Monograph:

Jalel, N.A. and Nicholson, H. (1990) Expert Systems for Fault Diagnosis in Nuclear Reactor Control. Research Report. Acse Report 412 . Dept of Automatic Control and System Engineering. University of Sheffield

Expert Systems for Fault Diagnosis in Nuclear Reactor Control

by

N. A. Jalel and H. Nicholson

Department of Control Engineering
University of Sheffield
Mappin Street
Sheffield S1 3JD

Research Report No. 412
November 1990

1 Abstract

An expert system for accident analysis and fault diagnosis for the (Loss Of Fluid Test) LOFT reactor, a small scale pressurised water reactor, has been developed for a personal computer. The knowledge of the system is represented using a production rule approach with a backward chaining inference engine. The data base of the system includes simulated time dependent state variables of the LOFT reactor model.

Another system is designed to assist the operator in choosing the appropriate cooling mode and to diagnose the fault in the selected cooling system. The response tree, which is used to provide the link between a list of very specific accident sequences and a set of generic emergency procedures which help the operator in monitoring system status, and to differentiate between different accident sequences and select the correct procedures, is used to build the system knowledge base.

Both systems are written in Turbo Prolog language and can be run on an IBM PC compatible with 640k RAM, 40 Mbyte hard disk and colour graphics.

2 Introduction

Expert systems are intended to solve problems normally requiring the knowledge and skills of human expertise. They consist of a knowledge base and inference engine that provides the mechanisms for symbolic reasoning, search and explanation.

In general, an expert system can be defined as a computer program that is used to solve problems that involve the manipulation of knowledge in a given domain, perform difficult tasks at a level of experts, make decisions and provide explanations for conclusions reached. They have been developed for a wide variety of applications in various fields of science, engineering, management, finance and business. A survey of the application of artificial intelligent methodologies to control system engineering is described by Linkens [1].

Expert systems for fault detection, diagnosis and correction have been used in different areas including, computer malfunction diagnosis, power networks, faults in VLSI chips, chemical and nuclear plant fault diagnosis, software diagnosis, aircraft diagnosis and in manufacturing. A review of the capabilities and limitations of some of the existing diagnostic expert systems, with a survey of available expert systems shells is given by Scherer and Pau [2,3].

All the studies after the Three Mile Island (TMI) accident recommended the development of improved methods for selecting, processing, and displaying relevant information to the operator to allow him to more efficiently cope with abnormal situations. Researchers have begun to investigate the potential for using expert systems in the development of decision aids for reactor operators. In nuclear engineering, they have been used for fault diagnosis, accident analysis and to aid nuclear reactor operators to choose appropriate action.

Some survey papers in this field provide an overview of the application of expert systems in nuclear technology. Bernard and Washio [4] examined the motivation for utilizing expert systems, identified the areas to which they were being applied and provided an assessment of their utility. An overview of expert system applications in Westinghouse nuclear fuel activities, where they have been used in the fields of, fuel fabrication, Zirconium tubing, Zirconium production and core design, has been undertaken by Leech and Casadei [5]. The applications of expert systems in the nuclear industry in different countries is given by Majumdar in his survey paper [6]. A review of some of the expert systems in use or under development in the nuclear industry is given by Kretzschmar [7]. Expert systems for fault diagnosis in nuclear reactors are discussed by Sudduth [8] according to the approach used to build the knowledge base. A survey of expert systems intended to support control room personnel for achieving improvements to tactical response during plant emergencies, and viewed as strategic aids to accident management, is examined and illustrated by Cain [9].

This paper is divided into two parts, in the first the expert system used for fault diagnosis and accident analysis is illustrated while in the second the expert system used to assist the nuclear reactor operator in choosing the appropriate cooling mode is discussed and illustrated. In the first part, the general description of the system is discussed, the knowledge engineering of the system explained, the structure of the system illustrated and the performance of the system is examined through the use of a hypothetical simulated plant accident. In the second part, the knowledge of the system which includes the cooling system of the LOFT reactor and the response tree methodology is explained, the expert system components are illustrated and finally the performance of the system is tested using a hypothetical accident.

PART 1

3 General Description of the System

The main purpose of this prototype expert system is to monitor the state condition of a specific type of pressurized water reactor called the LOFT reactor [16,17] and to assist the operators in identifying and diagnosing different types of accidents that might occur in the reactor. This system also could be used as a training tool to help operators to diagnose different types of accidents.

The system is built using If-Then rules to represent the knowledge of the system with a backward chaining inference engine. Turbo Prolog language is used to develop the system which is implemented on an IBM PC compatible with 640 k RAM, colour graphics and a 40 Mbyte hard disk.

The input data for the system includes the state variables of the simulated LOFT reactor model at different intervals of time and additional information about the whole plant collected by interrogation using 'yes', 'no' answers. Two approaches have been used to input the state variables of the LOFT model; in the first, the 27th order model is simulated on a Sun workstation using C language, in which the state variables are stored in a file on a floppy disk which the system is able to read; while in the second, the simulated model and the expert system are installed on the same PC, and hence a communication between the two programmes has been achieved. The system starts the communication by sending the total time of simulation and the value of the integration interval to the simulation program where the values of the state variables will be calculated and sent back to the system to be analysed for any accident which might occur. The communication procedure between the two program continues until the end of the simulation.

The output of the system includes a list of some of the important values of the state variables and an alarm for any deviation in their values with a list of possible causes for the deviation and possible faults in the system which cause the deviation. During the consultation, the system will be able to send the user first, second and even third alarms for different accidents that might occur in the plant, until the system is able to identify the exact accident in the reactor.

4 Knowledge of the Problem

The diagnostic knowledge used to build the expert system is given as heuristic knowledge which is collected from past experience gained in dealing with the accidents and is usually given in the form of given evidence and conclusion.

This knowledge, which is obtained from different literature sources and mainly from [10,11], was difficult to obtain because very little relevant information has been published in the nuclear reactor field.

The knowledge of the system involves diagnosis for possible causes of the deviation in critical plant parameters which include, reactor power, reactor coolant temperature, reactor coolant flow, pressuriser system parameters, steam generator system parameters and feedwater heater parameters. It also includes an early detection of possible serious faults that might occur in the plant and finally contains a diagnostic study of all expected malfunctions and abnormal operations in a nuclear power plant.

5 Structure of the Expert System

The system is built to identify, diagnose and analyse any abnormal behaviour in the pressurized water reactor. The general structure and the calculation procedure performed by the system can be divided into several stages, where in the first stage the important state variables of the simulated LOFT reactor will be entered to the system. The process of reading the data from the floppy disk is illustrated in figure 1, and the system will read the first set of data and analyse these for any accident or abnormal behaviour that might occur in the reactor. However, if the system is unable to identify any accident it will return to the floppy disk to read the second set of data which normally will be at the second interval of time, and the procedure will continue until the system is able to identify the accident.

The second approach of entering the data to the system is based on an advanced technique of linking Turbo Prolog, which is used to build the system, with Turbo C language, which is used to simulate the LOFT reactor model. The communication procedure between the two languages for entering the state variables is illustrated in figure 2. The communication starts by sending the values of the simulation time and integration interval chosen by the operator from the system to the simulation program to calculate the first set of data which will be sent back to the system for diagnosing and analysing any possible accidents that might occur in the reactor. The system will then send a signal to the Turbo C program asking for the second set of data if no accidents have been identified, and the procedure will continue until the fault is identified.

In the second stage, the input state variables of the simulated model will be analysed for any deviation from the normal operation limit. The deviation could happen as a result of an increase or decrease in the values of the state variables due to abnormal behaviour in the reactor. To identify the deviation, two limits have been designed for each state variable as shown in figure 3, where the first indicates a decrease or increase in the value of the state variable while the other indicates further increase or decrease.

