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on
OPERATOR TRAINING
AND QUALIFICATIONS

Charlotte, N.C., United States
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PROCEEDINGS

Volume 2

Committee on the Safety of Nuclear Installations
OECD Nuclear Energy Agency
38, boulevard Suchet
75016 Paris
France

Nuclear Energy Agency
of the
Organisation for Economic Co-operation

The Nuclear Energy Agency (NEA) is a specialised Agency of the Organisation for Economic Co-operation and Development (OECD) in Paris. The NEA committee on the safety of Nuclear Installations (CSNI) is an international committee made up of scientists and engineers who have responsibilities for nuclear safety research and nuclear licensing. The Committee was set up in 1973 to develop and co-ordinate the Nuclear Energy Agency's work in nuclear safety matters, replacing the former Committee on Reactor Safety Technology (CREST) with its more limited scope.

The Committee's purpose is to foster international co-operation in nuclear safety amongst the OECD Member countries. This is done essentially by:

i. exchanging information about progesss in safety research and regulatory matters in the different countries, and maintaining banks of specific data; these arrangements are of immediate benefit to the countries concerned.

ii. setting up working goups of task forces and arranging specialist meetings, in order to implement co-operation on specific subjects, and establishing international projects; the output of the study groups and meetings goes to enrich the data base available to national regulatory authorities and to the scientific community at large. If it reveals substantial gaps in knowledge or differences between national practices, the Committee may recommend that a unified approach be adopted to the problems involved. The aim here is to minimise differences and to achieve an international consensus wherever possible.

The main CSNI activities cover particular aspects of safety research relative to water reactors and fast reactors; probabilistic assessment and reliability analysis, especially with regard to rare events; siting research; fuel cycle safety research; various safety aspects of steel components in nuclear installations; and a number of specific exchanges of information.

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The Institute of Nuclear Power Operations (INPO) is a non-profit, independent organization created in 1979 by the nuclear utility industry. INPO is dedicated to promoting safety in operations in nuclear power plants.

Every U.S. utility with an operating license, a construction permit or a limited work authorization for a nuclear power plant is a member of the Institute. INPO's membership is broadened further with the inclusion of utilities that are co-owners of nuclear power plants. Participation is also extended to non-U.S. nuclear organizations and to domestic nuclear suppliers and engineering firms.

INPO was founded to assist nuclear utilities in achieving a high level of excellence in safety of nuclear power operations. Offices are located in Atlanta, Georgia.

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The CSNI Specialist Meeting on Operator Training and Qualifications Proceedings have been printed in two volumes. Volume I contains the conference agenda, introductory remarks, and proceedings of Sessions I and II. Volume II contains proceedings of Sessions III-VI, the Program Group, and the List of Participants.

Additional copies may be obtained by writing the Institute of Nuclear Power Operations, 1820 Water Place, Atlanta, Georgia 30339.

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SUMMARY

SESSION III - SIMULATORS

CHAIRMAN: P. B. MYERSCOUGH

Summary of Principal Points of Interest in the Papers Presented and Subsequent Discussion

(by P. B. Myerscough)

Mr. Blomberg's description of the development and use of a compact simulator illustrated the useful role which this type of machine could play in the overall training program. It is being used to supplement both the classroom training and the essential full-scope simulator. Particular interest was shown in its application to the training of staff other than plant operators, its use at the power station, and its adaptability for use by operating staff without its necessity for an instructor. Although the machine has a special merit in preparing engineers for training on a complete, full-use simulator, it appears currently to have support only in Sweden and at one training center in the United States.

Dr. Lindauer discussed the problems involved in updating a full-scope simulator for PWR training. This was a paper of particular interest to a licensing authority which may be contemplating specifying training simulator design initiative. The methods used in comparing the performance of the present simulator with that of the plant was of particular interest, using operational experience and accident analysis. Emphasis was placed upon the importance of introducing only those improvements in the simulator which were necessary for training purposes, a point which was appreciated in the subsequent discussion.

Comment was also made on the fact that the study illustrated the useful cooperation between trainers of plant operators and the difficulties in simulating accurately multiple malfunctions which have not yet occurred at the plant.

The consideration of the major issues involved in specifying a training simulator, by Mr. Grant, raised several fundamental considerations and probably could have had a greater impact if presented as the initial paper in the session. The discussion following the paper clearly indicated how essential it is that the design of subsequent use of the simulator should both form an integral part of the training program and the design of the program itself following a careful consideration of the role of the operator. The approach suggested by Mr. Grant could be useful (perhaps essential) for the NRC in specifying a simulator that would be accepted for licensing examination.

The final paper presented by Mr. Luffey was a detailed consideration of the design of a simulator improvement to rectify the limited capability of current machines to simulate two-phase flow conditions in the coolant system. The resulting modular machine will undoubtedly assist training in this vital area of PWR operation, and the comparison of the transient results with those obtained with the "St. Lucie" type of cooldown was of interest to operators. However, the lack of questioning following the paper probably indicated that this was not the sort of meeting to be considering detailed hardware and software design.

IMPLEMENTATION OF A BASIC PRINCIPLE SIMULATOR

J. F. DE GREEF, SCK/CEN BELGIUM

0. ABSTRACT

During the last twenty years, the Belgian Nuclear Energy Centre (SCK-CEN) gained a lot of experience in mathematical modelling and simulation of nuclear reactor power plant operation. The framework in which these activities took place, was mainly concerned with security analysis of these devices and with training operation personnel of these plants. In the last years a fourth generation hybrid computer including an AD/FIVE analog computer of Applied Dynamics Inc. (ADI) and a PDP-11/34 digital computer of Digital Equipment Corporation (DEC) were installed. Furthermore, a home made operator's desk simulating the elements of the control room of a nuclear power plant completes the computer system for real-time training purposes.

The described equipment is typically that of a Basic Principle Simulator, a device that quickly and effectively imparts the basic concepts of a nuclear power plant operation. Cost and concept are quite different from that of the better known Full Scope Simulator. Also the purpose and the implementation of a basic simulator is quite different. In a full training package the full scope simulator is merely involved in the sequence of training practice on the particular plant, whereas the basic simulator is inserted in the sequence of training insight in the basic behaviour of a plant in general. The basic simulator is consequently independent from a particular plant and has therefore advantages to cope with particular training needs.

The described mathematical model is also typically that of a basic simulator. Its generation has been kept simple and limited to the underlying fundamental physical laws. The advantages of this simple model are in the first place the capability for the computer to calculate in real time (this is mandatory due to the human being in the loop) and in the second place the ability for the operator to gain physical experience in fundamental plant behaviour.

The described exercises reflect the most elemental aspects of neutron kinetics, thermics and hydraulics. They initiate several types of perturbations, while the trainee observes the indicators and reacts on the operator's desk. Exercises are divided into four groups :

- at zero power operation	1. Loading and start-up.
- at full power operation	2. Balances of reactivity.
	3. Dynamical behaviour (automatic control).
	4. Dynamical behaviour (manual control).

Experience gained in operator training by a basic simulator led to the conclusion that this type of training can be a very useful link between theoretical teaching and practice training in a full training package.

I. INTRODUCTION

In the set of nuclear power plants simulators, four main groups may be defined as an attempt to classify simulators according to their typical implementation characteristics.

In the first group, there are the so called full scope simulators on which all practice training can be performed, including incidents and accidents. The operator's desk of this type of simulator is necessarily a replica of the control room of the particular plant to which the simulator is associated. This type of simulator is therefore strongly a dedicated one. The cost of such a simulator lies in the order of ten million of USA dollars. Authorities may impose for security reasons upon each particular plant the implementation of its particular full scope simulator in order to train and retrain the own operating personnel.

In a second group, we have what is called the functional simulator, a smaller but more general purpose type. Each particular plant can be simulated on this type of simulator, but in its functional behaviour only. Besides the normal functional operations, incidents and accidents may nevertheless be included depending upon the sophistication of the available software. This software makes the simulator to be general purpose : each plant has its particular software package on it. The cost lies in the order of one million of USA dollars. Costs for software development for each particular plant have to be considered separately. Managers of grouped plants of different types may acquire such a device for commun use.

In a third group, we find the basic principle simulator which simulates a plant only in its basic behaviour in normal operation. This device is still smaller but general purpose, as well in its software as in its hardware. Particularities may nevertheless be introduced by varying parameters in the available software. Also, according to the degree of sophistication of the available software package, some aspects of particuliar functional behaviour of a plant can be emphasized. The cost of a basic simulator lies in the order of one half a million of USA dollars. Training Centers, Institutes for nuclear energy promotion, Universities and the like, may acquire this type of simulator as a common device for the training of beginners, students, etc., non excluding the retraining of professionals and the solution by simulation of particular problems in the field of nuclear safety studies.

In a fourth group are situated the devices known as CAE (Computer Aided Education) devices which, as for any other application, may be implemented for training operators as a particular application to that device. No longer real time operation is however available. The access is in a conversational mode in a question - answer relation. The price lies in the order of ten thousand of USA dollars. Acquisition is merely a matter of individuality and personal teaching.

In a full training package, two main sequences may be defined. In the first sequence we find theory training, whereas practice training can be found in a second sequence. It is clear that the training in practizing the operation of a particular plant can only be done on the associated full scope simulator. However theory teaching and training can be performed independently from a particular plant and more general purpose means can be implemented. Here, the general purpose basic simulator can play an important role for the trainee in getting insight and feeling in the plant behaviour after theory teaching and before practice training. Moreover, notwithstand its simplicity, this simulator still keeps the real time facility. As a man in the loop the operator not only experiences insight in, but also physical feeling with the fundamental plant behaviour.

In a real time simulation, the main problem is not directly the size nor the sophistication of the simulation model, but the ability for the computer to operate in real time. This is mandatory, due to the man in the loop with the computer. No execution delay is admissible. The man in the loop is asking for a respons from the computer compatible with the speeds of its onw movements and reflexes. Like the real plant operation is conceived to permit the operator to move in its own world, so the simulator has to be conceived in the same way. Formally, a simulation works in real time if one cycle of the corresponding comuter program is executed in a time smaller than the particular value assigned to parameter Δt in the program. A cycle is one execution of the program to calculate the variable's updating from t_i to t_{i+1}, where $\Delta t = t_{i+1} - t_i$. In order to satisfy this real time constrain, care must be taken when selecting a computer system for real-time simulations. It is the merit of an hybrid computer system composed of an analog part and a digital part to meet this particular requerement.

II. SCOPE

An AD/FIVE analog computer has been linked to a PDP 11/34 digital computer to form the Hybrid Computer System AD-511. The hybrid software package available on this system allows the programming on it of real-time simulations of industrial processes. A hybrid computer program has been developped to simulate on the AD-511 system the operation of a PWR plant at full power. A particular version of the program has been derived to simulate also but separately the operation of the plant at zero power. Together with the adjoined operator's desk, the programmed AD-511 computer constitues a BASIC PRINCIPLE SIMULATOR for the purpose of training PWR plants operating personnel.

In these simulations an attempt is made to display the most relevant characteristics of the dynamical behaviour of a PWR plant at full power and at zero power, in the particular situation where the plant is operated by a human being.

At full power operation, to operate the simulated plant, the operator can deliberately act upon the following parameters by means of a series of keys on the operator's desk :

- the up and down movement of the control rods,
- the injection and dilution of boron,
- the mass flow in the primary circuit,
- the feed water flow in the secundary circuit,
- the steam dump in the secundary circuit,
- the rotation speed of the electricity generating group,
- the power demand by the grid,
- the synchronization between the group and the grid.

Indicators on the operator's desk allow the operator to keep track of the following parameters :

- For the reactivity balance :
  - the boron and the xenon concentration,
  - the fuel and coolant temperatures,
  - the control rod position.
- For the neutron kinetics :

- the nuclear power,
- the period.

- For the primary circuit :
  - the heat and mass flow,
  - the mean temperature,
  - the primary pressure.

- For the steam generation :
  - the water level,
  - the heat power,
  - the saturated steam temperature.

- For the turbine :
  - the steam flow,
  - the steam dump flow,
  - the feed water flow.

- For the electricity generator :
  - the rotation speed,
  - the electrical power.

All these and other parameters of the simulated model can be displayed individually on a monitor by simple use of the standard read-out-selector of the AD-511 system.

To perform a training, several types of perturbations may be initiated on the simulated plant, while the trainee observes the indicators and reacts accordingly at the operator's desk. During the run of an exercise, the simulation can be accelerated to gain time, or be freezed to permit discussion and explanation, or be reset to the initial state for re-run or initiation of another run. Also, parameters are registered for display, graphical output and documentation.

The simulated plant has capabilities to perform the following typical perturbations, starting at an initial steady state :

- At nominal power, to modify the position of the control rods or the concentration of boron.
- At nominal power, to decrease the demand power while in synchronism with the grid.
- At other power levels, to rise to nominal power demand.
- At nominal power, to cause a load drop.

- At nominal power, to cause an isolation from the grid.
- At low power, to bring the generating group to synchronism and to connect the group to the grid.
- At nominal power, to decrease the flow in the primary circuit.

For three parameters, operations consequent to the perturbations initiated, can be executed either automatically, or manually. These three parameters are : the position of the control rods, the feed water flow and the steam dump.

A glance at the simulation model described in the next chapter will reveal more details on the scope of simulation at full power.

At zero power, to operate the simulated plant, the operator acts upon the following parameters by means of two keys on the operator's desk :

- the up and down movement of the control rods,
- the injection and dilution of boron.

Indicators on the operator's desk allow the operator to keep track of the following parameters :

- for the reactivity balance : - the boron concentration,
  - the position of the rods,
- for the neutron kinetics : - the neutron flux,
  - the neutron flux decade.

To initiate and to perform start-up training exercises, several types of initial steady states can be introduced by means of a set of parapeter input values :

- the neutron source,
- the invested reactivity,
- the initial position of the rods and their antireactivity,
- the concentration of boron and its antireactivity,
- the temperature in the core and its negative temperature coefficients.

The neutron flux source and the subcritical reactivity balance will automatically establish the initial neutron flux level and the corresponding flux decade (log flux).

Subsequent to the establishment of a subcritical steady state, the trainee can, while observing the indicators and a graphical display, proceed by means of either the rod key or the boron key, or both :

- to reach criticality without passing alarm,
- to reach power level,
- to stabilize at any subcritical flux level.

As for the simulation at full power, during the run of an exercise the simulation can be accelerated to gain time, or be frozen to permit discussion and explanation, or be reset to the initial state for re-run or initiation of another run.

III. THE MODEL

In this chapter, the underlying equations of the mathematical model describing the behaviour of the real plant, are written.

a. Reactor kinetics

$$\dot{c}_i = \lambda_i Q_n - \lambda_i c_i \qquad \text{(Delayed neutrons)}$$

$$Q_n = \frac{\delta k}{\beta} Q_n + \Sigma \frac{\beta_i}{\beta} c_i \qquad \text{(Nuclear power)}$$

b. Reactivity and period

$$\delta k = \delta k_o + \delta k_d - \alpha_f T_f - \alpha_c T_c$$
$$- k_b C_b - k_x x_e \qquad \text{(Reactivity)}$$

$$\dot{D}_e = 26 \frac{\lambda \delta k}{\beta - \delta k} \qquad \text{(Period)}$$

c. Control rods

$$\dot{D}_i = \alpha_i (T_{mc} - T_m) \text{ or } \pm \alpha_m$$

$$D_p = \alpha_p (T_{mc} - T_m)$$

$$D_d = \alpha_d (Q_s - Q_n)$$

$$D = D_i + D_p + D_d \qquad \text{(PID controller)}$$

$$\delta k_d = k_d v_f (D - D_f) \qquad \text{(Rods antireactivity)}$$

$$v_f = \frac{0.008 D^2 - 250}{D - D_f} \qquad D \leq 125$$

$$v_f = 0.008 (D_f - D) \qquad D \geq 125$$

d. Xenon and boron

$$\dot{i} = \lambda_i Q_n - \lambda_i i \qquad \text{(Iodine concentration)}$$

$$\dot{x}_e = \frac{\phi_o \gamma_x \Sigma_f}{X_{eo}} Q_n + \frac{\lambda_i I_o}{X_{eo}} i$$

$$- \lambda_x x_e - \sigma_x {}_o x_e Q_n \qquad \text{(Xenon concentration)}$$

$$\dot{C}_b = \pm k_{cb} \qquad \text{(Boron)}$$

e. Primary circuit

$$(MC)_f \dot{T}_f = P_{no} Q_n - (UA)_{f-c}(T_f - T_c) \quad \text{(Fuel temperature)}$$

$$(MC)_c \dot{T}_c = (UA)_{f-c} (T_f - T_c) - WC_o.F (T_{co}-T_{cin}) \quad \text{(Coolant temperature)}$$

$$T_{co} = 2 T_c - T_{cin} \quad \text{(Core outlet temperature)}$$

$$(MC)_{up} \dot{T}_{ro} = WC_o.F (T_{co} - T_{ro}) \quad \text{(Reactor outlet temperature)}$$

$$(MC)_{hl} \dot{T}_{bin} = WC_o.F (T_{ro} - T_{bin}) \quad \text{(Hot leg outlet temperature)}$$

$$(MC)_b \dot{T}_b = WC_o.F (T_{bin} - T_{bo}) - (UA)_{b-sw}(T_b-T_{sw}) \quad \text{(Steam generator primary temperature)}$$

$$T_{bo} = 2 T_b - T_{bin} \quad \text{(Cold leg inlet temperature)}$$

$$(MC)_{cl} \dot{T}_{ri} = WC_o.F = (T_{bo}-T_{ri}) \quad \text{(Reactor inlet temperature)}$$

$$(MC)_{lp} \dot{T}_{cin} = WC_o.F (T_{bo} - T_{ri}) \quad \text{(Core inlet temperature)}$$

$$T_m = 0.5 (T_{ro} + T_{ri}) \quad \text{(Primary mean temperature)}$$

$$\dot{F} = \pm K_f \quad \text{(Primary heat flow factor)}$$

f. Pressurizer

$$w_{sg} = \frac{\alpha V_o}{v_{swp}} \dot{T}_m \quad \text{(Surge)}$$

$$E_p = M_{sp} h_{sp} + M_{swp} h_{swp}$$

$$\dot{E}_p = P_h - (h_{sp} - h_{spr}) w_{spr} \quad \text{(Energie balance)}$$

$$M_p = M_{sp} + M_{swp}$$

$$\dot{M}_p = w_{sg} - w_{sp} + w_c \quad \text{(Mass balance)}$$

$$V_p = M_{sp} v_{sp} + M_{swp} v_{swp} \quad \text{(Volume balance)}$$

$$H_p = \frac{1}{(\rho S)_p} M_{swp} \quad \text{(Water level)}$$

$$H_{pc} = A_p + B_p Q_s \quad \text{(Water level set point)}$$

$$\tau_p \dot{w}_c = (H_{pc} - H_p) - k_p w_c \qquad \text{(Water level control)}$$

g. Steam generator

$$E = M_s h_s + M_{sw} h_{sw}$$

$$\dot{E} = (UA)_{b-sw} (T_b - T_{sw}) - P_{so} Q_s \qquad \text{(Energy balance)}$$

$$T_{sw} = a_p P + b_p \qquad \text{(State equation)}$$

$$M = M_s + M_{sw}$$

$$\dot{M} = W_{fw} - W_s \qquad \text{(Mass balance)}$$

$$V = M_s v_s + M_{sw} v_{sw} \qquad \text{(Volume balance)}$$

$$\tau_b \dot{H}_b = k_{ks} W_s - k_{bw} W_{fw} - k_b H_b \qquad \text{(Steam bulb layer)}$$

$$H = \frac{1}{\rho S} M_{sw} + H_b \qquad \text{(Water level)}$$

$$\tau_{fw} \dot{W}_{fw} = G_{fw} (H_c - H) + W_s - W_{fw} \qquad \text{(Water level control)}$$

$$P_s Q_s = h_s W_s - h_{fw} W_{fw} \qquad \text{(Steam power)}$$

h. Turbine

$$W_d = k_d O_{vd} P \qquad \text{(Steam dump flow)}$$

$$W_h = k_h O_{vh} (P - P_r) \qquad \text{(High pressure steam flow)}$$

$$W_s = W_h + W_d \qquad \text{(Total steam flow)}$$

$$\tau_r \dot{P}_r = W_h - W_l \qquad \text{(Pressure reheater)}$$

$$W_l = k_l O_{vl} P_r \qquad \text{(Low pressure steam flow)}$$

$$\tau_{hc} \dot{C}_h = k_{lc} W_h - C_h \qquad \text{(High pressure torque)}$$

$$\tau_{bc} \dot{C}_l = k_{lc} W_l - C_l \qquad \text{(Low pressure torque)}$$

$$C_t = C_h + C_l \qquad \text{(Total torque)}$$

i. Electricity generator

$C_u = C_t - k_f N$ (Net torque)

$Q_e = k_e C_u N$ (Electric power)

$O_{vh} = O_{vhi} + O_{vhp}$ (Opening valve)

Isolated :

(Torque auxiliaires) $C_{el} = k_e \left(1 + \frac{0.8}{N_o}(N - N_o)\right)$

(Rotation Speed) $\tau_n \dot{N} = C_u - C_{el}$

$\dot{O}_{vhi} = k_{rni} (N_c - N) - k_{rth} O_{vhi}$

(Valve opening control) $O_{vhp} = k_{rnp} (N_c - N)$

Synchronism :

(Control of synchronisme) $\tau_q \dot{N} = Q_e - Q_{ec} - k_q (N - N_o)$

$\dot{O}_{vhi} = k_{rpi} (Q_{rc} - Q_e) - k_{trh} O_{vhi}$

(Valve opening control) $O_{vhp} = k_{rpp} (Q_{ec} - Q_e)$

IV. COMPUTER PROGRAM

In this chapter the computer program for implementing the simulation hardware is roughly described. Because of the hybrid characteristics of the hardware involved, the program description has to be devided into two main parts :

A. The part which imparts the hardware programming of the analog and logical components of the AD-5 analog computer, the hardware link between the AD-5 and the PDP-11 digital computer and the link between the AD-5 and the operator's desk.

B. The part which imparts the software programming in a FORTRAN code of the link between the AD-5 and the PDP-11, the calculation of that part of the model which can better be resolved digitally and the housekeeping of the several simulation parameters, including the control over the AD-5 and the Desk.

A. The hardware programming on the analog computer includes the following simulation subsystems :

1. The reactor neutron kinetics.
2. The primary circuit : a. the reactor core,
   b. the hot legs,
   c. the cold legs,
   d. the steam generator primary,
   e. the pressurizer.
3. The secundary circuit : a. the steam generator,
   b. the turbine.
4. The electrical circuit : a. the electricity,
   b. the electricity grid.
5. The link of the AD-5 analog computer to the operator's desk :
   a. desk indicators to AD-5 trunks,
   b. desk keys to AD-5 trunks,
   c. AD-5 trunks to desk recorders,
   d. time base for X, Y, Z recorder.
6. The link of the AD-5 of the AD-5 analog computer to the PDP-11 digital computer :
   a. the clock pulse generation for synchronism between the AD-5 and

the PDP-11,

b. the reset-run-freeze control and the speed-up of the simulation mode,

c. the multiplexing (demultiplexing) of the desk-key control lines to the digital computer.

B. The software programming includes the following simulation subsystems :

1. The interactive communications with the user for plant parameter input.
2. The interactive communication between the AD-5 and the PDP-11 for synchroneous calculation.
3. The control over the program as a result of the position of the keys on the operator's desk.
4. The digitally calculations of :
   - the initial steady state of the simulation,
   - the balance of reactivity,
   - the boron concentration,
   - the rods position an control,
   - the xenon cencentration,
   - the primary heat flow,
   - the energy control of the pressurizer,
   - the feed steam generator flow and control,
   - the steam dump and control,
   - the setting point of the generation group's rotation speed,
   - the grid power demand.

The simulation of the zero power operation is a particular set of the full power operation. The same computer program holds. The particular set is obtained by means of interrupting features initiated by the digital part of the program and causing the full power simulation to be frozen at the zero power level. Because of the rather few equations to be calculated in a zero power simulation, all calculations are programmed in the digital part of the system, the real time operation being preserved because of the low amount of computing volume.

V. THE EXERCISES

The exercises at zero power operation are essentially the approach to critical operation, starting from a given neutron source and the consequent initial subcritical flux level. The operator acts on the rods position and the boron concentrations. Equilibrium at intermediate subcritical levels between the initial level and criticality may be reached. At each level the antireactivity of the rods and/or the boron concentration and the inverse of the flux level are deduced from the simulator displays. These parameter values permit to drawn the S-curve of the rods and the approach to the critical position of the control rods. This may be repeated at several values of boron concentration and temperature levels. At criticality the flux may be brought to power level and stabilizing at intermediate levels is possible.

At power level operation a first sequence of exercises includes the examination of the reactivity balances and their consequences. Parameters such as temperature coefficient, Doppler coefficient, rods position and effectiveness, boron concentration and effectiveness, xenon effect, can be varied to show their importance against each other and their effect upon the steady state of the plant.

In a second sequence, load variations are induced in order to examine and explain the subsequent transients of the plant. During this sequence the trainee operator has to operate the simulated plant to reach the desired final situation after a load perturbation, f.i. a load drop or an increase of power demand from the electricity grid. During this sequence, the emphasis lies upon the examination of the transient behaviour of the reactor together with the steam generator, in other words the behaviour of the primary circuit. Also in this sequence, transients are initiated and operated from the side of the reactor, while the secundary power demand remains constant.

In a third sequence the secundary side of the plant is examined. Here the behaviour of the steam generator and its water level control, the behaviour of the turbine and its valve opening control and the behaviour of the electricity generator and its rotation speed control are displayed. Again the trainee operator has to operate to reach the desired end steady state. Perturbations are f.i. isolation from the grid, bringing the electricity group to synchronisme and variations of power demands by the grid.

VI. FINAL CONSIDERATION

The strong safety constrain situation in the case of a nuclear power plant at one hand and the relatively high part of the human factor in an accident at the other hand, require thoroughly training of the plant personnel. Among others two aspects are therefore of main importance in this matter : insight and feeling, reflection and reflex. The development of these two aspects can be supported by the two main advantages of a basic simulator : it ability to analyse in detail and thoroughly the fundamental behaviour of the plant and its real-time mode of operation.

APPLICATION OF COMPACT SIMULATORS IN TRAINING PROGRAMS

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1. History

The need for extensive simulation of nuclear power plants became apparent to the Swedish utilities when their first units were to be ordered in the late sixties. The first simulator model, STUDS, developed by STUDSVIK and the Swedish State Power Board in cooperation, dates from that period. Many modifications and complements have been introduced during the passed years and the STUDS simulators now in use model BWR and PWR systems quite extensively.

At the early stage the STUDS-models were used for transient analysis and stability studies of different kinds. The models at that time gave a first and valuable insight into the dynamical behaviour of different systems, the coupling between the variables and the quantitative effects of various system parameters on the transients.

In the design and construction phase of the first Swedish reactors the utilities made extensive use of the models for system design verification. In particular the control systems were investigated and in some cases alternative designs were initiated by the results of the simulation. Other applications included investigations of methods for identification of BWR core dynamics and development of an integrated control system for power and pressure control of a BWR using optimal control theory.

A growing field of application is the testing of on-line techniques for disturbance analysis, man-machine communication systems and as a tool for operator performance experiments. The STUDS-simulator models are extensively used for these purposes at the OECD Halden Reactor Project in Norway.

The recognition of the need for effective facilities for operator training was manifested by the establishment of a utility owned training center, AB Kärnkraftutbildning, AKU, at Studsvik and by the order in 1971 of the first full scope simulator of a BWR-type, soon followed by an order of a PWR full scope simulator. At the same time the first courses at Studsvik on basic training commenced with the use of a simulator based on the STUDS-models and - for the purpose - tailored operator consoles, both of BWR- and PWR-types. The value of using this type of a compact simulator for basic training was well appreciated by the utilities. Until the time when compact simulators were placed on-site at nuclear power plants, essentially all operators of the Swedish plants were attending the various training courses (~100) given at Studsvik.

The first on-site compact simulator was installed during spring 1979 at the Ringhals nuclear power station. It reproduces the Ringhals 2, Westinghouse PWR unit. A pioneering experience has since been obtained in using such a facility on-site for the training of various categories of technical staff.

The next compact simulator was installed in September 1980 at the Forsmark nuclear power plant. An intensive training program for the Forsmark 1 and 2 unit operators has taken place with this advanced version of a BWR-simulator. An equivalent compact simulator is under construction to be installed at the Oskarshamn plant before the end of 1981. Application of compact simulators in US-based training programs has been reported elsewhere (1,2) and will not be dealt with in this report.

With the compact simulator a new element has been introduced in the nuclear training programs. The present experience covers courses for among others the operating crew, management staff, maintenance personnel, regulatory people and students and many courses serve the purpose of preparations for full scope simulator training.

2. Compact Simulator

The compact simulator represents a functionally complete but simplified nuclear power plant. It is a self contained training tool incorporating all major systems, components, controls, permissives and trips and is designed to simulate virtually all normal operating evolutions. The compact simulator consists of a reactor operator's console, an instructor's console (TTY-terminal), a computer, visual display units (normally 3 CRT's) and a nuclear power plant software model. The lay-out of the instruments and controls on the Forsmark 1-2 simulator console is shown in Fig 1, together with a list of components and systems contained in the simulator. Normally the console lay-out and the software are tailored to the specific plant operating characteristics.

The compact simulator has many features which can be used to maximize training effectiveness such as:

- Full range of operation, from cold shutdown to hot full power (see fig 2 for PWR compact simulator operations during a start-up)
- Operation in real time, fast time, or slow time to enhance training effectiveness
- Displays of the time histories on colorgraphics CRTs of virtually any parameter
- Freeze/restart capabilities to interrupt simulation for purposes of discussion
- Backtrack feature to automatically generate new sets of initial conditions as simulation progresses
- Snapshot capability to allow any conditions simulated to be used as an initial condition
- Instant replay to initiate simulation with ensuing operator actions from a previous snapshot
- Ready modification of program constants for parametric studies
- A number (~20) of instructor-initiated malfunctions
- Annunciation of alarm and trip conditions and capability to bypass reactor and turbine trip conditions.

The simulator has proved to be a valuable tool for the training of both fresh and more experienced personnel. The less experienced trainees aquire a first hand practical understanding of plant dynamics and of the basic modes of operation. The more experienced trainees are brought to a deeper understanding of the main plant characteristics by the concentrated sequence of events and consequences which are shown on the simulator displays and which are not usually experienced in the daily operation of the plant. Furthermore they are given the opportunity to evaluate and discuss the course of events, without having to take the regular actions. A widely used training scheme is to first perform a classroom investigation of an actually occurred disturbance transient and later have the students handle the same disturbance transient by operating the compact simulator.

3. Application in training programs

There is within the nuclear field a growing awareness of the advantages in expanding the use of simulators in the training programs. This is generally manifested by the increased number of full scope simulators presently being installed in the world, but also, as experienced in Sweden by taking into use on-site placed compact simulators.

The training courses with the compact simulator which over the years have been carried out in Studsvik, have been attended by almost every category of technical staff in nuclear plants. The simulator is simple enough to quickly convey operational concepts to inexperienced personnel, while at the same time complex enough to challenge experienced personnel. Although primarily designed as an operator training device, the simulator's usefulness in orientation programs allows its application to a wide range of personnel, see fig 3. The limited size and the comparative simplicity of the operator console, which visualizes the main circuits by mimic diagrams allows the trainee to get acquainted with the operation of the simulator within an hour.

In general the training sessions last over a period of 2 to 3 days. In the STUDSVIK courses emphasis has been placed on the reactor operation and on detailed studies of reactivity effects. Reactor start-up, control rod maneuvering, approach to criticality, heating and power range operation with control rods, recirculation flow (BWR) and boration or dilution (PWR) are exercised. Attention is paid to reactivity feedback due to void formation, fuel and moderator temperature and Xenon. Fundamental relationships are illustrated and put into context with the integrated plant operation.

The Ringhals PWR compact simulator was procured by the utility mainly to support the training program for the PWR shift group personnel. The initial training program includes the so called reactor operations course at which the simulator is used. Subjects covered are reactor dynamics, transientanalysis and application of operational instructions at normal operation and at plant disturbances. One goal is to convey understanding of interactions between the reactor and the turbine at transients and disturbances. In addition results of various transients such as reactor trip, excess feedwater, loss of off site power, partial loss of reactor coolant flow, control rod withdrawal and control rod drop are studied.

Since the installation of the simulator in May 1979 the following simulator based courses have been effectuated at Ringhals:

- Basic course for 20 candidate reactor operators for unit for and for 4 members from the operation department (before commissioning of unit 4)

- Retraining courses for unit 2 and 3 shift group members, ~60 persons
- Preparatory courses for full scope simulator training at AKU attended by each shift group of unit 3, ~30 persons
- Basic course for unit 2 and 4 candidate reactor operators, ~30 persons plus one AKU candidate instructor
- General course for students (~25) from The Gothenburg Institute of Technology

Apart from these organised ocurses individual training is regularly taking place on the operator's own initiative.

Since the installation of the BWR compact simulator at Forsmark simulator training has been going on almost on a continuous basis. Refresher training programs, set up for the operators, typically consist of a yearly one week course at the full scope simulator and a three week program on-site with courses in which the compact simulator plays an important role.

In total the compact simulator courses performed at Forsmark comprise:

- 14 shiftgroups	basic course for each group, 2 days, 7x14 persons
- Commissioning staff	studies of test operation procedures, 2 days
- Candidate reactor operator	concentrated course, 2 days
- Maintenance staff	start-up procedure and full power operation 3 days, 10 persons
- 14 shiftgroups	transient analysis of actually occurred disturbances, 1 day each, 7x14 persons
- Candidate turbine and reactor operators	basic training courses, 3 days, 6 persons

There is a consensus among the users of the compact simulator about the effectiveness of its capabilities in the training process. It is also agreed that improved structuring of training exercises including expanded documentation and new application would enhance its usefulness still further.

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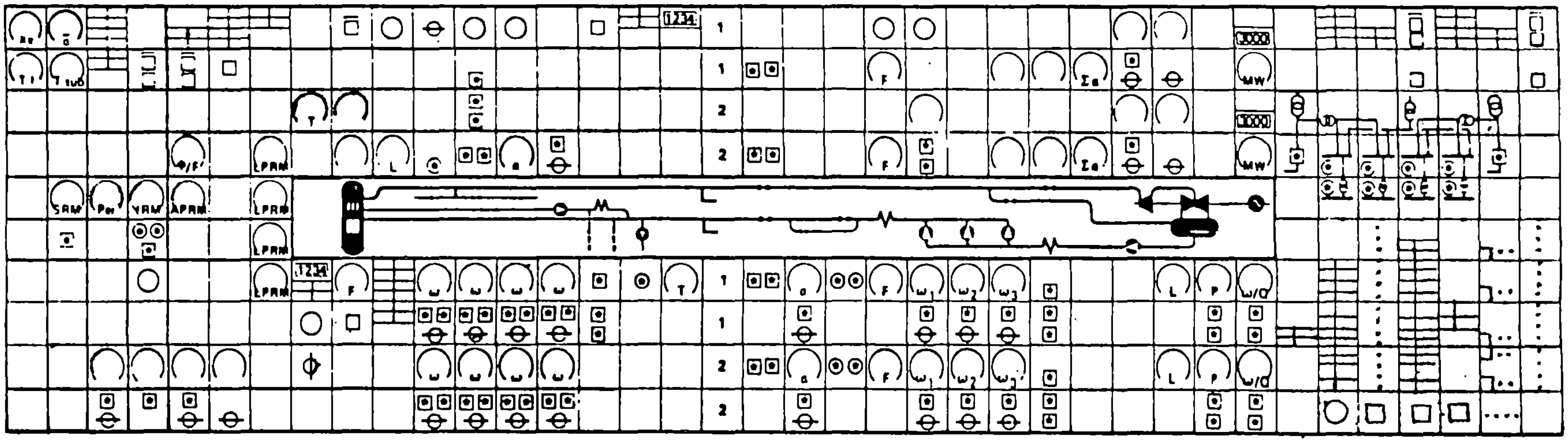


- Reactor Vessel and Core
- Recirculation Pumps
- Control Rods
- Feedwater Pumps
- Feedwater Isolation Valves
- Feedwater Bypass Valves
- Feedwater Control Valves
- Auxiliary Feedwater System
- Generator Output Breaker
- Reactor Relief Valves
- Reactor Pressure Control Valve
- SRM, IRM, APRM, LPRM's
- Reactor Protection System
- Shutdown Cooling System
- Low Pressure Injection
- Draining
- Main Steam System
- Turbine Generators
- Condensers
- Condenser Vacuum System
- Circulation Water System
- Generator Load and Speed Control

Fig 1. Forsmark compact simulator components and systems.

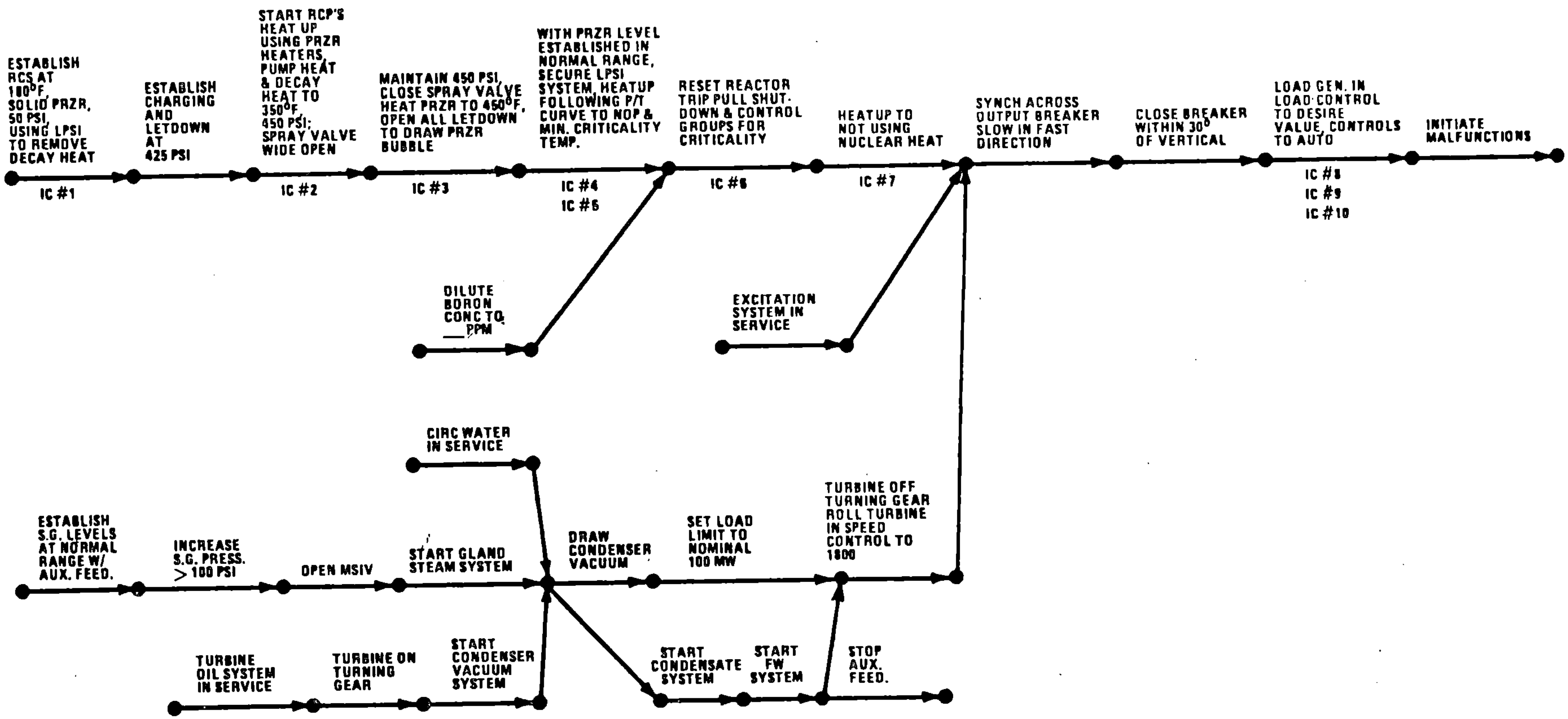


Fig 2. Simulator evolutions in PWR startup from cold shutdown.

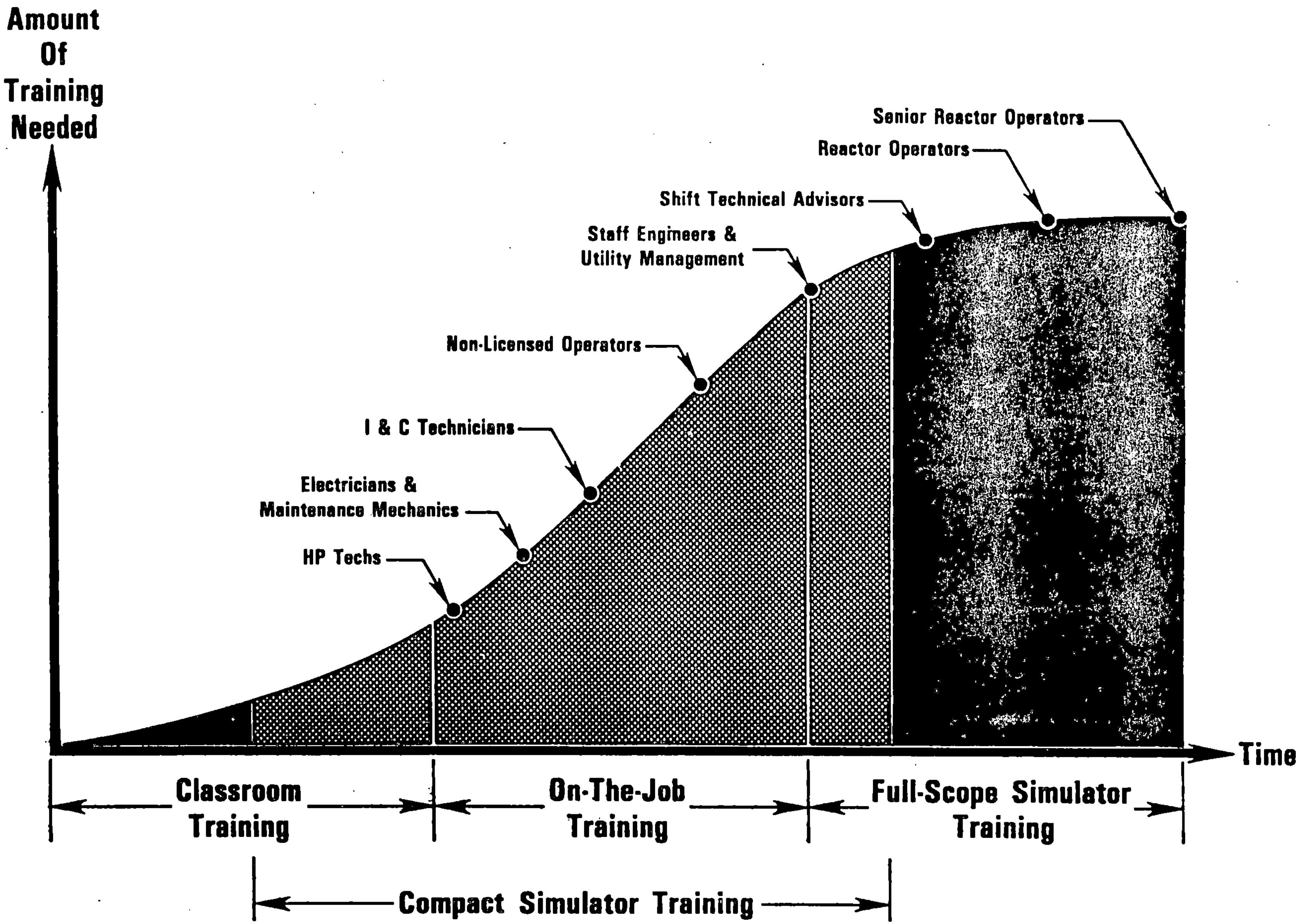


Fig 3. Compact simulator integrated into site training programs.

QUESTIONS TO PEHR E. BLOMBERG

W. F. Witzig **Q:** Please tell us the cost of the simulator you have just described.

A: This depends on the extent of tailoring work and the volume of additional systems desired by the customer, but normally the price settles below $500 thousand.

Rafael Vargas **Q:** Which kind of operational instructions do you use for your generic compact type of simulator, a simplified set from the reference plant or a special developed one? Which has been the impact on the shift groups the use of such set?

A: There is an operational manual available aiding the start-up of the simulator, setting the initial conditions, time scales, operational modes, etc., to the most part operated from the instructor's console, i.e., the teletypewriter terminal. There are also training manuals available, and specific ones are successively added as results of the ongoing training programs presently conducted at the various compact simulator users. Certain training programs are aimed for training candidates in the use of the regular operational instructions of referenced plant. The courses are very much appreciated, but an expanded set of documented exercises are desired.

IMPROVEMENTS IN SIMULATOR TRAINING FOR PWR

Dr. E. Lindauer, M. Simon, D. Reppmann

(Federal Republic of Germany)

The paper mainly deals with a study, which was carried out on possible improvements in simulator training in the FRG.

The aim of these investigations was to check the accidents and malfunctions already available at the simulator by means of systematic evaluation of operational experience and accident analyses and to suggest improvements resulting from this investigation.

The investigations carried out so far which utilized the available operational experience for Pressurized Water Reactors (PWR) and accident analyses mainly consideres the following aspects:

- Increase of the spectrum of simulated accidents and malfunctions
- Enlargement of the simulation volume for particular accidents and malfunctions
- Improvement of simulation accuracy

The paper discribes the approach of the study, the results gained so far and future goals. Besides, a short survey on the situation in the training simulator field in the FRG is given.

1. INTRODUCTION

In the Federal Republic of Germany, three simulators are used for the training of NPP personnel at present. Two of these simulators are operated at a simulator center organized by the licensees, the Power Plant School at Essen (KWS). The two facilities - a PWR simulator and a BWR simulator - have been in operation since late 1977. NPP Brunsbüttel (BWR) and NPP Biblis A (PWR) were used as reference plants. However, the control rooms ot these simulators did not imitate those at Brunsbüttel and Biblis. Instead, new control rooms were conceived which also take the control room designs of other nuclear power plants in the Federal Republic of Germany into consideration, since the personnel for all German NPP licensees is to be trained at this school.

The third simulator is located at KWU. It is used for the training of KWU's own commissioning personnel and the operating personnel of foreign customers. This is a PWR simulator which was put into operation in 1978 and corresponds to a great extent to that operated by KWS. However, its control room was already designed along the lines ot that planned for the future PWR standard plant.

A detailed descirption of the simulator would be excessive here. In general terms, one may say its equipment corresponds more or less to that of US simulators of comparable years of construction.

In early 1980, the Federal Ministry of the Interior (BMI) awarded GRS and KWU a contract for a study to investigate possibilities of improving both the existing PWR simulator and the training carried out with it. The TMI II accident was one of the major reasons for launching the study. It concentrates on avoiding, detecting and coping with malfunctions, the aspect of training for normal operation was disregarded.

The major aspects of the investigation may be characterized by the following two questions:

- Does the simulator incorporate all the mafunctions which are important for training purposes ?
- Is the simulation of the various malfunctions sufficiently realistic for the purpose ot training ?

2. REVIEW OF THE SCOPE OF SIMULATION

We considered an evaluation of malfunctions which have already occurred at PWR's as the most appropriate method to assess the appropriate scope of simulation. A purely theoretical approach to this question seemed to us to be less promising.

Thus, a great deal of operational experience gathered at both German and US PWR plants was evaluated in order to carry out the first part of the study, which was completed by early 1981. The evaluation was aimed at setting up a catalog of malfunctions and accidents which the simulator should have at its disposal. By way of comparison with the malfunctions already incorporated in the simulator, the study was to find out

- whether all relevant malfunctions are incorporated in the simulator, and

- whether these malfunctions are modeled completely, as far as initiating events and sequences or consequences are concerned.

With respect to operational experience at PWR plants, the following sources were available:

(1) Malfunctions at German plants, which either were reported by the licensees to the respective regulatory authorities and then stored by GRS in a data bank, or were described elsewhere. Approx. 1,400 events were evaluated.

(2) A collection of US events in Nuclear Power Experience (NPE), which is based on Licensee Event Reports (approx. 4,500 events were evaluated).

These events were classified and assessed as to whether or not they were important for simulator training. The assessment was performed on the basis of the following selection criteria:

- the importance of the event for the safety of the plant;
- the occurrence frequency of a malfunction;
- the consequences in terms of plant dynamics;
- whether or not the event is suitable for simulation;
- the importance of the event for the operational availability of the plant;
- events at plants abroad had to be transferrable to German plants.

With respect to the last item, it should be mentioned that the evaluation of US events was not particularly productive from the point of view of selecting malfunctions for a German PWR simulator. In many cases, the cause or sequence of a malfunction was found to be not transferrable to German PWR's. This is due to differences in plant technology e.g. in the design and control of operational and safety systems including the reactor protection system.

Results of the Review:

The evaluation showed that many of the malfunctions were already incorporated in the PWR simulator. This was to be expected since, for example, the KWU PWR simulator incorporates 150 different malfunctions. However, simulated malfunctions and actual malfunctions observed at a PWR were frequently found to differ with respect to both the initiating event (i.e. the history) and the sequence of the malfunctions concerned. Thus, the comparison between simulated malfunctions on the one hand and operational experience and/or accident analyses on the other led us to a number of general and a number of specific findings regarding simulator backfittings or improvements.

Specific Findings:

Notwithstanding the fact that the simulator already incorporates 150 malfunctions, another 60 malfunctions were found which we considered to be of importance for training purposes and which should therefore be incorporated in the simulator.

In the description of these accidents and malfunctions, major emphasis is put on the sequences involved, and they are assessed from the points of view of plant technology, plant dynamics and what is of importance in training. Furthermore, a specification was prepared for the extensions and improvements to be carried out on the simulator.

Without listing these additional malfunctions and accidents in detail, they can be classified as follows:

- leaks or breaks of various sizes and locations;
- malfunctions involving a mismatch between power generation and power output;
- malfunctions as a result of inadvertent actuations;
- malfunctions as a result of operator errors;
- malfunctions due to erroneous operation of controls;
- malfunctions involving multiple failures.

Even the inclusion of only some of these additional malfunctions means that the extent of simulation has to be correspondingly enlarged with respect to both the systems modeled and the quality of the models used.

As far as the modeled plant systems are concerned, this means:

- An extension of those systems which already exist in the simulator but have not yet been modeled to the necessary degree. For example, the containment air control, waste gas and the process controller are represented by imperfect models.
- Systems which are not yet included in the simulator such as the separate shutdown and residual heat removal system

to cope with external impacts, the coolant storage and purification system, and the automatic unit for testing the turbine protection system, have to be added.

- Extension of the simulator by the incorporation of important components which have a safety or availability function and [illegible] are not so far simulated such as:
- safety valves (only the most important have so far been simulated);
- check valves; and
- redundant auxiliary components (e.g. standby pumps).

As far as the models are concerned, the investigation showed that some of the existing models are not sufficient to simulate the malfunctions which have already occurred in practice. This refers to malfunctions in the reactor coolant system and/or the pressurizer, as well as to combinations of malfunctions.

General Findings

A comparison of the malfunctions gathered from operational experience and the simulated malfunctions shows that, in principle, most of the malfunctions are already incorporated in the simulator. However, the simulation frequently covers only the failure of a system or component and the consequences of this failure but not its history. For example, the malfunction "loss of one reactor coolant pump" may have a number of causes e.g. the failure of any one of the auxiliary systems required for the operation of the pump in question. A complete simulation would also allow for training of the timely detection and prevention of malfunctions.

Furthermore, it was found that multiple malfunctions or combinations of malfunctions frequently occur in pratice. Although, in principle, such combinations can also be simulated (the KWU simulator accepts the simultaneous input of up to 12 malfunctions), but they have not yet been checked out. Only the sequence of single malfunctions is confirmed experimentally or by design and safety analysis codes.

Thus, the following improvements are possible:

- the modeling of the initial causes of malfunctions
- the simulation of consequential faults in the sequence of malfunctions
- the simulation of combinations of malfunctions.

In addition, the evaluations gave a number of indications, how to improve the training as such. However, these will not be discussed here in detail.

3. REVIEW OF SIMULATION MODELS

The second part of ths study is aimed at reviewing the accuracy of the simulated sequences of malfunctions. However, work on this part is still in the initial phase.

