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PROBABILISTIC STUDY OF LOFA IN ETRR-1 REACTOR

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ABSTRACT

In evaluating the safety of a research reactor an analyses of the reactor response to a wide range of postulated initiating event must be carried out, these events could lead to an anticipated operational occurrences or accident conditions. These disturbances include decrease in heat removal by the reactor coolant system, which may be due to Loss Of Flow Accident(LOFA), loss of coolant or loss of heat sink. LOFA is considered for this study with application to tank type research reactor; as an example of this type is Egypt Test and Research Reactor Number 1(ETRR-1). The probabilistic method is used in this work. ETRR-1 reactor is provided with engineering safety feature systems(ESFs) to respond to possible accidents and perform mitigating functions. The possible malfunctions, component failures, or operator errors leading to a LOFA as an initiating event are investigated (e.g. pipe blockage, valve closing, pump failure..etc). The basic initiating event frequency/probability is calculated using appropriate probability models. The logic Event Tree model is constructed to illustrate all possible accident scenarios. Fault Tree technique is adopted to determine engineering safety systems failure probabilities. The results show the possible minimal cut sets corresponding to each system failure condition. Accident sequences leading to clad rupture and the resulting release categories are presented. Sensitivity of plant state to component failures, operator errors and system failure are presented. Possible weak points in the design are ,consequently, highlighted .

INTRODUCTION

The safety analyses and the deign of the reactor are complementary processes that are carried out interactively(1). The safety analysis is mainly used to demonstrate how the design and related operational procedure will contribute to prevention and mitigation of accident. It include analyses of the response of the reactor to a range of postulated initiating events and as a result, it provides a significant contribution to the selection of operational limits and conditions and of design specifications for components and systems, So these analyses are necessary to demonstrate that the overall risk of operation, or the safety margins are acceptable(2).

The well accepted approach to developing a safety analysis for a research reactor is to consider limiting credible accident initiating events in a deterministic manner for estimating the possible release to the environment. The Probabilistic technique ,however, serves for: (a)evaluating which accident sequences are of higher likelihood and for (b)evaluating relative risk ranking. As a consequence it serves for identifying which sequence will result in highest risk. Such sequence will then be analyzed using the deterministic approach referred to above. In addition through probabilistic technique one can: (a)identify hidden weakness of the design, (b) quantify the value of possible improvement or modification(3). The event tree serves as a logic model to combine system success and failure probabilities with the probability of a postulated initiating event occurring(4). The system failure probabilities are derived through the use of Fault Trees. Fault Tree analysis is a systematic procedure used to examine systems in order to determine component failure modes and other events, that can individually or in combination with each other cause system failure(5).

Nuclear reactors are provided with various engineering safety features (ESFs) such as Reactor shutdown system, Core cooling system and Fission product Confinement system(6). With all ESFs operating at their minimum design basis, accident sequence probability is quite small, but with any one or all of the ESFs not performing their designed function, a broad spectrum of accident sequences can occur, each with a probability and consequences dependent on the operability state of the various ESFs.

Disturbances which may occur in a reactor life time include: decrease in heat removal by reactor cooling system, loss of heat sink, loss of electric power supply and reactivity insertion accident[7,8]. The main contribution of the loss of heat removal are Loss of Flow Accident'LOFA' or Loss of Coolant Accident'LOCA'. The objective of the present work is to investigate the LOFA. Such accident refer to partial or total loss of forced flow. the total loss of flow will be presented.

SYSTEM DESCRIPTION

1-Primary Cooling Circuit

Fig.(1) represents the Primary Core Cooling Circuit 'PCCS' and Table 1 contains description of the circuit valves[9]. The circuit contains: 5 centrifugal pumps connected in parallel; two operating and three standby, 2 heat exchanger (H.E) connected in series, dearator, ion exchange filter(IEF), and pipe line. The flow(860 m3/hr) circulates through the reactor core in downward direction then into the outlet pipe (OL), 350 mm diameter. Part of the flow (10 m3/hr) is directed to IEF, the diverted flow leaves the filter to connect the second return line(RL2) ,200 mm in diameter, then to the reactor middle tank. The main flow is drived by the pumps and delivered to the first H.E and then to the second H.E, the flow leaving the H.E goes to the main return line(RL1) towards the reactor middle tank. Part of the flow(15%) is directed to the deaerator after which it joins the second return line towards the reactor middle tank.