After displaying the most sensitive plant parameters to the operator for each interval of time, the deviation in the plant parameters will be analysed to identify the possible causes of increase or decrease in the values of the state variables and to diagnose the possible accidents that might occur in the reactor.

In the last stage, the expert system is used to diagnose any alarms and accidents that might occur in the reactor. The fault tree for the different alarms and accidents is coded in the system using If Then rules in the form of

alarm (X) if
condition 1 and
condition 2.

For example, the fault tree given in figure 4 illustrates an accident due to failure in the rod drop which is coded by the system using Turbo Prolog language as

alarm (First alarm for rod drop fault) if
control rod position indicator is on.

and

accident (rod drop fault) if
(decrease in the reactor power) and
(decrease in the average coolant temperature) and
(decrease in the pressuriser pressure).

Figure 5 shows the overall structure of the expert system.

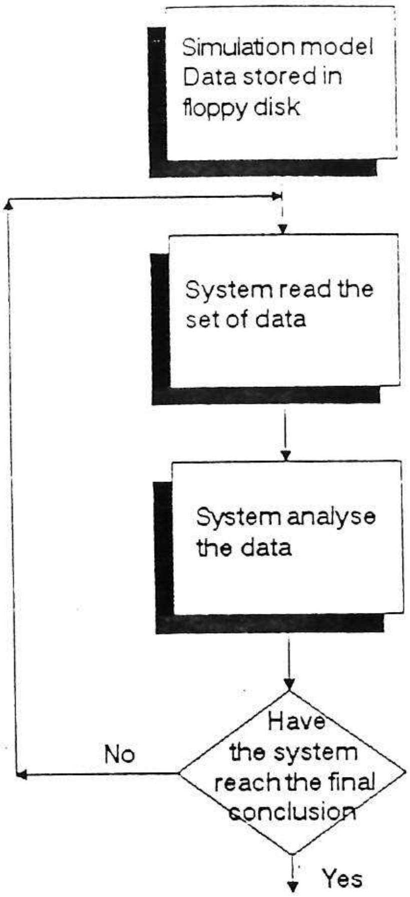

Figure 1: Entering the data to the system through the floppy disk

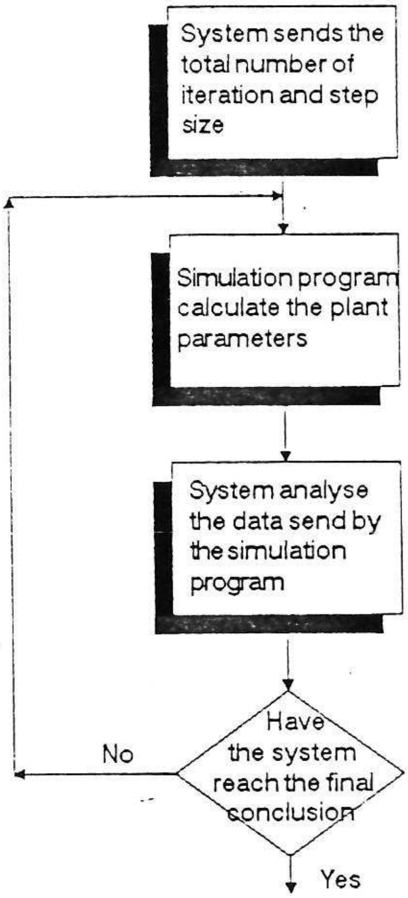

Figure 2: The communication between the system and the simulation program

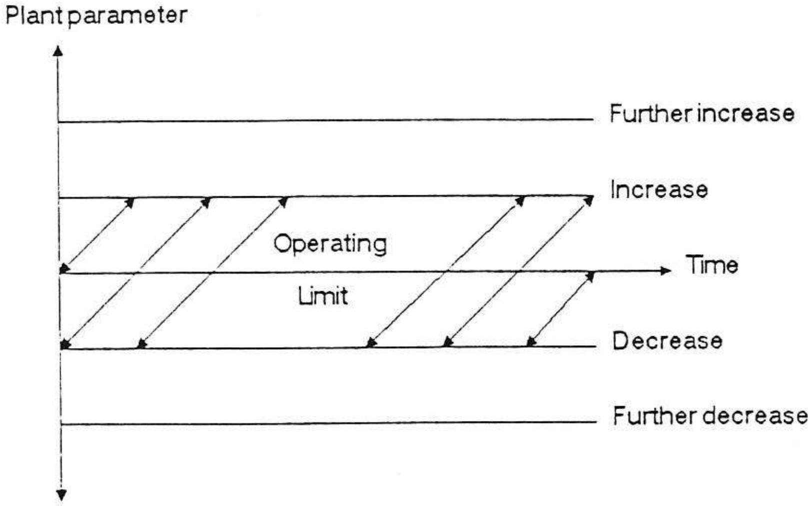

Figure 3: The deviation limit in the plant parameter

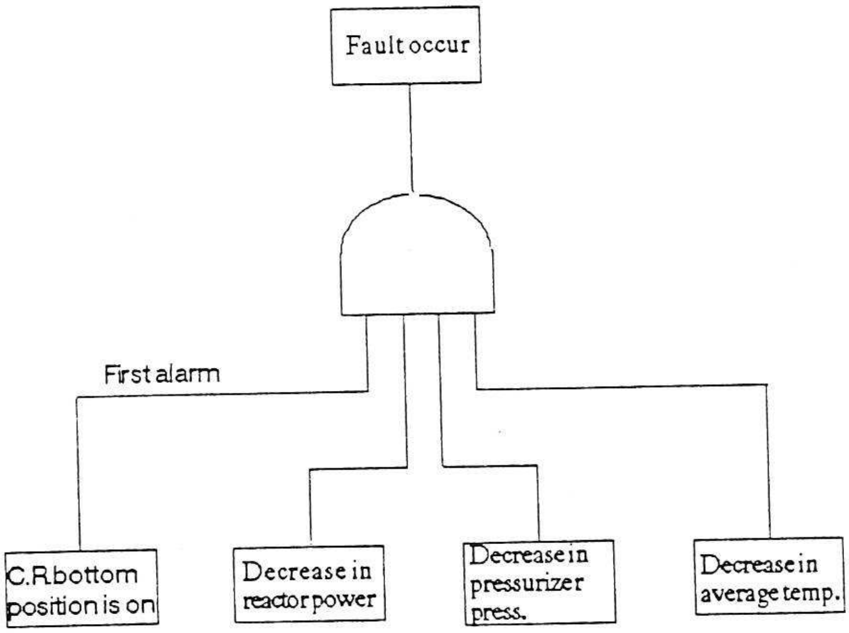

Figure 4: Fault tree for failure in the rod drop

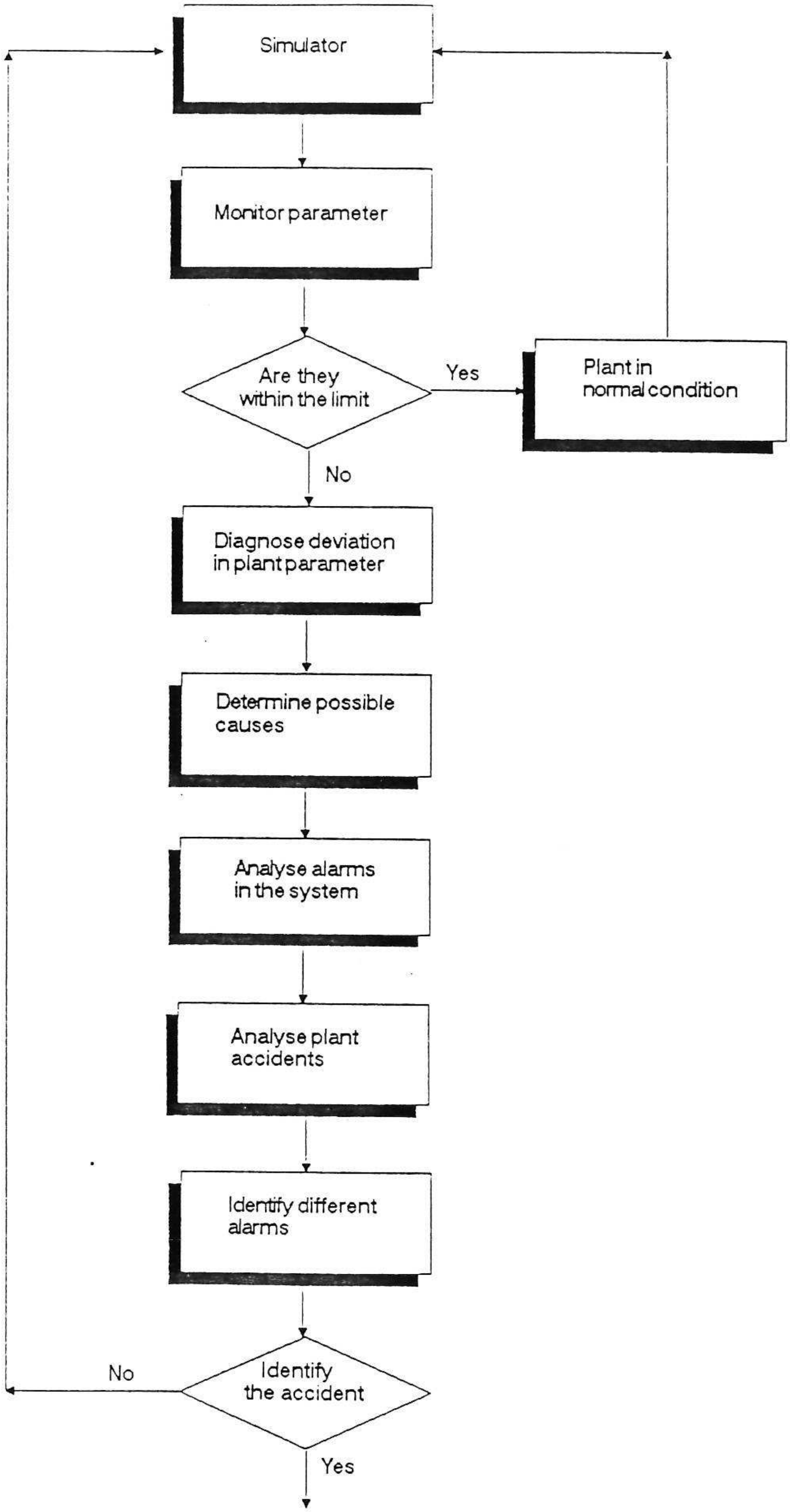

Figure 5: Configuration of the expert system

6 Expert System Performance

A simulated hypothetical accident is used to exercise and test the performance of the expert system for detecting and diagnosing any accident that might occur in the reactor. The accident is assumed to be a leakage in the feedwater piping and initiated by dropping the value of the feedwater flow at a time equal to 6 sec from the normal operating value to zero.

The symptoms for this type of accident involve decreases in steam generator water level, feedwater flow, and feedwater discharge pressure, an increase in the core coolant average temperature, and the feedwater contact breaker open. The fault tree for this type of accident is illustrated in figure 6. During normal operation the value of the steam generator water level is 126.0 (in), the value of the feedwater flow is 58.0 (lbm/sec), while the value of the feedwater discharge pressure is 511.0 (psia) and the core coolant average temperature is 553.0 ($^{\circ}F$). As described earlier, for each of these values, two value limits have been designed to indicate a decrease or increase from the normal operation limit and further increase or decrease in the value of the state variables. For the decrease in the steam generator water level, the first limit which is an alarm limit, is given equal to -1.3 (in) and the second limit which indicates a further decrease in steam generator water