Above all, the simulator has to reflect the overall behavior of the plant in the case of malfunctions. The accuracy of the individual parameters, which is required for design and safety analysis codes, is only of secondary importance here. In the Federal Republic of Germany, there is no such guideline as the ASN 3.5 Standard which prescribes a certain accuracy of simulation. Such a guideline is not to be expected in the near future nor, in our opinion, is it necessary. As already mentioned, the malfunctions and accidents available in the simulator have been checked out, i.e. they have been reviewed by means of commissioning tests and design and safety analysis codes.

The situation is different with respect to malfunctions not yet available in the simulator or not scheduled for simulation, such as various combinations of malfunctions. In these cases, it has to be examined whether or not the capacity of the existing models is sufficient.

It is not exactly known where the limits of the existing models are, where they can still be improved or to what extent they have to be replaced by new and better models. The evaluation of operational experience shows that malfunctions have to be modeled with sufficient precision even if the boundary conditions for these malfunctions change. For example, the "spurious opening of the main steam bypass" must be capable of being simulated:

- for the different states of operation (zero load/power operation),
- for the different burn-up conditions of the core (BEGINNING OF LIFE/ MIDDLE OF LIFE/ END OF LIFE), and
- for different valve flow rates.

It is obvious that this requires that the simulation of malfunctions should be carried out as little sequenceorientated as possible and that simulation should be based on physical models and technological relations.

We know that this is not the case for all malfunctions. However, the quality of simulation does not depend solely on the quality of the models used. A successful integration of the various models, i.e. the adaptation of the models to each other, is at least equally important.

To be able to investigate the capability and degree of integration of the models, a number of malfunctions were selected which we believe to be especially suited for this test. The results supplied by the simulator will be reviewed with the special transient codes which are used by GRS and KWU as design and safety analysis codes.

One calculation has already been made. It concerns a subcooling transient which occurred at the simulator's reference plant and for which measured data are available. However, difficulties arose both in the practical implementation of the comparative calculation between simulator and transient models and in the interpretation of the results obtained. The test had to be repeated several times until it was certain that the same starting conditions applied to both the simulator and the transient codes. The reason was, that the system technology, expecially the influence of controls, were not modelled identically in the simulator and the transient codes.

When evaluating the results, it was found that the data supplied by the simulator regarding "pressurizer level" and "reactor coolant pressure" deviated from the results of the transient codes. Although sufficient agreement with the real plant was achieved with the transient codes, an adaptation of parameters was necessary for this purpose. Such an adaptation of parameters could not be done in the simulator. It has not been possible to state beyond doubt the causes of the differences between simulator and transient codes, as the results always include the integral behavior of the entire plant. However, a thorough understanding of the differences is of course indispensable.

As a result of these difficulties, we feel that the integral comparison of the course of transients is, as such, not sufficient to review the quality of the models used in the simulator. In addition, it will be necessary to test the individual models separately without having to consider the behavior of the whole plant in the assessment. However, such an approach requires interference with the simulator software, with a disturbance being introduced at a suitable point in the program. Further interventions in the program then serve to suppress the influences of controls, limitations and reactor protection in order to eliminate the control system's feedback mechanisms which are undesired features in this model test. The pressure, temperature and level curves, plotted after the introduction of the disturbance, show the transient behavior of these variables und allow an evaluation of the simulator models by means of comparative calculations with qualified computer codes. Reviews of the following thermohydraulic models are planned:

- Reactor coolant system model
- Pressurizer model
- Steam generator model.

The course of a transient is decisively determined by reactions of the control systems. Consequently, it was decided to review the following controls:

- ACT control
- Main steam control
- Feedwater control.

In addition to the review of the individual simulator models, it is also intended, as already mentioned, to simulate the integral responses to selected malfunctions in order to be able to review the integration of the various models and/or the simulation of the overall behavior of the plant. For this purpose, priority should be given to malfunctions for which measured data are available from real plants.

4. FUTURE TRENDS:

In parallel to the investigations mentioned above, a programme for upgrading the simulator in the training center of the utilities (KWS) was started. Since it is not yet finished, no details can be given in this paper. Should it turn out, that there are fundamental obstacles to implement the required improvements on a reasonable expenditure. The construction of a third generation simulator might be advisable. The differences in the technology of the new plants have also to be considered in this context.

Such a third generation simulator was ordered by Nuclebras, Brazil, for the PWR plant at Angra II, and is already under construction. The plant is being constructed by KWU and essentially corresponds to future German PWR plants. KWU is the general contractor for the construction of this simulator until delivery to the licensee. The software, however, was subcontracted by KWU.

The following is a discussion of the major differences between this simulator and those at Karlstein and KWS.

Control room design: the concept of the control room corresponds to that of the new KWU standard PWR plant with a main control panel, an instrument rack with five CRT's and three secondary control panels. The main control panel and the instrument rack have been simulated 100% and the secondary control panels 90%.

Contrary to the old simulators, the redundant channels of the reactor protection system are simulated. This allows to perform functional tests on the simulator.

As far as the simulated systems are concerned, ten more systems have been modeled than in the KWS/KWU simulators where these additional systems are either not available at all or only available in very simplified form, such as the additional startup and shutdown system, the additional boration system, the containment air control system and the coolant storage and purification system.

A total of 14 automatic units for start and stop of subsystems are fully simulated, as compared with only 2 at the PWR simulators of KWU and KWS.

The simulation scope of the process controller was greatly extended in the Nuclebras simulator.

Equipment in Terms of Hardware and Software:

As far as hardware is concerned, four 32-bit processors are used as compared with three 16-bit processors for the KWU/KWS simulators.

The interface has 18.000 I/O, as compared with 7.000 I/O in the KWU/KWS simulators.

The programming language is Fortran IV, as compared with Assembler for the KWU/KWS simulators. This, and an advanced structurization of the programs, will facilitate software changes.

Malfunctions:

150 major malfunctions and accidents can be simulated on the Nuclebras simulator. The results of the GRS/KWU study were taken into consideration here. For example, new malfunctions were incorporated which are not yet available in the KWU/KWS simulators. A number of new malfunctions resulted from the different power plant technology of the new plants. Another 200 so-called simple malfunctions, i.e. erroneous alarms and answer-back signals which do not have any direct effect on the plant, are available. In addition, there are 250 so-called instructor functions which can be used to modify external parameters (such as coolant temperature or phase difference) or the position of manual valves or the availability of components (whether under repair or in an operable state).

Thus, a great number of possible combinations or variations of malfunctions result.

Models:

The Nuclebras simulator already uses more advanced models than the KWU/KWS simulators:

- The neutron-kinetic behavior of the core is described by a 1-D model instead of the point-kinetics model used by the KWU/KWS simulators. It is thus possible to calculate power distributions in the axial direction.

- The model for the thermohydraulic behavior of the pressurizer is a two-phase flow model.

- The model of the steam generator was improved.

Use of the simulator:

As the completion of Angra II will be considerably delayed and the Nuclebras simulator will be ready in early 1982, a training simulator embodying the latest technology will for some time be available at the KWU training center as of late 1982.

TRAINING SIMULATORS - MAJOR ISSUES REMAIN

Gary M. Grant

ABSTRACT

The issues surrounding nuclear power plant simulators continue to be the focus of considerable industry attention and energy. Fundamental training issues include the use of performance objectives in simulator training and the importance of plant-specific simulators in achieving these objectives.

The difficulty in addressing these issues is the result of a wide variety of circumstances. Perhaps most important is the evolving role of the operator in power plant safety. Training needs cannot be defined until the role of the trainee is thoroughly understood. It was not clear until after the accident at Three Mile Island Unit 2 (TMI-2) that the role of the operator may have been improperly defined. As a result, the performance objectives of simulator training and the impact of plant-specific simulators became a serious issue in the training community. From another perspective, TMI-2 graphically demonstrated that the conservatism of design codes ignored the "real" plant behavior and had a potentially negative influence on operational safety. This led to concerns over the models used in simulators and brought the technical community into the debate. Finally, the public wants and deserves the best possible assurances that nuclear plants are operated by highly trained, competent personnel. To this end,

regulatory agencies and other governmental bodies have been involved in the resolution of the issues surrounding nuclear power plant simulators.

To address these issues, examples of simulator performance objectives considered appropriate for operator training are developed in this paper. These are predicated on a defined operator role. The use of performance objectives in determining requirements for simulator hardware, software, and use is also examined. Finally, recommendations for further treatment of these subjects are made.

INTRODUCTION

"How do you get the most benefit out of a simulator training program?" Since the accident at Three Mile Island, this question has been raised in training circles perhaps as frequently as any other. Even today, two and one-half years later, a clear, concise answer is not available. In fairness, addressing the question in an atmosphere of conflicting studies, commercial pressures, and changing requirements is not easy, and, in fact, a single answer may never be reached. A method of addressing the question on a local level and in a way that meets the objectives of the users is developed here.

What are the objectives of simulator training? In order to answer the question of how to get the most benefit from a simulator training program, the performance objectives* to be accomplished must be determined.

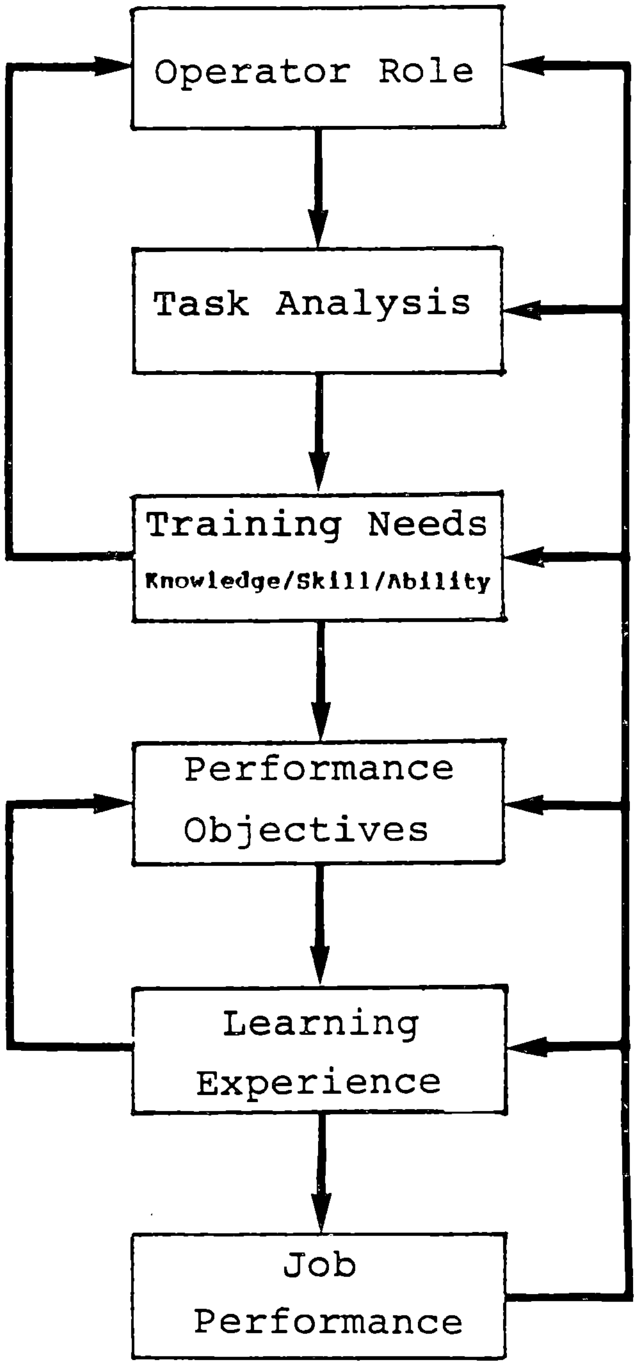
Performance objectives are concise statements of (1) what it is that a trainee who has mastered the objectives will be able to do, (2) under what conditions he will be able to do it, and (3) the criteria against which he will be evaluated. As designed, performance objectives are useful to trainees, instructors, and program developers. For trainees, performance objectives

*Terms such as "behavioral objective," "learning objective" or "instructional objective" are often used interchangeably with "performance objective" in training program development.

eliminate the "fuzziness" of assignments and exercises and the need to guess what is important and what is not. As a result, studying is more efficient, and trainees can evaluate their own progress toward achieving the objectives and can seek assistance if needed. For instructors, performance objectives organize the wealth of materials into categories such as objectives, clarification, nice to know, and irrelevent information. This allows for more effective instruction and better testing and evaluation. For program developers, a well-constructed set of performance objectives provides the necessary basis for making decisions about program content and learning experience. For management, the development and use of performance objectives provides valuable insight to the cost-effectiveness of simulator training.

The Operator's Role

Before performance objectives can be developed, training needs must be identified as the result of a systematic task analysis. Figure 1 shows the overall training program development process and the relationship of performance objectives to other equally important elements.



Training Program Development Process

Figure 1

In order to demonstrate the usefulness of performance objectives in the determination of simulator requirements, an aspect of the operator's role in nuclear safety has been chosen. Corcoran[1] describes the operator's role in the following terms:

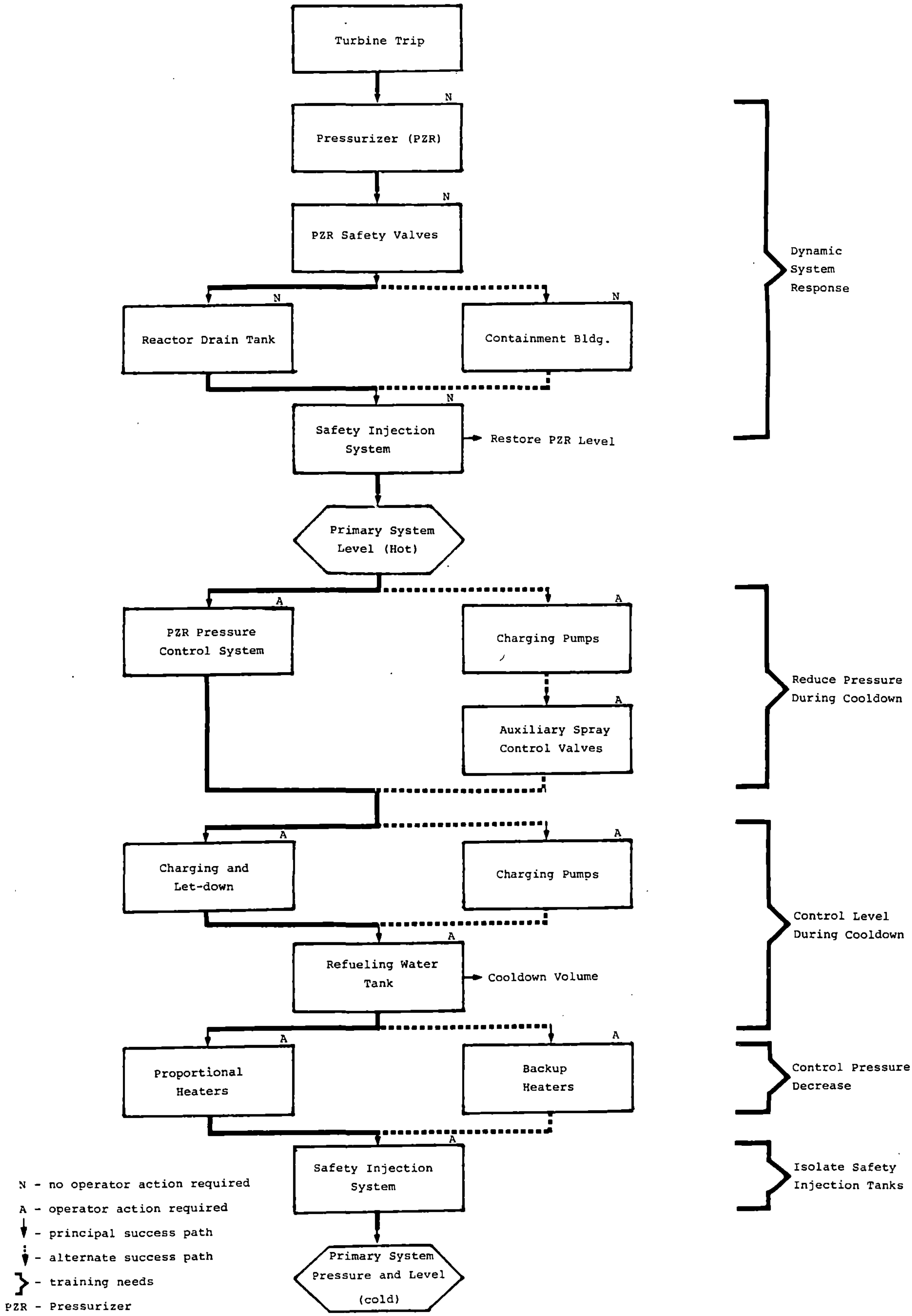
1. keep the plant set up so that it will respond properly to disturbances

2. operate the plant so as to minimize the likelihood and severity of event initiators and disturbances, and

3. assist in accomplishing safety functions during the event

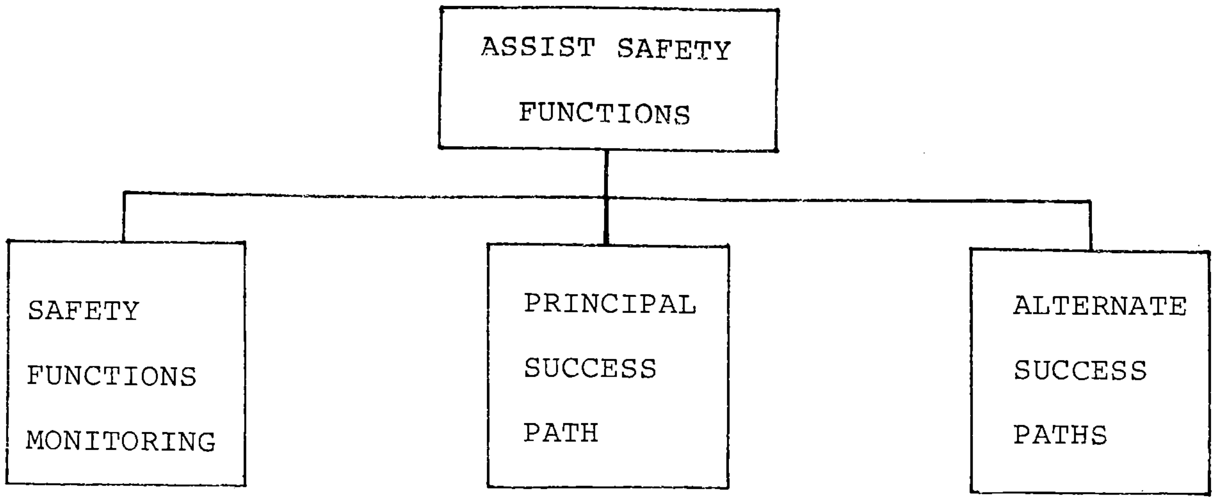
These global statements are useful as a basis for the performance of a task analysis. To demonstrate the process, only one role will be explored, that of assisting in accomplishing safety functions during an event.

First, the operator's role must be described in more specific detail. This step has been provided[1], as shown in Figure 2. Safety functions are defined as one or more actions that prevent core damage or minimize radiation releases to the general public. Success paths are the logical organizations of these actions. Each safety function normally has a principal and alternate success paths.



Primary System Pressure and Level Control Success Paths

Figure 51



THE OPERATOR'S ROLE

FIGURE 2

For example, a turbine trip from power is an event that results in an upset condition involving the control of primary system pressure and water level, two safety functions. As shown in Figure 3, multiple success paths are available to ensure ultimate, long-term control of primary system pressure and water level. These paths have been analyzed to identify required operator action. This analysis serves as a task analysis from which training needs and performance objectives will be identified and developed.

Performance Objectives

Performance, conditions, and criteria are the fundamental components of well-constructed performance objectives. Job performance is the ultimate measure of success in any training program and is an essential element in any performance objective. To be useful, performance objectives must include a description of measurable trainee performance. The following examples serve to illustrate this concept of measurable performance:

> The trainee shall understand how to control primary system pressure . . . .

While the objective does include required performance, "understand" is not directly measureable. Alternatives are available such as the following:

> The trainee shall control primary system pressure . . . .

or

> The trainee shall describe in writing how to control primary system pressure . . . .

A useful rule of thumb for evaluating performance requirements has been provided by Craik[2]. She recommends that after the performance objective has been prepared, an attempt should be made to visualize the trainee achieving the objective. If the performance can not be visualized, the objective should be rewritten in more measureable terms.

Conditions and criteria are qualifiers. In some cases, stating non-trial conditions and/or criteria in the performance objective may not be possible. In each case, however, they should be included if needed to better communicate the intent of the objective. Using a measurable performance statement from above, conditions (indicated by parentheses) and criteria (indicated by brackets) are added as follows:

> (Using appropriate references), the trainee shall be able to control pressurizer pressure within allowable limits (during plant cooldown using the auxiliary spray-control valve).

In order to achieve this performance objective, the trainee must have achieved certain enabling objectives, such as the ability to use procedures or pressure-temperature graphs. A detailed task analysis is necessary to identify the complete set of training needs for the development of both performance and enabling objectives.

Referring to Figure 3, the performance objectives may be identified as follows:

P.O.1. For a turbine trip event, the trainee shall be able to verbally identify (within one minute) a deviation of normal pressurizer pressure response greater than 200 psi and to identify (within five minutes) the cause of the deviation.

P.0.2. Using appropriate references, the trainee shall be able to control pressurizer pressure within allowable limits during plant cooldown, using the auxiliary spray control valves.

P.0.3. Using appropriate references, the trainee shall be able to control pressurizer pressure within allowable limits during plant cooldown, using the pressurizer pressure control system.

P.0.4. Using appropriate references, the trainee shall be able to control pressurizer level within allowable limits during plant cooldown, using the charging pumps and charging line backpressure control valve.

P.0.5. Using appropriate references, the trainee shall be able to control pressurizer level during plant cooldown, using the changing pumps and letdown flow controller.

P.0.6. Using appropriate references, the trainee shall be able to control the rate of decrease of pressurizer pressure within allowable limits during plant cooldown, using the pressurizer backup heaters.

P.0.7. Using appropriate references, the trainee shall be able to control the rate of decrease of pressurizer pressure within allowable limits during plant cooldown, using the pressurizer proportional heaters.

P.0.8. Using appropriate references, the trainee shall be able to depressurize (by draining or venting) and isolate the safety injection tanks during plant cooldown.

Achieving the Objective

At this point, the foundation for training program development has been laid. First, the operator's role was defined. Second, a task analysis was used to identify the training needs, and third, performance objectives were developed to meet those needs. Using this information, the program developer must design a learning experience that will "best" achieve the objectives.

The design of learning experiences is an iterative process involving complex multiple decisions and tradeoffs. Learning experiences may involve formal classroom instruction, on-the-job training, training on operational equipment, or training on a device such as a simulator. During the design of a learning experience that involves a simulator, the program developer must analyze the requirements imposed on the simulator by the performance objective. The net result of this analysis is a list of simulator requirements that must be satisfied in order to accomplish the performance objective. Analysis of several of the previously developed performance objectives is shown below.

Objective Analysis	Simulator Requirements
P.O.1	
1. deviations greater than 200 psi	1. (pressurizer) model fidelity

2. time measured criteria	2. device location, similarity
3. deviation cause	3. system design including alarm cues

<u>P.O.2</u>

1. control within allowable limits	1. (pressurizer) model fidelity
2. using auxiliary spray control valves	2. system design

<u>P.O.8</u>

1. depressurize and isolate	1. system design

Given the list of requirements, it can now be determined whether or not a non-plant-specific simulator can be used in the training program. For example, P.O.8 requires compatibility of system designs (in the area of safety injection tanks) between the trainee's home plant and the simulator reference plant. If such compatibility exists on a non-plant-specific simulator, the objective can be accomplished, regardless of other differences. P.O.2 imposes additional requirements in that the pressurizer pressure model must produce responses that are comparable to the trainee's home plant response, and the reference system design must include auxiliary spray control valves.

The process of comparing requirements to simulator capabilities and characteristics is an important part of program development. The result is a clear statement of what can and cannot be accomplished on a given simulator. The process is an iterative one and could result in a restatement of the performance objective or a decision that a different or additional learning experience is needed. It may also lead to the decision that a plant-specific simulator is needed that meets the requirements now defined.

Caution and Conclusion

Without question, the availability of a plant-specific simulator eliminates some work for the program developer. Having one, however, does not eliminate the need for systematic training program development. Many programs in the past have been built on the assumption that effective training is a function of the training device, rather than a product of a systematic development process. A United States Air Force study[3] offers a pertinent general conclusion:

> Understanding of the relation of simulator design features to simulator training effectiveness is quite limited. It is clear that it is not entirely a matter of duplicating an aircraft. Instead, it appears to be a matter of providing a learning environment in which precisely specified training objectives may be addressed.

A more recent study in the United States nuclear industry[4] gives the following conclusion:

> Decisions made on training tend to be made more on a subjective basis. The behavioral characteristics, the objectives to be met by each element of training, and the criteria for selection and satisfactory performance all tend to be subjectively defined on the basis of experience.

Likewise, the study found that "simulator requirements and simulator training programs are specified largely by subjective judgement" (rather than by a systematic methodology).

In conclusion, a systematic approach to training program development should include the development of performance objectives. In addition to their usefulness to trainees and instructors in general, performance objectives are essential for determining training device (simulator) requirements. Once the training requirements are known, a rational basis exists for making decisions regarding the need for a plant-specific simulator or the suitability of a non-plant-specific simulator. Existing programs, regardless of simulator type, could be made more effective by the development and use of performance objectives.

ACKNOWLEDGEMENT

The author would like to acknowledge the suggestions of Peter W. Steele, Walter M. Guinn, Albert M. Mangin, Ronald L. Fritchley, and E. L. (Red) Thomas during the review of this paper.

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QUESTIONS TO GARY M. GRANT

Persensky **Q:** Regarding your model, doesn't the task analysis identify job "performance objective," which can then be translated into K/S/A, which then relates to training "performance objectives"? Or can you discriminate between job and training "performance objectives?"

A: In some cases, the description of performance standards developed through job/task analysis may be very similar or even identical to a performance objective which is achieved using a simulator. However, a clearer distinction exists in areas where simulator training is inappropriate.

Robert C. Evans **Q:** Do you believe that there will, in the near future, be a regulatory requirement for operator evaluation, using the simulator?

A: Yes, such a requirement now exists in the U.S.A. Full implementation and evaluation of results will take some time.

Authors: F. C. Luffey
P. E. Meyer
J. H. Murphy

ABSTRACT

Advanced Techniques for Real Time Simulation of Reactor Coolant System Two-Phase Transients

Current nuclear power plant training simulators have limited capability for the simulation of two-phase flow in the reactor coolant system (RCS). Increased attention to operation sequences or accidents leading to two-phase conditions in the RCS indicated a need to enhance this capability in plant simulators. This paper represents the major features of Westinghouse simulator technology by describing the hardware and software used in current Westinghouse-designed simulators, and the software under development for the advanced simulator. The advanced simulator is an all digital solution method and permits Westinghouse simulators to compute the two-phase fluid conditions in a reactor coolant system following significant losses of primary coolant, and other similar related transients. Simulator model requirements address both single- and two-phase flow, forced and natural circulation, phase separation, reliability, and real-time capability. The Westinghouse advanced simulator reactor coolant system model meets these requirements through non-equilibrium stratified fluid nodes, drift flux flow links, and global compressibility. These computations in real time are made possible by significant advances in numerical techniques. This advanced simulator is modular in concept, and permits backfitting into existing Westinghouse simulator designs. Transient results are presented in this paper for several loss of coolant accidents, and "St. Lucie" type cooldown

INTRODUCTION

Increased attention to operation sequences and accidents leading to two-phase flow conditions in the Reactor Coolant System provided the need to develop two-phase flow models for Advanced Technology Simulators. Currently, all simulators have limited capability in the simulation of two-phase flow in the Reactor Coolant System (RCS). Presently, Westinghouse Simulators can simulate two-phase transients in sufficient detail for operator training. However, the Advanced Simulator model extends this capability in sufficient detail to allow engineering analysis, and procedure evaluation and verification in two-phase regions. Also, the Advanced model contains sufficient detail in plant parameters to use as a tool for training engineers in two-phase flow transients. The major emphasis of this paper is to describe the Westinghouse Advanced Simulator RCS model which addresses both single and two-phase flow, and phase separation in real-time. Preliminary results and evaluations of loss of coolant accidents (presented later) indicated an additional need to improve the steam generator and reactor fuel rod models to be thermally and hydraulically compatible with the two-phase reactor coolant system model. These models are still in a development phase and therefore will be described briefly. Development of the Advanced Simulator has emphasized real-time capability, system modularity for retrofitting into Westinghouse Simulator designs, and reliability through overall system software and hardware design. The present RCS model is capable of running two times real-time on a SEL 32/77 computer.

SYSTEM DESIGN OVERVIEW

All Advanced Simulator models are programmed in Fortran and are implemented into a dual Systems Engineering Laboratory (SEL) 32/77 digital computer (32 bit) system shown in Figure 1. Each computer contains a central processing unit (CPU) and an internal processing unit (IPU), which is primarily a "number crunching" processor, slaved to the CPU, but operating in parallel with the CPU. The dual system computers are interfaced with either shared memory or via data link. This configuration effectively results in having up to four processing units performing real-time tasks in parallel (i.e., 4 tasks). With the Advanced Simulator operating, approximately 50% of this capacity is used. The system hardware design provides sufficient flexibility for the operator training facility and nuclear plant engineering development programs by allowing sufficient computer margin for facility operations management.

RCS BASIC EQUATIONS

The basic equations comprising the Advanced Simulator Reactor Coolant System (RCS) model provide for single and two-phased flow during forced and natural circulation conditions. For each non-equilibrium stratified interior node, there are conservation equations for the mixture and gas region mass and energy. The total energy equation is rewritten to solve for enthalpy. An assumption of global compressibility requires that all fluid properties be evaluated using global pressure and the local nodal region enthalpy. For each nodal flow link, there is a momentum equation for the time derivative of the net mass flow rate. Also, there are equations for the pressure and enthalpy of each boundary fluid node and for the net mass flow rate in each boundary flow link. Finally, there are equations for the liquid and vapor mass flow rates in each flow link, and mass and heat transfer rates in each interior nodal region.

Global Pressure

The globally compressible assumption involves using a simplified equation of state in which the fluid properties are evaluated at the global (or system) pressure and at local enthalpies. That is, it is assumed that during the course of a transient, density variations due to pressure changes are small or that local pressures differ little from the global pressure. Since time steps longer than the longest sonic transit time are used, this assumption is reasonable.

The mathematical significance of the global compressible assumption is that the pressure is not an evolutionary variable. At any time, it must be determined simultaneously at all spatial points. An important implication of this to the finite difference solution of the equations of thermally expandable flow is that the discrete (or local) pressures must be determined simultaneously. The method used to calculate link net flow rates and pressure drops is described in detail in Reference 1.

RCS Model Components

The model components consist of a network of interior fluid nodes, boundary fluid nodes, non-critical flow links, and critical flow links. A specific representation of the RCS can be constructed by using the components to form a network of multiple fluid nodes, appropriately interconnected by flow links. The interior nodes provide for mass and energy storage; the boundary fluid nodes provide for pressure and enthalpy boundary conditions; the flow

RCS Model Components (continued)

links provide for mass and energy convection. Figure 2 shows a four loop PWR nodal configuration. An interior fluid node is defined as a fixed control volume containing some mass and energy of fluid. No flow (only mass and energy inventories) is associated with a fluid node. An interior fluid node may be connected with other fluid nodes by flow links. In effect, each node represents a major system component, such as node (4) Figure 2 represents the pressurizer.

A boundary fluid node is defined as a control volume containing fluid at a specified pressure and enthalpy. A boundary fluid node has no volume or mass associated with it. It may be connected with interior fluid nodes by critical flow links. These nodes are used to interface with auxiliary systems such as safety injection, accumulators, etc.

A non-critical flow link is defined by a momentum conservation equation for the net rate of change of the link mass flow rate. No mass and energy inventories (only flow) are associated with a flow link. A non-critical flow link always connects two interior fluid nodes.

A critical flow link is defined as a path for fluid flow where the net mass flow rate is a specified function. A critical flow link always connects an interior and a boundary fluid node. These flow links are used in conjunction with a boundary node to interface the RCS system with auxiliary safety, charging and letdown systems.

RCS NUMERICAL SCHEME OVERVIEW

The following scheme is used to solve the basic equations comprising the Advanced Simulator Reactor Coolant System (RCS) model.

The mass conservation equations for the mixture and gas regions of each interior fluid node are integrated explicitly first. The energy conservation equations (written in terms of enthalpies rather than internal energies) for the two regions of each interior fluid node are integrated implicitly second. A non-linear algebraic equation for global compressability, P*, is solved next. This equation represents the conservation of total RCS volume, i.e., the total volume occupied by the fluid in all nodes equals the total volume of all nodes. Then, the time rate of change of mass in each node is found using a method which assures mass conservation at the system boundaries and corrects any nodal volume error. Next the total mass conservation equation in all interior nodes and the momentum equations in all non-critical flow links are solved simultaneously for the net mass flow rates and pressure drops in all non-critical flow links. Finally, the liquid and vapor mass flow rates in all flow links are obtained from a drift flux model. The above numerical and solution techniques are more fully described in Reference 1.

Advanced Simulator Reactor Core System

A two-phase flow, single channel fuel rod model has been developed for the Advanced Simulator modelling in order to simulate normal reactor operations and postulated LOCA accidents which are limiting for simulation by a single phase flow model.

The fuel rod thermal calculation is based on a five radial node model coupled to a single average point in the coolant fluid. For each time step, thermal conductivities, heat capacities, heat transfer coefficients, and fuel gap width are updated to reflect the current temperature condition. The active core region is further divided into four segments such that two-phase flow mixture level can be defined. Various heat transfer phenomena exist in the clad outer boundary. Heat transfer correlations for forced convection, nucleate boiling, transition boiling, and film boiling are incorporated in the model calculations. The heat fluxes from the fuel rods to the coolant fluid are calculated for each axial segment and are integrated to yield the total heat fluxes to the mixture region and gas region for the Reactor Coolant System (RCS) thermal calculations.

Steam Generator Model

The steam generator model consists of 5 nodes per generator. Nodal components are for the downcomer, subcooled non-boiling region, boiling region, riser region, and steam dome region. The present version of the model provides for system blowdown, feedwater, and steam line and tube breaks.

Transient Results

The math model described has been applied successfully in modelling an RCS model for the Westinghouse Advanced Simulator. It has been verified via selected single and two phase transients as listed below:

Results from a typical 1-inch break are shown in Figures 3 through 5.

Results for a 4-inch LOCA are shown in Figures 6 through 8.

Results for a "St. Lucie" type cooldown to natural circulation are shown in Figures 9 through 10.

SUMMARY

In the past few years increased emphasis has been placed on the engineering aspects and systems response of nuclear plants under transient conditions. The NRC requirements have changed regarding personnel who must be in the control room (including a Shift Technical Advisor), and the type of simulators required to train these personnel.

The Advanced Simulator system models and hardware provides an effective tool for training plant operators as well as other technical support personnel.

ACKNOWLEDGMENTS

The authors wish to thank Harry Julian of Westinghouse NTD for his assistance in providing input information and analysis of the transient data presented. We, also, wish to thank Thierry Constantieux of Framatome for his assistance in the development of the RCS system software and generation of the model verification transients.

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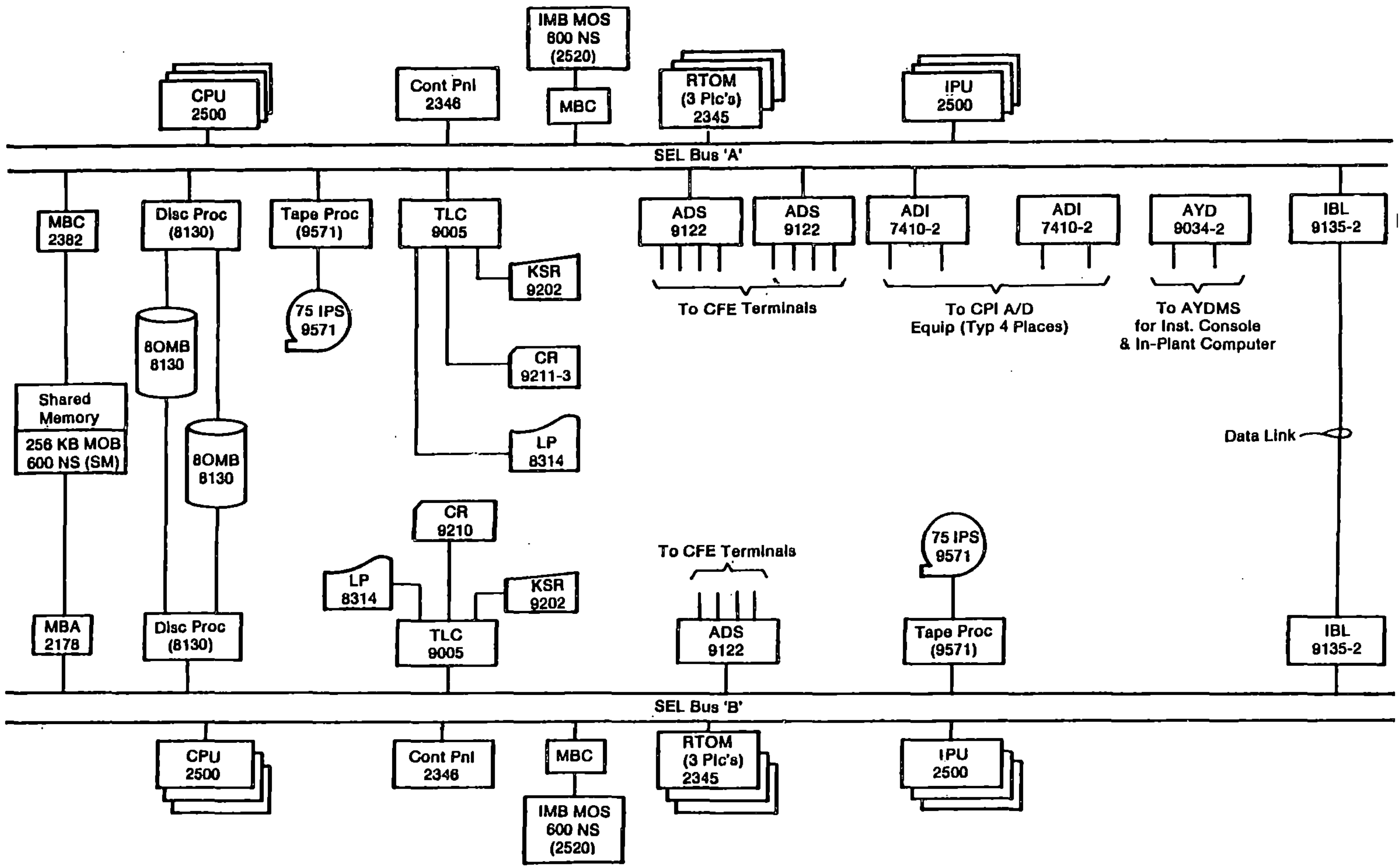

Figure 1.

T19659-1

FOUR-LOOP RCS MODEL

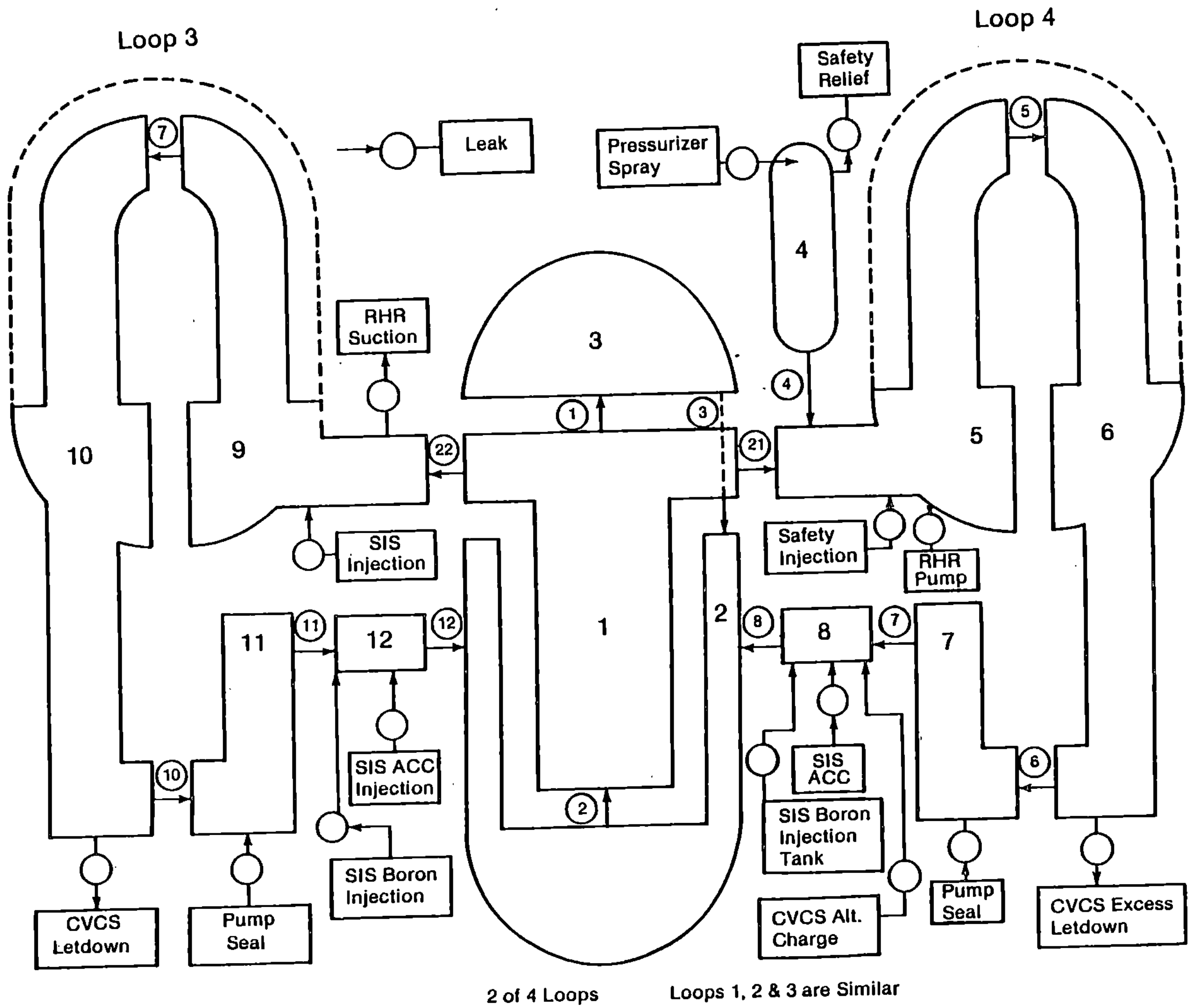

Figure 2.

T19659-2

TYPICAL 1-INCH COLD-LEG BREAK — TIME VERSUS PRESSURIZER PRESSURE

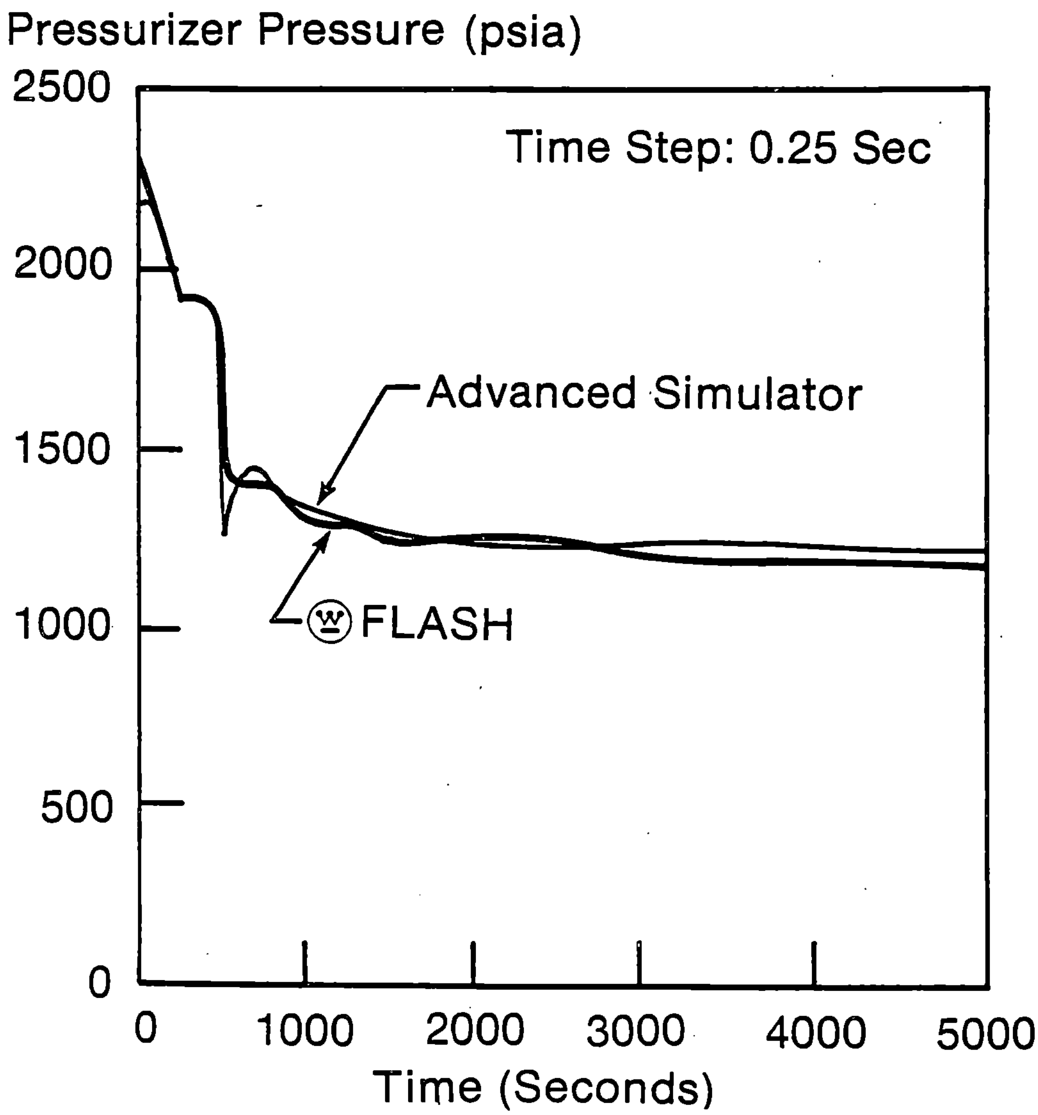

Figure 3.

T19659-4

TYPICAL 1-INCH COLD-LEG BREAK — TIME VERSUS BREAK MASS FLOW RATE

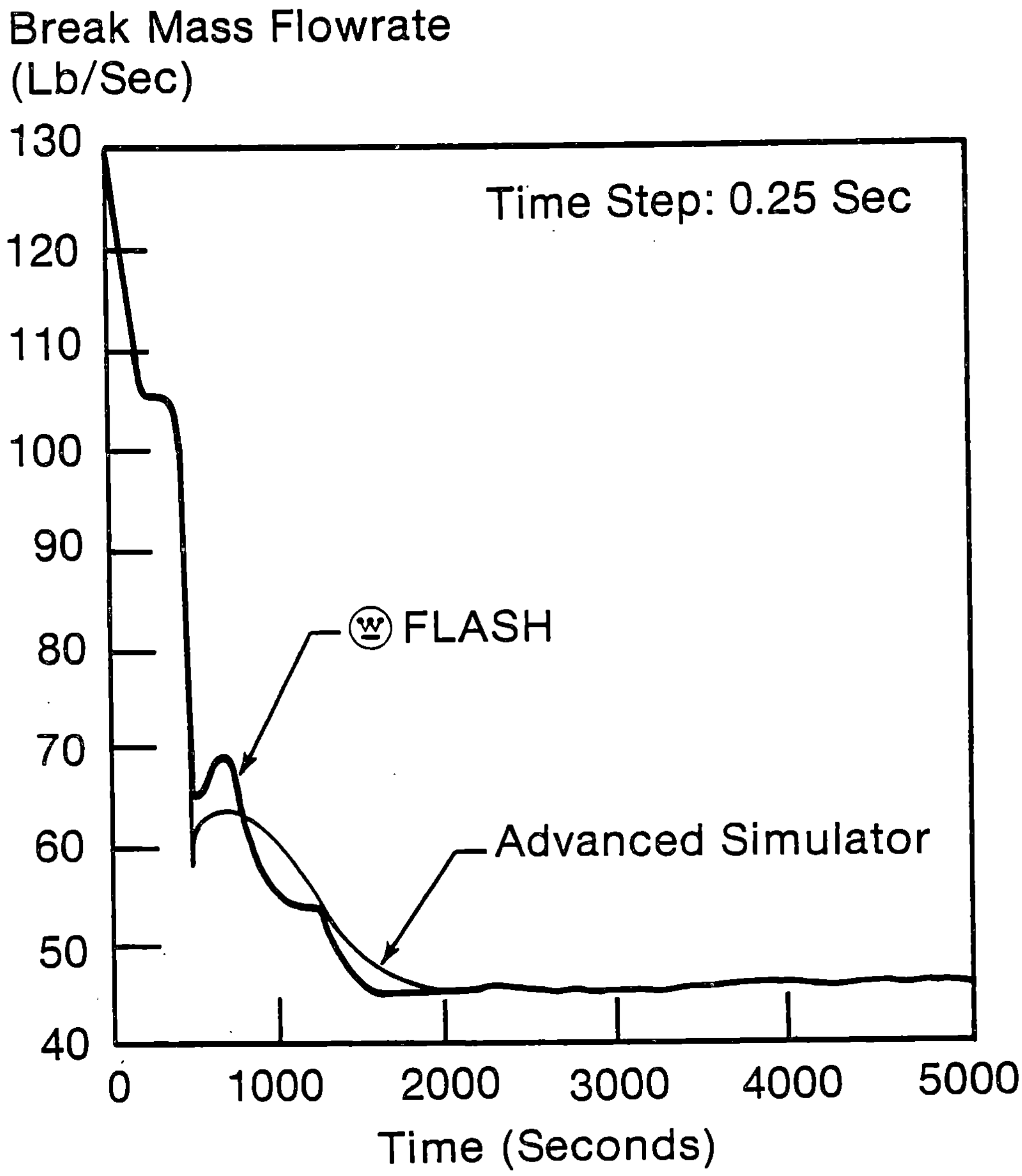

Figure 4.

