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FINAL TECHNICAL REPORT
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STRENGTHENING THE FISSION REACTOR NUCLEAR SCIENCE AND ENGINEERING PROGRAM AT UCLA:
A Matching Grant Program with PG&E

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DISCLAIMER

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Introduction

This is the final report on DOE Award No. DE-FG03-92ER75838 A000, a three year matching grant program with Pacific Gas and Electric Company (PG&E) to support strengthening of the fission reactor nuclear science and engineering program at UCLA. The program began on September 30, 1992. The program has enabled UCLA to use its strong existing background to train students in technological problems which simultaneously are of interest to the industry and of specific interest to PG&E. The program included undergraduate scholarships, graduate traineeships and distinguished lecturers.

Undergraduate Scholarships

UCLA has never had an undergraduate degree in nuclear engineering. Its program was graduate in nature, and heavily oriented toward Ph.D. quality students. A few nuclear-related courses were offered to seniors who take the power engineering option in the mechanical engineering program for a Bachelor of Science degree.

During the first year of the program, five $1,000 scholarships were granted to each of the following students:

Rudy Dahbura
Jessica Hoffman
Phillip Kwong
Matthew Quach
Achilles Young

Distinguished Lecturers

The lectures on frontier research topics were as follows:

1992-93:

1. On February 4, 1993, Professor S.G. Bankoff of Northwestern University visited with Professor Dhir and gave a seminar on instability and rupture of thin, heated liquid films. This seminar was supported in part by the matching grant program.

2. On May 6, 1993, Dr. Robert Henry of Fauske Associates, Inc. visited with Professors Okrent, Dhir and Kastenberg, and the students involved in their research, and gave a seminar on issues related to Level 2 diagnosis, and suggested avenues of attacking portions of this complex problem.

1993-94:

1. "Numerical Simulation of Complex Fluids," Professor Pushsendra Singh, Chemical and Nuclear Engineering Department, University of California, Santa Barbara.

2. "Computational Material Sciences," M.I. Baskes, Sandia National Laboratories.

1994-95:

1. "Experimental and Theoretical Studies in Microscale Engineering," Professor Arun Majumdar, Department of Mechanical and Environmental Engineering, University of California, Santa Barbara.

2. "Systems Engineering: An Approach to Information-Based Design," Dr. George A. Hazzelrig, Directorate of Engineering, National Science Foundation.

V **Research Summary and Graduate Traineeships**

Four topics were selected for research the first year, with the benefit of active collaboration with personnel from PG&E. These topics remained the same during the second year of this program. During the third year, two topics ended with the departure of the students involved (reflux cooling in a PWR during a shutdown and erosion/corrosion of carbon steel piping). Two new topics (long-term risk and fuel relocation within the reactor vessel) were added; hence, the topics during the third year award were the following:

- Reflux condensation and the effect of non-condensable gases.
- Erosion/corrosion of carbon steel piping.
- Use of artificial intelligence in severe accident diagnosis for PWRs (diagnosis of plant status during a PWR station blackout scenario).
- The influence on risk of organization and management quality.
- Considerations of long term risk from the disposal of hazardous wastes.
- A probabilistic treatment of fuel motion and fuel relocation within the reactor vessel during a severe core damage accident.

[illegible] candidates for graduate traineeships was matched to the topics selected for re[illegible]ld be of mutual interest to PG&E and UCLA. This is discussed in the paragraphs wh[illegible]ow.

Prof[illegible]ir guided the research on reflux cooling of a PWR during shutdown. Mr. Benja[illegible] and then Mr. Dengshan Wang were the graduate students who were awarded trainees[illegible]worked on this research, until the research was terminated with the departure of Mr. Wa[illegible]

Professor N. Ghoniem guided the research on erosion/corrosion of carbon steel piping. Mr. Hanchen Huang was the PhD candidate who received a traineeship and worked on this research up to its completion in late 1994. Dr. Huang received the PhD in December 1994 and accepted a position at Lawrence Livermore National Laboratory.

Professors D. Okrent and W. Kastenberg guided the research on determining plant status after the onset of severe core damage using artificial intelligence for a station blackout scenario in a PWR. Mr. Zheng Wu, a PhD candidate, received a traineeship and worked on this research. He currently has to complete the writing of his thesis. Two papers have been presented at international conferences based on his research. The papers are reproduced in Appendix C-1.

Professors D. Okrent and G. Apostolakis guided the research on the influence of the quality of organization and management on risk. Ms. Yongjie Xiong, the PhD candidate who received a traineeship, completed her thesis in late 1995 and works at PLG Inc, Newport Beach. Five papers have been presented at international conferences based on her research. They are reproduced in Appendix D-1.

Professor D. Okrent is guiding the research on considerations of long term risk in the disposal of hazardous wastes. Mr. Zhongbin Shu is the PhD candidate who initiated his research on this complex topic during the final year of this award.

Professors V. Dhir and D. Okrent are guiding the research on a probabilistic treatment of fuel motion and fuel relocation within the reactor vessel during a severe core damage accident. Mr. Xuegao An received a traineeship and began a review of this complex subject during the final year of this award.

One other PhD candidate, Mr. Leiming Xing, received partial support for a couple of months during the spring quarter of 1993, while he was completing his thesis on diagnosis of ATWS events using neural networks and an expert system. Dr. Xing now works for PLG Inc., Newport Beach.

The status of the research on reflux condensation at the time it was ended is reproduced in Appendix A.

The status of the research on erosion/corrosion of carbon steel piping, as of the time of completion of the PhD thesis of Dr. Huang, is summarized in Appendix B.

The research on the use of artificial intelligence for diagnosis of plant status during a severe core damage accident is summarized in Appendix C.

The research on the effect on risk of the quality of organization and management is summarized in Appendix D.

The research objective and approach for the work on considerations of long term risk from the geologic disposal of hazardous wastes are presented in Appendix E.

The research objective and approach for the work on fuel motion and relocation within the reactor vessel during a severe core damage accident are given in Appendix F.

APPENDIX A

An Investigation of Reflux Cooling in
a Pressurized Water Reactor During Shutdown

Reflux Condensation And The Effects of Noncondensable Gas

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Recent plant experience has included many events occurring during outages of PWR. During outages the power is low, the coolant system may be in a drained state with air or nitrogen present and various primary system closures may be unsecured. With the residual heat removal system(RHR) operating, the core decay heat is readily removed. However, if the RHR system capability is lost and alternate heat removal means can not be established, heatup of the coolant could lead to core coolant boil-off, fuel rod heatup, and core damage.

By identifying the possible plant conditions and cooling methods that might be used, the controlling thermal hydraulic processes and phenomena include

1. Gravity drain into the reactor coolant system;
2. Core water boil-off;
3. Reflux condensation cooling processes

This report identifies and analyzes one of the important thermal hydraulic phenomena following loss of RHR system in the PWR, reflux condensation. As one of the alternative heat removal schemes, reflux condensation uses the steam generator as the heat sink. The reflux cooling processes include (a). The initiation of the reflux cooling; (b). The effect of non-condensable gas.

In the reflux condensation cooling mode, core decay heat is removed by boiling. The steam flows to the steam generators where it is condensed on the inner surfaces of the steam generator U-tubes. The condensed water from the up flow side of the U-tube flows downward, against the upward flowing steam, into the steam generator inlet plenum, hot leg, reactor vessel upper plenum, and back to the core. The condensed water from the down flow side of the U-tube returns to the cold leg. Assume that at least one steam generator is operational, which means that at least one SG secondary side must contain cold water and nozzle dams must not be present in the hot and cold leg.

a. Initialization of the reflux condensation

To establish reflux cooling, the core boiling must sufficiently pressurize the RCS to compress the nitrogen or air in the upper regions of the RCS to expose condensing surface to the steam flow. At least one steam generator is operational. Furthermore the possible need for draining or venting of the primary system in order to obtain a stable reflux cooling mode will be studied.

b. The effect of noncondensable gas

When a condensable vapor is condensing in the presence of a noncondensable gas, the vapor must diffuse through the gas, requiring a decrease in vapor partial pressure toward the liquid vapor interface. Thus, interface saturation temperature is significant below the temperature of the main vapor gas mixture. A significant decrease in heat transfer coefficient results from the presence of very small amounts of noncondensable gas. Many experiments on this effect have been done such as PKL, FLECHT-SEASET, EPRI/SRI facilities. Most recent are the University of California, Berkeley's[6] and Massachusetts Institute of Technology's[5] experiments to investigate the effect of noncondensable gas. Details will be shown in the following section.

