measurable and have apparent

safety significance will be used to modify the generic rates. The new HERs developed from this effort will be utilized in the HASAP sensitivity code (Reference 5) to define impact on risk.

Third, with the refinements in the basic HER, there will be a need to develop statistical confidence bounds and attempts should be made to estimate common cause HERs.

It is concluded that with the above refinements to the task analysis, data, and statistical descriptions, the needed HERs will be available to the engineer. Until this is completed, the information in this report should be considered only as a guide to the user.

REFERENCES

1. Data Summaries of Licensee Event Reports of Pumps at U.S. Commercial Nuclear Power Plants - May 1, 1975 to April 30, 1978 - NUREG/CR-1205, EGG-EA-5044, January 1980, by W.H. Sullivan and J.P. Poloski of EG&G Idaho, Inc.*

2. Data Summaries of Licensee Event Reports of Valves at U.S. Commercial Nuclear Power Plants - January 1, 1976 to December 31, 1978 NUREG/CR-1363, EGG-EA-5125, June 1980, by W.H. Hubble and C.F. Miller of EG&G Idaho, Inc.*

3. Reactor Safety Study, WASH-1400 (NUREG-75/014) Appendix III, Section 6.1 entitled Human Reliability Analysis (Table III 6-1, General Error Rate Estimates), October 1975.**

4. Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications - NUREG/CR-1278 Draft Report for Interim Use and Comment, October 1980 by A.D. Swain and H.E. Guttmann, Chapter 13 (Valving Operations) and Chapter 20 (Derived Human Error Probabilities and Related Performance Shaping Factors).*

5. Sensitivity of Risk Parameters to Human Errors in Reactor Safety Study for a PWR - Final Draft, NUREG/CR-1879, December 1980 by P.K. Samanta, A.L. Swoboda, and R.E. Hall of Brookhaven National Laboratory.*

*Available for purchase from the NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, and the National Technical Information Service, Springfield, VA 22161

**Available free upon written request to Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555

APPENDIX A

LER DATA SUMMARY CODES USED IN APPENDIX A*

VEND - Reactor VENDor

- B - Babcock & Wilcox
- C - Combustion Engineering
- W - Westinghouse
- G - General Electric

PLNT - Reactor PLaNT (Only those in Appendix A)

Pressurized Water Reactors (PWRs)

B CR3 Crystal River 3	W DC2 D.C. Cook 1
B DB1 Davis Besse 1	W IP2 Indian Point 2
B OE1 Oconee 1	W IP3 Indian Point 3
B OE2 Oconee 2	W JF1 J.M. Farley 1
B OE3 Oconee 3	W KE1 Kewaunee 1
B RS1 Rancho Seco 1	W NA1 North Anna 1
C CC1 Calvert Cliffs 1	W PR2 Prairie Island 2
C MY1 Maine Yankee 1	W RO2 H.B. Robinson 2
C SL1 St. Lucie 1	W SU2 Surry 2
W BV1 Beaver Valley 1	W TR1 Trojan 1
W HN1 Connecticut Yankee	W TU4 Turkey Point 4
W DC1 D.C. Cook 1	W ZI1 Zion 1

Boiling Water Reactors (BWRs)

G BR1 Brunswick 1	G EN1 E.I. Hatch 1
G BF2 Browns Ferry 2	G FP1 FitzPatrick 1
G CO1 Cooper 1	G NM1 Nine Mile Point 1
G DA1 Duane Arnold 1	G PB2 Peach Bottom 2
G DR2 Dresden 2	G QC2 Quad Cities 2
G DR3 Dresden 3	G VY1 Vermont Yankee 1

LER CONTRL NUMBER - six digit control number assigned to each LER by the NRC to uniquely identify each record in the data file.

LER EVENT DATE - actual date the LER event occurred.

SYST - Reactor Safety SYSTem (only those in Appendix A).