TYPICAL 1-INCH COLD-LEG BREAK — TIME VERSUS CORE MIXTURE LEVEL

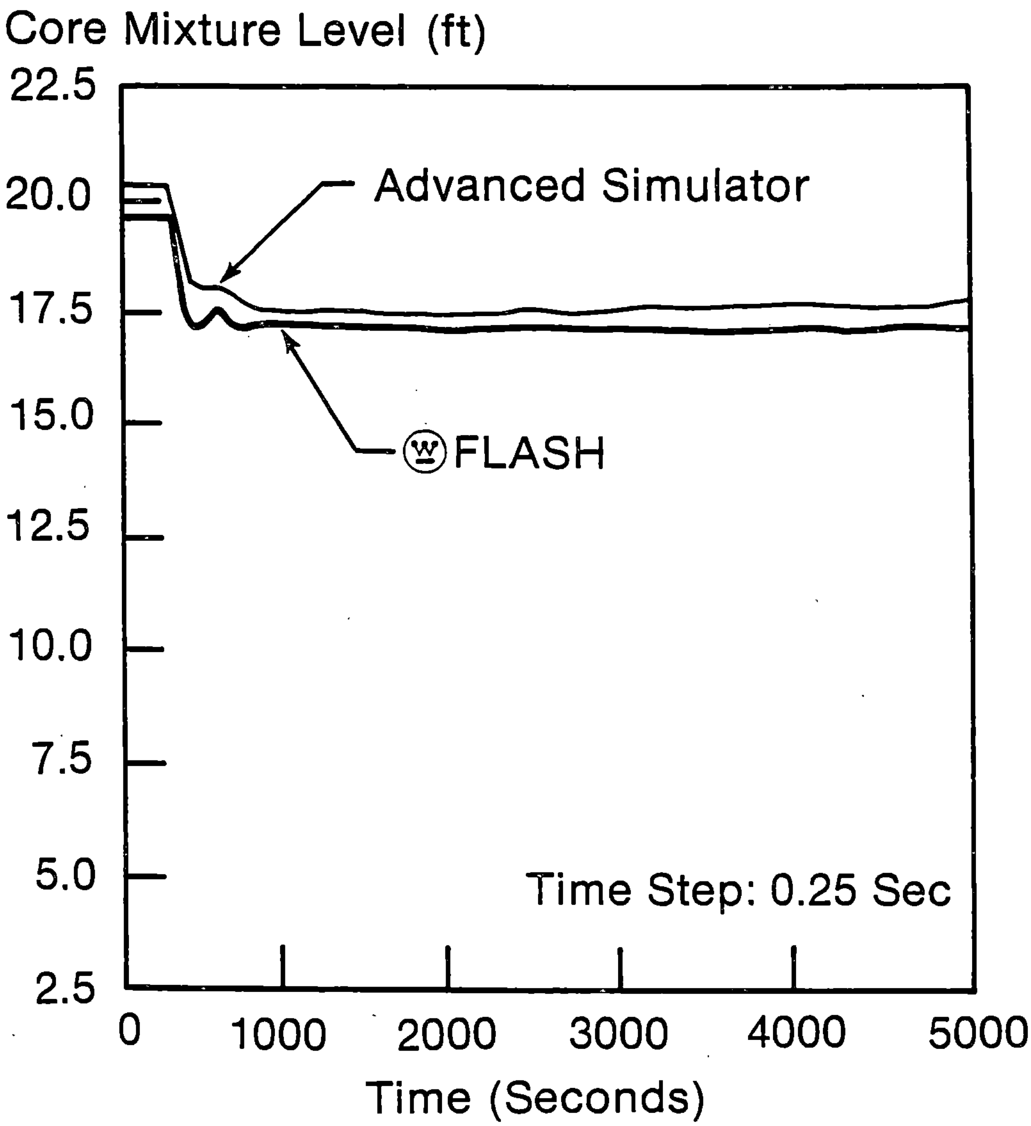

Figure 5.

PRESSURIZER PRESSURE FOLLOWING A 4-INCH COLD LEG BREAK

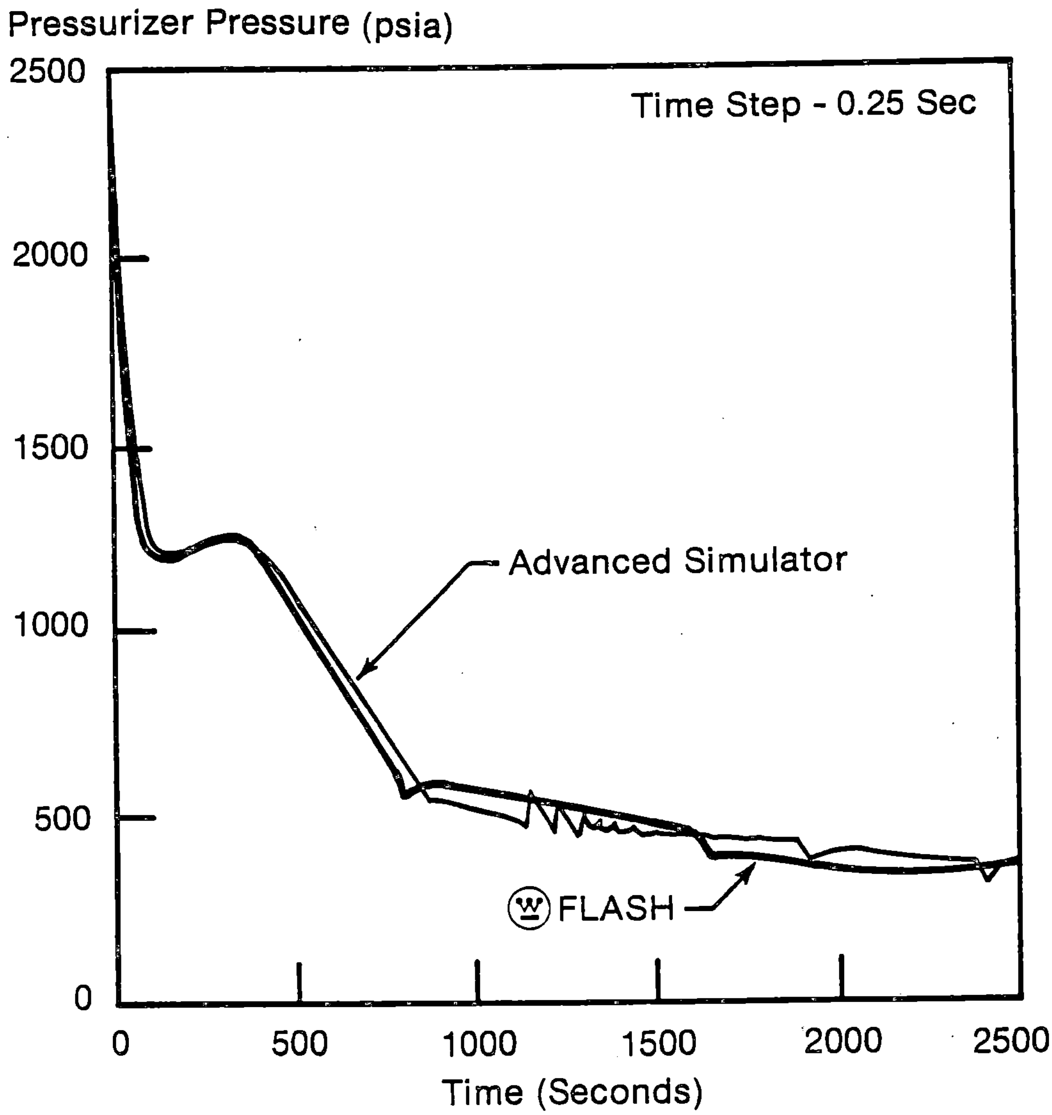

Figure 6.

T19659-7

CALCULATED FUEL CLAD TEMPERATURES FOLLOWING A 4-INCH COLD LEG BREAK

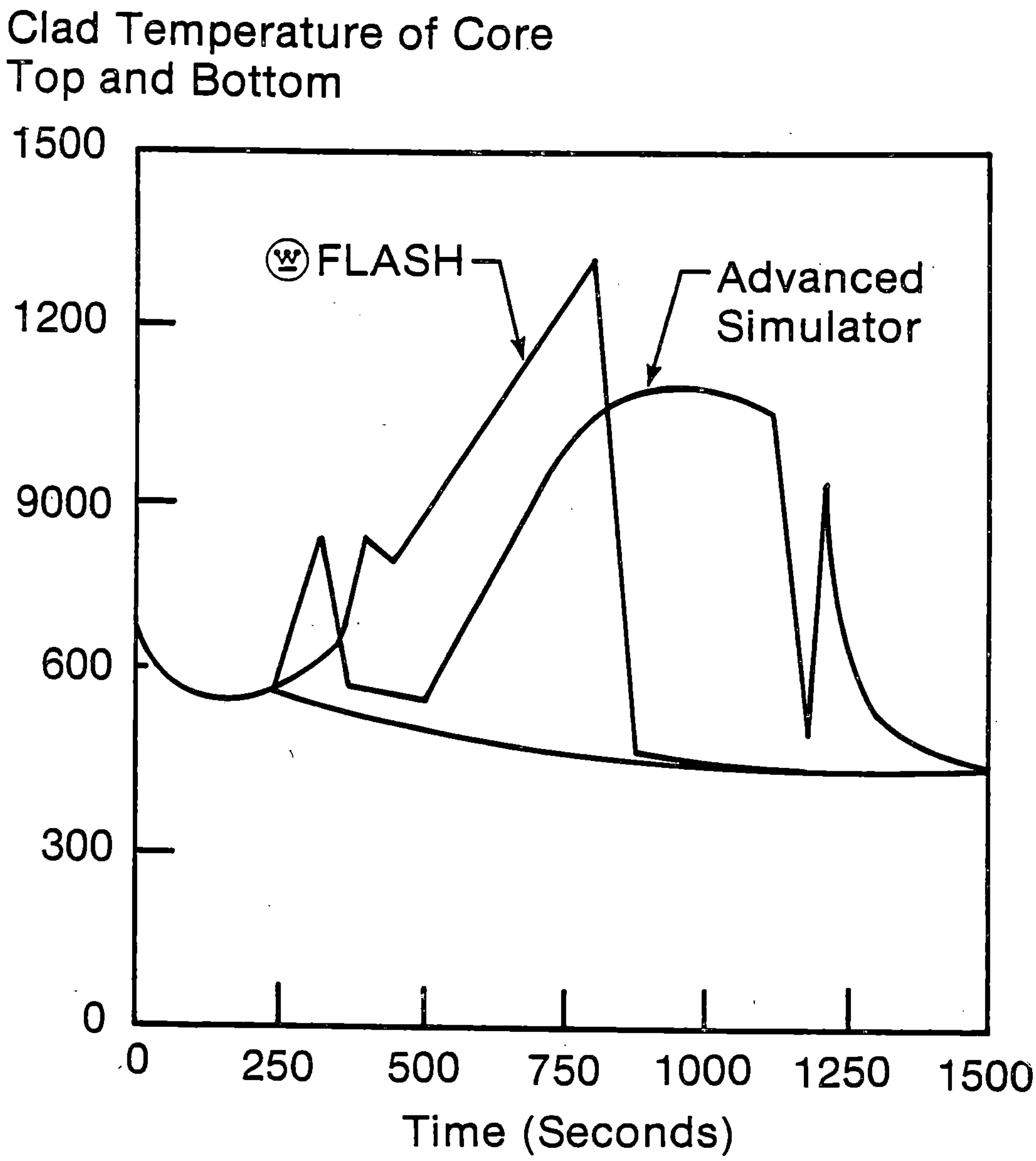

Figure 7.

T19659-6

CALCULATED CORE MIXTURE LEVEL FOLLOWING A 4-INCH COLD LEG BREAK

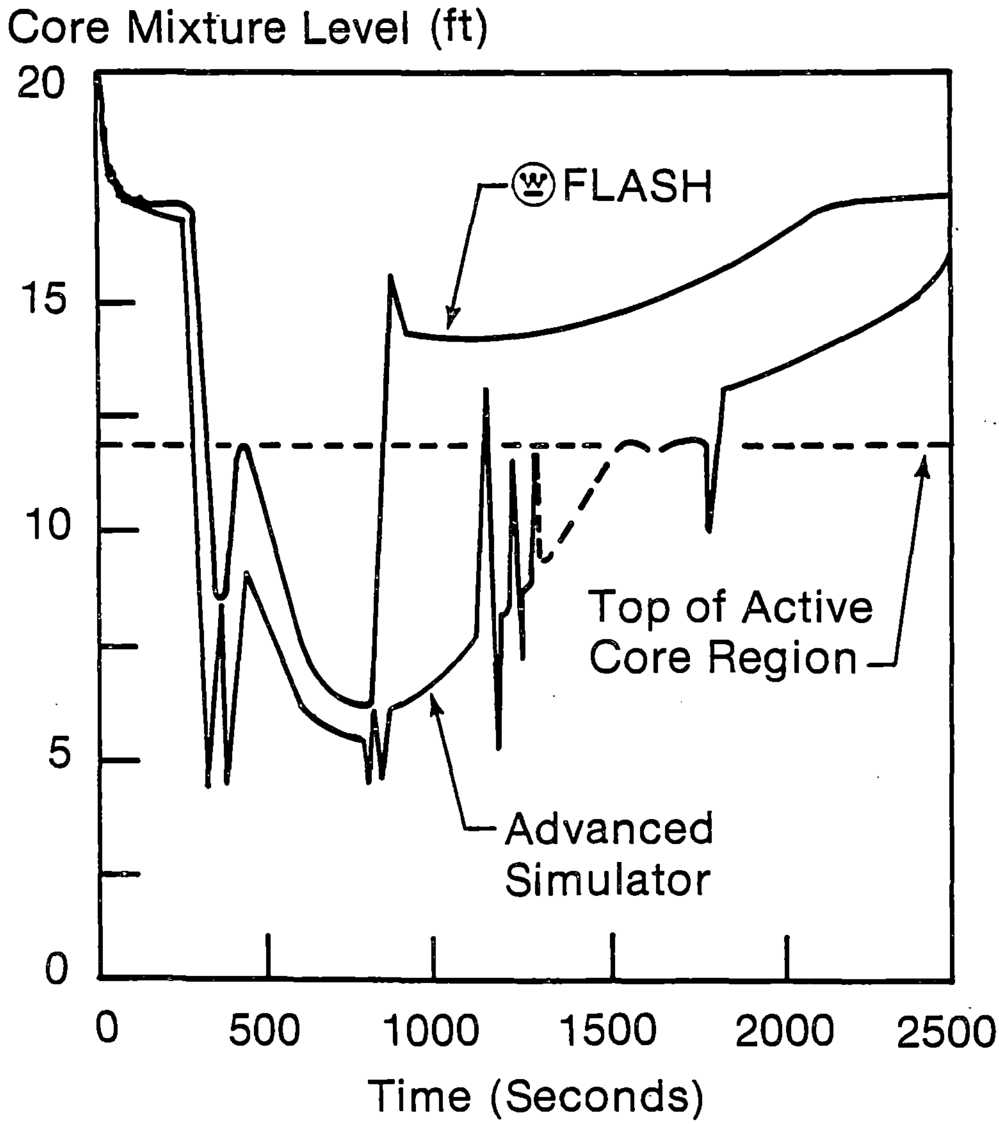

Figure 8.

PRESSURIZER PRESSURE DURING A "ST LUCIE" TYPE EVENT

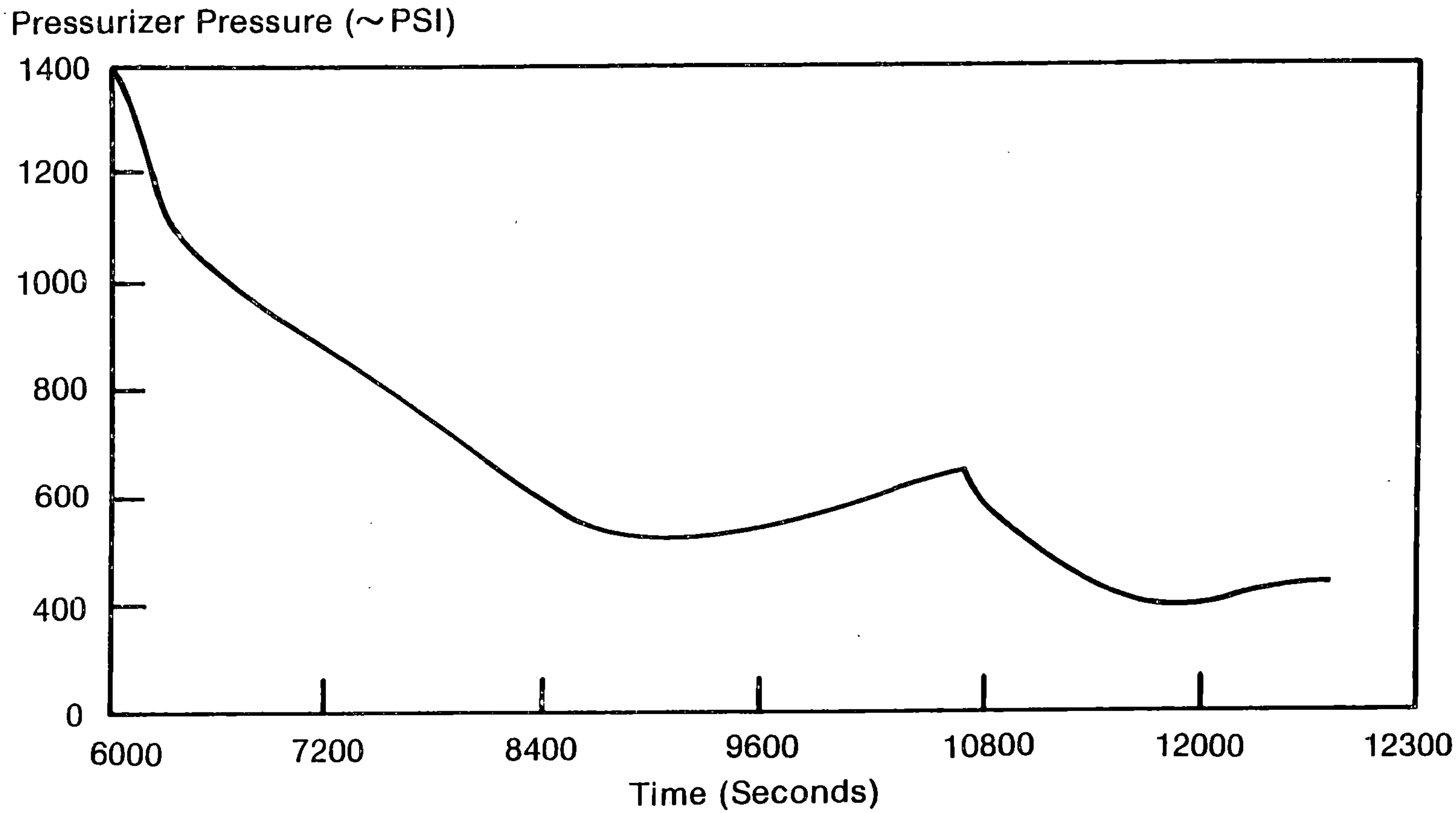

Figure 9.

T19659-9

PRESSURIZER MIXTURE LEVEL DURING A "ST LUCIE" TYPE EVENT

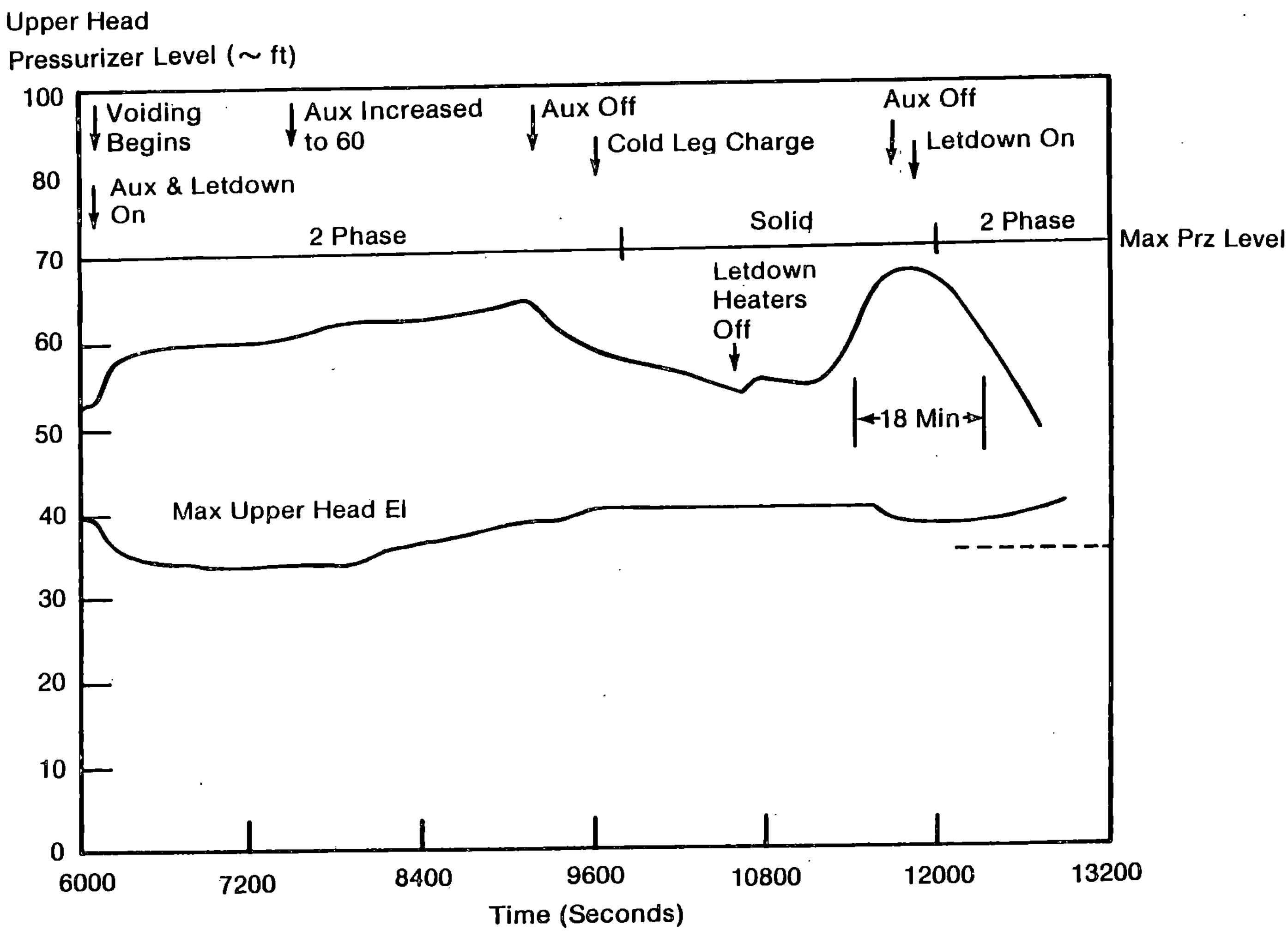

Figure 10.

T19659-10

QUESTIONS TO F. C. LUFFY

Gary Grant

Q:

1. What changes have been made to the core models in the advanced simulator?
2. Can you address degraded core conditions?

A:

1. We have improved the core thermal hydraulics (single channel fuel rod model includes fuel, clad, and coolant temperature calculations) to be compatible with two-phase flow. We are also improving other functional modules relating to the core, such as one dimensional space kinetics, decay heat, cross-section calculations, transverse X-Y flux tilt, Xenon and Iodine concentration, thermocouple temperature calculations, and delay neutrons.
2. At this time we do not have that capability.

REMARKS OF WILLIAM S. LEE

PRESIDENT, DUKE POWER COMPANY

EVENING BANQUET, OCTOBER, 13, 1981

This week marks the second birthday of INPO; INPO was born out of a new realization and acute awareness that we are all in the same boat. No nuclear plant stands alone. An event at any affects all. Yet, no nuclear plant is like another. Each organization has its own unique combination of strengths and weaknesses. INPO exists to identify each of our strengths and our weaknesses and then to transfer strengths to displace weaknesses wherever found.

Your presence here is an expansion of this concept of sharing in the search for excellence--from plant to plant and now from nation to nation. We face together the challenge of the quest for excellence--a quest that knows neither the boundary of a plant fence nor an international border. Therefore, I am encouraged by your participation in this meeting--your recognizing that we are our brother's keeper. We learned from TMI that an accident in one country can seriously affect others. There is no hiding from our common responsibility to help one another--to make nuclear a safe and reliable energy resource worldwide. To do otherwise may jeopardize the health and welfare of our publics, the large capital investments, the future role of nuclear power to serve all mankind, and the proud history of international contributions.

I emphasize the word "international" because this industry was only made possible by the contributions of many scientists, educators, designers, and operating personnel all over the world.

- The German-born Einstein made discoveries fundamental to the whole subject of atomic energy while working in a Swiss patent office.
- Madame and Pierre Curie startled the world with findings from a small lab in France.
- Others including Bohr, Cockcroft, Teller, and Fermi are further testament to what can be accomplished through shared knowledge and experience.

I am glad to see this tradition continuing here today because improving operator training and qualifications is so vitally important to our providing assurance that nuclear plants are operated safely.

In this mutual quest for excellence, you are exchanging ideas about operators.

- their selection
- their qualifications
- their training
- their education
- their use of simulation techniques
- their interface with machines
- their continual upgrading and requalification

If this exchange proves useful to you--if each of you goes home with an idea of how to strengthen your team--then this conference has been a success and serves as a constructive forerunner of future cooperative endeavors together in our mutual quest for excellence in the whole spectrum of nuclear energy. This experience can begin a renewed spirit of international cooperation.

Not emphasized in your program are two very special challenges that we have together. One is the supply of needed manpower, and the other is the gap between perceptions and realities.

Even the very best training programs are of little value if you do not have enough people to fill the classrooms. There is a worldwide shortage of qualified people to plan, design, build, test, regulate, and staff nuclear plants. In some countries, including the United States, the problem is critical. Our nuclear engineering schools have empty seats. Perhaps young people perceive a slackening of career opportunities in an industry where new commitments have slowed or stopped. Perception of nuclear risks by the young may be having a negative influence on course and career selection. Most of us failed in our manpower planning to anticipate the galloping regulatory requirements demanding more staff, more staff, and yet more staff. In some cases, ambitious construction programs were not accompanied by adequate personnel planning for start-up and operation. We are also finding that qualified people have become disenchanted with increasing personal pressures and are seeking careers outside the nuclear arena.

The solution to the manpower problem requires coordination within the nuclear community; those involved must communicate and share information and resources. First, our leadership must recognize that the problem is not unique to one plant or one utility but is widespread. Each entity has the clear responsibility to provide its full share of training and education rather than steal from others, which only spreads the shortage and makes the shortage mobile and self-perpetuating. Each must also be willing to give on-the-job training to less experienced employees or newcomers to nuclear ownership. We all must work more closely with our educational institutions to give them the tools to attract students. This can be in the form of grants, scholarships, vacation employment with utilities, and coordination of course content. We must provide employees incentives to reduce turnover rates and show them their clear path of progression from entry level to as high as their capabilities can carry them. We can solve the manpower problem if the highest-level executives will recognize its priority and, through action, make it happen.

And now to the gap between perception and realities. Its solution will help the manpower problem as well as permit nuclear energy to play its necessary role in meeting the world's energy needs.

Let me illustrate the gap. The sidewalks are packed with pickets carrying banners and placards of protest. Shrill voices decry the new and dangerous energy technology. It can KILL! The electronic and print media carry inflammatory reports of the protest and of the danger. Investors panic, the company's stock plummets, and the New York Stock Exchange suspends all trading in the company's securities. People do not understand the new energy technology nor how its dangerous and unseen emanations affect the human body, much less how they are measured. The public is frightened.

Sound familiar? The scene is Philadelphia. The year is 1879 - - 102 years ago. The company was Wannamakers Store that had just announced it would replace gas lights in their store with electric ARC lights. The new technology was electricity, and it was widely known that electric shock could be lethal. The telegraph and newspapers spread the news. It was difficult to understand voltage and electromagnetic radiation. The perceived risk to public health and safety brought about by electricity created fright.

Today, a century later, the public understands the benefits of electricity, as well as its dangers. Even though it takes hundreds of lives each year, electricity is now a widely accepted and even essential energy technology. Because it is now well understood, the gap between perceived risk and real risk has disappeared.

Today, the new technology is nuclear energy. Having been introduced to the world in the form of a horrible weapon, the very subject is fearsome to many. I personally am convinced that nuclear energy's benefits outweigh its risks, and to do without it would bring about catastrophic economic and social consequences to the world. Yet, we will not have the nuclear option without public conviction that it is the wisest choice. With strident voices in opposition in the aftermath of the trauma at Three Mile Island, the public is confused and concerned. They need the facts told sensitively, forthrightly, and accurately. Public policy decisions about energy will reflect public opinion. For wise decisions, we need a broad understanding of the facts and the choices, not hypochondria. The greatest drawback of nuclear power is its complexity and mystery, which if not clarified, may cause society to forego the nuclear option. Professionals and opinion leaders like you must tell it like it is. You must place high personal priority in communicating. We must do all we can to increase public understanding about energy options. If we do not do it with vigor, they will only hear other voices, progress towards our economic and social objectives will be halted, and calamitous world policy may result.

As leaders and professionals, I call upon you to become familiar with all energy issues and to actively inform others. For if the public is not armed with the facts, society may make wrong policy decisions. You and I will not make the decision as to whether nuclear energy--or any other alternative--is acceptable. Public policy decisions about energy will reflect public opinion through the political process. These decisions will be wise only if based on broad understanding of the facts and the choices. You and I can contribute to that understanding, not only by what we do about safety, about operators, but how hard we personally work at helping others understand. We can make nuclear safe. We can also close the gap between perceptions and realities.

SUMMARY

SESSION IV

SELECTION AND REQUIREMENTS

J. B. FECHNER

The four papers presented during Session IV treated subjects from the areas of selection, competency development, and human factors research.

Major characteristics and the benefits of a plant operator selection battery developed within 30 months by the Personnel Decisions Research Institute and the University of Minnesota were outlined. The 3-hour battery, which consists of a previous experience questionnaire, a series of brief ability tests and a personnel questionnaire, covers knowledge, skills, and personal stability of the candidate operator, summarized in an overall potential index. The selection battery has been validated on the basis of supervisors' ratings of job performance of 3,336 job incumbents from nuclear, hydro, electric and fossil-fixed plants. Four job performance scores (criteria) were used for the rating: emotional stability, operators' competence, problem-solving ability, and overall performance. The battery is to be applied at entry level.

The competency system developed by the Ergonomrad AB for Swedish nuclear power plants was described. It constitutes a set of closely related requirements and actions aiming for the acquisition and maintenance of the operators' ability, to meet adequately all situations and states of the plant including malfunctions, transients, and accidents. This system has been accepted by the Swedish Nuclear Power Inspectorate as a substitute for licensing each individual operator, and it has been implemented by the plant owner since mid-1980. The system was developed by starting with job descriptions (about 150 typical tasks), identifying job requirements, and specifying knowledge and skill requirements for shift supervisors, control room and turbine operators. The implementation of specific training and retraining programs, including an appropriate training organization, is in progress. The job training will be founded on five years' basic technical training.

Scope and content of the associate degree in Nuclear Engineering offered by the Pennsylvania State University since 1970 were presented, including the contents of the six relevant series of the two-year program and the laboratory facilities used. The courses are intended to develop a broad and basic technical knowledge, to improve the individual's communication skills, the written and oral skills, and the cognitive problem-solving skills. The program is one of the three existing programs accredited on the basis of the ANS - Accreditation Board for Engineering and Technology (ABET) criteria. It can be enlarged toward leading to a baccalaureate degree through two years of additional work.

A partial progress report was given for some of the areas in which the Human Factors Society has investigated (for NRC) to develop a comprehensive long-term human factors plan. Diverging industry and NRC views have been identified for many subjects, the reason in most cases being the lack of objective measures of effectiveness. Lack of objective performance criteria seems to have led to changes of the design of control rooms and procedures upon NRC request, which did not take into account overall performance. The operational relevance of scope, contents, and scoring of licensing exams for operators were questioned; a relationship between licensing scores and job performance needs to be established. No reliable requirements or data relevant to the optimal duration, changeover, and work/rest problem for shift work were found to be available. Management attitudes and practices, which do not provide for credit or recognition of the operator's work, need to be drastically improved - together with better payment for shift work - in order to raise the operator's motivation. Operator error reporting needs better identification of root causes.

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PAPER IV-1

A METHOD FOR OPERATOR COMPETENCE DEVELOPMENT IN NUCLEAR POWER PLANTS

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Abstract

The paper presents a method for operator competence development. Competence is defined as the ability of the operator to meet adequately all situations and states of the plant including normal operation, disturbances, transients and accidents. A set of closely related requirements and actions aiming for the acquisition and maintenance of operator competence is called the "competency system". There are the following parts in this system:

- Job description and job requirements.
- Recruitment requirements.
- Job training content development.
- Job training programme development including courses, training means and follow-up procedures.
- Requirements on plant training organization including training management and instructors.
- Registration, storing, processing and reporting of competency data.
- Recurrent checks and revisions of the "competency system".

A description is given of the procedures to develop a "competency system" under the following headings:

- Specification of some steering factors for a particular "competency system" solution, including present operator recruitment and job training etc.
- Specification of job requirements through job analysis and descriptions.
- Specification of knowledge and skill requirements for operating the plant.
- Specification and implementation of a job training programme including training resources and a training organization of the plant.

The development of the competency system for Swedish nuclear power plants was organized in work groups with participation from the Nuclear Power Inspectorate, the utilities and Ergonområd AB.

Although it is too early to draw any firm conclusions from the use of the competency system some statements about the method as such and from the work with it in the plants can be given:

- The terminology, e.g. on state diagram, typical tasks, knowledge objects is easily understood by different specialists in the plant.

- The terminology and the structure offered by the competency system facilitates communication between plants and between the plant and the safety authority.

- Operator job training has gained in recognition among plant management probably because the competency system makes it easier to see important relations between operator training and plant operation.

Background

Operator licensing for nuclear power plant personnel has been enforced by safety authorities in several countries, e.g. in the U.S.A. since 1954 and in FR Germany since 1974. In Sweden the Atomic Law has regulated nuclear power since 1959. This law gives the complete responsibility for safety to the reactor owner. It means that the reactor owner is also responsible for the personnel who operate the plant.

In 1976-77 the Swedish Nuclear Power Inspectorate carried out a feasibility study on the regulation of operator competence. In July 1978 a project was given to Ergonområd AB to develop principles for operator competence under the condition of the Atomic Law and under the condition that there are utilities with different operational organizations, operator recruitment and training traditions.

A method including principles and procedures for specifying operator competence was developed and has been implemented in ten Swedish nuclear power blocs since mid-1980. The work is done on behalf of the Nuclear Power Inspectorate which is the governmental authority for nuclear safety. The work is carried out in close cooperation with the utilities.

The method has a wider applicability than to nuclear power plant operators and can be used for safety and availability purposes in other processes and in other operator jobs.

Aim of the paper

This paper is a technical presentation of a method for the acquisition and maintenance of operator competence taking the above mentioned conditions concerning law and utilities into consideration. Some principles behind the method, procedures and an organization for the work involving the method are presented.

Operator competence

Competence in this context is defined as the ability of the operator to face adequately all situations and states of the plant including normal operation, disturbances, transients and accidents. The overwhelming proportion of these situations involve one or several processes and technical systems. The operator and his shift collegues have to face this. But it is important to realize that operator competence is not limited to the technical side of the plant. It also engages professional contacts with other specialists from inside or outside the plant, e.g. maintenance people, fuel specialists, radiological specialists and guard of the plant. There is also an operator interface with documentation, regulations etc.

What builds up operator ability? The answer is, of course, complex. But a set of knowledge and skill belongs to the most important factors. This knowledge and skill is a necessary condition for success in the job. Thus, it is an important step to analyse and define them.

There are factors other than knowlege and skill which may play an important role in adequate operator behaviour, e.g. personality factors, control room factors, work procedures and work organization factors but these factors are not taken into consideration in relation to operator competence. Some of the factors ought to be considered in separate studies and developments like control room design and man-machine communication. Other factors are difficult to take up because of lack of interest, opposition or large resources needed. Operator personality is probably such a factor. It may be very important under certain conditions, e.g. in stressful situations. But it is difficult to define relevant personality dimensions. It may cost quite a lot to develop adequate tests or measuring devices and there is probably a strong resistance among operators to them.

The purpose of the competency work was plant safety. But contrary to the distinct technical safety features, e.g. barriers in the plant, consequence reduction systems, redundancy and diversification of systems, it is not possible to separate between what is safety related and what is not safety related in the operation of the plant. It is a well documented experience from nuclear power plants that safety incidents or accidents often start with something which is not safety related. Through interrelations between systems a non-safety related incident may develop into a safety incident or an accident. Because of this, no distinct separation was made between safety and availability related operator competence.

A system concerning competence

A method for operator competence consists of a set of interrelated concepts which are called "the competency system". Part of the work has been allocated to the development of concepts and a structure for this system (1). The main features of it are described in figure 1.

A fundamental principle in the present method is that competency requirements of the operator should be based on what the operators actually do in their jobs. If one knows what the operators do in their jobs and knows the more important job circumstances like timing, task frequency and load one can also derive knowledge and skill needed for the operator to be able to carry out his tasks successfully. Thus, a starting point is the job analysis and job description. This is input for considerations on recruitment requirements and on job training requirements. The job training requirements are met through a job training programme, which contains all courses and other types of training which are needed to fulfill the training requirements. A training organization with sufficient resources to realize the training requirements is also an important part of the "system". Thus, the "competency system" contains the following parts.

- Job descriptions and job requirements.

- Recruitment requirements.

- Job training content development.

- Job training programme development including courses, training means and follow-up procedures.
- Requirements on plant training organization including training management and instructors.
- Registration, storing, processing and reporting of competency data.
- Recurrent checks and revisions of the competency procedures.

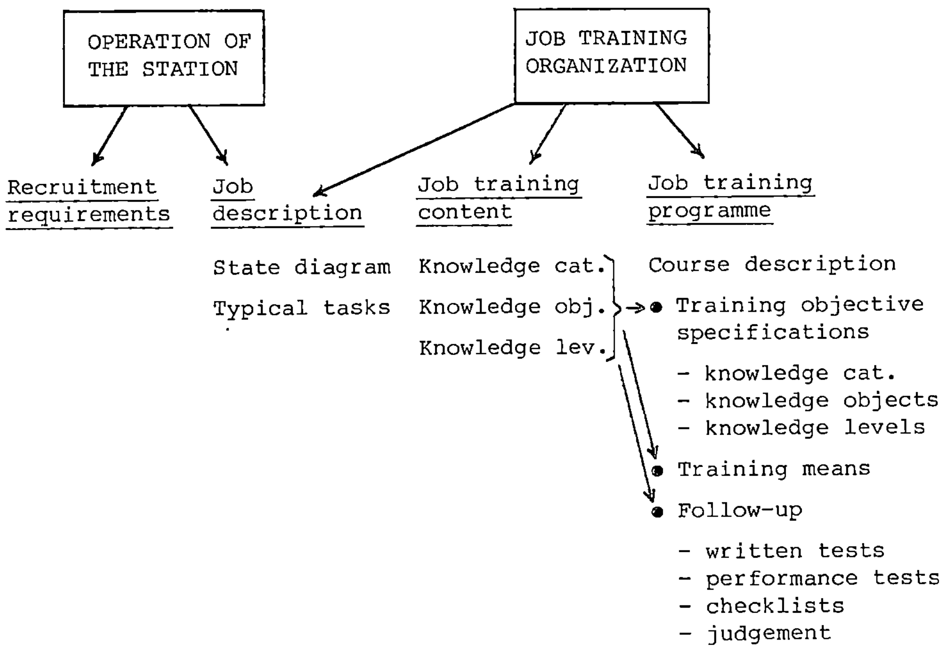

Figure 1. Competency and training for nuclear power operators.

Competency specification procedures

There are four main procedure steps in applying the competency method to particular operator jobs.

1. Specification of steering factors for a particular "competency system" solution.
2. Specification of job requirements through job analysis and job descriptions.
3. Specification of knowledge and skills requirements for operating the plant.

4. Specification and implementation of operator job training programme(s) including training resources and the training organization of the plant (or other organizations involved in operator training).

Competency steering factors

Some important factors which will influence the particular competency system solution must be identified at the very beginning of the development. To a large extent these factors belong to what can be called company policy and inherited factors. These factors can be identified through describing certain present state conditions within the company. They should be considered in the competency work either through accepting them as steering factors or through changing them when they are hindering efficient operator competence planning and development.

Information about these factors are found within the utility or the plant to which the analysed jobs belong. Data can be collected in interviews with plant or site management in the following factors:

(1) Present operator recruitment; requirements, principles and procedures, estimated need for future operator recruitment.

(2) Present operator training; basic operator training programme, retraining programme, courses, training aids like simulators.

(3) Present training organization; who takes care of training within the organization and how, availability of instructors and training time.

(4) Present operator situation; number of operators, operational staff organization.

(5) Present operator competency; competency levels and categories, amount of on-the-job experience.

(6) Objectives for operator competency; safety, availability and/or job satisfaction objectives, safety authority's, management's and operator's view of operator competency.

Procedures for the specification of job requirements

The job analysis technique used has been reported elsewhere (2). It is illustrated in Figure 2 which says that the body of knowledge needed to carry out a job can be defined through a limited number of tasks. These knowledge loaded tasks are called typical tasks. Together with some job situation demands like timing and precision a representative set of typical tasks forms the job requirements.

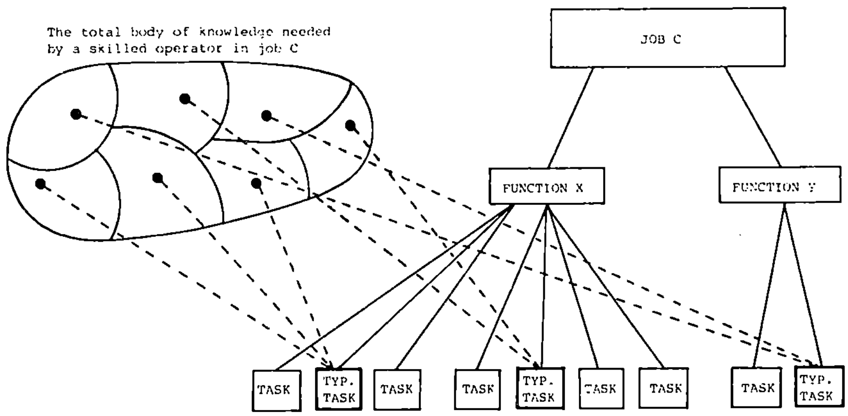

Figure 2. Typical tasks of a job and the body of knowledge.

The procedure to generate typical tasks is based on a system analytical approach. Starting in the analysis of the power generating system, which is described in a so called state diagram (see Figre 3) containing distinct plant states based primarily on the situation in the reactor and in the turbine of the plant.

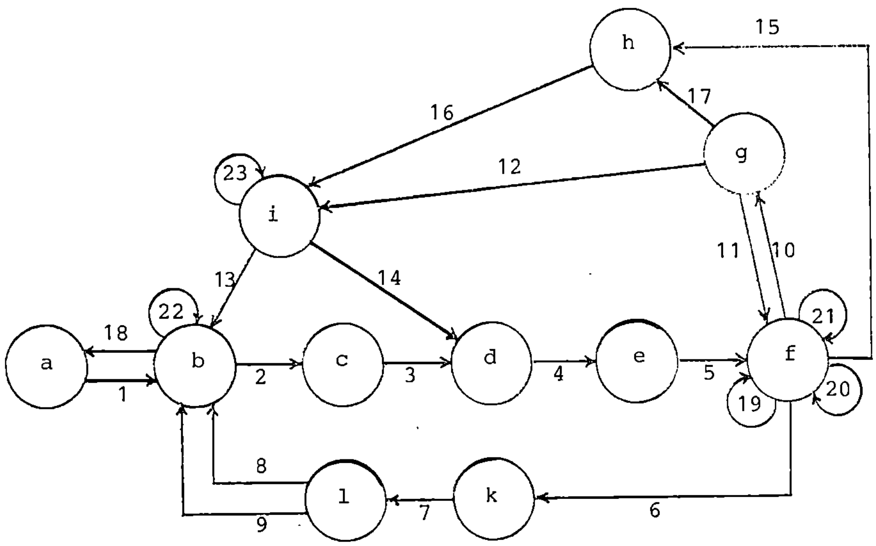

Figure 3. States and procedures for a BWR nuclear power plant (see text below).

There are eleven distinct states:

- a Reactor after refuelling
- b Cold subcritical plant
- c Heated subcritical plant
- d Hot critical, reactor power 5% turbine not running
- e Turbine at nominal speed
- f Normal operation
- g Disturbed operation
- h Emergency operation
- i Hot, tripped, subcritical reactor
- k Hot, critical reactor, power 5%
- l Hot, subcritical reactor

State transitions can be produced through manual or automation process control or through disturbances in the process or in the control system. The operator tasks are subunits to state transitions or in activities aiming for the preservation of some state:

Transition	Procedure
1.	General plant preparation
2.	Preparation for start up (heating of reactor using residue heat)
3.	Start nuclear heating and increase power to 5% Start aux. feedwatersystem to control waterlevel in reactor tank Heat steam pipes and continue nuclear heating using control rods Dump steam to condenser
4.	At 5% power, switch from aux. feedwater system to feedwater system Bring turbine to nominal speed
5.	Syncronization and loading of generator Increase generator power to 20%
6.	Decrease power to 5%
7.	Shut down to hot subcritical reactor
8.	Cooling by dumping steam to condenser

9. Cooling by dumping into containment

10. Incident causing disturbed operation

11. Return to normal operation after disturbed situation

13. Cooling of subcritical reactor

14. Start up of hot reactor

15. or 10. 17. Incident causing emergency situation

16. or 12. SCRAM or manual shutdown

18. Refuelling

19. Change of control rod pattern

20. Increase or decrease of power level

21. Change of shift

22. Maintaining state b (residue heat cooling)

23. Maintaining state i to be able to perform transition 14 later

The state diagram is the basis for task generation, which preferably can be done through interviews with plant operators, supervisors, operator instructors and system engineers. Operators are an important source of task information when there is plenty of operational experience. If operational experiences are lacking, process engineers and control system engineers become the most important information source.

The job analysis is conducted in four phases.

<u>Phase 1 - descriptions of the main system</u>: The aim of the description is to identify all interaction surfaces between the operator and the main system. The main system is not limited to the technical system for direct power production - which is the object in, e.g. the state diagram - but comprises the complete plant. Thus, the main system is considered an organizational system in which the technical process of power generation is a subsystem. To find the interaction surfaces of the three nuclear power operator jobs the main system was described in operational terms, in technical terms and as an organization.

Information about the goals of the plant should be collected, too. As a general rule quantitative as well as qualitative goals like "80% plant availability" can be used to generate more precise operator requirements which can be useful, e.g. when training requirements are derived.

The work in this phase starts with the localization of the analysed jobs. It is important to get a clear picture of where the jobs are situated within the organization of the plant. The analyst also has to get an overview of the general content of the jobs. Does a job include operation, maintenance, planning, supervision or other activities? The answer to this question indicates the type of system descriptions needed for the derivation of all interaction surfaces between the operators and the main system.

Phase 2 - task generation: The aim of the second phase is to generate operator tasks. This can be done in several ways. The most convenient way is through interviews with operators, supervisors and other personnel who cooperate with the operators. A matrix, such as Figure 4, can be used to guide the overview. Along one of the axis of the matrix there is the state diagram (or mission profile) of the main system. Along the other there is a number of possible interaction surfaces like systems, documents and other personnel.

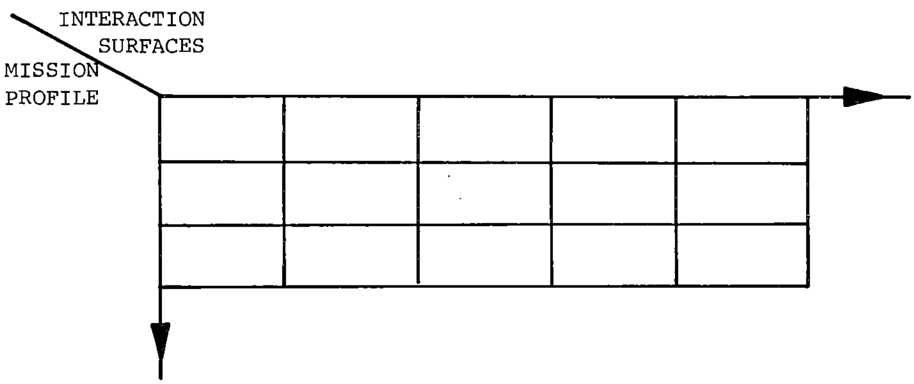

Figure 4. A first matrix used for operator task generation.

Phase 3 - job structure generation: The aim of this phase is to formulate operator tasks which are even, i.e. have the same or nearly the same degree of resolution. A rule of thumb is that the task statements should tell what is done in the task. A statement which tells how the task is done is too precise and means that the job content can not be described with a reasonable number of tasks. The typical tasks will also be organized according to what they will be used for. The state diagram makes them easy to communicate with the operational staff and training planners of the utility. All the collected task statements should be evaluated by an experienced operator. It is important to reformulate statements which can be missunderstood.

The outcome of this phase is a set of preliminary typical tasks, which will represent all operator functions or main activities of the analysed job.

Phase 4 - performance requirements generation: The aim of the fourth phase is to evaluate the relevance of each preliminary typical task and to formulate performance requirements. This can be done through another contribution from the interviewee mentioned above in phase 2 - task generation. Each typical task is judged on relevancy and operator performance. Also in this phase badly formulated tasks can be reformulated and a few new tasks can be added to the set of preliminary tasks.

The typical tasks generated in the job analysis are the main result of the job analysis. The set of typical tasks in a job also covers the knowledge and skill content of the job as was illustrated previously in Figure 2. If the operator knows these typical tasks he knows the job which means that he fulfills the competency requirements.

The job analyses performed in nuclear power plants resulted in around 150 unique typical tasks each for the turbine operator and the reactor operator. This shift supervisor had around 100 unique typical tasks. There is a certain overlap in content between the jobs. The reactor operator must know some of the tasks of the turbine operator. The shift supervisor must know all the tasks of both the reactor operator and the turbine operator. As deputy supervisor the reactor operator must know an extensive part of the tasks of the shift supervisor. A sample of typical tasks for the reactor operator is found in Figure 5.

Specification of knowledge and skill requirements

The typical tasks generated in the job analysis describe what the operator must be able to do to carry out the job successfully in accordance with demands to run the station safely and with high availability. The second step of the competency method concerns the transformation of the typical tasks into knowledge terms. The principles and procedures for this transformation has been reported previously (3).

Knowledge terminology

The aim is to express the typical task in terms which are relevant for the planning of personnel recruitment, operator training and follow-ups connected with these activities. A set of knowledge related terms was generated for analysing the typical tasks on knowledge and skill content for nuclear power plant operators. It is presented below and the terms are related to nuclear power plants. But it is likely that they can also be used in other process industries with minor modifications.

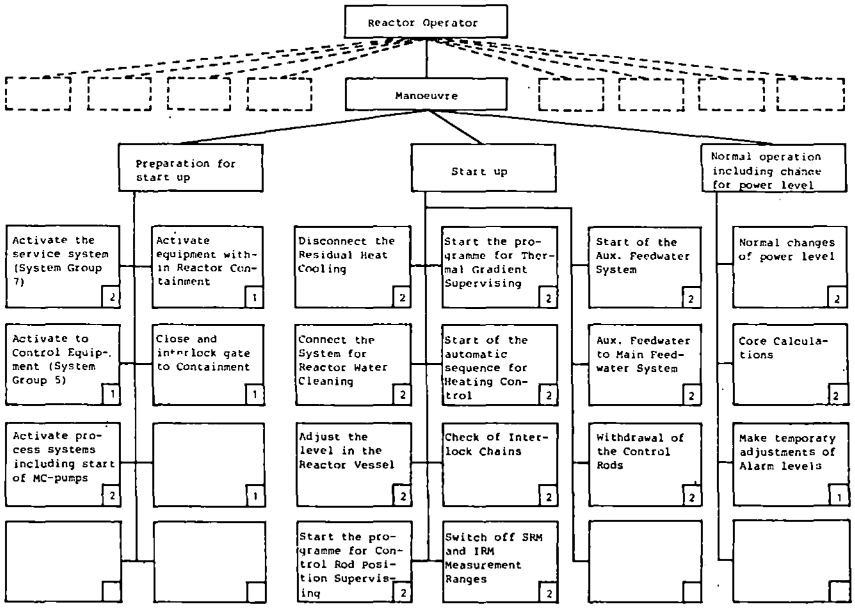

Figure 5. A sample of typical tasks.

A. Knowledge categories: 13 knowledge categories were formulated. They were judged by training and operator experts to be relevant for the nuclear operator jobs. A relatively precise definition was given to each category:

(1) Knowledge on plant layout.

(2) Component knowledge.

(3) Knowledge on manoeuvring.

(4) System knowledge.

(5) Process knowledge.

(6) Reactor core knowledge.

(7) Knowledge on localizing and identifying disturbances.

(8) Knowledge on normal operation and measures at disturbances.

(9) Knowledge on measures at plant fire, serious accidents and sabotage.

(10) Organizational knowledge.

(11) Administrative knowledge.

(12) Knowledge on safety regulations.

(13) Knowledge on supervision.

B. Knowledge object: 5 object categories were formulated. All objects within the categories are defined and are listed in the plant documentation (with exception for Actions):

(1) Technical systems according to, e.g. the System List of the plant.

(2) Organizational units or persons according to, e.g. the Organizational Chart of the plant.

(3) Documents according to, e.g. the Document List of the plant.

(4) Disturbances according to disturbance lists of the plant.

(5) Actions according to, e.g. the List of Typical Tasks of the jobs considered.

C. Knowledge depth: The depth of the knowledge or skill is defined in three levels:

3. Thorough knowledge or skill means learning to the extent that the material can be activated without use of instruction, advice or any other aid.