level is -21.4 (in). The deviation limits for the feedwater flow are equal to -0.9 and -9.0 (lbm/s) respectively while the feedwater discharge pressure is -10.0 (psia) and the core coolant average temperature is -1.0 and -1.5 ($^{\circ}F$) respectively. The limit values for a pressurized water reactor power plant are given in reference [10], however the values used in the expert system have been modified to suite the LOFT reactor which is a small scale pressurized water reactor; in addition they have been coded in the system in a way that can be altered very easily without affecting the whole program code of the system.

Table 1 illustrates the first list of output of the system which are the values of the state variables at a normal operating limit after reading the first set of data from the floppy disk, while table 2 illustrates the starting consultation between the user and the system in which the user enters the total number of iterations and the step length to call and start the simulation program, and as a result of that the system provides the user with important values of the state variables during normal operation.

The values of the state variables provided by the system at a time equal to 10 sec are shown in table 3, and it is clear that there is a decrease in the feedwater flow since its value is given equal to zero. From the information available to the system at this time, it is suggested that the accident could be a loss of a feedwater pump or blockage of feedwater flow or failure of a feedwater pipe or loss of a condenser pump, and this is shown in table 4. The deviations in the values of the state variables are analysed to supply the user with the possible causes of deviation, and table 5 analyses the possible cause of increase in the pressuriser water level and the possible cause of decrease in the feedwater flow. In addition to the information obtained from the simulated model, the system will ask the user about some important condition in the power plant to achieve the full diagnosis of any alarm or accident that might occur, and this is shown in table 6. At this time, the system is unable to diagnose any alarms or accident in the reactor, so the system will start reading the next set of data. Table 7 illustrates the values of the state variables at a time equal 20 sec while table 8 indicates the accidents that might occur in the reactor and at this time the system adds the possibility of loss of condenser or governor valve closed accidents. Tables 9 and 10 illustrate the diagnoses in the critical plant parameters and the input information provided by the user to the system respectively. The alarm and accident analyses are illustrated in table 11 in which the system provides the user with alarms for leakage in feedwater piping and partial loss of flow due to a blockage in a fuel assembly and with the belief that the accident is a leakage in the feedwater piping.