This report sets up models to simulate reflux condensation. The effect of noncondensable gas will be considered. A code in FORTRAN will be developed for the simulation.

I. Separate models

The first part will set up models for the separate parts of the system.

1. Pressurization of the reactor vessel.

The decay heat of the core depends on the initial power and the time after the rector shutdown.

$$Q = 0.095\ Po\ (3600t)^{-0.26} \tag{1}$$

where	Q	Decay heat
	Po	Power before the plant shut down
	t	Time after the plant shut down

The relationship between the saturation temperature and the pressure:

$$P = \exp(16.2834-3816.44/(T-46.13)) \tag{2}$$

where	P	Saturation pressure of the steam
	T	Saturation temperature

The steam generation rate

$$\underline{Mv} = (Q - (M_l C_{pl} + M_{stl} C_{pstl} + M_{gs} C_{pgs} + M_v C_{pv})\ dT/dt)\ /h_{fg} \tag{3}$$

where	$\underline{Mv}$	Steam generation rate
	M_l	Mass of water
	M_{stl}	Mass of reactor vessel

M_{g} Mass of noncondensable gas
M_v Mass of vapor
h_{fg} Latent heat of vaporization at the pressure of interest

The temperature and pressure

$$P_1 = M_{g1} R\, T_1 / V_1 + M_{v1} R\, T_1 / V_1 \tag{4}$$

$$P_2 = M_{g2} R\, T_2 / V_2 + M_{v2} R\, T_2 / V_2 \tag{5}$$

$$M_{g2} = M_{g1} - M_{gout} \tag{6}$$

$$M_{v2} = M_{v1} + \underline{Mv} - M_{vout} \tag{7}$$

where M_{gout} The mass of gas flow to the hot leg
M_{vout} The mass of vapor flow to the hot leg
Subscripts : 1 — at time t
2 — at time $t + \Delta t$

Notes: 1. $M_{out} = M_{gout} + M_{vout}$ depends on the system parameters such as the pressures in the SG and the reactor vessel etc. It will be given at the hot leg model. We can give the mass fraction of the noncondensable gas by $W_{air} = M_{gout} / M_{out}$.
2. $V_2 = V_1$.

Calculation strategy: Give T_0, P_0, V_0, M_{g}, we can get the arguments needed: T_2, P_2, $\underline{Mv}$ etc. Calculations begin with a given T_2^*, By the equation (3) get $\underline{Mv}$, From (5)-(7), get P_2^*. Compare P_2^* with P_2 calculated by the (2) and (4). If $P_2^* > P_2$, let $T_2^* = T_2^* - \Delta T$; If $P_2^* < P_2$, let $T_2^* = T_2^* + \Delta T$. Iterate until $P_2^* = P_2$.

2. The mixture level in the core

The mixture level in the core is very important for the flow regime of the hot leg and the heat transfer model of the steam generator.

Define j_g as superficial vapor velocity, V_{gj} as vapor drift velocity

$$j_g = \underline{Mv} / (\rho_v A_{fc}) \tag{8}$$

where A_{fc} Flow area in the core
ρ_v Density of Vapor

$$V_{gj} = 1.41\, [\sigma\, g\, (\rho_l - \rho_v) / \rho_l)]^{1/4} \tag{9}$$

where σ Surface tension of water, can get by (14).

Mixture level

$$Mxlvl = (lvl0 - lvlcore * \alpha)/(1-\alpha) \quad (10)$$
$$\alpha = j_g / V_{gj} \quad (11)$$

where
- α Void fraction
- Mxlvl Mixture level in the core
- lvl0 Water level without bubbles
- lvlcore The level of the core (consider the middle of the core)

Note: If the Mxlvl > hot_leg_level, then the hot leg uses slug flow model. If Mxlvl < hot_leg_level, the hot leg uses stratified flow model.

3. The steam generator and the effect of noncondensable gas

The steam generator(SG) is the most complex and most important part of the model. We set up two models that are (1). Non-flooding model, (2). Flooding model. Flooding means when the condensation rate and the steam flow rate are very high, because the U-tube is very thin, the condensed water blocks the steam flow. For the non-flooding condensation, we can use the following model.

The heat transfer rate[1] from outside of the tube to the water on the secondary side

$$Q_1 = \mu_l h_{fg} [(\rho_l-\rho_v) g /(g_c \sigma)]^{1/2} [C_{pl} (T_{wo}-Ts) / (C_{sf} h_{fg} Pr)]^3 \pi D_o \quad (12)$$

where
- $g_c = 1.0$
- T_{wo} Outside Temperature of the U-tube.
- Ts Temperature of the water of the SG secondary side
- $C_{sf} = 0.013$
- D_o Outside diameter of the U-tube
- L Length of the calculation (The U-tube is calculated by several sections. The length of each section is L).

$$Pr = \mu C_p / k \quad (13)$$
$$\sigma = 0.2358(1-T_s/647.15)^{1.256} [1-0.625(1-T_s/647.15)] \quad (14)$$

The heat transfer rate from the inside of the tube to the outside of the tube.

$$Q_2 = 2\pi K (T_{wi} - T_{wo}) L / \ln(D_o/D_i) \quad (15)$$

where
- T_{wi} Inside temperature of the U-tube
- D_i Inside diameter of the U-tube

K Thermal conductivity of the tube wall

The heat transfer r..te is equal to :

$$Q_3 = \underline{Mvc}\ h_{fg}\, \pi\, D_i L \tag{16}$$

$$Q_4 = h(x)(T_i - T_{wi})\, \pi\, D_i L \tag{17}$$

where $\underline{Mvc}$ Condensation rate
T_i Temperature of the bulk gas-steam mixture
$h(x) = Nu(x)\, k\, D_i$

According to MIT result[5]:

$$Nu(x) = 6.123 Re^{0.223} [(Wa,w - Wa,b)/Wa,w]^{1.144}\, Ja^{-1.253} \tag{18}$$

where Wa, w Mass fraction of the noncondensable gas at inside wall of the tube.
Wa, b Mass fraction of the noncondensable gas in the mixture.
Re Reynold number
Ja Jakob number

Or use UCB result[6]:

$$h(x) = 0.005 Re_{cond}^{0.45}\, W_a^{-1.1}\, h_{Nusselt} \tag{19}$$

where W_a Mass fraction of the noncondensable
$h_{Nusselt}$ Heat transfer coefficient for pure steam

and $$W_a(x) = M_a/(M_a + M_v(x)) \tag{20}$$

where M_a noncondensable gas flow rate
$M_v(x)$ vapor flow rate at x

$$M_v(x) = M_v(x-L) - \underline{Mvc} \tag{21}$$

and $$Re(x) = V(x)L/\nu \tag{22}$$

where $$V(x) = (Ma + Mv(x))/\rho\, A_f \tag{23}$$

and A_f flow area of the U-tube

The pressure at different location in the tube are related as:

$$P(x) + 0.5\rho_1 V(x)^2 = P(x+L) + 0.5\rho_2 V(x+L)^2 + H_f \tag{24}$$

where $P(x)$ Pressure at x

H_f Pressure loss by friction

$H_f = f \rho V^2 L/(2D)$

$\rho = PM/RT$ density of the mixture

$$\rho = \rho_v + \rho_g \qquad (25)$$

Since $Q_1 = Q_2 = Q_3 = Q_4$, we can calculate the $\underline{Mvc}$, T_{wi}, T_{wo}, $Mv(x)$, T_i, and Q etc.

Calculation strategy: Calculate section by section. For each section, we begin with (19)-(23), get h(x), by (2), (4), (5) and (24), (25), we can get P(x), T_i, then by (12)-(17), we get Q_i, T_{wi}, T_{wo}, T_s etc.

When the U-tube floods, the situation will be different and the heat transfer and flow process will be unstable.

4. The hot leg model

The mixture level of the reactor vessel and the steam condensation will determine the flow regimes in the hot leg. Different models will be used for stratified and slug flow models.

5. The cold leg and any other parts

Because the water seal in the leg, the flow scheme in cold leg will be different from that in hot leg.

II. Couple the models together

This part will couple those models together to see the behavior of the whole system and the effect of noncondensable gas. be compared with the result of the most recent experiment.