2. Knowledge or skill means that the material can be activated with use of instruction, advice etc.

1. Orientation means familiarity with the material normally without demand on performance.

Each one of the typical tasks from the job description are then analysed with regard to its knowledge content in terms of significant knowledge categories, knowledge objects and knowledge depth.

The outcome of the knowledge and skill analysis will be a list of typical tasks and its related knowledge content. Together they define the competency requirements of the job according to the present method. There are various ways to summarize and present these competency requirements.

The analysis concerning nuclear power plant operators was performed by personnel of each utility representing the operational staff and training specialists. Rules and advice for the analysis were formulated by Ergonområd.

Specification and implementation of an operator job training programme

The competency system concerning recruitment requirements and operator training is being implemented in all Swedish nuclear blocs.

Recruitment requirements have been expressed in terms of mathematics, physics, chemistry, technology and techiques. A High School education specially made for process technicians and operational personnel has been decided upon as the minimum basic education before entering job training for nuclear power plant operators.

The operator training offered by the utility should be based on this recruitment requirement. The operator training is divided into three categories:

A. Basic operator training.

B. Retraining.

C. Continued operator training.

The content of the basic operator training was specified in terms of typical tasks, knowledge categories, knowledge objects and knowledge depths. Quite an extensive part of the training is carried out in simulators, full scope simulators as well as more limited simulators.

The retraining is especially important in tasks, knowledge and skills which are seldom practiced on-the-job, e.g. in fault localization and identification and actions in disturbances and accidents. The need for retraining can be found through knowledge tests and questionnaires to operators. The retraining can be carried out, e.g. once a year.

It is important to realize that the content of a job in a large plant is never static. There are always new things, technically and organizationally, concerning regulations or new operational experiences which have to be taught to the operator. Therefore, there is also a need for updating of the operator in these new aspects. This training is called continued on-the-job training. It should be given with certain time intervals, e.g. a year.

The competency system also regulates how the utility shall follow up the individual competency. Different tests should be given to a student which will make it possible to demonstrate to the student himself and to the utility that the demanded knowledge and skills have been acquired.

The competency system developed for the Swedish nuclear power production also has some administrative procedures which made it possible for the Nuclear Power Inspectorate to fulfill its role as a safety authority.

The implementation of this competency system in the utilities has started from July 1, 1980. Each utility is responsible for the development of recruitment procedures, an operator training programme with courses and follow-up procedures and for a training organization including instructors, training aids and other resources needed to carry out the programme. It can be mentioned in this context that together the utilities are running a school for operator training which houses two so called fullscope training simulators.

Work organization for competency development.

The present method for operator competency work is adapted to be used by the utilities and operators. The work on the development and implementation of the competency system in the Swedish nuclear power stations has been carried out in close cooperation between Ergonområd AB, the Nuclear Power Inspectorate which is the safety authority and the utilities. There has been a working group at Ergonområd and a working group in each one of the utilities. Every part of the system has been thoroughly worked through in the working groups before it is accepted by the Inspectorate and is sent back to the utilities as a regulation. The work organization is presented in Figure 6.

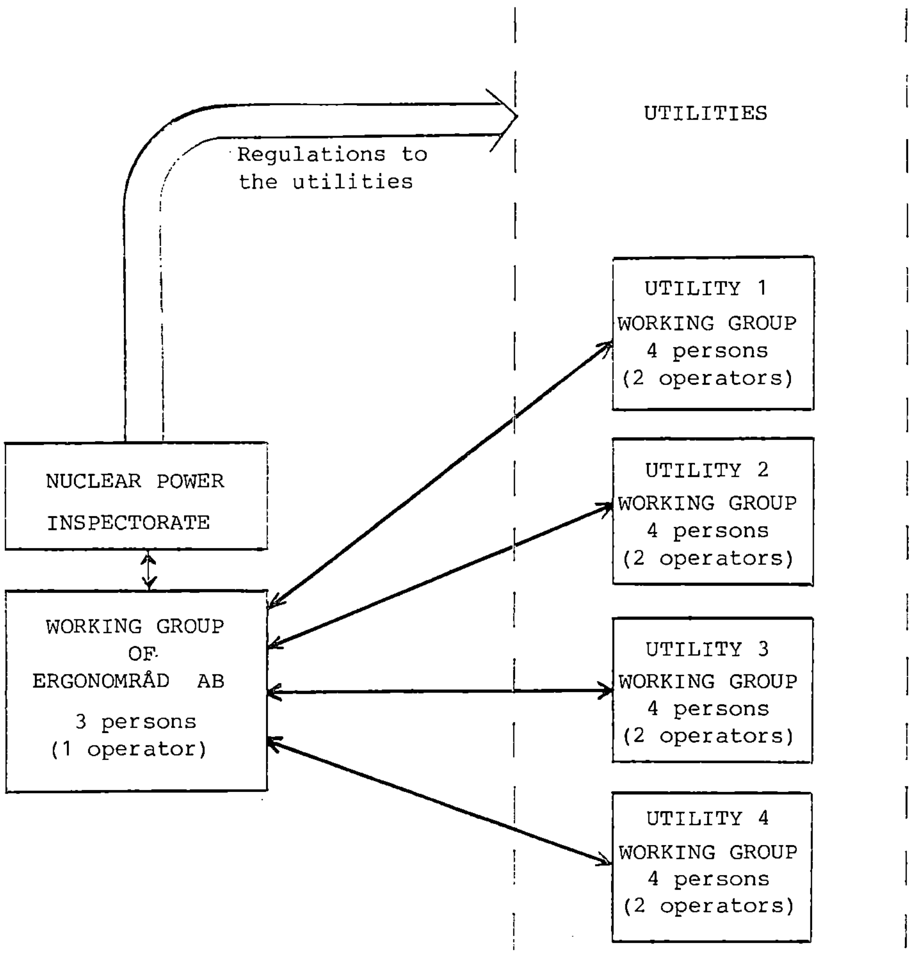

Figure 6. Work organization for the competency system.

Concluding remarks

These presented principles and procedures for competency in nuclear power plant operator jobs have a first order importance for system safety.

However, the applicability is not limited to safety in nuclear power. The concept of operator competency, consisting a number of interrelated factors as recruitment requirements, operator training and related follow-up procedures has a more general applicability.

The job analysis method can be applied more generally to operator jobs especially where there is a significant demand for safety or availability.

The principles for knowledge and skill analysis can be transferred without extensive modifications to other process operator jobs.

It it too early to draw any firm conclusions about the competency system but there are at the present moment some preliminary statements which can be made about the method as such.

- The job analysis method built up around the state diagram and the set of typical tasks is useful. The technique and its procedures can easily be communicated to operators, supervisors, instructors, plant management, system engineers and designers. The information needed to generate typical tasks can be collected with moderate costs and resources in comparison to other job analysis techniques like critical incident techniques and time sampling techniques.

- The transformation of typical tasks into knowledge and skill content can, with the present procedures, be a somewhat difficult job, although it has at the moment been carried out adequately in five plants in Sweden. There is no difficulty for operators and training planners of the plant in understanding the meaning of the knowledge terms as such, but there is a risk of some loss in the meaning of the terms between different persons who in some way or another are involved in operator training planning. This problem has been tackled by giving some training. A couple of three-day courses have been given to the instructors, operators and supervisors who are involved in the work.

- The common terminology and the structure for competency and operator training have a positive impact on operator training as a whole. It makes it easier to communicate between different plants and utilities on operator training. It has been demonstrated several times that experience gained in one plant can be transferred in an easy way to other plants. It is very likely that the common terminology and the structure offered by the competency system have enhanced this transfer.

- It has been observed during the work with the "competency system" that operator training has gained status on the plants. It is difficult to say to what extent this is an effect of the analysis and conclusions from the TMI-accident. But it is likely that the procedure to relate operator training to plant operation has made it easier for plant management to realize that operator training is an important support activity to plant operation which can not only influence safety but also plant availability. This obvious linking of operator training to plant operation has probably given training more recognition.

- One of the main purposes of the competency system is to also give the safety authority a tool to regulate and audit operator competence. A leading idea with the competency system is that the authority must not overemphazise operator performance as such but also consider other factors which contribute in a substantial way to operator competence, e.g. operator recruitment, training content, training means and operator follow-up procedures. The competency system offers tools for the authority to also audit these parts. It is not possible to evaluate the competency system from this point of view yet, but it is clear that the competency system and its terminology and structure facilitates the communication between the authority and the utilities in the same way that the communication in competency and operator training issues was facilitated between the utilities.

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QUESTIONS TO DR. JAN WIRSTAD

Joachim Fechner

Q: You have listed "leadership knowledge" as one of the knowledge categories. What is done with respect to training in order to develop this knowledge?

A: Each power plant is now working on a complete operator training programme in accordance with the Swedish "competency system," including leadership training. Leadership knowledge represents a new concept in connection with nuclear power plant operator training and requires some special efforts.

At the moment, I cannot give a precise answer to how the courses in leadership training will be arranged. Barseback, one of the plants, has, however, an outline of such a course.

A Plant Operator Selection System for Evaluating Employment Candidates' Potential for Success in Electric Power Plant Operations Positions

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Overview

The Plant Operator Selection System is the culmination of a thirty month research effort sponsored by Edison Electric Institute and carried out by Personnel Decisions Research Institute. Seventy investor-owned utility companies participated in the program. Research information was obtained and analyzed from thousands of company officials, supervisors, and plant operations personnel working in hundreds of plants.

The Plant Operator Selection System is a battery of tests and questionnaires that can be administered to job candidates in less than three hours. Various components of the battery measure what a job candidate has accomplished in previous educational and work situations, how well a candidate compares with others on a number of important aptitudes or abilities, and whether or not a candidate possesses the kind of personal stability required in power plant operations positions.

A job candidate's answers to the tests and questionnaires of the Plant Operator Selection System are scored and converted to an OVERALL POTENTIAL INDEX. Values of the OVERALL POTENTIAL INDEX [OPI] range between 0 and 15. Candidates with high OPI values are much more likely to become effective and successful plant operators than candidates with low OPI values.

It is possible to estimate the financial advantages to a company of using the Plant Operator Selection System in evaluating candidates for plant operations jobs.

Details of Procedure

The following activities were carried out during the development and validation of the Plant Operator Selection System.

1. A comprehensive review of the scientific and trade literature revealed behavioral constructs that had been found by other investigators to be important for success in power plant operations and other process control jobs.

2. Job descriptions of all operating positions were examined, and PDRI staff members visited ten geographically dis-bursed power plants for the purpose of conducting on-site job analyses. These were followed by a series of ten two-day job information meetings held with company officials from all participating companies. All the foregoing information was used to develop a comprehensive 506 item Plant Operator Job Task List. A section of this task list relevant to nuclear operations is shown on the following page.

3. The Plant Operator Job Task List was used by 2,710 job incumbents to describe the salient features of their jobs. Job incumbents were selected in such a way as to assure that each distinct operations job title in each participating plant was described by at least two job incumbents. Dimensional analyses of task list descriptions revealed five relatively independent operations job areas cutting across all participating companies. These five areas include:

    - Hydroelectric and Switchboard Operator positions;
    - Nuclear Plant Operator positions, including Control Room Operators;
    - Boiler and Turbine Operator positions;
    - General Fossil Plant Operator positions, including Control Room Operators; and
    - Beginning Level and Fossil Plant Trainee positions.

TASKS BE SURE TO USE A NO. 2 PENCIL ONLY AND FILL ALL CIRCLES COMPLETELY	A Darken circle if you do NOT perform task (zero time spent and zero importance)	B Indicate Relative Time Spent Much less than other tasks / About the same as other tasks / Much more than other tasks	C Indicate Relative Importance Much less than other tasks / About the same as other tasks / Much more than other tasks	D Task is typically performed On site in plant / Both places equally / In control room
14. Monitor and operate gaseous waste system	N	1 2 3 4 5	1 2 3 4 5	1 2 3
15. Handle and replace contaminated filters, manually and/or with remote handling equipment	N	1 2 3 4 5	1 2 3 4 5	1 2 3
16. Monitor and operate radioactive area ventilation exhaust system	N	1 2 3 4 5	1 2 3 4 5	1 2 3
17. Monitor process and effluent radiation monitoring systems, and evaluate indicators	N	1 2 3 4 5	1 2 3 4 5	1 2 3
18. Participate in testing reactor protective systems	N	1 2 3 4 5	1 2 3 4 5	1 2 3
19. Inspect, monitor, and operate control systems related to reactor safety	N	1 2 3 4 5	1 2 3 4 5	1 2 3
20. Monitor and operate primary coolant leak detection systems	N	1 2 3 4 5	1 2 3 4 5	1 2 3
21. Monitor and operate moisture separator/reheater control systems	N	1 2 3 4 5	1 2 3 4 5	1 2 3
22. Monitor and operate containment cooling systems	N	1 2 3 4 5	1 2 3 4 5	1 2 3
23. Shut down reactor to hot standby condition	N	1 2 3 4 5	1 2 3 4 5	1 2 3
24. Cool down plant from hot standby to cold shutdown condition	N	1 2 3 4 5	1 2 3 4 5	1 2 3
25. Receive and inspect new fuel	N	1 2 3 4 5	1 2 3 4 5	1 2 3
26. Test and operate new and spent fuel handling equipment	N	1 2 3 4 5	1 2 3 4 5	1 2 3
27. Prepare plant for refueling	N	1 2 3 4 5	1 2 3 4 5	1 2 3
28. Preposition fuel prior to refueling	N	1 2 3 4 5	1 2 3 4 5	1 2 3
29. Perform reactor refueling operations	N	1 2 3 4 5	1 2 3 4 5	1 2 3
30. Prepare plant for service after refueling	N	1 2 3 4 5	1 2 3 4 5	1 2 3
31. Perform hot functional and low power physics tests following refueling	N	1 2 3 4 5	1 2 3 4 5	1 2 3
32. Perform sipping tests to identify defective fuel	N	1 2 3 4 5	1 2 3 4 5	1 2 3
33. Operate radiation survey instruments	N	1 2 3 4 5	1 2 3 4 5	1 2 3
34. Compute radiation dose rates and personnel radiation exposure	N	1 2 3 4 5	1 2 3 4 5	1 2 3
35. Follow radiation safety procedures	N	1 2 3 4 5	1 2 3 4 5	1 2 3
36. Don and work in respiratory protective equipment and/or anti-contamination clothing	N	1 2 3 4 5	1 2 3 4 5	1 2 3
37. Remove and dispose of contaminated clothing	N	1 2 3 4 5	1 2 3 4 5	1 2 3
38. Launder contaminated clothing	N	1 2 3 4 5	1 2 3 4 5	1 2 3
39. Participate in equipment decontamination activities	N	1 2 3 4 5	1 2 3 4 5	1 2 3
40. Participate in personnel decontamination activities	N	1 2 3 4 5	1 2 3 4 5	1 2 3
41. Operate and monitor loose parts detection system	N	1 2 3 4 5	1 2 3 4 5	1 2 3
42. Monitor core vibration detection system	N	1 2 3 4 5	1 2 3 4 5	1 2 3
43. Perform calculations associated with load changes and transients	N	1 2 3 4 5	1 2 3 4 5	1 2 3
44. Perform calculations associated with estimating critical conditions	N	1 2 3 4 5	1 2 3 4 5	1 2 3
45. Monitor and control boric acid concentrations in PWR cooling and support systems	N	1 2 3 4 5	1 2 3 4 5	1 2 3
46. Inspect and monitor PWR boric acid heat trace system	N	1 2 3 4 5	1 2 3 4 5	1 2 3
47. Monitor and operate PWR chemical and volume control system	N	1 2 3 4 5	1 2 3 4 5	1 2 3
48. Inspect, monitor, and operate PWR steam generators	N	1 2 3 4 5	1 2 3 4 5	1 2 3
49. Monitor and operate BWR Traversing In-core Probe Subsystem (TIPS) to monitor local power distribution and calibrate nuclear instrumentation	N	1 2 3 4 5	1 2 3 4 5	1 2 3

4. Three workshop series were conducted with company officials and plant operators for the purpose of defining quite exactly the dimensions of job success and job failure in and the financial consequences of success or failure in plant operations.

- The first series (Series I) consisted of four meetings with nuclear plant officials for the purpose of exploring the nature and relative frequency of instances of aberrant, unreliable, or deviant job behavior on the part of personnel working in nuclear plants. Table 1 summarizes six broad patterns of aberrant behavior found to characterize such examples. Though extremely infrequent, such instances are sufficiently serious to warrant continued study of selection programs designed to screen out persons who show tendencies toward such behaviors.

- A second series of workshops (Series II) consisted of three meetings with operations supervisors and training department officials held for the purpose of exploring not only examples of aberrant job behavior but the full range of job performance examples illustrative of either unusual effectiveness or unusual ineffectiveness in plant operations. Over the three meetings, 667 such examples were accumulated. Content analyses of these examples yielded the seven categories of operator job performance shown in Table 2.

  In addition, participants in these Series II workshops rated the relative importance of personal characteristics--knowledges, aptitudes, interests, and temperament factors--for becoming successful in power plant operations positions.

- A third series of workshops (Series III) consisted of two meetings with high company officials held for the purpose of developing estimates of the magnitude financial costs likely to be associated with ineffective operator job performance and financial savings likely to be associated with effective job performance.

  The importance of Series III workshops has to do with the desirability of documenting the business necessity of selection procedures. We addressed the concept of business necessity by documenting that there are very

Table 1. Six Major Manifestations of Emotional Instability Derived from Analysis of Instances of Aberrant Job Behavior in Nuclear Plants

Manifestation	Description	Contributing Factors
HOSTILITY TOWARD AUTHORITY (anti-social conduct)	Refusal to work as a team member; resentment of supervisory direction; verbal or physical aggression against others.	. hot temper . hostility . interpersonal isolation . defensiveness . vandalism
IRRESPONSIBILITY and IMPULSIVENESS (unreliable conduct)	Horseplay; failure to take job seriously; refusal to comply with regulations; impulsive actions taken without concern for consequences	. impulsiveness . inability to cope with job structure . failure to take work seriously . inappropriate reaction to crises
DEFENSIVE INCOMPETENCE (withdrawal conduct)	Reluctant to carry out the job because of lack of knowledge or skill; defensive efforts to cover up one's own incompetence	. does not interact with others . does not inform others . covers up mistakes . fears taking action without detailed instructions
PSYCHOPATHOLOGY (incapacitating emotional instability)	Irrational actions; extreme fear; incapacitating emotional reactions; uncontrollable aggression, depression, etc.	. nervousness, moodiness . depression . panic in emergency situations . irritability, aggressiveness . inappropriate emotional response . periods of deteriorating job performance
COMPULSIVE INCOMPETENCE (obsessive control and compulsive conduct)	Extreme and compulsive attention to detail; demand for absolute control over job tasks; refusal to share knowledge with others.	. refusal to delegate activities . refusal to rely on other persons . demands to "check" everything for correctness . won't accept help from others . quick to argue or fight with others

Table 1 (cont'd.)

Manifestation	Description	Contributing Factors
SUBSTANCE ABUSE (chemically induced erratic conduct)	Erratic and unpredictable behavior induced by excessive use of alcohol and/or other drugs.	. erratic behavior from accumulated stress . alternately irritable, melancholy, aggressive, and depressed . appears at work drunk or high on other drugs

Table 2. Job Performance Categories Summarizing the Content of 667 Operator Performance Examples Gathered During Series II Workshop Meetings

Performance Category	Description
A. SYSTEM COMPREHENSION	Knows plant equipment, plant processes, and plant operating procedures. Possesses complete knowledge of relationships between all types of plant equipment and their functions in generating electrical energy. Knows operating characteristics of overall system and how all parts of the system fit together.
B. RESPONSE TO CRITICAL, HIGH RISK, AND/OR EMERGENCY SITUATIONS	Operates equipment correctly during critical times and/or high risk situations. Diagnoses causes of emergency malfunctions under severe time pressure and high risk. Correctly assesses criticality of situation by considering effects on entire system. Takes appropriate action to maintain system or to return system to normal operating conditions.
C. MAINTAINING STANDARD OPERATIONS: MONITORING, INSPECTING, TESTING AND ADJUSTING EQUIPMENT	Inspects condition of equipment routinely, systematically, and thoroughly. Monitors equipment to confirm proper operating conditions and detects valid indicators of non-standard operating conditions. Recognizes situations likely to develop into problems and corrects conditions to prevent problems from occurring.
D. ADMINISTRATIVE RECORD KEEPING	Documents actions as required and develops and maintains records of operations. Explains reportable occurrences in writing. Handles equipment maintenance requests, supply requisitions, etc. Reads logs, procedures manuals, training materials, other manuals, etc. to keep properly informed.

Table 2 (cont'd.)

Performance Category	Description
E. INFORMING OTHERS	Keeps superiors, peers, and others fully informed of relevant information. Conveys information accurately, clearly, and unambiguously regardless of circumstances, whether they are relaxed or under serious time pressure.
F. RELATIONSHIPS WITH CO-WORKERS	Gets along with others. Cooperates fully with supervisors, peers and others and works as a team member. Exerts extra effort to help out in special situations. Willingly helps others through showing them how to carry out job tasks, filling in for them when necessary, etc.
G. COPING WITH JOB CIRCUMSTANCES	Accepts structure, procedures, regulations and rules of plant operation. Accepts authority from others and responds constructively to problem situations. Controls emotions under even unusually difficult circumstances.

large economic consequences associated with both effective and ineffective levels of operator job performance.

Thus, it is possible to evaluate the validity of the Plant Operator Selection System according to the financial benefits likely to accrue through improved selection.

Table 3 shows lower bound estimates of the standard deviations in dollars of plant operator job performance for five operations positions.

5. All the foregoing information was used to select tests and inventories to be administered, on an experimental basis, to plant operators working in the 70 companies participating in the research project. Table 4 shows the qualities measured by the instruments chosen for inclusion in the experimental test battery.

6. Our purpose, of course, was to administer tests to a large number of plant operators and to correlate their test scores with supervisors' ratings of their effectiveness in performing their jobs. Thus, we developed several rating scales designed to measure all aspects of what we had learned about operators' effectiveness/ineffectiveness on their jobs. Accordingly, our rating scales tapped the manifestations of emotional instability shown in Table 1, the dimensions of operator job performance shown in Table 2, and various of the personal qualities listed in Table 4. All these rating scales were combined into a Job Performance Appraisal Booklet that was used by supervisors in describing their subordinates.

7. Coordinators in each of the participating companies arranged to administer the experimental battery of tests to job incumbents in various plant operations positions. Completed tests were obtained from a total of 3,413 plant operators.

   Coordinators also were asked to obtain two performance ratings for each job incumbent who had taken the experimental tests. Two such ratings were, in fact, obtained for a total of 2,677 plant operations employees. One supervisory rating was obtained for each of an additional 665 job incumbents.

Table 3. Lower Bound Estimates (lower limit of 99% confidence intervals) of Standard Deviation (in dollars) of Operator Job Performance for Five Operations Positions

Position	Lower Bound Estimate of Standard Deviation of Operator Job Performance
Nuclear Plant Control Room Operator	$112,000
Fossil Plant Control Room Operator	$ 67,500
Hydroelectric Plant Operator	$ 14,000
Nuclear Plant Operator (unlicensed)	$ 21,000
Fossil Plant Operator (Plant level such as boiler operator, turbine operator, etc.)	$ 19,000

Table 4. Qualities Measured by Tests and Inventories Chosen for Inclusion in the EEI Plant Operator Experimental Battery

1. Numerical Aptitude
2. Spatial Visualization (three dimensions)
3. Speed of Perception and Accuracy (detail orientation)
4. Reasoning Ability (inductive reasoning and deductive reasoning)
5. Knowledge of Mechanical Principles
6. Fluency of Ideas for Problem Solving
7. Verbal Ability
8. Attentional Selectivity (field independence)
9. Spatial Memory (visual screening)
10. Reading Comprehension
11. System Comprehension
12. Care and Accuracy in Following Directions
13. Sociability
14. Leadership Orientation
15. Freedom from Anxiety
16. Playfulness
17. Self Control
18. Acceptance of Routine
19. Adjustment to Shift Work
20. Willingness to Accept Authority
21. Defensiveness
22. Psychopathy
23. Impulsiveness
24. Dependability/Conscientiousness
25. Sleep/Wakefulness Physiology
26. Habits of Forgetfulness
27. Absorption
28. Risk Taking Orientation
29. Emotional Maturity
30. Hard Work/Accomplishment
31. Confidence/Self Esteem
32. Interest in Things/Ideas (e.g., Practical, Scientific, Artistic Interests)
33. Changes in Life Circumstances
34. Check scales to detect inattention in completing tests, effort to "look good", and deliberate random responding.

Table 5 shows the numbers of operators for whom both supervisory ratings and test information were available for analysis.

Analyses Performed and Results Obtained

We turn now to a summary of the statistical analyses performed and the results obtained:

1. First, we learned that supervisors agreed quite well with one another in sizing of their subordinates. This is shown by coefficients of rater agreement ranging from .59 for ratings of emotional stability to .74 for ratings of overall job performance.

2. Supervisors' ratings were summarized to form four job performance (or criterion) scores for each job incumbent. Criterion score 1, called Emotional Stability, reflected supervisory ratings of an operator's stability and reliability on the job.

   Criterion score 2, called Operations Competence, reflected supervisory ratings of an operator's effectiveness in carrying out the job according to the categories shown in Table 2.

   Criterion Score 3, called Problem Solving Ability, reflected supervisory ratings of those personal qualities which had been shown to be important determiners of job success in operations positions.

   Criterion score 4, called Overall Performance, reflected supervisory ratings of an operator's overall effectiveness when all performance areas are considered.

3. Extensive analyses were carried out to discover the particular combinations of ability tests, background or experience measures, and characteristics of temperament that were most accurately and efficiently related to these four criterion scores. Table 6 shows validity coefficients obtained for various ability experience, and personality measures against each of the four criterion scores.

4. Validity generalization analyses showed that variances in validities across companies are larger by only trivial amounts than what we would expect on the basis of differ-

Table 5. Numbers of Plants Represented and Numbers of Job Incumbents Tested and Rated by Supervisors During Development of the Power Plant Operator Selection System

Type of Plant	Number of Plants	Number of Job Incumbents Tested and Rated
Nuclear	25	492
Fossil	183	2668
Hydroelectric	34	176
Totals	242	3336

Table 6. Validity Coefficients* between Components of the Plant Operator Selection System and Four Measures of Plant Operator Job Effectiveness

	Supervisory Rating Scores of Job Performance			
	Criterion 1 Emotional Stability	Criterion 2 Operations Competence	Criterion 3 Problem Solving Ability	Criterion 4 Overall Performance
APTITUDE COMPONENT	.26	.27	.42	.28
EXPERIENCE COMPONENT	.23	.28	.31	.30
PERSONALITY COMPONENT	.21	.15	.05	.15

*Validity coefficients have been corrected to take account of measurement errors in the criterion scores. The reliabilities for each of the four criterion scores are shown below:

Criterion 1	.59
Criterion 2	.66
Criterion 3	.67
Criterion 4	.74

ences in sample sizes (sampling error) alone. Validity generalization analyses yielded similar conclusions for variances in validities across sex and race subgroups; however, validity variances across job types are larger than would be expected from sampling error alone. Validities for predicting success in operator job success in fossil plants are consistently somewhat greater than those for predicting operator job success in hydroelectric and nuclear plants. The typical validity of the Plant Operator Selection System for all operations jobs in fossil plants is .40. The corresponding typical validity for operations jobs in nuclear and hydroelectric plants is .30.

Interpretation of Scores on the Plant Operator Selection System

The Plant Operator Selection System is comprised of three components requiring no more than three hours to administer.

The first component is the Previous Experience Questionnaire, consisting of a single booklet containing questions about an employment candidate's previous experiences, as they may be indicative of confidence, past patterns of effectiveness, work orientation, stability, and acceptance of structure.

The second component consists of a series of brief ability tests yielding scores that are indicative of mechanical knowledge, reading and arithmetic skills, quickness in learning, and ability to catch on and understand new situations.

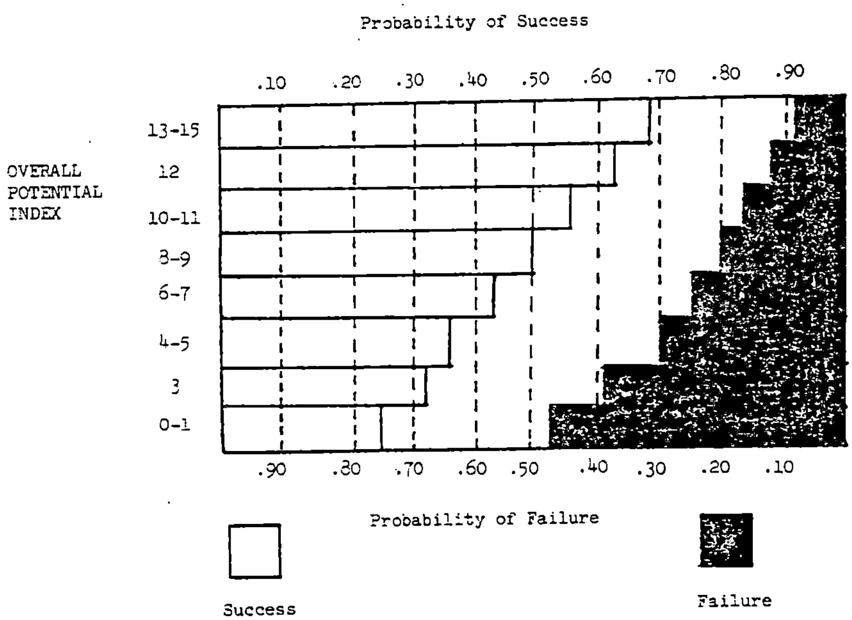
The third component is the Personnel Questionnaire, consisting of a single booklet containing statements indicative of an employment candidate's standing on such aspects of temperament as impulsiveness, self control, stability, socialization, and riskiness.

A candidate's scores on the three components of the Plant Operator Selection System are combined into an OVERALL POTENTIAL INDEX (OPI). OPI values range from 0 to 15, corresponding to percentile ranges as shown in Table 7. Each OVERALL POTENTIAL INDEX value has been shown empirically to be associated with varying probabilities of success or failure in different plant operations positions. For example, the chart shown in Figure 1 shows how various OVERALL POTENTIAL INDEX scores relate to the likelihood of success or failure in nuclear plant control room operator positions. Information contained in Figure 1 is based on defining SUCCESS to include performance as good or

Table 7. Percentile Scores Equivalent to Various OVERALL POTENTIAL INDEX Scores

Index Score	Percentile Scores Equivalent to Index
15	Higher than 99
14	98-99
13	96-97
12	90-95
11	80-89
10	70-79
9	60-69
8	50-59
7	40-49
6	30-39
5	20-29
4	10-19
3	5-9
2	3-4
1	1-2
0	Lower than 1

Figure 1. Probabilities of Success and Failure Associated with Various OVERALL POTENTIAL INDEX Scores Obtained by Candidates for Nuclear Control Room Operator Positions



Note: Success corresponds to a performance level equivalent to that shown by the top half of presently employed Nuclear CROs. Failure corresponds to a performance level below that shown by 80 percent of presently employed Nuclear CROs.

better than the average performance of present operators. [That is, SUCCESS corresponds to performance equivalent to the top 50% of present operators.] FAILURE is defined as performance poorer than that shown by 80% or more of present operators. [That is, FAILURE corresponds to performance levels equivalent to the bottom 20% of present operators.]

Financial Benefits from Improved Selection

Many years ago, Cronbach & Gleser (1965) developed the equation shown in Figure 2. This equation allows us to determine the gain in dollars per year to an organization for each person selected on the basis of this new Plant Operator Selection System. We have already seen from company officials' estimates (see Table 3) a conservative (lower bound) figure for the standard deviation (in dollars) of nuclear control room operators' performance is $112,000 annually.

Table 8 shows financial outcomes that might be expected with use of the Plant Operator Selection System to select nuclear control room operators under a number of different, but typical personnel selection policies.

As can be seen, the economic implications, industry wide, of adopting a validated selection strategy such as the EEI Plant Operator Selection System are quite great. If only 200 nuclear CROs per year were to be selected, the annual financial advantage per year to the industry would range between 5 million and 13 million dollars depending on the selection ratios used by different companies.

Use of such systematic selection procedures over a ten year period would yield an estimated return to the nuclear industry of well over a half billion dollars.

Reference

Cronbach, L. J. and Gleser, G. C. Psychological Tests and Personnel Decisions. Urbana, Illinois: University of Illinois Press, 1965.

Figure 2. Equation for Estimating Annual Incremental Utility in Dollars from Adopting a Validated Selection System

$$\Delta \bar{U}/\text{selectee} = r_{xy} SD_y \bar{z}_x - C/p$$

Where:

$\Delta \bar{U}$/selectee	--the gain to a company in dollars per year per person selected as a result of using a new selection procedure
r_{xy}	--the validity of the new selection procedure; the correlation between scores on the procecure and job performance for a group of incumbents not selected with the procedure. This validity estimate may be the raw validity of the procedure, or it may be the so-called true validity--the value given after corrections have been made for such attenuating artifacts as criterion unreliability, restriction in range, etc.
SD_y	--the standard deviation of job performance among incumbents not selected with the procedure
$\bar{z}_x$	--the average standard score on the selection procedure of those selected by the procedure
C	--the cost of applying the selection procedure for one applicant
p	--the proportion of applicants who are selected

Table 8. Estimates of Expected Annual Savings in Dollars per Selectee from Use of Plant Operator Selection System for Nuclear Control Room Operator Job Candidates

Proportion of Job Candidates Chosen	$\bar{z}_x$	Annual Saving/ Selectee
.05	2.06	$67,216
.10	1.76	$58,136
.25	1.27	$42,272
.50	.80	$26,680

Note: 1. The above estimates assume that the administrative costs associated with the Plant Operator Selection System would amount to $100 for each candidate evaluated.

2. The validity of the Plant Operator Selection System for Nuclear CRO candidates is assumed to be .30.

HUMAN FACTORS SOCIETY STUDY GROUP

PROGRESS REPORT

Robert C. Sugarman, Ph.D.[1]

Robert R. Mackie, Ph.D.[2]

The Human Factors Society is conducting a study for the Nuclear Regulatory Commission to develop a comprehensive human factors plan for the next 10 years. This plan will meet the diverse requirements for human factors consideration imposed by the different regulatory functions and responsibilities of the various NRC program offices. Four technical areas are being examined in both the activities of the NRC and the nuclear power utilities and vendors. The areas are:

a. human engineering

b. training and training devices

c. manpower and personnel

d. procedures and operator aids

Extensive data collection is being undertaken via interviews, site visits, and examination of government and industry reports. Although the four technical areas are not independent of each other, this report will describe the human factors perspective of some of the major concerns that have been identified in the training and manpower areas.

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Several of the problem areas identified thus far have as their central issue the lack of performance standards by which performance of nuclear power plant personnel can be judged. Performance standards are a key component of task analyses, a keystone of systems analysis as it applies to human functions. The task analysis is used in all human factors tasks dealing with equipment design, development of procedures, development of training objectives and materials, test construction, and even management and policy decisions related to job code descriptions, staffing and promotion requirements, and so forth.

It appears to be the case that no complete task analyses have been conducted that apply to any of the several personnel functions which are relevant, directly or indirectly, to nuclear power plant operations. Those functions not only include reactor and auxiliary operators, but also maintenance, health physics, and other personnel who interact with operations personnel during any phase of plant operation.

The lack of performance criteria resulting from the lack of task analyses has a major impact on the entire licensing processing including:

a. the validity of the exams
b. examination updating and quality control
c. utility of the training which is designed specifically to prepare examinees

d. requalification requirements

e. coordination of NRC in-house expertise relevant to examine procedures

f. incoming skills and knowledges required for qualification (this impacts significantly on manpower sources, seniority systems, career path, in-service training, etc.)

g. design of training aids, devices, and simulators

h. procedures for change of shift

i. determination of which personnel functions (job codes) require licensing or certification

We have noted secondary effects, namely a high personnel turnover rate resulting from career advancement obstacles in the form of education requirements. Those requirements are difficult to defend from the standpoint of their relationship to operational criteria.

Our knowledge is incomplete in at least two areas needed to assist in defining realistic human performance criteria: 1) the reliability of human performance which must be anticipated; and 2) the decay of skills and knowledge with disuse. This need is far from unique to the nuclear power business, however.

Other problem areas have been noted that apply to management decisions which have a potential impact on plant safety. Questions of work-rest cycles and, in particular, practices wherein operating personnel are required to work extensive overtime leading to decrements in performance, are not being adequately addressed.

Another of our concerns is that the management variables involved in safe nuclear power plant operation have not been identified nor have they been tied to any criteria of safety or operational performance.

These and many other topics are the subject of our study. The final results of this program will be published in early 1982.

PAPER IV-4

The Associate Degree in Nuclear Engineering--
What Does It Offer to the Training of Reactor Operators?

by

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The Pennsylvania State University

for presentation at

CSNI Specialist Meeting on Operator

Training and Qualifications

Charlotte, NC

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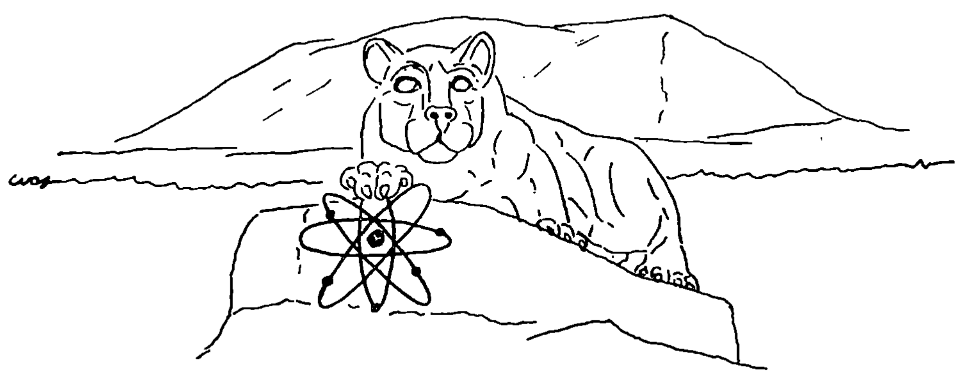

The Associate Degree in Nuclear Engineering

What Does It Offer to the Training of Reactor Operators?

by A. J. Baratta, J. L. Penkala, W. F. Witzig

The analysis of the accident at Three Mile Island by the Kemeny Commission[1] has focussed attention on the need to improve operator training and education. While considerable progress has been made in the upgrading of training programs, there is still considerable debate over how to improve the formal education of operators. At one extreme, one hears arguments that every operator should, and must, have a Baccalaureate Degree. At the other, it is argued that formal education provides little, if any, benefit. Instead, what is needed are highly experienced, well-trained, individuals.[2]

This paper examines the case for degreed personnel in the control rooms. Specifically, the authors consider that as a minimum, reactor operators should possess an Associate Degree in Nuclear Engineering Technology from an accredited program. It is not the purpose of this paper to consider the educational needs of shift supervisors and other personnel. These needs are discussed elsewhere.[2]

For the purpose of this paper, such a program will be defined as one meeting the American Nuclear Society--Accreditation Board for Engineering and Technology (ANS-ABET) criteria.[3,4] While it is recognized that other criteria and accreditation agencies exist, the authors' familiarity with those of the American Nuclear Society and those of the Accreditation Board for Engineering and Technology form the basis for this discussion.

Today there are three ABET accredited two-year nuclear programs in the United States. These programs are located at the Wentworth Institute of Technology, the Hartford State Technical College, and The Pennsylvania State University.[5] Each of these programs requires the equivalent of two years of full time resident academic work beyond high school. The curricula are technologically oriented and include applications of the physical sciences and technique of mathematics to the solution of practical problems in the areas of nuclear, electrical, and reactor technology. The instruction includes both laboratory and classroom work.

As required by the ANS-ABET criteria, the specific course work must include the equivalent of one-half academic year of basic science and mathematics including algebra, trigonometry, and concepts of calculus. There must be at least one year of technical courses in addition to those in basic science and mathematics. Also, each program includes one-third of an academic year of non-technical subjects such as oral and written communications, social sciences, and humanities. The remainder of the two years is devoted to those areas determined appropriate by the institution. The curriculum must consist of at least seventy semester credit hours of course work.

A typical program is shown in Figure 1. The program is the Penn State Nuclear Engineering Technology Program. The Penn State Program was initiated in 1970. At that time, it was entitled, "Nuclear Technology Program." Since then it has undergone several revisions intended to update and improve the overall program. The name of the program was changed to Nuclear Engineering Technology so as to more accurately reflect the course content.

Figure 1

2 Year Nuclear Engineering

Technology Program

Program Graduates receive an Associate Degree in Nuclear Engineering Technology. The program is accredited by the Accreditation Board for Engineering Technology (ABET).

First Term	Credits
E.G. 1, Engineering Drawing	2
*Engl. 4, Basic Writing Skills; or Engl. 10, Composition and Rhetoric I	3
=Engr. 2, Engineering Orientation	1
=Math 801, Technical Mathematics	3
=Phys. 150, Technical Physics	3
	12

Second Term	Credits
=Cmp. Sc. 1, Basic Computer Programming	1
=E.E. 801, Fundamentals of D.C. Circuits	3
=E.E. 809, D.C. Circuits Laboratory	2
=Math 802, Technical Mathematics	3
=Phys. 151, Technical Physics	3
	12

Third Term	Credits
=Chem. 11, Introductory Chem.	3
=E.E. 814, Electrical Circuits	4
*Engl. 10, Composition and Rhetoric I or Engl. 20 Composition and Rhetoric II	3
=Math 803, Technical Calculus	3
	13

Fourth Term	Credits
=NucE 800, Nuclear and Atomic Science	2
=NucE 804, Principles of Measurement	3
Social Science selection	3
Sp. Com 200, Effective Speech	3
	11

Fifth Term	Credits
*Engl. 826, Report Writing	3
=M.E. 807, Heat Transfer	3
=NucE 801, Radiological Safety	2
=NucE 802, Elements of Nuclear Technology	2
Humanities selection	3
	13

+Sixth Term	Credits
=NucE 803, Elements of Nuclear Power Generation	3
=NucE 804, Introduction to Reactor Technology	3
=NucE 812, Nuclear Technology Laboratory	3
=NucE 814, Reactor Technology Laboratory	3
	12
NucE 830, Health Physics (Optional Course)	3

*Students are placed in Engl. 4 or 10 on the basis of English Placement Test scores. Students who are placed in Engl. 4 also must take Engl. 10. Students who begin with Engl. 10 are encouraged to take Engl. 20. Engl. 826 is required for all students in the program.
+Sixth term is to be taken at the University Park Campus.
=Denotes courses applicable to 60 semester hour requirement of second proposed revision 2 to Regulatory Guide 1.8. Total applicable is 53 credits. Balance of seven credits can be easily accommodated by rearrangement of program.

At present, the program includes 6 semester credit hours of physics, 3 credits of chemistry, 9 credits of mathematics, and 37 credits of nuclear and related technologically oriented courses. In addition, the program includes 9 credits of english, 3 credits of social sciences, 3 credits of the humanities, and 3 credits of speech.

Included in the required courses is a mixture of both classroom and laboratory course work. In the case of Penn State, the laboratory work includes extensive work with a TRIGA reactor and other facilities at Penn State's Breazeale Reactor.

The technical portion of the Penn State program is easily broken into two parts. The first half includes courses in the basics of engineering, science and mathematics. These courses include the physics, mathematics, computer science, engineering drawing, engineering orientation, and chemistry courses. Typically, students take these during their first academic year. These courses heavily stress the development of a basic understanding of those physical principles that underlie the solution of typical engineering problems. The courses rely heavily on problem solving to both develop an understanding of these basic principles and at the same time develop an individual's problem solving skills and abstract reasoning ability. The course work a student encounters during this part of the program provides the basics for the second half of the program.

The second half of the program applies those principles learned in the first half to the area of nuclear technology and related subjects. The material covered includes specifics of how a nuclear reactor operates, the fundamentals of radiological health, basic reactor thermodynamics and heat transfer, the fundamentals of radiation detection and measurement, and basic radiochemistry. Again, there is an emphasis on problem solving. Throughout the program, there is an emphasis on not only "how" things work but also "why." It should also be pointed out that all courses are taught using algebra and basic calculus. This allows the material to be taught in a quantitative manner rather than qualitative.

In addition to those courses intended to develop a broad and basic technical knowledge, the program also includes courses intended to improve an individual's communication skills. Courses are required which cover both oral and written communication. Course work in both the technical and non-technical liberal arts courses exercise both written and oral skills. The value of such work is evident to anyone who has tried to write clear and concise operating procedures for use around a reactor.

The accredited program outlined above offers several advantages over training programs currently in use. The most significant of these is the development of an individual's cognitive problem solving skills. Because of the heavy emphasis on problem solving encountered in this type of program, an individual's problem solving and abstract reasoning ability are developed. Such skills and abilities are needed if an individual is to respond properly to a new unexpected occurrence. In addition to these skills, the program also provides the background needed to readily analyze and solve unexpected technical problems in a reliable manner.

Another advantage offered by an accredited program is its acceptance by the public. Repeated public opinion surveys show that, in general, college programs are viewed as credible. Graduates of such programs are generally perceived as competent. As a result, one would expect that degreed utility personnel should be viewed by the public as more competent than those without degrees.

It is also expected that the use of existing accredited programs by utilities should help <u>prevent, or at least reduce, increases in training costs</u>. For example, currently proposed revisions to the NRC Regulatory Guide 1.8 requires operating personnel to have from 45 to 60 college credits in selected technical areas.[6] A review of the proposed requirements shows that a substantial portion of the credit requirements are met by the Associate Degree Program. For example, a review of the Penn State program, shown in Figure 1, shows that 53 of the 73 credits should be applicable to the proposed academic requirements. Thus, little, if any, expense would be incurred by a utility in developing a program to meet these requirements.

The Associate Degree may also be used as a "step" towards a baccalaureate degree. For example, at Penn State it is possible to complete a two-year program in NET and then transfer all credits to a Bachelor of Engineering Technology (BET) program. Such students enter the BET program in the junior year. They are thus able to obtain a baccalaureate degree upon completion of two years of additional work.

In conclusion, an Associate Degree helps the reactor operator by the development of a basic knowledge of the technical areas of interest. In addition, the program develops and improves those cognitive problem solving skills needed to cope with the unexpected technical problems. Since these programs exist today, they provide an approach to improving the educational level of operators which requires little developmental effort and are most cost effective.

References

1. "The Report of the President's Commission on the Accident at Three Mile Island," J. G. Kemeny, Chairman. Washington, D.C. (October 1979)
2. "The value of an Engineering Degree for Nuclear Plant Operation." W. F. Witzig and G. E. Robinson. ANS International Conference. (November 1980)
3. "ANS-ABET Criteria for the Accreditation of 2 Year Nuclear Technology Program." ANS, LaGrange, IL and Accreditation Board for Engineering and Technology, New York, NY.
4. "Manual of Evaluation Procedures of ABET." Accreditation Board for Engineering and Technology, 345 East 47th Street, New York, NY 10017.
5. Technology Accreditation Commission Report EC-21, December 1980. ABET Publications Office, New York.
6. See for example, Revision 2 to Regulatory Guide 1.8 dated September 1980, USNRC, Office of Standards Development, Washington, D.C.

QUESTION TO W. F. WITZIG

Robert Mackie

Q: What feedback have you had from industry reflecting the value to them of personnel who have received the associate degree? It seems to me that the financial benefits are only realized if industry feels it can greatly shorten the training program for these people.

A: The feedback has been very favorable. The graduates are eagerly sought by nuclear utilities and industry. In fact, the demand exceeds supply by three to five times now.

SUMMARY

SESSION V - PERFORMANCE MEASUREMENT

CHAIRMAN: R. M. Koehler

Use is being made of nuclear plant simulators for performance analyses of power plant operators, their procedures, and their control room environments. Studies sponsored by the EPRI have demonstrated that computer assessment of operator actions on a simulator can generate human factors improvement evaluations, can assist in standards development, and can evaluate the timeliness and accuracy of operator response to accidents.

Computer assisted data processing techniques are available to utilities for the complex evaluations of occupational (task) data. These data are usable in establishing performance standards and other information of value to trainers and nuclear plant management.

Performance of operators in control rooms is dependent on their environments. The recently published NUREG-0700 will serve as a basis for total job design including crew size and crew organization, factors not generally considered flexible in human factors evalutions.

HUMAN FACTORS RESEARCH USING THE EPRI PERFORMANCE MEASUREMENT SYSTEM

E. J. Kozinsky
General Physics Corporation
Chattanooga, Tennessee

A computer-based evaluation system has been developed for objectively measuring certain elements of an operator's performance on a power plant simulator. The Performance Measurement System (PMS) is designed to help the instructor in his total evaluation by providing measurements and documentation of certain essential elements of a trainee's performance. Some measurements of operator performance can be applied to Human Factors research on the operator - control room interface.

The computerized system provides measurements of how the operator has responded to plant indicators and made switch manipulations. Magnetic tape data containing indications of all the control room gauges, annunciator lights, and switch and knob positions is collected, with time to one-second accuracy. When any change occurs, a data record is written. The resulting data is a sequence of "snap-shots" of the simulator, each containing the status of every light, meter, switch and knob on the simulator. By evaluation of a series of data records, operator time response, errors, and continuous control can be evaluated.

TRAINING EXERCISES

The development of evaluation software for a training exercise must start with a clearly defined idea of which plant evolutions are to be involved. The exercise must cover a discrete facet of plant operation suitable for evaluation and compatible with available initial conditions in the simulator. Ideally the exercise will follow the operating and emergency procedures with only a single correct path of operator actions which will deliver the plant to the desired condition (e.g., Reactor Startup). However, there may be steps in which there are multiple paths to accomplish a given requirement. These cases must be specifically noted, in order to account for that variability of action in the programming. Exercises which do not have well-defined procedural steps or recognized operator actions lack the necessary bases for evaluation tools.

With a definition of correct operator performance, evaluation software to detect deviations or errors is developed. The outputs for training consisted primarily of time line printouts of operator errors in the context of major plant milestones. Figure I is a typical output format. Other summaries of errors were also developed. The errors were deviations from the 'correct'

DATE:25-FEB-80
RUN : 1

EVENT/ERROR CHRONOLOGY
SEQUOYAH SIMULATOR
REACTOR STARTUP

	TIME HR:MIN:SEC	EVENT OR ERROR
	00:00:00	START OF EXERCISE.
	00:12:45	OPERATOR BEGINS ROD WITHDRAWAL.
**	00:12:45	FAILED TO TEST ALL ANNUNCIATORS PRIOR TO STARTUP.
**	00:12:45	FAILED TO MONITOR THE HIGH SOURCE RANGE FOR AUDIO COUNTS.
**	00:12:45	COMMENCED STARTUP WITH T-AVG BELOW 541 DEGREES.
**	00:27:55	ALLOWED T-AVERAGE TO DROP BELOW 538 DEGREES.
**	00:28:12	CONTINUED STARTUP WITH T AVERAGE BELOW 538 DEGREES.
**	00:28:28	CONTINUED STARTUP WITH T AVERAGE BELOW 538 DEGREES.
**	00:31:16	CONTINUED STARTUP WITH T AVERAGE BELOW 538 DEGREES.
**	00:31:47	CONTINUED STARTUP WITH T AVERAGE BELOW 538 DEGREES.
**	00:34:06	CONTINUED STARTUP WITH T AVERAGE BELOW 538 DEGREES.
**	00:34:15	CONTINUED STARTUP WITH T AVERAGE BELOW 538 DEGREES.
**	00:40:00	CONTINUED STARTUP WITH T AVERAGE BELOW 538 DEGREES.
**	00:51:22	CONTINUED STARTUP WITH T AVERAGE BELOW 538 DEGREES.
**	00:53:38	CONTINUED STARTUP WITH T AVERAGE BELOW 538 DEGREES.
	01:02:27	REACTOR CRITICAL.
**	01:07:59	FAILED TO RECORD SOURCE RANGE AT STARTUP.
**	01:08:04	FAILED TO RECORD INTERMEDIATE RANGE FOR STARTUP.
	01:14:15	P-6 PERMISSIVE LIGHT
	01:16:40	SOURCE RANGE BLOCK
**	01:16:40	FAILED TO RECORD BOTH INTERMEDIATE RANGES BEFORE SOURCE RANGE BLOCK.
**	01:16:58	RECORDED POWER RANGE PREMATURELY (BEFORE 1% POWER).
	01:25:56	REACTOR POWER AT 10**-8 AMPS.
**	01:28:37	EXCEEDED A 100 DEG/HR. HEATUP RATE.
	01:28:42	REACTOR ADDING HEAT (POAH).