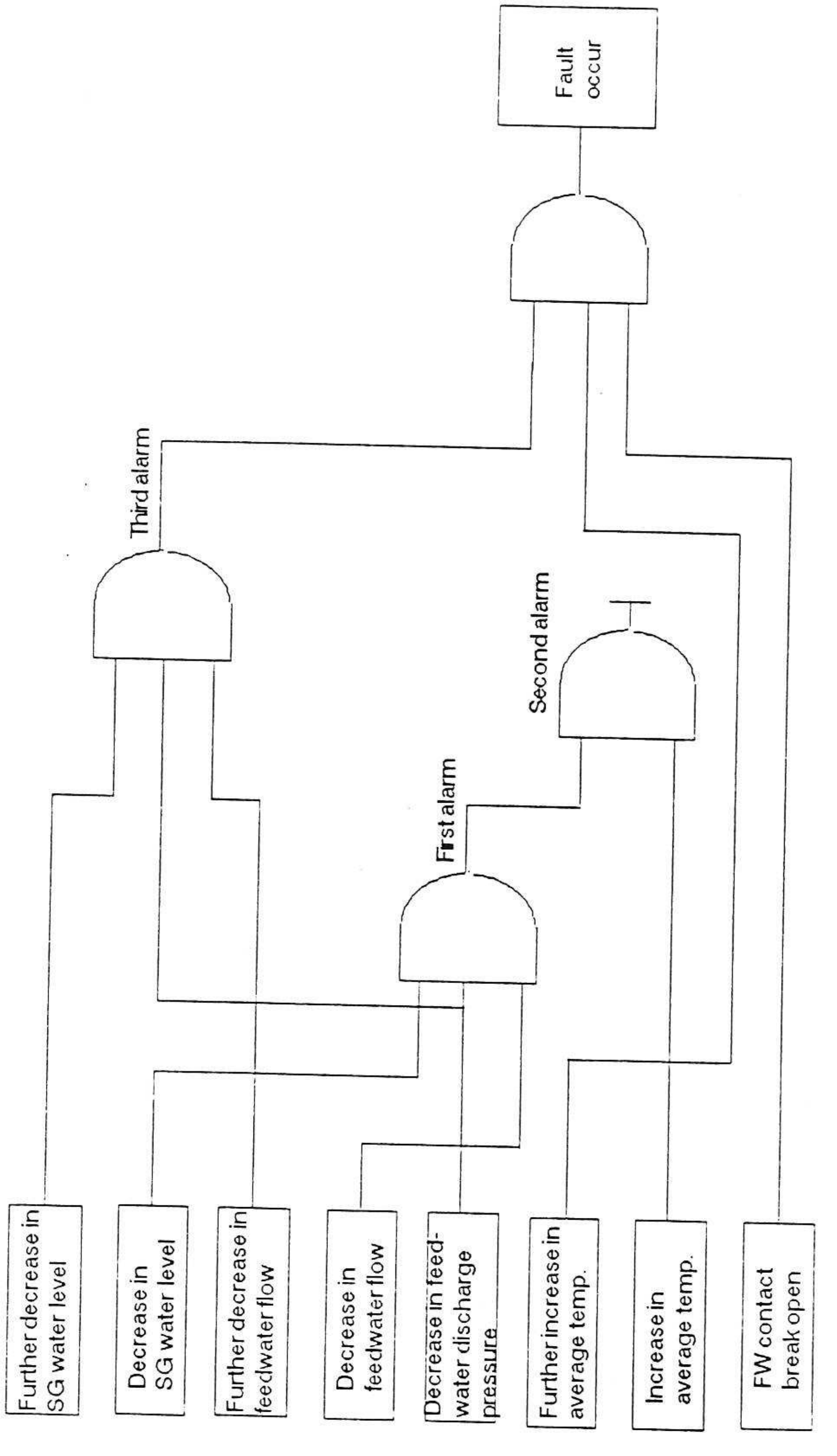

Figure 6: Fault tree for leakage in feedwater piping

At this time 0 the values for the
reactor power is 49
hot leg temperature is 571
cold leg temperature is 535
average temperature is 553
loop flow is 3655546
pressuriser pressure is 2280
pressuriser level is 47
steam flow is 58
steam generator pressure is 750
steam generator water level is 126
feedwater flow is 58
feedwater temperature is 415
primary loop pressure is 2280
press any key to continue

Table 1: Values of the state variables at time 0 sec

Enter the total number of iterations
800
Enter the step value
0.05
press any key to continue

Table 2: Communication between the user and the system

At this time 10 the values for the
reactor power is 49
hot leg temperature is 571
cold leg temperature is 535
average temperature is 553
loop flow is 3655471
pressuriser pressure is 2265
pressuriser level is 48
steam flow is 59
steam generator pressure is 763
steam generator water level is 118
feedwater flow is 0
feedwater temperature is 415
primary loop pressure is 2266
press any key to continue

Table 3: Values of the state variables at time 10.0 sec

It is found that the failure at this time is
Loss of feedwater pump or
Blockage of feedwater flow or
Failure of feedwater pipe or
Loss of a condenser pump
press any key to continue

Table 4: General suggestion of accidents

It is found that there is an increase in the pressuriser level
This is the result of one of the following
Reactor power increase
Pressuriser pressure decrease
Pressuriser water heater current rate decrease
Decrease in charging pump speed or in number of pumps in operation
It is found that there is a decrease in feedwater flow
This is the result of one of the following
Blockage of feedwater flow line
Failure of feedwater pipe
Loss of a condenser pump
Steam pressure increase
press any key to continue

Table 5: Analysis of plant parameters

Is the Control Rod (C.R.) bottom position indicator on
n
Is there disagreement between the Reactor Control Cluster Assembly (RCCA) position indicators
n
Is there loop steam generator pressure increase
n
press any key to continue

Table 6: Entering information to the system

At this time 20 the values for the
reactor power is 47
hot leg temperature is 572
cold leg temperature is 537
average temperature is 554
loop flow is 3650097
pressuriser pressure is 2264
pressuriser level is 51
steam flow is 60
steam generator pressure is 777
steam generator water level is 95
feedwater flow is 0
feedwater temperature is 416
primary loop pressure is 2265
press any key to continue

Table 7: Values of the state variables at time 20 sec

It is found that the failure at this time is
Loss of condenser vacuum or
Turbine throttle or governor valve closed
Loss of feedwater pump or
Blockage of feedwater flow or
Failure of feedwater pipe or
Loss of a condenser pump
press any key to continue

Table 8: General suggestion of accidents

It is found that there is an increase in the pressuriser level
This is the result of one of the following
Reactor power increase
Pressuriser pressure decrease
Pressuriser water heater current rate decrease
Decrease in charging pump speed or in number of pumps in operation
It is found that there is an increase in the steam generator pressure
This is the result of one of the following
Governor valve closed
Loss of condenser vacuum
Steam generator water level increase
Hot leg temperature increase
It is found that there is a decrease in the steam generator water level
This is the result of one of the following
Steam flow decrease
Feedwater flow decrease
Steam generator steam pressure increase
Hot leg temperature increase
It is found that there is a decrease in the feedwater flow
This is the result of one of the following
Blockage of feedwater flow line
Failure of feedwater pipe
Loss of a condenser pump
Steam pressure increase
press any key to continue

Table 9: Analysis of plant parameters

```
Is the C.R.bottom position indicator on
n
Is there disagreement between the RCCA position indicators
n
Is there disagreement between loop steam generator pressure
n
Is the feedwater pump contact breaker open
y
Is there a change in the temperature distribution
n
Is there a change in the power distribution
n

press any key to continue
```

Table 10: Entering information to the system

```
This is
the third alarm for leakage in feedwater piping
This is
the first alarm for a partial loss of flow due to blockage in a fuel assembly

I believe that the accident is
leakage in feedwater piping

press any key to continue
```

Table 11: Final conclusion

PART 2

7 The Cooling System of the LOFT Reactor

Four major systems can be used to cool the LOFT reactor [18] during normal and emergency conditions:

1. Primary coolant system.
2. Secondary coolant system.
3. Purification system.
4. Emergency core coolant system (ECCS).

7.1 Primary Coolant System

This is used to cool the reactor during normal operation. Two coolant pumps are used to force cooling water into the reactor vessel downcomer, into the lower plenum and then through the reactor core. Heated water will leave the reactor vessel from the upper plenum and flow to the steam generator.

7.2 Secondary Coolant System

This is used to remove heat from the primary coolant system during normal and emergency conditions. The main feedwater pumps receive water from the condensate receiver and the subcooler, after that the water will return to the steam generator and be converted to steam through the heat exchanger. The steam will change to water through the air cooled condenser, and the water will then be pumped back again to the condensate receiver.