Table 1 Valve description of P.C.Cs.

Valve no.	Description	Normal position
1,2,3,4,5,17	Motor operated	Normally opened
6,7,8,9,10,18,19	Manual operated	Normally opened
25,26,27,35	Motor operated	Normally closed
28,29,30	Manual operated	Normally closed

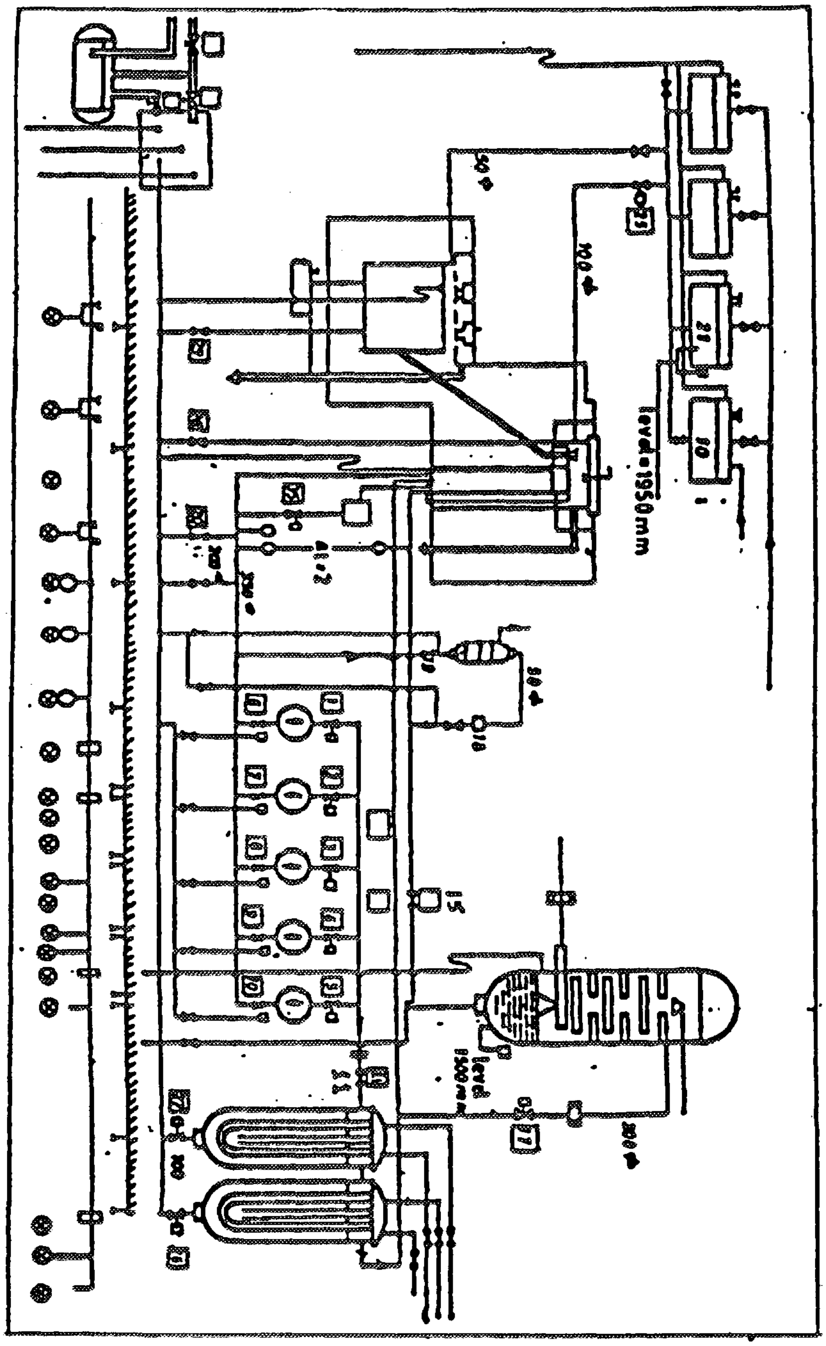

FIG. (1) FIRST CIRCUIT LINE DIAGRAM

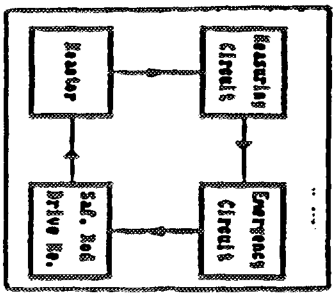

Fig. 2 Reactor Shutdown Circuit

2-Shutdown System

The reactor shutdown system(RSD) fig.2 consists of measuring channels, emergency logic circuit and safety rod drive mechanism[9]. The safety rod mechanism fig. 3 consists of 48v D.C. motor, electromagnet, drum, connecting gears and change over switch. The safety rod configuration inside the core is shown in fig. 4. A safety rod is withdrawn by its own motor at a speed of 2cm/sec at normal operation during start up, and is hold in its upper position by the electromagnet. In case of emergency shutdown(SCRAM) the rods are dropped under the force of compressed spring, and its weight within 0.2 sec. In normal shutdown the rods are inserted at a speed of 2.0 cm/sec.

3-Confinement System

The confinement system consists of:-

a- the reactor hall structure

b-the ventilation system

a-The reactor hall

the reactor hall has dimension of 30.5 x 21.5 x 19.95m. Access to the hall is through 5 staircases; 2 in the first floor, 2 in the second floor, and one in the third floor. All access are provided with sealed door. The glass windows of the reactor hall are sealed by past and are never opened. The opening for access for large object is provided with a sealed double door. The walls on the side of the hall building are made of 1.0 m thick concrete, the thickness of other walls is 0.5 m, the roof is vaulted shape reinforced concrete with tie.

b-Ventilation System

It is installed ,see fig. 5, to remove radioactive gases from the areas above and under the reactor, the pump room, and the hot cells, via air rarefaction. It includes 6 groups of ventilators, ventilation control room, and stack of 43 m height. Each group contains two ventilators(blowers) one is working and the other is stand by, two electric valves are installed before and after each ventilator. No air filter nor iodine traps are included in the system.