PMS Training Output
Figure I

operator path.

While a single correct task path may appear overly stringent, the transients generally trained for on a simulator have specific procedures which govern proper operation action. The "Achilles' heel" of this approach is that single or multiple malfunctions may be superimposed on an exercise scenario so that operators encounter many different situations. It is impossible for the person who develops an exercise to anticipate all the possible malfunctions which may be selected by the instructor, and it is impractical to develop evaluation software which includes combinations of all those that are anticipated. This limited evaluation to relatively fixed exercises, reduced the flexibility available to the instructor. The result was relatively poor instructor acceptance.

A second problem area with this type of exercise development was its dependence on plant procedures. The 'correct' operator path was defined by procedures for a given scenario. As procedures are subjected to continual revision the 'correct' path changes, requiring evaluation software revisions. High software maintenance costs result, and evaluation programs continually become out of date.

Although operator performance evaluation of this type is possible, there are limitations in instructional flexibility and maintenance costs which must be recognized. Alternate performance measures, not dependent on procedures may offer more promise.

CONTINUOUS VARIABLES

One of many operator tasks is to control plant states in a stable manner as well as staying within operating limits. Certain continuous variables tend to reflect stability during manual operations; but absolute criteria for stable performance within technical specification limits are argumentative. Operator smoothness and control strategy varies as does the opinions of instructors. While most instructors would agree on what constitutes highly unstable performance, quantification of the degree of smoothness is subjective. In this kind of situation the best approach to developing measurement is to find out what operators and trainees really do (i.e.: how they control quantitatively). Given quantitative data, criteria can be derived for training performance assessment based on those measures which reveal the differences between experts and novices.

Continuous variable work in the EPRI project concentrated on developing measurement data forms for defining useful measures of performance. The parameters selected for continuous variable measurement analysis were power during reactor startup and manual steam generator water level control. Two performance data formats for initial analyses were developed for each exercise, (a) time history plots of continuous variables and (b) state-space representations of the system states (variables) which are being controlled. Donald Vreuls provided the major direction for this research.

State-Space Plots

Figure II is a state space (phase plane) representation of reactor power control during a startup. It plots class intervals of Intermediate Range Power along the horizontal axis against class intervals of Startup Rate along the vertical axis. IR Power ranges from 1 x 10^{-10} to 5 x 10^{-4}. The plot is a 21 x 19 cell matrix. The data in each cell of the matrix is the percent of time (in tenths of a percent of total exercise time) that the two system states occurred in their respective class intervals at the same time. For example, during the whole run represented, Figure II shows that 3.4% of the time the IR Power was in upper third of the 10^{-9} meter range (5 x 10^{-9}) while the Startup Rate was in the range of 0.6. The matrix can be thought of as a two dimensional time histogram of the correlation between Startup Rate and IR Power, or a Phase Plane of a parameter and its derivative.

The state space plots are a useful data form because there are definite regions which represent questionable and/or unstable performance. For example, the operator's task is to smoothly bring IR Power up to 10^{-8} and hold, then bring it up to approximately 10^{-4} without exceeding a Startup Rate of 1.0. Figure II represents the stable performance of a qualified operator. Figure III, however, reveals the unstable performance of a trainee. The trainee was overcontrolling the rods as seen in the irregular state space plot curve.

Steam Generator Water Level Control

During a plant startup on some PWR designs the operator manually controls steam generator water level. The job is to match the feed flow into the generators with the steam flow out of the generators to maintain the desired steam generator water level within limits. The manual part of the job terminates about 15% power when the feed valves are placed in automatic control.

Figure IV is a typical state-space representing control of four steam generators. The horizontal axis of the plot scales steam generator water level in 2% class intervals from 16% to 60%. The vertical axis represents Feel Flow minus Steam Flow in "million pounds mass per hour" units from +0.30 to -0.30 in class intervals of 0.02 units. The row labeled "RX" shows reactor power level during the exercise against the scale in the row labeled "POWER", which represents a range of 0-22% power in 1% class intervals. Any

ALL SEGMENT SYSTEM STATES IN PERCENT TIME X 10

IR POWER IN TENTHS

S/U RATE X 10	X 10-10 1	2	5	X 10-9 1	2	5	X 10-8 1	2	5	X 10-7 1	2	5	X 10-6 1	2	5	X 10-5 1	2	5	X 10-4 1	2	5	TOTAL
13																						0
12																						0
11			+			+			+			+			+			+				0
10																						0
9																						0
8																						0
7			+			+			+			+			+			+				0
6						34	38	36	30	14	22	36	38	39	23							315
5				23	41	2			2	17	14				13	40	45	1				202
4		19	53	21					1	8								38				141
3	16	36	+			+			3+			+			+			+				56
2	41								3													45
1	28								3													32
0									206													206
-1			+			+			+			+			+			+				0
-2																						0
-3																						0
-4																						0
-5																						0
TOTAL	86	55	53	45	41	36	38	36	250	40	36	36	38	39	36	40	45	39	0	0	0	999

Reactor Startup Phase Plane

Figure II

ALL SEGMENT SYSTEM STATES IN PERCENT TIME X 10

IR POWER IN TENTHS

S/U RATE X 10	X 10-10 1	2	5	X 10-9 1	2	5	X 10-8 1	2	5	X 10-7 1	2	5	X 10-6 1	2	5	X 10-5 1	2	5	X 10-4 1	2	5	TOTAL
13																						0
12																						0
11			+			+			+			+			+			+				0
10																						0
9																						0
8				1	1																	2
7			+	¦	2	+			+			+			+			+				3
6				2	15	19	19	20	12													89
5				4					¦	22	23	25	25	25	23	26	4					180
4				7					1	4	¦						25	7				46
3		29	44+	3		+			1+			+			+			¦+				79
2	28	10	25	11					2									¦				79
1	24		3	¦					2									6				37
0	13		19	4					150									234				423
-1			25+			+			¦+			+			+			+				25
-2			24																			24
-3			4																			4
-4																						0
-5																						0
TOTAL	66	40	146	36	19	19	19	20	171	27	24	25	25	25	23	26	29	250	0	0	0	996

Reactor Startup Phase Plane

Figure III

data beyond the limits of the plot are truncated to the respective limits. The plot is a 23 x 32 cell matrix containing percent of exercise time in which each of the two system states occurred in their respective class intervals at the same time.

These data tend to reflect stability and tightness of control as well as performance near the limits. When water level is high, the operator should be underfeeding the generators (a negative value of feed minus steam); the dispersions shown in Figure IV reveal that this was not always the case, and that Feed-Steam variability was high. Conversely, when the water level is low, the operator should overfeed, and Figure IV reveals that this was the case. One may conclude that this operator overcontrolled Feed-Steam slightly because of the vertical dispersion on the plot, and that he appeared to be overfeeding more than he should have at high water levels.

By contrast, Figure V shows another operator's performance. It can be seen that the dispersions are much smaller than in Figure IV, revealing tighter control by this operator. Water level was maintained at a lower value (actually, an average of 38% as opposed to an average of 42% for the previous operator), but this operator came dangerously close to a low-level trip. It can easily be seen why: This operator tended to underfeed more at lower water levels than the previous operator.

INFORMATION PROCESSING

In order to address operator information processing, data tapes were processed to extract information on the rate at which information is being presented to the operator in various scenarios. Evaluating a large LOCA, the accident is accompanied by 8 bits of information to the operator (lights). The next four seconds give the operator 112, 283, 97, and 25 bits of information. Five seconds into the accident the operator has been presented with 576 flashing lights to reveal to him the state of the plant. His job is simply to integrate and interpret this information and take appropriate action. During the following thirty seconds, the information rate is 28 bits per second, then decreases to 1 bit/second. The normal operations bit rate was about 1 bit per minute.

Taken alone, this data serves only to illustrate and document the extremely high data rate to which the operator is subject in a major casualty. This would tend to support findings in surveys that in major casualties the operator tends to "tune out" the mass of information being presented and specifically seek out cardinal bits on which to base decisions. This data can also be used to develop alarm filter systems or Disturbance Analysis and Surveillance System (DASS) software.

```
FEED-,---------------------------------------------------------------------,
STEAM¦ ALL STEAM GENERATUR   SYSTEM STATES        IN PERCENT TIME X 10     ¦
 MIL ¦---------------------------------------------------------------------¦
 PPH ¦                         WATER LEVEL IN PERCENT                      ¦
X 100¦ 16 18 20 22 24 26 28 30 32 34 36 38 40 42 44 46 48 50 52 54 56 58 60¦
   30¦                                                                      ¦
   28¦                                        ¦                             ¦
   26¦           +              +             ¦+              +             ¦
   24¦                                     ¦  '                             ¦
   22¦                   1  ¦        '  ¦                                   ¦
   20¦                   ¦  3  1  '  '  '  '  ¦  1  '                       ¦
   18¦                ¦  '  1  ¦  1  '  ¦  1  2  1  '                       ¦
   16¦           +    2  '  ¦  '+ ¦  2  1  1  1+ '  1     '  +              ¦
   14¦                ¦     ¦  '  ¦  2  2  1  1  1  2  2  1                 ¦
   12¦                1  1  2  1  3  2  2  2  2  2  3  4  2                 ¦
   10¦             2  1  2  3  5  4  5  3  4  5  2  3  2  1  1  ¦           ¦
    8¦             1  5  4  2  1  4  5  6 10  6  6  4  1  2  2  1           ¦
    6¦           + 1  6  5  3  4+ 2  4  4 10  9+11  9  3  4  4+ 3           ¦
    4¦             ¦ 14  4  2  6 10 11  5 14 12 14 14  7  8  2  5  ¦        ¦
    2¦                8  4  1  ¦  3 15  7 14 15 10  9 13  5  7  4  '        ¦
    0¦             ¦  4  3  2  3  7 27 26 20 25 23 22 21  7  5  6  2        ¦
   -2¦                   '  ¦  1  1  2  6  7 12  9  9  7  3  2  1           ¦
   -4¦           +          ¦  '+ ¦  2  3  2  5+ 7  5  4  3  1+ 1  '        ¦
   -6¦                      '  2  1  '  1  1  ¦  2  2  2  1  '  1  '        ¦
   -8¦                            '     '  ¦  ¦  2  1  1  1     1           ¦
  -10¦                      '  '  ¦  '  '  ¦  '  2  ¦  '  1  1  '           ¦
  -12¦                         ¦  1  ¦     ¦  2  1  1     '  2              ¦
  -14¦           +             ¦+ ¦  1  '     1+    1  '  ¦  1+    '        ¦
  -16¦                         ¦  ¦  1  '  '  ¦     '  '  1  '              ¦
  -18¦                            '  '  ¦  1        ¦  1  1  '              ¦
  -20¦                            '  '  1  ¦  '  ¦  1  ¦  ¦  1  '  '        ¦
  -22¦                            '  '  1  2  2  1  1  1  1  ¦  ¦  '        ¦
  -24¦           +              +     '  1  1  2+ 2  2  2  1  1+ ¦  '  '    ¦
  -26¦                                        '  '  '  1  ¦  ¦  '     '     ¦
  -28¦                                                                      ¦
  -30¦                                                                      ¦
  RX>             16 88160106163169118 48 80 15  8  9 14                    <
POWER¦  0  1  2  3  4  5  6  7  8  9 10 11 12 13 14 15 16 17 18 19 20 21 22 ¦
     ¦----------------------------------------------------------------------¦
```

Steam Generator Level Control State-Space Plot

Figure IV

```
FEED-,----------------------------------------------------------------------,
STEAM| ALL STEAM GENERATOR   SYSTEM STATES        IN PERCENT TIME X 10      |
 MIL |----------------------------------------------------------------------|
 PPH |                       WATER LEVEL IN PERCENT                         |
X 100| 16 18 20 22 24 26 28 30 32 34 36 38 40 42 44 46 48 50 52 54 56 58 60|
   30|
   28|
   26|            +              +              +              +
   24|
   22|
   20|
   18|
   16|            +    '  '  |  1+ 1  1         +              +
   14|     |  |  1  2  2  3  2  |  |  |  '
   12|     1  4  3  6  5  3  3  1  '     1  1  |  1  |
   10|     1  4  5  7  4  3  3  5  3  2  1  1  1  2  1  '
    8|  1  5  5  4  4  5  6  9  9  7  6  4  2  2  1  1  1
    6|     5  3  4+ 3  2  4  5  7+11 12  5  5  4+ 2  2  |      +
    4|     5  5  4  4  2  5  2  7  3  1 14 17  5  4  2  |
    2|     3  3  2  6  2  6  2  6  5  |  9 31 13  5  3  1  1
    0|     1  8  3  5  2  3  5  5 27  9  3 51 16 55 18 11  4
   -2|        2  4  7  4  5  1  3  9 14  '  |  5  3  3  4  1  '
   -4|           2+10  4  6  3  3+11 22  4  |  |+ 1  1  '  |  |+ '
   -6|           '  2  6  7 12 13 16 13  2  2  1  1  |  '
   -8|              '  2  4  7  8  8  4  2  2  1  1  1
  -10|                 '  |  |  2  2  2  2  1  2  1  1  '
  -12|                             '  |  |  1  |  '  '  '
  -14|            +              +       '  '   + '            +
  -16|
  -18|
  -20|
  -22|
  -24|            +              +              +              +
  -26|
  -28|
  -30|
   RX>339       29213198120 38                                          59<
POWER|  0  1  2  3  4  5  6  7  8  9 10 11 12 13 14 15 16 17 18 19 20 21 22|
     |----------------------------------------------------------------------|
```

Steam Generator Level Control State-Space Plot

Figure V

STANDARDS DEVELOPMENT

Using PMS data, a detailed analysis of operator performance during a large LOCA and other casualties was conducted. This analysis provides empirical support for the establishment of guidelines for assignment of safety-related actions to operator or automatic functions. That data was presented to the working group developing ANSI Standard N660. Such data on operator time responses provides unique objective evaluation of operator performance.

CONCLUSION

The use of PMS as a research tool offers unique opportunities to address many human factors questions in nuclear control rooms. There are many potential applications in operator training programs to provide objective, standardized measures of operator performance.

RELATED PUBLICATIONS

Performance Measurement System for Training Simulators, EPRI NP-783, Electric Power Research Institute, Research Project 769-1, May 1978.

Criteria for Safety-Related Nuclear Plant Operator Actions: Initial Pressurized Water Reactor (PWR) Simulator Exercises, NUREG/CR-1908, September, 1981.

QUESTION TO MR. KOZINSKY

Mr. Schlegel

Q: If you can identify "correct behavior" so precisely, why not automate it?

A: In the simulator, I have the advantage of knowing what the malfunctions are, so I can precisely determine the correct response. In the plant, the operator does not have that advantage. The problem is not automation of plant response, but automation of diagnosis.

SAFETY-RELATED OPERATOR ACTIONS IN NUCLEAR POWER PLANTS*

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The Safety Related Operator Actions Program at Oak Ridge National Laboratory is intended to provide the U. S. Nuclear Regulatory Commission (USNRC) with quantitative and qualitative data on the performance of nuclear power plant operators during accident events. The data are necessary to support licensing decisions, standards development, and research in a number of areas related to operational safety. The program, which was developed after a preliminary assessment of available historic data,[1] consists primarily of three tasks: (1) collection and assessment of data from controlled simulator exercises; (2) collection and assessment of field data; and (3) calibration of simulator to field data.

The simulator exercises are being conducted by General Physics Corporation (GPC) at the Tennessee Valley Authority (TVA) Training Center near Soddy-Daisy, Tennessee. Licensed nuclear

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plant operators from different utilities perform a controlled series of exercises that involve a number of transient events (typically 6 to 10 during an eight-hour period) with periods of "normal operation" in between. Data are recorded automatically using the Performance Measurement System (PMS) developed previously by GPC under contract to EPRI.

The field data collection is being carried out by Memphis State University Center for Nuclear Studies (MSU/CNS). Licensee Event Report (LER) summaries provided by the Nuclear Safety Information Center at ORNL are used as an indicator to identify applicable events. Subsequent site visits are conducted to compile all available records - control room logs, supervisor logs, computer output, plant upset records, etc. - that relate to the event. As much as possible, an attempt is made to review the details of the event with site personnel, especially any available who were involved with the event of interest.

The initial series of ten "exerimental runs" on the PWR (Sequoyah Plant) simulator has been completed, and a report summarizing results will be published in the near future.[2] The PWR field data collection is not yet complete. However, an example of the kind of quantitative data that is being extracted is provided in Fig. 1. The figure is a cumulative probability plot comparing results of simulator exercises to field data collected during the preliminary study reported in Ref. 1 for the event "Inadvertent Safety Injection at Power" (I.S.I.).

The ordinate values are the time in seconds for the operating crew to perform the first required action (reset S.I.) correctly. Since a linear fit on a log scale approximates the data points reasonably well, a log-normal distribution is assumed for both samples. The log-mean value of response time is approximately 1.9 minutes and 4.8 respectively for the field data and simulator data. A similar analysis for the event "Dropped Rod" gave log-mean response times of 0.7 minutes and 1.6 minutes respectively for the field data (sample size N = 19) and simulator data (N = 8). In both cases the mean response time was greater for these initial simulator runs than for the field data.

The current program of simulator experiments and field data collection should be viewed as one necessary element in NRC's overall approach to improving operational safety of nuclear power plants. It will provide specific information on operator response times useful for the near-term licensing and standards needs and will initiate development of a realistic data base on operator performance necessary for a broad range of analysis and assessments currently in progress or planned. The results and analysis of data completed at this time are very preliminary and should not be used to form general conclusions. However, they do demonstrate that the approach to development of a data base from field-calibrated simulator results should be successful.

References

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2. "Safety Related Operator Actions: Initial PWR Simulator Exercises," A General Physics Corporation Report prepared for Oak Ridge National Laboratory (to be published as ORNL/NUREG/TM-434).

PAPER V-3

A REPORT ON THE PILOT TEST TO DEMONSTRATE THE CAPABILITIES OF THE COMPREHENSIVE OCCUPATIONAL DATA ANALYSIS PROGRAM (CODAP)

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Institute of Nuclear Power Operations

INTRODUCTION

CODAP is an acronym for the Comprehensive Occupational Data Analysis Program. It is a computer assisted occupational analysis system developed by the US Air Force Human Resources Laboratory which stresses the quantification and empirical testing of human performance factors for a given job or group of jobs. The Institute of Nuclear Power Operations (INPO) is conducting a pilot test of CODAP with Alabama Power Company to demonstrate the capabilities and usability of CODAP. The pilot test is being conducted on the mechanical maintenance job positions.

CODAP SOFTWARE CAPABILITIES

A job analysis is conducted to develop a task inventory. The resulting task statements, along with background questions, are used to form clusters or groups of tasks, define categories of tasks and produce prioritized lists of information meaningful to managers. Typical examples of CODAP reports include composite job descriptions, differences between the job requirements of two or more groups of workers, summaries of human performance variables (such as relative time spent performing tasks and the consequences of incorrect task performance).

CODAP uses sophisticated data analysis techniques such as hierarchical clustering, inter-rater reliability measures, and regression analysis to induce field data to information useful to managers.

CODAP receives its data input from occupational surveys administered to job incumbents and their supervisors. Occupational surveys administered to job incumbents generally consist of three sections: background questions, equipment and tool list, and task inventory. Background questions are used to identify various subgroups within the total population of workers or supervisors surveyed. Background questions used in the pilot test for mechanical maintenance include:

- Present job title
- Job at which initially hired
- Number of months employed at Alabama Power
- Number of months at current location
- Number of months in current job level
- Number of months experience as a mechanic prior to joining Alabama Power
- Number of months experience in a fossil plant
- Years of education
- Number of technical courses taken
- Number of courses taken leading to a higher level of formal education
- Relative amount of time spent working in each of the following areas: fitting, machining, welding, rigging

Other background questions frequently asked include grade or salary level and location of job.

By identifying groups of workers, it is possible to cluster the tasks performed by those workers and thus produce unique job descriptions.

The equipment and tool list is used to identify the equipment and tools used by each subgroup of workers. Later, this information can be merged with the background data to create yet another variable for producing job descriptions.

The task inventory contains a list of all tasks performed by incumbents in each job and is developed using traditional job analysis techniques. Workers are asked to indicate those tasks they perform in their current job and the amount of time they spend performing each task, relative to all other tasks performed. A nine-point scale is being used in the pilot test for this rating.

Supervisors may be asked to respond to the task inventory on one or several task factors. In the pilot test, supervisors are asked to indicate for each mechanical maintenance task the "consequences of incorrect task performance." This scale provides a measure, from low to high, of the consequences of incorrect worker performance. Supervisors are also being asked to indicate the "current training emphasis" vs. "desired or required training emphasis for each task." Other scales less

In the pilot test, CODAP software routines will be used to develop job descriptions for each group of mechanical maintenance workers. A job description is developed by first deciding which background items best describe a specific group of workers. A typical job description might begin with the following set of background variables:

- High school graduate
- Less than 12 months on the job
- No previous technical courses
- Job title of Apprentice Machinist, Level 1

A CODAP routine scans the responses of all workers (in this example, machinists) and selects only those tasks performed by the specified workers (i.e., Apprentice Machinist, Level 1). The routine computes hierarchical clusters according to the percent of time workers spend performing each task. These clusters show the tasks and relative time spent performing for the specified subgroup of workers. Thus a description of the job in terms of tasks performed is created. CODAP can also compare the job description for a subgroup to a full group and determine what percentage of workers in the full group appear in the subgroups.

Another CODAP routine computes and prints for each task the following values:

- Number of workers performing each task
- Percent of workers performing each task

- Average percent time spent performing each task by all workers
- Cumulative sum of the average percent time spent performing each task by all workers

Managers often use this report to select which tasks to include in a training program.

CODAP can evaluate the between-group difference for any pair or job descriptions. For example, this routine compares the job description of two levels of journeymen (such as electricians) and computes the between-group differences.

This information aids in determining distinct job types within an occupational area. The values reported include:

- Difference in percent of time spent performing each task
- Difference in percent of time spent on each duty (a duty is a cluster of closely related tasks)
- Difference in percent of workers performing each task
- Difference in number of tasks needed to account for a specific percent of total group time
- Difference in number of duties (clusters of closely related tasks) needed to account for a specific percent of total group time
- Difference in the average number of tasks performed by each group of workers

A separate CODAP routine computes an average value for a selected variable for all workers that perform each task. Examples include the average number of months on the job or the average number of years of education. An additional routine computes an average percent value for a selected variable such as pay grade or skill level. This program uses discrete data with a value range of 1 to 9.

A unique routine in CODAP produces a treelike diagram that visually displays the order in which groups of tasks merge during the hierarchical clustering process that produces job descriptions.

A CODAP routine computes the total percent time spent performing a duty, the number of tasks in each duty, and the percent of an individual worker's responses for each duty. These duty values can be saved and used as a new background variable when producing job descriptions.

CODAP will also compute the overlap between a pair of job descriptions and report the comparison as a matrix. This routine computes the degree of overlap in average percent time spent performing each task, or in terms of the number of tasks performed in common. This program is useful in determining the degree to which the same training can be used for two separate groups of workers, or for determining where duties and responsibilities overlap and potentially cause conflict.

For the supervisory data collected in a pilot test, a CODAP routine will be used to compute the average inter-rater reliability coefficient or individual supervisors, and the reliability coefficient for the total group. This information will be used to delete those supervisors with divergent survey responses.

CODAP can be used to extract up to 100 background or computer variables and compute a correlation matrix or regression problem. The curve of best fit can be computed when one variable (Y) is predicted from another variable (X) using polynomials. Scattergrams of actual observations can also be plotted.

STATUS OF THE PILOT TEST

Alabama Power Company has conducted a job analysis of the mechanical mantenance positions. The resulting task inventory has been formatted as a CODAP occupational survey. Mechanical maintenance workers at each level will be asked to complete the survey indicating the relative amount of time they spend performing each task. The survey also includes a section on biographical background data and an equipment list. Mechanical maintenance job supervisors will complete a separate survey of the "consequences of incorrect task performance" and "current training emphasis" vs. "desired training emphasis." The surveys will be conducted on a schedule that will not interfere with the plant maintenance requirements. The usefulness of the additional information gathered using CODAP, in assessing job training program content, will be reported to the nuclear utility industry.

QUESTION TO J. R. HALE

Don Milley

Q: The programme you describe is called CODAP. There are probably other similar programmes. In Ontario Hydro, we have used a programme called TRAG. Would you comment on the relative merits of the other programmes?

A: CODAP and TRAQ are identical programs.

PAPER V-4

ANALYTICAL TECHNIQUES FOR CREATING A JOB DESIGN BASIS FOR A NUCLEAR POWER PLANT OPERATING CREW

BY

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CSNI SPECIALIST MEETING
ON
OPERATOR TRAINING AND QUALIFICATIONS

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INTRODUCTION

Since the accident at TMI, the nuclear industry and its regulators have been in constant dialogue over the question of the qualification and training of nuclear plant personnel. From the early TMI Action Plan (NUREG-0660) to the most recent Commissioner's Proposal, SECY 81-84, that dialogue has lead to a wholesale increase in the qualifications required of the nation's Reactor Operators, Senior Reactor Operators, and Shift Supervisors. Two and a half years later, that dialogue is at the point of suggesting that Reactor Operators be degreed with no indication that a degree requirement is the light at the end of the tunnel. While not implying that the dialogue has been without merit, this paper will nevertheless present a fundamentally different agenda for the post-TMI, "lessons learned" dialogue in the hope of starting a discussion that, unlike its predecessor, has an end-point in sight.

Presenting a fundamentally different agenda sounds like a task for at least a team of nuclear luminaries, not a solitary author of limited credentials. Fortunately for the author, however, the insight on which this paper is based requires neither technical sophistication nor a wealth of experience. It simply requires exposure to both military (submarine) and commercial operating crews and enough inquisitiveness to wonder why they are so fundamentally different.

Notwithstanding the differences in size and design between military and commercial plants, or even the training and experience of the operating crews, there seems to be a fundamental structural difference between the commercial and military crews. A commercial control room crew is structured so that each of the crew members has "panoramic" job responsibility.[1] The military control room crew, on the other hand, has "focused" job responsibility at the control panels with only a single, senior individual with panoramic job responsibility.[2] With this fundamental structural difference in mind, it does not take too much reflection on the TMI phenomenon called "sensory overload" to force the questions: could TMI be symptomatic of an issue more basic than training and qualifications; could TMI be an issue of Job Design itself?

With these questions in mind, the first item on the OECD meeting agenda that was published in the Call for Papers is particularly relevant. That first agenda item called for a discussion of the "functions, role and organisation (sic) of control room personnel as a crew and as individuals . . . ." What that agenda item suggests is that what I call "the more basic issue of Job Design," may now be ripe for discussion.

Ideas are never the property of a single individual. As might be expected, the author's investigation revealed that the idea of Job Design or better Job Re-Design, has had wide currency since TMI, albeit in another context. Therefore, in deference to the more experienced technologists who have commented on Job Design, this paper contains a substantial review of their opinions. The paper begins with an overview of how the notion of Job Design fits into the overall operational problematic. Then, the opinion review. The paper will then discuss analytical techniques that are recommended as appropriate in designing jobs for nuclear power plants and close with conjecture on Job Design changes that may occur if the suggested methodologies are implemented.

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1 Panoramic job responsibility implies a crew structure in which each crew member has a responsibility for operations on all the panels in the control room.

2 Focused job responsibility implies a crew structure in which individuals have operating cognizance over a limited segment of the control room.

JOB DESIGN WITHIN THE OPERATIONAL PROBLEMATIC

Job Design is a simple idea that the tasks associated with a certain job can be designed <u>before</u> the job is put into practice. Interestingly, most jobs are designed <u>after</u> the fact.[3] In everyday experience, most of us are "thrown" at a job and expected to "carve out" a position. As the Willis paper indicates, most of the time this works because of the adaptability of the human subject to a job, particularly in <u>non-time sensitive</u> situations. However, in <u>real-time</u> plant situations, the "sensory overload" phenomenon noticed at TMI suggests that human adaptability is limited[4] (at least quantitatively) and that attention needs to be given to time-basing the tasks of the operator.[5] This limitation of human adaptability seems to relate to the information processing capability of the operator. Since the operator is evidently limited to making X decisions in Y minutes, it follows that a required number of decisions per minute will require <u>by design</u>, a certain number of operators. This design requirement for a certain number of operators will, of course, be based on peak decision making demand rather than on average demand. The point here, is that the number of operators can be defined by the number of decisions per unit of time that the machine requires. In quantitative

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3 Willis, J. L., "Nuclear Power Plant Operator Task and Skills Analysis - A Call for Innovation"; ANS, Gatlinburg, Tennessee: April, 1981.

4 Meister, David, <u>Human Factors: Theory and Practice</u>; Wiley: New York, 1971, p. 53. "Another common attitude is that even when there are design inadequacies, the human will adapt, will 'muddle through.' The ability to overcome inadequate design characteristics is a most fortunate result of the same human flexibility which presumably also produces variable errors. Of course, the ability to adapt occurs only when operating conditions do not stress the operator excessively. Stress occurs when undesirable operational conditions . . . force the operator to respond at or near the limit of his abilities."

5 ibid., p. 63. "If in addition we know the performance duration required for each function, we can plot the Functional Flow Diagram along a time continuum. This is useful later in determining whether the human can perform the function."

terms what emerges is a choice. Given a certain machine design[6] that demands X decisions in Y minutes in scenario Z[7], a certain number of operators will be needed. Conversely, given a fixed number of operators acting in scenario Z, the machine design is limited as to the number of decisions per minute it can expect from those operators — other decisions must be made by the machine itself. The idea that system design involves specifying numbers of operators is put forward most succinctly by Meister when he discusses "error causes":

> To design a system requires that one <u>specify</u> not only the items of equipment, but also the <u>number</u> and <u>types</u> of personnel using the equipment; their <u>background</u> and <u>training</u>; appropriate data resources; logistics; and maintenance programs. (op. cit., p. 23, emphasis added).

Decisions by machine vs. decisions by the operator oversimplifies the equation. As Meister points out (above), system design affects both number and <u>types</u> of operators. This means that the operator side of the equation can be both quantitatively manipulated (i.e., by adding more operators) and can be qualitatively manipulated (i.e., by enhancement: procedures, computers, training, etc.)[8]. The machine side of the

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6 By "machine design," we are, of course, referring to a nuclear power plant and its control room. Note the methodological shift here. We are discussing the operational problematic from the point of view of the plant - the need of the plant to be operated - rather than from the point of view of the operator.

7 Scenario Z is a point in time at which the maximum number of 'Off-Normal' events are occurring simulataneously. What hopefully will emerge in this paper is that defining Scenario Z is a major item for each plant. The author feels that understanding Scenario Z is a sine qua non for operator and control room design.

8 The quantitative/qualitative distinction is not as concise as the words suggest. Qualitative decision making enhancers like training may, by raising the decision making level, actually slow down the decision making process. Quantitative decision making enhancers like procedures and computers (which contain pre-set decisions) may actually reduce the quality of decisions made.

equation can also be manipulated by assigning certain decisions to plant self-protective devices thereby eliminating the need for operator action.

Whatever the resolution of the man-machine equation, the key idea here is that Job Design is an integral part of the operational system design problematic. This means that plant and control room design have serious implications for operator design (as to numbers and type) and conversely, that existing operator design (or lack thereof) has serious implications for hardware design. Consequently, there needs to be a discussion between designers and operators relative to the allocation of decision making functions between the human, the machine, or both. If that discussion is to be beneficial, it must have a ground rule. We propose a rule that in the real time world of plant operations, the starting point for operational analysis be the plant and its needs rather than the operator and his. Certainly the operators needs will have to be accounted for. However, they should not serve as the starting point for analysis.

The idea that Job Design does indeed have a place within the operational problematic finds support in the professional literature, particularly from human factors experts like Meister who have historically been involved in Man-Machine design in the military and in aerospace. Since Three Mile Island, there has been discussion within the nuclear industry on adapting military and aerospace experience to nuclear power plants. What is discussed in the following section is the degree to which the idea of Job Design has found support within the utility industry itself and from the Industry's major vendors, laboratories, and consulting organizations.

EMERGENCE OF JOB DESIGN SINCE TMI

Immediately after TMI, Job Design was apparently not the fundamental issue, qualifications and training for existing jobs were. However, the TMI Action Plan (NUREG-0660) did recognize the need to " . . . increase the capability of shift crews in the control room by assuring that a proper number of individuals with the proper qualification and fitness are on shift at all times" (emphasis added).[9] The document also indicated that:

> NRR will develop requirements and issue instructions to operating plant licensees and operating license applicants to assure the necessary number and availability of personnel to man the operations shifts. The requirements will include administrative procedures to govern the movement of key individuals about the plant to assure that qualified individuals are readily available in the event of an abnormal or emergency situation. They will also include new administrative procedures that limit overtime (emphasis added).[10]

This statement implies that an an administrative document would be prepared by NRR that would contain a design basis for operations shifts. However, rather than issuing such a design document, NRR issued on July 31, 1980 its "Interim Criteria for Shift Staffing," a document that contains generic PWR and BWR staffing criteria by license category and administrative restrictions on overtime. Absent a crew design basis from NRR, it would remain for other documents to surface the fundamental question of basic Job Design.

Job Design did reappear, but in the control room design review dialogue that had its roots in NUREG-0660. Evidently, just as NUREG-0660 had passed over Job Design enroute to training and qualification issues, so too did NUREG/CR-1580[11] pass over Job Design enroute to what one

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9 "TMI Action Plan" (NUREG-0660), U.S. Nuclear Regulatory Commission, Washington, D.C., Draft 3: February, 1981, p. I.A.1-1.

10 ibid., p. I.A.1-3.

11 "Human Engineering Guide to Control Room Evaluation" (NUREG/CR-1580), U.S. Nuclear Regulatory Commission, Washington, D.C.: July, 1980.

paper calls "cosmetics".[12] Fortunately, however, the Guidelines (NUREG/CR-1580) received widespread industry comment that refocused their attention in many cases on the Job Design issue. Conveniently (for this author) those industry comments were published in Appendix A of the "Staff Supplement to the Draft Report on Human Engineering Guide to Control Room Evaluation" (NUREG-0659).[13] Appendix A, then, becomes the source of the industry comments that illustrate (below) the emergence of the idea of Job Design. The author, of course, can take no credit for the industry-wide base that the Appendix A comments represent. He can only express appreciation to the commentors and let the fact that such a wide spectrum of commentors chose to address Job Design speak for itself.

Commenting on the overall control room design approach in NUREG/CR-1580, J. L. Anderson of the I and C Division at ORNL widens the control room concerns to include Job Design (NUREG 0659, App. A, p. A-11):

> The approach outlined (1580) appears to deal exclusively with ways to evaluate or perform currently defined tasks better. Most of the industry attention seems oriented this way, and this is certainly needed, but may not be sufficient. A great deal more attention is needed to evaluate the defined tasks to determine if they are really what the operator should be doing, or whether the tasks could best be performed by automated systems with operator supervision at a different level. The operator should not be required to perform a task just because he is capable of it, but only if he can contribute a degree of performance or safety that is impractical to automate.

Another widening of viewpoint is expressed by S. J. Ditto, also of

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12 Starkey, R. L. and Brown, A. W.; "Man-Machine Interface - More than Cosmetics and a Control Room Review," American Power Conference, Chicago: April, 1981.

13 op. cit; U.S. Nuclear Regulatory Commission, Washington, D.C.: March, 1981.

ORNL's I and C Division who obviously shares his colleague's perspective as evidenced by similar comments which add an interesting analogy (ibid., p. A-109):

> The first significant observation (of 1580), from the standpoint of a control engineer, is that there is no explicit statement regarding the intended relationship between the operator and the plant. Just what role is the operator expected to play in normal as well as abnormal operation of the power plant? It is not at all clear that a control room can be evaluated properly until such a question is explored. It cannot be answered in such general terms as "the operator has total responsibility for all phases of the operation." There are many control loops and sub-loops that continue to operate and influence plant behavior without knowledge, consent, or aid from the operator. Analogies are automobile chokes, timing control, and even steering geometry that gives stability to the automobile's directional control.

R. W. Pack of INPO also widens the viewpoint of the discussion in specific words that criticize an atomistic approach (in 1580) that may overlook the Job Design problem (ibid., p. A-90):

> The proposed guidelines (1580) offer an atomistic approach to solving acknowledged human factors problems. While all such remedial actions will reduce the probability of human error, this writer for one, does not have any confidence that, in toto, they will address basic underlying problems, namely, what are the optimal levels of system automation vs. manual control, how can we improve the diagnostic process so that the operator proceeds unerringly to the correct diagnosis and solution to the wide multitude of anticipated and unanticipated problems that can arise. These problems are being addressed on several fronts and it might be advisable to wait for the answers to such questions before going beyond surface changes to existing boards.

T. M. Anderson, Manager of the Nuclear Safety Department at Westinghouse, addresses the Job Design question with explicit language that opens the issue of the organization of control room personnel (ibid., p. A-165):

> The multi-person control room poses some unique problems related to crew assignments, movement patterns, and crew coordination. This issue of the organization of control room personnel can affect the details of control room layout, etc. This situation needs to be addressed and appropriate guidance needs to be included.

Stephen H. Howell, Chairman of the A.I.F. Committee on Power Plant Design, Construction and Operation, also suggests widening the Scope of NUREG/CR-1580's inquiry (ibid., p. A-1):

> It is likely that the review will draw conclusions as to the adequacy of this manning and the assignment of responsibilities.

In placing Job Design within the operational problematic (above), this paper suggests that the "sensory overload" phenomenon at TMI was symptomatic of a lack of understanding of the number of tasks that could be imposed on the operators by the machine within a strict time frame that is beyond the operators control. Several of the NUREG 0659 commentors seem to share this view. Henry W. Pielage, Vice President Engineering Applications at Entor Corporation addresses the time basing aspect of Job Design squarely (ibid., p. A-66):

> One of the major factors which, in my opinion, is of greatest concern, and which is not addressed as an item of particular interest in the evaluation (1580) is the question of "sensory overload" on the operators when the plant goes into an upset condition. No amount of replacement, relocation and remarking will eliminate this problem.

W. G. Counsil, Senior Vice President, Northeast Utilities, looks at the time-basing aspect of Job Design in terms of information processing in his comments (ibid., p. A-107):

> For these reasons, we recommend that changes be made to NUREG/CR-1580 to recognize that retraining and delegation

> of the task of information processing to specifically assigned individuals can be a reasonable alternative to mechanical backfits.

Mr. Counsil's obvious and legitimate concern with backfits and his position that job (re)design may be a reasonable alternative to them is seconded by both Commonwealth Edison and Vepco whose similar words recommend the following (ibid., p. A-43 and p. A-151):

> Review each guideline in Volume II and indicate the satisfactory alternatives for backfit requirements in place of modifying hardware.

Mr. Counsil's comment that links "retraining" and "task delegation" shows that Job Design is more than a question of numbers of personnel. A more conclusive list of the constituent parts of Job Design is offered by G. F. Flanagan of the Engineering Physics Division of ORNL (ibid., p. A-108):

> Much research needs to be done before such a list of guidelines (1580) are useful. Finally, the "real" problems in the control room are not associated with human engineering in the sense expressed here, but with training, procedures, instrumentation and control, computer software/hardware, and data validation as well as management attitude, crew structure, and "tradition", all of which require extensive research before forcing backfits and which appear to have extensive safety implications.

Appendix A contains more than comments on the need for Job Design. Several commentors make suggestions and critoques on a proposed analytical technique for articularing a design basis for operations jobs. What emerges in Appendix A then is the beginning of a dialogue on the pros and cons of operator task analysis. The pro side of the operator task analysis debate is taken by GPU (ibid., p. A-74):

> NUREG/CR-1580 does not emphasize the importance of assessment of operator needs. It is important in any human factors evaluation that the needs of the operator, control room or otherwise, be known if proper study and evaluation are to be done.
>
> One extremely useful tool in the assessment of operator needs is task analysis.

The GPU position places a high premium on interviews and talk-throughs. Cautions against this approach can be found chiefly in the ORNL comments:

> Page 23, 3.2 Operator input is valuable but do not exalt it. The operator is analyzing himself. He will remember where and when he had concerns in operating the plant. (E. W. Hagen, p. A-113).
>
> Another point that appears questionable (at least it may not provide significant safety benefit for rare and extensive scenarios) is the implication that operators can reveal important shortcomings through interviews. It appears to this reviewer that operators are ingenious and will find effective ways to operate systems that are awkward and faulty. Witness lines drawn on recorder windows, extra labels, etc. However, these usually involve the routine and not the upset conditions. (Ditto, op. cit.).
>
> (page 17) Task analysis (Sect. 2.5.6) assumes that the steps of the procedures to be executed are correct and that the goals or objectives of the procedure are correct. If the steps are wrong or objectives inappropriate, the task analysis is invalid. This point has not been considered in the document. (Kisner, op. cit.).

Certainly, the position advocated by GPU is worthy of merit. In fact, a thorough evaluation of existing operations practices will reveal important design defects and provide a baseline against which Job Design changes can be evaluated. However, that is not the point. The position of ORNL might be expressed as follows:

> It is important in any human factors evaluation that the

needs of the plant be known if proper study and evaluation are to be done. One extremely useful tool in the assessment of plant needs is operational analysis.

ANALYTICAL TECHNIQUES FOR CREATING A JOB DESIGN BASIS FOR A NUCLEAR PLANT OPERATING CREW

The task of designing a Nuclear Plant Operating Crew involves defining the legitimate role of the operators, what they do and do not contribute to plant operations - which is to say, how they interact with the "mind of the plant." From the ground rule discussed eariler, this section of the paper will have to discuss the following questions:

- How do we define the operational needs of a plant?

- Once defined, how do we allocate the operational needs of a plant to the human, to an automatic function or to both?

These questions can be asked in the context of a new plant design in which the control systems and control room are not "fixed in concrete." In such a case, several iterations of definition and allocation could take place. These questions can also be asked in the context of an existing plant where the control logic and control room are fixed. In this case, a definition of plant operational needs involves articulating that which already exists. What remains is allocating those needs to determine the number and types of operators that are needed to contend with the plant as it exists. This second case reflects the needs of our utility executive, quoted earlier, that " . . . retraining and the delegation of the task of information processing to specifically assigned individuals can be a reasonable alternative to mechanical backfits."

Since this paper is addressed to an audience concerned with either operational plants or plants whose design is nearly complete, the methods we will discuss will pertain to fixed plant design, with the understanding that the principles involved could be used for new plants in an iterative design process. In the discussion that follows, the paper will attempt to give some general direction to answering our two questions. These sections will suggest ways in which plant operational needs can be defined and allocated. The methodology for making these suggestions will involve comparing three current system analysis procedures (academic, NRC, and industry) and combining the best features of each approach.

OPERATIONAL NEEDS DEFINITION

Several sources suggest methods that are appropriate for this task. We will review: Meister's determination of system requirements and functions, the suggestions of the NRC in NUREG-0659 Appendix B, and a System Operability Assessment Review Project currently being conducted on a PWR. Appropriate comments will follow.

Meister's procedure to determine operational needs is a two-step process that first determines system requirements and then determines system functions. It is in the system function determination that Meister comes to terms with "time-basing" the tasks of operators. Meister's two-step process is outlined as follows (op. cit., pp. 58-60):

1-0 Determine system requirements.

- 1-1 Determine what information is available concerning the system.
  - 1-1.1 Secure and examine available documents describing the system.
- 1-2 From the relevant documentation extract and list the following in detail:
  - (a) The system's mission or goal
  - (b) required system outputs
  - (c) required system inputs
  - (d) system capabilities and performance requirements demanded by the mission(s)
  - (e) environmental factors which may affect system performance

(f) constraints

2-0 Determine system functions.

2-1 For each system mission (see Step 1-2) list sequentially the individual major operations that must be performed to implement the mission.

By listing these operations in terms of sequential dependencies (e.g., to fly one must first take off, to take off one must first start the engine) and correlating them with the over-all time frame, they become stages in the accomplishment of the mission. In effect, what one must do to accomplish the mission becomes the individual system functions. (emphasis added).

2-2 Describe the resultant system functions in the form of a functional flow block diagram (FFD).

2-3 Determine the effect on system functions of the environmental factors, performance requirements and constraints noted in 1-2.

2-4 When additional functions are required by step 2-3, add the new functions and insert them in the diagram developed in 2-2.

2-5 Specify the inputs to, and the outputs from, each system function.

NUREG-0659, Appendix B. procedure to determine operational needs follows a different procedure than Meister. It begins with the definition of "Operating and Safety Functions" and then requires identification of Plant System Control Functions.

Paragraph 4.1.2 of this document contains the suggested procedure:

> the plant's operating and safety functions that must be controlled and monitored in the control room to achieve the control room objectives shall be identified and documented. The identified nuclear plant operating and safety functions may include:
>
> (1) nuclear reactor reactivity control
>
> (2) reactor core cooling
>
> (3) reactor coolant systems integrity

(4) primary reactor containment integrity

(5) radioactive effluent control

(6) power generation

(7) power transmission

More detailed identification of plant system control functions should then be made by considering operational situations and events that will or may confront operators in the control room. The operational situations and events to be considered, listed in terms of priority, should at a minimum consist of:

(1) All events required to be assessed by Section 15, "Accident Analyses," of the Standard Format (Reference 2).

(2) Anticipated transients without scram.

(3) Anticipated operational occurrences, including startup and shutdown of the plant.

(4) Failures in systems, subsystems, and components, and human errors.

(5) The sequence of failure events for transients and accidents analyzed to develop upgraded emergency procedures (Task Action Plan I.C.1, NUREG-0660 and NUREG-0737).

(6) Normal Operation of the plant.

The SOAR Project defines operational needs in a procedure that attempts to articulate in an integrated fashion, the overall logic of the power plant systems, how they interface with each other, and how they are controlled. The SOAR Project begins with the preparation of data packages which are then analyzed, as discussed below:

- A Physical System Set contains a drawing that shows all Process Computer inputs and outputs, it also shows in detail the interfaces between systems. A physical functional component breakdown is also prepared. General Arrangement drawings are also marked up as an aid to operator/designer reviewers. Finally, a physical input/output pathways diagram is prepared. This diagram categorizes potential system interconnects by relative pressure categories to clearly show the potential for fluid intermingling. Essentially, the mechanical design history is brought together here.

- A System Control Set begins with the preparation of a working document that shows all control clusters. It also contains operational modes and alignments" by system. This is a key document. It shows "allowable control statuses" in the form of "operational modes." It is important in that "operational modes" become the agenda for further analysis. Finally, a common mode failure document is prepared: loss and restoration of d.c., a.c., and instrument air.

- A System Man-Machine Interface Package is prepared. First, it contains a control board inventory with an allocation list; this list shows what information is available to the operator from what sources. Marked-up control board drawings and a list of computer processed information are also prepared.

- The final package coalesces much of the information gathered above. Titled the "Constraints, Requirements and Desirable List" it generates all the potential information processing tasks. It also contains the historical punchlist of operability problems by system.

The data packages are then brought together in a "Blowout Session" in which each of the "operational modes" articulated in the System Controls Set becomes an agenda item. Each of the operational modes is analyzed by both operators and engineers from the following perspectives.

- Loss of intended function. Each mode has an <u>intended</u> function. The participants look for ways for intended function to fail "by the numbers": controls, instrumentation, and system components. They also look at incident precursors and recovery options. Frequency (likelihood) and severity are assessed and documented. A judgment is made by the <u>probablistic risk assessments</u> participants that certain losses of intended function are certifiable off-normal scenarios.

- Provision for <u>unintended</u> function. This process is similar to "loss of intended function" but focuses on all the unintended functions that can occur. Again PRA is used to certify certain unintended functions as off-normal scenarios.

The Blowout Session also reviews each system with particular attention to how single barrier failures or misalignments can cause unintended fluid transport. The output of the Blowout Session is a list of certified off-normal scenarios and system deficiencies. This, together with normal operations, is essentially the "task list" that becomes the subject of allocation.

Comments and Recommendation on Operational Needs Definition. Each of the three processes discussed above attempts to generate a list of operational situations and events. Meister and SOAR begin with system analyses to determine the event list. The NRC suggested method begins with predetermined events and reviews the systems in light of the events. The NRC method pre-judges plant operational needs to ensure that FSAR Chapter 15 events and ATWS are included in the design basis list of operational needs. The SOAR and MEISTER procedure are more inclined to reveal the lower magnitude—higher probability events (i.e., the off-normal scenarios). We feel an argument can be made for combining these procedures. First, the NRC procedure mandates consideration of tasks that will eventually force the operator to interact with the plant's engineered safeguard features. The NRC procedures also mandate consideration of normal operation of the plant or what the SOAR procedure might call "consideration of intended function." In any event, joining the NRC and SOAR approaches would be a basis for ensuring inclusiveness of the operational needs list.

Meister adds a dimension not included in either the NRC or SOAR procedure. Namely, Meister insists that not only must the task list show sequential dependency, but that the list must be correlated to an overall time frame. Meister explains correlation to overall time frame by stating that:

> If in addition, we know the performance duration required for each function, we can plot the functional flow diagram (FFD)

along a time continuum. This is useful later in determining whether the human can perform the function. (op. cit., p. 63).

This paper recommends that when an attempt is made to define Operational Needs inclusively that the NRC and SOAR approaches both be used to assure the inclusiveness of the Operational Needs List. The author also recommends that the Operational Needs List be formatted in accordance with Meister's recommendation to insure a proper understanding of the information processing demands that can be placed on the operator by the machine. This understanding is essential if allocation of tasks to man and/or machine is to be made intelligently.

OPERATIONAL NEEDS ALLOCATION

The same three sources (Meister, NRC, and SOAR) were reviewed and compared for this section of the paper.

Once Operational Needs have been defined and placed in a format that adequately illustrates the sequence and magnitude of these operational needs, they must be allocated to automated plant systems, to the human, or to some combination of both. This step, which we call Operational Needs Allocation, is contained in the same three sources (Meister, NRC and SOAR) that were reviewed and compared above.

Meister's procedure to allocate operational needs takes into consideration both the strengths and weaknesses of man and machines in a procedure that is outlined as follows: (op. cit., pp. 63-67):

3-0 Allocate functions between men and machines.

3-1 Determine how system functions should be implemented.

3-2 Examine all presently available engineering documentation, for example, the Statements of Work, feasibility study reports, etc. to determine if equipment (as distinct from system) functions have already been decided upon by the customer or the contractor management on the basis of previous analyses — as they often are. Even if they have, however, the equipment functions may imply some operator relationship that should be analyzed since, as was pointed out in Chapter One, most equipments require some operation or maintenance.

3-3 For those system functions that have not been allocated as yet, differentiate between operator and equipment functions on the basis of human factors criteria.

3-4 Specify alternative man-machine configurations and functions.

A more realistic way of performing the function allocation is to concentrate first on listing and describing all the possible ways that the mission objective(s) can be implemented.

3-5 Verify that the human functions can be performed to system requirements.

3-5.1 Determine by reference to system requirements documents and/or conferences with system designers whether a quantitative operator performance requirement exists.

3-5.2 If the set of system requirements does not include an appropriate operator performance requirement, it will be necessary for the specialist to infer what that requirement is, based on his analysis of the over-all system requirements and of the configuration from which the function was extracted. (As was pointed out earlier, the absence of an operator performance requirement is very likely to be the usual case, except when human functions are critical to the system.)

General operator performance criteria can be categorized in terms of

(a) frequency of required outputs;

(b) speed of required outputs;

(c) physical requirements (e.g., strength, sensory discrimination capability, decision-making capability) for implementing the function;

(d) accuracy of required outputs.

3-5.3 Compare the operator performance requirement (e.g., speed) with average operator capability.

If an operator capability is less than that required by the potential system configuration, the function cannot be performed by the operator, and an equipment solution to the design problem must be accepted.

3-5.4 The comparison process is the same for the other three criteria.

NUREG-0659, Appendix B discusses the allocation of function in a fashion similar to that of Meister, but is specific with respect to the criteria for allocation of function to humans and machines. Par. 4.2 of this document discusses the allocation of function (op. cit., pp. B-16 to B-19):

> Human Factors engineering principles and criteria should be used to evaluate control room human-machine interfaces when analyzing performance requirements for plant control functions and for the allocation of functions to categories. Allocation categories should consist of:
>
> (1) Automatic operation by plant systems equipment.
>
> (2) Manual operation by control room operators and/or plant technicians.
>
> (3) Some combination of (1) and (2).
>
> The design evaluation allocation criteria should consider the capabilities and limitations of operators and systems. Table

B-1, Human/Machine Capabilities, provides qualitative categorization of actions where humans or machines excel.

