Evaluation of human error in high

temperature gas cooled reactor

by

Waleed Hussain Abul-Faraj

A Thesis Submitted to the

Graduate Faculty in Partial Fulfillment of the

Requirements for the Degree of

MASTER OF SCIENCE

Major: Nuclear Engineering

Approved:

Signatures have been redacted for privacy

Iowa State University
Ames, Iowa

1979

TABLE OF CONTENTS

ABSTRACT

The HEGAR classification of gas cooled reactor for coding human errors is developed. The classification describes the general systems, subsystems, and components of HTGR. This classification is flexible to permit expansion, change, added for any system, subsystem, and component and it can be adapted to any HTGR design.

This study shows the importance of human error in Fort St. Vrain, HTGR to safety analysis. Human errors contributed 38.4% to the total reported events. The major sources of human error based on manual review of LER's records from May 30, 1974 to December 30, 1977 are maintenance error in improper handling, did not check/test, or improper setting and administrative errors in procedural deficiencies. The systems most frequently involved in human errors are main reactor coolant system (38.4%), auxiliary electric power system (23.3%), reactor protection system (16.4%), and radioactive waste treatment system (9.6%).

Though the systems most frequently involved are different in HTGR and LWR, the components most frequently involved show the same pattern. Valves were involved in 15.1% of the human errors, while switches contributed 13.7% to the human error population. Pumps (8.2%) and control rods (6.8%) were the second leading category.

The failure causes were due to procedural deficiency (17.8%), improper handling (13.7%), and did not test/check (11.0%).

Pattern recognition techniques were utilized in the analysis of the data and the identification of generic and system specific problems.

I. INTRODUCTION

The relevance of human errors to radioactivity release and radiation exposure of both the public and workers is considered due to its importance to safety analysis (1). Human error and other combination of failures have recently contributed to the ill-controlled accident of the Three Mile Island nuclear power station of Metropolitan Edison Company (2).

The purpose of this study is to estimate the human errors, and failure (error) rates with 90% confidence bounds based on actual nuclear reactor operating experience of Fort St.Vrain reactor (HTGR), and to identify the most frequently occurring human errors, failure mode, and error consequences for the period from May 30, 1974 to December 30, 1977. Also, the impacts of human errors on plant systems, components, and on the environment are evaluated for the same duration of experience.

The Reactor Safety Study (RSS) WASH-1400 estimated the public risk in the operation of U.S. commercial power plants which could result from potential accidents in light water reactors (LWR's) (3).

The risks were estimated because there have been no nuclear accidents to date resulting in risk to the public. The methods used to develop the risk estimates were based on event trees and fault trees techniques which were used to define potential accident paths and their likelihood of occurrence. Since no actuarial data base for human error rates in nuclear power plants exists, it was necessary to obtain data from other industries for the relevant operation tasks and with good judgment of technical personnel.

Accident Initiation and Progression Analysis (AIPA) study has been applied to obtain guidance in choosing nuclear safety research and development that is most worthwhile for HTGR nuclear power plants (4, 5). The probabilistic techniques used are similar to those employed in the RSS for LWR's, WASH-1400, which are based on initiating event selection, event/fault/tree construction, block diagram, probability evaluation, and consequence evaluation. The study was divided into two phases. Phase one; the preliminary phase: (1) establishes a framework for the ranking of HTGR abnormal event sequences with respect to safety as an aid in the selection of future efforts, (2) provides quantitative data for the identification of risk and design tasks, (3) provides bases for selection of systems suitable for the studies of economic aspects of alternative design options for the safety-related systems, and (4) provides insights as to which aspects of the risk analysis need to be emphasized in the future to achieve necessary maturing of the probabilistic methodology. Seventeen initiating events were chosen and evaluated, these being representative of the complete spectrum of classes of potential radioactive sources in the plant and having potentially the highest probabilistics of release within each class. In phase two the study was extended to consider a much broader range of accident sequences in terms of both an increased number of initiating events and a wider spectrum of plant responses to core heat up transients.

Joos et al. (6) developed a computer program for storing the human error information for PWR's, BWR's between June 1, 1973 and June 30, 1975. Human errors, error rates, and 95% confidence interval were calculated.

Husseiny et al. (7) introduced a taxonomy of occurrences as a framework for data collection. Human factors effects on Fort St.Vrain were investigated in regard to human and systems interfaces. Also Mean Times Between Failure (MTBF) were calculated for both routine and vigelance tasks.

Sabri et al. (8) suggested a taxonomy of operation tasks and operator errors. A scheme for collection of data on reliability of nuclear power plants was outlined. The scheme was designed for sorting and storing of failure information in a data library for ease of retrieval by reliability analysis codes. Kalman filter techniques, to evaluate the operator performance, to predict and to update human failure rates were introduced.

Danofsky et al. (9) developed a model to examine the influence of operator performance on reactor shutdown system reliability. The model provides a tool to monitor the operator response in different operational tasks. Also the model allows the use of existing data on human response.

A systematic approach (10) was developed to analyze operation experience in commercial nuclear power plants by providing a measure of the gain and deterioration in the operation skill. The model provides a tool to deal with collective performance of plant operators or to examine the skill of individual operators. The model was tested for Dresden I, Yankee Rowe, and for LWR experience in 1974 and 1975.

Kherich et al. (11) reviewed nuclear power plant experience to determine causes, frequency, and duration of forced shutdowns of commercial nuclear power stations. Correlations were made among component failure rates, human factors, plant sizes, reactor types, and plant downtime. The study provides further improvement on availability which can be achieved by identification of causes of forced outages through parametric analysis of relevant operation data.

NSRG (1) developed a computer data management program for storage, handling, updating and retrieval of information and compiled data extract from the LER's. An operator-analysis statistical information system (OASIS) using alphameric encoding scheme of LER records, and a general event classification system (GENCLASS) containing more detail description of event were presented. An analysis of the interrelationships between operator and human error rates, and different operation parameters were conducted. Several statistical techniques for use in data smoothing and for interpretation of LER statistics were examined, and a linear recursive Kalman filter were obtained for data smoothing, prediction and updating of operator error rates base on a learning model.

Sabri and Husseiny (12) introduced a human model based on cybernetic interactions and allowed for use of available data from psychological experiments. The operator model was identified and integrated in the control and protection system. The availability and reliability were given for different segments of the operator tasks and for specific period of the operator life.

Cho (13) developed a computer system to retrieve historical and current data from LER's. The Weibull Probability Plotting method was applied to operator error data, and estimates of scale and shape parameters were obtained and compared with computer results. It was concluded that the Weibull plotting method is suitable to estimate Weibull parameters for operator errors that have occurred during operation. Three commercial power plants were used to evaluate the code and operator errors.

Azarm (14) constructed a model in prediction of future data for operator error rate for two different types of LWR's (PWR's, BWR's) with respect to power rate and time, depending on smoothing the data extracted from LER's, estimating a static model, and estimating a dynamic model. It was concluded that the learning process of BWR's is almost independent of power, but in PWR's the power of the reactor is one of the important factors on the learning process.

The general description and the safety features of typical HTGR are given in Chapter II. Chapter III describes the classification of the general systems, subsystems, components of HTGR's, for coding the human error based on information provided in LER records. Appendix A lists the subsystem abbreviation, components, failure mode, and other classification codes. Brief description of the events, their causes and consequences, and the error population for each system based on extracted data from LER between May 30, 1974 and December 30, 1977 for Fort St.Vrain reactor HTGR are given in Chapter IV. Method of analysis, failure significance, number of errors, failure (errors) rates, 90% confidence bounds, and the analysis of the distribution of human error over 43-month period are given in Chapter V. Finally, conclusions and recommendations for further work are provided in Chapters VI and VII.

II. HTGR GENERAL DESCRIPTION

High temperature gas-cooled reactor (HTGR) is an advanced thermal reactor, that produces steam, using helium as the primary coolant, graphite as the neutron moderator and as the structural portion of the fuel elements, and a uranium-thorium fuel cycle. The fuel cycle is based on highly enriched uranium for the initial and make-up fissile material, thorium for the fertile material, with the bred U-233 being recycled at the earliest opportunity. Advanced system may use gas turbines. Ceramic fuel is used in the form of coated thorium/uranium carbide or oxide pellets (15). The use of thorium in the fuel cycle results in low fuel cost, conservation of fuel and in adding the large deposit of thorium available to fuel reserves.

The high-temperature and high thermal efficiency (about 39%) of HTGR's result in economic and environmental advantages; such as

1. high performance through conservation of fuel,
2. competitive cost,
3. lower thermal discharge because of its higher efficiency,
4. lower release of radioactive waste because of the high integrity of fuel and the inert gas used as coolant, and

5. low consumption of raw materials because of high efficiency and use of thorium in the fuel cycle.

Gas cooling makes it possible to achieve high operating temperatures at moderate pressures. Helium has the fundamental advantage that it always remains in the same phase, making complete loss of coolant no longer a problem. Among the important special features of helium are

1. pure single-phase operation, no voids
2. it absorbs essentially no neutrons
3. inertness, both chemical and radioactive
4. compatibility with water, air, and fuel
5. total coolant loss is impossible, only depressurization (adequate cooling after shutdown is available)
6. optical transparency permits visual control during fueling and maintenance operation.

Graphite is an excellent moderator, which has been used in thermal reactors. Its low neutron-capture cross section places it high among moderator candidates. No neutrons are lost wihin the core through absorption in metallic fuel cladding or structural material supports. In addition to its nuclear characteristics, graphite is ideally suited to high-temperature operation, since, unlike most materials, it increases in strength at higher temperatures, reaching a maximum at about 4500°F, well above the reactor operating range, and continues to maintain significant strength at much higher temperatures (16, 17). The attractive features of graphite can be summarized as follows:

1. very low neutron capture cross section,
2. excellent thermal conductivity,
3. excellent mechanical strength even at temperatures well beyond the HTGR range,
4. high specific heat, and
5. ease of fabrication.

Graphite provides an adequate sink in case of loss of coolant pressure. Nevertheless, high temperature graphite technology is still in the developmental stage. Cracks have occurred in the relatively low power plants operated until now. Thus, quality of graphite in HTGR's is yet to be improved.

The fact that experience in boilers and water coolants far exceeds utility experience in gas and liquid metal coolants have affected the development of HTGR's and Liquid Metal Cooled Reactors. This is since earlier efforts of development have been devoted to Light Water Reactors (LWR's).

III. HUMAN ERROR CLASSIFICATION, HEGAR: I

In order to provide a format for coding failures, root causes, and consequences related to human errors in HTGR's it is convenient to develop a classification system appropriate for computer analysis and for data collection. The classification of Human Errors in Gas-cooled Reactors (HEGAR) developed here the general systems, subsystems, components of HTGR, for coding the human errors based on information provided in the licensee event reports (LER's). This classification does not provide all systems for each particular design of HTGR, but it mainly includes most safety related systems. However this classification is flexible to permit expansion, change, added for any system, subsystem, and component, and it can be adapted to any HTGR design.

The HEGAR classification is divided into ten systems, each system is represented by an alphabetical character as a letter coding (18-25). Name, abbreviation, description, redundant trains and equipments, function and purposes is presented for each subsystem. For the sake of comparison the WASH-1400 Code (3), and NCR Code (15) is given. The system list is given in Table 3.1. The classification is given in Tables 3.2 to 3.9.

Table 3.1. System code

Code	System	Abbreviation
A	Auxiliary Electrical Power System	AEPS
B	Reactor Protection System	RPS
C	Emergency Cooling System	ECS
D	Main Reactor Coolant System	MRCS
E	Prestressed Concrete Reactor Vessel System	PCRVS
F	Auxiliary System	AS
G	Radioactive Waste Treatment System	RWTS
H	Instrumentation and Monitoring System	IMS
I	Other Systems	O
J	Unknown	U

Table 3.2. Subsystem of A: Auxiliary electrical power system

Subsystem	Abbreviation	Description and Function	WASH 1400 CODE	NRC CODE CODE
Off-Site AC Power System	EPS-Off	Two AC power sources to supply the emergency power needs for EPF systems, and shutdown. (The preferred source.)	Q	EA
On-Site AC Power System	EPS-DG	There are two standby diesel generator sets to provide emergency, in-house power in sufficient quantity to drive all electrical auxiliaries that are essential for shutdown cooling. These generators are started automatically on loss of both off-site sources.	Q,X	EB
On-Site DC Sources	EPS-DC	Two station batteries provide separate and normally independent sources of power for essential DC-powered auxiliaries and services. Each battery is large enough to supply all shutdown direct-current loads for not less than one hour following loss of all alternating-current power.	Q	EC
Auxiliary Equipment for AEPS	EPS-EQP	This includes transformers, cables, buses, protective devices, etc.	Q	EG

Table 3.3. Subsystem of B: Reactor protection system

Subsystem	Abbreviation	Description and Function	WASH 1400 CODE	NRC CODE
Reactor Protection Control Rod System	RPERS	This system is made up of the control rod drive mechanisms, the control rods, and the hydraulic control modulus; these components provide for the rapid insertion of the control rods when a trip condition exists.	3	RB
Reactor Protection Logic System or Scram System	RPLS SS	A general two-of-three logic system is used in the scram circuits of the plant protection system. Three independent sensing circuits are provided for each scram parameter (neutron flux, temperature, pressure, moisture, etc.).	3	IA
Coolant Loop Protection System	CLPS	Tripping of the two helium circulators and shutting off feed water and steam to and from the steam generator of one loop. The reactor remains in operation, and the control system limits plant power to 50% of rated power during a loop shutdown. A logic system (two out of three) is provided for the loop to shutdown in the event of high moisture level in one loop, both loop circulators tripped, etc.	3	CJ
Circulator-Trip System	CTS	It is a shutting down of a single helium circulator. The reactor and both cooling loops remain in operation following the trip of a single circulator. The tripping of two circulators in a loop results in a	3	CJ

Table 3.3 (Continued)

Subsystem	Abbreviation	Description and Function	WASH 1400 CODE	NRC CODE
		loop shutdown. A logic system is provided for each of the four circulators to shut them down in the event of loss of bearing water, over-and-under speed, and other malfunctions.		
Steam-Water Dump System	SWDS	The detection of moisture in a helium loop or high primary-coolant pressure results in feedwater shutoff and steam and water dump of a steam generator. Each stem generator has its own dump system. The sequence of events in a dump is first, to close off the feedwater supply while simultaneously tripping the reactor; then second, to open both parallel dump valves.	3	--
Reactor Reserve Shutdown System	RRSS	A reactor reserve shutdown system functionally independent of the normal control rod system is provided. Neutron absorbing material, in the form of spheres of boron carbide in graphite, will be stored in a hopper in each refueling penetration. This absorber material can be released from hoppers into the core, if required. The system operates by breaking a rupture disc with gas pressure, and letting the boron carbide fall into the reserve shutdown hole in the control element.	3	IC

Table 3.4. Subsystem of C: Emergency cooling system

Subsystem	Abbreviation	Description and Function	WASH 1400 CODE	NRC CODE
Primary Coolant Loop Helium Circulator	PCL-HC	See Main Reactor Coolant System. Each helium circulator of MRCS can use to circulate sufficient coolant to remove decay heat. MRCS is designed as residual heat removal.	M,B	CF
Primary Coolant Loop Steam Generator	PCL-SG	See Main Reactor Coolant System. The function of SG is to transfer the required decay heat load.	M	CF
Core Auxiliary Cooling System	CACS	CACS is an independent mean of cooling the reactor in the event that none of the primary coolant loops are available. CACS has sufficient cooling capacity to provide effective core cooldown and prevent damage to either the core or primary coolant system components. CACS is designed as another separate system for residual heat removal. CACS consists of two independent auxiliary core cooling loops for the 2000 MWt reactor. Each loop contains an auxiliary circulator, an auxiliary heat exchanger, an auxiliary primary cooland shutoff valve, and an auxiliary circulator service system.	M	SF
Core Auxiliary Cooling System Auxiliary Circulator	CACS-AC	The auxiliary circulator consists of electric-motor-driven compressor installed in auxiliary loop penetration. AC pumps the cold helium through the upper cross-duct	M	SF

Table 3.4 (Continued)

Subsystem	Abbreviation	Description and Function	WASH 1400 CODE	NRC CODE
		into the core top plenum for circulation through the core before beginning another auxiliary cooling loop cycle.		
Core Auxiliary Cooling System Auxiliary Circulator Service System	CACS-ACSS	The auxiliary circulator service system provides cooling to the circulator motor starter windings. ACSS transports heat from the heat exchanger to an ultimate heat sink which operates on either the water/air or water/water principle.	M	SF
Core Auxiliary Cooling System Auxiliary Heat Exchanger	CACS-AHE	The auxiliary heat exchanger is designed to provide adequate heat removal during all postulated transients and accidents. The loss of one CACS loop will not prevent the remaining heat exchangers from providing the required heat removal capability.	M	SF
Core Auxiliary Cooling System Auxiliary Primary Coolant Shutoff Valve	CACS-APCSV	The function of the auxiliary primary coolant shutoff valve is to limit back-flow through an auxiliary coolant loop when the main circulators are operating.	M	SF

Table 3.5. Subsystem of D: Main reactor coolant system

Subsystem	Abbreviation	Description and Function	WASH 1400 CODE	NRC CODE
Main Loop Cooling System	MLCS	The system consists of primary and secondary coolant systems. Its function is to generate steam in a main superheat-reheat steam cycle for subsequent conversion into electrical energy by turbine-generator unit. Steam is generated by circulating primary coolant through the core in series with the steam generators via the main circulators, which are driven by cold reheat steam. In addition to the power operation function, the main cooling loops constitute the principal heat removal system during shutdown of the reactor.	1	CB
Primary Coolant System	PCS	The primary coolant system consists of four primary coolant loops for the 2000 MWt reactor. Each loop contains a steam generator, a circulator, and a main helium shutoff valve. The function of PCS is to remove core heat by circulating helium through the core in series with the steam generators. The PCS is contained within the PCRV so that no primary coolant helium leaves the PCRV during its main function of transferring heat from the core to secondary coolant system.	1	CB

Table 3.5 (Continued)

Subsystem	Abbreviation	Description and Function	WASH 1400 CODE	NRC CODE
Primary Coolant Loop Helium Circulator	PCL-HC	HC consists of a single-stage axial-flow helium compressor and single-stage steam turbine drive together with a water-lubricated bearing system and a helium buffer seal system. The circulators are designed to operate under normal and abnormal conditions, including 1. Normal plant operation between rated load and minimum load. 2. Plant startup. 3. Routine plant shutdown for refueling or other maintenance. 4. Plant shutdown following a reactor scram, turbine trip, loop shutdown, steam leak, or primary coolant system depressurization.	1	CB
Primary Cooland Loop Main Circulator Service System	PCL-MCSS	MCSS provides the following 1. A continuous noninterruptible supply of high-pressure, clean, cooled water for circulator bearing support while the circulator shaft is rotating. 2. A continuous cooling capability to remove heat from the water used to support the circulator bearings. 3. A drain water recovery capability which prevents reduction of system water inventory when supplies from external sources are interrupted. 4. A circulator brake and static seal actuating capability.	1	CB

Table 3.5 (Continued)

Subsystem	Abbreviation	Description and Function	WASH 1400 CODE	NRC CODE
		5. A supply and purge from each main helium circulator of buffer helium to the lower helium labyrinths. 6. A supply of clean, high-pressure cooled water for system make up purpose and for backup to the water supplies described under item (1).		
Primary Cooland Loop Steam Generator	PCL-SG	The six steam generation modules are grouped together and called one steam generator. Each steam generation module has an evaporator-economizer-superheater section and a reheater section. Hot helium flows through the generator entering at the reheater and leaving at the economizer end. Steam and water flow counter-current to helium through the economizer, evaporator, and reheater section, but co-current through the superheater section.	1	CB
Primary Cooland Loop Main Helium Shutoff Valve	PCL-MHSV	The purpose of MHSV is to prevent reverse flow through a nonoperating circulator. Each main circulator is provided with a primary coolant shutoff valve.	1	CB
Secondary Coolant System	SCS	It consists of the following major components: consenser, low-pressure feedwater heater, deaerating feedwater heater heater, steam generator feedpump and drive, high pressure feedwater heaters, plant and loop feedwater piping, main steam sections of	1	CC

Table 3.5 (Continued)

Subsystem	Abbreviation	Description and Function	WASH 1400 CODE	NRC CODE
		the steam generators, loop and plant main steam piping, high pressure turbine and main bypass system, cold reheat plant piping, circulator turbines, cold reheat steam at temperation, cold reheat loop piping, reheater sections of the stem generators, loop and plant hot reheat piping, and intermediate-pressure and low-pressure turbines and bypass system which discharge steam to the condenser.		

Table 3.6. Subsystem of E: PCRV system

Subsystem	Abbreviation	Description and Function	WASH 1400 CODE	NRC CODE
PCRV	PCRV	It includes the concrete, reinforcing bars, prestressing systems, cavity and penetration liners, penetration closures, and lines cooling tubes. The PCRV contains the primary coolant system, with its associated equipment for controlling circulation, and portions of secondary coolant system. The vessel is leak tight within specified limits and is capable of resisting normal operating loads plus loads under upset, emergency, and faulted conditions.		SA
PCRV-Pressure Relief System	PCRV-PRS	It is provided for backup protection against overpressure in the incredible event that all plant protection system action should fail. The system will limit the PCRV to the maximum cavity pressure. The system consists of two redundant pressure relief trains. Either of which is adequate to prevent exceeding the PCRV maximum cavity pressure in the event of any credible overpressure accident.	S	SA
PCRV-Cooling System	PCRV-CS	The PCRV concrete requires protection against thermal damage from reactor heat. To insure this, cooling tubes, divided into two system, are fastened to the concrete side of the steel membrance by continuous fillet. Each of the systems is capable alone of controlling concrete temperature within safe limits.		SB

Table 3.7. Subsystem of F: Auxiliary system

Subsystem	Abbreviation	Description and Function	WASH 1400 CODE	NRC CODE
Circulator Auxiliary System	CAS	It supplies buffered helium for injection, bearing water and emergency drive water for the water turbine. All major equipment in both the buffer helium subsystem and the bearing water subsystem--pumps, compressors, heat exchangers, and filters--are duplicated.		AD
Helium Purification System	HPS	It provides purified helium for circulator seals and the top PCRV seals. It purifies helium after the initial fill at plant commissioning, before it is pumped to storage, and after a boiler-tube failure. The system also collects active impurities passing through it. The system has two, full-size processing trains; one is operating, while the other is in activity decay, regeneration, or standby.		PC
Helium Storage System	HSS	It provides storage volume for plant helium inventory during depressurization, furnishes a supply of high-pressure purified helium for plant use. It consists of the following major components: High pressure supply tanks, low pressure storage tanks, oil absorber, transfer compressor, and valves and pumps.		AD
Liquid Nitrogen System	LNS	It is provided in the plant to furnish the temperature required by the low-temperature charcoal absorbers in the helium purification system and a source of cold, dry gas for use in the primary coolant moisture		PC

Table 3.7 (Continued)

Subsystem	Abbreviation	Description and Function	WASH 1400 CODE	NRC CODE
		monitors. The system consists of nitrogen recondensers, storage vessels, and interconnecting piping system.		
Reactor Plant Cooling Water Systems	RPCWS	They are comprised of three systems; viz: 1. Two closed demineralized-water loops serving the PCRV, core support structure, helium purification fuel storage, fuel purge, and the liquid nitrogen system. 2. The service water system serving the water turbine coolers, helium transfer compressors, buffer helium and circulator bearing coolers, radwaste pumps and compressors, the closed demineralized-water loop coolers, and the booster service water system.	E	WB
Reactor Building Ventilation System	RBVS	It has restricted leakage features through the use of appropriate joint designs and sealing materials. It is a "once through" type with separate supply and exhaust systems. The supply system has two 100% capacity subsystems. The exhaust system has three 50% capacity subsystems.	Z	AA

Table 3.8. Subsystem of G: Radioactive waste treatment system

Subsystem	Abbreviation	Description and Function	WASH 1400 CODE	NRC CODE
Waste Processing System--Liquid Handling	WPS-LH	The (liquid waste handling) system provides limited holdup capacity to permit monitoring and controlled release of the liquid wastes, because significant quantities of liquid radioactive waste arise only as the result of planned operations (decontamination of equipment following refueling, or adsorber regeneration). The system provides two 3000-gal liquid-waste-receivers and one 3000-gal liquid-waste-monitoring tank. In addition a pair of demineralizers, with the resin contained in replaceable cartridges, would be provided to permit processing of any solutions having unexpectedly high radioactivity. Disposal of liquid wastes could be achieved by pumping at controlled rate into the cooling-tower blowdown line for dilution prior to release from the plant site. Alternatively the use of filters and a reverseosmosis unit in the liquid-waste-purification train to supplement the resin-bed demineralizer could provide high-purity water suitable for reuse within the plant. There would be no liquid-waste release from the system, and all radioactivity removed could be shipped off-site in solid form.	N	MA