References:

1. Warren M. Rohsenow, James P. Hartnett, Hand book of heat transfer
2. NUREG/CR5855 EGG 2671 thermal-hydraulic processes during reduced inventory operation with loss of residual heat removal. April, 1992
3. C. Calia and P. Griffith, "Modes of circulation in an inverted U-tube array with condensation," Thermal hydraulics in nuclear power technology vol 15. page 35-44
4. B. Chow and V. K. Dhir "An investigation of reflux condensation in a PWR during reduced inventory with loss of residual heat removal"

5. Mansoor Siddique, Michael W. Golay, and Mujid S. Kazimi, "Local heat transfer coefficients for forced convection condensati :n of steam in a vertical tube in the presence of a noncondensable gas," MIT, 1992
6. K. M. Vierow, " Behavior of steam-air system condensing in concurrent vertical downflow ", Department of Nuclear Engineering, University of California at Berkeley. 1990.
7. V. K. Dhir, R. C. Amar and J. C. Mills, "A one dimensional model for the prediction of stratification in horizontal popes subjected to fluid temperature transient at inlet," Nuclear Engineering and Design. vol 107. Page 307-314, 1988.
8. Isao Kataoka and Mamoru Ishii, "Prediction of pool void fraction by new drift flux correlation" ANL-86-29
9. Chapman, heat transfer
10. M. Siddique, Michael W. Golay, and Mujid S. Kazimi, "The effect of noncondensable gases on steam condensation under forced convection condition," MIT-ANP-TR-010, MIT, 1992

Including the Influence of Organizational Factors Into Risk

Yongjie Xiong and David Okrent

1. Introduction

The effect of organizational factors on the risk of a nuclear power plant has received more and more attentions in recent decades. The ultimate goal is to incorporate the influence of organizational factors into risk analysis. To achieve this goal, the following three questions must be answered: (1) what are the organizational factors and what are their characteristics; (2) how to measure these factors; and (3) how to quantitatively include the impact of these factors into risk analysis. To address these three questions, this year we have carried on the work in these three aspects. First, operational experiences expressed in the ASSET reports are examined in terms of twenty organizational dimensions (factors) proposed by the Brookhaven National Laboratory and Pennsylvania State University [1,2]. We hope this study can be potentially used in areas like NRC and INPO inspection in determining the organizational performance's contribution to the root causes. Second, the Behaviorally Anchored Rating Scales (BARS) method have been used to develop the measurement scales for some of the categories of one important organizational factor, deep technical knowledge. BARS for seven subcategories of deep technical knowledge have been finished. This study provides a measurement method for organizational factors and can be extended to some other organizational factors. Third, the most important step, is how to use these measurement scales to include the influence of organizational factors into Probabilistic Safety Assessment (PSA). The Work Process Analysis Model [3,4] (WPAM) has been proposed. It is an analysis tool to quantitatively including the impact of organizational factors on nuclear safety through the key work processes in a nuclear power plant. This year we continued the work on the design change work process.

2. Operational Experience Study

In the past, we studied the LERs of Plant FitzPatrick (LERs of 1988-1991) and the LERs in 1985's Precursor analysis [5], hoping to obtain some statistical data on organizational and management factors. But information in LERs is usually very brief and does not go deeply into organizational factor's root cause analysis in many cases. For this reason, we then searched other literature. Accident analysis reports and the ASSET reports are very useful in this sense. Three accident analysis reports and three ASSET report were chosen for detailed study initially. Recently, as a result of a series of research activities, a collection of twenty organizational dimensions (factors) has been identified [1,2]. Brookhaven National Laboratory (BNL) have worked out the definitions for these dimensions. These dimensions represent a comprehensive although overlapping taxonomy of organizational elements that related to the safe operations of nuclear power plants. In the study of operational reports, we use these twenty organizational dimensions.

For several years the IAEA has offered its help to assist nuclear power plant operating organizations by means of the Assessment of Safety Significant Events Team (ASSET) program. More than twenty nuclear power installations have invited the IAEA to send ASSET teams of experts to perform reviews of operating experiences. The team prepares a written report to the nuclear installation, a report which focuses on the effectiveness of the organization in correcting problems, and, in particular, the depth and adequacy of root cause analysis. The purpose of ASSET is to review the operating organization and provide conclusions on the appropriateness and completeness of the planned and implemented corrective action. Generic lessons are drawn and suggestions are offered when necessary to improve plant management control on prevention of incidents and thus to enhance the overall level of operational safety. We picked three ASSET reports for preliminary study. The following is one of them.

Leningrad Nuclear Power Plant Accident Analysis [6]

EVENT: Fuel damage followed by release of unfiltered gases outside the plant

I. Problems

(1) (page 56 of Report [6]) "Unexpected closing of regulating valve."

"In general the valves fulfilled the regulating functions in a correct way for many years. Problems were encountered when valves had to perform an isolating function. For that reason many valves had already been replaced before the event."

(2) (page 52) "There was no logic system present to avoid release of untreated

radioactive gases to atmosphere."

(3) (page 56) "Procedure fails to give guidance in this situation."

II. Suggested Relevant Dimensions:

Above problems show a lack of deep technical knowledge in design, design modification, and response actions to the event. The previous analysis of replacing the regulating valve did not point to the possibility of blocking flow. This might be caused by lack of deep technical knowledge, or not having good testing program. In case of no the logic system for channel rupture (this is lack of deep technical knowledge in design), and lack of guidance of procedures in this situation, the operating staff were unaware of the hazardous situation. Although there was a basic understanding that the logic existed, no attention was given to the position of the valves until the signal of radioactivity at the roof. This shows the operators lack of deep technical knowledge, maybe also training.

Another problem is communication. In this event, the reactor panel engineer, the system panel operator, the radioactivity systems control operators, the operators for the filtration system and so on are involved. But they are "in limited responsibility areas, in physical areas isolated from one another, and those actions and decisions are coordinated through supervisors, not through interactions amongst themselves" (page 71).

Therefore the suggested relevant organizational Dimensions are: Formalization, Communication-Interdepartmental, Communication-Intradepartmental, Technical Knowledge, Training, Problem Identification and Safety Culture.

The root causes of the events studied in ASSET reports, show a strong relationship with organizational factors. Centralization, communication, formalization, problem identification, resource allocation, roles-responsibility, technical knowledge and training are the key organizational factors which influence the plant safety. This study is only the beginning in doing, further study may lead to the potential use of the organizational factors (dimensions) in AEOD analysis of the operational experience. It can also be potentially used to help NRC and INPO inspection in determining the organizational factors' contribution to the root causes.

3. The Measurement of Organizational Factors

After identifying the organizational factors which will impact the plant safety, the next question is how to measure these factors. Based on the previous operational experiences study, one important organizational factor, adequate or deep technical knowledge, has caught our attention and received detailed study as an example of developing measurement methodology. In an effort to provide an initial basis for further examination of deep technical knowledge, technical knowledge was divided into six broad categories, some of which are subdivided into two or three subcategories as follows:

(1) PRA: Level 1, Level 2, Level 3

(2) Details of Plant: Structures, Systems, Components

(3) Transient Behavior: Reactor Physics, Thermal Hydraulic

(4) Severe Accident Management

(5) Physical Science: Health Physics, Chemistry, Materials

(6) Safety Basis: Design Basis, Technical Specification, Regulation and Industry Standards

Currently, Structured Interview Protocol, Behavioral Checklist and Behavioral Anchored Rating Scales (BARS) are methods used in organizational factors measurement. BARS[7 8] has been identified as a potentially valuable instrument in the measurement of various attributes important to an evaluation of the quality of organization[9]. BARS is a performance evaluation device that incorporates behavioral examples with general performance dimensions. Specifically, each scale represents an area of performance (in this case one of subcategories of deep technical knowledge). The behaviors are designed to facilitate the user's interpretation of poor, average, and high on each of the scales. In the development of BARS, experts are brought together to define the dimension and provide behavioral examples.

For each subcategory, a generic set of performance measures has been prepared, each representing a differing combination of aspects of deep technical knowledge applicable to the particular performance dimension. For the dimension "reactor physics", which is a subcategory of transients, ten performance measures, which take the place of the behavior examples usually formulated in a BARS application, were developed to make a list from which selections have been made to provide preliminary five-point BARS for ten different positions at the plant. The draft ten performance measures are the following:

(1) Good working knowledge (quantitative) of steady state and transient neutronics (kinetics and dynamics), including all relevant reactivity contributors, deep familiarity with all plant specific reactivity control features, reactivity accident potential (e.g. phenomenological course of ATWS) and criticality considerations under severe accident conditions, and quantitative grasp of the interaction between thermal-hydraulic and neutronic phenomena.