Table B-1

Human/Machine Capabilities
(condensed list)

Humans Excel In	Machines Excel In
Ability to reason inductively	Deductive processes
Improvising and adopting flexible procedures	Exerting great force, smoothly and with precision
Ability to react to unexpected low-probability events	Storing and recalling large amounts of information in short time-periods
Applying originality in solving problems: i.e., alternative solutions	Performing complex and rapid computations with high accuracy
Ability to continue to perform when overloaded	Doing many different things at one time

SOAR function allocation. As of this writing, the SOAR Project team has not developed a firm set of criteria for functional allocation. However, they have listed the allocation outputs to be:

- allocation to hardware
- allocation to procedurally enhanced operator
- allocation to training enhanced operator
- allocation to computer

Comments and Recommendation on Operational Needs Allocation

All three allocation processes will ultimately attempt to differentiate between functions best performed by humans and functions best performed by machines. Interestingly, there is an explicit difference of opinion between Meister and the NRC on allocation criteria. The NRC uses the "qualitative categorizations" shown above in Appendix B. Meister takes issue with these qualitative criteria:

> The human factors criteria referred to are those developed by Fitts (1951) and cited by almost every human factors text, presumably for lack of anything better to recommend. These criteria compare the capabilities of men with those of machines in terms such as, "men are better at inductive reasoning, machines are better at deductive reasoning . . ." As was pointed out in Chapter Two, such criteria are practically useless in making any meaningful, practical function allocation decisions because (a) the criteria are overly general and (b) they assume that functions will be performed either by machines alone or by men alone. However, the Fitts list is a useful starting point (but only that).

What Meister suggests is needed are quantitative allocation criteria in terms of frequency, speed, accuracy, etc. Unfortunately, such criteria do not exist in the nuclear industry. Meister underlines this need and makes an appropriate recommendation:

> The essence of function allocation is, as we have seen, the comparison of operator capability with a requirement for operator performance. It has already been pointed out that, except in systems in which human functions are especially critical, most system requirements do not include quantitative operator performance requirements . . . (But) even if an operator performance requirement has been specified or inferred, the specialist may have some difficulty in securing from the behavioral literature the appropriate operator performance data with which to verify the proposed human function . . . The author considers that research to provide these data should have first priority. (op. cit., pp. 74-75).

Apparently, a computer based system for measuring human performance in serious operational situations has been developed by EPRI. What this paper suggests, and what the author recommends, is that the EPRI

Performance Measurement System (PMS) be applied to a yet-to-be-defined list of operational needs so that performance criteria can be developed. The performance criteria will ultimately become the basis on which operational needs can be allocated.

CONCLUSION

The primary thrust of this paper has been to make a case for Job Design and to suggest analytical techniques for doing it. In making the case, the author has attempted to illustrate two technical points.

1. Number and type of personnel are design basis considerations that are intimately associated with equipment and system design.

2. The fundamental data of Job Design are information processing requirements, both their definition and their allocation.

This paper may also suggest to some how crew structures may change if the analytical methods suggested above are implemented. Although conjecture of this type may be inappropriate in certain technical papers, the author feels that is appropriate here since it reveals the thinking that began the review process resulting in this paper.

The idea behind Operational Needs Allocation is that it is both needless and unwise to duplicate human and machine function unnecessarily. If a machine is designed to adequately self-protect by automatic function, it may not be necessary or desirable to either train the operators or structure their crews to respond to the self-protect feature. If this principle is applied to the TMI accident, the lack of a cogent and consistent philosophy of non-intervention with engineered safeguards features emerges. If this principle of unwarranted duplication of function is further applied to the Three Mile Island accident, it may be seen that much of the reactor oriented training of the operators was rendered superfluous by the plant's SCRAM circuits. Where training appears to have been inadequate is in the area of thermodynamics; indeed, the operator's function was to assist the plant in finding a desirable heat sink. This understanding has lead the industry and its regulators to add a substantial amount of training in the thermodynamics area. What has not happened, however, is a corresponding design-basis reduction in the training required for "reactor understanding". Furthermore, what function allocation suggests is that even in the area of thermodynamics the plant is capable of finding a heat sink (however undesirable) without the intervention of an operator. If this is true, (and the TMI incident suggests that the plant in fact could have self-protected without operator intervention), then it appears that there is some cause to seriously circumscribe the role of the operator seriously. In short, the job needs to become smaller not larger.

If operational needs definition and allocation do take place, what may emerge is a control room structure with a similar number of operators but where total crew responsibility is siginificantly reduced and where that total crew responsibility is further subdivided and specifically assigned to individuals. If this occurs, then ". . .retraining and the delegation of the task of information processing to specifically assigned individuals" as a "reasonable alternative to mechanical backfits" may become a practical reality for the industry. Should this be the case, then a new regulatory philosophy will have to emerge that defines the role of the operators as being an integrated crew with the responsibility to direct the plant to a more desirable or safer state than that which it would proceed if left to its own design.

A final note on Job Design and its effects relates to the SECY 81-84 "degree dialogue" to which this paper refers in its Introduction[14]. In

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14. Ahearne, J.; "Operator Qualifications and Licensing Proposed Rule" (SECY-81-84); U.S. Nuclear Regulatory Commision, Washington, D.C.: June 9, 1981.

that "dialogue," the Commissioner indicates that a BS or BE degree " . . . is a strong indicator of the technical knowledge, general aptitude, sense of responsibility, and commitment that . . . is important for reactor operators to have, particularly during an unanticipated emergency situation when procedures may not apply." The author agrees that degrees can be such an indicator but disagrees with the conclusion that they are a sine qua non for reactor operators.

Generally speaking, a professional engineering curriculum develops in a student an abstract reasoning or information processing capability that mirrors the mathematical and linguistic skills it imparts. Theoretically, a calculus based thinker should be better equipped to conceptualize rate-of-change sensitive scenarios and naturally look for the problems associated with non-linearity of function. Similarly, a linguistically sophisticated thinker should be better equipped to manipulate the temporal "aspects" or the "logic" associated with diagnostic thinking.

The theory that degree related skills may help operators in "unanticipated emergencies situations when procedures may not apply" may be seriously limited, however. Meister suggests that training is of limited value in overcoming situational factors like poor human engineering, over-load conditions, and task complexity (op. cit., pp. 45):

> The number of predisposing situational factors in Table 2-4 and the errors that may result (Table 2-5) suggest that we cannot rely on training alone to overcome inadequate situational factors. Since training is directed at modifying the individual, it only indirectly reduces the impact of situational demands by enabling the operator to cope with them more efficiently. Hence additional training or better personnel selection will never completely catch up with the situational demands. It is possible by training to mitigate the negative effects of poor human engineering or excessive task complexity, but it is impossible to eliminate these effects completely. Although we would never suggest reducing training (which is, in any case, needed to perform the job), it is apparent that only by reducing job/equipment demands (e.g., simplifying design) can

the balance between situational demands and personnel responses to these demands be accomplished.

Were Commissioner Ahearne to follow this line of thinking relative to job design and training rather than following the "point of comparison" to airline pilots on which he bases his degree argument, he might hypothesize that specific training, coupled with good human engineering and management of task complexity, can create the response capability needed to handle the unanticipated emergency situations that are his legitimate concern.

If this hypothesis is viable (I believe it is capable of being tested), then the inverse is viable as well. That is, if the information processing requirements of the reactor operator can be defined (even in the "unanticipated emergency situations" to which the Commissioner refers) and task complexity can be properly managed, then the mathematical, linguistic and scientific "constructs" needed by the reactor operator to support that processing capability can be defined as well. The relationship between information processing requirements and mental constructs is one that lends itself to direct inference; it is a relationship that can be tested. Therefore, it is a relationship that can lead us to specific, disciplined course requirements for R.O.'s[15] rather than to the gross course generalizations associated with degrees.

What the evidence suggests is that if detailed job design is undertaken by the industry, ultimate decisions will have to be made on the management

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15. We hasten to add that the "R.O." job title is unsupported by design basis evidence that "reactor operator" is indeed the essence of the control room job(s). In fact, the TMI accident illustrates that the plant automatic scram circuits circumscribe one of the most important R.O. functions, leaving the operator free to deal with the decay heat transport problem. Whether or not the training and qualifications are adequately designed to reflect this "man-machine" reality is another matter.

of task complexity. What Meister suggests is that training only helps the problem of task complexity by giving an operator the ability to "tune-out" extraneous information. Accordingly, if training is only an attenuator in the information processing or task management problem, training cannot be the sole basis on which nuclear safety is assured. What appears to be needed in addition to integrating the capabilities of the human with the self-protective ability of the nuclear power plant is computer assistance with scenario detection, i.e., operator focusing. Taken together, machine design, human design (in terms of number and types), computer assisted scenario detection, integrated procedures, and overall management of operational task complexity may provide enough opportunities for creative job reconfiguration may obviate the need for expensive and unwarranted redesign of basic nuclear power plant systems and control rooms.

QUESTION TO D. F. SHEA

Joachim Fechner

Q: Don't you admit that the systematic review process suggested by NUREG 0700 could as well help to improve the situation with respect to control room design for older plants, as some of the changes found to be necessary certainly can be retrofitted to old plants as well?

A: Yes, I agree. The review process for older plants is the same as for new plants. In both cases, the review process will reveal the operational problems inherent in the plant design. Those "problems" need to be allocated to the plant's "autonomic" function or to the "conscious" function of the operator. The allocation process in new plants has more options in this regard. In older plants, the allocation process will be constrained by existing hardware. It is my opinion that in older plants we should adopt a philosophy that job redesign (and training) should be the preferred choice for handling the "problems" revealed by the NUREG 0700 review process. In this regard, the comments of W. G. Council in NUREG 0659, Appendix A, page A-107 are particularly enlightening.

SUMMARY

SESSION VI - HUMAN FACTORS ASPECTS

CHAIRMAN - A. C. CARNINO (CEA, FRANCE)

This session included papers on a variety of job aids and design criteria of importance to operator efficiency: principles of designing written procedures (contributed by a CSNI group of experts); designing optimal computer graphic displays; setting shiftwork patterns in relation to human biological rhythm; and a trial of a prototype automated display of safety system status, which operators have found useful for self-training.

EVALUATING HUMAN RELIABILITY IN THE EXECUTION OF ROUTINE NPP TASKS - DESIGNING PROCEDURES TO IMPROVE IT

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for

the CSNI Group of Experts on Human Error Data and Assessment

ABSTRACT

CSNI formed the Group of Experts in November 1978, giving it the tasks of assembling information on task structure and typical errors in routine tasks and of laying the groundwork for international exchanges of human error data.

In 1980, the Group took up the question of what constitutes good practice in writing procedures for such NPP operational support activities as test, calibration and maintenance.

Based on the contributions made to a two-day Workshop on Task Analysis and Procedure Design in April 1980, a Guide has been completed that gives practical advice (that is not in any way a regulatory specification) to those persons, typically engineers, who actually write the test, maintenance and calibration procedures used in nuclear power plants.

The Guide is divided into several sections:

- preparing to write or revise a procedure
- drafting the step sequence
- completing the master version
- checking a new or revised procedure
- preparing user copies
- encouraging error-free use of procedures

The paper briefly discusses the rationale behind many of the items in the Guide, and the problems that arose in agreeing on several of them because of differences between countries in the philosophy of information presentation and in the "normal" behaviour expected of the personnel that use such procedures.

INTRODUCTION

Human errors made in performing routine operational tasks such as maintenance, calibration, and testing pose an obvious potential threat to the safety of a nuclear power plant, particularly if safety systems are involved. However the significance of human error in a specific case must be considered in the context of the impact of single and common mode equipment failures also to be expected. This was brought out by a case study of a real test procedure in a French power station performed by an earlier CSNI Group[1]. That study underlined the importance of considering information feedback from system to human when quantifying human reliability. Criteria were identified that the structure of a task must satisfy if quantification is to be successful. Also identified was a general need to collect more quantified information on classes and rates of human error, potential for self-correction and potential of task management to generate common mode errors.

CSNI formed the current Group (the present members of which are given in Annex I) in November 1978, giving it the two tasks of assembling information on task structure and typical errors in routine tasks (along with compensatory measures in system and procedure design), and of studying the feasibility of, and laying the groundwork for international exchanges of human error data. The Group has developed and is in the process of making trial applications of a set of categories for describing human error. They can be used in industrial incident and event reports to ensure that adequate information is collected to support improvement of human work situations and man-machine interface systems, as well as attempts at quantifying human error rates.[2] The Group is now also developing a set of guidelines on what quantification of human error is currently feasible and useful, given the different needs of reactor design, operating, and regulatory organisations.[3]

In 1980, the Group took up the question of what constitutes good practice in writing procedures for such NPP operational support activities as test, calibration, and maintenance. There was seen to be a clear need for such guidance, because procedures constitute one of the main interfaces between man and proces, and in any one plant they are often written by several people both within and external to the plant staff. Because operators tend to develop a mental image of the system they are operating, it is important that manipulations on different systems be described in a consistent manner. Procedures must give complete and unambiguous descriptions, as each user will interpret them in his own fashion. Operating personnel often move from plant to plant; thus such persons may encounter wide variations in procedure format.

The Group organised a two-day Workshop on Task Analysis and Procedure Design, held at the OECD in Paris in April 1980. Participants included national representatives from reactor research, design, operating and regulatory organisations, as well as from the chemical industry and a University Department of Applied Psychology.

Based on the contributions made to the Workshop, two successive drafts of a proposed Guide were circulated for comment, and a final version was completed in mid-1981. The result was a booklet containing practical advice (and NOT in any way a regulatory specification) to those persons, typically engineers, who actually write the test, maintenance and calibration procedures used in nuclear power plants.

The Guide is divided into several sections, as may be seen from the Table of Contents reproduced as Annex II. Following are brief descriptions of the sections, with selected examples of the points made and the rationale behind them.

Sections of the Guide

I. INTRODUCTION

The Guide begins by noting what limited regulatory requirements exist in the area of procedure design. It is pointed out that procedures for routine maintenance, calibration and testing of equipment should not be presented in precisely the same way as operating procedures. More detail must be included, for instance, but not at the expense of clarity and conciseness. Greater reliability in executing these procedures can increase the reliability and availability of equipment involved, and may be important to maintaining adequate performance of highly reliable safety-related systems.

The recommandations given in the Guide are based on general human factors principles and studies of real plant incidents. Many of them may appear obvious - but they are no less important for all of that. In effect, the 86 specific points constitute both a guide to preparing new procedures and a checklist for evaluating existing procedures with respect to good human factors practices.

II. PREPARING TO WRITE A PROCEDURE

This section discusses several factors that should be kept in mind when one begins to design the documents for a procedure, such as its precise goal, who will use it, what the consequences of erroneous use would be, and so on.

For example, one point poses the question:

How many people are required to perform the procedure?

If several people are involved, make it clear which person is responsible for overall co-ordination. If different versions of the procedure are to be written for the various persons involved, each version should incorporate a brief summary of the other person's actions, as well as the order in which they are to be carried out. The person in charge should have a complete version of the procedure.

Another point suggests that one should:

Consult potential users.

It is valuable to compare your description to their conception of how the equipment involved functions. Users will often be able to advise you on whether the proposed procedure is practical, and they will be less tempted to deviate from a procedure if they have helped to design it. They may point out errors in design, or ways of simplifying the procedure. However, if they are very familiar with the system(s) concerned, they may take for granted certain details of the task that should be included for completeness.

III. DRAFTING THE STEP SEQUENCE

This part of the booklet takes up questions that arise when one actually prepares a first draft of the complete sequence of steps in a procedure: overall structure, content of the individual steps and how they are laid out.

[Note: It is considered that a procedure is divided into several phases (e.g. system isolation, calibration, and restoration). Each phase will ordinarily consist of a group of steps which must be completed in a prescribed order. Each step consists of one (or a very few) elementary actions].

For instance, it is recommended that one:

Structure the procedure on two levels:

(i) Use headings giving the goals of each phase of the procedure (e.g. "Isolate Safety Injection Signal Train 3");

(ii) Under each heading, give the corresponding series of elementary steps to be done.

Both experienced and apprentice users will thus have a clear picture of the logic underlying the sequence of elementary steps. Out-of-place, inaccurate or missing steps will be more apparent, thus making procedure design and verification more reliable.

One important point here is that one should:

If possible, link important steps to other actions which have immediately apparent consequences if omitted.

The user can easily and inadvertently omit steps which are functionally only weakly related to the primary goal of the procedure (e.g. a check of standby channels before a circuit is isolated for testing). Equipment redesign may be called for to ensure a positive system response to manipulation. Redundancy in system response helps the user to verify that steps have been completed correctly.

Another basic point is that:

Each action should consist of a short, simple, affirmative verb in the active voice.

Avoid negative forms, passive voice or converting verbs into abstract nouns (e.g. Don't direct that, "Rotation of Knob A should not be continued after the indicator lamp B is extinguished". Rather, tell the user to "Rotate Knob A until indicator lamp B goes out").

The pros and cons of various styles of checkoff are considered in some detail:

Decide on what type of checkoff will be sufficient to confirm that all steps and groups of steps have been successfully completed.

It may be adequate just to have the user sign off at the end of the procedure. However, it may be wise instead to have the person check off each group of steps (or even each step), depending on their importance and what feedback the user has from the system. Self-verification is important; equipment should be designed as far as possible to respond positively and unambiguously to user actions.

A second person should check that important actions are completed correctly. For instance have any vital or complex calculation verified and signed off by a second person. If the procedure ends with important steps (e.g. restoration of switches and valves to service status), have a second person verify the status of the equipment and check off that the steps have been completed. (Steps near the end of a procedure are particularly vulnerable to omission).

Avoid insisting on verification of everything by a second person because the procedure could easily become cumbersome - obviously unnecessarily - and unpopular to perform. Users may be tempted to take less care in executing it if they know that all they do will be checked.

The format of each individual step is obviously very important; it was generally agreed that it is much preferable for one to:

Write the procedure in the form of a list, and in columns rather than paragraphs.

The list form gives a procedure a clear horizontal structure, thus taking into consideration that the user works through each step in the step sequence from left to right.

The column format is simpler to follow than the paragraph. The user can find his place again more easily after an interruption. He is more likely to notice omitted information. You can easily incorporate a checklist if you wish, and space to record data.

Each step should contain space for much more information than the simple instruction itself:

- step number
- checkoff mark
- the action(s) to be performed
- where the user will observe system response
- the normal system response (including, for readings, quantitative limits of acceptability)
- system setpoints and, if adjustments are required, recording of the as-left condition of the system
- recording of readings, quantitative limits of an acceptable result, and any hand calculations

- abnormal system response

- what the user should do if he obtains an abnormal system response, unacceptable reading or result of calculations

- user comments.

Even though it may be an onerous task to verify the exact style of lettering on all equipment mentioned in the procedure, it is very important that one:

> Be sure that references to equipment correspond exactly to the labels on them (including being abbreviated, in capital or small letters, arabic or roman numerals, etc.).

IV. COMPLETING THE MASTER VERSION

This section deals with those details that must be taken care of in completing the entire procedure document: information at the beginning and end, format of inserted tables, drawings, graphs, etc.

Thus for example it is recommended that one:

> Include the following items on the first page(s) of the procedure:
>
> - power plant and unit identification
> - procedure title, number and revision number
> - a place for the signature of the person authorising use of the procedure
> - date of last review and of next scheduled review of the procedure
> - list of modifications made in the procedure following previous reviews
> - if the procedure is for temporary use, the date or conditions of expiry
> - a table of contents
> - a summary statement of the goal/function of the procedure
> - explicit identification of the equipment to be worked on, and its location (room and place in the room)

- frequency with which the procedure is to be repeated (if periodic)

- prerequisite plant, system, or equipment conditions

- other actions or procedures to be completed before the procedure is used

- number and qualifications of users required, and where they are to be when performing the procedure

- precautions to be taken when the procedure is performed

- other reference documents needed

- a list of equipment and tools needed.

At the end of a procedure a simple, important, and often overlooked point is to:

> Include a notation such as "END OF PROCEDURE" after the last procedure step as an indication that a complete version of the procedure instructions has been used.
>
> The last page of the procedure is the one most vulnerable to easily unnoticed loss.

The subsection on format takes up questions of readability of a procedure, for instance, that one should:

> Indicate the relative importance of different information in the procedure by using different type sizes or type faces, indenting, underlining, frames, or lines in the margin.
>
> Spacing out words is less effective. Capital letters stand out, but the eye tires rapidly of reading them, so they should be used sparingly. <u>Consistent</u> use of a few (about three) type styles can give effective variable emphasis, but use of more will likely be confusing. Coloured printing, used sparingly, can give impact to important information.

In a similar vein, information (lists, tables, etc.) in addition to the sequence of actions to be performed may be needed. Their layout is important, too.

Thus, for example:

If you include graphs, be sure that:

- the lines on the graph paper are clearly reproducible on the copying machine to be used
- handwritten letters and numbers are well-formed and that typewritten characters are unbroken and unfilled
- in the final version, letters and numbers will be at least 3 mm (1/8") high
- the scales are compatible with the divisions on the graph paper (to avoid the need for approximate interpolation).

V. CHECKING A NEW OR REVISED PROCEDURE

A newly-created procedure must be systematically checked for coherence, completeness and accuracy. It may only then become possible to check certain important aspects. Thus one should:

Consider who will review the outcome of the procedure and what information he will need.

Adequate and unambiguous data must be recorded during execution for the needs of the person assessing the results (and for the plant archives).

Verification involves more than a straightforward double-check of the sequence of actions in a procedure. Hence one should:

Have a prospective user do a "walk-through" test execution of the procedure in your presence. This is vital to ensuring that the procedure is accurate, complete, coherent and practical.

There are two practical constraints on this. Before reactor startup, there may be little consequence to executing a faulty procedure, but it may be difficult to set up realistic test conditions and system status. When the reactor os operating, one must avoid jeopardising plant safety by performing still unproven procedures. If an actual test execution is not possible, have a prospective user simulate performing it on on the spot.

VI. PREPARING USER COPIES

Assuming that an immaculate master version of a procedure has been prepared, certain physical limitations may lead to use of documents of poor quality on the job. This section of the Guide briefly discusses these impediments, which include for instance degradation in legibility due to repeated generations of reproduction or handling.

VII. ENCOURAGING ERROR-FREE USE OF PROCEDURES

Even with a correct, well laid-out set of procedure documents, a person may employ them in such a way that the procedure is not performed correctly. The users should thus be briefed on how to execute the procedure, for instance, that they should:

- take extra care if they are interrupted while executing a procedure, and not try to remember a result until they can record it later. When they resume execution, they should verify that all the steps completed have been checked off, because it is easy to forget one's place when resuming the procedure,

and that they should:

- not "improve" the procedure or reorder the steps for convenience.

Changing the way a procedure is carried out can make secondary effects more significant (e.g. adding instrument recorders may load signal sources; substitute materials or equipment may seem at first "close enough" to that specified in the procedure, but there may be subtle, unacceptable differences).

Document control practices can affect the reliability with which procedures are executed. Even a slow updating process may become significant, because it is important to:

<u>Keep procedures up-to-date</u>

<u>This is vital</u>. Users will shun using obsolete, faulty procedures. Handwritten changes to correct outdated procedures often cause user uncertainty and incorrect execution. Whoever assesses the results of procedure use should analyse any handwritten notes to see if they indicate a need to modify the procedure. If possible, have equipment and procedures cross-referenced so that all procedures affected by plant evolution, e.g. equipment modifications or changes in reference setpoints can be quickly identified. All procedures should be reviewed

periodically (ANSI N18.7-1976 suggests every two years) and revised if necessary. Keep the delay in issuing revisions as short as possible.

VIII. AFTERWORD

Error-free execution of procedures depends on more than the documentation principles described in the Guide. Just as important are the conception of the whole work situation itself, user training, staff and work organisation and ergonomic design of equipment. Task execution is strongly affected by such simple things as poor or missing equipment labels, glare from unshielded illumination, proximity and comfortable height of instruments and controls, and having a place to put down the procedure documents while performing the required manipulations. Recent studies by EPRI [4, 5, 6] have brought into sharp focus the general principles involved here as they relate to power plant situations.

Such questions are beyond the scope of the Guide and only a detailed task analysis can identify all the important factors involved in each specific case.

Appendix II in the Guide contains an example of a recently rewritten procedure for a French power station that incorporates many of the ideas presented.

DEBATED POINTS IN THE GUIDE

Considerable discussion was required at the Workshop to arrive at a mutual understanding and consensus view on some points in the Guide. There are differences between countries in the philosophy of what constitutes the optimum content and design of procedures. For instance, in one country it may be considered adequate to provide the procedure user with the complete set of elementary steps and a checklist, thus in principle rendering the procedure usable by almost anyone and essentially "goof-proof". However another country which has a long tradition of catering to highly-trained craftsmen may consider it necessary to provide, in addition, sufficient information on the goals of the various phases of the procedure so that the user is able to understand the significance of the system response he observes. Thus in the section of the Guide dealing with Design Criteria, only the following general advice could be given when one is considering who the user of the procedure will be:

> While procedures should be written in unambiguous, consistent terms, even a "complete" description of the task inevitably assumes some foreknowledge on the part of the user. The qualifications (i.e. skills, knowledge and reading ability) of the <u>least</u>-qualified intended user will determine the

level of detail that you should include, at least for those sections that may influence plant safety. Consider whether physical or organisational constraints will affect when, how or by whom the procedure will be performed.

Another example of differences between countries is the use of specialised symbols and identification codes to identify actions and equipment, a practice which is used to varying degrees in different countries. Thus the following recommendation in the Guide represents a compromise between some experts who feel that an identification code is more accurate than a plain-language identification, and others who prefer the latter method of identification because of the risk of mis-reading a pure, non-redundant equipment code:

> Identify the specific pieces of equipment to be manipulated in each step. Do not refer to an identification in a previous step.
>
> > For instance, specify both name and identification code of valves (e.g. "Close isolation valve IV-01").

Similarly, there was some divergence of views on the question of whether each step should consist of one versus "a few" elementary manual actions. This debate led to the following recommendation in the Guide:

> Try to include only one action in each step.
>
> > Some experts feel that up to three actions can satisfactorily be combined in one step <u>if</u> they are tightly related. (e.g. "Turn switch A to position 5, observe value on level indicator B and record the value" comprises three actions). If you do sometimes include more than one action per step, then even so on average there should be less than about 1.5 actions per step over the whole procedure.

Thus the Guide constitutes more a list of items that should be kept in mind when writing a procedure than a universal prescription. It should be respected in spirit, rather than letter. National educational practices and philosophy must be taken into account as well when it is used in any particular country.

Finally, it should be pointed out that there are studies now in progress in several countries aimed at improving the documentation for procedures used in nuclear power stations. Hence while the guidance given in the Guide should definitely aid in improving currently common documentation practices, it most certainly does not constitute the last word that will be said on the subject.

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ANNEX I

Current Composition of the CSNI Group of Experts on Human Error Data and Assessment

A. Carnino (1981 Chairman)	Commissariat à l'Energie Atomique, France
H. Roggenbauer	Österreichisches Forschungszentrum Seibersdorf, Austria
F. Léonard	Centre d'Etude de l'Energie Nucléaire, Mol, Belgium
J. Rasmussen (1979-1980 Chairman)	Risø National Laboratory, Denmark
P. Gagnolet	Electricité de France
M. Griffon	Commissariat à l'Energie Atomique, France
P. Namy	Framatome, France
W.E. Büttner	Gesellschaft für Reaktorsicherheit, Federal Republic of Germany
E. Lindauer	Gesellschaft für Reaktorsicherheit, Federal Republic of Germany
W. Kutsch	Rheinisch-Wastfälische Elektrizitätswerke, Federal Republic of Germany
L. Noviello P. Moretti	Ente Nazionale per l'Energia Elettrica, Italy
G. Finetti	Comitato Nazionale per l'Energia Nucleare, Italy
T. Tobioka	Japan Atomic Energy Research Institute
J.P. Bento	Nuclear Safety Board of the Swedish Utilities
L. Skärstrom	Swedish State Power Board
G. Hensley	British Nuclear Fuels Corporation, United Kingdom

D. Hunns	United Kingdom Atomic Energy Authority, Safety and Reliability Directorate
A.D. Swain	Sandia Laboratories, United States
W.E. Vesely	United States Nuclear Regulatory Commission
G. Mancini	Commission of the European Communities, Joint Research Centre, Ispra Establishment
M. Stephens (Secretary)	OECD Nuclear Energy Agency

Also incorporated in the procedure-writing Guide are ideas contributed by the following persons who participated in the Workshop:

E. Bohr	TUV Rheinland, F.R. of Germany
Mr. Krotil	KWU, Federal Republic of Germany
Mr. Modemann	RWE, Federal Republic of Germany
K.D. Duncan	UWIST, United Kingdom
B.W. Eddershaw	ICI, United Kingdom

ANNEX II

Table of Contents of the Guide to Writing Maintenance, Test and Calibration Procedures

QUESTIONS TO A. CARNINO

Jukka Laaksonen

Q: I have more of a comment than a question. I appreciate your effort in writing guidelines, which tell how to write the procedures. However, I think there is still one more step to be taken. If you would give your typewritten procedure to a professional used to making commercial brochures and ask him to put it in a printed form with various type and size of letters, he could improve it a lot. In an example I can mention, the Westinghouse owners' group has put their emergency procedures to a new printed form, and if you compare the new procedures with the same procedures in the old typewritten form, you can readily recognize how much easier it is to use the new form.

A: I certainly agree that professional people could help in preparing the format of the procedure. But technical content cannot be addressed by those people, and our guide would be useful in this respect, at least.

Jochim Fechner **Q:** You mentioned quantification of human errors as not being necessary in all cases. Could you, please, give us a few more details on cases where you thought quantification is not needed?

A: To answer this question, I have to illustrate it by the example treated by the group of EOCD on rare events. During the assessment of a test on a safety system on a French plant, we had performed a task decomposition into 186 actions. By the observation of the task itself, we found that in these actions some of the errors were to be recovered before continuing the test. Then doing an error consequence analysis, we only kept eight actions as creating the system unavailability. As this was used for an overall reliability of the system under study, the errors significant for this purpose had to be common modes affecting all the channels of the system. There, only three actions could lead to these common modes. It is therefore feasible for quantification to focus on these actions and to collect data only on these.

ADVANCED DIAGNOSTIC GRAPHICS

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ADVANCED DIAGNOSTIC GRAPHICS

INTRODUCTION

This paper reports U.S. NRC-sponsored research at the Idaho National Engineering Laboratory (INEL) involving evaluation of computer-based diagnostic graphics. The specific targets of current evaluations are multivariate data display formats which may be used in Safety Parameter Display Systems (SPDS) being developed for nuclear power plant control rooms. The purpose of the work is to provide a basis for NRC action in regulating licensee SPDSs or later computer/cathode ray tube (CRT) applications in nuclear control rooms.

The subjects' ability to detect normal or abnormal display conditions in three SPDS formats was evaluated for this paper. This method was selected because NUREG-0696, Functional Criteria for Emergency Response Facilities, says, "The primary function of the SPDS is to aid the operator in the rapid detection of abnormal operating conditions."

The focus of this experiment is upon the ability of subjects to detect abnormal conditions displayed briefly on a pseudo-CRT screen. Measuring the subjects ability to detect abnormal conditions requires that the ability to detect be separated from the subjects willingness to respond, the response criterion. The Theory of Signal Detection (Swets, Tanner and Birdsall 1961; Green 1960) offers a solution to this problem through calculation of independent measures of sensitivity, d^1 and decision criterion, β. The reader is referred to Van Cott and Kincade, Human Engineering Guide to Equipment Design, for a short, readable explanation of the mechanics of signal detection theory.

This paper discusses the experimental method used, results, and conclusions.

Experimental Method

Subjects

Ten adult volunteers were used as subjects in this investigation. Five of the subjects are currently qualified reactor operators from the Loss-of-Fluid Test reactor plant. They have a mean 9.4 years of reactor operating experience. Each operator received his initial reactor training in the U.S. Navy. The other subjects are EG&G Idaho, Inc., engineers. They were not trained in the details of the LOFT plant or the significance of the parameters displayed on the SPDS formats. This was not considered a limitation because the detection task requires only identification of normal or abnormal display states based on color or shape changes.

Instructions to Subjects: Pretest (Detection)

Pretest instructions to subjects were generally as follows:

This is a visual-recognition experiment in which we are attempting to determine the value of various display configurations. The type of displays we are currently interested in are Safety Parameter Displays (SPD) for nuclear power plants.

During the test, you will be asked to observe the screen and report when you detect an abnormal parameter on the SPD. You will be able to control the display's appearance on the screen. You can merely identify the state that the display represents i.e., all normal parameters or some abnormal parameters. There will be three different configurations for SPD used in this experiment. Figures 1a and 1b show a typical bar-graph display in both normal and abnormal states. Note that the abnormal states are represented by red bars and by red numerical readings, which indicate the actual state of the parameters. Figures 1c and 1d are normal and abnormal meter configurations. Meter needle positions and colored numerical readings indicate normal and abnormal conditions. Normal and abnormal star configurations are shown in Figures 1e and 1f. Star shape and colored numerical readings indicate normal and abnormal conditions.

The displays will be shown to you for only a brief period of time. If you cannot determine the state of the display, abnormal/normal, make your best guess. The display will then be shown to you for a slightly longer period of time. This will continue until you are consistently making the correct response. That is, correctly identifying normal displays as "normal" and abnormal displays as "abnormal".

Apparatus

A dual-channel tachistoscope (Gerbrands Model G1180) equipped with an automatic slide changer (Model G1180) and adaptation field logic interface (Model G1159) were selected for stimulus presentation. This device was equipped with a four channel timer (Model 300-4T), two shutters, one beam splitter, and an associated shutter drive console (Gerbrands Corp. Arlington, Mass.).

Illumination Levels

Illumination levels were measured with a Gossen Cadmium-sulfide cell light meter. A hemispherical diffuser was used to measure ambient room illumination levels from the subject's test position. Spot attachments of 15° and 7.5° were used as necessary to reduce the meter's angle of acceptance when measuring illumination levels on specific areas of the rear projection screen.

On the simulated CRT display the red and green information was at an illumination of 700 LUX with an average screen illumination of 525 LUX. Average ambient room illumination throughout all presentations was 1.75 LUX.

Stimuli

The stimuli used in the experiments are 35mm slide photographs of reactor transient data displays on a cathode ray tube.

The photographs were taken with a Contax Model RTS camera using a Zeiss Planar f2.8, 60mm, macro lens. The CRT image was displayed through a Dunn Instrument Camera 631 system. Ektachrome 200 color slide film was used.

The stimuli are described in three parts: content, parameter format, and display configuration.

Content:

Stimuli content refers to the actual reactor transient data pictured on the test slides. The data come from recordings of plant instrument readings during experiments on the LOFT reactor.

The LOFT reactor is a 50-MWt pressurized water reactor used in reactor transient testing for the U.S. Nuclear Regulatory Commission. This testing has included small- and large-break loss-of-coolant experiments and other operational transient tests. The slides used in display evaluation picture normal conditions before experiments and abnormal conditions during the following types of experiments: steam load decrease, loss of primary flow, steam load increase, loss of feedwater, and small break loss-of-coolant.

Parameters:

Test slides were made in the three different formats, each format displaying the same plant parameters. These formats display data giving an overview of reactor plant conditions. Those parameters most often listed in the emergency procedures of the LOFT plant operating manual as symptons of plant transients were selected for the displays. Parameters selected for display were pressurizer level, hot leg pressure, primary coolant system flow, cold leg temperature, hot leg temperature, feed flow, steam flow, steam generator level, and steam generator pressure. The normal (green), caution (yellow), and alert (red) parameter limits were identical for each format.

Normal values and ranges of the parameters are for steady-state operations. Thus, a "normal" operation at LOFT such as a slow-power ascension may cause one or more parameters to leave the prescribed normal range temporarily.

All of the displays represent normal values as a central value in the display with the range bracketing that normal value. For some parameters the normal value and range are fixed and for others the normal value and range are a direct function of reactor power (i.e., the normal value of feed flow increases with increasing reactor power).

The central 85% of the range is green, the 10% adjoining the green is yellow and the outer 5% of the range is red in all test displays. The green is for normal, yellow for caution, and red for alert in the color standard used for these displays.

Display Configuration

Three representative SPD formats were used in the test slides:

1. The star diagram, analogous to the Westinghouse Electric Corporation's iconic CRT display.

2. The deviation bar format, used in at least one power plant SPDS design (Palo Verde).

3. The meter display, which was developed because groups of meters may be used to provide seismically qualified SPDS backup (NUREG-0696).

Each display format shows control rod status in a box to the left, date/time in the lower left, and reactor power at the bottom. The only difference in the displays is the method used to show normal values, ranges, and interrelationships between nine analog parameters. The display formats are shown in Figure 1.

Deviation Bar (Figures 1a and 1b). In this display a central vertical line indicates the normal value. Parameter deviations from this value show as bars to the left or right of normal. High- and low-range values are shown as vertical lines. Parameter descriptions and digital values are on the right of the display. As parameter values reach the 85% (green-yellow) and 95% (yellow-red) barriers, the bar indicator and digital values on the display change to the appropriate color. On this display, parameters for the primary coolant system are grouped at the top and secondary system parameters are grouped at the bottom.

Meter Display (Figures 1c and 1d). Parameter values on this display are represented by needle positions on nine meters drawn on a cathode ray tube. The green, yellow, and red ranges are shown on the meters with only the color corresponding to the current parameter value shown. Digital values (color coded) and parameter descriptions are inside each meter.

Circular Plot (Star) (Figures 1e and 1f). This display represents parameter values as positions on the spokes of a circle. A small inner circle represents range minimums with an outer circle representing maximums. Current-value spoke positions are tied together to form a nine-sided polygon. Digital values and parameter descriptions are shown around the outside of the maximum-range ring. Background rings show the 85% range values. Digital parameter indications change color corresponding to 85% and 95% values.

Visual Angle. From the subject's test position, the simulated CRT display subtended a horizontal visual angle of 13.4° and a vertical angle of 11.4°.

Procedure: Operator Training

Operator subjects were given more extensive training than engineer subjects. This training was to prepare them for future display testing that will be more complex than detection testing. Each of the operators was briefed on the three SPDS formats and on the normalization schemes

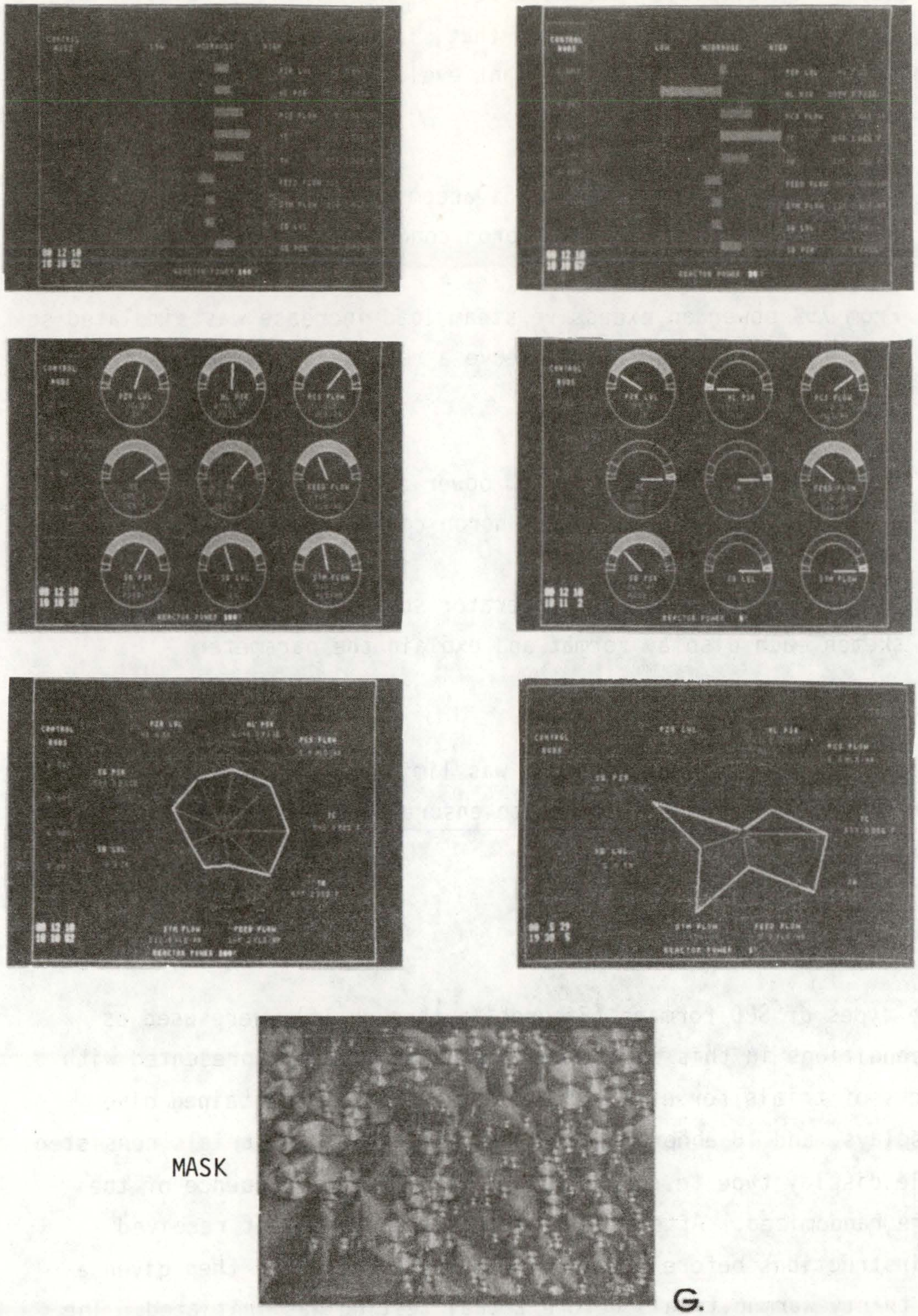
NORMAL ABNORMAL

BARS

METERS

STARS

MASK



A. B. C. D. E. F. G.

Figure 1. Test Display Examples.

(normal values) for displayed parameters. An engineering simulation of LOFT was used to drive each display so that, in real time, each operator subject observed the same simulated plant evolutions on each display format. The evolutions were:

1. Power ascension from 50 to 75% accompanied by changing and draining to reduce primary boron concentration.

2. From 75% power an excessive steam load increase was simulated so that the operators could observe a reactor scram caused by low pressure.

3. Power descent from 100 to 75% power accompanied by charging and draining to increase primary boron concentration.

Following simulation training, each operator subject was required to correctly sketch each display format and explain the parameter normalizations.

Training of the engineer subjects was limited to their viewing each display in normal and abnormal states to ensure that they knew how these states were represented.

Procedure: Testing

Three types of SPD formats (Figures 1a through 1f), were used as separate conditions in this experiment. Each subject was presented with three blocks of trials for each condition. Each block contained nine normal displays, and 18 abnormal displays. Each block of trials consisted of a single display type (e.g. meters). The order and sequence of the trials were randomized. After their training, each subject received detailed instructions before the session began. They were then given a series of thirty warmup trials before actual testing was initiated. The test display was first presented for 5.0 ms. The exposure duration was then increased by 10 ms per display until the subject made no errors during three successive blocks. The subjects responses were recorded at each exposure duration. Between every presentation an intertrial masking slide was displayed to eliminate the possibility of establishing latent images. The mask consisted of a photograph of a color pattern which conveyed no information (Figure 1g). The order of presentation of the test blocks was balanced across subjects and type of display configuration. The subjects were given a 15 minute rest between display configuration changes.

Design

The experiment was designed to manipulate the following independent variables:

- o Type of display configuration (bar, meters, and star)
- o Display condition (abnormal and normal)
- o Type of subjects (operators and engineers)
- o Exposure duration.

The dependent variables for the pretest were:

- o Response accuracy--percent correct
- o The subject's perceptual sensitivity (d')
- o The subject's response criterion (β).

RESULTS

The data from this experiment are shown in Figure 2. An analysis of variance of these data was conducted in three separate parts. The first part anlyzed the perceptual sensitivity (d') of the subjects as a function of display type and exposure duration as shown in Figures 2a and 2d. The analysis revealed that both display type ($F(2,16) = 10.88$, $p<.01$) and exposure duration ($F(7,56) = 21.17$, $p<.01$) are significant main effects, i.e., sensitivity changes with display type and exposure duration. In addition, a significant interaction was shown for the display type and exposure duration ($F(14,112) = 2.15$, $p<.01$), i.e., sensitivities changes as a function of two variables, display type and exposure duration.

Since one of the objectives of this experiment was to evaluate the three display formats, orthogonal planned comparisons of the data were conducted. The first comparison was meters versus bars and star. This comparison revealed a significant difference ($t(16) = 4.oz$, $p<.01$), i.e., bars and stars were better for detection than meters. The second comparison--bars versus star--showed no significant difference for detections.

The second part of the analysis considered the accuracy of the subject's responses in terms of percent correct. These data are plotted in Figures 2b and 2e. As with sensitivity, two main effects were found, display type ($F(2,16) = 12.94$, $p<.01$) and exposure duration ($F(7,56) = 24.13$, $p<.01$). Again, a significant interaction was found for display type by exposure duration ($F(14,112) = 5.15$, $p<.01$). Orthoganal planned comparisons of the data on the type of display format showed the same pattern of results as the first part of the analysis, i.e., the difference between meters versus stars and bars was significant ($t=4.90$, $p<.01$). The second comparison, bars versus star, showed no significant difference.

The third part of the analysis examined the data in terms of the subject's response criterion (β). These data are plotted in Figures 2c and 2f. The only significant main effect was exposure duration ($F(7,56) = 9.58$, $p<.01$). No interactions were found to be significant and no comparisons were conducted on these data.

CONCLUSIONS

Star and bar formats, treated together, are better than the meter format for the detection task. This conclusion is directly supported by statistical comparisons.

Another conclusion is that the star format is better than the bar format for the detection task. This is not directly supported by statistical comparisons but does represent the trend of the data shown in Figures 2a, 2b, 2d and 2e.

Thus the star format apparently transmits information concerning parameter conditions better than the bar format which is, in turn, better than the meter format. Differences are shown in viewer requirements for longer exposures (more information) to accurately assess the condition of the display.

Interestingly, the rate of change in the subject's ability to extract information seems greater with the meter format, perhaps due to a ceiling effect on the subjects responses using the star and bar formats. Given that the order of presentation for exposure duration was fixed (5 to 75ms), the subjects may have been engaging in more perceptual learning from the meter displays than from the other two formats.

Longer exposure times increased the measured sensitivities of the subjects and produced more correct responses. This is not surprising, since the amount of information available for making a decision would usually increase with a longer exposure, and the more information available, the better the decision would be.

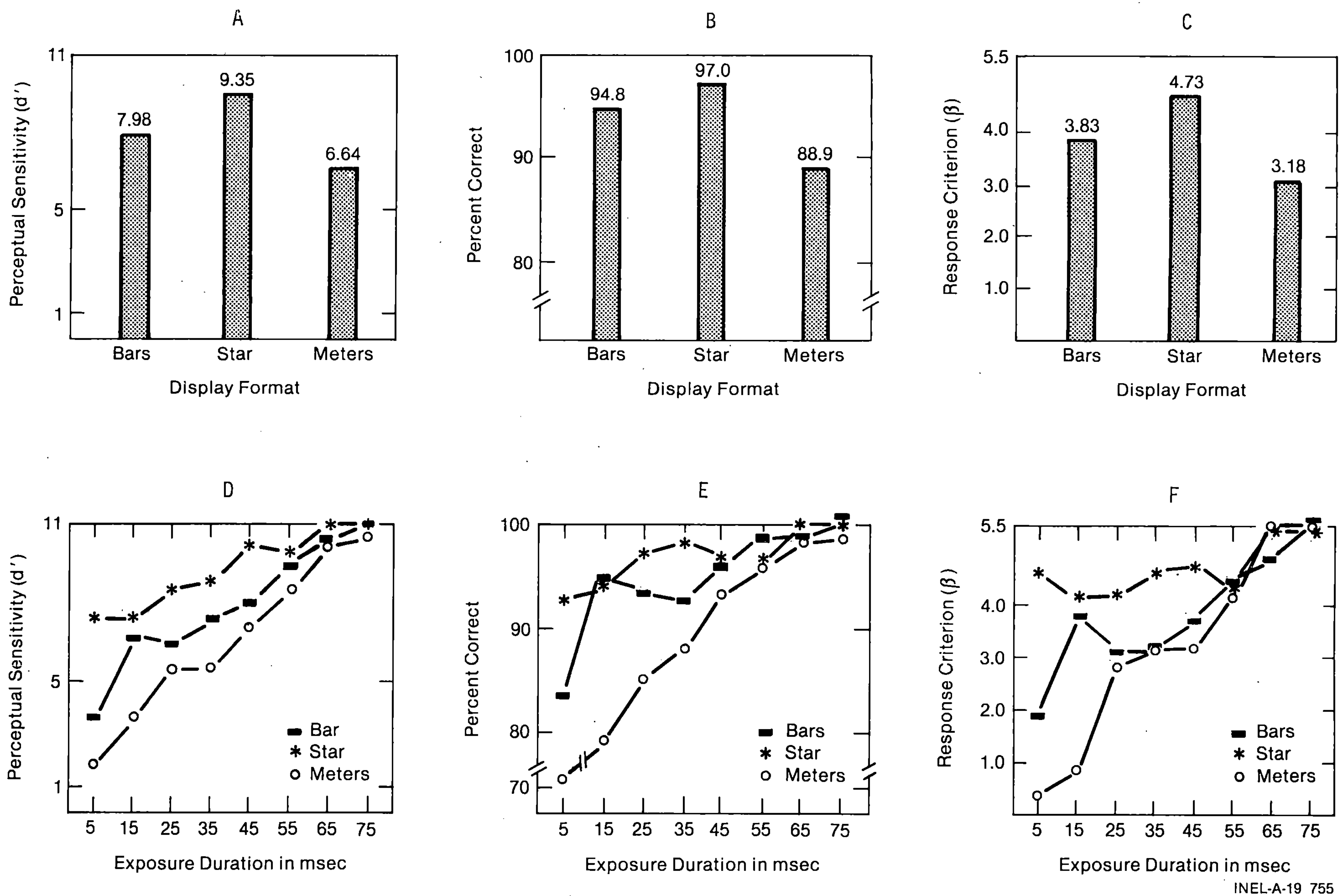


Figure 2. Experiment Results

The background and experience of the subjects was not a significant variable in this task. Operators could not be distinguished from nonoperators on the basis of performance. Therefore, we can assume that the detection task is a purely perceptual in nature and is not impacted by the differences in training and experience these two groups.

Finally, there was no differential effect of display format on the response criterion used by the subjects. Therefore, the differences shown in the accuracy measure are not due to shifts in the strategy of subject responses. It must be recognized that detection represents only the first stage of cognitive processing. Later stages of cognitive processing may have larger role in determining the overall effectiveness of modern decision-aiding techniques such as, the SPDS. The reader is therefore cautioned not to extrapolate the results of this study beyond the context of visual detection. This work is a first step in evaluation of advanced diagnostic graphics, such as an SPDS, using performance measures of cognitive processes found in a nuclear control room, i.e. detection of abnormal conditions. The next step in the evaluation work is to examine subjects ability to recognize specific parameter conditions using various SPDS formats.

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QUESTIONS TO MICHAEL A. BRAY

Pierre Lienart

Q: Your tests were performed to select the best mean about selection. What about identification?

A: Currently, tests are being run using parameter identification times and accuracy as performance measures. Future plans include use of event classification (i.e., identification) as a performance measure.