Table 3.8 (Continued)

Subsystem	Abbreviation	Description and Function	WASH 1400 CODE	NRC CODE
Waste Processing System-- Gas Handling	WPS-GH	Gaseous wastes result primarily from regeneration of the low-temperature absorption beds in the helium purification system. The gas contains essentially only ^{85}Kn. The gas handling system typically provides two 1100-ft^3 surge tanks together with compressors to permit temporary storage and decay of gaseous waste. The gas can be processed by the radioactive-gas recovery system, which separates the gas into a nonradioactive stream for release to the atmosphere and a radioactive concentrate. This concentrate is stored within the operating helium-purification system. The radioactive gaseous wastes can be withdrawn from the system for disposal by off-site shipment or by controlled release to the atmosphere.	N,8	MB
Waste Processing System-- Solid Handling	WPS-SH	Solid radioactive waste, other than spent fuel, has a low radioactivity level. Tritium can be removed from the helium-purification system as solid with the titanium getter unit. Other solids include laboratory waste, filters, and used reflector blocks.	N	MD

Table 3.9. Subsystem of H: Instrumentation and monitoring system

Subsystem	Abbreviation	Description and Function	WASH 1400 CODE	NRC CODE
Area Radiation Monitoring System	ARMS	The ARMS has about 20 area monitors. They are: 1. Refueling--machine control room 2. Hot-service-facility platform 3. Hot-service-facility blower suction 4. Instrument room-analytical board 5. Valve-operating stations (2) 6. Radiochemical laboratory 7. Stairwells (3) 8. Walkways (4) 9. Operating area (2) 10. Reactor-plant-exhaust filter room 11. Office area 12. Control room 13. Condensate-demineralizer area Monitors are chosen to match the expected radiation levels. They are including equipment monitors, liquid monitors, gas monitors, and particulate and iodine monitors.		BA
Process Radiation Monitoring System	PRMS	Its function is to monitor certain plant processes to detect radioactivity in excess of acceptable limits. 1. Steam and water dump tank monitoring 2. Loop reheat steam header monitoring 3. PCRV relief-valve piping monitoring 4. Loop reheat steam header condensate monitoring 5. Helium circulator bearing water drain monitoring		MC

Table 3.9 (Continued)

Subsystem	Abbreviation	Description and Function	WASH 1400 CODE	NRC CODE
		6. Radioactive-liquid and gas-waste exhaust monitoring 7. Radioactive-gas-waste compressor cooling water monitoring 8. Reactor-plant ventilation (low and long range) monitoring 9. Building-air radioactivity monitoring 10. Air-ejector exhaust line monitoring 11. Primary coolant system monitoring 12. Helium-purification-system outlet monitoring		
Reactor Control and Information System	RCIS	The control-rod drives are controlled by the operator with switches on the reactor control board. Electric power to each drive mechanism is supplied through a reversible motor starter. The contactor coils are energized directly by control switches. There are three sets of controls. 1. individual rod control 2. rod-group selection and control 3. automatic rod control Each set can be operated under appropriate conditions. Position indicators are provided, one for each drive. A flashing feature on the limit lights indicates: which rods are being driven at any instant and in which direction.		IE

Table 3.9 (Continued)

Subsystem	Abbreviation	Description and Function	WASH 1400 CODE	NRC CODE
Reactor and In-core Instrumentation	RICI	1. Nuclear measurements: a. Neutron sensors: They are out-of-core to measure neutron flux and gamma exposure rate. b. Installation sensors: They are installed inside the PCRV. c. Channels: They use to prevent control rod withdrawal without source indication and to produce the flux-controller input signal. 2. Temperature measurement: a. Thermocouples: Used to increase various core, coolant, and steam generator, circulator, and core-support temperatures. b. Acoustic thermometry: (Peach Pottom) As an added in-core temperature-measuring system. 3. Neutron sensors for in-core monitoring: It provides information on the neutron flux distribution. 4. Failed-fuel-element detection: They are located by physical inspection during fuel handling. 5. PCRV-instrumentation and data acquisition: The system includes signal conditioning, programming, excitation, and alarms required for monitoring vibrating-wire strain gages etc. Output of the system appears on printed tape and	C	IB

Table 3.9 (Continued)

Subsystem	Abbreviation	Description and Function	WASH 1400 CODE	NRC CODE
		punched paper tape for further computer processing. The system can continually or intermittently monitor all the channels or selected channels.		

IV. DATA CLASSIFICATION AND ERROR POPULATION

For the 43-month period from May 30, 1974 to December 30, 1977, the LER's were manually reviewed to extract events related to human error for the Fort St.Vrain, HTGR. An event occurring during startup period or normal operation (construction period not included) which has involved human errors; such as, operator error, human maintenance error, or administrative error is recorded. Human error in design or installation was not considered.

Operator errors include only those errors directly attributable to licensed operators only which are not caused by deficiencies in procedures or by failure of component or instrumentation. Operator errors must be distinguished from system errors which involve inadequate or the lack of operation procedures, instruments, logistics or personnel (6, 26). An operator performing a task incorrectly because a checklist procedure gives the operator incorrect operation instruction is not considered as an operator error. Errors in chemical analysis, testing and calibration are considered as operator errors.

Maintenance errors include only those directly attributable to maintenance personnel and not those caused by deficiencies in maintenance procedures. Component failures that may have been avoided by more stringent preventative maintenance are not considered. Errors in filter changing and installation of equipment after commercial operation began is included among maintenance errors.

Administrative errors basically include all those human errors not directly attributable to operator or maintenance or installation personnel. Examples include errors or deficiencies in procedures, calculational errors, judgmental errors and insufficient control over plant operations.

The taxonomy given in Table 4.1 is designed such that information of great importance are included. Emphasis is placed upon subsystems and components involved, number of components involved, the failure mode, the effect of the error on the subsystems and components involved, the effect of the error on environment, mode of the human error (omission/commission, common mode/recurring unusual), and a very brief description of each event, its cause and consequence. The classification in Chapter III and component code, failure mode code, subsystem abbreviation in Appendix A are used for coding human errors in Table 4.1.

The human events are then classified for analysis. Error population for each subsystem and component involved, type of error, failure mode, type of the effect of the error on the involved subsystem and component, type of the effect of the error on environment, and the type of mode of the human error are collated and categorized by system, and they are presented in Table 4.2 through Table 4.19.

Table 4.1. Error leading to significant events

Ident. No.	Page	Ref. No.	Facility I.D. Type	Date	System Involved	Component Involved	No. of Components Involved	Effect on System	Effect on Component
1	36	092176	FSV	053074	RPCRS	Rod	2	N	N
2	37	093028	FSV	060374	PCL-HC	Valve	U	D	D
3	37	093706	FSV	061074	RPCRS	CRD (drive mechanism)	1	D	F
4	38	095383	FSV	081674	EPS-DG	Switch	1	F	F
5	40	096069	FSV	092474	RPCRS	Connectors	U	N	F
6	42	097144	FSV	103174	EPS-DG	Switch	1	F	D

Failure Mode - Type of Error	Omission/ Commission	Duration	Rad. Release to Environ.	Rad. Exposure	Common Mode Recurring Unusual Events	Further Description
Procedural Deficiency-A	O	N/A	NO	NO		2 rod groups were withdrawn as a test to determine the difference in critical rod height with the core in air and helium. This movement violated tech. spec.
Improper Setting-M	O	N/A	NO	NO	R	Several small valve packing leaks which were repaired caused a significant loss of plant helium.
Improper Operation-M	O	N/A	NO	NO		Inadv. operation of CRD drive motor with shipping/manual tool in placed resulted in damaging 3 bolts & 2 dowel pins & limit switches.
Carelessness-M	C	N/A	NO	NO		A loss screw in a switch caused improper switch operation.
Installation-M	O	N/A	NO	NO	C	During removal of control rods, a small quantity of moisture contacted the connectors. This caused to destroy the pressure sealing capability of the connectors.
Misadjustment-M	O	N/A	NO	NO		The shutdown mechanism, actuated by tow oil pressure or engine over speed, had tripped. This caused the engine of diesel generator failed to start.

Table 4.1 (Continued)

Ident. No.	Page	Ref. No.	Facility I.D. Type	Date	System Involved	Component Involved	No. of Components Involved	Effect on System	Effect on Component
7	43	097491	FSV	112074	RCS	Event	-	NO	-
8	44	099094	FSV	120174	WPS-GH	Valve	1	D	F
9	44	099095	FSV	120974	WPS-GH	Rupture disc	1	D	F
10	45	100077	FSV	010475	PCL-HC	Valve	1	D	N
11	47	099706	FSV	020475	WPS-GH	Rupture disc	1	D	F

Failure Mode - Type of Error	Omission/ Commission	Duration	Rad. Release to Environ.	Rad. Exposure	Common Mode Recurring Unusual Events	Further Description
Event-A	O	N/A	NO	NO		A conflict in the limits for moisture in the primary coolant
Left Closed-O	O	N/A	RWL	NO	U	The inlet valve of gas waste tank B had been handjacked shut from last release to prevent leakage. This caused a release uncontaminated gas to the ventilation system exhaust.
Wrong Proced.-A	O	N/A	RWL	NO		The high flow exceeded the capacity of both gas waste compressors and blew the rupture disc. This caused a release uncontaminated helium.
Deenergized-O	O	N/A	NO	NO		While removing temporary jumpers, it contacted station ground blowing a fuse. This deenergized a relay causing to release the static seal on helium circulators
Wrong Proced.-A	O	N/A	RWL	NO	U	The regeneration compressor was started, two valves opened disc in the gas waste vacuum tank to fail from overpressure. The procedure failed to specify the proper valve line-up prior to starting the compressor.

Table 4.1 (Continued)

Ident. No.	Page	Ref. No.	Facility I.D. Type	Date	System Involved	Component Involved	No. of Components Involved	Effect on System	Effect on Component
12	48	099663	FSV	020775	EPS-DG	Relay	1	F	F
13	48	105284	FSV	030475	RCS	Event	-	D	-
14	52	103052	FSV	050775	EPS-DG	Engine	1	D	D
15	54	103638	FSV	061775	EPS-DG	Engine	1	NO	NO
16	58	106330	FSV	090575	EPS-DG	Engine	1	NO	NO
17	58	100029	FSV	092275	RPCRS	CRD	1	NO	NO

Failure Mode - Type of Error	Omission/ Commission	Duration	Rad. Release to Environ.	Rad. Exposure	Common Mode Recurring Unusual Events	Further Description
Improper Setting-M	O	N/A	NO	NO		Diesel generators A started but the A bus relay sensing voltage was low, this caused a start signal to be not generated for B unit.
Inadvertent Actuation-O	O		NO	NO		Inadvertent release of one hopper into the core and excessive moisture in main coolant caused a reactivity anomaly to be observed.
Did Not Test-M	O	N/A	NO	NO		Diesel engine tripped from high water jacket temperature.
Procedural Deficiency-A	O	N/A	NO	NO		During DG overspeed testing, the engine ran at 1900 RPM and the max. recommended is 1650. This caused to shutdown the engine.
Improper Setting-M	O	N/A	NO	NO		The different pressure in fuel rack setting between A and B engines caused to shutdown one of the engines automatically.
Procedural Deficiency-A	O	N/A	NO	NO		Procedures did not require a check of the orientation position caused control rod drive to be inserted in improper rotation.

Table 4.1 (Continued)

Ident No.	Page No.	Ref. No.	Facility I.D. Type	Date	System Involved	Component Involved	No. of Components Involved	Effect on System	Effect on Component
18	59	106623	FSV	092675	EPS-DC	Cable	-	NO	NO
19	60	109201	FSV	092975	RPCRS	CRD	1	N	D
20	62	108272	FSV	111875	EPS-EQP	Breaker	1	F	F
21	62	109250	FSV	111875	PCL-HC	Seal	1	F	N
22	63	108522	FSV	120475	EPS-DG	Engine	1	F	N
23	66	110311	FSV	011576	PCRV-PRS	Valve	-	N	N
24	67	110312	FSV	012676	PCL-HC	Seal	-	F	N
25	69	112722	FSV	031876	RPS-CTS	Switch	-	F	F

Failure Mode - Type of Error	Omission/ Commission	Duration	Rad. Release to Environ	Rad. Exposure	Common Mode Recurring Unusual Events	Further Description
Carelessness-M	O	N/A	NO	NO		A sharp metal cut through a cable insulation causing the battery charger tripped open.
Inadvertent Actuation-M	C	N/A	NO	NO		Inadvertent admitted water into PCRV saturated the gas, some entered the hopper and leaked part of the boric acid out of the balls.
Did Not Check-M	O	N/A	NO	NO		Loose electrical connection on the bus of supply fan breaker caused to fail to start the fan.
Lack of Repair-M	O	N/A	NO	NO		Repair work caused to release the static seal.
Communication-O	O	N/A	NO	NO		Losing the information about setting the switches caused to fail to start.
Communication-A	O	N/A	NO	NO		Modification to oil system relief valves made without proper approval.
Carelessness-M	-	N/A	NO	NO		Either the holding circuit was interrupted or the reset circuit was made up. This caused to release seal set on helium circulator.
Improper Handling-O	O	N/A	NO	NO	C	Each trip was initiated by buffer seal differential pressure switches.

Table 4.1 (Continued)

Ident. No.	Page No.	Ref. No.	Facility I.D. Type	Date	System Involved	Component Involved	No. of Components Involved	Effect on System	Effect on Component
26	69	112722	FSV	031876	RPS-CTS	Switch	-	F	F
27	69	112722	FSV	031876	RPS-CTS	Switch	-	F	F
28	69	112722	FSV	031876	RPS-CTS	Switch	-	F	F
29	69	112722	FSV	031876	RPS-CTS	Switch	-	F	F
30	70	112699	FSV	040176	PCL-HC	Valve	-	F	N
31	71	113197	FSV	041476	EPS-DG	Clutch	1	D	D
32	72	113958	FSV	051376	PCL-HC	Valve	1	N	F

Failure Mode - Type of Error	Omission Commission	Duration	Rad. Release to Environ.	Rad. Exposure	Common Mode Recurring Unusual Events	Further Description
Improper Handling-O	C	N/A	NO	NO	C	Each trip was initiated by buffer seal differential pressure switches.
Improper Handling-O	C	N/A	NO	NO	C	Each trip was initiated by buffer seal differential pressure switches.
Procedural Deficiency-A	C	N/A	NO	NO	C	Each trip was initiated by buffer seal differential pressure switches.
Procedural Deficiency-A	C	N/A	NO	NO	C	Each trip was initiated by buffer seal differential pressure switches.
Did Not Follow Proced.-M	C	N/A	NO	NO		The valves were closed rapidly causing a pressure surge in sensing lines. This caused to trip helium circulator.
Procedural Deficiency-A	O	N/A	NO	NO		The clutch control linkage was set too close to center and clutch engaging torque was too high. This caused to move the clutch in declutch direction.
Improper Permission-A	C	N/A	NO	NO		Shift supervisor gave improper permission for the valve striking with loop 1 operation, caused to trip the circulator.

Table 4.1 (Continued)

Ident. No.	Page No.	Ref. No.	Facility I.D. Type	Date	System Involved	Component Involved	No. of Components Involved	Effect on System	Effect on Component
33	73	113959	FSV	051376	PCL-HC	Pump	-	D	F
34	73	114636	FSV	060176	PCL-HC	Pump	-	D	F
35	74	114635	FSV	060776	PCL-HC	Pump	-	D	F
36	76	115505	FSV	061776	PCL-HC	Event	-	D	F
37	77	115500	FSV	062676	PCL-SG	Valve	1	N	N
38	78	115506	FSV	062676	PCL-HC	Pumps	-	F	F

Failure Mode - Type of Error	Omission/ Commission	Duration	Rad. Release to Environ.	Rad. Exposure	Common Mode Recurring Unusual Events	Further Description
Improper Handling-O	O	N/A	NO	NO		Operator didn't have surge tank makeup from either the bearing water makeup pumps or from the makeup bearing water system. This caused to the circulator to trip.
Improper Handling-O	O	N/A	NO	NO		A plugged filter in buffer gas return line resulted flow blockage which caused to trip helium circulator.
Procedural Deficiency-A	C	N/A	NO	NO	R	Steam turbine trips were reset. The procedure for reset had not been revised. This caused to trip the circulator.
Did Not Follow Proced.-O	O	N/A	NO	NO		Procedure not followed caused in an anticipated trip signal to two helium circulators.
Left Open-O	O	N/A	NO	NO	U	The operator noticed that a steam water dump valve in each loop was open and the alarm light was lit. No cause has been found.
Improper Handling-O	C	N/A	NO	NO	R	The operator didn't make the surge tank makeup supplied from either the bearing water makeup pumps or from backup bearing water system. This caused to trip both helium circulators.

Table 4.1 (Continued)

Ident. No.	Page No.	Ref. No.	Facility I.D. Type	Date	System Involved	Component Involved	No. of Components Involved	Effect on System	Effect on Component
39	79	116263	FSV	062876	PCL-HC	Bearing	1	F	F
40	80	115253	FSV	071676	PCL-HC	Breaker	1	F	N
41	80	115884	FSV	072676	PCL-H	Valve	1	F	F
42	81	118726	FSV	093076	WPS-LH	Pump	1	N	N

Failure Mode - Type of Error	Omission/ Commission	Duration	Rad. Release to Environ.	Rad. Exposure	Common Mode Recurring Unusual Events	Further Description
Procedural Deficiency-A	O	N/A	NO	NO		Changing rapidly in backup bearing water pressure resulted a negative buffer differential pressure. This caused to trip the circulator.
Mispositioning-O	O	N/A	NO	NO		The handles for adjusting voltage and the output breaker were identical and above each other, the operator closed the breaker when he intended to adjust the voltage this caused to trip the circulator
Did Not Follow Proced.-A	C	N/A	NO	NO	U	Scheduling this test without shutting down the circulator auxiliary system caused to trip the circulator.
Procedure Violation-A	O	N/A	RWL	NO	U	A reactor building sump pump was found to be running concurrently with a liquid waste release in violation to technical specifications because the operator failed to remember the note which omitted one step which appeared early in the procedure.

Table 4.1 (Continued)

Ident No.	Page No.	Ref. No.	Facility I.D. Type	Date	System Involved	Component Involved	No. of Compo- nents Involved	Effect on System	Effect on Component
43	82	120092	FSV	101476	PCL-HC	Transmitter	1	F	F
44	83	119004	FSV	102076	EPS-DG	Tube	-	F	D
45	84	120090	FSV	110976	IMS	Detector	-	N	N
46	85	121547	FSV	010677	EPS-DG	Tube	-	F	D
47	86	121603	FSV	010677	PCL-HC	Filter	1	F	F
48	86	121548	FSV	011477	RPLS	Connector	1	F	F
49	87	121602	FSV	011977	ECS-CACS	Pump	1	D	F

Failure Mode - Type of Error	Omission/ Commission	Duration	Rad. Release to Environ.	Rad. Exposure	Common Mode Recurring Unusual Events	Further Description
Calibration-M	O	N/A	NO	NO	U	The main drain to buffer helium differential controller caused the circulator high pressure separator to flood allowing water enter the main cooling system.
Did Not Check-M	O	N/A	NO	NO	C	Dirty tubes in heat exchangers caused to trip both engines on diesel generators.
Procedure Violation-A	O	N/A	NO	NO		It was found that a conflict existed between 2 technical specs. on moisture limits.
Did Not Check-M	C	N/A	NO	NO	R	Dirty tubes in heat exchangers caused to trip both engines.
Improper Handling-O	O	N/A	NO	NO	R	The filter in line was plugged caused to trip the helium circulator.
Did Not Connect-M	O	N/A	NO	NO		The connector in the molded plug assembly was not proper contact. This caused the shutdown logic to fail.
Did Not Connect-M	O	N/A	NO	NO		The speed switch drive cable was disengaged from the spindle. This caused the interim auxiliary cooling method diesel driven pump to shutdown.

Table 4.1 (Continued)

Ident. No.	Page No.	Ref. No.	Facility I.D. Type	Date	System Involved	Component Involved	No. of Components Involved	Effect on System	Effect on Component
50	89	121598	FSV	011977	PCL-HC	Event	-	F	-
51	91	122399	FSV	021077	EPS-DC	Battery charger	1	D	F
52	91	122210	FSV	021477	PCL-HC	Valve	1	D	F
53	93	122432	FSV	021477	SCS	Valve	1	D	D
54	93	122432	FSV	021477	IMS	Detector	1	D	F
55	93	122433	FSV	021477	PCRV-CS	Switch	1	D	F
56	93	122580	FSV	021477	PCL-HC	Seal	1	D	F
57	94	123161	FSV	022277	EPS-EQP	Breaker	1	F	F

Failure Mode - Type of Error	Omission/ Commission	Duration	Rad. Release to Environ.	Rad. Exposure	Common Mode Recurring Unusual Events	Further Descriptions
Procedural Deficiency-A	O	N/A	NO	NO	R	Procedural deficiency caused the circulator to trip on loss of bearing water.
Improper Handling-O	O	N/A	NO	NO	U	Improper switching and battery charger failure interrupted instrument bus voltage which caused to an automatic reactor scram.
Did Not Check-M	O	N/A	NO	NO		Grease and dirt on valve stem caused the valve to fail to close.
Did Not Check-M	O	N/A	NO	NO		Fine rust colored deposit caused the steam outlet valve to fail to operate properly.
Improper Handling-M	O	N/A	NO	NO		The circuit board became partially unplugged from excessive handling. This caused moisture monitor response to be unsatisfactory.
Did Not Check-M	O	N/A	NO	NO		Dirt in the hydraulic oil reservoir level switch caused to trip the pumps.
Ruptured-A	O	N/A	NO	NO	R	Fatigue-cycle caused the helium circulator static seal bellows to fail.
Connection-M	O	N/A	NO	NO		Loose terminal block connections caused the breaker to fail to open automatically.

Table 4.1 (Continued)

Ident. No.	Page No.	Ref. No.	Facility I.D. Type	Date	System Involved	Component Involved	No. of Components Involved	Effect on System	Effect on Component
58	95	123162	FSV	030977	EPS-DG	Breaker	1	D	F
59	97	124335	FSV	042577	AS-LNS	Event	-	F	-
60	102	126017	FSV	071377	WPS-LH	Radiation monitors	2	D	F
61	102	126016	FSV	071377	PCL-HC	Event	-	F	-
62	103	126978	FSV	072177	WPS-LH	Event	-	D	-
63	105	128936	FSV	090277	EPS-EQP	Switch	1	F	N
64	106	130092	FSV	100377	AS-RBVS	Switch	1	D	N
65	106	130093	FSV	100677	Cooling pond	Event	1	N	N

Failure Mode - Type of Error	Omission/ Commission	Duration	Rad. Release to Environ.	Rad. Exposure	Common Mode Recurring Unusual Events	Further Descriptions
Procedural Deficiency-A	O	N/A	NO	NO		Control device broken during maintenance work, caused the DG output circuit breaker to fail to close.
Event-A	C	24	NO	NO	U	Late delivery of liquid nitrogen caused to shut down the reactor.
Did Not Monitor-M	O	N/A	RWL	NO	C	Radiation monitor filter plugged which caused to release liquid waste with no monitoring.
Did Not Follow Proced.-O	O	N/A	NO	NO		The operator failed to follow procedures. This caused to trip the circulator.
Calculation-A	O	N/A	RWL	NO		Incorrect calculations caused a liquid waste to be released.
Improper Setting-M	O	N/A	NO	NO		Electrical connector improperly mated caused steam pipe rupture detector to fail to trip.
Carelessness-O	O	N/A	NO	NO	U	The control switch for the louver group found in an intermediate position. This caused a group of louvers to be inadvertently opened.
Procedural Deficiency-A	O	N/A	NO	NO		Inadequate procedures caused the circulating water storage pond level to be below limit.