PWR	BWR
B Auxiliary Feed	D Low Pressure Core Spray
F Containment Spray Injection	F Containment Spray Injection
H High Press. Cool. Injection	H High Press. Cool. Injection
L Low Press. Cool. Injection (RHR)	L Low Press. Cool. Injection (RHR)
G Chemical Volume Control (Make-up) - for Charging Pumps only	Q Reactor Core Isolation Cooling

*See References 1 and 2 for complete listing of LER data summary codes.

COMP - COMPonent type

PUMPS

PM - Pump, Motor Driven
PT - Pump, Turbine Driven

VALVES

AV - Pneumatic - Operated Valve
HV - Hydraulic - Operated Valve
MV - Motor - Operated Valve (Electric)
RM - Remote - Operated Valve (but AV, HV, or MV not specified)
UV - Operator Type or Function Not Stated
XV - Manual - Operated Valve

Note that the reactor safety system remotely operated valve (ROV) related group of LER data summaries include the reactor safety system AV, HV, MV, and RM related summaries. In addition, the ROV group includes those UV reactor safety system related summaries which appear to be ROV rather than XV. The reactor safety system manual valve (XV) related group of LER data summaries include all those other UV related summaries which appear to be XV in addition to those already identified as XV related.

FAIL MODE/ MECH CODE -

Failure Mode (left character)	Failure Mechanism (center and right characters)
PUMPS	
B - Does not start	02 Personnel error
C - Loss of function	14 Procedural Discrepancy
D - Does not continue to run	
VALVES	
F - Failed to operate as required	01 Personnel (Operation)
T - Technical Spec. Violation	02 Personnel (Maintenance)
V - Improper Valve Configuration	03 Personnel (Testing)
	06 Procedural Discrepancy

PERSONNEL ERROR RELATED LER DATA SUMMARIES
ASSOCIATED WITH REACTOR SAFETY SYSTEM
REMOTELY OPERATED VALVES (ROV)
(from January 1, 1976 to December 31, 1978)

VEND	PLNT	LER CONTROL NUMBER	LER EVENT DATE	SYST	COMP	FAIL MODE MECH CODE	FAILURE MODE	FAILURE MECHANISM
B	CR3	017322	030777	H	RM	VO1	"B" LOOP OF HPI SYS. ISOL. CONT. TO T.S.FOR MAINTENC	PERSONNEL ERROR
B	DB1	021176	043078	L	UV	VO1	VLVS DH11&12 WERE NOT OPENED WHEN REQ. DURING S/D	PERSONNEL ERROR & PROCEDURE INADEQUACY
B	OE2	015535	080976	L	UV	VO1	LOW PRESS INJECT TRAIN A NOT ISO PROPERLY	PERSONNEL FAIL TO FOLLOW T.S.REQUIREMENTS
B	DB1	019984	121677	B	UV	VO3	AUX.FDWTR VLVS AF599&608 SHUT;MS107A,AF3871&72BK.O	PERSONNEL ERROR;IMPROPER VLV LINEUP
C	MY1	019261	102077	L	UV	VO1	IMPROPER VALVE LINEUP LPSI PUMPS	PERSONNEL ERROR
W	DC1	016648	121776	F	MV	VO1	CONTAINMENT SPRAY PUMP SUCTION VALVE FOUND CLOSED	PERSONNEL LEFT VALVE SHUT AFTER A TEST
W	NA1	022565	100478	B	UV	VO1	IMPROPER VALVE LINEUP,AUX FEED PUMP TO A,B,C STMGN	OPERATING PERSONNEL ERROR
W	SU2	015962	082076	H	UV	VO1	OUTSIDE ISOLATION VALVE FOUND OPEN/NORMALLY CLOSED	PERSONNEL/FAILED TO FOLLOW ADMIN CONTROLS
W	TU4	019421	091377	L	RM	VO1	TWO VALVES INADVERTENTLY OPENED/RWST LEVEL DECREAS	PERSONNEL ERROR/POSSIBLE PROCEDURE PROBLM
W	IP3	014604	042976	F	RM	VO3	INCRT.VLV LINUP FOR TEST,CNTMNT SPRY ACCID. TURN ON	PERSONNEL ERROR
G	QC2	021561	052178	D	RM	VO1	2A CORE SPRAY SUC.VLV. MO 2-1402-3A CLOSED WHEN REQD	TO BE OPEN//PERSONNEL ERROR
G	VY1	014230	021776	L	MV	FO1	RHR VLVS. 25A&B WOULD NOT STROKE ELECTRICALLY	PERSON.ERROR(MUST STRO.VLVS.MANUALLY 1ST)
G	FP1	021227	050878	D	MV	FO2	POWER REMOVED FROM CORE SPRAY VALVE MOTOR	PERSONNEL ERROR
G	CO1	018894	072577	L	RM	TO3	4 ISOLATION VALVES NOT LEAK TESTED	PERSONNEL ERROR
G	DR3	018551	072577	L	RM	TO3	SURVEILLANCE TEST NOT COMPLETED ON LPCI VALVES	PERSONNEL WERE 6 DAYS LATE COMPLETNG TEST
G	EN1	014106	010276	D	RM	TO3	CORE SPRAY TEST NOT DONE IN REQUIRED TIME	PERSONNEL ERROR
G	FP1	017659	022477	H	UV	VO3	VALVES ON HPCI ROUTE FROM COND TK TO TORUS L/U IMP	OPERATOR ERROR/TESTING PERSONNEL
G	VY1	016997	011877	L	RM	VO3	"B" RHR PMP DISCH.VLVS. SHUT WHEN REQ. TO BE OPEN	PERSONNEL ERROR