7.3 Purification System

When the reactor is shutdown, decay heat can be removed from the primary coolant system by the purification system. Primary coolant is removed from the system at the blowdown hot leg. It passes through the regenerative heat exchanger and the non regenerative heat exchanger where the heat is removed by the Primary Component Cooling (PCC) system. The cooled water is returned to the primary system at the primary coolant pump suction.

7.4 Emergency Core Coolant System (ECCS)

This is used to remove decay heat from the primary coolant system. It is composed of three subsystems:

- High Pressure Injection System (HPIS)
- The Accumulator System
- Low Pressure Injection System (LPIS)

1. HPIS
   Is used to cool the primary coolant system at pressures above 190 psi. HPIS is initiated automatically for Loss Of Coolant Accident (LOCA) but can be implemented manually for non-LOCA conditions. In the HPIS the water source is the Borated Water Storage Tank (BWST). The HPIS pumps inject water into the primary system at the downcomer, lower plenum or cold leg injection points.

2. The Accumulator System
   This consists of a pair of pressurized tanks holding cold water. The tanks are pressurized to 600 psi and isolated from the primary coolant system by check valves. When the primary system pressure falls below 600 psi the accumulator water begins to flow to the primary system into the downcomer and the lower plenum.

3. LPIS
   This is used to cool the primary coolant system at pressures less than 190 psi. Two LPIS pumps take suction from the Borated Water Storage Tank (BWST), but during recirculation situations suction can take place from the Blowdown Suppression Tank (BST), Blowdown Hot Leg (BHL) or Pressure Reduction Sump (P R Sump). Injection to the primary system is through the cold leg, lower plenum or down injection point.

8 Response Trees

To prevent the release of radioactive materials from the fuel elements of the reactor, the fuel elements must be kept cool. This is done by forcing water through the reactor vessel, between the fuel rods, and out of the reactor vessel to a heat sink.

Nuclear power plants have more than one system designed to remove heat from the fuel elements, and during normal operation, the primary and secondary coolant systems function together to remove heat from the reactor fuel. However, other systems such as the auxiliary feedwater system, the purification system, and the emergency core coolant system (ECCS) can be used to cool the reactor during other modes of operation.

The minimum set of components used to cool the core can be called a cooling mode. Usually a cooling mode consists of five elements:

1. A heat sink
   Which is the fluid or location where the heat which has been removed from the fuel is ejected from the system.

2. Water source
   Is the tank or vessel which holds the cooling water used by the cooling mode.

3. A pump
   To remove the water from the water source and drive it towards the primary coolant system.

4. Piping route

5. Injection point
   Is the location at which the water is finally delivered to the primary coolant system.

The cooling mode element can be arranged as shown in figure 7. A cooling mode number is assigned based on the effectiveness of the cooling mode and the ease of its implementation. Cooling modes with lower mode numbers are used before modes which have higher mode numbers.

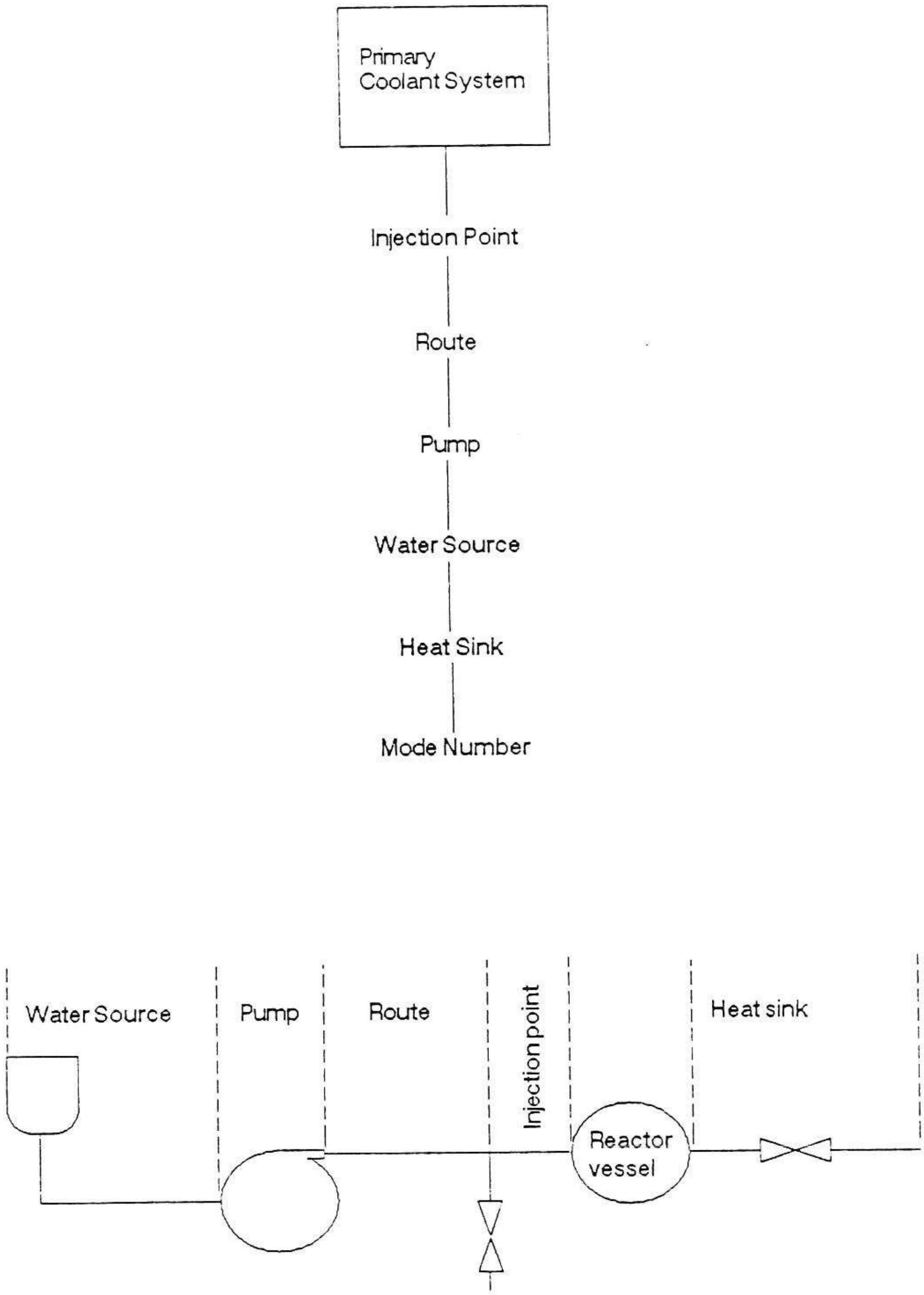

Figure 7: Cooling mode component

A response tree is simply a pictorial representation of a number of different cooling modes, where common elements of different modes are joined together near the top of the tree to show the interconnections between the modes, while at the bottom of the tree the common elements are separated to preserve the boundaries of distinct cooling modes.

A response tree is used to assist the operator in performing the following tasks:

1. Diagnose the event.
2. Determine an appropriate response.
3. Evaluate the effectiveness of the response.
4. Select a decay heat removal (DHR) mode.
5. Evaluate the effectiveness of the DHR mode.

The idea of a response tree is that, during normal operation the reactor is cooled using the primary and secondary cooling systems. If an accident happens it may be necessary to shift to another mode to cool the reactor or indeed if a component is disabled then the cooling mode which utilizes this component can not be used and in this case the cooling mode with the smallest mode number which is still available must be selected instead.