Table 4.1 (Continued)

Ident. No.	Page No.	Ref. No.	Facility I.D. Type	Date	System Involved	Component Involved	No. of Components Involved	Effect on System	Effect on Component
66	107	130762	FSV	103177	Support structure	Snubber	12	N	F
67	108	130761	FSV	110477	PCL-HC	Valve	1	F	N
68	108	131768	FSV	120577	PCL-HC	Event	-	F	-
69	109	132170	FSV	121477	PCL	Event	-	N	-
70	110	132154	FSV	121477	EPS-DG	Relay	1	F	F
71	111	132153	FSV	121477	PCL-HC	Circuit	1	D	D

Failure Mode - Type of Error	Omission/ Commission	Duration	Rad. Release to Environ.	Rad. Exposure	Common Mode Recurring Unusual Events	Further Descriptions
Did Not Check-M	O	N/A	NO	NO	C	Hydraulic snubber found inoperable due to leak.
Improper Setting-M	O	N/A	NO	NO		Feed water valve set too high caused to trip the circulator.
Improper Action-O	O	N/A	NO	NO		Circulator auxiliary controls were failed by personnel error.
Improper Handling-A	O	N/A	NO	NO	U	Excessive level of carbon monoxide and dioxide caused primary coolant impurities to exceed limit.
Procedural Deficiency-A	O	N/A	NO	NO		Improper procedures caused to actuate the relay and to trip the generator.
Midadjustment-M	O	N/A	NO	NO	U	Electronic circuit unbalance caused circulator speed indication reading incorrect.

Table 4.2. Error leading to significant events in auxiliary electric power system

Subsystem	Component	Type of Error	Failure Mode
EPS-DG	Switch	Maintenance	Carelessness -
EPS-DG	Switch	Maintenance	Misadjustment -
EPS-DG	Relay	Maintenance	Improper Setting -
EPS-DG	Engine	Maintenance	Did not test -
EPS-DG	Engine	Administrative	Procedural Deficiency-
EPS-DG	Engine	Maintenance	Improper Setting -
EPS-DC	Cable	Maintenance	Carelessness -
EPS-EQP	Breaker	Maintenance	Did not check -
EPS-DG	Engine	Operator	Communication -
EPS-DG	Clutch	Administrative	Procedural Deficiency-
EPS-DG	Tube	Maintenance	Did not check -
EPS-DG	Tube	Maintenance	Did not check -
EPS-DC	Charger	Operator	Improper Handling -
EPS-EQP	Breaker	Maintenance	Connection -
EPS-DG	Breaker	Administrative	Procedural Deficiency-
EPS-EQP	Switch	Maintenance	Improper Setting -
EPS-EQP	Relay	Administrative	Improper Deficiency-

Effect on System	Effect on Component	Omission Commission	Radiation Release to Environment	Common Mode Recurring Unusual	ID. No.
F	F	C	No		4
F	D	O	No		6
F	F	O	No		12
D	D	O	No		14
N	N	O	No		15
N	N	O	No		16
N	N	O	No		18
F	F	O	No		20
F	N	O	No		22
D	D	O	No		31
F	D	O	No	C	44
F	D	C	No	R	46
D	F	O	No	U	51
F	F	O	No		57
D	F	O	No		58
F	N	O	No		63
F	F	O	No		70

Table 4.3. Error population in auxiliary electric power system

	No. of Events
Subsystem	
EPS-DG	11
EPS-EQP	4
EPS-DC	2
EPS (other)	0
Component	
Switch	3
Relay	2
Engine	4
Cable	1
Breaker	3
Clutch	1
Tube	2
Charger	1
Type of Error	
Maintenance	11
Administrative	4
Operator	2
Failure Mode	
Carelessness	2
Communication	1
Connection	1
Did not test/check	4
Improper Handling	1
Improper Setting	3

Table 4.3 (Continued)

	No. of Events
Misadjustment	1
Procedural Deficiency	4
Effect on Subsystem	
Degraded	4
Failed	10
None	3
Effect on Component	
Degraded	5
Failed	8
None	4
Omission/Commission	
Omission	15
Commission	2
Radiation Release to Environment	
No Release	17
Release within Limits	0
Release Exceeds Limits	0
Common Mode Recurring Unusual	
Common Mode	1
Recurring	1
Unusual	1

Table 4.4. Error leading to significant event in reactor protection system

Subsystem	Component	Type of Error	Failure Mode	
RPCRS	CRD	Administrative	Procedural Deficiency	-
RPCRS	CRD	Administrative	Procedural Deficiency	-
RPCRS	CRD	Maintenance	Improper Operation	-
RPCRS	Connectors	Maintenance	Installation	-
RPCRS	CRD	Administrative	Procedural Deficiency	-
RPCRS	CRD	Maintenance	Inadvertent Activation	-
RPC-CTS	Switch	Operator	Improper Handling	-
RPC-CTS	Switch	Operator	Improper Handling	-
RPC-CTS	Switch	Operator	Improper Handling	-
RPC-CTS	Switch	Administrative	Procedural Deficiency	-
RPC-CTS	Switch	Administrative	Procedural Deficiency	-
RPLS	Connector	Maintenance	Did not connect	-

Effect on System	Effect on Component	Omission Commission	Radiation Release to Environment	Common Mode Recurring Unusual	ID. No.
N	N	O	No		1
N	N	O	No		1
D	F	O	No		3
N	F	O	No	C	5
N	N	O	No		17
N	D	C	No		19
F	F	O	No	C	25
F	F	C	No	C	26
F	F	C	No	C	27
F	F	C	No	C	28
F	F	C	No	C	29
F	F	O	No		48

Table 4.5. Error population in reactor protection system

	No. of Events
Subsystem	
RPCRS	6
RPC-CTS	5
RPLS	1
RPS (other)	0
Component	
CRD	5
Connector	2
Switch	5
Type of Error	
Administrative	5
Maintenance	4
Operator	3
Failure Mode	
Did not Connect	1
Improper Handling	3
Improper Operation	1
Inadvertent Actuation	1
Installation	1
Procedural Deficiency	5
Effect on Subsystem	
Degraded	1
Failed	6
None	5

Table 4.5 (Continued)

	No. of Events
Effect on Component	
Degraded	1
Failed	8
None	3
Omission/Commission	
Omission	7
Commission	5
Radiation Release to Environment	
No Release	12
Release Within Limits	0
Release Exceeds Limits	0
Common Mode Recurring Unusual	
Common Mode	6
Recurring	0
Unusual	0

Table 4.6. Error leading to significant events in emergency cooling system

Subsystem	Component	Type of Error	Failure Mode	
ECS-CACS	Pump	Maintenance	Did not connect	-

Effect on System	Effect on Component	Omission Commission	Radiation Release to Environment	Common Mode Recurring Unusual	ID. No.
D	F	O	No		49

Table 4.7. Error population in emergency cooling system

	No. of Events
Subsystem	
ECS-CACS	1
ECS (other)	0
Component	
Pump	1
Type of Error	
Maintenance	1
Administrative	0
Operator	0
Failure Mode	
Did not Connect	1
Effect on Subsystem	
Degraded	1
Failed	0
None	0
Effect on Component	
Degraded	0
Failed	1
None	0
Omission/Commission	
Omission	1
Commission	0

Table 4.7 (Continued)

	No. of Events
Radiation Release to Environment	
No Release	1
Release Within Limits	0
Release Exceeds Limits	0
Common Mode Recurring Unusual	
Common Mode	0
Recurring	0
Unusual	0

Table 4.8. Error leading to significant event in main reactor coolant system

Subsystem	Component	Type of Error	Failure Mode	
PCL-HC	Valve	Maintenance	Improper Setting	-
PCL	Event	Administrative	Event	-
PCL-HC	Valve	Operator	Deenergized	-
PCL	Event	Operator	Inadvertent Activation	-
PCL-HC	Seal	Maintenance	Lack of Repair	-
PCL-HC	Seal	Maintenance	Carelessness	-
PCL-HC	Valve	Maintenance	Did not Follow Procedure	-
PCL-HC	Valve	Administrative	Improper Permission	-
PCL-HC	Pump	Operator	Improper Handling	-
PCL-HC	Pump	Operator	Improper Handling	-
PCL-HC	Pump	Administrative	Procedural Deficiency	-
PCL-HC	Event	Administrative	Did not Follow Procedure	-
PCL-SG	Valve	Operator	Left Open	-
PCL-HC	Pump	Operator	Improper Handling	-
PCL-HC	Bearing	Administrative	Procedural Deficiency	-
PCL-HC	Breaker	Operator	Mispositioning	-
PCL-HC	Valve	Administrative	Did not Follow Procedure	-
PCL-HC	Transmitter	Maintenance	Calibration	-
PCL-HC	Filter	Operator	Improper Handling	-
PCL-HC	Event	Administrative	Procedural Deficiency	-
PCL-HC	Valve	Maintenance	Did not check	-
SCS	Valve	Maintenance	Did not check	-
PCL-HC	Seal	Administrative	Ruptured	-
PCL-HC	Event	Operator	Did not Follow Procedure	-
PCL-HC	Valve	Maintenance	Improper Setting	-
PCL-HC	Event	Operator	Improper Activation	-
PCL	Event	Administrative	Improper Handling	-
PCL-HC	Circuit	Maintenance	Misadjustment	-

Effect on System	Effect on Component	Omission Commission	Radiation Release to Environment	Common Mode Recurring Unusual	ID. No.
D	D	O	No	R	2
N	-	O	No		7
D	N	O	No		10
D	-	O	No		13
F	N	O	No		21
F	N	-	No		24
F	N	C	No		30
N	F	C	No		32
D	F	O	No		33
D	F	O	No		34
D	F	C	No	R	35
D	F	C	No		36
N	N	O	No	U	37
F	F	C	No		38
F	F	O	No		39
F	N	O	No		40
F	F	C	No	U	41
F	F	O	No	U	43
F	F	O	No	R	47
F	-	O	No	R	50
D	F	O	No		52
D	D	O	No		53
D	F	O	No	R	56
F	-	O	No		61
F	N	O	No		67
F	-	O	No		68
N	-	O	No	U	69
D	D	O	No	U	71

Table 4.9. Error population in main reactor coolant system

	No. of Events
Subsystem	
PCL-HC	23
PCL-SG	1
SCS	1
PCL (other)	3
Component	
Valve	9
Event	7
Seal	3
Pump	4
Bearing	1
Breaker	1
Transmitter	1
Filter	1
Circuit	1
Type of Error	
Administrative	9
Maintenance	9
Operator	10
Failure Mode	
Calibration	1
Deenergized	1
Did not Check	2
Did not Follow Procedure	4
Event	1
Improper Actuation	1
Improper Handling	5
Improper Permission	1

Table 4 .9 (Continued)

	No. of Events
Improper Setting	2
Inadvertent Actuation	1
Lack of Repair	1
Left Open	1
Misadjustment	1
Mispositioning	1
Procedural Deficiency	3
Ruptured	1
Carelessness	1
<u>Effect on Subsystem</u>	
Degraded	11
Failed	13
None	4
<u>Main Reactor Coolant System</u>	
Degraded	3
Failed	12
Non	7
<u>Omission/Commission</u>	
Omission	21
Commission	6
<u>Radiation Release to Environment</u>	
No Release	28
Release Within Limits	0
Release Exceeds Limits	0

Table 4.9 (Continued)

	No. of Events
<u>Common Mode Recurring Unusual</u>	
Common Mode	0
Recurring	5
Unusual	5

Table 4.10. Error leading to significant event in prestressed concrete reactor vessel system

Subsystem	Component	Type of Error	Failure Mode	
PCRV-PRS	Valve	Administrative	Communication	-
PCRV-CS	Switch	Maintenance	Did not check	-

Effect on System	Effect on Component	Omission Commission	Radiation Release to Environment	Common Mode Recurring Unusual	ID. No.
N	N	O	No		23
D	F	O	No		55

Table 4.11. Error population in prestressed concrete reactor vessel system

	No. of Events
Subsystem	
PCRV-PRS	1
PCRV-CS	1
PCRV (other)	0
Component	
Valve	1
Switch	1
Type of Error	
Administrative	1
Maintenance	1
Operator	0
Failure Mode	
Communication	1
Did not Check	1
Effect on Subsystem	
Degraded	1
Failed	0
None	1
Effect on Component	
Degraded	0
Failed	1
None	1
Omission/Commission	
Omission	2
Commission	0

Table 4.11 (Continued)

	No. of Events
Radiation Release to Environment	
No Release	2
Release Within Limits	0
Release Exceeds Limits	0
Common Mode Recurring Unusual	
Common Mode	0
Recurring	0
Unusual	0

Table 4.12. Error leading to significant events in auxiliary system

Subsystem	Component	Type of Error	Failure Mode	
AS-LNS	Event	Administrative	Event	-
AS-RBVS	Switch	Operator	Carelessness	-

Effect on System	Effect on Component	Omission Commission	Radiation Release to Environment	Common Mode Recurring Unusual	ID. No.
F	-	C	No	U	59
D	N	O	No	U	64

Table 4.13. Error population in auxiliary system

	No. of Events
<u>Subsystem</u>	
AS-LNS	1
AS-RBVS	1
AS (other)	0
<u>Component</u>	
Switch	1
Event	1
<u>Type of Error</u>	
Administrative	1
Operator	1
Maintenance	0
<u>Failure Mode</u>	
Carelessness	1
Event	1
<u>Effect on Subsystem</u>	
Degraded	1
Failed	1
None	0
<u>Effect on Component</u>	
Degraded	0
Failed	0
None	1
<u>Omission/Commission</u>	
Omission	1
Commission	1

Table 4.13 (Continued)

	No. of Events
Radiation Release to Environment	
No Release	2
Release Within Limits	0
Release Exceeds Limits	0
Common Mode Recurring Unusual	
Common Mode	0
Recurring	0
Unusual	2

Table 4.14. Error leading to significant events in radioactive waste treatment system

Subsystem	Component	Type of Error	Failure Mode	
WPS-GH	Valve	Operator	Left Closed	-
WPS-GH	Rupture Disc	Administrative	Wrong Procedure Followed	-
WPS-GH	Rupture Disc	Administrative	Wrong Procedure Followed	-
WPS-LH	Pump	Administrative	Procedure Violation	-
WPS-LH	Monitor	Maintenance	Did not Monitor	-
WPS-LH	Monitor	Maintenance	Did not Monitor	-
WPS-LH	Event	Administrative	Calculation	-

Effect on System	Effect on Component	Omission Commission	Radiation Release to Environment	Common Mode Recurring Unusual	ID. No.
D	F	O	RWL	U	8
D	F	O	RWL		9
D	F	O	RWL	U	11
N	N	O	RWL	U	42
D	F	O	RWL	C	60
D	F	O	RWL	C	60
D	-	O	RWL		62

Table 4.15. Error population in radioactive waste treatment system

	No. of Events
Subsystem	
WPS-GH	3
WPS-LH	4
WPS (other)	0
Component	
Valve	1
Rupture Disc	2
Pump	1
Monitor	2
Event	1
Type of Error	
Administrative	4
Maintenance	2
Operator	1
Failure Mode	
Calculation	1
Did not Monitor	2
Left Closed	1
Procedure Violation	1
Wrong Procedure Followed	2
Effect on Subsystem	
Degraded	6
Failed	0
None	1

Table 4.15 (Continued)

	No. of Events
Effect on Component	
Degraded	0
Failed	5
None	1
Omission/Commission	
Omission	7
Commission	0
Radiation Release to Environment	
No Release	0
Release Within Limits	7
Release Exceeds Limits	0
Common Mode Recurring Unusual	
Common Mode	2
Recurring	0
Unusual	3

Table 4.16. Error leading to significant events in instrumentation and monitoring system

Subsystem	Component	Type of Error	Failure Mode	
IMS	Detector	Administrative	Procedure Violation	-
IMS	Detector	Maintenance	Improper Handling	-

Effect on System	Effect on Component	Omission Commission	Radiation Release to Environment	Common Mode Recurring Unusual	ID. No.
N	N	O	No		45
D	F	O	No		54

Table 4.17. Error population in instrumentation and monitoring system

	No. of Events
Subsystem	
IMS	2
Component	
Detector	2
Type of Error	
Administrative	1
Maintenance	1
Operator	0
Failure Mode	
Improper Handling	1
Procedure Violation	1
Effect on Subsystem	
Degraded	1
Failed	0
None	1
Effect on Component	
Degraded	0
Failed	1
None	1
Omission/Commission	
Omission	2
Commission	0

Table 4.17 (Continued)

	No. of Events
Radiation Release to Environment	
No Release	2
Release Within Limits	0
Release Exceeds Limits	0
Common Mode Recurring Unusual	
Common Mode	0
Recurring	0
Unusual	0

Table 4.18. Error leading to significant events in other system

Subsystem	Component	Type of Error	Failure Mode	
Cooling Pond	Event	Administrative	Procedural Deficiency	-
Support Structure	Snubber (12)	Maintenance	Did not check	-

Effect on System	Effect on Component	Omission Commission	Radiation Release to Environment	Common Mode Recurring Unusual	ID. No.
N	N	O	No		65
N	F	O	No	C	66

Table 4.19. Error population in other system

	No. of Events
Subsystem	
Cooling Pond	1
Support Structure	1
Component	
Snubber	12
Event	1
Type of Error	
Administrative	1
Maintenance	1
Operator	0
Failure Mode	
Did not Check	1
Procedural Deficiency	1
Effect on Subsystem	
Degraded	0
Failed	0
None	2
Effect on Component	
Degraded	0
Failed	1
None	1
Omission/Commission	
Omission	2
Commission	0

Table 4.19 (Continued)

	No. of Events
Radiation Release to Environment	
No Release	2
Release Within Limits	0
Release Exceeds Limits	0
Common Mode Recurring Unusual	
Common Mode	1
Recurring	0
Unusual	0

V. DATA ANALYSIS

A. Failure Significance

The data show a cumulative total of 190 errors occurring during the 43-month period reviewed. This review of the LER records shows that about 38.4% of the causes are human errors which are related to administration, maintenance, or operator. Maintenance, administrative, and operator errors represent 41.1%, 34.2%, 24.7% of all human errors respectively. The systems most frequently involved in human errors are the main reactor coolant system, auxiliary electric power system, reactor protection system, and radioactive waste treatment system. About 38.4%, 23.3%, 16.4%, and 9.6% of the total human errors are related to main reactor coolant system, auxiliary electric power system, reactor protection system, and radioactive waste treatment system respectively. Also, 41.1% of the total human errors caused part of the systems to fail, and 35.6% of the total human errors caused to degrade part of the systems. The components most frequently involved in human errors are valves, switches, pumps, and control rods. About 15.1%, 13.7%, 8.2%, and 6.8% of the total human errors related to valves, switches, pumps, and control rods respectively. Also, 55.4% of the total human errors caused components to fail, and 13.8% of the total human errors caused to degrade components. The effect on the systems and components indicates the significance of the human errors because of its great importance in safety and reliability analysis. Such errors cause delay in operation, increase in down time and reduction in plant availability factor. Improvement in human errors would require a careful study of the frequency of the failure modes.

The failure modes most frequently resulted in human errors are procedural deficiency, improper handling, did not test/check, and improper setting. About 17.8%, 13.7%, 11%, and 6.8% of the total human errors related to procedural deficiency, improper handling, did not test/check, and improper setting respectively. Any faulty action resulted in human errors is described by two categories, omission, and commission. Also, 80.6% of the total human errors related to omission, and 19.4% of the total human errors related to commission.

Common mode, reoccurring, or unusual events of human errors have a direct effect on safety and reliability, especially in estimating the reliability of redundant systems. About 15.5%, 14.1%, and 8.5% of total human errors related to unusual, common mode, and reoccurring events respectively. Only 9.6% out of the total human errors resulted in some form of radiation release to environment within limits.

B. Method of Analysis

The failure (error) rate per month for specific task, $\hat{\lambda}$, is defined as

$$\hat{\lambda} = \frac{n}{T} \text{ per month} \tag{1}$$

where

n = failure (error) count

T = reactor-month surveyed.

The 90% confidence bounds on λ are

$$\hat{\lambda}\,\frac{X^2_{0.05,2n}}{2n} \leq \lambda \leq \hat{\lambda}\,\frac{X^2_{0.095,2(n+1)}}{2n} \tag{2}$$

where $X^2_{\alpha,r}$ is the α-percentile of the chi-square distribution with r degrees of freedom.

Human failure (error) rates in the operation of the Fort St.Vrain, high temperature gas cooled reactor (HTGR) during the time period from May 30, 1974 through December 30, 1977 are calculated and the results are given in Table 5.1. Table 5.1 presents the actual number of errors, n, committed during the 43-month period, the estimated error rate, $\hat{\lambda}$, in errors/month, and 90% confidence bounds on λ based on $\hat{\lambda}$ and n, also in errors/month. In Table 5.1, number of errors, error rates, and 90% confidence bounds are given for the following,

1. for each system involved
2. for each component involved

3. for each type of error (administrative, maintenance, operator)
4. for each failure mode
5. for each type of effect on systems and components (failed, degraded, none)
6. for each mode of human error (omission/commission, common mode/reoccurring/unusual).

Figures 5.1-5.22 show the distribution of human error over the 43-month period of the plant age for the most frequent systems, components, types of error, types of effect on systems and components, and modes of human error involved (27-29).

Table 5.1. Number of error, error rate, and 90% confidence

	No. of Events (n)	Failure Error Rate (λ)	90% Confidence	
FOR ALL PLANT				
System Involved				
Auxiliary Electric Power System	17	.3953	.2519	.5929
Reactor Protection System	12	.2791	.1610	.4522
Emergency Cooling System	1	.0233	.0012	.1103
Main Reactor Coolant System	28	.6512	.4628	.7578
Prestressed Concrete Reactor Vessel System	2	.0465	.0083	.1464
Auxiliary System	2	.0465	.0083	.1464
Radioactive Waste Treatment System	7	.1628	.0764	.3058

Table 5.1 (Continued)

	No. of Events (n)	Failure Error Rate (λ)	90% Confidence	
Instrumentation and Monitoring System	2	.0465	.0083	.1464
Other System	2	.0465	.0083	.1464
Component				
Switch	10	.2326	.1262	.3945
Relay	2	.0465	.0083	.1464
Engine	4	.0930	.0318	.2129
Cable	1	.0233	.0012	.1103
Breaker	4	.0930	.0318	.2129
Clutch	1	.0233	.0012	.1103
Tube	2	.0465	.0083	.1464
Charger	1	.0233	.0012	.1103
Control Rod	5	.1163	.0458	.2445
Connector	2	.0465	.0083	.1464
Pump	6	.1395	.0608	.2754
Valve	11	.2558	.1435	.4234
Event	10	.2326	.1262	.3945
Seal	3	.0698	.0190	.1803
Bearing	1	.0233	.0012	.1103
Transmitter	1	.0233	.0012	.1103
Fitter	1	.0233	.0012	.1103
Circuit	1	.0233	.0012	.1103
Rupture Disc	2	.0465	.0083	.1464
Monitor	2	.0465	.0083	.1464
Detector	2	.0465	.0083	.1464
Snubber (one time)	12	.2791	.1610	.4522

Table 5.1 (Continued)

	No. of Events (n)	Failure Error Rate (λ)	90% Confidence	
Type of Error				
Administrative	25	.5814	.4042	.8120
Maintenance	30	.6977	.5022	.9463
Operator	18	.4186	.2706	.6207
Failure Mode				
Carelessness	4	.0930	.0318	.2129
Communication	2	.0465	.0083	.1464
Connection	1	.0233	.0012	.1103
Did not Test/Check	8	.1860	.0926	.3357
Improper Handling	10	.2326	.1262	.3945
Improper Setting	5	.1163	.0458	.2445
Misadjustment	2	.0465	.0083	.1464
Procedural Deficiency	13	.3023	.1788	.4807
Did not Connect	2	.0465	.0083	.1464
Improper Operation	1	.0233	.0012	.1103
Inadvertent Actuation	2	.0465	.0083	.1464
Installation	1	.0233	.0012	.1103
Calibration	1	.0233	.0012	.1103
Deenergized	1	.0233	.0012	.1103
Did not Follow Procedure	4	.0930	.0318	.2129
Event	2	.0465	.0083	.1464
Improper Permission	1	.0233	.0012	.1103
Improper Actuation	1	.0233	.0012	.1103
Lack of Repair	1	.0233	.0012	.1103
Left Open	1	.0233	.0012	.1103
Misposition	1	.0233	.0012	.1103
Ruptured	1	.0233	.0012	.1103
Calculation	1	.0233	.0012	.1103

Table 5.1 (Continued)

	No. of Events (n)	Failure Error Rate (λ)	90% Confidence	
Did not Monitor	2	.0465	.0083	.1464
Left Closed	1	.0233	.0012	.1103
Procedure Violcation	2	.0465	.0083	.1464
Wrong Procedure Followed	2	.0465	.0083	.1464
Effect on Part of the System				
Degraded	26	.6047	.4237	.8390
Failed	30	.6977	.5022	.9463
None	17	.3953	.2519	.5929
Effect on Component				
Degraded	9	.2093	.1092	.3652
Failed	36	.8372	.6221	1.1058
None	20	.4651	.3082	.6759
Omission/Commission				
Omission	58	1.3488	1.0733	1.6756
Commission	14	.3256	.1968	.5090
Radiation Release to Environment				
No Release	66	1.5349	1.2407	1.8802
Release Within Limits	7	.1628	.0764	.3058
Release Exceeds Limits	0	.0000	.0000	.0000
Common Mode/Reoccurring/Unusual				
Common Mode	10	.2326	.1262	.3945
Reoccurring	6	.1395	.0608	.2754
Unusual	11	.2558	.1435	.4234