PERSONNEL ERROR RELATED LER DATA SUMMARIES
ASSOCIATED WITH REACTOR SAFETY SYSTEM
MANUAL VALVES (XV)
(from January 1, 1976 to December 31, 1978)

VEND	PLNT	LER CONTROL NUMBER	LER EVENT DATE	SYST	COMP	FAIL MODE	MECH CODE	FAILURE MODE	FAILURE MECHANISM
B	DB1	019381	102077	B	UV	V01		BOTH AUX.FD.PMP.SUCTS.LINED UP TO DEAERATOR STR.TK	PERSONNEL ERROR;IMPROPER COOL.WTR.LINEUP
B	RS1	023502	121578	L	UV	V01		INCOMPLETE VALVE LINEUP CAUSED SEVERE H2O HAMMER	PERSONNEL ERROR
B	CR3	023095	100478	L	XV	V03		ISOL VALVE TO FLOW CONTROLLER SHUT	PERSONNEL ERROR/VALVE REQUIRED TO BE OPEN
C	SL1	022908	110378	L	UV	V02		ISOL VALVES INADVERT OPENED BETW SPRAY&SHUTDOWN COOLING SYSTEM//	PERSONNEL ERROR
W	DC1	019569	111877	F	UV	V01		TWO CONTAINMENT SPRAY VALVES FOUND CLOSED	PERSONNEL ERROR/CLOSED UNIT 1 INSTEAD OF 2
W	DC2	021079	041678	L	XV	V01		HANDWHEEL CAME OFF RHR VALVE POSITIONING IT WRONG	PERSONNEL FAILED TO REPOSITION THE VALVE
W	DC2	022838	101078	B	XV	V01		AUX FEED VALVE FOUND UNLOCKED & OPEN/WRONG LINE-UP	PERSONNEL ERROR
W	TR1	019580	101677	L	XV	V01		RHR DISCHARGE VLV (87288) FOUND LOCKED CLOSED	PERSONNEL ERROR/IMPROPER VALVE LINEUP
W	ZI1	017255	021877	L	UV	V01		IMPROPER VALVE LINEUP/UNIT-1 VAL CLOSED INSTEAD 2	PERSONNEL ERROR/NEITHER VAL TAGGED FOR ID
W	JF1	020993	032578	B	UV	V02		MDAFP & TDAFP RECIRC BYPASS ISO VLVS OPEN	PERSONNEL ERROR
W	IP2	015920	080576	H	UV	V03		SI PMPS COMMON SUC.VLV(846)IN IMPROPER POSITION	PERSONNEL ERROR/DID NOT OPEN VLV AS REQR.
G	DR2	021158	042878	L	XV	V01		VALVES DNA LPCI & CORE SPRAY IMPROPER LINEUP	OPERATING PERSONNEL ERROR
G	QC2	018575	071977	L	UV	V01		VALVE FOUND CLOSED BETWEEN A & B LOOP/LOSS OF LPCI	PERSONNEL ERROR/INADVERTANT CLOSURE