The response tree is continually updated within reactor operation when components are disabled or repaired to ensure that the most suitable cooling mode is being implemented [19,20,21]. Figure 8 illustrates the response tree for the primary, secondary and purification systems.

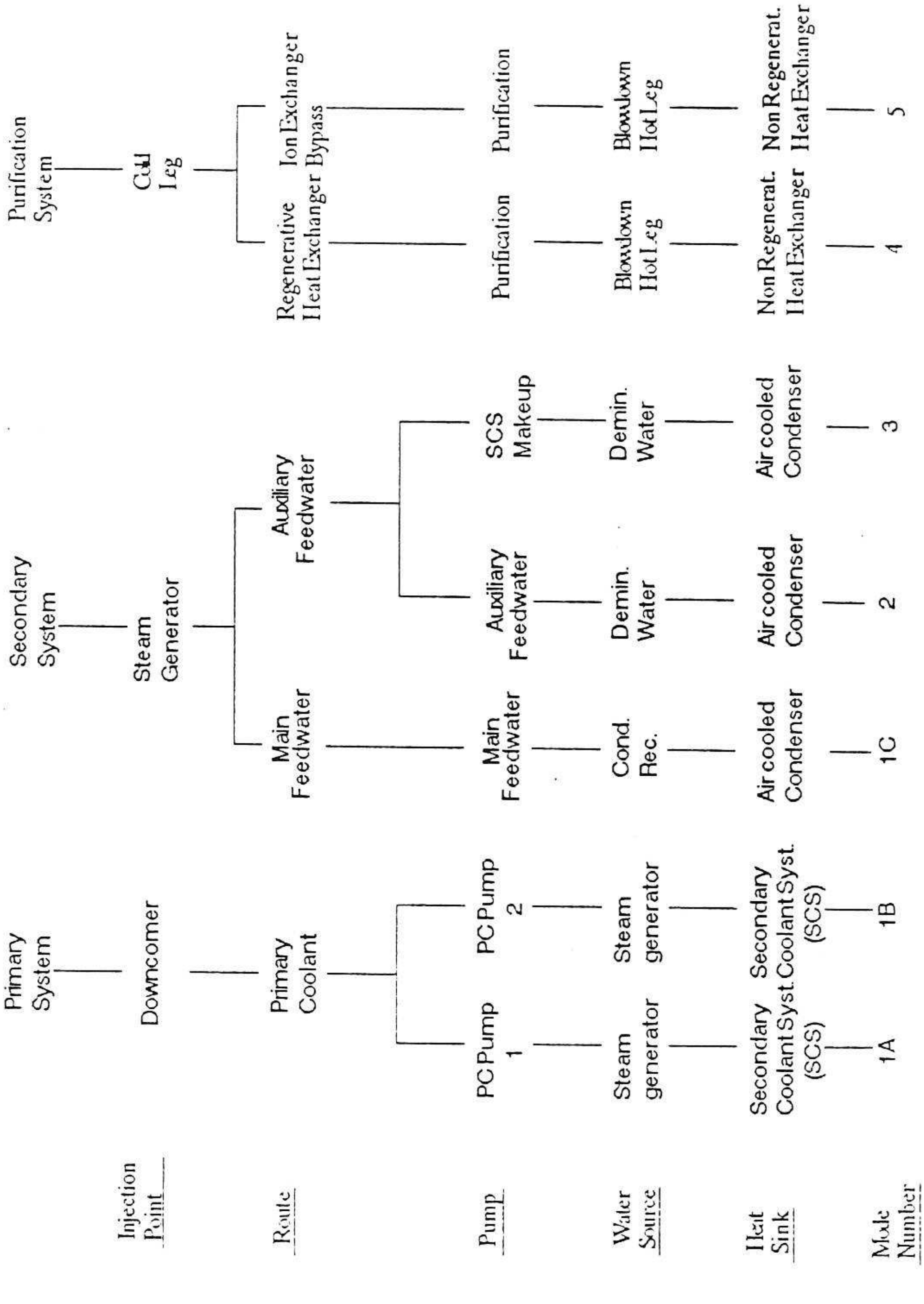

Figure 8: Response tree for the primary, secondary and purification systems

9 Description of the Expert System

The main purpose of the system which is built using both Turbo Prolog and Turbo Prolog Toolbox is to assist the nuclear reactor operator to identify quickly the appropriate cooling mode to cool the reactor. During any accidents or faults that might occur, the operator must first take action to cool the reactor core to prevent the release of radioactive material by choosing the available and appropriate cooling mode. Usually the operator chooses and switches the desired cooling mode by first gathering all the evidence about the reactor situation which is available to him, and then chooses the cooling mode according to the list of instructions. With this approach, an operator error can easily occur causing a serious accident which will damage the reactor. Knowledge based computer programs will indeed help the reactor operator to easily choose the appropriate cooling mode and avoid any operator mistake that might occur.

The expert system is built using a production rule approach with a backward chaining inference engine, so the knowledge base of the system is represented in the form of If-Then rules using the response tree described earlier, where each single cooling mode is represented by a single rule and coded in the system in such a way that the rule representing the cooling mode with the lower mode number is coded first in the system. By doing this the rule for the lower cooling mode number will fire first since the control strategy of the Turbo Prolog system is to fire the first appropriate rule. The expert system rules are given for example in the form:

Use the (purification system first line) IF
(cold leg injection (v-4060 and cv-p140-6) is working) AND
(commercial power is working) AND
(instrument air is working) AND
(regenerative heat exchanger route (cv-p140-4, v-4053, v-4056
and cp-ix-2) is OK) AND
(purification pump (cv-p140-5) is working) AND
(blowdown hot leg water source (cp-hx-2, cp-hx-1, cv-p140-8
and cv-p140-9) is OK) AND
(non-regenerative heat exchanger heat sink is working).

Expert system rules are divided into five parts according to the main five cooling systems in the LOFT reactor, where the first part is for the normal operating cooling system, the second is for the purification system, the third for the high pressure injection system, the fourth is for the accumulation system and the last is for the low pressure injection system.

The expert system is designed to first ask the operator about the reactor situation which includes five different situations of reactor operation ranging from normal operation to an emergency situation as shown in table 12. The operator has to choose one of these situations and by so doing will help the system to identify the appropriate cooling system to cool the reactor. After that, the system has to choose the right cooling mode using the knowledge base rules, and the system is able to achieve that by asking the user about the availability of the support system and the different components of the cooling modes in which the user answer will be either 'Yes' for their availability or 'No' for their disability.

The output of the system will be the selection of the appropriate cooling mode which the user has to use to cool the reactor. The system is able to provide the user with the reason for this selection, if requested, and a list of the correctly working components and the faulty components of the LOFT reactor will be shown and according to which rule in the knowledge base the system has matched to reach this conclusion.

The system has a good colour graphics environment which is achieved using the Turbo Prolog Toolbox, where the selected cooling mode will be shown graphically to the user. In the first stage the response tree for the selected cooling system will be shown with the recommended cooling mode. In the next stage the selected cooling system will be shown schematically with the chosen cooling mode highlighted in a different colour and the faulty equipment within the cooling system will be identified.

10 System Performance

A hypothetical accident has been assumed to occur in the LOFT reactor, to test and examine the performance of the expert system in selecting the appropriate cooling mode to cool the reactor. The accident is assumed to be a small break Loss Of Coolant Accident (LOCA) in which HPIS pump A fails to start.