Table 5.2. Error committed during 43-month period for all systems of the plant

System	Reactor Months																					
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22
MRCS	1	0	0	0	0	1	0	0	1	1	0	0	0	0	0	0	0	1	0	1	0	0
EPS	0	0	1	0	1	0	0	0	1	0	0	1	1	0	0	2	0	1	1	0	0	0
RPS	3	0	0	1	0	0	0	0	0	0	0	0	0	0	0	2	0	0	0	0	0	5
WPS	0	0	0	0	0	0	2	0	1	0	0	0	0	0	0	0	0	0	0	0	0	0
PCRVS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	0
AS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
IMS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
OS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
ECS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

Reactor Months																					Total of Events
23	24	25	26	27	28	29	30	31	32	33	34	35	36	37	38	39	40	41	42	43	
1	2	6	2	0	0	1	0	0	2	3	0	0	0	0	1	0	0	0	1	3	28
1	0	0	0	0	0	1	0	0	1	2	1	0	0	0	0	0	1	0	0	1	17
0	0	0	0	0	0	0	0	0	1	0	0	0	0	0	0	0	0	0	0	0	12
0	0	0	0	0	1	0	0	0	0	0	0	0	0	0	3	0	0	0	0	0	7
0	0	0	0	0	0	0	0	0	0	1	0	0	0	0	0	0	0	0	0	0	2
0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	0	0	0	1	0	0	2
0	0	0	0	0	0	0	1	0	0	1	0	0	0	0	0	0	0	0	0	0	2
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2	0	0	2
0	0	0	0	0	0	0	0	0	0	1	0	0	0	0	0	0	0	0	0	0	1

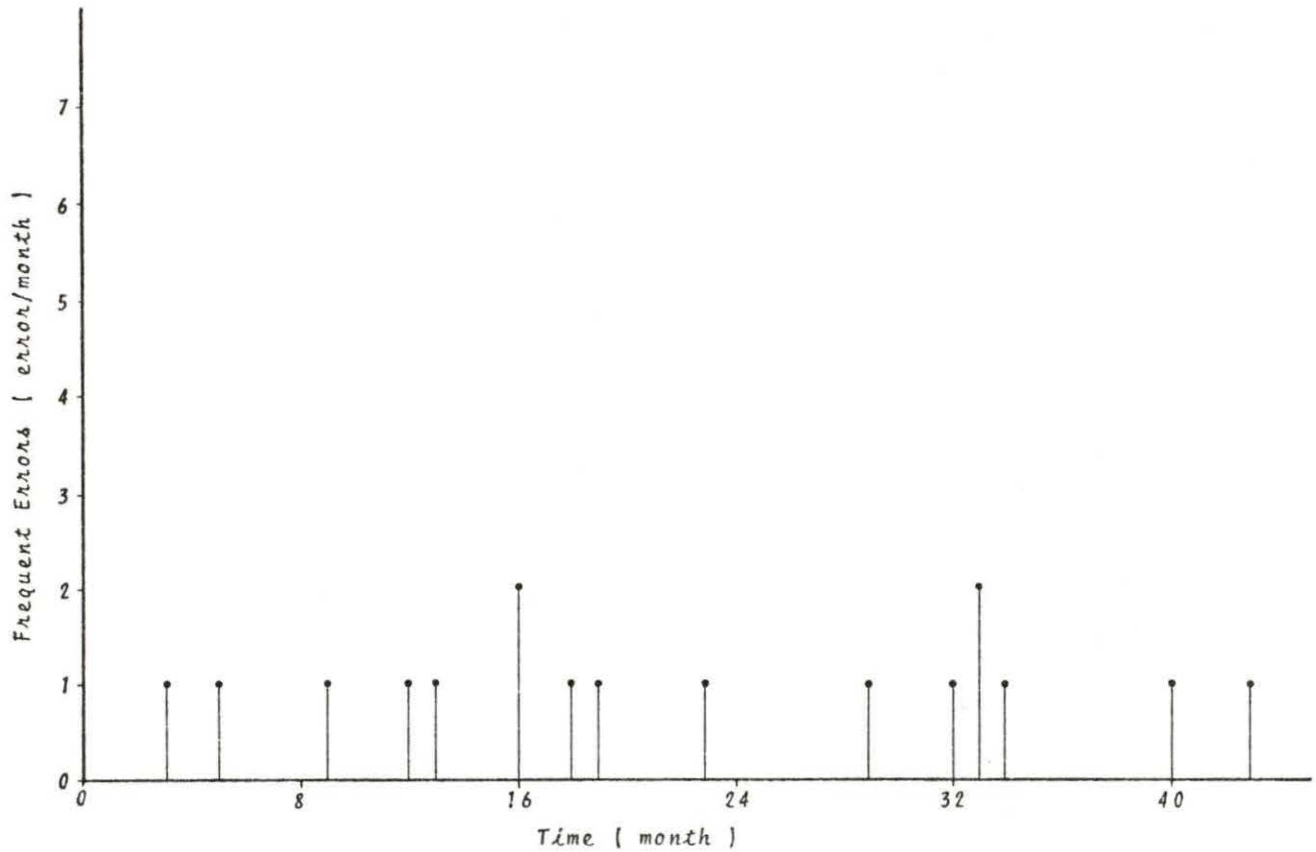

Figure 5.1. Distribution of human error for auxiliary electric power system

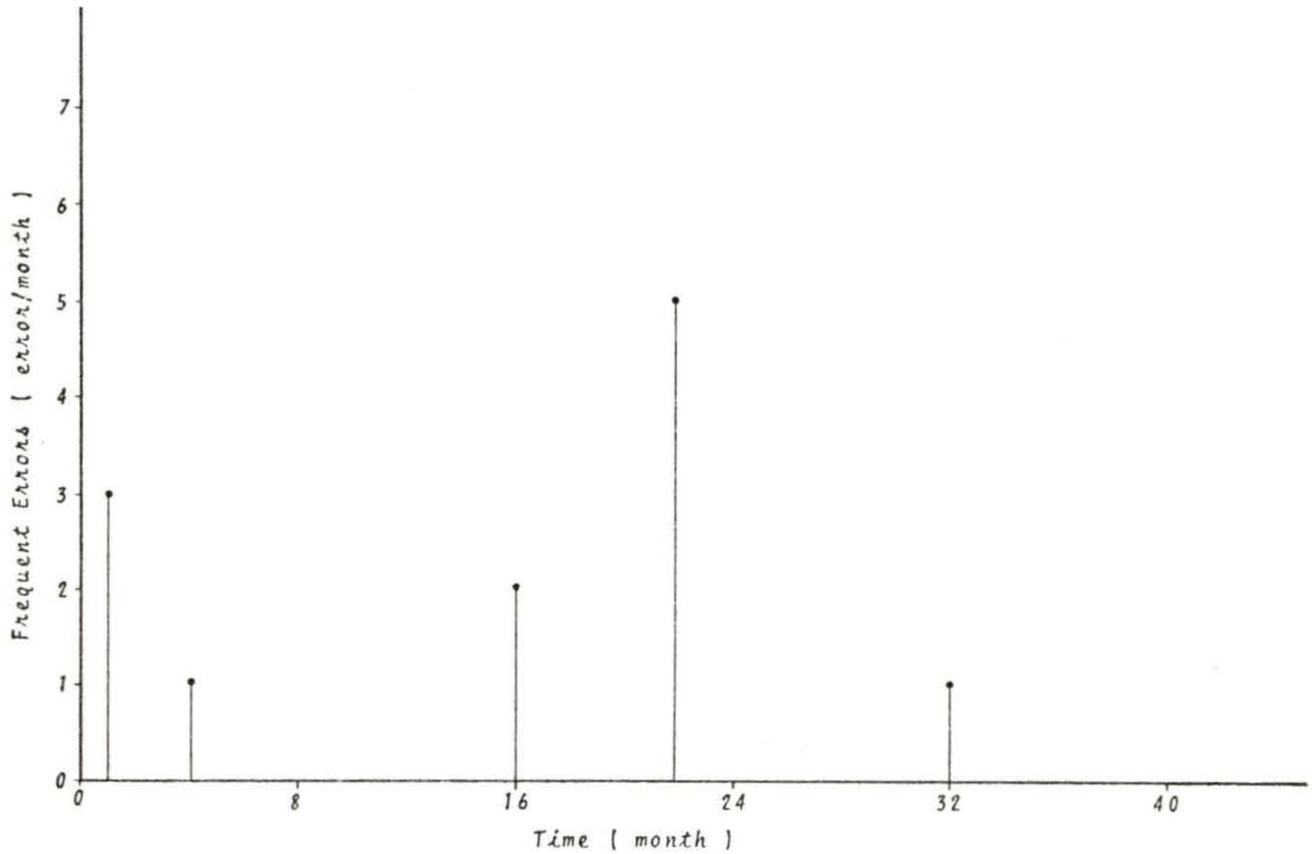

Figure 5.2. Distribution of human error for reactor protection system

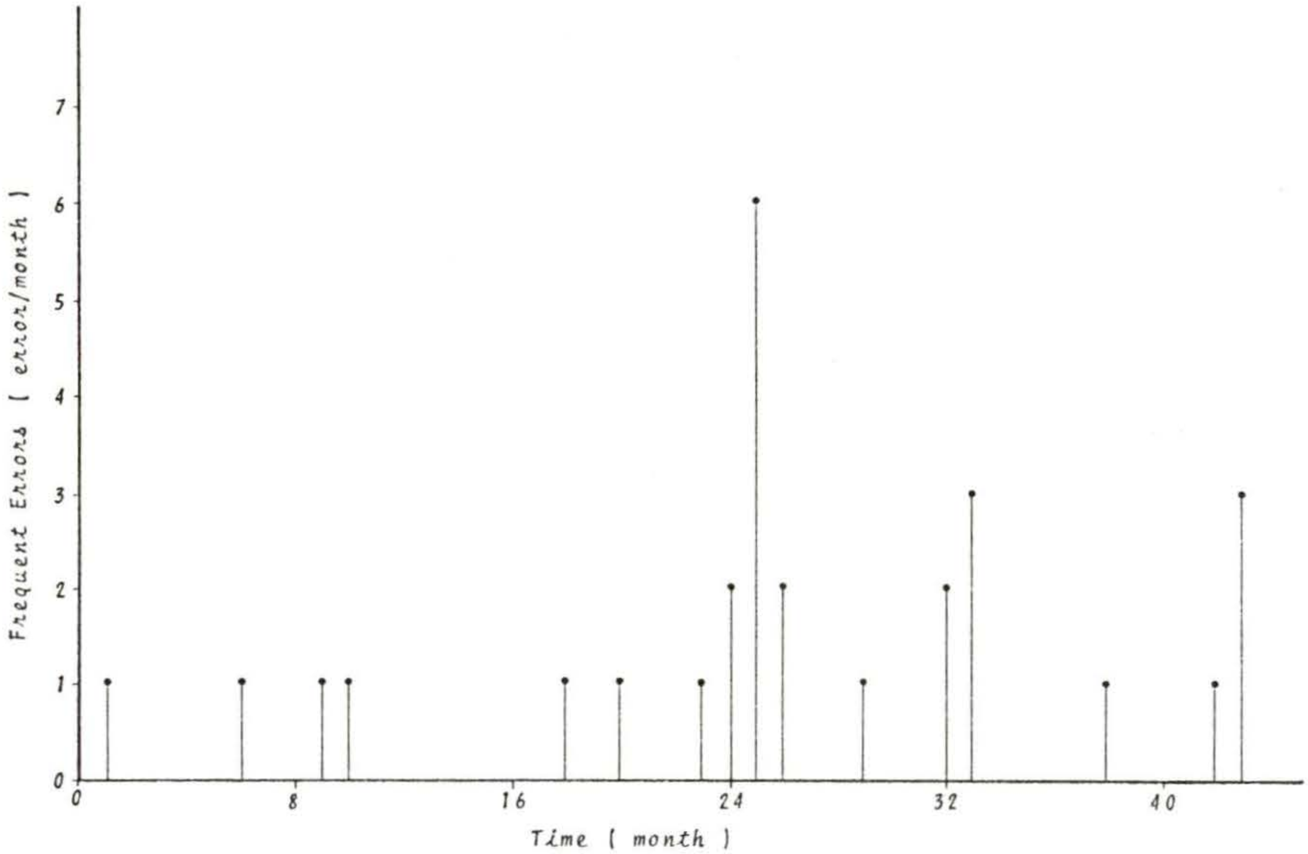

Figure 5.3. Distribution of human error for main reactor coolant system

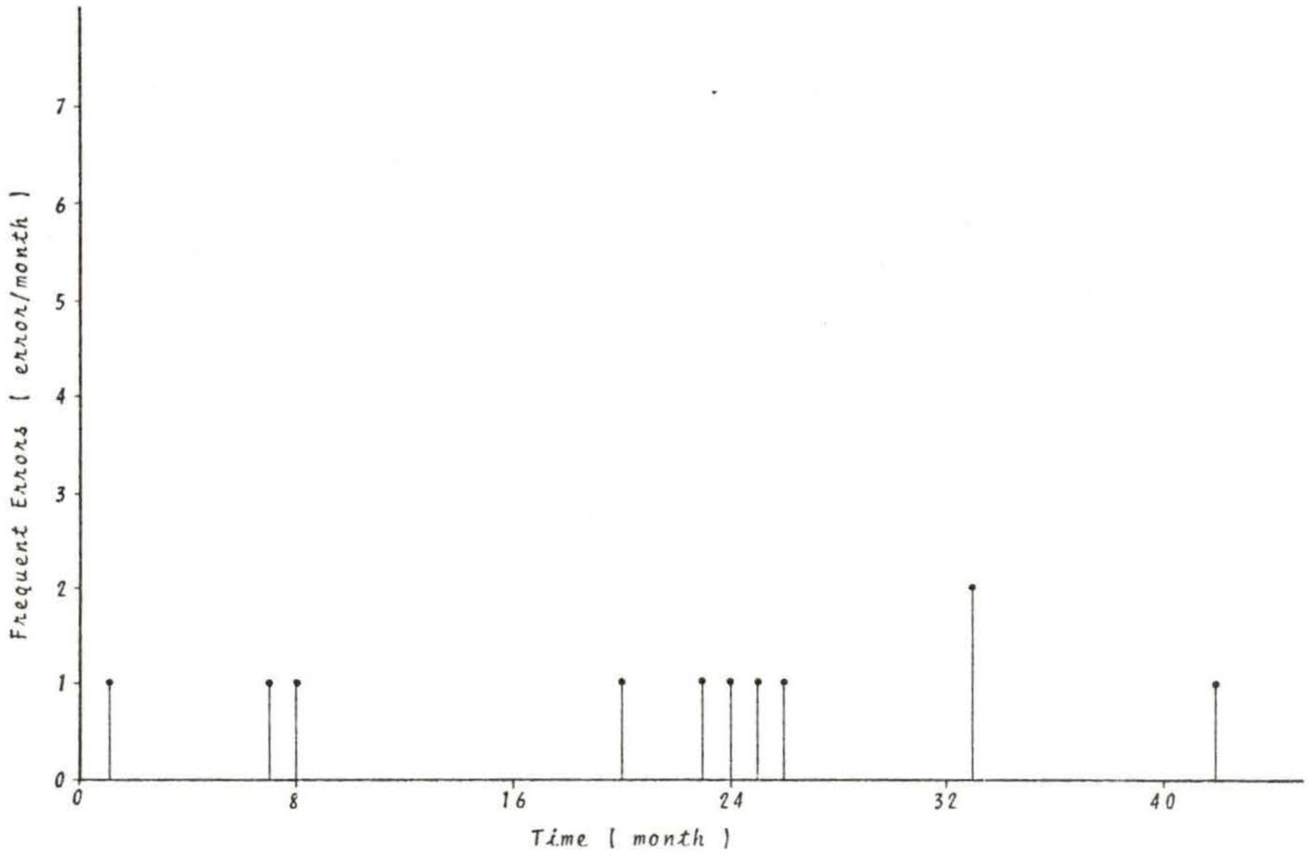

Figure 5.4. Distribution of human error for valve (component) involved

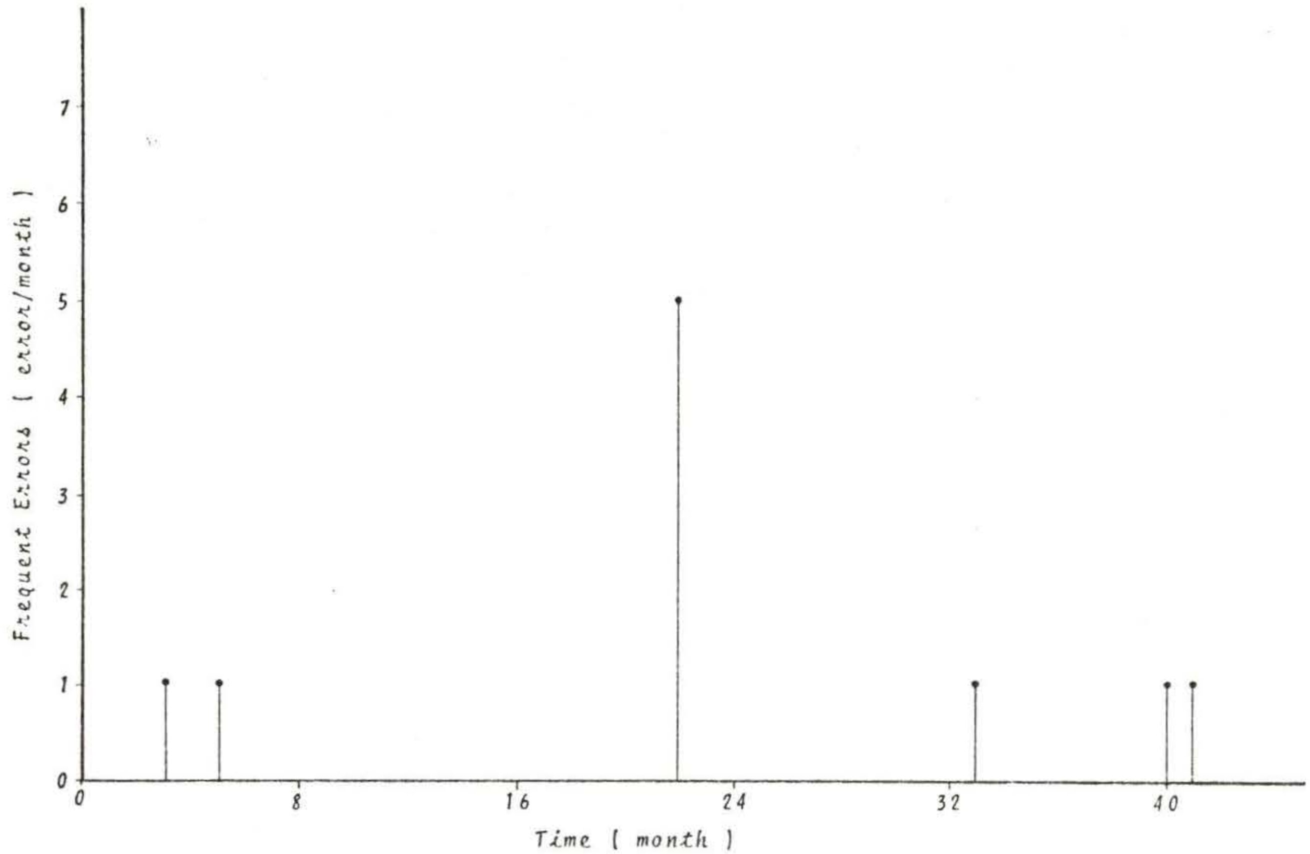

Figure 5.5. Distribution of human error for switch (component) involved

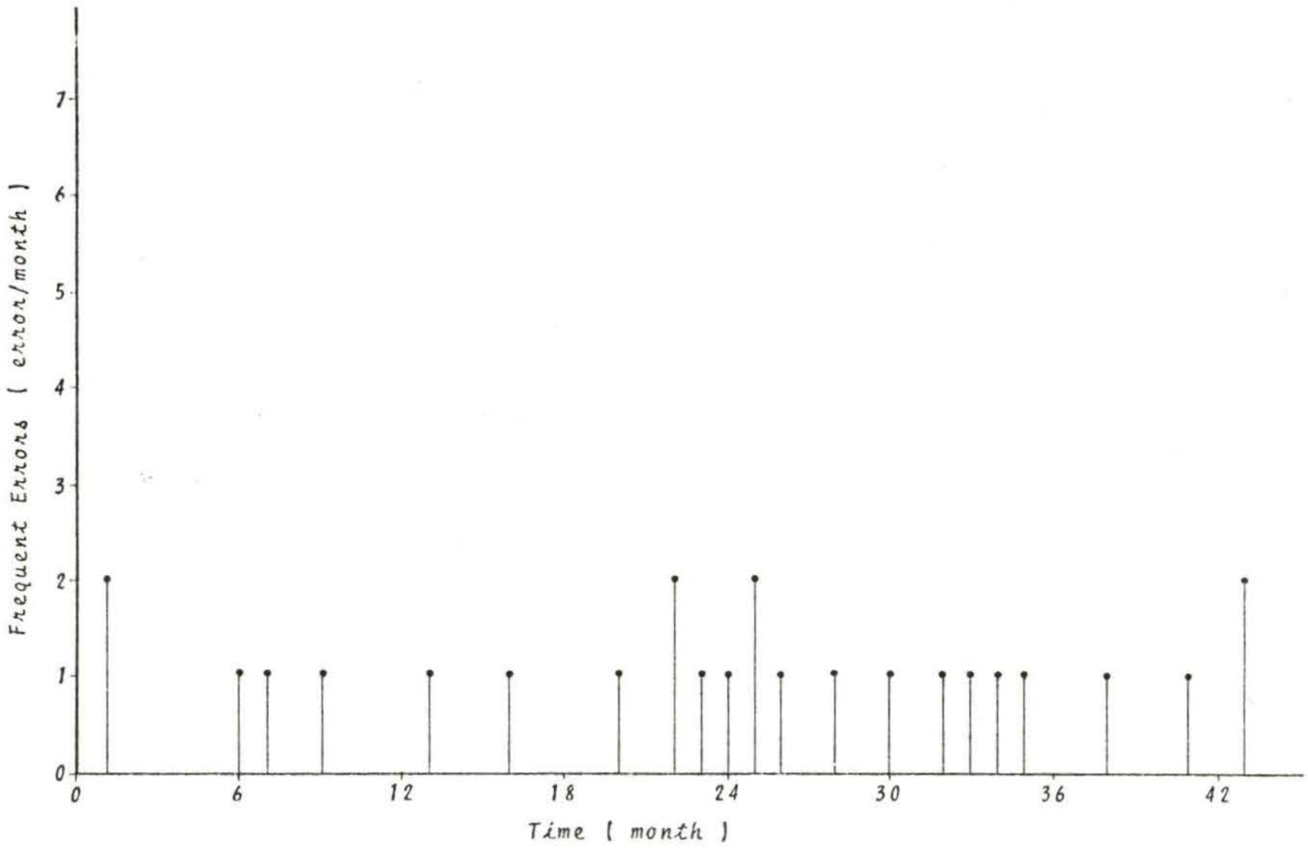

Figure 5.6. Distribution of human error for type of error (administrative) involved