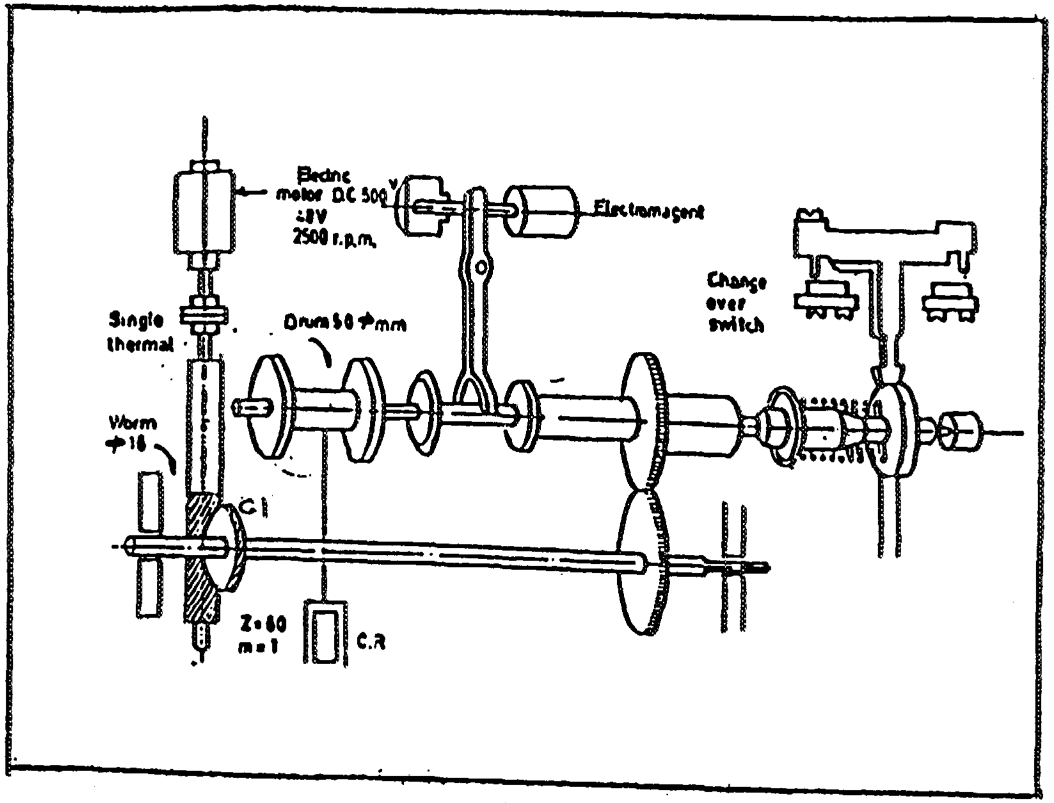

Fig. 3 Safety Rod Mechanism

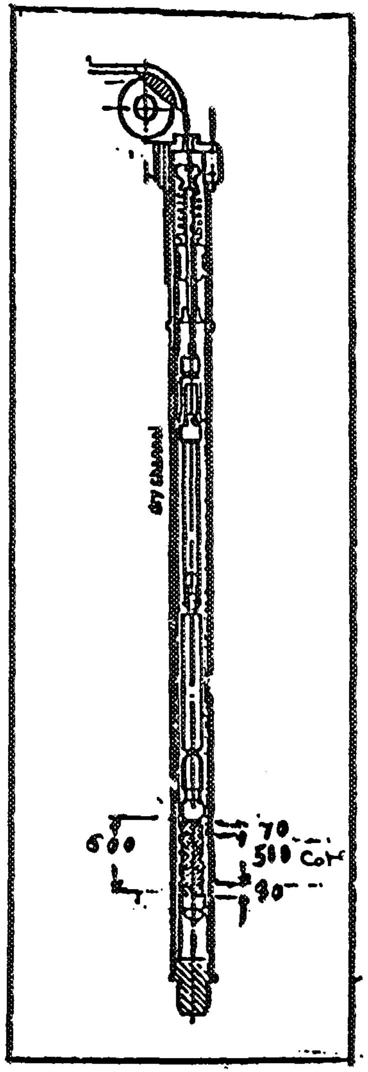

Fig. 4 Safety Rod Cofiguration Inside the Core
(Dims. In mm)

LOGICAL MODEL

a-Event Tree

Event Tree is constructed in fig. 10 for total loss of flow accident. The basic event annual probability (TLOFA) is calculated in the next section. The reactor shutdown system(RSD) is activated by a scram signal if the flow is reduced by amount 20% of its nominal value. As analyzed in [10], the decay heat liberated after shutdown is transferred to the surrounding air across the H.Es via loop natural (NCC) at a rate of 10,600 Kcal/hr. Following a TLOFA The average core clad temperature reaches 47 ^{o}c after 100 sec from reactor shutdown[11]. On the other hand if NCC outside the core is unavailable the decay heat is transferred through pool natural circulation into reactor water tank[10], under this condition the core water temperature reaches the boiling point 90.4 ^{o}c after 70 hrs from reactor shutdown. If the RSD failed, the average clad temperature reaches 125 ^{o}c at 30 sec from the failure of the system. This accident at power condition can develop dry out, clad rupture and release of radioactive material into cooling water. In the above cases the confinement ventilation system is required to function.

b-Fult Tree

The block diagrams illustrating the safety rod mechanism, reactor core cooling system, ventilation system, and reactor hall are shown in figs. 6,7,8 and 9, while the corresponding fault trees are given in figs. 11,12,13 and 14. The main contribution of failure of RSD are no scram signal, break of the gears; break of the spring, break of the drum, energizing of the electromagnet, cut of the probe, blockage of the dry channel. The failure of NCC may due to closing of the valves v1;v2;v6;v7;v11, failure of pumps p1 and p2, blockage of the outlet pipeline OL, blockage of the main return pipeline RL1, H.E.1 blockage or H.E.2 blockage. The failure of the confinement ventilation system may occur as a result of closing of duct opening, fan failure, blower failure, stack closed or duct break. The reactor hall building insulation system BS fails if any of the five doors or the main door is not closed perfectly, or break of the hall windows.