According to this accident the symptoms received by the operator will include, decreasing reactor vessel pressure with no pump outlet flow detected . From these symptoms the operator will choose number 3 from the list given in table 12. After choosing the number the program will start the consultation by asking the operator some questions concerning the availability of the cooling mode equipment in which the user reply will be either 'y' or 'n' as shown in table 13. The system output will include the recommended cooling mode to be used to cool the reactor and according to this accident the system will select the first automatic downcomer line to cool the reactor. The system will also provide the user with the reason for this selection which involves a list of correctly working equipment and the list of the faulty equipment in the cooling mode, and according to rule number 10 in the knowledge base the system has reached this result as illustrated in table 14.

The system will be able to help the operator to interpret the final conclusion graphically. Figure 9 illustrates the response tree for the high pressure injection system, and the selected cooling system. The cooling system which is used to cool the reactor is shown schematically with the selected cooling mode identified as shown in figure 10. Finally, the faulty equipment in the selected cooling system is identified as illustrated in figure 11.

11 Conclusion

In the first part it is clear that the system at time equal to 20 sec is able to identify the accident, which starts at 6 sec, caused by dropping the feedwater flow to zero. The system has been designed to start the consultation and undertake the analysis every 5 sec, however this could be easily altered as desired by the user. The diagnostic ability of this system for alarms and accidents in nuclear reactor plant is very encouraging and the deviation in the plant parameters can be detected easily. In addition, the system has shown a good ability to diagnose deviation in the plant parameters. The knowledge base of the system can be modified easily to include additional rules for other accidents that can be added to the system without affecting the whole coding program style of the system.

The second system performance in choosing the appropriate cooling mode is also very promising. The response tree methodology is a useful approach for describing the cooling system of the nuclear reactor, and the graphical environment gives considerable assistance to the operator in choosing the appropriate cooling mode and avoids any possible misunderstanding of the system's final conclusion.

RESPONSE TREE	
Select one of these numbers	
Is the system in:-	
1	normal operation (operating pressure = 15.5 mpa)
2	reactor shut down but not emergency operation
3	emergency operation with reactor shut down and operating pressure = 11mpa
4	emergency operation with reactor shut down and operating pressure = 4.1 to 1.4mpa
5	emergency operation with reactor shut down and operating pressure less than 1.3mpa
3	

Table 12: The situation of the system

RESPONSE TREE
Type y for yes and n for no
Is the cold leg v2169 cvp12063 working? y Is the downcomer cvp120110 cvp120109 working? y Is the vital power A working? y Is the vital power B working? y Is the normal a v6474 v3042 cvp128117 v2253 working? y Is the normal b v5134 v2255 cvp128116 v3043 working? y Is the hpis a pump v2222 cvp12811 working? n Is the hpis b pump v2224 cvp12814 working? y Is the bwst v5225 working? y Is the cold water working? y

Table 13: A dialogue between the user and the expert system

RESPONSE TREE
It is better to use hpis through the first automatic downcomer line Would you like to know why we select this response (y/n)? We select this response because :- Incorrectly is the hpis a pump v2222 cvp12811 working Correct is the cold water working Correct is the bwst v5225 working Correct is the hpis b pump v2224 cvp12814 working Correct is the normal b v5134 cvp128116 v2255 v3043 working Correct is the normal a v6474 cvp128117 v3042 v2253 working Correct is the vital power b working Correct is the vital power a working Correct is the downcomer cvp120110 cvp120109 working Correct is the cold leg v2169 cvp12063 working The system matches the above result with rule no. 10