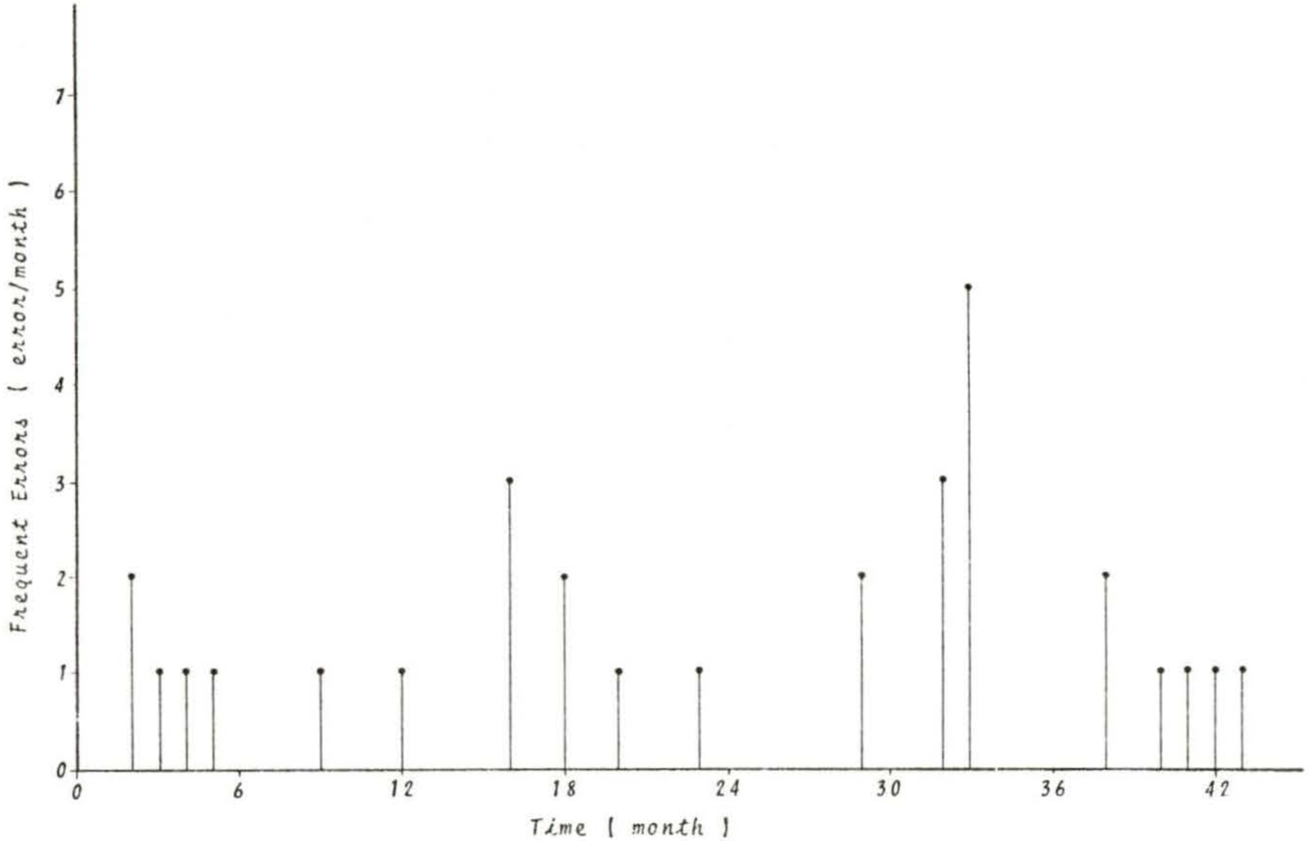

Figure 5.7. Distribution of human error for type of error (maintenance) involved

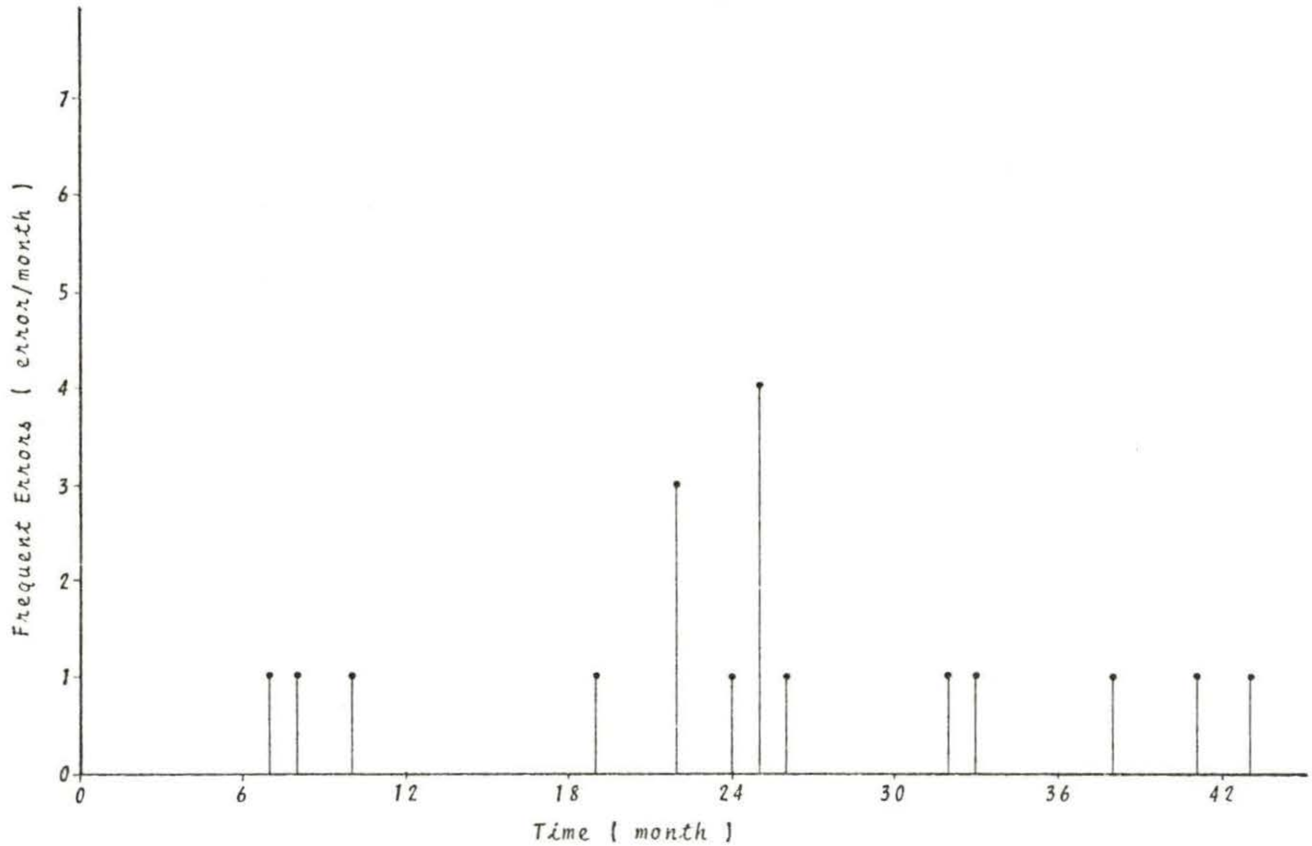

Figure 5.8. Distribution of human error for type of error (operator) involved

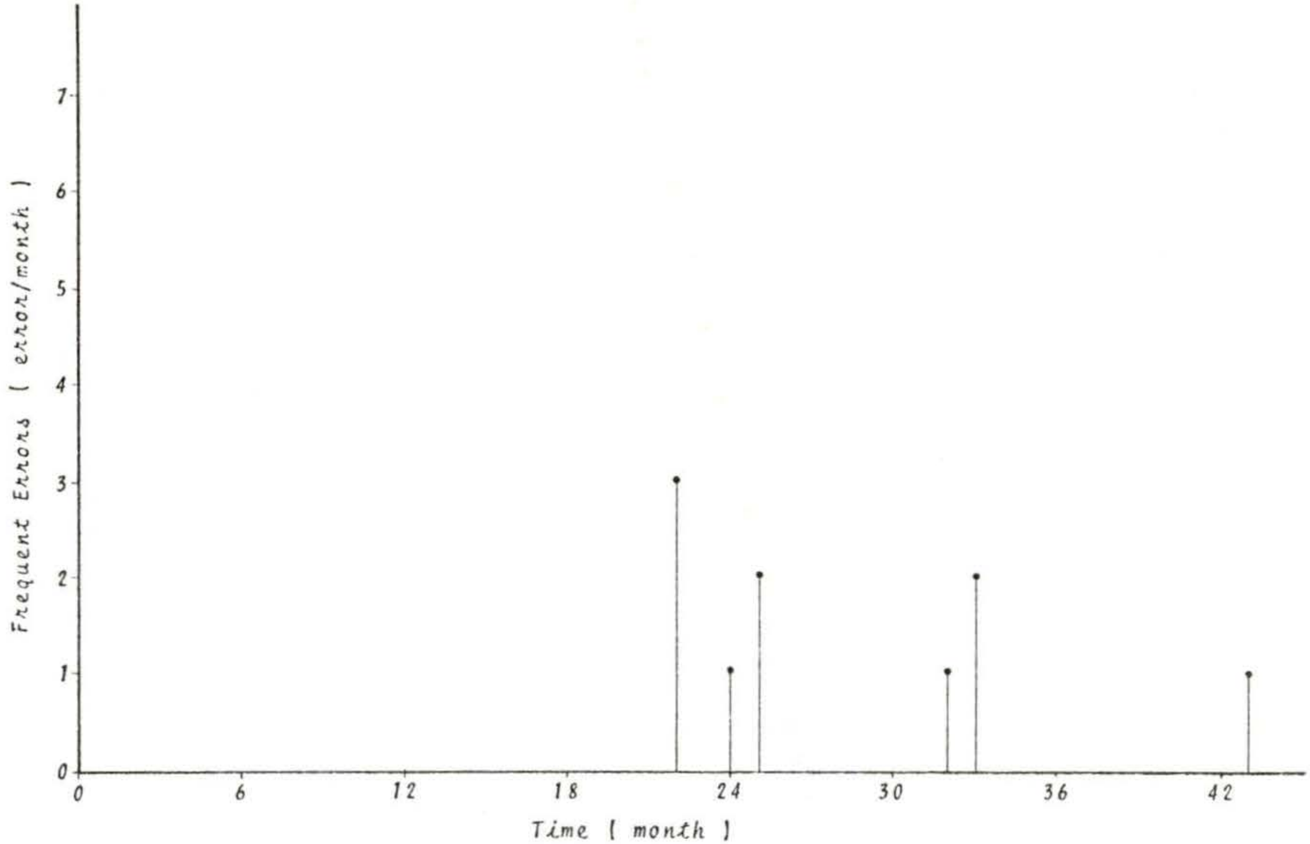

Figure 5.9. Distribution of human error for failure mode (improper handling) committed

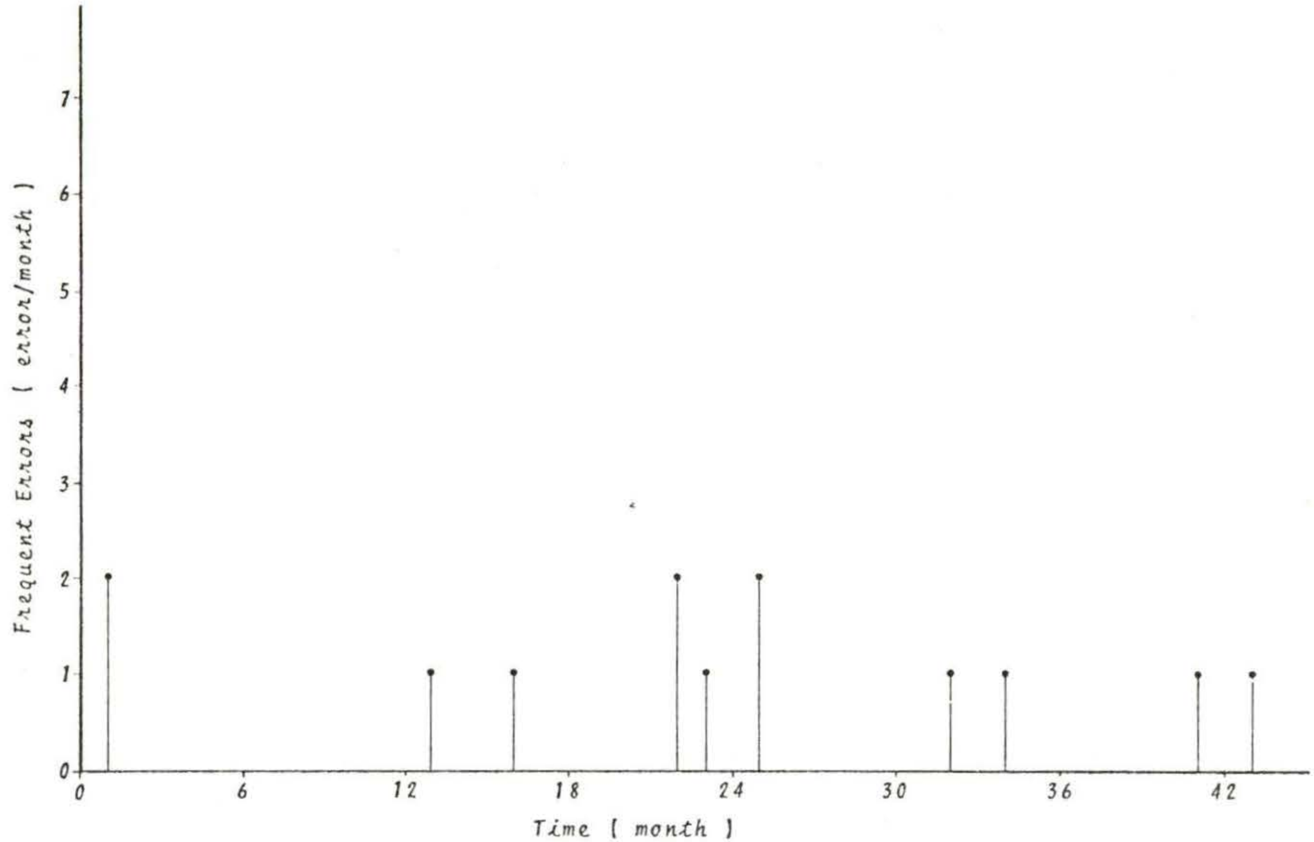

Figure 5.10. Distribution of human error for failure mode (procedural deficiency) involved

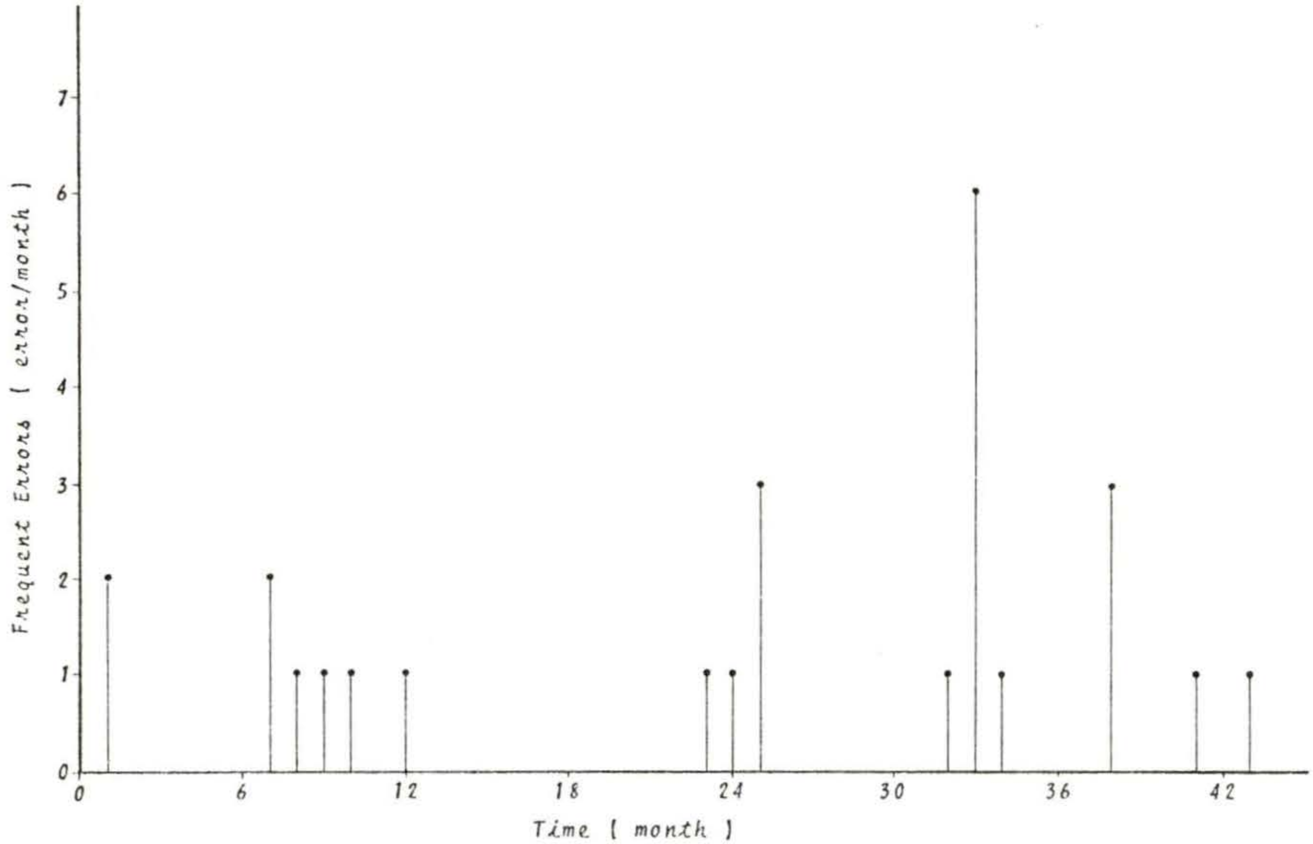

Figure 5.11. Distribution of human error for type of effect (degraded) of the error on the involved system

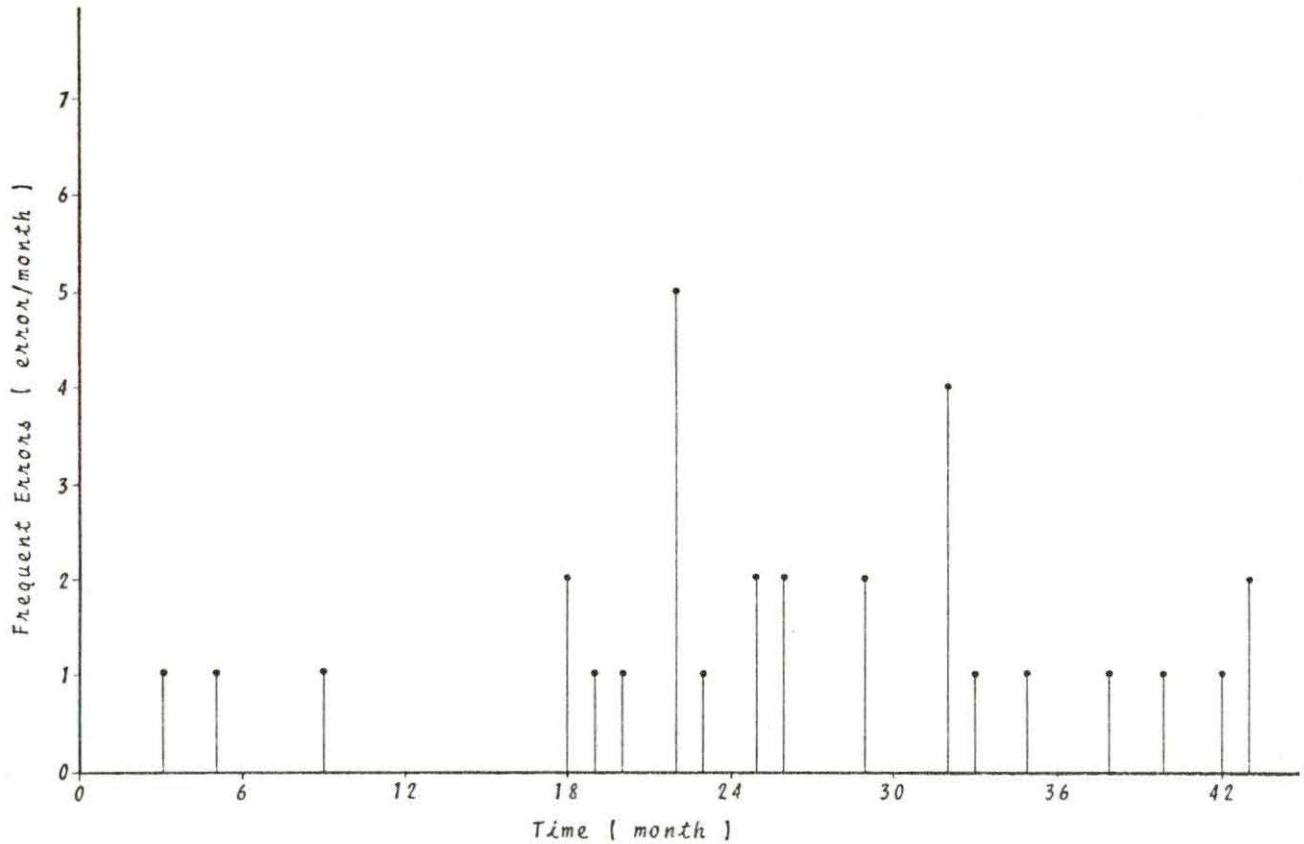

Figure 5.12. Distribution of human error for type of effect (failed) of the error on the involved system

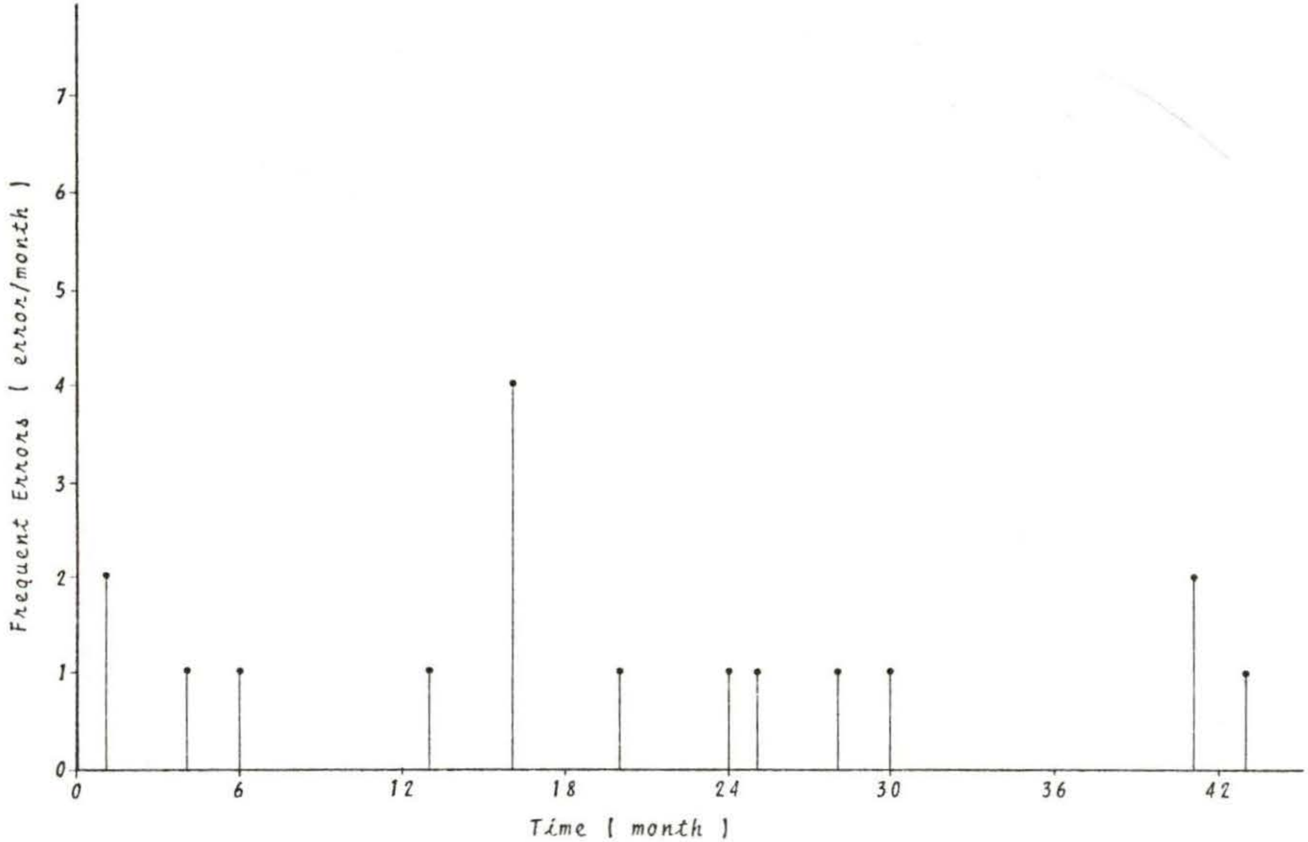

Figure 5.13. Distribution of human error for type of effect (none) of the error on the involved system