Patterns of Shift Work in the Power Industry: The Need for Circadian Chronohygiene in Bioengineering at the Man-Machine Interface

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Patterns of Shift Work in the Power Industry: The Need for Circadian Chronohygiene in Bioengineering at the Man-Machine Interface

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ABSTRACT

A common and continuing problem in the power industry is the selection of optimally designed schedules of rotation--or of "ROTAS"--for shift-workers. A concomitant problem is the selection and monitoring of the worker and the ministering to his needs in order to maximize on-the-job performance and minimize hazards to his health. A survey that we have made of the world power industry shows that the ROTAS most commonly in use are those that combine slow rotation with phase advance. From the theoretical point of view, from practical experience, and from experimental studies of simulated shift work reported below, these are conditions that assure bad circadian chronohygiene and that cause circadian dyschronism and poor performance with respect to visual acuity and elementary cognitive function. In animal studies that modeled closely all of the major shift work rotation protocols commonly used in the power industry, significant differences were seen among the protocols in the ability of the animals to adjust their circadian rhythms to the shifts. Four types of rotation protocols were investigated over the course of more than half a year, during which time about 17,000 independent measurements of deep-body temperature and of food consumption were made on each of 36 separately housed rats in a long-term, residential, controlled environment, data acquisition system. The protocols simulated: (1) slow rotation by phase advance (most commonly used in the US power industry and in use at Three Mile Island 2 years ago), (2) slow rotation by phase delay (the preferred method, see below), (3) rapid rotation (1-1-1) in which a new shift occurs daily, and (4) rapid rotation (2-2-2) in which a new shift is introduced every other day. (These rapid rotation schedules are commonly used in Japan where two operator related accidents occurred earlier this year.) Dietary measures designed to accelerate circadian phase shifts were studied by comparing: (1) high protein breakfasts versus high carbohydrate breakfasts, and (2) groups of animals consuming their meals on "days off" (e.g., "weekends") at times appropriate to the day shift versus groups of animals consuming their meals on "days off" at times anticipating their next shift. These experiments clearly showed that rotation by phase delay with anticipatory meal-timing on "days off" resulted in the most rapid phase adaptation to a new shift with a minimum

disturbance of the normal circadian temperature rhythm. All of the other protocols, including especially the widely used 1-1-1 and 2-2-2 rapid rotation schedules were seriously dyschronogenic--a condition not unlike jet-lag--and therefore as we extrapolate to the human experience, to be avoided by all workers who may have critical decisions to make. Accordingly, modified conventional ROTAS, and new ROTAS that approach optimal design will be discussed, along with tested chronohygiene mitigation and circadian measurement techniques that deal effectively with shift-work fatigue and its hazards.

INTRODUCTION

The most serious single omission in extant studies of human error rate prediction is in the failure to identify the contribution made by the stresses and stressors associated with shift-work. Furthermore, where shift work studies have been made, they have been chiefly in the form of descriptive characterizations of problems arising from the various ROTAS rather than in the form of interventions designed specifically to ameliorate such problems. In this paper, through an interplay of experiments involving animal models as simulators and human subjects we want first to characterize in biochemical and physiological terms the connections of circadian rhythms to the problems of the shift worker, and next to show some of the fairly simple steps that can be taken to ameliorate such problems.

Circadian Glossary

Ultradian- less than a day $\tau < 24$ h

Circadian- about a day $\tau \approx 24$ h

Infradian- more than a day $\tau > 24$ h

Chronotype; Zeitgeber; Chronobiotic; Dyschronism; Dyschronogenic

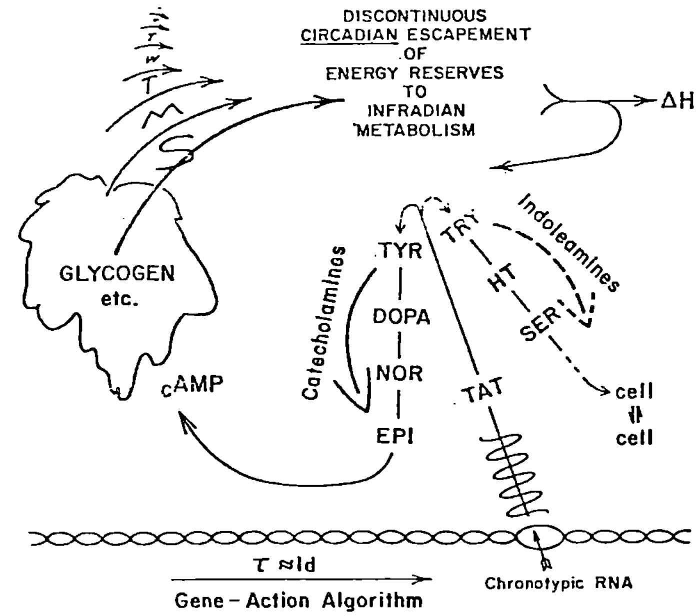

Figure 1. The circadian clock is pictured as an energy reserve escapement, with alternate path options for the active phase (catecholamines pathway dominant) or the inactive phase (indoleamines pathway dominant) in the circadian cycle. (Reproduced from Reference 1 with permission of Williams and Wilkins, Baltimore, 1977).

TABLE 1 ARGONNE SHIFT WORK SCHEDULING SURVEY

	North America		Western Europe, Africa and Asia	
	Nuclear	Non-Nuclear	Nuclear	Non-Nuclear
Satisfied with Schedule Now in Use	9	7	14	5
Need for Minor Improvements	10	9	7	1
Need for Major Improvements	6[a]	11[a]	0	1[b]
No Opinion Expressed	3	7	2	0
Shifts Rotate by Phase Advance ($\Delta\phi^{+}$)	15	22	4	4
Shifts Rotate by Phase Delay ($\Delta\phi^{-}$)	6[c]	7[c]	12[c]	-
Shifts Rotate by Both $\Delta\phi^{+}$ and $\Delta\phi^{-}$	4	3	-	1
Fixed Shifts (in some operations)	(2)	(3)	0	0
12 h Shifts, Rapid Rotation (2-3 d)	0	1	0	0
8 h Shifts, Rapid Rotation (2-4 d)	0	0	8	4
8 h Shifts, Slow Rotation (5-7 d)	28	33	15	3

[a]Of these 17 plants, 11 rotated by phase advance, and 3 by both phase advance and phase delay.

[b]This is one of the 4 plants in this group rotating by phase advance; some respondents failed to give information regarding phase change of the rotas (-).

[c]Of these 25 plants, 19 expressed "entirely satisfied", and of the remaining 6, 4 expressed only the need for more time for training purposes.

(after Table 1, Reference 2)

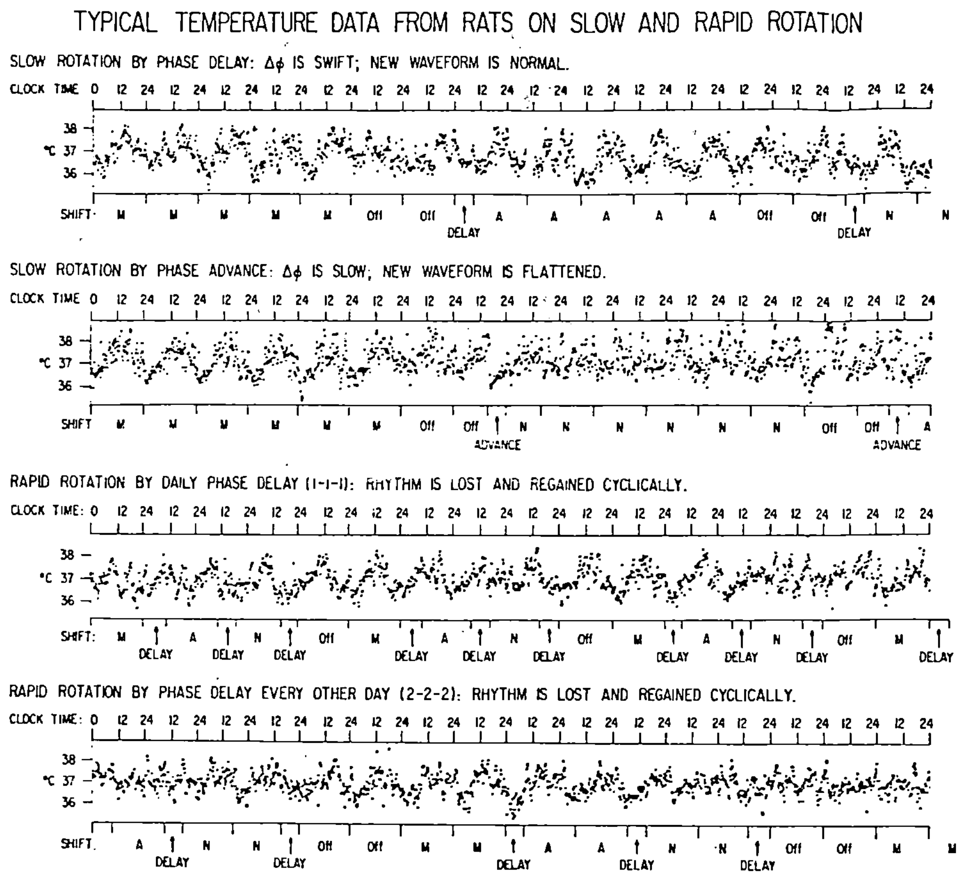

Figure 2. In this study we have simulated four of the most commonly used shiftwork protocols and determined their effect on the circadian body temperature rhythm in rats. Slow rotation by phase delay resulted in the most rapid phase adaptation to the new shift with minimal disturbance of the normal circadian temperature rhythm. (From Reference 6)

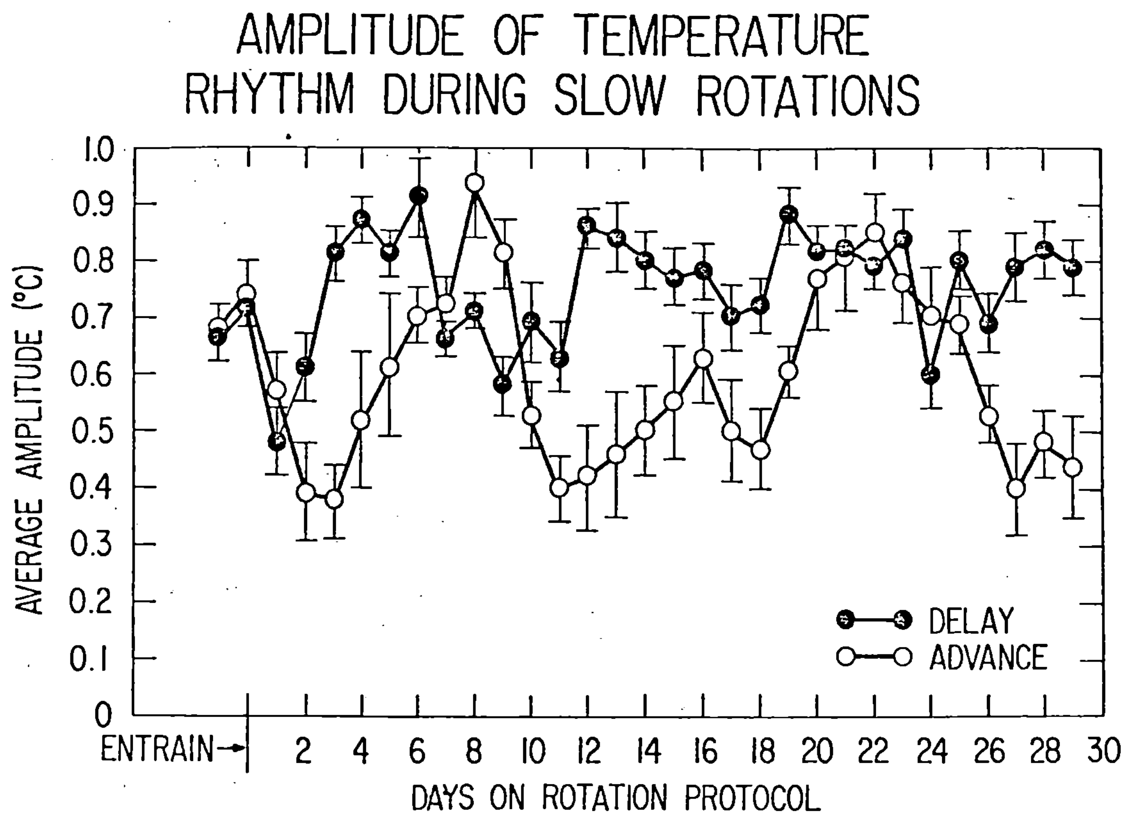

Figure 3. During slow rotation by phase advance the amplitude of the circadian temperature rhythm is reduced to a greater extent than during slow rotation by phase delay. The rhythm is more labile under phase advance conditions as shown by the large, continuing fluctuations in amplitude.

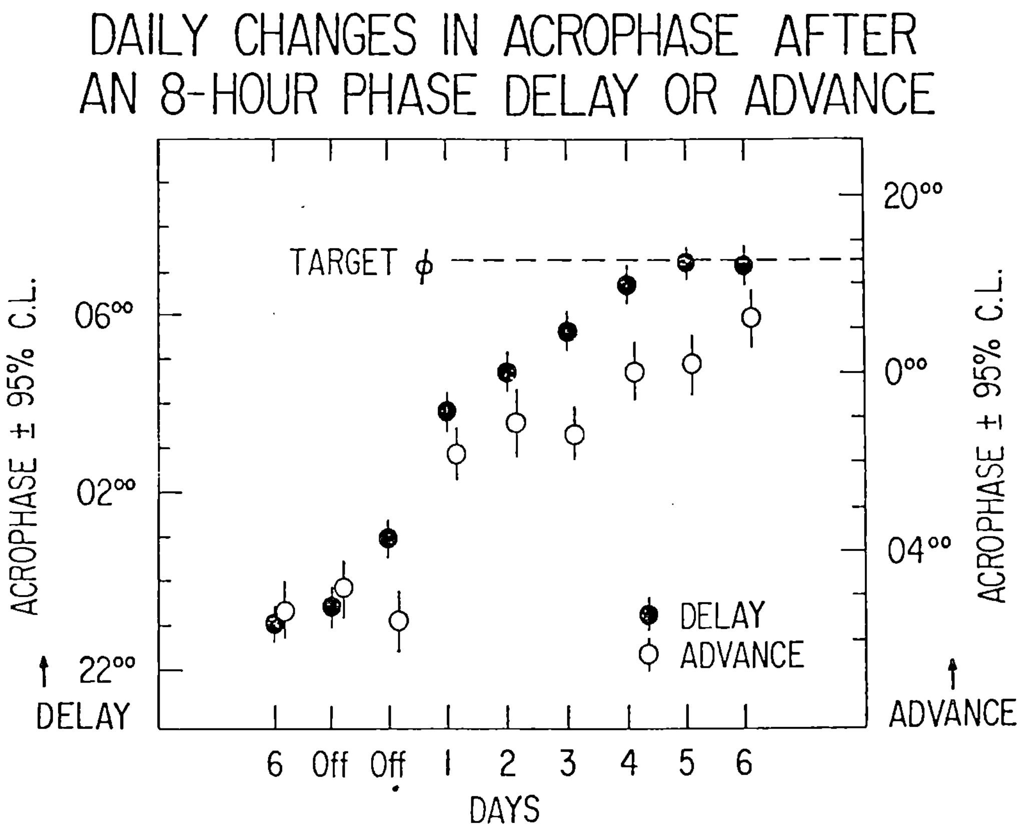

Figure 4. Rats subjected to an 8-hour phase delay adjust the acrophase of their temperature rhythm more quickly and completely than rats subjected to an 8-hour phase advance. Less disruption of the circadian rhythm (as shown by smaller 95% confidence limits) occurs with phase delay than with phase advance.

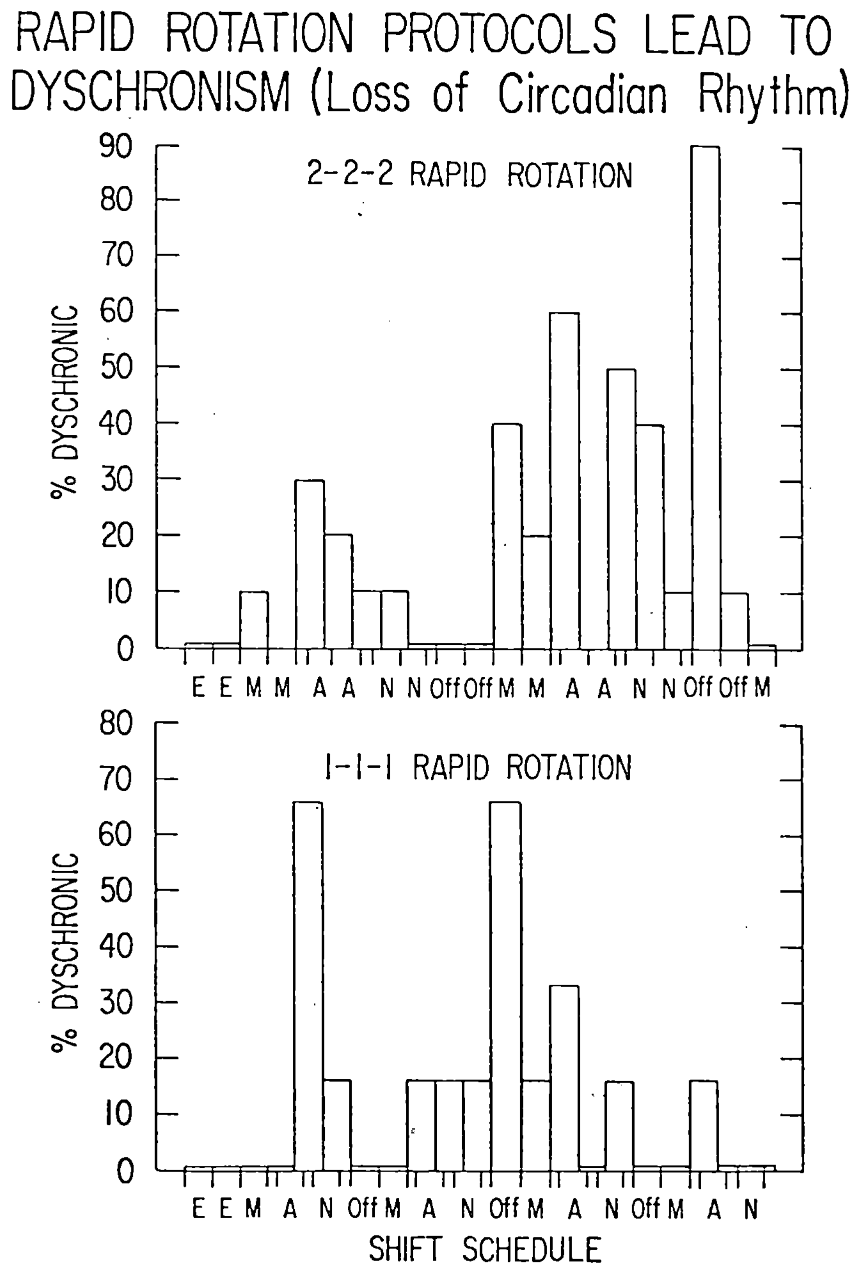

Figure 5. Rapid rotation protocols lead to dyschronism (loss of circadian rhythm). The percent of animals with no significant circadian temperature rhythm (as determined by cosinor analysis) increases as a function of the length of time spent on a rapid (2-2-2) rotation protocol. Animals also become dyschronic on a (1-1-1) rapid rotation schedule, but after a while the number of dyschronic animals decreases, as many of them begin to free-run. Complete loss of the circadian temperature rhythm was not seen in any animals subjected to slow rotation protocols.

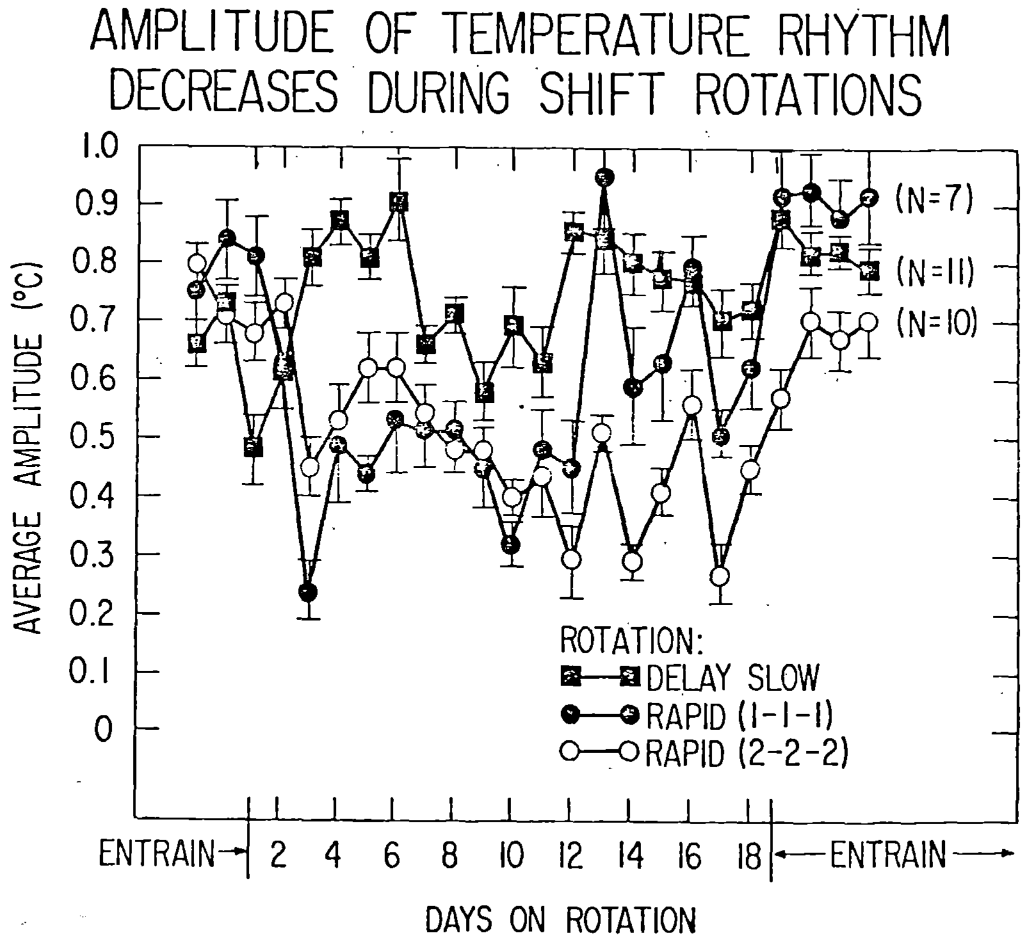

Figure 6. All the shift rotation protocols disturbed the normal circadian temperature rhythm as evidenced by decreases in the amplitude of the rhythm. However, slow rotation by phase delay was markedly better in maintaining a high amplitude than either of the rapid rotation protocols. Initially both rapid rotation protocols significantly reduced the amplitude to the same degree. But as the animals on (1-1-1) rapid rotation began to free run, the amplitude recovered and was similar to that observed for slow rotation.

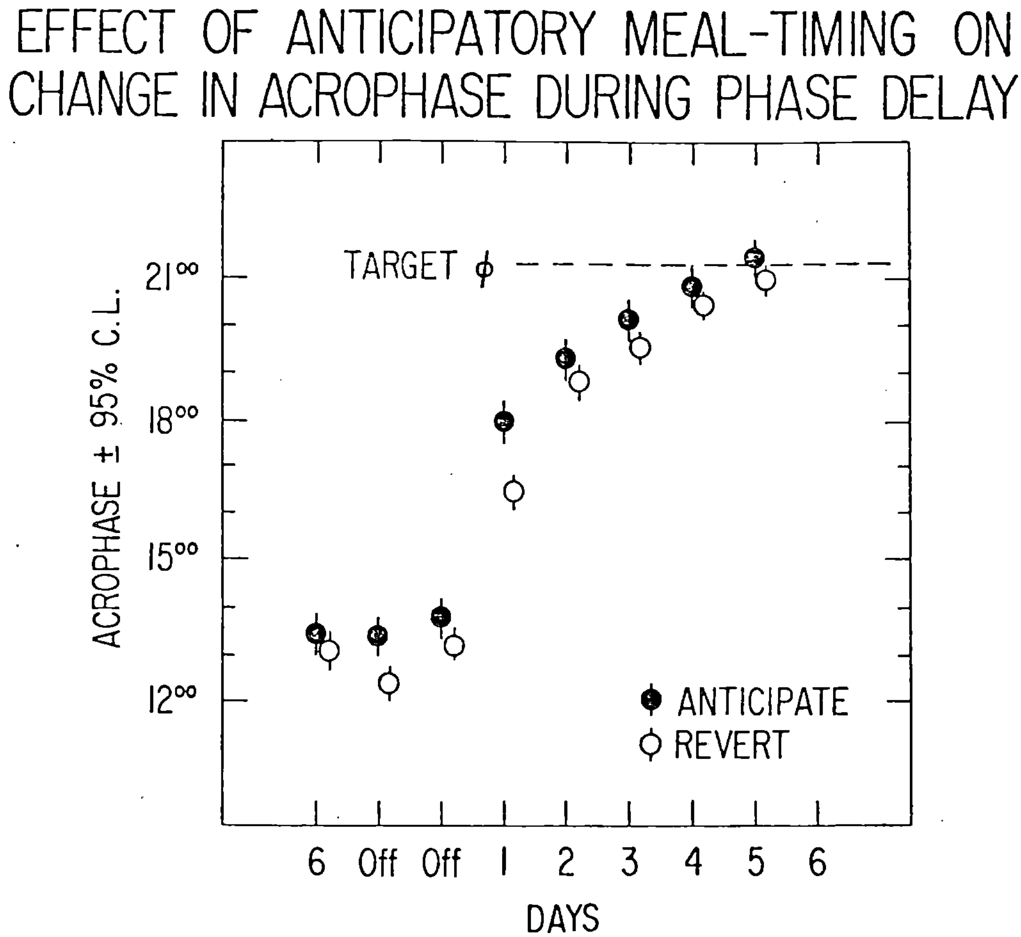

Figure 7. Anticipatory meal-timing on "days off" had a slight but consistent effect in accelerating the adjustment of acrophase to the target phase as compared to groups of animals who "reverted" to consuming their meals on a fixed morning shift schedule.

THU Old Shift	FRI Last Day, Old Shift	SAT Day Off	SUN Day Off	MON New Shift
ROUTINE	FEAST	FAST	BREAKFAST NEW TIME	ROUTINE

(see below)

Day Off #1 SATURDAY Old Time		Chronophase Delay → 8 hours →	Day Off #2 SUNDAY New Time	
MORNPHASE	EVENPHASE	MORNPHASE	NOONPHASE	EVENPHASE
• Sleep late • Several cups of black coffee or tea	• Eat sparingly • Avoid carbohydrates (light salads, fruit OK) • No coffee or tea unless decaff	• Lights on, exercise • Big high-protein breakfast • No caffeine	• Big high-protein lunch	• High-carbohydrate/low-protein supper • To bed on new time

Figure 8. Chronohygiene for Delay Rotation: A diet plan to implement circadian phase delay during slow rotation.

MORNPHASE	NOONPHASE	RESTPHASE	EVENPHASE
m 0700 *a* 1500 *n* 2300	*m* 1200 *a* 2000 *n* 0400	*m* 1530 *a* 2330 *n* 0730	*m* 1800 *a* 0200 *n* 1000
BREAKFAST	LUNCH	TEATIME	SUPPER
• Lights on • Exercise • Big, high-protein breakfast • No caffeine	• No naps • Big, high-protein lunch • No caffeine	• Coffee, tea or cocoa • Sweets	• High carbohydrate, low protein • No caffeine • Low-key exercise

Figure 9. Routine workday chronohygiene for the shiftworker. A guide to an orthochronally proper daily routine for shiftworkers on the morning (m), afternoon (a), or night (n) shift. All times that are given may easily be scaled to advance or delay by ±1 h to adjust to local conditions and preferences.

CONCLUSIONS

In a study with animal models simulating the major shift rotation schedules (ROTAS) currently in use in the nuclear power industry, we compared different slow rotation protocols (weekly rotation) with one another and with rapid rotation protocols (rotation daily, "1-1-1", or every other day, "2-2-2"). The direction of rotation for the slow ROTAS was either by phase advance or by phase delay. We found that 1) phase delay is "better" than phase advance, in agreement with the human "satisfaction index" in our earlier survey (Table 1). 2) Anticipatory meal-timing is slightly "better" than reverting to the morning shift during days off. 3) Slow rotation is "better" than rapid rotation. 4) Rapid rotation (1-1-1) is "better" than rapid rotation (2-2-2). In each case, "better" means more rapid and complete adaptation of acrophase of temperature rhythm to new shift; minimal reduction of amplitude of temperature rhythm, and minimal incidence of dyschronism (loss of rhythm in body temperature).

It is clear that many of the circadian connections to the shift work problem have now been identified in basic physiological, biochemical and behavioral terms, and that these connections have far-reaching implications in the proper design of shift rotation schedules on the one hand, and of chronohygiene methods on the other for workers in shift-work industries.

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QUESTION TO DR. EHRET

Joechim Fechner

Q: Did you investigate the influence that rest-phases might have on a desyncronization on the circadian rhythmicity already present?

A: Other investigators, especially Naitoh at the U. S. Naval Research Laboratory, have looked into this. If "rest" is so long as to include REM sleep, then it appears to be always counter productive--on the day of the "long nap," as well on subsequent days since sleep at the wrong time is also a zeitgeber. If "rest" is brief, and does not include sleep, it appears desirable to have recuperative affects. The fine tuning of this problem, in terms of the underlying ultradian episodes of each active phase of the day, remains to be done.

IMPLEMENTATION OF AN AUTOMATED STATUS ANALYSIS SYSTEM IN AN OPERATING NUCLEAR POWER PLANT

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Contents

Text of a paper presented at the OECD Nuclear Energy Agency CSNI meeting on Operator Training and Qualifications at Charlotte, North Carolina, U.S.A., October 12-15, 1981.

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IMPLEMENTATION OF AN AUTOMATED STATUS
ANALYSIS IN AN OPERATING NUCLEAR POWER PLANT

J. Christenson, T. Graae, H. Roggenbauer

1. INTRODUCTION

A basic axiom of every power reactor operator is that he must always try to operate the reactor in conformance with the Technical Specifications requirements established by the regulatory agency from which he receives the authority to operate the reactor. Despite the best intentions of operators, the actual execution of this simple axiom proves to be a demanding, time-consuming task, and plant operating histories show numerous examples in which the axiom has been violated. In Sweden, and most other countries, the Technical Specifications are stipulated by the regulatory authorities and establish the framework (or envelope) within which the plant can be legally (and safely) operated. The specifications consist of a set of regulations which fall into four categories:

1. Administrative reporting requirements
2. Acceptable limits of process parameters
3. Routine periodic testing requirements
4. Conditions and restrictions on reactor operations related to the functional status of various safety systems.

Accurate and reliable execution of the regulations in the last category is made difficult by the combined complexity of the safety system (which may consist of as many as 5,000 functional elements) and the regulations (typically 50 or more typewritten pages). This paper describes an experimental system designed to both increase the reliability with which regulations in this category are executed and to also make this task easier for the operator to carry out. The system has been implemented on the Forsmark 1 Reactor and its efficacy is now under evaluation. The experimental system is known as "RGB-F" and is the result of a cooperative research project involving Forschungszentrum Seibersdorf (formerly Studiengesellschaft fur Atomenergie), Austria, OECD Halden Reactor Project, Norway, and ASEA-ATOM, Sweden.

The letters RGB are derived from the German phrase "Rechnergestutztes Betriebshandbuch" which was the name used for the original system concept. The first experimental system of this type was developed for the Austrian Zwentendorf reactor, and was conceived as having a broader scope(1).

In the experimental version of RGB-F which has been implemented at Forsmark 1, the size of the system has been restricted to selected reactor systems and the scope of the status analysis has been truncated so that it is performed only with respect to the requirements of the Technical Specifications. Significant benefits to the reactor operator were demonstrated during the course of the operation of the RGB-F system. However, it is also evident that even further benefits can be realized from a full-size, full-scope system, and some of these aspects are discussed in Section 7.

The central philosophy of the Technical Specifications is that the status of the reactor's safety systems must be such that for any Postulated Initiating Event (PIE), the system is capable of automatically providing 100% of its design basis Protective Action (PA), and that it must be capable of this response even in the event of the failure of any single active element of the system (2-4). Because of this requirement, safety systems are designed with significant amounts of redundancy and diversity, and as a result can often carry out their design basis PA's even when some of their elements are either in a degraded state or inoperable. Under these circumstances the requirements of the Technical Specifications place certain limitations on the continued operation of the reactor. These limitations take the form of different time limits for repair of the faulty elements, requirements for extra testing or surveillance, lower power levels, or in certain cases reactor shutdown within a specified time limit. The specific limitation depends upon the number and combination of inoperable elements (degraded elements are assumed to be inoperable from the standpoint of applying the Technical Specifications). The purpose of the RGB-F system is to automatically analyze any arbitrary status of the systems included in the experiment and to inform the operator of the limitations the Technical Specifications place on that status.

The RGB-F system contains as its main elements a carefully programmed computer and an interactive input/output terminal designed for use by the reactor operator. By means of just a few keystrokes the operator can input into the RGB-F system the status of any element in the safety system and receive in return a complete analysis of the limitations and restrictions placed on the reactor as a result of the current status of the safety system. The RGB-F system also has several other components and supporting functions. However, the fundamental objective of the system is one just described. By utilizing the RGB-F system the reactor operator can quickly and reliably determine the constraints implied by the Technical Specification requirements because of the current status of the reactor safety system.

Figure 1 shows a generalized version of the system's inputs and outputs.

2. FUNCTIONS OF THE RGB-F SYSTEM

The experimental RGB-F system is designed to accomplish the following reactor operator oriented tasks for all parts of the reactor safety system and Technical Specifications which are included in the experimental system:

1. To allow the operator to enter into the system the state of all safety system elements. Safety system elements are defined as arrangements of one or more components which in aggregate perform a particular function associated with the operation of the safety system. Elements have two possible states: operable and inoperable.

2. To allow the operator to enter into the system the current operating status of the reactor (power, nuclear heating, hot standby, hot shutdown, cold shutdown, etc.).

3. To record the information from tasks "1' and "2' above (including time of entry) and to allow the operator to obtain systematic displays showing the current state of all system elements and the reactor mode.

4. To allow the operator to establish two distinct modes of system operation: "Test" and :Record". In the Test Mode operator entries are considered only as trials to investigate the results of proposed changes in either the state of safety system elements or the reactor mode. In the Record Mode operator entries are treated as reflecting genuine changes in the state of the reactor systems.

5. To advise the operator upon request of all limitations that current state of the safety system elements places upon the reactor remaining in the current reactor mode. Limitations in this context include time limits and special tests and procedures.

6. To advise the operator whenever the status of safety system elements requires the termination of the current reactor mode and to also advise him on the reactor modes which can be legally entered when the current mode is terminated.

7. To furnish advisory messages and alarms at specified times prior to any requirement for reactor mode termination whenever the current state of the reactor safety system will

cause the future occurrence of a mode termination requirement.

The foregoing tasks are all carried out at the operator input/output station operating in an interactive, real-time mode, which means that all system responses to operator instructions occur within one second after the instruction has been entered into the system. Physically the station consists of a Nord Color Terminal (NCT) including the color TV-screen, controller and a keyboard with alphanumeric, numeric block and special function keys.

3. STRUCTURE OF THE RGB-F SYSTEM

Figure 2 shows the hardware configuration of the experiment. Reactor operator communication takes place at an input/output (I/O) station located in a room next to the main control room where work permits are prepared. The I/O station consists of a keyboard and a color CRT-screen which are connected to a Color Terminal Controller and a Modem which transmits and receives information to a similar unit in the Administration Building. The Modem in the Administration Building communicates with the main computer (a NORD-100) which contains the programs and data for RGB-F system. The computer is also connected to various other I/O devices in the Administration Building which are used to monitor and record the results of the experiment.

At the operator I/O station, information and commands are entered into the RGB-F system via the keyboard using normal alphanumeric keys and 8 special function keys. Output information is presented on the color CRT-screen.

Roughly speaking, the contents of the RGB-F system computer consist of utility communication routines, analysis programs and "Operability Tables". The Operability Tables contain lists of elements. A single element is actually aggregate of components which form functional unit (e.g., "pump A" includes the pump itself and the assocated control unit and switches). In practice all of the elements are assigned specific names (6-9 alphanumeric coded characters) and the names are entered into various columns in the Operability Table.

Each Operability Table column contains as its entries the names of all of the elements in a safety system channel required to perform a particular safety function. In a reactor safety system, there are several redundant channels for each function, and the columns describing each redundant channel are grouped together into a Table. In this way, every safety function required by the Technical Specifications is described by a Table which has as its columns the elements that make up each redundant channel for that function.

Figure 3 shows the basic structure of a typical table, in this case consisting of four columns indicating four redundant channels, each of which has the function of sensing a particular reactor condition and delivering the appropriate signal to the reactor scram system when that condition occurs. The analysis program determines the operability of any particular channel based on the status of the column-row entries in the column corresponding to that channel. The basic rule for column operability is that a column becomes inoperable whenever any one of its column-row entries is declared inoperable, and conversely a column is operable only when all of its column-row entries are operable. In many cases a column-row consists of only a single element and the operability status of the column-row is equivalent to that of the element. In other cases, a column-row entry may consist of two or more elements as is shown in Figure 3 where there are two elements within each channel which can measure the neutron flux in the intermediate range (elements 531K976 and 531K978 for Channel A). In this particular case the operability rule for this column-row entry is that at least one of the elements in the column-row be operable. In other cases, column-row entries consist of many elements and the operability rule for the column-row is that some specified minimum number of the elements be operable. Based on operability rules of this type, which may vary from column to column and from Table to Table, the RGB-F analysis program determines the operability status of all of the safety functions, compares these to the Technical Specification requirements and informs the operator of any restrictions which the Technical Specifications place on reactor operatons because of the status of the safety system. The most stringent restriction on reactor operations (generally the minimum amount of time which the reactor is allowed to reamin in its current operating mode) is continually displayed in the top field of the CRT screen, and other information of the status of safety system elements is available to the operator on request.

One of the important tasks of the RGB-F system is to make the operator aware of all the elements which are not operable or for which the operability is limited in some way. This task is complicated by the fact that elements may consist of many components which from the standpoint of RGB-F are undefined entities. An inoperable element then may contain any number of inoperable componenents, and the restoration to operability of a particular component of an element may not indicate that the element itself is operable. A simple example is a pump which could have its inlet valve closed and its control unit disconnected. Obviously, neither opening the valve nor reconnecting the control unit will individually restore the pump to operability. In the RGB-F system, this situation is handled by requiring that an "inoperable entry" be made into system every time a component of an element becomes inoperable. Associated with each entry of this type is a unique "form number" (originating from a serial numbered set of report forms). The RGB-F analysis program will not declare an element to be operable until an operator entry has been made explicitly canceling each of the active inoperable component "form numbers" associated with that element.

4. EXTENT OF THE RGB-F EXPERIMENT

The current RGB-F experiment has deliberately been limited in scope so that the experimental system could be implemented quickly and feedback obtained from actual operating experiences at the Forsmark 1 Reactor. One way in which the scope has been limited is to restrict the analysis performed by the system to those aspects of the Technical Specifications which deal with the operability of the reactor safety system. A second limitation in scope is obtained by only including particular functions of the reactor safety system elements that are responsible for carrying out the following functions:

Incore Flux Measurement
Reactivity Control and Activation
Emergency Core Cooling (ECC) Hardware
Stand-by-Power Capability
Activation Sensors and Safety Chains for ECC Hardware

Even with these limitations in scope the RGB-F system includes 1700 functional safety system elements which are analyzed according to 63 separate Forsmark 1 Technical Specification operability rules.

The RGB-F system takes into consideration all faults and fault combinations which are within the range of the experiment. Faults are handled on a functional element basis: Any component of a functional element which breaks down that function is considered as rendering the functional element inoperable. Examples of functional elements are a measuring channel which would include as components the transducer, the transmitter, the supervising chain and other associated equipment. Only functional elements appear as explicit entries in the RGB-F tables. The number of elements to be identified is thus kept to a minimum, which speeds up the system response to operator entries, saves computer memory space and simplifies the administrative procedures.

In the experimental version of RGB-F the results of the analysis are presented to the system operator in the form of color-coded alphanumeric text. In subsequent versions of the system it is expected that this information will be presented in a combined alphanumeric-graphical format.

5. THE REACTOR OPERATOR INTERFACE OF THE RGB-F SYSTEM

The reactor operator interface of the RGB-F system consists of the I/O station described in Section 3 together with the associated procedures which the operator must be familiar with in order to use the system. A design basis requirement for interface has been that it can be used by any person familiar with the reactor plant without any further complicated training. Consequently, the interface has been kept as simple as possible with communication between the operator and the system based on dialogues which appear in a standard location (or field) on the CRT screen. This field always contains a question or an instruction for the next step to be carried out by the operator. This field also contains an image of the operator's alphanumeric input to the system as he generates it by keystrokes at the keyboard. If the input is incorrect, a clear text descriptive error message appears in the left corner of the dialogue field. For operators with some experience with the system, the message is usually the only direction that they need to correct the input. If further direction is needed, the operator can depress a "special function" key labeled "HELP" which will cause a list of all legal inputs to be displayed on the CRT screen.

All of the output information for the operator appears on the CRT screen in the form of different pictures which have the same basic structure: four horizontal fields. The uppermost field is used to continuously present the shortest repair time. The second field always contains the status of the reactor and the mode of the RGB-system. The third (or middle field) is reserved for the textual material that appears in the different pictures which the operator can request. The fourth (and lowest field) is the dialogue field. The pictures themselves are of five types:

1. The "overview picture" that lists the safety systems and indicates whether or not their status implies special test requirements or limitations on reactor operation.

2. The "input report" picture that is used whenever the operator wishes to change the operability status of a safety system element.

3. The "inoperable element list" picture, that lists all inoperable safety system elements, the form number on which they were reported, the time at which they were declared inoperable, and the operating time restrictions produced by the elements inoperability.

4. The "system detail" picture for a particular system that displays information about the inoperable elements within the system and describes in clear text the restrictions on reactor operations and extra tests which the condition of the system requires. The picture also displays a reference to the paragraph of the Technical Specifications that leads to these requirements.

5. The "help" picture which displays a list of all legal inputs from the operator I/O station to the RGB-F system.

The sequence of events which follows the discovery of an inoperable element in the plant is the following. Suppose that during a periodic test an inoperable element is found. If the fault in the element can effect the safety of the plant it is reported to the control room on a special serial-numbered form. The form number together with the element name is used as a unique fault identifier and is entered into the RGB-F system. The RGB-F system records this information together with the time of entry and can combine subsequent reports dealing with the same element with the original fault report. Other forms might refer to work permits, list of closed or opened valves, etc. The operator can localize all these forms by calling the original fault report. All of the forms are cancelled after the fault has been repaired and the element has been restored to an operable status.

The RGB-F I/O operator station is also designed to be used for planning repair or maintenance activities and for the training and instruction of the reactor operating staff. In this case the system is operated in the "test" mode and only the identification for the failed element needs to be entered into the system. The operator can freely enter into the system any fault or fault combination. As output he gets information on the corresponding Technical Specification restrictions and requirements. This mode of operation is expected to be particularly useful during the preparation for annual revision (or refueling outage) when there are numerous entries into the RGB-F system. All of these "faults" can be planned with the help of the RGB-F system operating in the "test" mode. When the planning is ready and the consequences of closing valves, stopping pumps, etc., have been checked, all information can be transferred from test-mode over to real-mode by depressing a single special function key. Since several hundred (or more) entries may be involved, a great deal of potential duplication is avoided.

6. IN-PLANT OPERATING EXPERIENCE

The RGB-F system was placed in operation at Forsmark 1 at the end of 1980 and operated by plant personnel on an experimental basis for six months. At the end of this period an evaluation of the experiment was made by interviewing the system's users and examining its operating history during the experiment. Users reacted positively to the RGB-F concept, but felt that in a full-scale system there should be close coordination with the administrative forms used for plant operations. The operating history showed that the principal use of the system was to plain maintenance and test activities using the Test Mode. These results will be used to guide the design of a full-sclae Automated Status Analysis System for the Forsmark station.

7. CONCLUSIONS

The newly developed and implemented RGB-F system is expected to become an important tool in assisting the operators of the FORSMARK 1 plant in reliably and routinely assuring that the plant is operated in conformity with the Technical Specifications. The system is not an attempt to replace human analysis, but rather is designed to increase the assurance that human analysis and decisions are done correctly and reliably. The RGB-F system will contribute to this goal by presenting to the operator the Technical Specification implications of any arbitrary state of the systems included in the RGB-F experiment within one second after that state has been defined at the operator I/O station. Achieving the same result using normal control room procedures can require extensive examination and comparison of the Technical Specifications, plant drawings and system descriptions. In the RGB-F system all the foregoing sources have been "pre-analyzed" so that the RGB-F system is able to present immediately to the operator just the essential information that he will need to make a decision.

The traditional use for computers in nuclear power plants if for process monitering, event recording, and so forth. Computers are not generally formally accepted as being responsible for the activation of the protective action functions of the reactor safety system (5). The RGB-F system is a computerized aid for the reactor operating staff that enables them to supervise the status of the reactor safety system with respect to the Technical Specifications by analyzing the consequences of known faults and combinations of known faults. Thus, the RGB-F system is not directly associated with the reactor safety system, but it serves to increase the reliability with which the safety system is operated and is a step toward the more general application of programmable computers in the nuclear power plants. In the post-TMI era it seems clear that there will be an increasing need for supporting systems for reactor operators, and the successful implementation of the RGB-F system is expected to lead to the development of a full scale system for automated system status analysis at the Forsmark station.

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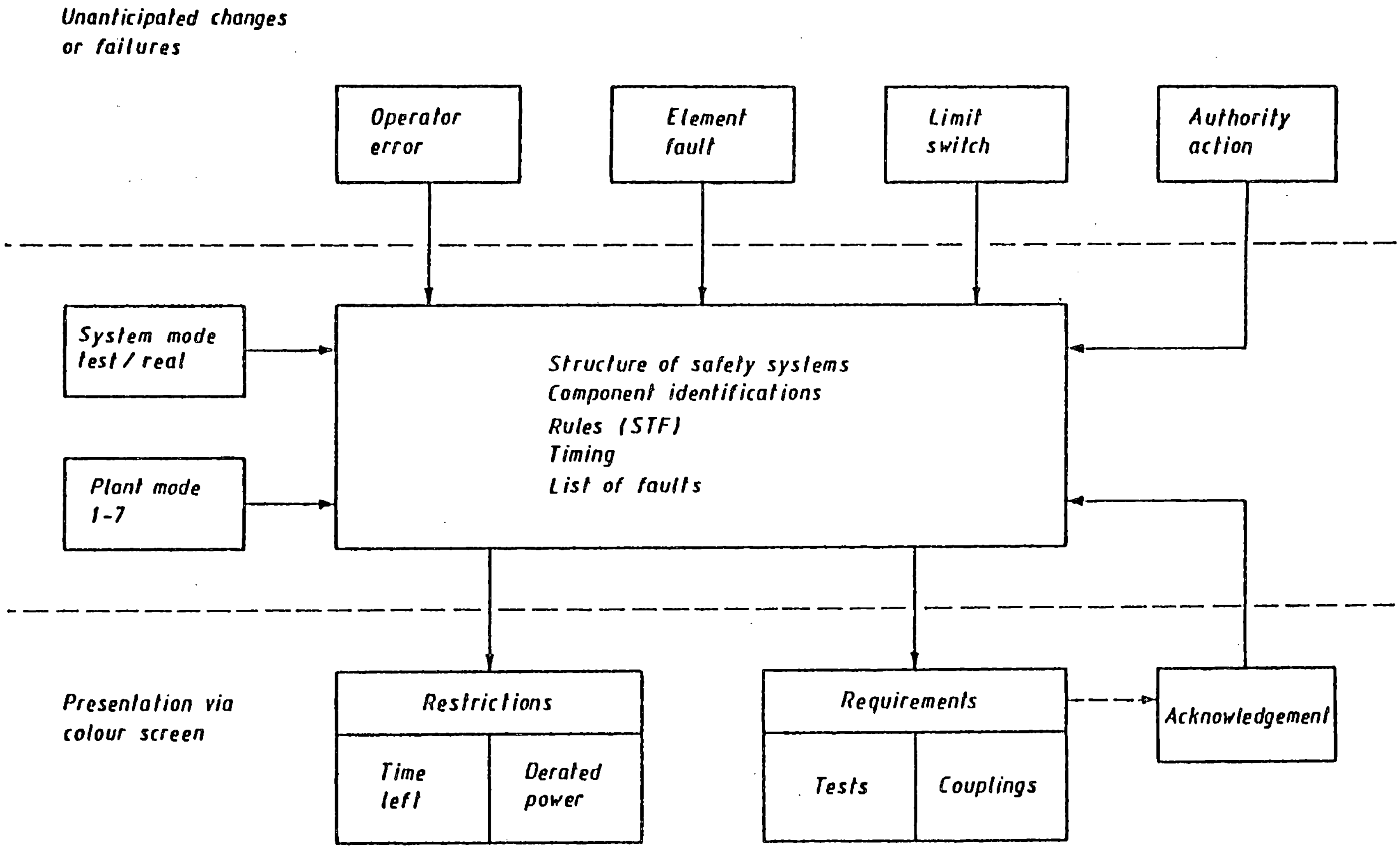

Figure 1. General Structure of the Automated System Status Analysis System

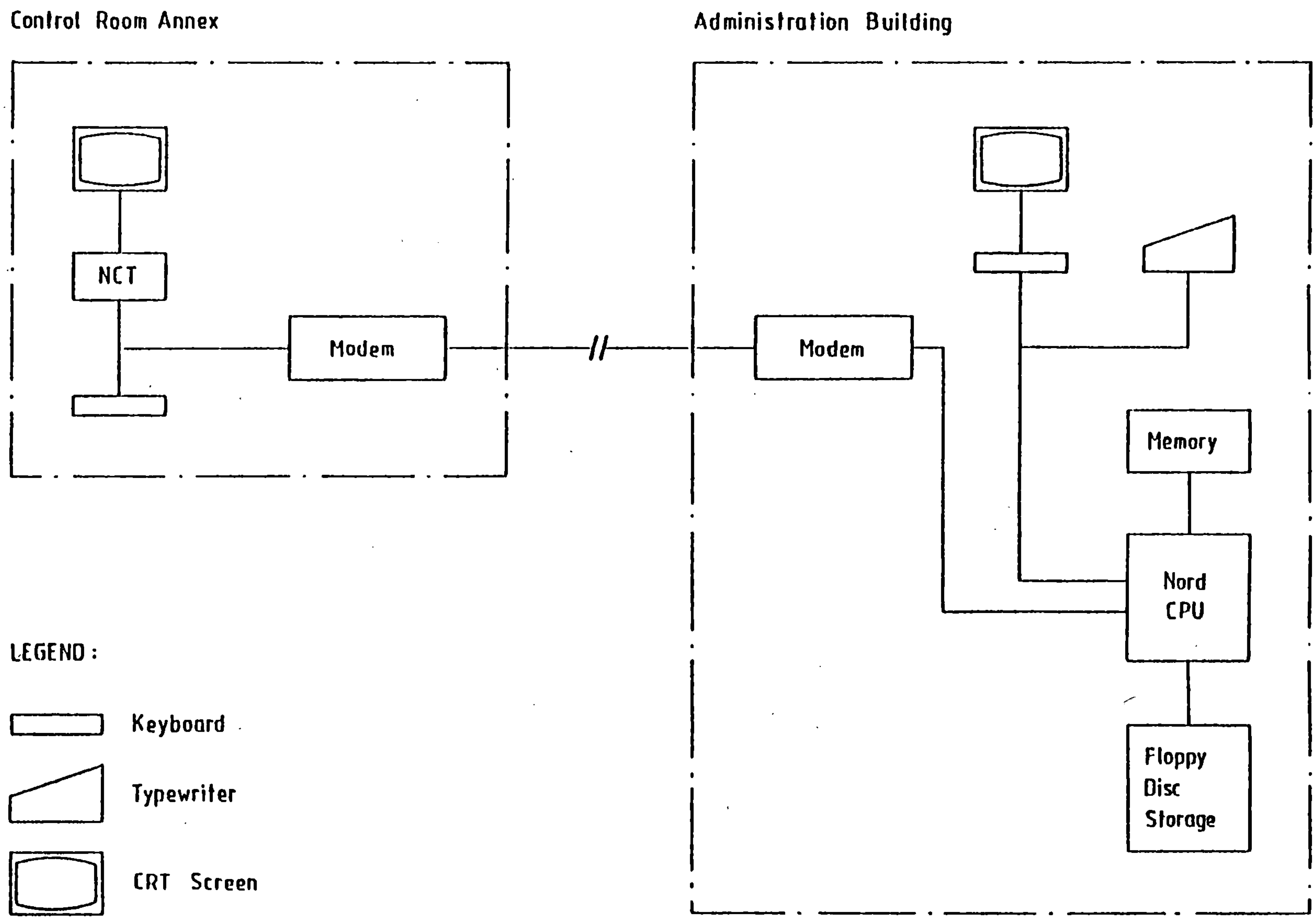

Figure 2. Hardware Configuration of the RGB-f Experiment

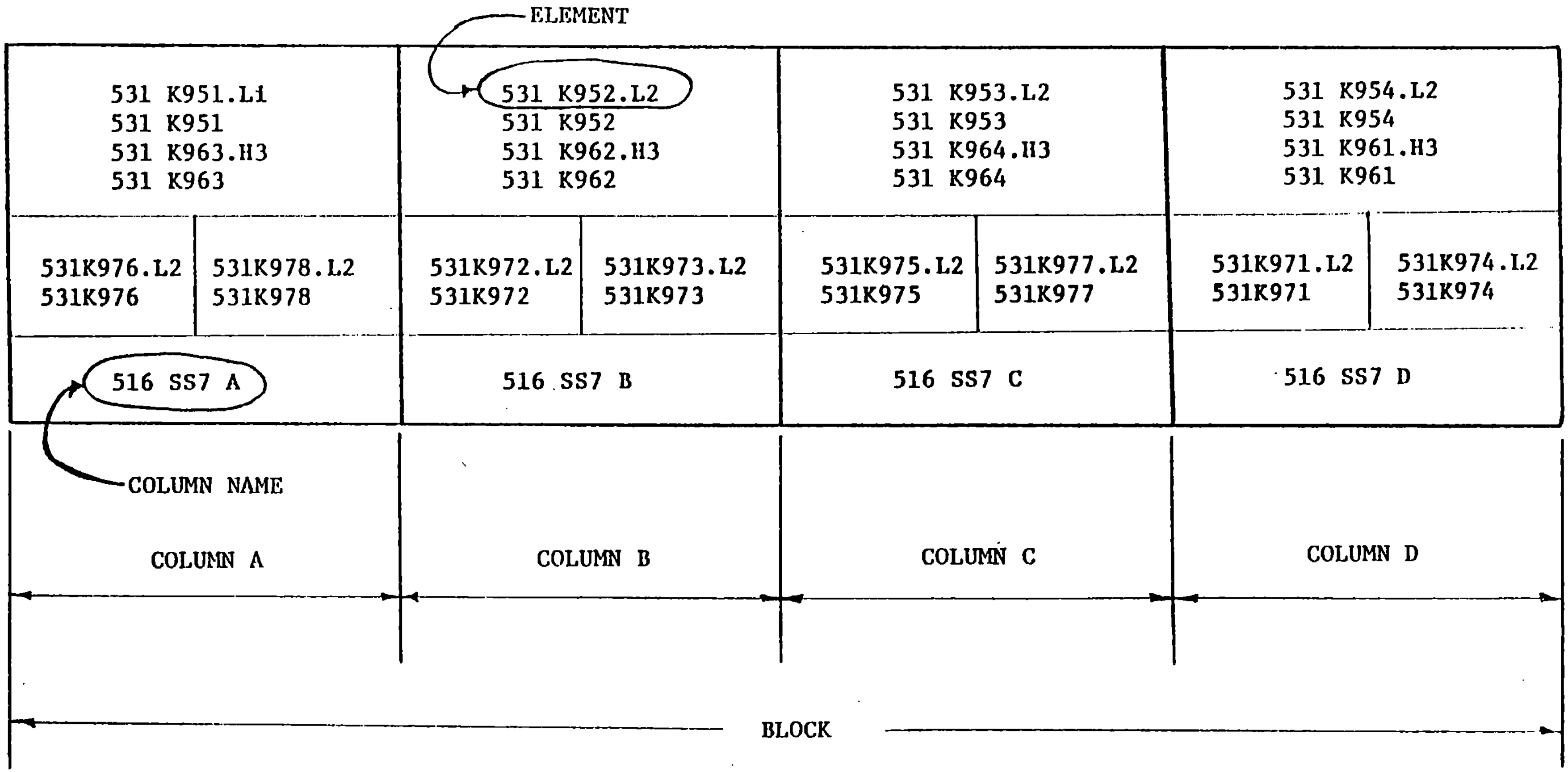

Figure 3. Basic Structure of RGB-f Tables

CSNI SPECIALIST MEETING ON

OPERATOR TRAINING AND QUALIFICATION

Charlotte, North Carolina, United States

12-15 October 1981

REMARKS for the PANEL

"THE FUTURE: MAN'S ROLE in the NUCLEAR POWER PLANT"

by

P. Courvoisier*

ABSTRACT

The author takes a look at the meaning of some frequently used terms, such as: "training," "qualification," who is "man," and what is his "role" in a nuclear power plant. He presents his views on these terms and proposes explanations for a number of further terms. The author sees that "qualification" is used in two ways and proposes that the term "selection" be used exclusively to refer to the process of finding (good) candidates for training and education, and the term "licensing" to their final admission to work in the envisaged post. In the view of the author, "training" stands between drill and education. Drill is good and even necessary for more reflex-like actions that are to be taken in operating situations which have been previously and

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fully analyzed and which are hence amenable to this very specific form of training. Education is necessary to enable mainly the upper levels of operating crews to handle events in a nuclear power plant which go beyond pre-envisaged scenarios.