(2) Phenomenological understanding (semi-quantitative) of all important reactivity related effects in steady state, start-up, shutdown and accident conditions, including severe accident re-criticality, details of plant specific reactivity control features including indirect reactivity control effects. Capable of understanding the interaction between neutronic and thermal hydraulic phenomena.

(3) Capable of recognizing abnormal reactivity conditions, performing an estimated critical position, estimating the magnitude of changes in power associated with anticipated transients, (e.g. drop rods, loss of feedwater, etc.).

(4) Continuing familiarity with major relevant reactor physics concepts, (e.g. multiplication, burnup, fission product poisons, reactivity feedbacks), familiarity with reactor physics role in safety for specific plant.

(5) Capable of visualizing the plant response to change in reactivity due to plant activities (e.g. startup, shutdown), anticipate abnormal reactor states (e.g. high flux tilt, inoperable control rods, etc.) and thermal hydraulic effects on power (e.g. cool-water accident, loss of feedwater heating).

(6) Some familiarity with concepts of criticality, shutdown, reactivity feedback, reactivity transients, influence of system failure on ability to shutdown, understand safety function of critical components in systems important to reactivity control.

(7) Familiar with the concept of reactivity control (rods, boron, etc.), understands the important systems and components in controlling reactivity during normal plant operation and accident conditions.

(8) Some familiarity with the concept of fission, reactor control and systems for controlling the fission process, understands the importance of maintenance on reactivity control system, especially maintenance on redundant trains.

(9) Understands the basic fission process and concept of criticality. Can name the major systems related to shutdown of the reactor.

(10) Knows the plant uses nuclear energy as a heat source, can find his way through the plant, understands the concepts of safety (similar to a general employee).

The draft performance measures for the positions of shift technical advisor and maintenance foreman follow. (Note that an excellent rating is not appropriate for each position for each dimension.)

Shift Technical Advisor:

Excellent	1	Generic Measure (2)
	2	Generic Measure (3)
Good	3	Generic Measure (4)
	4	Generic Measure (5)
Poor	5	Generic Measure (6)

Maintenance Foreman:

Excellent	1	Generic Measure (5)
	2	Generic Measure (6)
Good	3	Generic Measure (7)
	4	Generic Measure (8)
Poor	5	Generic Measure (9)

The method has been applied thus far in draft form for seven dimensions (or subcategories) of deep technical knowledge: PRA level 1, PRA level 2, plant structures, plant systems, plant components, reactor physics and thermal hydraulic. Ten or twelve generic measures appeared to suffice for ten positions; however, it is anticipated several more generic measures would be useful to cover twenty different plant positions. The study on deep technical knowledge provides a feasible measurement method. This method can be extended to some other organizational factors.

4. The Work Process Analysis Model (WPAM) for Design Change Work Process

Work Process Analysis Model (WPAM)[3,4] studies the influence of organizational factors on safety through key work processes in nuclear power plant. The purpose of WPAM is building the link between the existing PRA and the organizational factors through the work processes in nuclear power plants. As we know, a large portion of the information-based decision processes at NPP organizations follows routine flowpaths. Formally, a work process is defined as a standardized sequence designed within the operational environment of an organization to achieve a specific goal. Operations, maintenance, engineering (design change as one of it), and plant support work processes are most important safety related work processes in a NPP.

The predictable nature of the work processes suggests that a systematic analysis can be conducted to identify the desirable characteristics of a given process and to develop performance measures with respect to the strengths and weaknesses in the process. Furthermore, since work processes are closely related to plant performance, it is possible to conduct the analysis in such a way so as to facilitate the integration of organizational factors and PSA methodology. In order to address these issues, the work process analysis model has been divided into two parts. WPAM-I consists of a mostly qualitative analysis of a given work process. WPAM-II, on the other hand, presents a mathematical algorithm for the quantification and incorporation of organizational factors into PSA.

4.1 The Design Change Work Process

The actual design control and modification activities vary from plant to plant, but the key elements are similar. The design change work process includes all activities associated with design control, the design, installation and testing of plant modifications. In our study, design change work process refers to the design control and modification activities of a nuclear power plant including: (1) field change: procedures and other document modifications which do not alter plant function, or design bases; (2) minor modification: minor design change activities which involve simple changes or small scopes of work; and in which conceptual and preliminary engineering packages are not required and formal cost estimating and design alternative consideration are not required; (3) design change: change other than above two.

Design change work process activities of a nuclear power plant vary widely in the level of complexity, scope and multi-organizational review requirement. The design change work process typically involves five major steps: design change request initiation, review and scope assessment, package generation and approval, field implementation and document close out (starting from receiving a design change request).

4.2 WPAM-I Task Analysis and Organizational Factor Matrix

WPAM-I consists of mainly qualitative analysis of a given work process. It proceeds by asking the following basic question: how can unsafe attitudes or unsafe decisions made in the work processes defeat the defenses and barriers of the organization and be translated into noticeable unsafe events of either hardware failures or human errors? The first step of WPAM-I is to conduct a task analysis. This analysis focuses on understanding the following three elements of the work processes under investigation: (1) Tasks that are involved in the work process and the plant personnel involved in each task; (2) Actions involved in each task and their failure modes; (3) The defenses or barriers involved in each task and their failure modes. Task Analysis results a cross-reference table. Table 1 gives cross-reference table for the design change work process. The second step of WPAM-I is to define the organizational factors matrix for the studied work process, which shows the organizational factors that might impact the safe performance of each task in the work process. The matrix is an assessment of the importance of the role of organizational factors in the overall quality and efficiency of the work process. The organizational factors matrix for the design change work process is given in Table 2.

4.3 Work Process Analysis Model - II for Design Change Work Process (WPAM-IIa)

The goal of WPAM is to qualitatively include organizational factors into existing PRA.

Task	Action/Barrier	Department	Personnel
Design Change Request Initiation	Document Assembly	Nuclear Eng. Design Organization (NEDO)	System Design Engineer (SDE)
	Task Initiation	Various Dept.	Variable
	Review	NEDO	NEDO Manager
Review and scope assessment	Review	NEDO	Group Supervisor (GS) SDE
Package Generation and Approval	Conceptual Engineering Package (CEP) Generation	NEDO and various Dept.	SDE and variable
	CEP Review and Approval	NEDO, Station Operation (SO), NES&L* Dept. Nuclear Generating Site (NGS) Dept., Design Review Committee (DRC), PMRC**	SDE, GSs, Discipline Manager (DM) Discipline Responsible Engineer (DRE) Technical Supervisor Engineer (TSE) Independent Review Engineer (IRE)
	Preliminary Engineering Package (PEP) Generation	NEDO and various Dept.	SDE and variable
	PEP Review and Approval	NEDO, SO, PMRC, NES&L Dept. NGS Dept	SDE, GSs, DM, DRE, TSE, IRE
	Meetings	Representative of Review Org.	SDE, DREs
	Design Change Package or Minor Modification Package	NEDO Construction Dept. Operation Maintenance	SDE, IRE, Integrated Plant Review Engineer and variable
Field Implementation	Document Assembly		
	Execution	Nuclear Construction	Variable

* NES&L: Nuclear Engineering Design Organization
**PMRC: Plant Modification Review Committee

Table 1. The Cross Reference Table for the Design Change Work Process

	Design Change Initiation				Package Generation & Approval						Imple-menta-tion		
	Document Assessbly	Task Initiation	Review	Review and Scope Assessment	CEP Generation	CEP Review & Approval	PEP Generation	PEP Review & Approval	Meetings	DCP G, R & A	Document Assembly	Execution	Document
Centralization		X	X	X	X	X	X	X	X	X			
Communication-External													
Communication-Interdepartmental	X	X		X							X	X	
Communication-Intradepartmental		X			X		X			X	X		
Coordination of work		X			X		X			X	X	X	
Formalization	X		X	X	X		X			X	X	X	
Goal Prioritization			X	X									
Organizational Culture	X	X	X	X	X	X	X	X	X	X	X	X	X
Organizational Knowledge		X			X		X			X		X	
Organizational Learning					X		X			X		X	
Ownership		X	X	X	X		X			X		X	
Performance Evaluation		X											
Personnel Selection			X	X	X		X			X		X	
Problem Identification		X	X	X		X		X	X	X			
Resource Allocation					X	X	X	X	X	X			
Roles Responsibility	X	X	X	X	X	X	X	X	X	X	X	X	X
Safety Culture		X	X	X	X	X	X	X	X	X		X	X
Technical Knowledge	X	X	X	X	X	X	X	X	X	X	X	X	X
Time Urgency		X	X	X	X		X			X	X	X	
Training		X	X		X		X			X		X	

Table 2. The Organizational Factors Matrix for the Design Change Work Process

This can be achieved by either adding organizational factors into an existing fault tree, or modifying existing fault tree entries. WPAM use the latter. It is argued that the organizational factors are already in PRA because, first, human error analyses are already in PRA, and second, failure data used in PRA are plant specific with organizational factors already considered. For this reason, our study is mainly focused on the organizational factor dependent failures and organizational common cause failures. Organizational factor dependent failures are defined as the hardware failures caused by organizational factors and those human errors which are also caused by organizational factors but not covered by existing human error study. Organizational factors common cause failures are defined as the failures of two or more components (either identical or not identical) caused by the same organizational factors.