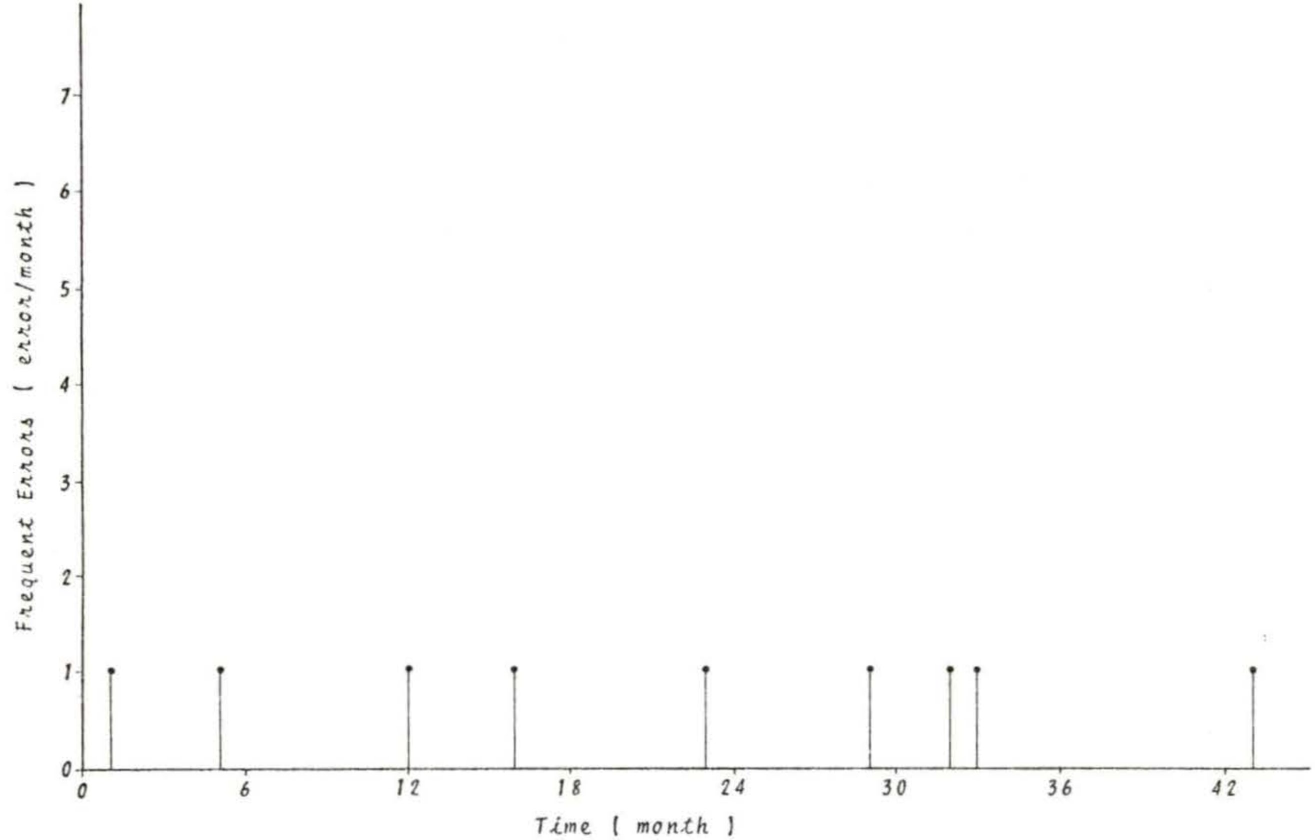

Figure 5.14. Distribution of human error for type of effect (degraded) of the error on the involved component

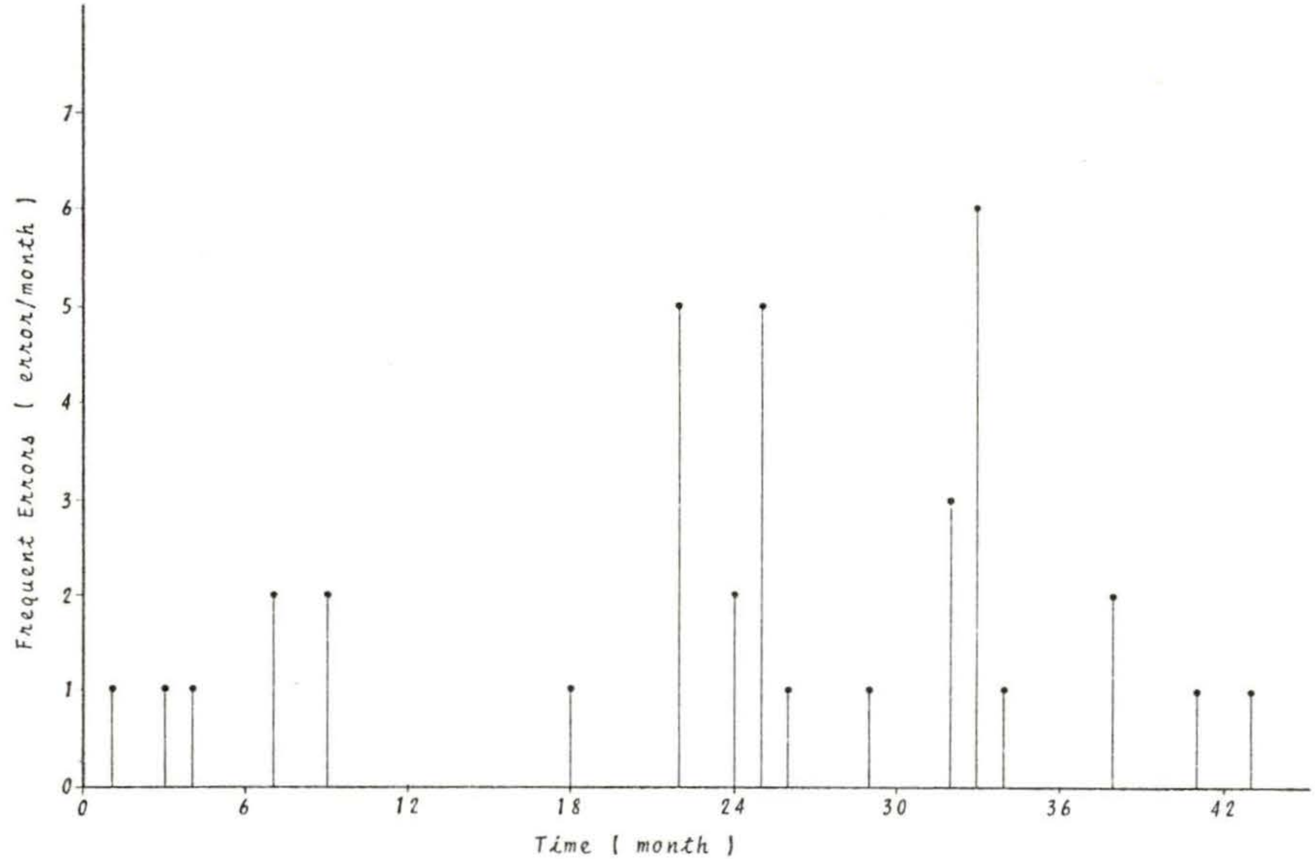

Figure 5.15. Distribution of human error for type of effect (failed) of the error on the involved component

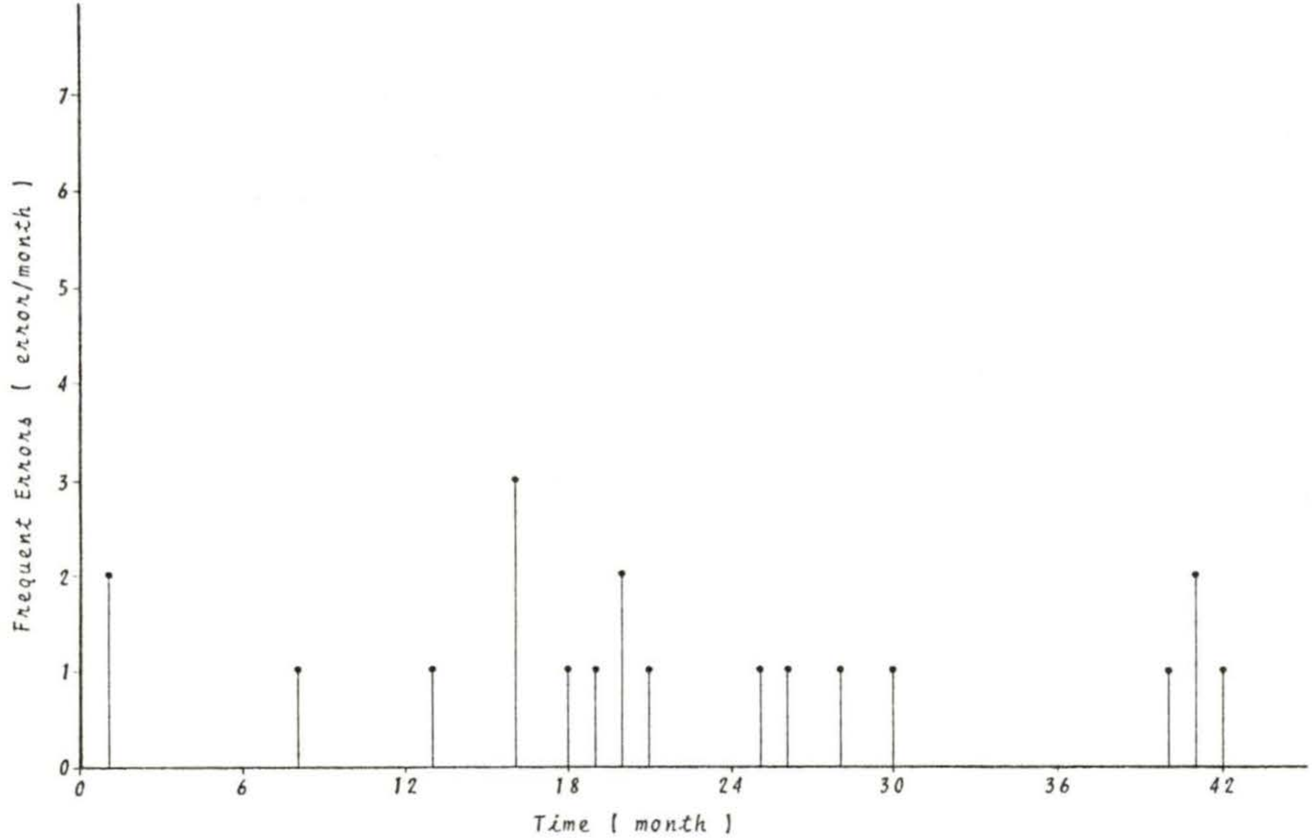

Figure 5.16. Distribution of human error for type of effect (none) of the error on the involved component

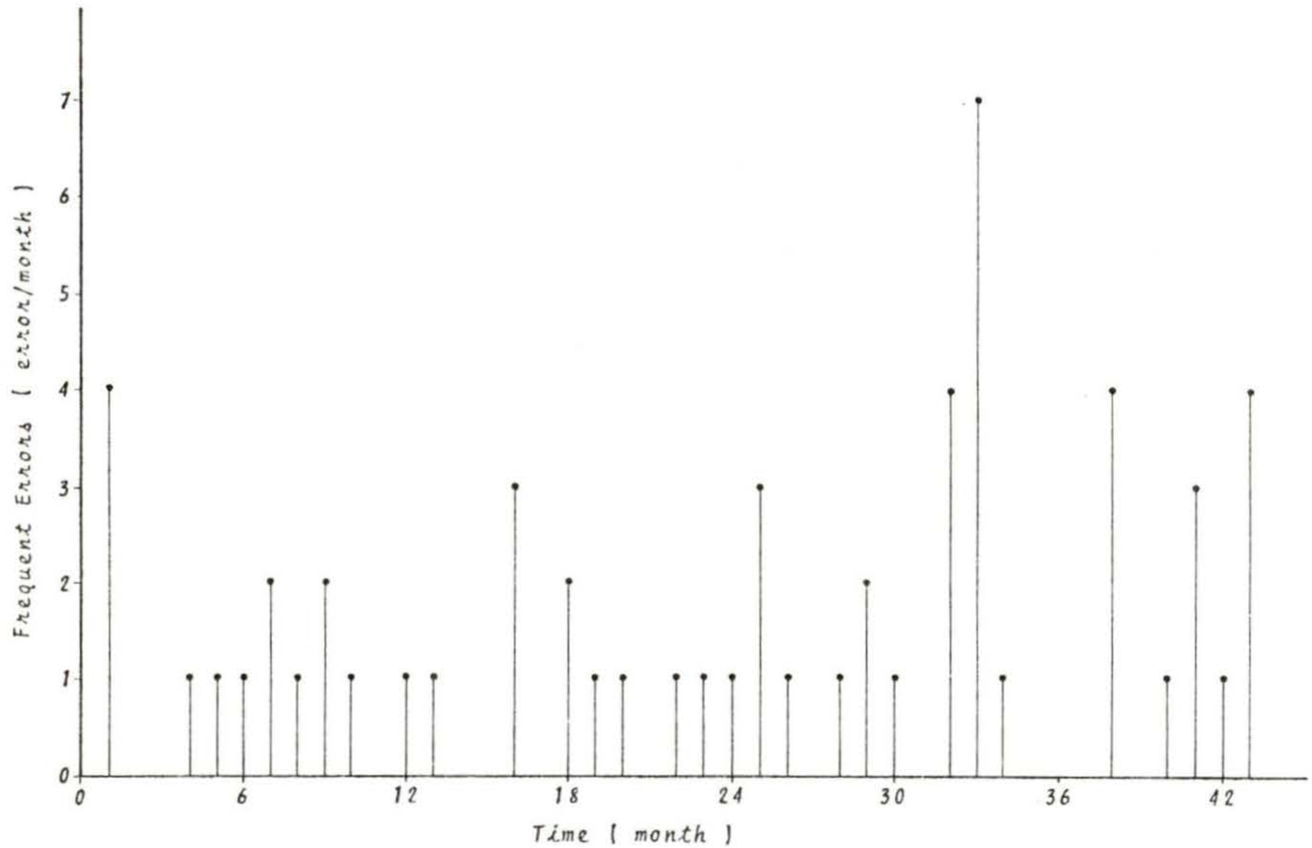

Figure 5.17. Distribution of human error for omission event

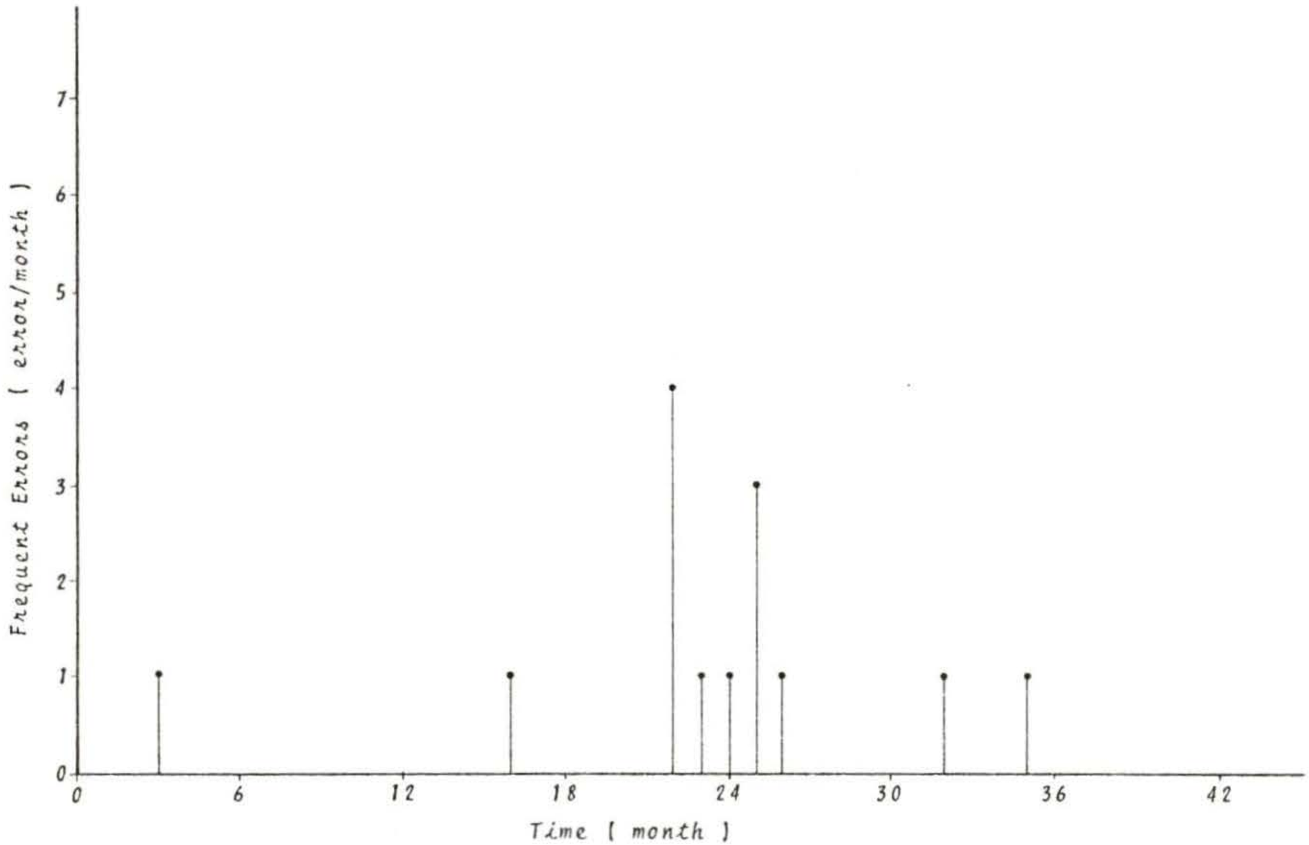

Figure 5.18. Distribution of human error for commission event

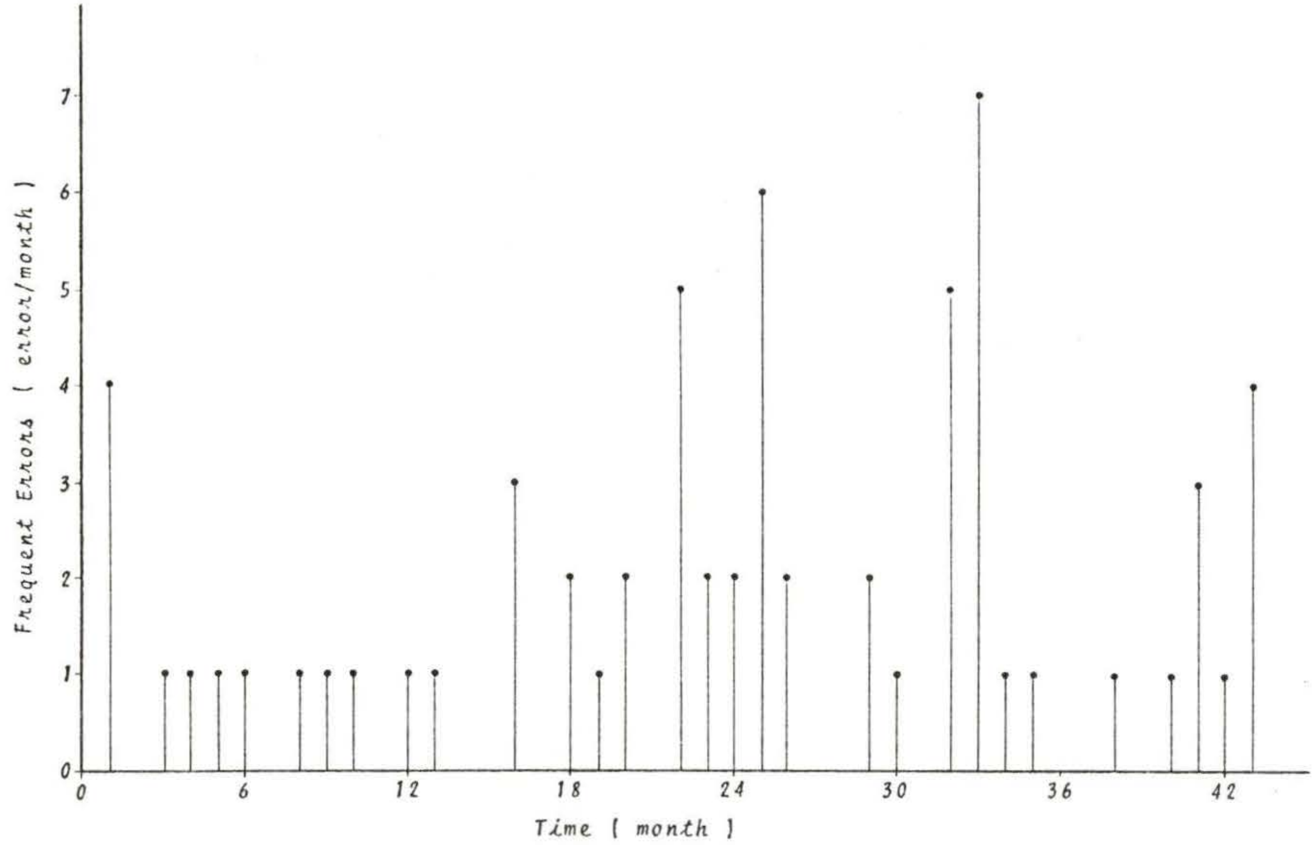

Figure 5.19. Distribution of human error for no radiation release to environment

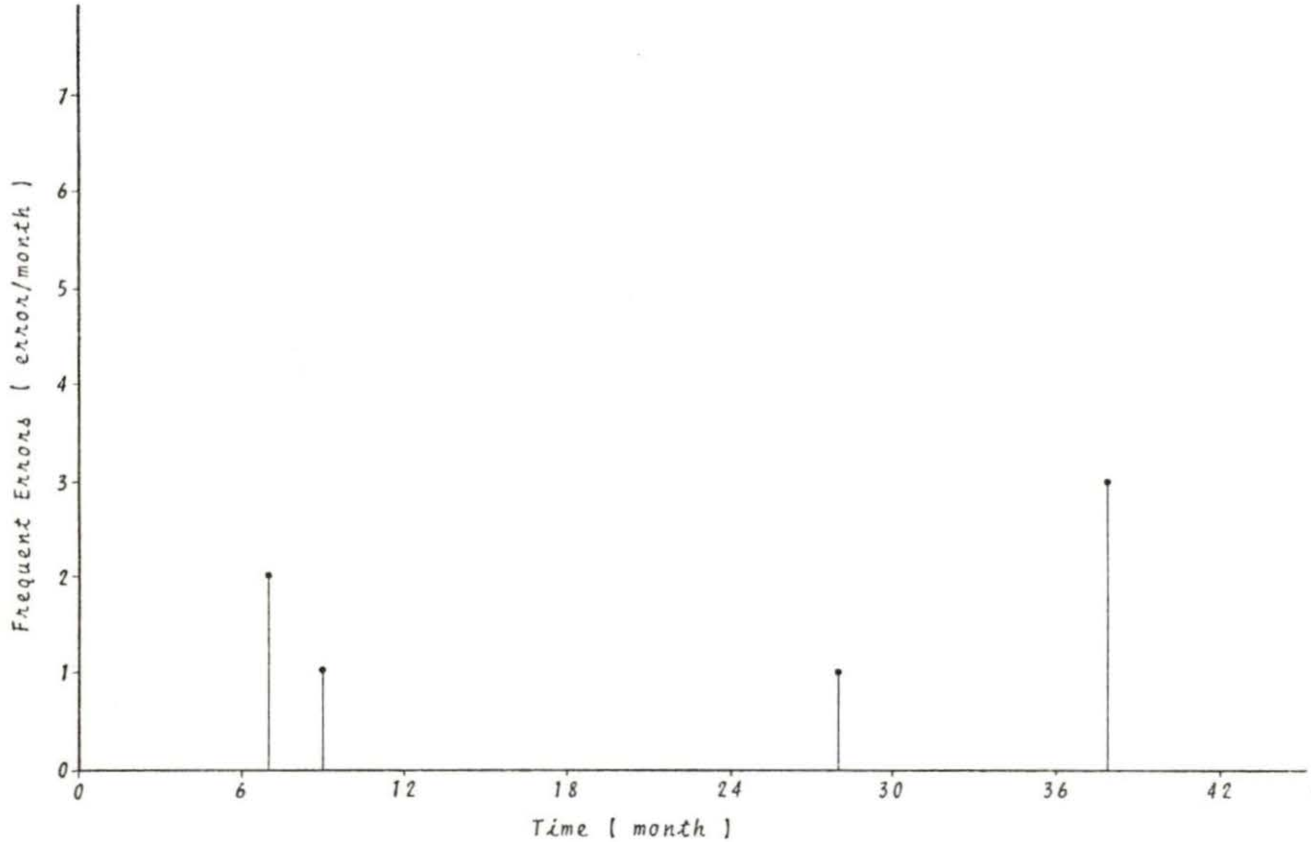

Figure 5.20. Distribution of human error for radiation release within limits to environment