At least the latter events call for operators with fully developed personalities, who dispose of a high degree of self-confidence without presumptuousness, who are very stable and who are ready to take responsibility for deciding the actions needed to maintain control of the system.

The author attaches great weight to this meaning of the notion of "man," and his concern is that candidates with the corresponding qualities be selected for education and training, as only they will be able to perform successfully in the most difficult part of their "role," i.e., during accidents. The author describes the three different parts of the role operators have to play in nuclear power plants as an approach to obtaining the definition of it needed to guide selection and training.

SCOPE OF THIS PAPER

The title for this panel contains three words which need to be considered in somewhat closer detail: (a) what is the future, (b) who is "man" in this context, and (c) what is his role when he is to operate a nuclear power plant (NPP)? In addition, the title of this specialist meeting is "Operator Training and Qualification." Hence, there are two further words which need explanation: (d) what is meant by training, and (e) what does qualification mean?

I will try to express my views on all of these words (or rather, notions) in a general, but hopefully not superficial way.

There is, however, the additional "word" NPP. It need not be considered here for the benefit of the members of this specialist meeting, but I think that whatever has to be said for man's role in the NPP is valid to a large degree for other nuclear installations as well.

I think that in all minds and without further discussion, the word "operator" stands for all jobholders who form part of the operating personnel of an NPP, including those at all levels upwards and downwards from the operator at the control desk, who are considered to be relevant to the safe operation of the plant.

This limitation itself is a problem, as it presents questions regarding maintenance, quality assurance, quality control, and radiation protection personnel, and is certainly not a consideration of minor importance.

THE "FUTURE" IN GENERAL

The future is at least the increment starting at the present, and it is generally and in all probability only a continuation of the last trends and developments. What was the past, and what trends are visible as regards man's role in the NPP? As specialists, you know this past and the present tendencies well; you have updated your knowledge by attending this meeting. There is no doubt that TMI has brought about a break with earlier tendencies here or there, and there are new (or at least newly formulated) tendencies visible. Hence, the future will not--and should not--be simply an extrapolation of past trends.

The present trend and the present tendencies, as reflected in the recent literature (including the abstracts for this specialist meeting) present abundant imperatives for improving the human part of the integrated man-machine system, which is an NPP. The key words which come up over and over again are: man-machine interface, human factors and human factors engineering, ergonomics (which is synonymous with the former), job analysis, training, psychological fitness, skills, aptitude, capability, competence (the latter three having very nearly the same meaning, as I see it).

Is this enough? Key words do not constitute a programme for the future. To keep within the theme of the present meeting, do the most prominent and most often repeated key words "training" and "qualification" of those who will have to run an NPP cover all that needs to be done?

Logically, yes--what else can one do than train the group of people who will work as operators of any sort of machinery (NPPs are nothing other than that) and train them to a sufficient degree that they meet certain qualifications considered necessary for their work?

But this logically correct approach implies that it is crystal clear how and for what the training has to be undergone, and what the goals are that must be satisfied. One of my main aims in this paper is to look at such questions and to speak out on them. I will return to these points mainly under the heading "role of operators," which to me is of greatest interest.

TRAINING

I do not want to discuss methods of training. They cover most of the existing literature, and they have been the subject of the greatest part of the papers given at this meeting. However, I am still uncertain what the authors of these many papers consider to be the extent to which the candidate's full range of mental capabilities should be addressed in the proposed training and whether the authors do ask this fundamental question at all. Is the proposed or described training meant to fill a candidate's intellect only or is it geared to entrain more of his personality? How do the various authors see the difference between "training" and "education"? Do they feel that such a difference should be made in the instruction for the different levels of jobs in an operating crew?

Qualification can also mean, at each step in a career, that a person has been selected and trained--or even educated--sufficiently well to fill the post under consideration and is hence qualified to actually take it over will all rights, duties, and responsibilities that are connected with it. This second use of the word "qualification" of persons is synonymous to "licensing" them for their posts, which more clearly implies that a formal procedure has to be carried through.

I propose to use exclusively the term "licensing" for this second step in order to eliminate the ambiguity of the word "qualification."*

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*An explanation of terms, somewhat more detailed than a simple glossary, would be of great help. To be defined are not only "selection" and "licensing," but also terms like "ability," "aptitude," "capability," "capacity," "competence," "gift," "talent," "proficiency," "skill," "education," "training," "drill," etc.--all of them explained in their "technical" use for nuclear energy production. A standard dictionary is just not precise enough to prevent misunderstanding. Technical people should not forget that these terms are non-technical in nature and cannot be used by them as they please.

"MAN" IS A FULL HUMAN PERSON

Let me say that it is my firm belief that the question of selection of a candidate by looking at his qualities as a human person is the most important one in the whole domain of man's role in the operation of NPPs, and that this question has to be continuously asked throughout every operator's career. A final answer to this question is not given once for all by just looking at the level of general education a candidate has received and the examinations he has passed so far (whether the answer depends on the sex of the candidate is beginning to be debated). I am convinced that the full answer can be given only by looking at the qualities of a candidate as a human person.

It may well be that the answer to this question of qualities of an operator is positive in the first phases of his career, then later in his working life becomes negative, leading to termination of his career advancement, and even to his dismissal from this type of work.

I say this in view of the fact that every man's personality develops throughout his life and may take directions one cannot know before they have manifested themselves. I say this also because of my strong conviction that an operator in an NPP must not be looked at as just a sort of robot, who by training, or better by drill, has been programmed to perform certain well-circumscribed functions. Quite to the contrary, he has to be looked at as a full human person with all the qualities, good or bad, which the psychologists can enumerate for us (viz., so far as they do know them already). There is no doubt that a part of man's personality is analogous to programmed robot behavior and that this part is of great importance for the operation of NPPs (certainly for so-called normal operation, but also and even more importantly so for many functions during all sorts of disturbances); I will come back to this question later on. But there is no doubt as well that, the more the situation of an NPP gets off-normal, the greater is the challenge to the personal qualities of its operators. The result toward which such off-normal situations will lead can be influenced in a decisive way by personal qualities of the operators, which are outside and beyond the domain of previous training of any kind, be it general or specific, and--this must be clearly admitted and recognized--also beyond the capabilities of selection techniques and licensing procedures of operators as they have developed to date.

After what I have said, it is clear that I attach great weight to the word "man" in the context of his role as operator of NPPs, and that I see here the importance of considering the full personality of every single operator (and not only psychological types that one might define for the different posts in an operating crew).

THE ROLE OF MAN

The next word in the title of this panel is "role." The word comes from the theatre world and means the part an actor has to play in a drama, the part being written on a rolled piece of paper. No doubt actors have to be selected according to the demands of these parts and must be instructed on all details by the producer. Otherwise, the performance of the drama will become a failure. Actors know that there are "unplayable" roles. Let us assume that it is certain that the dramas which may have to be played in NPPs--down to the last act--do not present such roles. Even so, like in the theatre world, one cannot be certain of this unless at least fairly all-inclusive final rehearsals have been carried out. Whoever rehearses "nuclear dramas," at a simulator e.g., should keep that point in mind.

To come back specifically to the nuclear world: the first thing to be done is to carefully select the candidates for the different levels of work in an operating crew. Once this is completed and all parties concerned have agreed that a person is qualified to become a candidate, training starts. Nevertheless, both selection and training have to be made in view of the question: selection and training to perform what role in the NPP? Selection and training, and finally licensing of operators cannot be separated from the role they will have to play. Thus, I would add "selection" to the very true, if short, finding in the abstract of the paper III-4 by G. M. Grant: "Training needs cannot be defined until the role of the recipient is thoroughly understood."

As you well know, selection and training of operators have long been problems for those who have had to do them. These tasks have become a problem in the eyes of the public mainly since TMI. But do all concerned--selectors, teachers, licensing authorities, and the public--know clearly, or at least sufficiently well, what situations operators will or may be confronted with and which they are expected to handle without hesitation and with full (or at least near) perfection?

The Call-for-Papers document for this specialist meeting invites discussion of "capabilities of operating staff in NPP control rooms and the problems they face there." Expressed in a somewhat longer fashion this means: "What does an operator have to bring into the control room of an NPP when he is to handle whatever may happen there?" This formulation is well and good, but it does not say what degree of success he is supposed to achieve by his actions and to what degree his actions have to be correct and perfect. It does not specify whether operators have to play their roles only within the framework of pre-envisaged scenarios (this being the situation of actors in a drama known from beginning to end before the curtain opens), or whether they have to perform perfectly--or at least reasonably well--even in situations which are outside, perhaps far outside, previous consideration.

When one sifts the literature, including the abstracts for the present specialist meeting, with the question of the role-to-be-played in mind, one is left with practically empty hands and especially so when one asks what the goals to be reached should be. There are only a few ideas expressed, however vaguely. I cite (without reference for brevity) the following ones:

- deal adequately with all situations and states of the plant
- cope effectively with emergencies
- respond flexibly to situations deviating from pre-analyzed ones
- take the best course of action in unforeseen situations

There is nothing wrong with these formulations of the goals to be reached--except that they are just too general to be of use to those who have to select the candidates, those who have to train them, and finally those who have to determine that a candidate is qualified for his job. R. C. Sugarman and R. R. Mackie (paper IV-3) express it thusly: "A central issue (is the) lack of standards by which the performance of NPP personnel can be judged." However, their point refers to human factors only, i.e., ergonomics, which is a more restricted field than what I have in mind.

AN APPROACH TO DEFINING THE "ROLE OF MAN IN AN NPP"

It is trivial to say that no clear, comprehensive, and correct answer can be given to a question which is vaguely formulated.

So let me try to present at least an approach to a clear formulation of the question of what man's role in an NPP is once and for all.

I see this role as being threefold:*

1. Firstly, there is what is called normal operation. Here the operators, mainly the operators in the control room, have as their minimum duty to survey the plant as it runs on fully automatic control, riding out minor disturbances, e.g., from the main grid. This activity is an actual direct watch by the operators in the control room. It is a checking sort of activity for operators at lower levels, who periodically or on special order verify the correct performance of the different systems or components throughout the plant. It is a surveillance activity by the leader of an operating crew, frequently called supervisor, whose duty during normal operation is to see that the whole field of operator activities is well and evenly covered.

2. But designers have not been able to make everything in an NPP automatic, nor build plants which can ride out whatever

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*I cite partly from the paper by P. Courvoisier, K. B. Stadie, and M. E. Stephens: "Qualification and Training of Nuclear Power Plant Operating Staff in the NEA Member Countries," IAEA International Conference on Current Nuclear Power Plant Safety Issues, Stockholm, October 1980.

disturbances might happen. If one or another of the systems of an NPP gets outside of its domain of automatic action, the operators will have to try to get it back to this domain on being summoned by appropriate alarms. In such cases, they become a prolonged arm of the automatic control system, using prescribed procedures which they have been trained to apply. But this form of training (better called "drill") logically cannot cover more than those abnormal situations which the designers have been able to foresee (up to and including the design basis accidents) and which have accordingly been included in the specific training of the operators. Reacting correctly to foreseen abnormal situations which have not been covered by any automatic action is a second and clearly very important aspect of the role of the operators of an NPP.

"Procedures" have been prepared for such foreseen abnormal situations--procedures which have been carefully worked out (and hopefully well written) by knowledgeable people, taking into account certainly more aspects than an operating shift would be able to think of under the pressure of time of an incident. These procedures should, therefore, be used to the extent that they are applicable and their imperatives "do . . ." or "do not . . ." should be followed. Not using them would be a very considerable human failure on the part of the operators.

Hence, in such cases the main role of the operators is to determine that they do have a situation which is covered by a procedure and then to use it--<u>but</u> always with open eyes and alert minds to all possible deviations from the scenario for which the procedure was written, a scenario with which they must be fully familiar. For this somewhat restricted, but still clearly definable role, the operators will have to be trained.

3. There remain the--hopefully rare--abnormal situations outside the domain of the foreseen ones and outside the catalogue of pre-analyzed scenarios for which safety systems have been provided and for which extensive training has been given. They have become a matter of great concern. If one of them should happen, it would call for a typical human capacity: man's capability to combine information before him by means of his intelligence and to draw conclusions which guide his actions. This specific human faculty has to come to bearing more and more as an abnormal situation in an NPP differs more and more from the normal or pre-envisaged situations and becomes more and more what is finally called an accident.

Clearly, pure drill and adherence to prepared or even prescribed procedures will no longer suffice under such conditions. The operators on shift, including whatever informed helpers will come in within useful time, will then have to show how well their education, their training, and their experience have led them to understand the characteristics of their plant and to know how they can make use of the different systems which the designers have put into their hands as tools to keep the plant under control.

The Kemeny Commission, in their report on TMI, expressed the view that the accident had gone too far to be tolerable, as it had put the operators at times into the position of experimenters. However this specific event should be regarded, I feel that wherever a situation in an NPP has developed which is outside the trained repertoire of pre-envisaged scenarios, the operators are inevitably in the position of experimenters, who have to analyze the situation by themselves with only partial backup by procedures, but still with the backup of pertinent things they have learned during their education and training. They have to devise the way to handle the situation on the spot, correcting it, if need be, as it develops further.

An argument that NPPs must be forbidden because their operators could get into the position of experimenters needs discussion.

I would like to submit the following view on it: Usually an experimenter in the laboratory expects his experiment will confirm results which he knows fairly well beforehand, with good reason in view of all he knows already about his domain of science and his previous experimentation. It is very rare indeed that an experimenter makes a step into a fully unknown field of possible events, be that on purpose or--more important to our comparison here--unknowingly and by pure chance. As regards NPPs, there have not been any events--including TMI--which have shown that NPPs are operated close to fully unknown fields of possible accidents into which one might get suddenly and which might mean disaster. So the label of "possible experimenter" for the operators of NPPs describes clearly and succinctly the third aspect of the role these operators may have to play (but there is no reason why this label should be transformed into the label "potential hazardeur").

However, just as inadequately equipped experimenters will not be able to come up with good experimental results, it is the duty of the designers of NPPs to design plants with basic characteristics which can be understood by so-called normal people, not only by supermen. They must also provide sufficient and adequate systems as the operators' tools. The operators may use them in "unorthodox" ways if a situation calls for this and lends itself to such use, but they cannot be expected to invent and to build new tools within a useful time, at least during the dramatic phases of an emergency.

Tools are adequate only if they fit well into the "hands" of the operators, i.e., only if ergonomics and like considerations as well as their intellectual capacity are taken into account. Adequate tools leave the operators sufficient time to select them as needed and to prepare them for use. A thorough drill in the use of the tools will be necessary in any case, as this use can certainly not be learned under the stress of the situation.

This third part of the role of the operators in an NPP, even if in all probability it will never have to be played, needs adequate general education, mainly for the upper levels of an operating crew, beyond the specialized training needed for the second part of their role. This third part would have to be played spontaneously and with only the general guidance: "Take what course of action you see to be the best, and always watch whether you are still on track!" This clearly implies transferring responsibility from the writers of procedures for pre-envisaged scenarios to the operators and to their teachers. Taking over this responsibility needs a full man's personality, a high degree of self-confidence without presumptuousness and much stability--and here one comes back to the question of selection of operators.

A FINAL REMARK

The nuclear community has long studied the safety of NPPs intensively as a technical challenge without having an overall, clear-cut, operationally usable definition of what "safety" is. Nevertheless, the operation of NPPs has been remarkably safe so far. By analogy, I see no reason for there being any call to stop the operation of all NPPs until all is known about the roles men have to play in their operation, even if we have to admit that at present we know much more about the hardware in the NPPs, including the behavior of components under stress, than we know about the "life-ware," i.e., the human beings, working for and in them and during dramatic situations under stress as well. In other words, we know much more about technology than about psychology. If I say "we," I mean at least the technical people, who are active in the nuclear field--and we have to make up for this deficit by all means.

APPENDIX

to the paper by P. Courvoisier

The following diagramme describes a model of the man-machine relation in order to illustrate what is said in the paper. This model is admittedly naive; everyone is invited to improve it as best he can.

To the right is indicated the domain of all the hardware of the NPP. Next to it is the domain of the "controls," comprising all the electric, electronic, and electromechanical devices used to make the hardware of the plant systems run correctly, along with the sensors installed to get information on their states. The flows of information from and actions on the hardware are indicated by arrows.

The latter domain has a visible surface, i.e., the panels for the instruments, indicators, and displays of all sorts as well as for switches, buttons, etc. This surface serves the operators in front of it as the medium by which they perceive the state of the plant and can physically act on it. This surface, together with its extension representing direct contacts between operator and hardware (manual valves, e.g.), is the man-machine interface. Close to it is the domain of ergonomics, frequently called "human factors." Ergonomically well-designed interfaces promote a clear and rapid perception of the situation an NPP is in and ease timely and error-free actions to alter this situation in a desired way.

Such actions should be reflex, and must hence be programmed, i.e., they must be learned through specialized drill. Actions of this type are represented in the diagramme by reflected arrows.

Other actions need mental preparation before they are taken. I see two broad groups of them:

- One group of actions is intended to cope with pre-envisaged events, up to and including design-basis accidents. They follow patterns, which the operators identify from their perception of certain key parameters indicating that one of the pre-envisaged events is going on. The operators remember the scenario for it and act, guided and aided by written procedures, according to the imperatives "do. . ." or "do not . . ." which these procedures contain. The operators have to select the procedures appropriate to the scenario that they have determined to be occurring. They should not act by relying on their memory alone throughout the whole course of the event.

  Specific training will suffice to lead to successful coping with such events.

- The other group of actions would be in response to unexpected events which present elements beyond the pre-envisaged ones of the first group. In such cases, the operators will have to act in a creative way, using the systems they have at hand as best they can imagine in order to bring the event under control. Procedures prepared for the above group of actions will aid, but only partially.

Education so that the operator has a full understanding of the characteristics of the NPP will be needed in order for him to diagnose the state of the NPP correctly and to cope successfully with such (rare) situations.

CSNI Specialist Meeting (CHARLOTTE)

10/12 - 10/15/81

Panel Discussion

(This record was transcribed from a tape recording and has been edited slightly for clarity.)

Chairman: K. B. Stadie

October 14, 1981

PANEL DISCUSSION

Ladies and gentlemen, I first applaud your endurance. I am impressed that you are all prepared to stay on after such a long meeting to hear what we have to say on "Man's Role in a Nuclear Power Plant." To begin with, I should perhaps clarify what we, the panel, mean by the term "operator." As a matter of fact, we use the term "operator" in a rather wide sense. It covers everybody who is in the control room or can be mobilized at a given time to help in a difficult situation. So to speak, we therefore include all the ROs, the SROs, the STAs, the PEs, and whatever else there may be. Before starting, I should very briefly introduce you to the panel. Starting from my left, there is first of all Warren Witzig, Penn State (USA), followed by Walt Gronow of NII (UK), Mr. Fechner of BMI (FRG), and going on to my right, Madame Carnino of CEA (France), Mr. Alonso of Junta de Energia Nuclear (Spain), and to the far right, Bob Smith of INPO (USA).

Immediately to my left is my colleague, Mike Stephens, and my name is Stadie. We both represent the Nuclear Energy Agency of the OECD. You will notice that there is evidently a preponderance of panel members from regulatory and other national safety authorities. This is in line with CSNI, which as I told you during my opening remarks, brings together national safety and regulatory authorities.

In opening this panel, I should perhaps briefly tell you how this meeting came about. The idea for this meeting was put forward 2-1/2 years ago when CSNI held a special meeting after TMI--if I remember correctly, at the beginning of June 1979. It was at that time that Harold Denton and Saul Levine faced for the first time their opposite numbers in other countries. Of course, many interesting insights on the accident were exchanged then and since, within the framework of CSNI, but what sticks in my mind until today is what Dr. Courvoisier observed during the meeting. He had noted that during the latter stage of the accident, more nuclear professionals had aggregated in the TMI control room than there exist in all of Switzerland. Therefore, there were more experts in the room than he could possibly get to help him in managing such an accident.

I mention Dr. Courvoisier here because we had asked him to prepare a thought-provoking paper, which was distributed yesterday, and which I hope you have all read. We hope that his ideas will contribute to our rather lively discussion, which we unfortunately have to conclude before 5 p.m.

I would like to begin by asking a very specific question, which I am very keen to ask myself. A few years ago, Alvin Weinberg put forward the notion that we would be obliged to create a new technical elite, or as he called it, a priesthood, to build and in particular, to operate nuclear power plants. My specific question would be, "To what extent do you think this notion holds true today, with regard to nuclear power plant operators?" Before you make your contributions, I should just like to inform you that the panel is being taped so that we may add it to the proceedings of the meeting.

Thank you. We will now turn to the other side of the ocean. Mr. Alonso has asked to add something here.

Mr. Alonso

Well, when Alvin Weinberg mentioned priesthood, he not only referred to the operation of power plants, but if I recall correctly, he was also referring to waste management. But anyway, as the subject of this meeting is training and operation, we have to take into account two particularities, two main characteristics, of priesthood in really any religion. The first one is the longevity of the institutions, and the second one is the dedication and the fidelity of the priests themselves. I think that we all have to recognize that monasteries are run very well, and that they can survive for centuries the many, many wars and the many, many revolutions that have destroyed other human organizations, including monarchies, democracies, and even, they have changed the frontiers of the countries, and some countries have disappeared. And still, monasteries have gone on living for hundreds of years. Well, I am not advocating with these ideas that nuclear power plants should be run by priests. But, really when you visit some of these plants, and I can refer to two specific ones in Spain, the Zorita and Garona plants, which are rather isolated, then you start to realize that the operating crews are close to priests. They are living in a close community; they dedicate their lives to the operation of this particular plant; and they do really behave as priests, in many ways. Well, if this is the case, and if this is the experience they got from these two power plants in Spain, I believe that there is certain sense in Mr. Weinberg's position, and I am going to propose that the utilities, when they go out recruiting people, one of the questions they may ask to the possible candidate is, would you mind to become a priest and live and work in a monastery? If the answer of the candidate is yes, I believe you have good chances of having a good operator.

Well, I would just like to reiterate the position which existed, exists in Spain to some extent, that in the very first days of nuclear power, it was necessary to select, if you like, the elite that the country could produce. But when one has a large number of plants, when the world perhaps is going to depend upon nuclear energy, the operator has to be a part of that system and therefore has to be seen to be a normal person by the public, if they are to believe that these nuclear plants are safe. And therefore, to put in elite personnel suggests that they are outside the control of normal people. I think that is a wrong philosophy. So, I would say that Weinberg's comment was not necessarily related to this particular aspect of the problem--he referred, of course, to a much wider perspective of design, as well as operation. I think we have to be careful that we make sure that operators are not elite, and in this conference, the way in which I have heard the developments of operator training, education, qualifications, leads me to believe that there are some difficulties which exist with the design, which we ought to put right.

Mr. Stadie

Thank you. Is there anybody else on the panel who would like to contribute?

Unidentified Speaker

Just to reinforce here, as we all know that when we started this business, we had to develop an aura of eliteness to get people to join the program, because we weren't paying them any more at that time. Now, we pay them more, at least in our country we do. We pay our operators a little more, so therefore that eliteness is not quite as required, but on the other hand, I might point out that to get somebody to agree to be a nuclear power plant operator, knowing full well the very difficult road it is to get qualified in the first place, the very difficult road it is to stay qualified in the second place, and working shift work for innumerable years in the third place, you have to have something beside dollars to intrigue these, or motivate these people. So, I am suggesting that we don't want to seem to make too common, or they won't want to join this elite organization.

Mr. Stadie

Thank you. I should now turn to the specific questions raised by Dr. Courvoisier. As he sees it, the operator has three main functions:

i) He observes the normal operation, which is probably a routine job.

ii) He has to react correctly to a number of abnormal situations, which the designers and the power company have foreseen based on their experience. For these situations, procedures are developed on how to handle them, and the operator has been drilled for these eventualities.

iii) Even before TMI and more so since, it was recognized that situations may develop which may fall outside the catalogue of pre-analyzed scenarios. How do you think operators should be prepared for such a situation, and even more important, what confidence have you that they will react properly?

Madame Carnino

I think that the problem of what is in the scenarios, which have been foreseen and predicted, is something very important. I think we can rely on and connect these to the philosophy of the safety. We have designed the safety of the plants taking into account the design-basis accidents, and then we have made scenarios from this. In this case, we don't always take into account all the failures of the system or the components, and if we look at the scenario that really happens in accidents or near accidents, then we can see that you have a lot of occurrences happening at the same time in a given short time. I certainly think we have to improve our knowledge about these scenarios and to know more and to train people, but I wonder if exercising people on these scenarios, which shouldn't happen very often, and if having them train repeatedly too much on this would not lead the operators to prediagnosis--normal human tendency is to try to fit what he sees, what we see, to some given information that we know about. And so I think it is more a tool for creating a reflective approach by the operators and more, giving them what we call a critical mind, which is something that is very important, and that's the way I would think personally that these scenarios should be used.

Mr. Stadie

Thank you. Are there any different views to be put forward?

Unidentified Speaker

I wouldn't dissent from the view. But I think it's very important to realize the least reliable component in our system is probably the operator. Perhaps that is a sweeping statement, but should we rely upon an operator as a safety barrier? I know that question has been raised in this conference, should not the operator be a redundant safety value, and therefore whatever training we can give him is there to assist in, if you like, cleanup actions where the system has gone beyond its design parameters and not one in which he would prevent the system from shutting down the plant or trying to deal with it in an adequate way. Therefore, the need to train them in these scenarios is not for the purposes of preventing the incident from occurring, but it is for cleanup; it is for taking appropriate action, and, therefore, we should ensure that really the operator is not stressed in the immediate events, and I believe in that case this hands-off principle is one which should be pursued in the design.

I would just like to add a small comment on that because it seems, at least to me, that the statements I made with respect to safety barrier in connection to the operator was slightly misunderstood. It was meant, and if one looks into the paper I get that should be quite clear from what is written there, it was meant exactly like this clean-up function. Because I am certainly convinced that in the first place, we have the design and even the next and the third barrier should be the design; even for the worst case one could imagine, even for a core melt, one could probably do a lot more. The operator should be looked at as just some kind of backup function in case the event does not take the preplanned course, which anyway has to be laid down or has to be the basis for the design one does, and for the first case where it deviates from that and even the barriers won't do what one has expected, then the operator should be capable of taking the appropriate action. Taking them without being stressed.

Mr. Stadie

There is apparently a consensus on this.

Let me just add one comment. I think man is a master of the machine. Now, Dr. Courvoisier starts out and he gives us three things. He says, 1) normal conditions--it would seem that we could mostly agree that under normal conditions the plant should run on automatic control. Man watches, man observes, man monitors, but the plant runs on automatic control.

This is in the future we're talking about--we're not talking about today. It doesn't even do that today.

The second is the foreseen abnormality. Here, if it's foreseen, it would appear to us that automatic control again can remain dominant and the one in charge. However, the scrutiny of man obviously must be increased, because if it should deviate from that pattern then move to the third, which is the unforeseen condition, man is in charge. We have got to run it. So, I think the only place we can get into debate with that system is where does one taper off, and where does the other start?

Adm. Smith

I guess I have been operating nuclear power plants since 1956, and what I am hearing today is very interesting, but it never entered into the picture before and not that it shouldn't have, but, of course, most of my experiences are in the Navy, and the Navy has been operating plants since 1955 and has never experienced a serious abnormality of any kind. The way they train their operators is not to go through a bunch of scenarios of what might happen and what could happen and all these far-out situations. The Navy system is to make sure that their operators are thoroughly, and I mean thoroughly, qualified on every single system and watch station in their plant. And I mean every teeny, tiny valve--every air line, every gauge line, so that when something happens, and it's a system abnormality, the operator standing in the control room has an immediate mental image of what needs to be done and, in fact, can personally direct the operators to take, what he feels, is appropriate action because he knows that plant. Now I have seen a lot of plants in the last few years, commercial plants, and I'll guarantee you, at least in this country, our operators do not know their systems anywhere as well as the Navy people do, and if they were to take a Navy qualification examination, they would not be operators. Now that doesn't apply to every single plant, of course. I am sure that some are better than others. But I will guarantee you that if you walk around any of the plants that I have been in and talk to the operators, you will soon find out that there is a definite lack of understanding of their systems and no manner of abnormality training is going to overcome a lack of system knowledge.

Unidentified Speaker

I would like to add one further point to that. I would like to utter some type of disagreement with this point--that nearly everything up to those events which are completely unplanned should be automated. And this would more or less meet with what Mr. Smith said. The operator is, or has been driven in the past, to a situation whereby extensive automation, his in-depth knowledge of what is going on in the plant with respect to single systems, has more or less been lost, and if we continue going on that direction automating every single function, this in-depth knowledge, this feeling for the plant, at least to my feeling, will be lost. And I am understanding the initiative in this NUREG 0-700, which has been presented by Mr. Stadie this afternoon exactly in this context. It is not meant as an initiative to even go further with automation; however, it is meant as, say, an approach to step back, to analyze things from the beginning, and to ask oneself whether it is really necessary to automate everything and wouldn't it be better to have things back to the operator, give them back to the operator, in order to increase his knowledge and understanding of the plant, in order to make the job more fascinating, so to say, to probably even increase motivation, to get rid of some of the boredom, and only automate those things which really go beyond his human capacities and capabilities to handle them.

Mr. Stadie

Well, I see that Admiral Smith has something to add to this topic.

Adm. Smith

Life is getting more complex. The Navy man doesn't have somebody looking over his shoulder. He doesn't have the community pressures except in his priesthood. His life is altogether different aboard ship--altogether different. And at the risk of being contradicted, the systems land based are more complicated. Now, we're learning to fly in space, and we don't fly in space the same way we flew when we were barnstorming in the biplanes. We fly a great deal more by computer today. And it just seems to me that we are talking about the future now--not just today--I can't see us moving any other way than increased automation, with man the master of the machine.

Charles Ehret

It is hard to contradict any of the things that have been said. I think generally speaking the systems are remarkably inert--the machine and the man. And when we plug into equations or what have you, human error prediction equations, we see that we are doing remarkably well. But, we have so many plug-in performance-shaping factors and things of this sort that can influence it dramatically. Now Bob Smith's point about qualifications is very well taken. And yet we almost lost the limits recently. For two reasons--drugs and bad shift schedules. And I think from the point of view of this man, the thinking man of which our German colleagues have been reminding us, we have to have this thinking man there. I didn't have time in my presentation to indicate that one of the manifestations of this state, the transient in our own head, is retrograde amnesia. That was surely one of the manifestations of the man landing on the moon and landing in Mexico City. So these rare, rare events which we are now working at, it's remarkable. They're still ahead of us. And we say which are the largest ones, and I would say man and his error-proneness on account of many of the things which constitute our very nature, are things to focus on. If this man is, in fact, performing very well and is very well-qualified, but he suddenly finds himself in such a state as I described a while ago, any one of us, at such a time will perform badly. You won't know your telephone number; you won't know your middle initial; you won't know any of things you are well-qualified for, and surely we must focus on many things that are your own specialties here. But this is one that is at the forefront, and it is at the forefront of this priestly character that we talk about--this dedicated character.

Mr. Stadie

Is there any other comment to this question?

Robert Carlson

It is very interesting to note, that in fact, man is error-actuated. In 1954, at Sydney University, we got from our professor of electrical engineering a definition of man. And this says that man is the most complex, non-linear, electrochemical and mechanical, error-actuated, negative feedback servosystem that is capable of mass production by unskilled labor. When you add that element into the equation, and you overlay it with all the emotionalism and all the things that have happened, and particularly the remarks of our speaker here on the biological side, I think this is one of the most important elements in understanding how he will perform, particularly when his adrenaline is pumping at about 10 times the normal. And this is the situation that we have really got, and he is one objective in the safety exercise, which might be in conflict with the utilities exercise, to keep that fuel cool. And if he does exactly that, he has achieved his safety objective and really, if you look at the reactor in its most simplest terms, all we need is a thermocouple. But we don't have the guts to put just a thermocouple in it. Because man can't design his equipment reliably enough. But can I make just one quick quotation about human communications: "I know that you believe that you understand what you think I said, but I am not sure you realize that what you heard is not what I meant."

Mr. Stadie

Are there any more comments regarding the three levels of responsibility of an operator and his back-up expertise in the control room?

Unidentified Speaker

I am not sure whether it is a question or a comment to Mr. Smith. But, I suggest that if the civilian nuclear program had the security system of the U.S. Navy, we wouldn't know about Three Mile Island either. And I agree that the plants are so complicated and are very hard to get into them to learn those systems when you really, I think, can't afford to shut them down and there's no port time when people can crawl through the pipes and things--they have a much more difficult situation than the Navy. We have to solve it somehow. They need to know it.

Adm. Smith

Bob, a quick response, if I may, because as an ex-Navy man yourself, you may be restrained. Very often, we tend to reach for the conspiracy when none exists. We find it somehow, in our human nature, to believe that we can explain things that either work very badly or work very well. I worked for the Navy for some 12 years under Rickover, and it was perhaps one of the most productive and most exasperating experiences I have ever had.

But I can tell you that there was one thing that came through throughout all of that experience, and I had a good number of them in the very early days. While what we were doing was classified, at no time was there any attempt to conceal, in any way or any manner, any equipment or personnel behavior operation whatsoever. It simply wasn't done. There was the concept of integrity.

Mr. Stadie

Perhaps, if you don't mind, we will go on to the next question. Dr. Courvoisier has stressed in this paper that what we need from nuclear power plant operators is that they are "full human persons." What do you think is the best way of selecting and training people to become full human persons?

Dr. Alonzo

Our discussions these two days to an outsider, in my opinion, could have given him the impression that we are asking for close to Nobel prize winners to run our power stations. Well, this is, of course, an exaggeration, but probably our intentions and also our needs will call for a high level of excellence. There is no question in my mind to that point. We all have to remember that nuclear reactors have indeed been run by Nobel prize winners and by outstanding scientists and outstanding engineers in many countries. I believe the first and best example is perhaps Chicago Pile number one run by Enrico Fermi. But I will go on to say, I dare to say that probably Enrico Fermi did not run their reactor only because he was a Nobel prize winner, but because he was also very well-treated. His mind was very well-treated to experimental work. And also because he was able to motivate his team well beyond other things. I really believe that perhaps the Fermi example sets the pattern for our reactor operators in just the expressions. The first one is that the operator should have technical and scientific competence. The second is that they must have natural skills to manipulate machines, and the third is that they will have to be mentally suited for accepting and performing the work.

Mr. Stadie

Thank you. I do not think there will be any disagreement here.

Warren Witzig

I am no more qualified to comment than the fact that I am a parent, and I have just been around the earth for a few decades. It seems to me that one of the things we don't do today as well as we could is establish the kind of reward system, the kind of thing that Bob was talking about earlier, or Alonso was talking about with the incentive, the teamwork, the motivation; it seems to me that we can do a lot better on that front, and there are obviously some changes underway today. But we have got to do much better in that area. What is it that makes individuals feel well satisfied with their job, with what they are doing? I think there are a few things that Alonso has mentioned and maybe one or two others. We want to be loved. You may laugh at that. But we want to be loved. We want to feel that we are making a contribution. We want to have a high quality of life and eat, drink, and be merry. These are all very human characteristics, and I think that as we reward reactor operators with these kinds of attributes that are associated with their job, we are going to get better and better operators. We need more women in the nuclear business. You know that in the entering freshman classes of nuclear engineering across this country, about 20 percent of the entering freshman are women. And when it comes to graduation time, it is only about 10 percent. They go off into other pursuits--business, etc., and I think that is a very detrimental sort of thing. I think this is one of the things we have got to fix in this whole matrix.

Unidentified Speaker

I won't add any comment to that. I was just thinking about something else. I don't know if there are any operators in this room and especially having followed the two and three days now of this conference, but I would like to know what they think of the way they have been treated here, and if they are still willing to be operators in our nuclear class. I would like to know about that. I don't know if there are any in the room.

Butch Colby

My name is Butch Colby. I am the manager of Power Operations for Singer-Link, and I received my senior operator license at a utility in the midwest. I have attended, in the last three years, some of these conferences, and most of it has to do with operator training and what have you. I think that the thing that I find most fascinating about this is probably the lack of operators at these conferences, and I think possibly that this is something that you, on the panel, should address.

Unidentified Speaker

I would like to add a few remarks along the lines of being loved or more neutral, being accepted. I guess what one has been addressing already through this conference was the acceptance by the management, in terms of positive feedback, of what he is doing. One could as well look at acceptance in terms of public acceptance, and in this area, a lot remains to be done. And, he has to be accepted by the regulatory authorities. I want to say that because I am coming from that side of the fence, and I guess we have done not too good a job during the recent years by imposing, so to say, all the requirements and ever more detailed requirements on the same kind of person; this has, at least in our country, led to some kind of considerable demotivation, I would say. We have contributed to some extent to many guys not being very willing to accept this responsibility any longer.

Unidentified Speaker

We had a chief executive officers conference at INPO about a month ago, and one of the subjects that came up was this very subject--how we motivate, and how we can keep operators happy. And this public perception business was the major thing that the chief executive officers wanted to talk about. In some parts of the country, it was rather bad that the families of the operators were, if not harassed, certainly ignored and not accepted socially in the community in some cases, which made it very difficult for the individual to stay in that business. You know, day after day, year after year, when his family was somewhat isolated from the local community. Now, I realize that this is not the case in every community, and it certainly is not the case where the communities are larger and people melt into the population. But in some of our reactor sites that are fairly well-isolated, the communities are small that serve that area, and we need to do a better job of public relations to make sure that our operators are accepted by the community and are looked up to as professionals and not looked down at as something carrying radioactivity around in his shoes or something. That scarlet letter A in Hawthorne's novel stands for atomic, not adultery.

Unidentified Speaker

Seriously, for a moment, the Sunday after the TMI accident, when I had been up with my friends and the rats working all night with a pair of graduate students calculating the fission product inventory in Three Mile Island-1, we dug out a code and you know how we do that and walked down the aisle in church, and as we passed one "friend" the remark was made, "Now there's the guy who is causing all that trouble down there." Now that's not a kind of friendly way--suppose you are the operator. Here I am, 150 feet away.

Mr. Stadie

I guess that is a recurring problem at any nuclear conference. Public acceptance is a stigma which many of us have lived with for a number of years. I think one could say a lot about this, but I think this is probably going a bit beyond what we are trying to discuss here today, and I see time is mercilessly running forward, so let me come to another question which is very brief, very short. What is the best way to reduce operator error? And then, some of the panel members asked, how can we measure it? Very simple, very straightforward, and probably very difficult to answer.

Madam Carnino

I think, from my personal point of view and from being a nuclear engineer for many years, we are at the stage where reliability assessments were about 10 years ago. When we performed our very first reliability studies of systems in nuclear power plants, we found some weak points--some problems due to systems and equipment. What did we do then? We improved our ways of dealing with these failures of components, and we know now how to technically design a good system in order to improve its reliability, and this means, at this time, we had dominant factors. Now we have moved forward. We know how to do it. So we think now that these reliability problems have been solved, or we know how to handle it. Now we find that having decreased these problems, we find that the human beings and the human reliability is a very important factor.

Because, we have decreased the other one. And I think we have to analyze the causes of the human errors, or so-called human errors. It's not very obvious, but sometimes we find the real causes. And we have a tendency to say that it is a human error, especially in the execution and action, but sometimes we could explain it by a bad design, by other design errors, bad procedure, bad physical inventory, and so on. And if we find such causes, then we can find engineering solutions without having to adapt the man to the machine we have designed. I think this is something perhaps that we have not discussed much here, except this morning. I think we can address, at the same time, the design problems and see if we can decrease the error rates, the human error rates, in this way, and then perhaps we'll find that we need a much more detailed and refined training. But I think now the training shares the causes of human errors as well as design errors and all the other factors that have existed.

Unidentified Speaker

I think that is a fairly comprehensive coverage of the problem. All I would say is that the operator says we have to live with the design mistakes and have to accommodate those design errors. It is a question of the feedback, and I think that what has been said about analyzing causes of operator error and that feedback, but of course, they will feed back into new designs, but I think one of the problems is the multiplicity of designs. Again, we come back to this problem of should plants be standardized and if benefits that can be derived from feedback can then be applied to standard plants. Certainly, there needs to be a much greater exchange of information, a better method of analyzing operator error, and one would hope that the international community can achieve that with the work that is being done by working parties, such as those run by CSNI. But I think operators do have to live with design errors, and it's a question of feedback and the process of improving them.

Mr. Stadie

If there are no more panel views, I should like to open this briefly to the audience.

Ken Elston

I am a station manager, but I think I am an old operator because I have operated many nuclear research reactors, as well as power plants on hands-on. What I would like to comment on is that I have heard an awful lot at this conference on operator error. There is another side that you should look at. We talk about automation. Our plants are very automated. But, I can confidently say that our production is much higher and our safety record is much higher because of the present operators and the actions that they take. So, I am concerned that people are talking about emphasizing operator error to the operators and maybe making it such that the operators will be very concerned about taking these actions when it must be 100 to 1 that they help production, and they help safety.

Mr. Stadie

Maybe we should end up asking what role could the simulator play in the training of operators, particularly with regard to what Dr. Courvoisier described as the "third level of responsibility by the operator." Again, I speak here of the operator in the broadest sense.

Unidentified Speaker

I think it is clearly obvious here that the simulator has a large role to play in the quantification and training aspect of commercial nuclear power. I think the one thing we have to be careful about is, and it was pointed out several times today, that the simulator certainly cannot answer all of our ills, if we have ills, and I don't think we should anticipate or expect it just because you have a site-specific simulator that you use, that you are necessarily solving your operating problems, even in a retraining situation. And getting yourself a lot of canned scenarios that the operators soon memorize, you may not think that they are very bright but they pick up those in a hurry, and they can just almost spot when one of these things are starting, even if you try to mix them up a little bit, there are only so many things that you can really do, and there are just so many things that the trainers are capable of entering into the simulators, not just because of the simulator, but because of their own mentality. There is a certain mind-set that we get into on this training. So, I would just like to say that having been brought up where simulators were a no-no in the world that I was brought up in, and realizing how much they are required in the commercial world, I think we have to be very careful that we get the most out of them but not really anticipate or expect that they are going to be even the capstone of our training program. I still would like to emphasize that solid system understanding, qualification, to my mind, is just as important as being able to run through some scenarios on a simulator.

Unidentified Speaker

I am not really going to talk about simulators. I am going to talk about another device, which in my opinion, is also very important for training operators. That is training reactors. There was a time a long time ago when a lot of small reactors were just built for training people. And here in this conference, we have been emphasizing simulators, and there has been a couple of papers or three papers and enough people, and also Professor Witzig mentioned his reactor and maybe some others. I believe that these reactors are very important for training people, because they are very close to reality. And if you look at the pattern that pilots operate certainly in their training, and I remember now one paper presented by a captain of Lufthansa in Stockholm who mentioned training in a small plane for a pilot is a must. So I believe that we should also emphasize using small reactors, and it seems to me not enough development in that area has taken place.

Unidentified Speaker

I couldn't endorse the previous speaker's point more. There is a difference when a man knows that he is operating a bunch of -----, or whether he actually has control of radioactivity. You can sense it when you watch him in each circumstance. If I can add, just for a moment, the role of the simulator not perhaps in the abnormal as you asked, Mr. Chairman, but in the other conditions, there are few universities that are going to be able to afford the five to ten million for simulators. So the concept simulator limited is very useful, and also I make the plea for those utilities to share with their neighboring universities some time on those simulators in the course work. It can be done; it has been demonstrated at several universities in the past year and it is very, very productive.

Mr. Stadie

I had hoped that there would remain a few minutes at the end of the panel discussion for anybody to ventilate any point which he felt needed to be made. Unfortunately, time is rapidly running out, and there are only one or two minutes left. Is there anybody who wants to make one final comment to this panel?

Dr. Alonso

I have been talking so I am very ready to give the floor to anybody, but since this is not the case, I would like to talk a little bit about interaction, and to me interaction is a very important thing. It not only concerns what is called the man-machine interface or the man-machine interaction but it is a very broad field, in my opinion, because there is a link going from let's say the highest authorities in the country to the electricity consumers and this link passes through the operator onto the machine that the operator is handling. And this is very clearly the case, and it was demonstrated very clearly in my opinion in the case of TMI. So, when you talk about inter-actions, you have to consider that the plant interacts with the site and vice versa, the site interacts with the operating crew and vice versa, and that the electricity consumers interact also with the plant operator. Probably one way to reduce human error is to do a study of these interactions because, after all, human persons are put into the chain.

Mr. Stadie

Thank you, Mr. Alonso, for concluding this panel with this rather noble perspective. I am afraid that we now have to close, and I can only voice my regret that we cannot go on, although I am sure there are many other aspects we could discuss here.

I should like to thank the panel members and the audience for their lively participation in this exchange, and I turn over the microphone to Bob Smith of INPO, who has been the vice chair-man of this meeting.

Adm. Smith

Thank you. I have two administrative announcements. I want to remind you that for the tour to the McGuire Power Station tomorrow morning, the busses leave right outside this room, these doors, right outside in this driveway at 8:00 in the morning.

However, they will be available at 7:30, if you would like to come down and put your luggage in the busses early and get rid of the baggage when you check out. There will be one of the busses that will be marked "airport," and what that will mean is that immediately upon completion of the tour, it will leave at noon directly for the Charlotte airport, for those of you who might want to catch a plane. I think the bus will be there no later than 1:00. One more announcement is that we will obviously have proceedings of this meeting, and as I understand it normally the attendees will get a copy; however, I think when you make a printing like this, it's almost as cheap to print a few extra copies, so if any of you would like to get some extra copies, if you will just leave your name and the number of copies you would like on a tablet that Karen has in the back of the room on your way out, I will see that you get some extra copies. It might make it easier for you to distribute these results within your own organizations, if you have extra copies, and certainly, like I say, if we are going to print 100, we might as well print 500, because the paper is cheap--it is the setup that is expensive. One more thing--I think that I have had several remarks made to me, and I am happy that this conference was run very smoothly, and I would like to ask the people who have provided the oil for the smooth operation to please stand up, and I think we can let them know what our approval is. Of course, we had Mike Stephens, Ron Wilson, Pierre Lienart, Karen Rawley, Candy Nunneley, Barbara Trott, and the three young ladies from Duke, Barbara Thomas, Nancy Demuro, and Delilah Suggs. Will you people stand up please so we can see who you are? And now, on behalf of Joel Kramer of the NRC, and Dennis Wilkinson from INPO, and of course, Klaus Stadie from OECD, I declare this meeting adjourned, and thank you very much for your participation.

CSNI SPECIALIST MEETING
ON OPERATOR TRAINING AND QUALIFICATIONS

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16. ABSTRACT *(200 words or less)*
The events during the accident at TMI-2, along with others identified in retrospect at other nuclear plants, re-emphasized the critical role of the reactor operator. Many countries are focusing greater attention on the capabilities of control room operating staff and on the problems they face. In view of the importance to safety on the subject, the CSNI Subcommittee on Licensing decided that a specialist meeting should be held on the broad aspects of operator selection and training and the functions and organization of operating staff. The meeting focused on the functions, role and organization of control room personnel as a crew and as individuals; selection and qualifications of personnel; operator training and requalification; evaluation of crew and individual performance; professional and career alternatives for control room personnel; and "concepts for the future" (e.g., implementation and impact of computer technology, advanced simulator concepts, off-site monitoring and support). Fourteen countries and three international organizations were represented. This report consists of two volumes.

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