Typically, PSA results include a set of dominant accident sequences presented in logical combinations of minimal cut sets (MCSs), which contain basic events, such as hardware failures, human errors. The first step of WPAM-II is to define the candidate parameter groups (CPGs) for the studied work process. Second, the dominant accident sequences are analyzed. Those minimal cut sets, whose basic-event parameters show strong organizational dependence, are highlighted. Then, the dependencies that are introduced by organizational factors (OFs) are evaluated by recalculating basic-event probabilities while accounting for the dependencies among the parameters that represent each basic-event, and thus, the MCS frequencies are reassessed.

4.3.1 Candidate Parameter Groups for Design Change Work Process

Different organizational factors play different roles in importance for different events, therefore a generic group of parameters must be identified to obtain the generic weights of organizational factors on these parameters. The Candidate Parameter Groups in the design change work process study are defined as a group of generic parameters of unsafe events, which are associated with design change work process, and to which failure modes in a minimal cut set are susceptible. A preliminary lists of these parameters are:

(a) Failure due to hardware change (FHC),

(b) Failure to return-to-normal after hardware change (FRHC),

(c) Failure to return-to-normal after hardware modification (FRM),

(d) Testing procedures deficiency (TPD),

(e) Calibration procedures deficiency (CPD),

(f) Operating procedures deficiency (OPD),

(g) Maintenance procedures deficiency (MPD).

This seven candidate parameter groups are only preliminary. Further study is needed. For example, the "Failure due to hardware change" is too big. It can split into generic groups, such as: wrong material, wrong system interaction, etc.

4.3.2 AHP Application in WPAMa-II to Obtain Organizational Factors Weights

The Analysis Hierarchy Process (AHP) [10] is a decision theory. This theory provides a method for multiple attributes to obtain priorities. AHP is used to obtain the organizational factor weights for the candidate parameter groups. Figure 1 is the hierarchy developed for the design change work process. Our goal is to obtain the organizational factors' weights (or priorities) for each Candidate Parameter Group. Therefore the first level of the hierarchy is Candidate Parameter Groups and the last level is organizational factors. The purpose of the hierarchy is to find the relationship between Candidate Parameter Groups and organizational factors. The second level is the design change work process tasks because the failure modes influenced by organizational factors are occurring while performing these tasks. The third level is the plant personnel/positions involved in these tasks. The last level is organizational factors, which influence the behavior of the personnel in the organization structure. After the hierarchy is developed, experts are asked to assign the pairwise comparison for the hierarchy. A computer code had been developed for the calculation of AHP. The final result is listed in Table 3.

The next step of WPAM-II is using the organizational factors rating of a plant and the weights obtained from the AHP process to screen out the organizational factors influenced minimal cut sets and modify their probabilities, i.e.,quantitatively including the organizational factors into PRA. This is still ongoing research for the design change work process.

5. Future Work

In the future, we will continue the work on the operational experience study and the

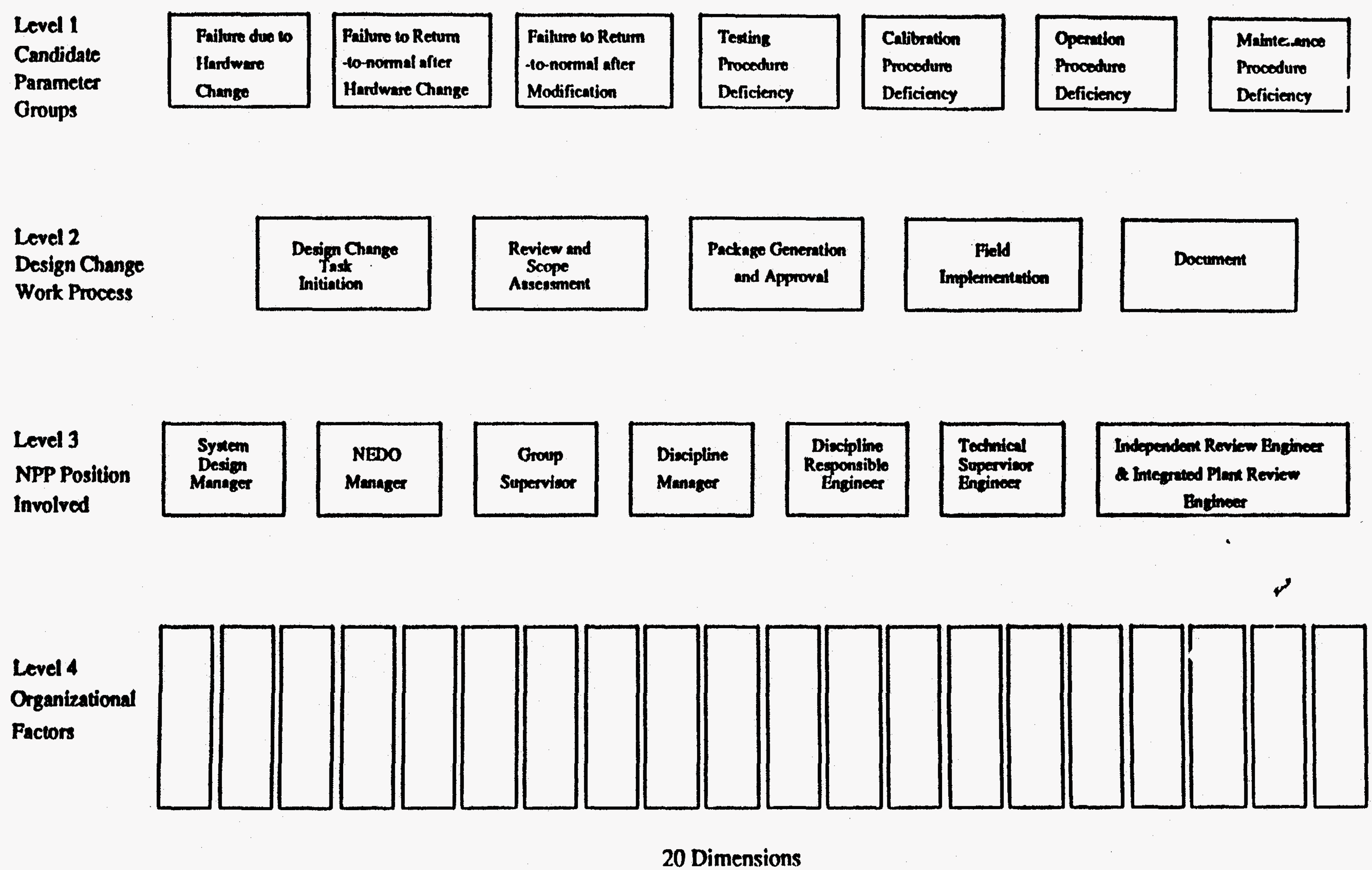

Figure 1 AHP Model for Design Change Work Process

Table 3. AHP Results: Final Candidate-Parameter-Group Weights

for Design Change Work Process

	FHC	FRHC	FRM	TPD	CPD	OPD	MPD
Centralization							
Communication-External							
Communication-Interdepartmental	.0634	.0771	.0771	.0676	.0676	.0676	.0676
Communication-Intradepartmental	.0856	.1018	.1018	.0907	.0907	.0907	.0907
Coordination of Work	.0697	.0729	.0729	.0693	.0693	.0693	.0693
Formalization	.1711	.1514	.1514	.1640	.1640	.1640	.1640
Goal Prioritization							
Organizational Culture							
Organizational Learning	.0891	.0756	.0756	.0856	.0856	.0856	.0856
Organizational Knowledge	.0916	.0869	.0869	.0826	.0826	.0826	.0826
Ownership							
Performance Evaluation							
Personnel Selection							
Problem Identification							
Resource Allocation							
Roles-Responsibilities							
Safety Culture	.0030	.0018	.0018	.0034	.0034	.0034	.0034
Technical Knowledge	.3454	.3425	.3425	.3483	.3483	.3483	.3483
Time Urgency							
Training	.0811	.0899	.0899	.0885	.0885	.0885	.0885

design change work process analysis. First, we will analyze each of a large number of ASSET reports in terms of the twenty organizational dimensions and identify those organizational dimensions which appear to play a significant role in the operating events chosen for detailed analysis by the ASSET teams. We also examine other significant operating experience discussed in the ASSET reports. We look for patterns and correlations for each NPP station; we also look for a correlation between these dimensions and the management related recommendations made by the ASSET teams. Based on this experience in analysis of ASSET reports, we will suggest how the influence of organization and management could be made a part of root cause analysis, and the results thereof carried over to other aspects of the plant different from those for which the selected operational events apply.