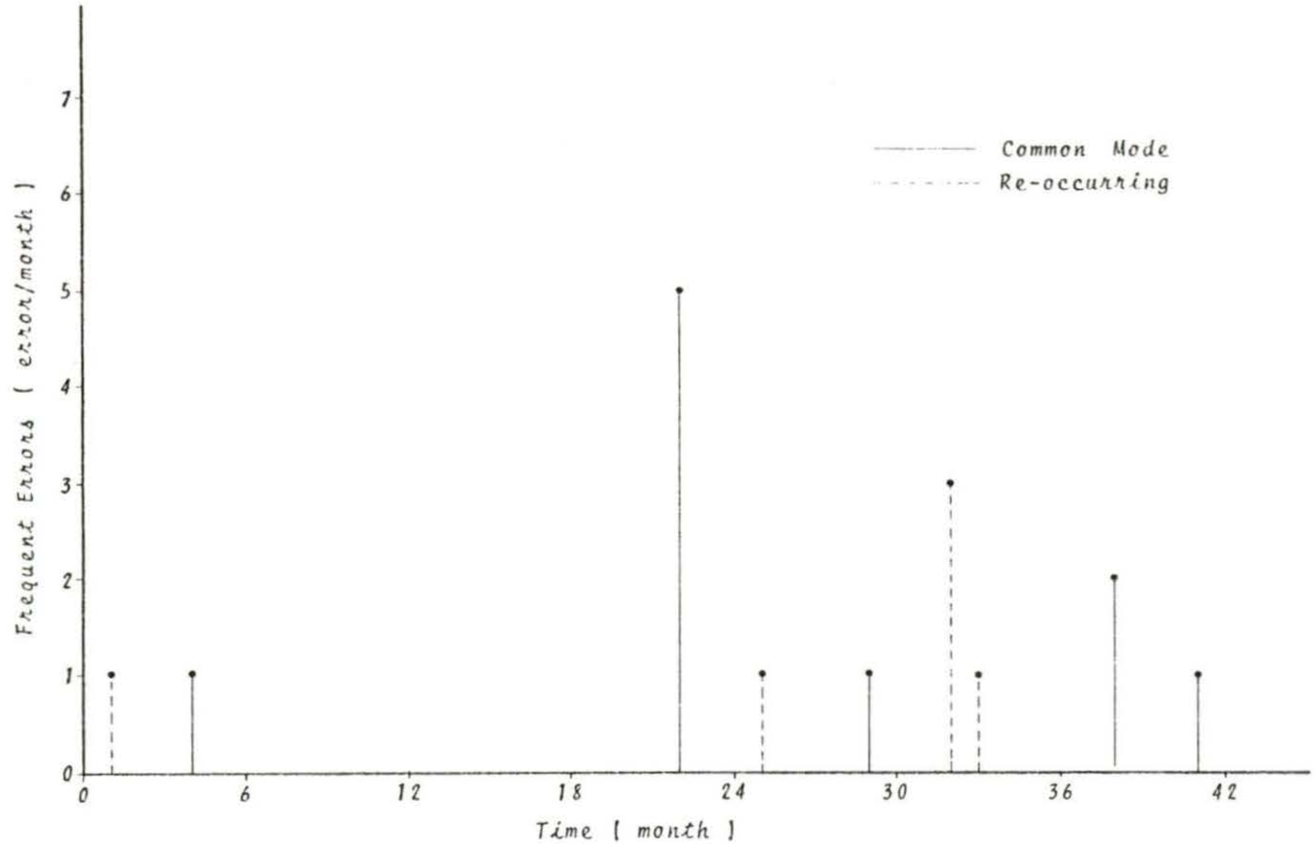

Figure 5.21. Distribution of human error for common mode and reoccurring event

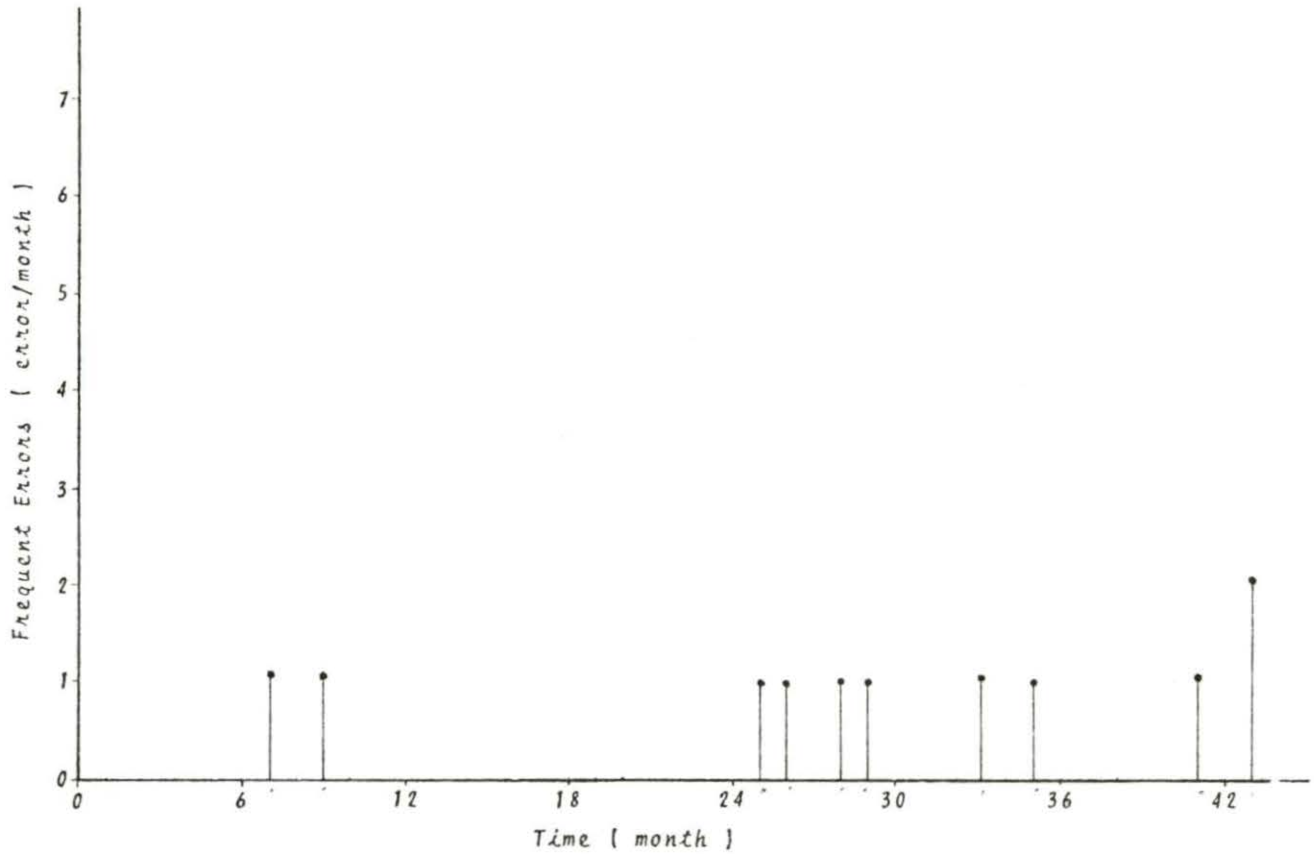

Figure 5.22. Distribution of human error for unusual event

C. Analysis of Human Error Distribution Curves by Pattern Recognition Principles

A simple and intuitive approach to the general analysis of human events in a nuclear power plant is to utilize the concept of pattern classification by distance functions. The representation of input data is by arranging measurements in the form of a measurement or pattern vector:

$$\underline{X} = \begin{matrix} x_1 \\ x_2 \\ \vdots \\ x_n \end{matrix} \tag{3}$$

The most obvious way of establishing a measure of similarity between pattern vectors, which we consider as points in Euclideon space, is by determining their proximity. The Euclideon distance between an arbitrary pattern vector $\underline{X}$ and the ith prototype is given by

$$D_i = \underline{X} - \underline{Z_i} = (\underline{X} - \underline{Z_i})^t(\underline{X} - \underline{Z_i}) \tag{4}$$

where $\underline{Z_i}$ is the cluster center (mean vector), and 't' is the transpose of a matrix.

The minimum-distance classifier computes the distance from a pattern $\underline{X}$ of unknown classification to the prototype (cluster center) of each class, and assigns the pattern to the class to which it is closest. In other words, $\underline{X}$ is assigned to class w_i if $D_i < D_j$, for all $j \neq i$. If $D_i > D_j$, $\underline{X}$ is assigned to class w_j.

The pattern recognition procedure presented here is called the K-means algorithm. This algorithm is based on the minimization of a performance index which is defined as the sum of the squared distances from all points in a cluster domain to the cluster center. This procedure consists of the following steps.

Step 1. Choose k initial cluster centers $\underline{Z_1}(1)$, $\underline{Z_2}(1), \ldots,$ $\underline{Z_k}(1)$. These are arbitrary and are usually selected as the first k samples of the given sample set.

Step 2. At the kth iterative step, distribute the samples ($\underline{X}$) among the k cluster domains, using the relation,

$$\underline{X} \in S_j(k) \quad \text{if} \quad \|\underline{X} - \underline{Z_j}(k)\| < \|\underline{X} - \underline{Z_i}(k)\| \tag{5}$$

for all $i=1,2,\ldots,k$, $i \neq j$, where $S_j(k)$ denotes the set of samples whose cluster center is $\underline{Z_j}(k)$.

Step 3. From the results of Step 2, compute the new cluster centers $\underline{Z_j}(k+1)$, $j=1,2,\ldots,k$ such that the sum of the squared distances from all points in $S_j(k)$ to the new cluster center $\underline{Z_j}(k+1)$ is minimized. In other words, the performance index

$$J_j = \sum_{\underline{X} \in S_j(k)} \|\underline{X} - \underline{Z_j}(k+1)\|^2, \quad j=1,2,\ldots,k \tag{6}$$

is minimized. The $\underline{Z}_j(k+1)$ which minimizes this performance index is simply the sample mean of $S_j(k)$. Therefore, the new cluster center is given by

$$\underline{Z}_j(k+1) = \frac{1}{N_j} \sum_{\underline{X} \varepsilon S_j(k)} \underline{X}, \quad j=1,2,\ldots,k \tag{7}$$

where N_j is the number of samples in $S_j(k)$. This indicates that the cluster centers are sequentially updated.

Step 4. If $\underline{Z}_j(k+1) = \underline{Z}_j(k)$ for $j=1,2,\ldots,k$, the algorithm has converged and the procedure is terminated. Otherwise go to Step 2.

A computer program has been written for the above algorithm in the FORTRAN language (Appendix B). The initial cluster centers are taken arbitrarily to be $\underline{Z}_1$ and $\underline{Z}_2$, i.e., k=2. The program analyses a set of pattern vectors and divides them into two classes or domains. The eventual result is the cluster centers of the two classes are obtained from the sample set.

The distribution of human error over the 43-month period of the plant age for systems, components, types of error, types of effect on system and component, and modes of human error which are most frequently involved are analyzed by using a simple pattern recognition computer program developed by NSRG. The output shows that seven of the distribution curves , Figures 5.2, 5.5, 5.8, 5.9, 5.10, 5.12, 5.15. 5.18, 5.22, characterized by Pattern (II) given in Figure 5.24, and 14 of the distribution curves, the rest, characterized by Pattern (I) given in Figure 5.23. The frequent errors (error/month) in Pattern I are greater than in Pattern II, but the two patterns show almost a constant rate followed by an increase of error rate with passage of time and then the increase decline afterward. This shows that there is a learning behavior.

The computer program listing and the output are given in Appendix B.

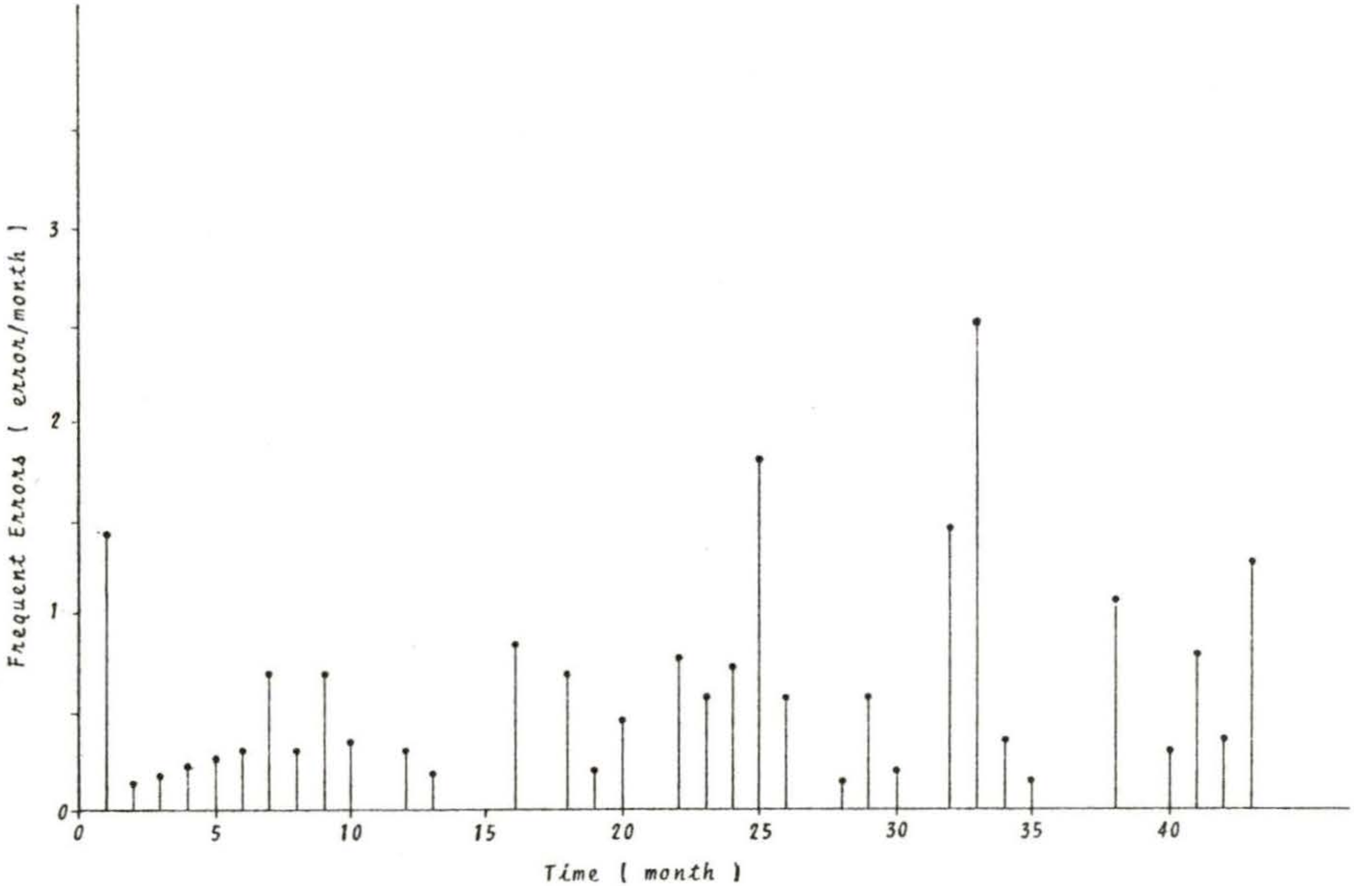

Figure 5.23. Distribution of human error pattern (I)

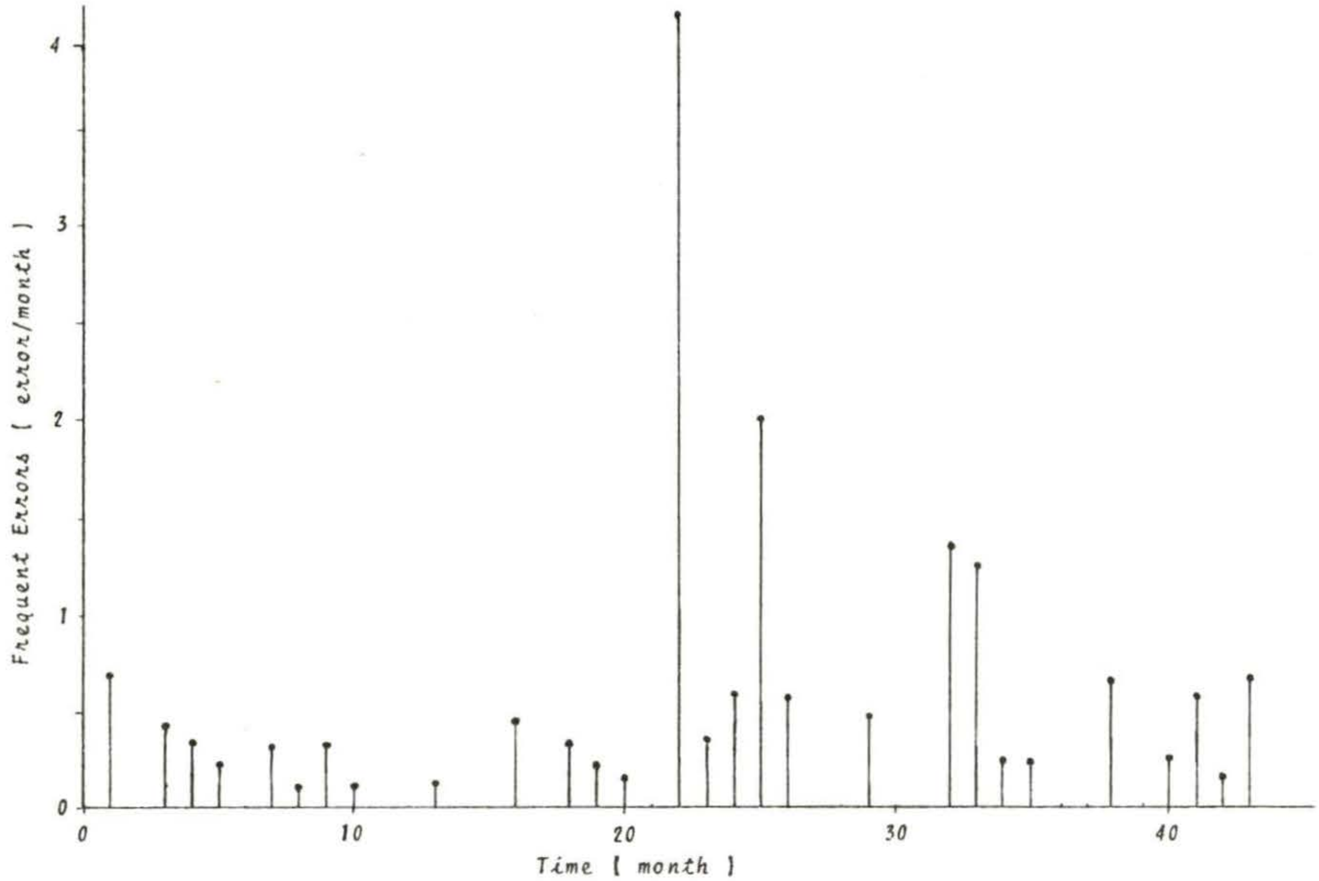

Figure 5.24. Distribution of human error pattern (II)

VI. SUMMARY, CONCLUSIONS, AND RECOMMENDATIONS

The HEGAR classification of gas cooled reactor for coding human errors is developed. The classification describes the general systems, subsystems, and components of HTGR. This classification is flexible to permit expansion, change, added for any system, subsystem, and component and it can be adapted to any HTGR design.

This study shows the importance of human error in Fort St.Vrain, HTGR to safety analysis. The major sources of human error based on manually review of LER's records from May 30, 1974 to December 30, 1977 are maintenance error in improper handling, did not check/test, or improper setting and administrative errors in procedural deficiencies. The systems most frequently involved in human errors are main reactor coolant system, auxiliary electric power system, reactor protection system, and radioactive waste treatment system.

So far this study is adequate for identifying the most frequently involved systems, components, failure modes, and their significance and effects on the plant and the environment, but not enough data have been accumulated on human errors for HTGR in the U.S. to warrant a risk assessment study based on actual data which is very necessary for safety analysis and the availability of nuclear power plants.

For improving human performance emphases should be placed on identification the most frequently involved systems, components, failure modes, and types of error and in the same time evaluation, training, updating, and decision-making programs should be carried out to provide means of reducing the human errors in the plant.

VII. SUGGESTIONS FOR FUTURE WORK

The following are suggestions for future work related to this study.

1. More data need to be collected on human errors for HTGR's to conduct probabilistic analysis based on event/fault/tree construction, block diagram, and probability consequence evaluation.
2. Theoretical and experimental studies have to be performed on data prediction and updating of human errors in HTGR power plants.
3. Evaluation, training, updating, and decision-making studies for system, components, failure modes, and error types which most frequently involved in human errors have to be fulfilled on HTGR plants to reduce human errors, to find out the impact of the studies on safety analysis and the availability of HTGR power plants, and to obtain guidance in optimum design of HTGR power plants.

VIII. REFERENCES

1. Z. A. Sabri and A. A. Husseiny. Evaluation of Gross Operator Error Rates Based on Past Experience in Commercial Nuclear Power Plants, ERISAR-PR-78001. Engineering Research Institute, Iowa State University, Ames, Iowa, 1978.

2. "The Ordeal at Three Mile Island". Nuclear News, Special Report, April 6, 1979.

3. U.S. Nuclear Regulatory Commission. Reactor Safety Study--An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants. WASH-1400. U.S. Government Printing Office, Washington, D.C., 1975.

4. San Francisco Operations Office, U.S. Energy Research and Development Administration. HTGR Accident Initiation and Progression Analysis Status Report, GA-A13617. U.S. Government Printing Office, Washington, D.C., 1976.

5. San Francisco Operation Office, Department of Energy 1. HTGR Accident Initiation and Progression Analysis Status Report, GA-A15000. U.S. Government Printing Office, Washington, D.C., 1978.

6. D. W. Joos, Z. A. Sabri, and A. A. Husseiny, Nucl. Eng. Des., 52, 265 (1978).

7. A. A. Husseiny, Z. A. Sabri, and R. A. Danofsky, Trans. Am. Nucl. Soc., 23, 313 (1976).

8. Z. A. Sabri, A. A. Husseiny, and R. A. Danofsky, Trans. Am. Nucl Soc., 23, 484 (1976).

9. R. A. Danofsky, Z. A. Sabri, and A. A. Husseiny, Trans. Am. Nucl. Soc., 26, 519 (1977).

10. A. A. Husseiny, and Z. A. Sabri, Trans. Am. Nucl. Soc., 30, 666 (1978).

11. S. Kherich, Z. Sabri, and A. Husseiny, Trans. Am. Nucl. Soc., 30, 636 (1978).

12. Z. A. Sabri and A. A. Husseiny, Annals of Nuclear Energy, 6, 309 (1979).

13. H. Y. Cho. Human error data retrieval for U.S. commercial nuclear power plants. M.S. thesis. Library, Iowa State University, Ames, Iowa, 1979.

14. M. A. Azarm. Techniques for data prediction, smoothing, and updating of operator error in commercial nuclear power plants. M.S. thesis. Library, Iowa State University, Ames, Iowa, 1979.

15. U.S. Nuclear Regulatory Commission. Instructions Preparation of Data Entry Sheets for Licensee Event Report (LER) File. U.S. Government Printing Office, Washington, D.C., Report No. NUREG-OIGI, July 1977.

16. M. C. Douglas. Energy Technology Handbook. McGraw-Hill Book Company, New York, 1977.

17. P. Fortescue, Nucl. Eng. Des., 26, 3 (1974).

18. R. N. Quade, Nucl. Eng. Des., 26, 179 (1974).

19. K. C. Lish, Nuclear Power Plant System and Equipment. Industrial Press, New York, 1972.

20. G. L. Wessman and T. R. Moffette, Nucl. Eng. Des., 26, 78 (1974).

21. R. W. Schleicher, "An Analysis of HTGR Core Cooling Capability", General Atomic Company, San Diego, Report No. Gulf-GA-A12504, 30 March 1973.

22. J. M. Harrer and J. G. Beckerley, "Nuclear Power Reactor Instrumentation Systems Handbook." Volume 1 and 2. U.S. Atomic Energy Commission, Washington, Report No. TID-25952-P1 & 2, January 1976.

23. G. Melagan and M. Metwally, "Commercial Nuclear Power Plant Systems--(Light Water Reactors)--A Manual for LER Analysis." Engineering Research Institute, Iowa State University, Ames, Iowa, Report No. ERISAR-M-78002, 1979.