PERSONNEL ERROR RELATED LER DATA SUMMARIES
ASSOCIATED WITH REACTOR SAFETY SYSTEM
PUMPS
(from May 1, 1975 through April 30, 1978)

VEND	PLNT	LER CONTROL NUMBER	LER EVENT DATE	SYST	COMP	FAIL MODE MECH CODE	FAILURE MODE	FAILURE MECHANISM
B	DB1	018711	080377	B	PT	BO2	AFP 1-2 DID NOT START ON A LOSS OF FEEDWTR TRIP	GOVENR VLV COMPLETELY CLOSED
B	DB1	018717	080677	B	PT	BO2	AFPT 1-2 FAILD TO START ON ACTUATION SIGNAL	DISCONNECT SWITCH IN "REMOTE"
B	DB1	020274	011278	F	PM	BO2	BOTH CONT SPRAY PMPS INOPERABLE R/X IN MODE 4	PUMP CKT.BKRS. RE-ENERGIZED
B	OE1	014806	012876	F	PM	BO2	REACTOR BUILDING SPRAY PUMP - INOPERABLE (1A)	INCORRECT SUBSTITUTE BREAKER INSTALLED
B	RS1	016711	112376	F	PM	BO2	RX BLDG SPRAY PUMP FAILED TO START	BRKR WAS NOT RACKED IN CORRECTLY
B	RS1	017233	021877	B	PT	DO2	STEAM DRIVE TO AUX FEED P318 TRIPPD P319 OOS	OPERATR FAILD TO RESET TRIP WHEN P318 TST
W	BV1	021646	042178	G	PM	DO2	1C CHG PUMP TRIPPED, NO BACK-LP WAS AVAILABLE	OPERATOR MADE IMPROPER ELEC LINE-UP
W	JF1	020993	032578	B	PT	BO2	AFPT FAILD TO START FOLLOWING R/X TRIP	TRIP/THROTTLE VALVE FOUND SHUT
W	RO2	017585	032477	F	PM	DO2	CONTMT SPRAY PMP A NOT ON SVC DURING CRITICALITY	PERSONNEL HAD REMOVED PUMP FROM SERVICE
W	RO2	019793	112377	H	PM	BO2	BREAKERS FOR BOTH SAFETY INJ PMPS FOUND RAKD OUT	FAILURE TO RECOG TS LIMIT FOR GT 200 F OP
W	TR1	016337	101976	B	PT	BO2	TURB-DRVN AUX FD PMP FAILED TO START - AUTOMATIC	OPERATOR FAILED TO RESET TRP/THRL VALVE
G	BR1	018167	061877	L	PM	DO2	2A RHR TRIPPD DUE TO OPERATOR HITTING SWITCH	ACCIDENTAL BUMPING OF CONTROL ROOM SWITCH
G	BR2	018676	062277	L	PM	BO2	RHR 2A CHARGN MOTOR SWITCH IN OFF POSITION	PERSONNEL LEFT SWITCH IN OFF POSITION
G	PB2	021081	042978	L	PM	BO2	UNIT 2 "B","D"RHR BLOCKED FOR 2 HRS.	OPERTR REMVD UNIT 2 INSTED OF UNIT 3 PUMP

PROCEDURAL DISCREPANCY RELATED LER DATA SUMMARIES
ASSOCIATED WITH REACTOR SAFETY SYSTEM
VALVES
(from January 1, 1976 to December 31, 1978)