Second, on the WPAM study, we will examine further the candidate parameter groups for the design change work process to obtain a complete set of the parameter groups. Using these candidate parameter groups, the minimal cut sets will be screened to highlight those minimal cut sets which are influenced by organizational factors. The algorithm needs to be finished to recalculate the frequencies of the highlighted minimal cut sets, that is, achieve the ultimate goal of quantitatively including organizational factors into PRA.

References

1. R. Jacobs and S. Haber, "Organizational Processes and Nuclear Power Plant Safety", *Reliability Engineering and System Safety*, **45** (1994) 75-83.

2. D. A. Shurberg and S. Haber, "Techniques to Assess Organizational Factors: Progress to date," BNL-A-3956-1-7/94, 1994.

3. K. Davoudian, J.S. Wu & G. Apostolakis, Incorporating organizational factors into risk assessment through the analysis of work process, *Reliability Engineering and System Safety*, **45** (1994) 85-105.

4. K. Davoudian, J. S. Wu, and G. Apostolakis, The work process analysis model (WPAM), *Reliability Engineering and System Safety*, **45** (1994) 107-25.

5. J. W. Minarick, J. D. Harris, P. N. Austin, J. W. Cletcher, and E. W. Hagen, "Precursors to Potential Severe Core Damage Accident: 1985, A Status Report", Oak Ridge National Laboratory, NUREG/CR-4674, ORNL/NOAC-232, Vol. 2, 1985.

6. International Atomic Energy Agency, "Report of the ASSET Mission to the Leningrad Nuclear Power Plant in Russia", IAEA/NENS/ASSET/93/R/03, 1993.

7. P. C. Smith and L. M. Kendall, "Retranslation of Expectations: An approach to the Construction of Unambiguous Anchors for Rating Scales", *Journal of Applied Psychology*, 47, pp 149-155, 1963.

8. R. P. Jacobs, D. Kafry and S. Zedeck, "Expectations of Behaviorally Anchored Rating Scales," *Personnel Psychology*, 33, pp. 595-640, 1980.

9. F. Landy and J. L. Farr, *The Measurement of Work Performance*, NY Academic Press, 1982.

10. T. L. Saaty, *The Analytic Hierarchy Process: Planning, Priority Setting, Resource Allocation*, McGraw-Hill International Book Co., New York, 1980.

APPENDIX B

Erosion/Corrosion in Carbon Steel Piping

PHENOMENOLOGICAL MODELING OF EROSION/CORROSION IN PIPING SYSTEMS OF AGING POWER PLANTS

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ABSTRACT

Carbon steels are extensively used in the piping of cooling systems used in fossil-fueled and nuclear power plants. Rapidly flowing pressurized water at high temperatures often causes mechanical damage to the internal surfaces of piping, and if the oxide protective layer is absent, the mechanical damage is particularly severe. The resulting deterioration of the load-carrying capability of pipe sections is described as erosion/corrosion, and usually causes substantially more severe degradation than either erosion or corrosion alone. The problem is economically very significant because unpredictable failures can lead to plant shut-downs, leading to electric power outages. Current predictive models of failures caused by erosion/corrosion are empirically-based, and hence are limited only to the range of available experimental data. Our effort is to develop phenomenological models to extend the range of predictive capabilities, and to provide a fundamental basis for current power plant inspection procedures. The research approach is composed of three parts; (1) A water flow model will be constructed where an advanced fluid flow computer code will be employed to analyze flow characteristics around bends, elbows, straight sections with gradients, and other critical components of the system. (2) A detailed boundary layer model will be developed for mass transport as a result of chemical attack, dissolution of pipe wall constituents, diffusion/convection across the boundary layer to the water main stream, and by mechanical impact of energetic particles in the water (3) An analytical stress analysis model will be developed to predict stress evolution in critical system locations. The effects of pressure and temperature variations will be considered.

1. Problem History and Relevance

In december 1986, a pipe burst occurred in the US Nuclear power plant Surry. This accident was directly caused by wall thinning by loss of steel constituents due to the corrosive effects of flowing water. The metal loss was found to have occurred over a period of nine years of actual operation time. Design stresses were exceeded, and rupture of the pipe ensued. the cost of this accident, including replacement of lost electric power generation was estimated to be about $50 Million . Other catastrophic failures were observed in fossil-fueled power plants, showing the generic nature and severity of the problem in the US and California power industries. The percentage of total leakage failures in power plants is given in Figure (1), where leakage-type failures caused by erosion/corrosion are shown to be quite significant (~22 %).

Physical phenomena associated with erosion/corrosion are complex and varied, and are not generally amenable to mechanistic modeling. However, it is extremely important for plant operators to access knowledge on the thinning rates of piping components. Two levels of models are possible for the description of the dependence of thinning rates on physical and operational variables. These are:

(1) *Empirical models*: where an extensive data base is gathered on the phenomenon under all reasonable circumstances, and a procedure is developed for interpolation of this data base. Note that extrapolations are not easily assured, and thus variations in the operational procedure or environment cannot be predicted.

(2) *Phenomenological Models*: which are based on physical phenomena, although not at the fundamental level. In such models, simplicity of the physical situation can be gained at the expense of sacrificing complex details. These models can be correlated with available laboratory and field data base, but will have a better chance for extrapolation outside the range of available data.

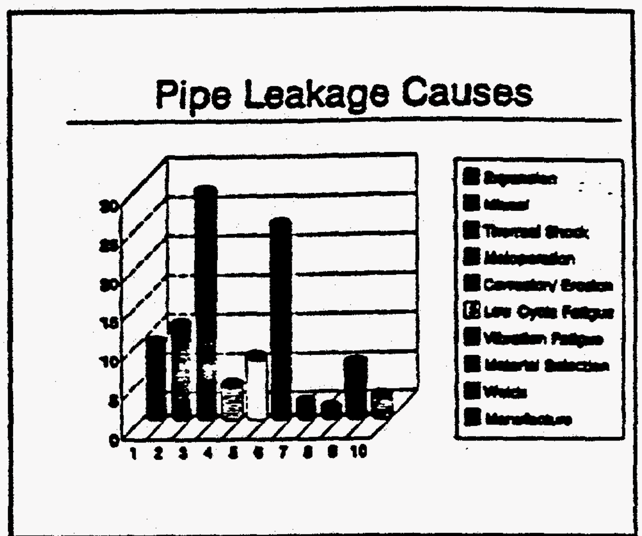

Figure 1
Percentage of pipe leak failures in power plants

In order to avoid the high cost of erosion/corrosion related accidents, one of the previous approaches would be necessary to help in the systematic inspection and evaluation of induced wall thinning. The following factors are identified to influence the corrosion/erosion process: (1) piping material, especially the Cr and Mo content: $h = Cr + Mo$ (%); (2) fluid velocity: w (m/s); (3) piping geometry: with a geometry factor k_c; (4) dissolved oxygen concentration: g (μg/kg); (5) water chemistry: pH = pH value; (6) water temperature: T (K).

The only available models are empirical at this time. Two different attempts have been made to correlate the thinning data. In the US, an EPRI-sponsored project resulted in a computerized data base for corrosion/erosion rates [1]. The work of Chexal and Horowitz culminated in the development of the computer programs CHEC for single phase, and

CHECMATE for two phase flow environments. In addition to a large library of geometry factors for elbows, T-sections, expansion sections, flow reducers, etc., the data is obtained from both laboratory and plant environments.