24. G. P. Mclagan and Z. A. Sabri, "An Operator Data Acquisition Methodology Applied to Licensee Event Reports From Nuclear Power Plants With Application For Improved Operator Performance." Engineering Research Institute, Iowa State Univ., Ames, Iowa, Report No. ISU-ERI-Ames-79184, May 1979.

25. A. A. Husseiny and Z. A. Sabri, "Operator ERRORs Summaries." Eng. Research Inst., Iowa State Univ., Ames, Iowa, Report No. ERISAR-PR-78--7, October 1978.

26. D. Meister, "Individual and System Errors in Complex Systems." Am. Psychological Assoc. Convention, St. Louis, Missouri, 1962.

27. A. Hald. Statistical Tables and Formulas. John Wiley & Sons, Inc., New York, 1952.

28. R. Fisher and F. Yates. Statistical Tables for Biological Agricultural and Medical Research. Longman Co., London, 1963.

29. N. Mann, R. Schafer, and N. Singpurivalla. Methods for Statistical Analysis of Reliability and Life Data. John Wiley & Sons, Inc., New York, 1974.

IX. ACKNOWLEDGMENTS

The author likes to offer his deep thanks to Professor Abdo A. Husseiny and Professor Zeniab A. Sabri, for suggestions, advice, encouragement, and unlimited help throughout the work. Thanks also to Professor Herbert T. David, Professor August Valfells, and the nuclear safety group of Iowa State University for their helpful discussions.

I thank King Abdulaziz University in Jeddeh, as well as the Saudi Arabian Educational Mission in Houston, Texas, for their financial support and encouragement.

X. APPENDIX A. HEGAR: II

Component Codes, Failure Mode, and Taxonomy

The following tables of subsystem abbreviation, components, failure modes, and classification codes are to assist the process of (LER) analysis for coding the human errors. The subsystem abbreviation is given in Table A.1 in alphabetical order to aid in identification of the system and in the classification of the human errors. The component list is given in Table A.2 to aid in coding the failed component due to human error. The list is divided into two parts to distinguish between mechanical and electrical components (3, 15, 23). The failure mode is given in Table A.3. It is self-explanatory (6, 23). A brief description of the major components of the classification taxonomy (24, 25) is as follows:

- Identification Number--I.D. assigned to each human event recorded
- Page--LER data printout page number
- Reference Number--LER-assigned event I.D.
- Facility Identification Number--number code assigned to each nuclear facility
- Type--reactor type; i.e., H-HTGR, P-PWR
- Date--date of event occurrence if given; otherwise date of report.

- System Involved--the system (subsystem) involved in the error.
- Component Involved--the major component involved in the error.
- Number of Components Involved--the number of similar components involved in the error.
- Effect on System--effect of the error on the involved system (subsystem): N - None' D - Degraded State; F - Inoperative or Failed State.
- Effect on Component--effect of the error on the component(s) involved: N - None; D - Degraded State; F - Inoperative or Failed State.
- Failure Mode--the action omitted or committed which resulted in error.
- Type of Error--A - Administrative or Procedural Error; M - Maintenance Error; O - Operator Error.
- Omission/Commission--delineates the error as being one of omission or commission.
- Duration--the time elapsed between error occurrence and discovery (in hours) when given: N/A - Not Available.
- Radiation Release to Environment--signifies if the error resulted in an environmental radiation release (quantity listed in Further Description): NO - No Release; RWL - Release Within Limits; REL - Release

Exceeds Limits; RU - Release but Unknown Limits.

- Radiation Exposure--signifies whether there was any radiation exposure as a result of the error (rate listed in Further Description): NO - No Exposure; XP - Exposure; N/A - Not Available/Not Applicable.
- Common Mode/Reoccurring/Unusual Events--C - Common Mode Error; R - an event which is Reoccurring for a particular facility; U - Unusual Event.
- Further Description--a short summary outlining relevant details of the occurrence.

The classification taxonomy with human error data is given in Table 4.1.

Table A.1. Subsystem abbreviations

ARMS	: Area Radiation Monitoring System
CACS-AC	: Core Auxiliary Cooling System-Auxiliary Circulator
CACS-ACSS	: Core Auxiliary Cooling System-Auxiliary Circulator Service System
CACS-AHE	: Core Auxiliary Cooling System-Auxiliary Heat Exchanger
CACS-APCSV	: Core Auxiliary Cooling System-Auxiliary Primary Coolant Shutoff Valve
CAS	: Circulator Auxiliary System
CLPS	: Coolant Loop Protection System
CTS	: Circulator-Trip System
EPS-DC	: On-Site D.C. Sources
EPS-DG	: On-Site A.C. Power System
EPS-EQP	: Auxiliary Equipment for Auxiliary Electric Power System
EPS-OFF	: Off-Site Power System
HPS	: Helium Purification System
HSS	: Helium Storage System
LNS	: Liquid Nitrogen System
MLCS	: Main Loop Cooling System
PCL-HC	: Primary Coolant Loop Helium Circulator
PCL-MCSS	: Primary Coolant Loop Main Circulator Service System
PCL-MHSV	: Primary Coolant Loop Main Helium Shutoff Valve
PCL-SG	: Primary Coolant Loop Steam Generator
PCRV-CS	: Prestressed Concrete Reactor Vessel-Cooling System
PCRV-PRS	: Prestressed Concrete Reactor Vessel-Pressure Relief System
PCS	: Primary Coolant System
PRMS	: Process Radiation Monitoring System
RBVS	: Reactor Building Ventilation System
RCIS	: Reactor Control and Information System

Table A.1 (Continued)

RICI	: Reactor and Incore Instrumentation
RPCPS	: Reactor Protection Control Rod System
PRCWS	: Reactor Plant Cooling Water Systems
RPLS	: Reactor Protection Logic System or Scram System
RRSS	: Reactor Reserve Shutdown System
SCS	: Secondary Coolant System
SWDS	: Steam-Water Dump System
WPS-GH	: Waste Processing System-Gas Handling
WPS-LH	: Waste Processing System-Liquid Handling
WPS-SH	: Waste Processing System-Solid Handling

Table A.2. Component code

Mechanical Components	
AC - Accumulator	RK - Refueling Water Storage Tank
BL - Blower	
BK - Boron Injection Tank	SL - Sluice Gate
AK - Chemical Addition Tank	SG - Steam Generator
CK - Condensate Storage Tank	ST - Subtree
CN - Condenser	SK - Surge Tank
CD - Control Rod Drive Unit	SP - Sump
FA - Cover Plate	TK - Tanks, Other
CM - Damper	TG - Tubing
CL - Diesel	TB - Turbine
DW - Drywell	CV - Valve, Check
XJ - Expansion Joint	EV - Valve, Explosive Operated
FL - Filter or Strainer	HV - Hudraulic Operated
GB - Gas Bottle	XV - Valve, Manual
BK - Gasket	MV - Valve, Motor Operated
HE - Heat Exchanger	AV - Valve, Pneumatic Operated
IP - Incore Probe	RV - Valve, Relief
OR - Orifice	SV - Valve, Safety
PP - Pipe	FV - Valve, Safety-Relief
CP - Pipe Gap	KV - Valve, Solenoid Operated
PV - Pressure Vessel	DV - Valve, Stop Check
PZ - Pressurizer	VV - Valve, Vacuum Relief
PM - Pump	VT - Vent
ED - Reactor Control Rod	WL - Well
RF - Refrigeration Unit	WW - Wetwell

Table A.2 (Continued)

Electrical Components	
AM - Amplifier	MS - Motor Starter
AN - Annunciator	ND - Neutron Detector
BY - Battery	PT - Potentiometer
BC - Battery Charger	RM - Radiation Monitor
BS - Bus	RC - Recorder
CA - Cable	RE - Relay
CB - Circuit Breaker	CN - Relay or Switch Contact
CL - Clutch	RS - Reset Switch
CO - Coil	RT - Resistor, Temperature Device
CS - Control Switch	AD - Signal Comparator
DI - Detector	PS - Switch, Pressure
DE - Diode or Rectifier	TS - Switch, Temperature
DC - DC Power Supply	QS - Switch, Torque
FS - Flow Switch	TM - Terminal Board
FU - Fuse	SB - Test Pushbutton
GE - Generator	OL - Thermal Overload
GS - Ground Switch	TI - Timer
HT - Heat Tracing	CT - Transformer, Current
HG - Heating Element	OT - Transformer, Potential
IM - Input Module	TR - Transformer, Power (or Control)
IV - Inverter (solid State)	
ES - Level Switch	TF - Transmitter, Flow
LT - Light	TL - Transmitter, Level
LA - Lightning Arrester	TP - Transmitter, Pressure
LS - Limit Switch	TT - Transmitter, Temperature
SW - Manual Switch	WR - Wire
MO - Motor	00 - Event (No Component Involved)

Table A.3. Failure mode code

AR - Improper Addition Rate
AS - Improper Assembly
CN - Carelessness/Negligence
CP - Left Partially Closed
CL - Left Closed
CD - Closed
CM - Communication
CC - Did Not Connect
CT - Connection (Other)
CB - Calibration
CA - Calculation
CO - Open Circuit
CS - Short Circuit
DG - Damage (Other)
DI - Inadvertent Damage
DD - Did Not, Deenergize/Disengage/Stop
DV - Inadvertent Deenergization
DL - Left Deenergized
DS - Deenergized/Disengaged/Stopped
ED - Did Not, Energize/Engage/Start
EI - Inadvertent Energization
EL - Left Energized
ES - Energized/Engaged/Started
EV - Event (No Failure Mode)
EX - Lack of Experience
FD - Did Not Fill
FO - Overfilled
FR - Improper Flow Rate
HI - Improper Handling
IA - Inadvertent Actuation
IS - Installation

Table A.3 (Continued)

LK - Leakage
LU - Exceeds Upper Limit
LL - Exceeds Lower Limit
LD - Did Not Lock
LB - Lubrication
MD - Did Not Monitor/Inattention/Failure to Observe
MT - Misinterpretation/Misunderstanding
MJ - Misjudgment
MS - Misadjustment (during repair)
MP - Mispositioning/Misalignment/Improper Setting
OP - Left Partially Open
OL - Left Open
OD - Opened
OI - Improper Operation (other)
OR - Overloaded
OZ - Overpressurized
OT - Overtorqued
PT - Painted
PG - Plugged
PR - Procedure Violation (other)
PD - Did Not Follow Procedure
PW - Wrong Procedure Followed
PU - Procedure Unfamiliar
RA - Did Not Record
RM - Incorrect Recording/Misread
RR - Did Not Remove After Repair
RB - Did Not Remove From Service
RC - Removed From Service
RJ - Inadvertently Removed From Service
RG - Improper Replacement
RI - Improper Repair

Table A.3 (Continued)

RL	- Lack of Repair
RE	- Erroneous Repair
RH	- Did Not Reset
RS	- Slow Response/Time-Too Long
RF	- Fast Response/Time-Too Short
RN	- No Response
RO	- Over Response/Overcompensation
RT	- Did Not Return to Service
RD	- Ruptured/Deformed
SQ	- Improper Sequence
SV	- Severed
TG	- Did Not Tag
TD	- Did Not Test/Check
TN	- Did Not Tighten
UL	- Left Unattended
VI	- Improper Verification
WP	- Left In Wrong Position (other)
WL	- Left Withdrawn
WI	- Improper Withdrawal
WR	- Wiring

XI. APPENDIX B. COMPUTER PROGRAM

```
              DIMENSION X(50,25),Z(50),ZZ(50),A(50),B(50),
             $    SUMS1(50),SUMS2(50),IX(50,25),LL(25),KK(25)
     3        READ (5,10,END=36,ERR=30) M,N
    10        FORMAT (2I3)
              DO 1 J=1,N
     1           READ (5,11,END=34,ERR=32)(IX(I,J),I=1,M)
    11              FORMAT(43I1)
              WRITE(6,85)
    85        FORMAT(26X,'INPUT DATA')
              WRITE(6,86)
    86        FORMAT('+',25X,10('_'))
              WRITE(6,87)
    87        FORMAT(//6X,23('FIG',2X))
              WRITE(6,88)(II,II=1,N)
    88        FORMAT(3X,23(I5))
              WRITE(6,89)
    89        FORMAT('+',5X,113('_'))
              DO 101 II=1,M
                 WRITE(6,90)(IX(II,JJ),JJ=1,N)
    90           FORMAT(3X,23(I5))
   101           CONTINUE
              WRITE(6,91)
    91        FORMAT('1',6X,'OUTPUT')
              WRITE(6,92)
    92        FORMAT('+',6X,6('_'))
              WRITE (6,41) M,N
    41           FORMAT(//6X,'M=',I4,'     N=',I4)
     C
     C
              DO 201 J=1,N
                 DO 202 I=1,M
                    X(I,J)=FLOAT(IX(I,J))
   202              CONTINUE
   201           CONTINUE
     C
```

```
            DO 12 I=1,M
               Z(I) = X(I,1)
               ZZ(I) = X(I,2)
      12       CONTINUE
      C
            WRITE (6,21) (Z(I),ZZ(I),I=1,M)
      21       FORMAT (//(2(3X,F10.5)))
      C
            MM = 0
      2     MM =MM+1
            IF (MM.GT.10) GO TO 20
            L=0
            K=0
            DO 24  J=1,M
               SUMS1(J)=0
               SUMS2(J)=0
      24       CONTINUE
      C
            DO 13  J=1,N
               SUM1=0
               SUM2=0
               DO 14  I=1,M
                  A(I)=X(I,J)-Z(I)
                  B(I)=X(I,J)-ZZ(I)
                  SUM1=SUM1+A(I)**2
                  SUM2=SUM2+B(I)**2
      14          CONTINUE
      C
               IF (SUM1.GT.SUM2) GO TO 15
               L=L+1
            LL(L)=J
               DO 16  JJ=1,M
                  SUMS1(JJ)=SUMS1(JJ)+X(JJ,J)
      16          CONTINUE
                  GO TO 13
```

```
C
15         K=K+1
       KK(K)=J
           DO 66  JJ=1,M
               SUMS2(JJ)=SUMS2(JJ)+X(JJ,J)
66             CONTINUE
13         CONTINUE
C
       DO 17  I=1,M
           SUMS1(I)=SUMS1(I)/L
           SUMS2(I)=SUMS2(I)/K
17         CONTINUE
C
       SS1=0
       SS2=0
       DO 18  I=1,M
           SS1=SS1+(SUMS1(I)-Z(I))**2
           SS2=SS2+(SUMS2(I)-ZZ(I))**2
18         CONTINUE
C
       WRITE (6,22)  (SUMS1(I),SUMS2(I),I=1,M)
22         FORMAT (/(2(3X,F10.5)))
C
       WRITE (6,23) MM
23         FORMAT ('0NO. OF ITERATIONS',I2)
       WRITE(6,80) L
       WRITE(6,81)(LL(II),II=1,L)
       WRITE(6,80) K
       WRITE(6,82)(KK(II),II=1,K)
80     FORMAT(3X,I2,'  :  ')
81     FORMAT('+',8X,25(I3))
82     FORMAT('+',8X,25(I3))
C
```

```
      IF ((SS1.EQ.0).AND.(SS2.EQ.0))  GO TO 20
      DO 100  I=1,M
         Z(I)=SUMS1(I)
         ZZ(I)=SUMS2(I)
100      CONTINUE
      GO TO 2
C
20    WRITE (6,25)  (SUMS1(I),SUMS2(I),I=1,M)
25       FORMAT (' CONVERGENCE IF MM < 10',/(2(3X,F10.5)))
      GO TO 3
C
30    WRITE (6,31)
31       FORMAT (//' IMPROPER SUBSCRIPTS')
         STOP
32    WRITE (6,33)
33       FORMAT (//' IMPROPER/NON-NUMERIC VALUES READ')
         STOP
34    NUMIN=(J-1)*M+I-1
         MXN=M*N
         WRITE (6,35) NUMIN,MXN
35       FORMAT (//' NOT ENOUGH VALUES READ IN',/,I5,' OUT OF',I4)
36       STOP
      END
```

INPUT DATA

FIG 1	FIG 2	FIG 3	FIG 4	FIG 5	FIG 6	FIG 7	FIG 8	FIG 9	FIG 10	FIG 11	FIG 12
0	3	1	1	0	2	0	0	0	2	2	0
0	0	0	0	0	0	2	0	0	0	0	0
1	0	0	0	1	0	1	0	0	0	0	1
0	1	0	0	0	0	1	0	0	0	0	0
1	0	0	0	1	0	1	0	0	0	0	1
0	0	1	0	0	1	0	0	0	0	0	0
0	0	0	1	0	1	0	1	0	0	2	0
0	0	0	1	0	0	0	1	0	0	1	0
1	0	1	0	0	1	1	0	0	0	1	1
0	0	1	0	0	0	0	1	0	0	1	0
0	0	0	0	0	0	0	0	0	0	0	0
1	0	0	0	0	0	1	0	0	0	1	0
1	0	0	0	0	1	0	0	0	1	0	0
0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0
2	2	0	0	0	1	3	0	0	1	0	0
0	0	0	0	0	0	0	0	0	0	0	0
1	0	1	0	0	0	2	0	0	0	0	2
1	0	0	0	0	0	0	1	0	0	0	1
0	0	1	1	0	1	1	0	0	0	0	1
0	0	0	0	0	0	0	0	0	0	0	0
0	5	0	0	5	2	0	3	3	2	0	5
1	0	1	1	0	1	1	0	0	1	1	1
0	0	2	1	0	1	0	1	1	0	1	0
0	0	6	1	0	2	0	4	2	2	3	2
0	0	2	1	0	1	0	1	0	0	0	2
0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	1	0	0	0	0	0	0
1	0	1	0	0	0	2	0	0	0	0	2
0	0	0	0	0	1	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0
1	1	2	0	0	1	3	1	1	1	1	4
2	0	3	2	1	1	5	1	2	0	6	1
1	0	0	0	0	1	0	0	0	1	1	0
0	0	0	0	0	1	0	0	0	0	0	1
0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0
0	0	1	0	0	1	2	1	0	0	3	1
0	0	0	0	0	0	0	0	0	0	0	0
1	0	0	0	1	0	1	0	0	0	0	1
0	0	0	0	1	1	1	1	0	1	1	0
0	0	1	1	0	0	1	0	0	0	0	1
1	0	3	0	0	2	1	1	1	1	1	2

FIG 13	FIG 14	FIG 15	FIG 16	FIG 17	FIG 18	FIG 19	FIG 20	FIG 21	FIG 22	FIG 23
2	1	1	2	4	0	4	0	1	0	0
0	0	0	0	0	0	0	0	0	0	0
0	0	1	0	0	1	1	0	0	0	0
1	0	1	0	1	0	1	0	0	1	0
0	1	0	0	1	0	1	0	0	0	0
1	0	0	0	1	0	1	0	0	0	0
0	0	2	0	2	0	0	2	0	0	1
0	0	0	1	1	0	1	0	0	0	0
0	0	2	0	2	0	1	1	0	0	1
0	0	0	0	1	0	1	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0
0	1	0	0	1	0	1	0	0	0	0
1	0	0	1	1	0	1	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0
4	1	0	3	3	1	3	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0
0	0	1	1	2	0	2	0	0	0	0
0	0	0	1	1	0	1	0	0	0	0
1	0	0	2	1	0	2	0	0	0	0
0	0	0	1	0	0	0	0	0	0	0
0	0	5	0	1	4	5	0	0	5	0
0	1	0	0	1	1	2	0	0	0	0
1	0	2	0	1	1	2	0	0	0	0
1	0	5	1	3	3	6	0	1	0	1
0	0	1	1	1	1	2	0	0	0	1
0	0	0	0	0	0	0	0	0	0	0
1	0	0	1	1	0	0	1	0	0	1
0	1	1	0	2	0	2	0	0	1	1
1	0	0	1	1	0	1	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0
0	1	3	0	4	1	5	0	3	0	0
0	1	6	0	7	0	7	0	0	0	1
0	0	1	0	1	0	1	0	0	0	0
0	0	0	0	0	1	1	0	0	0	1
0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0
0	0	2	0	4	0	1	3	0	2	0
0	0	0	0	0	0	0	0	0	0	0
0	0	0	1	1	0	1	0	0	0	0
2	0	1	2	3	0	3	0	0	1	1
0	0	0	1	1	0	1	0	0	0	0
1	1	1	0	4	0	4	0	0	0	2

OUTPUT

M= 43 N= 23

0.0	3.00000
0.0	0.0
1.00000	0.0
0.0	1.00000
1.00000	0.0
0.0	0.0
0.0	0.0
0.0	0.0
1.00000	0.0
0.0	0.0
0.0	0.0
1.00000	0.0
1.00000	0.0
0.0	0.0
0.0	0.0
2.00000	2.00000
0.0	0.0
1.00000	0.0
1.00000	0.0
0.0	0.0
0.0	0.0
0.0	5.00000
1.00000	0.0
0.0	0.0
0.0	0.0
0.0	0.0
0.0	0.0
0.0	0.0
1.00000	0.0
0.0	0.0
0.0	0.0
1.00000	1.00000
2.00000	0.0
1.00000	0.0
0.0	0.0
0.0	0.0
0.0	0.0
0.0	0.0
0.0	0.0
1.00000	0.0
0.0	0.0
0.0	0.0
1.00000	0.0

1.42857	0.66667
0.14286	0.0
0.21429	0.44444
0.28571	0.33333
0.35714	0.22222
0.35714	0.0
0.64286	0.33333
0.35714	0.11111
0.71429	0.33333
0.28571	0.11111
0.0	0.0
0.42857	0.0
0.42857	0.11111
0.0	0.0
0.0	0.0
1.42857	0.44444
0.0	0.0
0.64286	0.33333
0.28571	0.22222
0.71429	0.11111
0.07143	0.0
0.57143	4.11111
0.71429	0.33333
0.64286	0.55556
1.78571	2.00000
0.64286	0.55556
0.0	0.0
0.42857	0.0
0.71429	0.44444
0.35714	0.0
0.0	0.0
1.50000	1.33333
2.50000	1.22222
0.35714	0.22222
0.21429	0.22222
0.0	0.0
0.0	0.0
1.07143	0.66667
0.0	0.0
0.35714	0.22222
1.00000	0.55556
0.42857	0.11111
1.42857	0.66667

NO. OF ITERATIONS 1

14 : 1 3 4 6 7 11 13 14 16 17 19 20 21 23

9 : 2 5 8 9 10 12 15 18 22

1.42857	0.66667
0.14286	0.0
0.21429	0.44444
0.28571	0.33333
0.35714	0.22222
0.35714	0.0
0.64286	0.33333
0.35714	0.11111
0.71429	0.33333
0.28571	0.11111
0.0	0.0
0.42857	0.0
0.42857	0.11111
0.0	0.0
0.0	0.0
1.42857	0.44444
0.0	0.0
0.64286	0.33333
0.28571	0.22222
0.71429	0.11111
0.07143	0.0
0.57143	4.11111
0.71429	0.33333
0.64286	0.55556
1.78571	2.00000
0.64286	0.55556
0.0	0.0
0.42857	0.0
0.71429	0.44444
0.35714	0.0
0.0	0.0
1.50000	1.33333
2.50000	1.22222
0.35714	0.22222
0.21429	0.22222
0.0	0.0
0.0	0.0
1.07143	0.66667
0.0	0.0
0.35714	0.22222
1.00000	0.55556
0.42857	0.11111
1.42857	0.66667

```
NO. OF ITERATIONS 2
  14  :   1  3  4  6  7 11 13 14 16 17 19 20 21 23
   9  :   2  5  8  9 10 12 15 18 22
CONVERGENCE IF MM < 10
     1.42857       0.66667
     0.14286       0.0
     0.21429       0.44444
     0.28571       0.33333
     0.35714       0.22222
     0.35714       0.0
     0.64286       0.33333
     0.35714       0.11111
     0.71429       0.33333
     0.28571       0.11111
     0.0           0.0
     0.42857       0.0
     0.42857       0.11111
     0.0           0.0
     0.0           0.0
     1.42857       0.44444
     0.0           0.0
     0.64286       0.33333
     0.28571       0.22222
     0.71429       0.11111
     0.07143       0.0
     0.57143       4.11111
     0.71429       0.33333
     0.64286       0.55556
     1.78571       2.00000
     0.64286       0.55556
     0.0           0.0
     0.42857       0.0
     0.71429       0.44444
     0.35714       0.0
     0.0           0.0
     1.50000       1.33333
     2.50000       1.22222
     0.35714       0.22222
     0.21429       0.22222
     0.0           0.0
     0.0           0.0
     1.07143       0.66667
     0.0           0.0
     0.35714       0.22222
     1.00000       0.55556
     0.42857       0.11111
     1.42857       0.66667
```