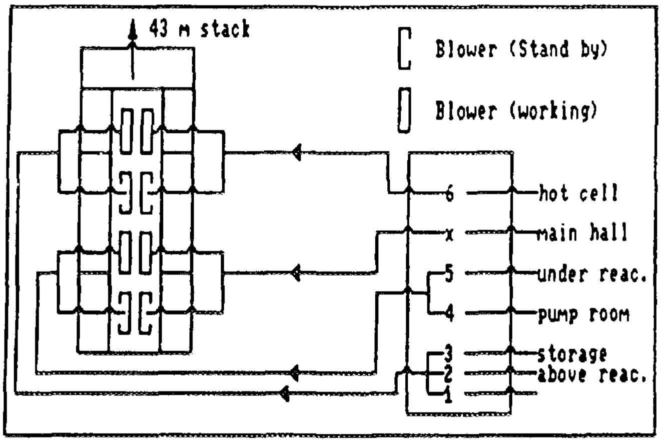

Fig. 5 Ventilation System

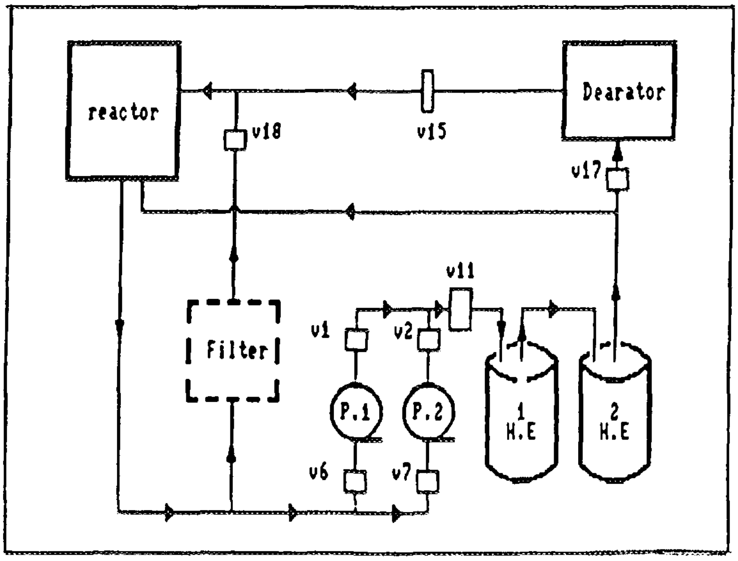

Fig. 6 Block Digram of The Reactor Cooling Circuit

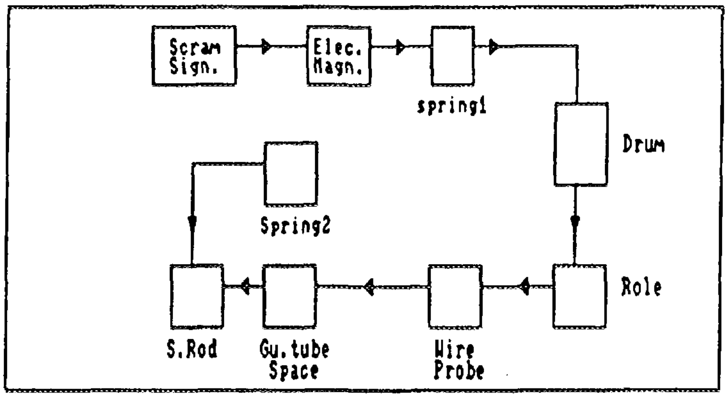

Fig. 7 Block Diagram of The Safety Rod Mechanism

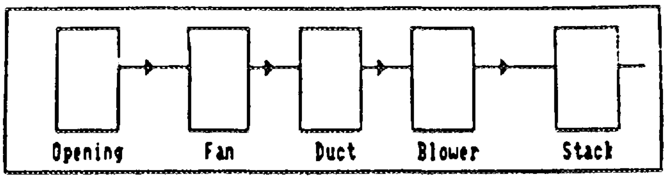

Fig. 8 Block Diagram of The Ventilation System

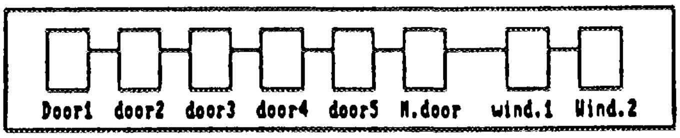

Fig. 9 Block Diagram of The Reactor Hall

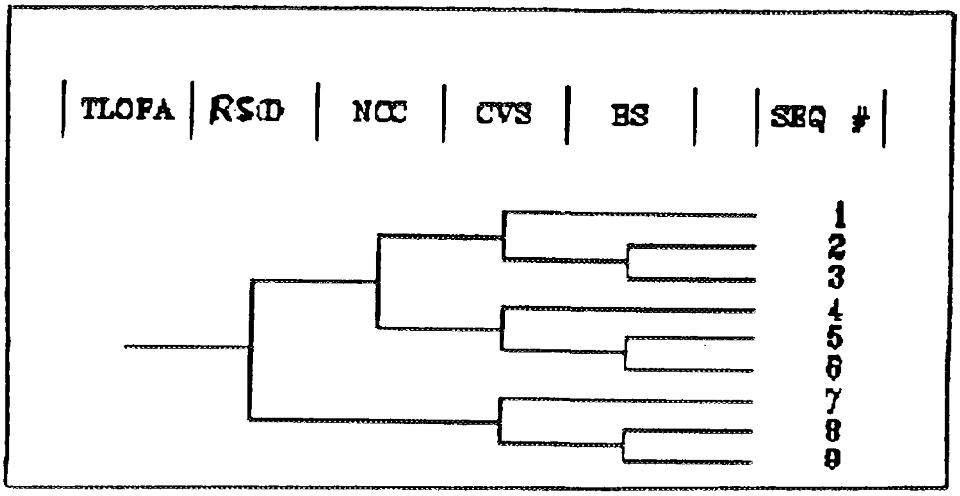

Fig. 10 TOTAL LOSS OF FLOW

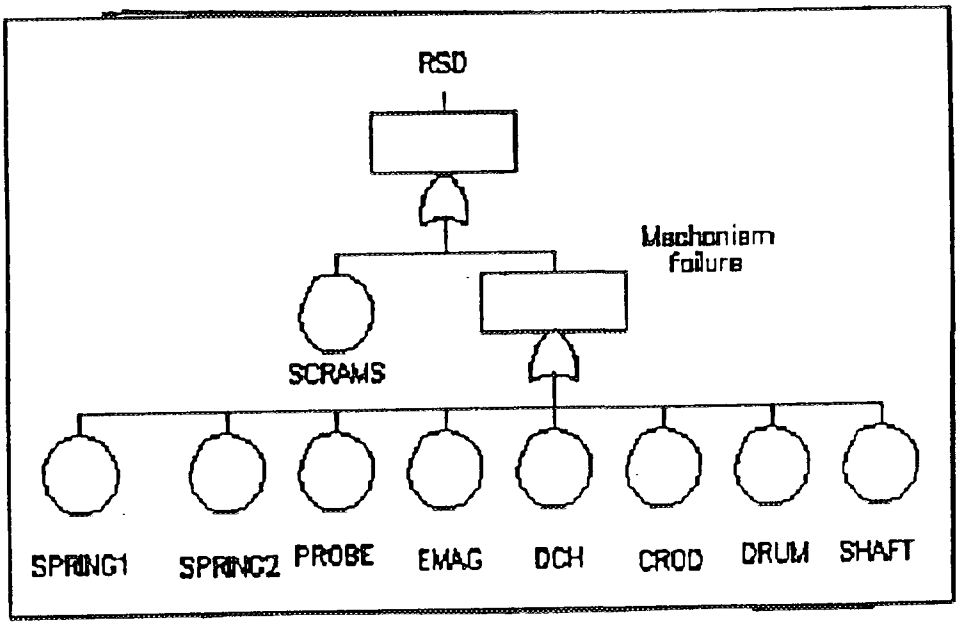

FIG. 11 RSD FAULT TREE

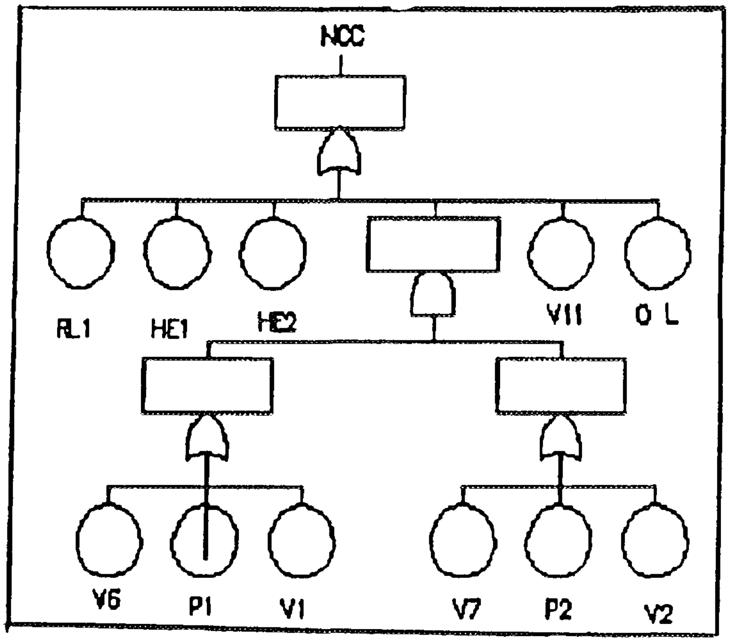

FIG. 12 NCC FAULT TREE

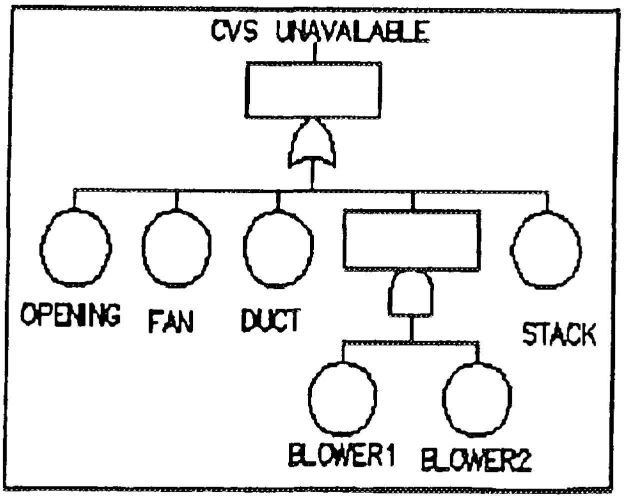