While the computer codes of Chexal and Horowitz do not give explicit functional forms for the thinning rates, Kastner and Riedle (Kraftwerk Union AG (KWU), Erlangen, Germany) have developed explicit thinning rate equations [2,3]. Their correlation for the surface erosion rate, r, can be put in the following form:

$$\begin{aligned} r &= k_c \times F(w, T, pH, g, h, t) \\ &= 6.25 k_c \{B e^{Nw} \left[1 - 0.175(pH-7)^2\right] 1.8 e^{-0.118g} + 1\} + [f(t)] \end{aligned}$$

where t is operating time in hours, and the following functions are defined:

$$\begin{aligned} B &= -10.5\sqrt{h} - 9.375 \times 10^{-4} T^2 + 0.79T - 132.5 \\ N &= -0.0875h - 1.275 \times 10^{-5} T^2 + 1.078T - 2.15 \quad for \;\; 0 \le h \le 0.5\% \\ N &= (-1.29 \times 10^{-4} T^2 + 0.109T - 22.07)0.154 e^{-1.2h} \quad for \;\; 0.5 \le h \le 5\% \\ f(t) &= \sum_{j=0}^{j=4} C_j t^j \end{aligned}$$

and the C_j are fitting constants:

The cooling water in power plants usually contains some corrosive components (e.g., CO_2), which result in corrosion through the action of two factors. These are the pH value and oxygen concentration in the fluid. A common component of piping systems, iron, reacts with H^+ in the following way: $Fe + 2H^+ \longleftrightarrow Fe^{++} + H_2$

Saturation of ionic iron (Fe^{++}) and formation of a hydrogen gas layer at the liquid-metal interface will tend to keep Fe from diffusing into the water. With addition of oxygen atoms, these two limiting factors will be eased. Oxygen can react with the hydrogen gas to form a gaseous layer, and hydrogen is reduced in this way. Oxygen can further oxidize the ferrous iron hydroxide to form a less soluble ferric hydroxide. This process reduces the ferrous hydroxide in the liquid and concentration of ionic iron becomes unsaturated. Addition of oxygen therefore encourages iron to go to the liquid and accelerates corrosion. Other elements in the liquid (e.g., chloride) and those in the metal (e.g., chromium) are also important. Their contributions vary case by case. If CO_2 is the corrosive agent, the dominant chemical reaction will be: $Fe + CO_2 \longleftrightarrow FeCO_3 + H_2$.

2. Flow Visualization

Initial modeling of fluid flow was performed with Algor CAD Fluid Flow System. It allows quick visualization of various combinations of possible flow geometry. For example, Fig. 2 shows the constant u-velocity contour plot of a 2-D flow through a sudden expansion (Re_D=400). As expected, fully developed parabolic flow detaches from the wall and reattaches downstream, returning to parabolic flow with the decreased center line velocity. Fig. 3 shows the velocity vectors within the enclosed area shown above in Figure 2. Vertical component of the flow near the wall can be seen at the reattachment region. Finite Element Analysis code was used to solves the Navier-Stokes equations for steady-state, Newtonian, incompressible flow.

3. Continuing Work

Under dynamic conditions, corrosion will be enhanced by erosion. When flowing particles are not very energetic, the particles mainly remove the oxide layer between the liquid and the metal. Metal atoms must cross the oxide layer in order for the chemical reaction (corrosion) to occur. Thinning of the oxide layer makes the corrosion process easier by shortening the diffusion distance. An oxide layer is usually formed on metal surface in a corrosive environment, and the metal oxide layer is much weaker than metal itself. When impacted by an energetic particle, removal of the layer is easier that of the metal, and hence the erosion process is enhanced by corrosion. When the flowing particles are very energetic, they can penetrate the oxide layer and damage the metal directly. The first particle may only shatter the oxide layer and the metal. Subsequent particles will remove these embedded fragments. These processes cause severe weight losses of the metal and the protective layer.

Mingling of erosion and corrosion processes makes the problem very difficult to model. Attempts have been made to phenomenologically model the erosion-corrosion processes. Natesan and Liu [4], and Abdulsalam and Stanley [5] focused on chemical reaction aspects, while Nesic and Postlethwaite [6-8] paid more attention to fluid patterns. Zeisel and Durst [9] developed a relatively complete model.

We plan to model mass transfer similar to the Zersel-Durst model, where corrosion processes are described by two mixing equations: one describing corrosion of bare metals [10], and the other describing corrosion controlled by diffusion across the protective layer [9]. Finnie's equation [11] are combined with Sundarajan-Shewmon's equation [12] to describe erosion processes. Finnie's equations account for erosion caused by the parallel component of the impacting particle's velocity, while the Sundarajan-Shewmon's equation accounts for contributions from perpendicular component of the velocity. Erosion and corrosion processes are correlated through the protective layer. The flow field will be obtained by solving Navier-Stokes equation, which is solved using the Patanker-Spalding numerical scheme [13], as in the widely used PHONENICS code [14]. Different geometries, elbow, T-section, expansion/contraction segment, etc. will be studied. Solutions of all these components will be combined to give a universal solution for any complex geometry.

Mass transfer across the boundary layer will influence fluid behavior through boundary conditions. In return, fluid behavior will determine mass transfer speed by changing concentration of concerned elements in the boundary layer. By coupling these two, one can solve for removal/deposition rates on metal surfaces of piping systems. Material losses will be used in stress analysis formulation to determine lifetime of piping systems.

References

(1) B. Chexal and J. Horowitz, "Flow assisted corrosion in carbon steel piping - parameters and influences," *Symposium on Environmental Degradation of Materials in Nuclear Power Plant Systems - Water Reactors*," Jeky;; Island, DA, Aug 1989.

(2) W. Kastner, M. Erve, N. Henzel and B. Stellwag, *Nucl. Engr. Design,* 119(1990)431.

(3) N. Henzel, W. Kastener and B. Stellwag, "Erosion/ Corrosion in Power Plants under single and two-phase flow conditions," *American Power Conference, 50th Annual Meeting*, Chicago, April 1988.

(4) K. Natesan and Y. Liu, "Erosion-Corrosion of Materials at Elevated Temperatures", *Mater. Sci. Engr. A* 121(1989)571-580

(5) M. Abdulsalam and T. Stanley, "Steady-State Model for Erosion-Corrosion of Feedwater Piping", *Corrosion-NACE* 48(1992)587-593

(6) S. Nesic and J. Postlethwaite, "Relationship Between the Structure of Distributed Flow and Erosion-Corrosion", *Corrosion-NACE* 46(1991)874-880

(7) S. Nesic and J. Postlethwaite, "A Predictive Model for Localized Erosion-Corrosion", *Corrosion-NACE* 47(1991)582-589

(8) S. Nesic and J. Postlethwaite, "Hydrodynamics of Disturbed Flow and Erosion-Corrosion. Part I - Single-phase Flow Study", *Canadian J. Chem. Engr.* 69(1991)698-703

(9) H. Zeisel and F. Durst, "Computations of Erosion-Corrosion Processes in Separated Two-phase Flows, in *Symposium on Corrosion,* Las Vegas, April 1990

(10) C. De Waard and D. Milliams, "Carbonic Acid Corrosion of Steel", *Corrosion-NACE* 31(1975)177

(11) I. Finie, "Erosion Behavior of Metals", *Wear* 3(1960)87-103

(12) G. Sundarajan and P. Shewmon, "A New Model for the Erosion of Metals at Normal Incidence", *Wear* 84(1983)237-258

(13) S. Patanker and D. Spalding, "A Calculation Procedure for Heat, Mass and Moment Transfer in Three-Dimensional Parabolic Flows", *Int. J. Heat and Mass Transfer* 15(1972)1787-1806

(14) S. Varnas, "The PHOENICS Model of a Low-Current Plasma Arc", CHAM Report, May 1989, pp304-342

Figures

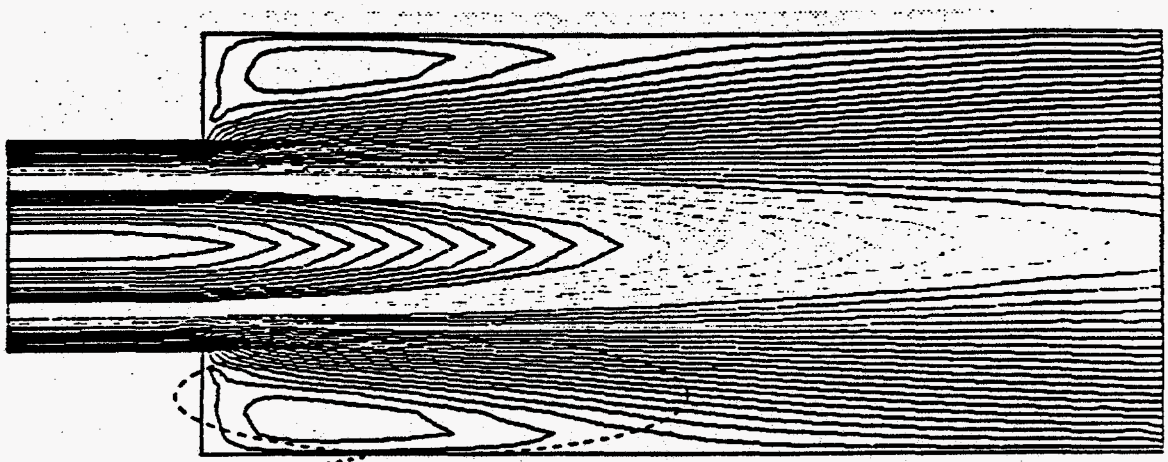

Figure 2. Constant horizontal velocity component contour plot.