VEND	PLNT	LER CONTROL NUMBER	LER EVENT DATE	SYST	COMP	FAIL MODE MECH CODE	FAILURE MODE	FAILURE MECHANISM
B	CR3	017176	020777	K	RM	V06	PER.CYCLED BSV-36 ALLOWING NAOH INTO THE RCS SYSTEM	PROCEDURAL INADEQUACY
B	DB1	019548	111777	B	UV	V06	AUX.FDWTR.VLVS AF599&608 CLOSED;T.S.VIOLATION	PROCEDURAL DEFICIENCY;TEST PROCEDURE REV1
B	OE3	019883	120677	L	XV	T06	VALVE LP-28 OPEN BUT NOT LOCKED AS REQD	INADEQUATE PROCEDURES
C	CC1	017476	031377	L	UV	V06	ISO.VLV ADD.TO AUX SPRAYLN DUR.PRIOR OUT.LEFT SHUT VLV.NOT INCLUDED IN VLV LINEUP/PROCED.ERR	
W	BV1	020496	012178	L	UV	V06	THE RWST WAS DRAINED TOO LOW DUE TO INCOR.VLV L/U	PROCEDURAL DISCREPENCY
W	JF1	020446	012178	L	RM	I06	VAL HCV-603A MECH STOP NOT REPLACED AFTER MAINT	ERROR IN TECH SPEC
W	JF1	022629	091878	L	RM	V06	RHR PUMP SUCTION VLV 8701A CLOSED WHEN REQ. OPEN	INADEQUATE PROCEDURES
W	KE1	021365	042378	L	UV	V06	RHR VENT VLV. OPEN,WHEN IT SHOULD HAVE BEEN CLOSED	DEFECTIVE PROCEDURES
W	NA1	021506	051478	B	AV	V06	AUX FEEDWTER VALS PCV-F2-159A,-159B IMP.LINEUP	CHECKOFF SHEET DOES NOT LIST NORM VAL POS
W	PR2	017889	051977	B	XV	V06	PUMP INLET VALVE FOUND NEARLY SHUT/AUX FEED INLET	POSITION REQUIREMENT WILL GO ON CHECKLIST

PUMPS
(from May 1, 1975 to April 30, 1978)

VEND	PLNT	LER CONTROL NUMBER	LER EVENT DATE	SYST	COMP	FAIL MODE MECH CODE	FAILURE MODE	FAILURE MECHANISM
G	DA1	019127	092177	H	PT	C14	HPSI WOULD NOT DEVELOP SUFFCNT RPM	EXCESSIVE OPENING OF L.O. THROTLE VALVES
G	NM1	017431	040477	L	PM	D14	WHILE LOWRNG R/X H2O LEVEL FOR MAINTNCE ON VESSEL	SHUTDWN COOLNG PMP TRPPD DUE TO LOW SUCT
W	HN1	015097	061776	L	PM	D14	R/X SHUTDOWN & REFUELING LOST PWR TO RHR PMP --2MIN	OVERLOADED 480V BUS PROCEDURES DEFICIENT

PERSONNEL ERRORS RELATED LER DATA SUMMARY

PERCENTAGES BY FAILURE MODE/MECHANISM

FAILURE MODE/MECHANISM	ROV			XV		
	PWR	BWR	BOTH	PWR	BWR	BOTH
V01 Improper valve configuration/ personnel (operations)	80.0	12.5	50.0	63.6	100	69.2
V02 Improper valve configuration/ personnel (maintenance)	0	0	0	18.2	0	15.4
V03 Improper valve configuration/ personnel (testing)	20.0	25.0	22.2	18.2	0	15.4
T03 Technical Specification violation/personnel (testing)	0	37.5	16.6	0	0	0
F01 Failed to operate as required/ personnel (operations)	0	12.5	5.6	0	0	0
F02 Failed to operate as required/ personnel (testing)	0	12.5	5.6	0	0	0

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16. ABSTRACT *(200 words or less)*

This report provides a methodology for an initial quantification of gross categories of human errors made in conjunction with several reactor safety system components operated, maintained and tested in licensed nuclear power plants. The resultant human error rates provide the first real systems bases of comparison for the existing, derived and/or best judgement equivalent set of such rates or probabilities.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

Human Errors
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17b. IDENTIFIERS/OPEN-ENDED TERMS

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