FIG. 13 CVS FAULT TREE

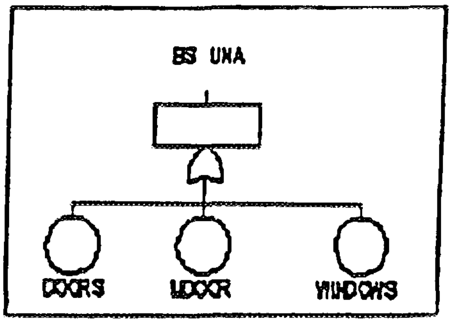

FIG. 14 BS FAULT TREE

CALCULATION

The total loss of flow basic event(TLOFA) may arise due to:-

1-superior valve closing 2-pump failure

3-pipe blockage 4-H.E. blockage

1-Superior valve closing

As given in table 1 the valves 6, 7 and 11 are manually operated and normally opened. The valves 1 and 2 are motor operated and normally opened. During normal operation the accessibility of operator to pump room is prevented, so closing of manual valves 6,7 and 11 due to human error is impossible. The reactor cycle is 48 hrs/2 weeks then,

The annual operating time(to) =1248hrs, (1)

From reference[12] we have:-

manual valve failure rate (vlam)=8.9 E-8 /hr (2)

and using the commutative probability formula:-

p = 1- exp (- vlam *to), (3)

therefore:-

p(v6) = 1.1107 E-4. (4)

The manual valves v6 , v7, v11 are identical.

For motor operator valves 1 or 2 the failure rate is given by[12],

vlam=2.7E-7/hr, (5)

therefore:-

p(v1)=3.369 E-4. (6)

2-pump failure

The pump failure may be due to failure of pump motor, failure of pump coupling, or failure of pump control logic. From reference [12] we have:-

the pump failure rate = 1.5 E-5 /hr, (7)

therefore:-

p(p1) =p(p2) = 1- exp (- 1.5 E-5 *1248) = 1.8546 E -2. (8)