Table 14: The final result of the system

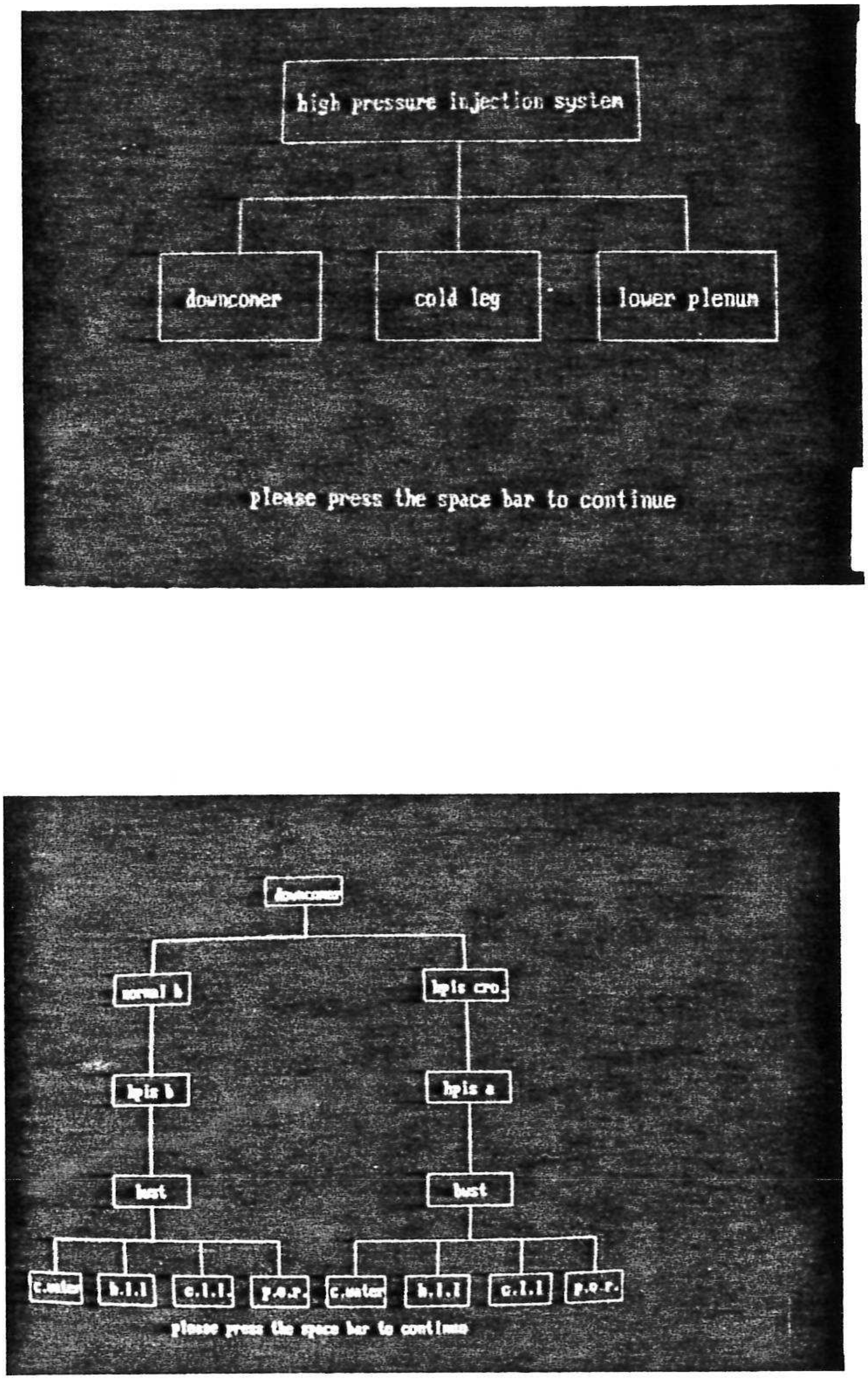

Figure 9: The response tree of the selected cooling system with the cooling mode

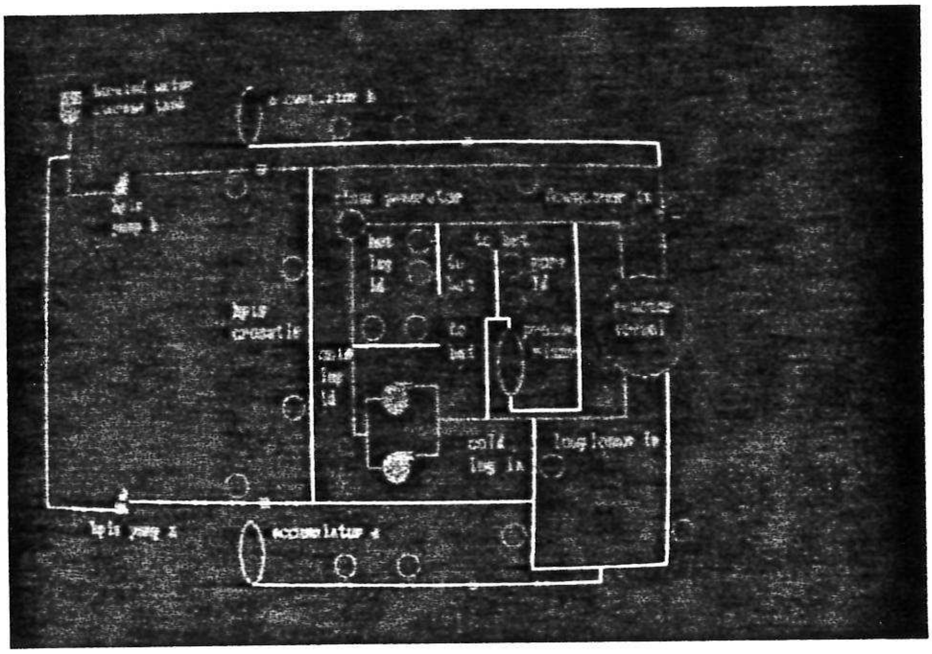

Figure 10: The cooling system with the recommended cooling mode

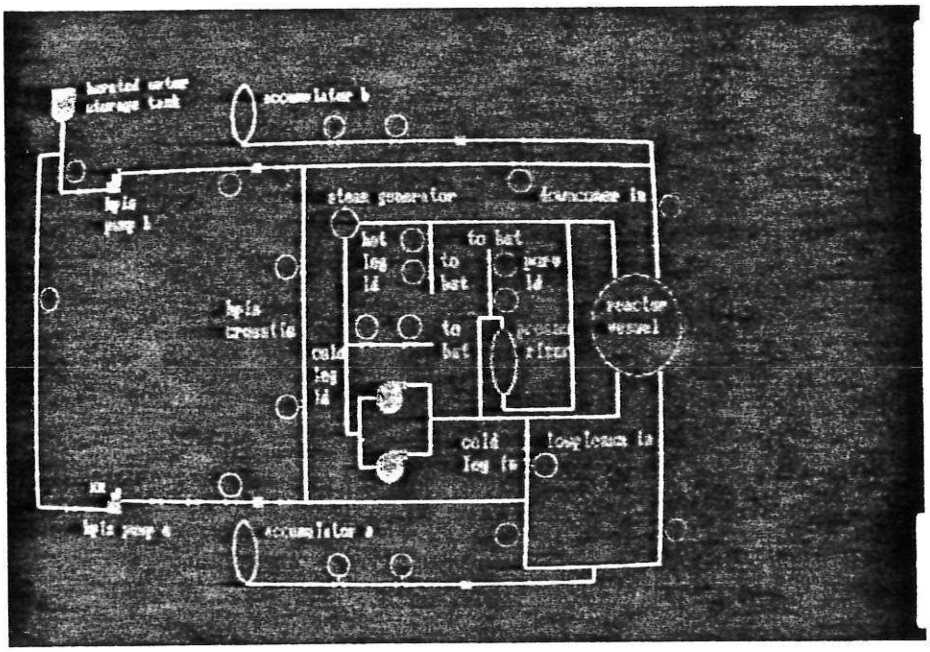

Figure 11: The cooling system with the faulty equipment

References

[1] Linkens, D. (1990). *AI in Control Systems Engineering.* The Knowledge Engineering Review, Vol. 5, N0. 3, pp. 1-34.

[2] Pau, L. (1986). *Survey of Expert Systems for Fault Detection, Test Generation and Maintenance.* Expert Systems, Vol. 3, No. 2, pp. 100-109.

[3] Scherer, W. and White, C. (1987). *A Survey of Expert Systems for Equipment Maintenance and Diagnostics, in Fault and Reliability Knowledge Based and Other Approaches. Eds. Singh, M., Hindi, K., Schmidt, G. and Tzaftestas, S.* Pergamon Press.

[4] Bernard, J. and Washio, T. (1988). *An Examination of Expert Systems Activates Within the Nuclear Industry.* Transactions of the American Nuclear Society, Vol. 57, pp. 240-241.

[5] Leech, W. and Casadei, A. (1989). *Overview of Expert Systems Applications in Westinghouse Nuclear Fuel Activities.* Transactions of the American Nuclear Society, Vol. 59, pp. 60-61.

[6] Majumdar, D. (1989). *Artificial Intelligence Applications in the Nuclear Industry: an International View.* Nuclear Engineering and Design, Vol. 113, No. 2, pp. 181-193.

[7] Karwatzki. J. (1987). *Expert System in Engineering.* Computer-Aided Engineering Journal, Vol. 4, No. 2, pp. 99-102.

[8] Sudduth, A. (1989). *Knowledge-Based Systems in Process Fault Diagnosis.* Nuclear Engineering and Design, Vol. 113, No. 2, pp. 195-209.

[9] Cain, D. (1989). *Artificial Intelligence Applications in Accident Management.* Nuclear Engineering and Design, Vol. 113, No. 2, pp. 251-7.

[10] Ashmawi, EL. (1972). *Use of On-Line Computers for Malfunction Diagnosis and Fault Detection in Nuclear Power Plant.* PH.D.Thesis, North Carolina State University.

[11] Hassberger, J. (1986). *Simulation-Based Expert System for Nuclear Power Plant Diagnostics.* PH.D. Dissertation, University of Michigan.

[12] Jackson, P. (1986). *Introduction to Expert Systems.* Addison-Wesley.

[13] Borland International, Inc. (1986). *Turbo Prolog Owner's Handbook.* Borland International.

[14] Borland International, Inc. (1986). *Turbo Prolog Toolbox.* Borland International.

[15] Alty, J. and Coombs, M. (1984). *Expert Systems Concepts and Examples.* National Computing Centre.

[16] Tylee, J. (1980). *Low Order Model of the Loss Of Fluid Test (LOFT) Reactor Plant for Use in Kalman Filter Based Optimal Simulation.* Proceedings of the 4th Power Plant Dynamics, Control and Testing Symposium, Gatlinburg, pp. 1-31.

[17] Tylee, J. (1980). *Low Order Model of the Loss Of Fluid Test (LOFT) Reactor Plant for Use in Kalman Filter Based Optimal Estimators.* EGG-2006, EG & G Idaho Falls, Idaho.

[18] Aerojet Nuclear Company (1974). *LOFT Integral Test System Final Safety Analysis report.* ANCR-1142, Idaho Operations Office, Idaho.

[19] Nelson, W.R. (1980). *Response Trees for Detection, Diagnosis and Treatment of Emergency Conditions at the LOFT Facility.* Masters Thesis, University of Washnighton, USA.

[20] Nelson, W. (1982). *Reactor: An Expert System for Diagnosis of Nuclear Reactor Accidents.* Proceedings of the International Conference on Artificial Intelligence, Pittsburgh, PA, pp. 296-301.

[21] Nelson, W.R. (1984). *Response Trees and Expert Systems for Nuclear Reactor Operations.* NUREG/CR-3631, EGG-2293, EG and G Idaho, Inc., Idaho Falls, USA.