3-pipe and heat exchanger blockage

The blockage of pipes or heat exchanger by human error is responsible for this event, this may occur during maintenance, It is assumed, while the primary circuit is opened for maintenance, an object may slip or fall inside. If we consider[12] :-

rate of human error per operation = 1.0 E -3 , (9)

annual maintenance frequency =1, (10)

therefore;-

p(pipe block or H.E block) = 1.0 E-3 * 1 = 1.0 E-3, (11)

p(OL) = p(RL1) = p(H.E1) = p(H.E.2) = 1.0 E -3 , and (12)

p(PL1)=p(PL2)=p(p1)+p(v6)+p(v1) = 1.89939 E -2 (13)

Then the TLOFA can be calculated as follows:-

p(TLOFA) = p(RL1) + p(HE1)+p(HE2)+p(v11)+p(OL)+p(PL1*PL2) (14)

p(TLOFA) = 4.4718 E -3 (15)

RESULTS AND DISCUSSIONS

Using the PSAPACK code[12] with the constructed fault trees(fig. 11,12,13 and 14), the failure probabilities and the minimal cut sets of order 1 or 2 for the systems RSD, NCC, CVS, and BS are given in table 2. The obtained top event probabilities of the previous ESFs, the system event tree, and the postulated basic event frequency of occurance of TLOFA are processed by the code, nine accident sequences were obtained. The results are stated in table 3 with emphasize on clad failure probability and release category.

From above results we reached to :-

1-The reactor system is unaffected, if total loss of flow occur, in condition that it is scrammed, natural circulation is available through the primary circuit, and ventilation system is working.

2-If the reactor scrammed due to TLOFA , both NCC and CVS are not available , the radioactivity level inside reactor hall may increase above normal level with probability of order 2.0E-5 and with probability 1.3E-7 outside reactor building.

3-If the reactor doesn't scrammed due to TLOFA and the CVS is unavailable, the clad rupture may occur the reactivity level increases to above normal condition in the restricted areas with probability of .02

4-The worst scenario for TLOFA occur when the reactor doesn't scrammed and both CVS and BS are fail, the radioactivity level increases in and outside reactor hall with probabilities of 7.8E-5 and 5E-7 respectively

5-Main contribution to TLOFA is due to pump failure and human error

6-The loss of main electric power supply is not considered as a cause of component failures. It will be investigated separately in future work.

7- Radio active removal components(filters) have to be added into The CVS

Table 2 Failure Probability of E.S.Fs

System	Probability	No of Mcs* of order 1	No. of Mcs of order 2
RSD	.02	9	-
NCC	.0052	5	9
CVS	.0039	4	1
BS	.0064	3	-

* Minimal cut sets

<u>Definition of the number of Mcs</u>

No 6 (Failure of: Probe-Electro magnet-Spring-Scram signal-Guide tube-Control rod-C.R. spring-Drum-Shaft)

No 5 (Failure of : Return pipe-First H.E.- Second H.E.-Output pipe-Valve 11)

No 9 (Failure of: Pipe1 and pipe2 -Valve2 and pipe1-Pipe 2 and valve 6-Pipe 2 and valve1-Valve7 and Pipe1- Valve7 and Valve6 - Valve2 and Valve6 - Valve2 and Valve1 - Valve7 and Valve1)

No 4 (Failure of : Fan- Opening- Stack- Duct)

No 1 (Failure of : Blower1 and blower2)No 3 (Failure of : Doors-Windows- Main door)

Table 3 Sequence Probability and Release Category+

Sequence no	Probability	Clad Rupture	Release Category
1 -	------	No	Normal Operation, No release
2-	.0039	No	Small R.A. level increase for operator*
3-	2.5E-5	No	Small R.A. level increase for operator and reactor site
4-	.0052	No	No R.A. level increase for operator and small increase for public** after 70 hrs
5-	2.0E-5	No	Medium R.A. level increase for operator after 70hr
6-	1.3E-7	No	Medium R.A. level increase for operator and reactor site*** after 70 hrs
7-	.02	Yes	Large R.A. level increase for public after 30 sec
8-	7.8E-5	Yes	Large R.A. level increase for operator after 30 sec
9-	5E-7	Yes	Large R.A. level increase for and reactor site after 30 sec

+ It is should be noted that the release categories are obtained when there is no radioactive removal filters as part of the CVS.

*operator refer to restricted areas as reactor hall and pump room

** public refer to people outside reactor site and in the region of down flow stream from stack

*** reactor site refer to areas outside reactor hall and in the site location

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