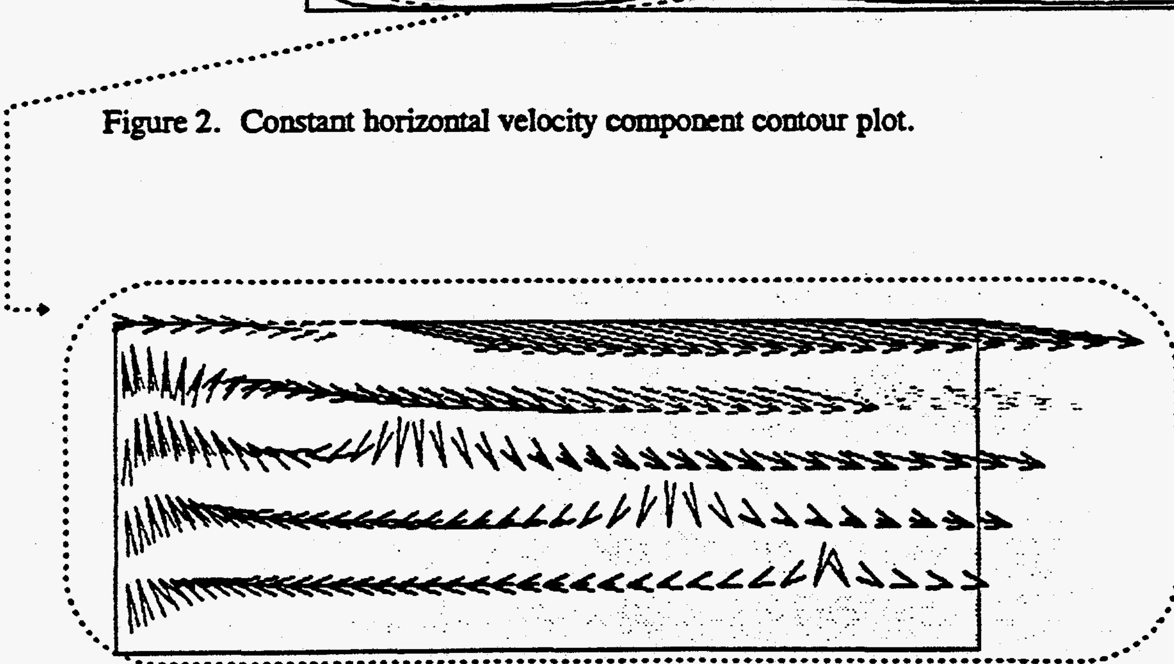

Figure 3. Velocity vector plot.

Use Of Artificial Intelligence In Severe Accident Diagnosis For PWRs

Zheng Wu, D. Okrent and W.E.Kastenberg

Introduction

Severe accident management has been recognized as essential element to enhance nuclear power plant safety and large effort has been devoted on related issues [1,2,3]. Silverman and Klopp used a neural network-based expert system for the purpose of severe accident management [4]. The system was used to predict parameters important for the accident management during loss of coolant accidents (LOCA), e.g., the time available to core support plate and reactor vessel failure and time remaining until recovery actions were too late to prevent core damage. Guarro et al. have proposed an accident management advisor system (AMAS) as a decision aid for interpreting the instrument information and managing accident conditions in nuclear power plant [5]. The modified logical flowgraph methodology was used to interpret the instrument readings to derive the plant parameters and the plant status was determined through the Baysian Belief Network (BBN). Recently, artificial neural networks have been used for the BWR ATWS transients pattern recognition [6]. Core power, vessel pressure, number of open safety relief valves, and suppression pool temperature have been chosen to define the four patterns for the training of the networks. The results show that the neural networks can successfully retrieve the patterns even with large random noise and partial loss of the input information. As indicated by the authors, this kind of error resistance might be useful in severe accident situations where the instrumentation may not be available because of the harsh environment. Neural networks have also been used in many other areas of nuclear power plants, including transient diagnostics, sensor validation, plant-wide monitoring, check valve monitoring, vibration analysis [7]. In most of these applications, multi-layer, feed-forward backpropagation neural networks are used. A dynamic node architecture scheme for neural network training was proposed by Basu and Bartlett to optimize the neural network structure [8]. For a three layer backpropagation neural network, while the neuron number of input and output layer is usually determined by the diagnostic problem, the number of neurons for the hidden layer is added or deleted dynamically during the training until the optimal criteria are met with a certain number of hidden neurons. Neural networks with schemes other than backpropagation have also been applied to fault diagnosis. Specht's probabilistic neural networks [9] were modified and used to integrate with influence diagrams for power plant monitoring and diagnostics [10]. Marseguerra and Zio proposed a stochastic neural network (boltzmann machine) and used it to diagnose a pipe break in a simulated auxiliary feedwater system [11].

It is important for the personnel in charge of accident management during the accident to understand the status of the power plant and the progression trend of the accident in order to evaluate and implement effective prevention or mitigation strategies. While there are lots of efforts on diagnostic systems for accidents before core damage [12,13,14], there is a general lack of diagnosis methodologies for severe accidents where the core would undergo severe damage and accidents might progress beyond vessel breach.

The objective of the proposed work is to build a diagnostic expert system which would monitor the progression of the severe accident and provide necessary plant status information to assist the plant staff in accident management during the accident. Station blackout type accident would be used as the case study. The current phase of research focus is on distinguishing different primary system failure modes and following the accident transient before and up to vessel breach.

Station Blackout Accident

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Preliminary analysis indicates that primary system pressure undergoes more or less distinct dynamic responses with various failure modes during station blackout [3,15]. After the initial transient period, there is a decrease of the primary system pressure because of energy transfer to the secondary system before the dryout of steam generators and possible energy loss through the primary system opening (e.g. RCP seal leaking). After the dryout of the steam generators, the primary system pressure will increase to the PORV setpoint when the PORVs start cycling. The primary system pressure will fluctuate accordingly. For the case of large RCP seal failure, the pressure drop might be so large that it will no longer go up to the PORV setpoint. Depending on different primary failure modes, there might be a different primary system pressure history. In addition, there are other sensor readings which could be used to distinguish different failure modes [16]. For example, when ISGTR occurs, the pressure, temperature, and radiation level of the secondary side of the steam generator will normally increase. In summary, the combination of the primary system pressure history and other instrumentation indications could be used to diagnose various primary system failure modes during station blackout accidents.

Methodology

There are basically two fundamental problems for the diagnostic task, i.e., detection of a failure and identification of the failure. The detection process would uncover a possible primary system failure from abnormal sensor readings and the identification process would determine which failure actually occurs from the time series of the signals. It is important to distinguish these two steps of the diagnosis because it usually takes more data to identify what exactly happens after the detection. In our case, it is rather easy and quick to tell that the reactor vessel has been breached or hot leg/surge line fails from the sudden large decrease of the primary system pressure, whereas it is not so easy to see right away what happens for some other failures. For the case of PORV Stuck Open, the failure could not be detected for some sustained period of time until the sensor readings show substantial abnormality. The same situation applies to ISGTR without radiation reading of the secondary side of steam generators. Large uncertainties have to be considered during the accident progression. First, there is uncertainty regarding which failure occurs. For example, during a station blackout accident, the auxiliary feedwater system may either be in operation or fail at the initiation of the accident. After uncovery of the top of the active fuel, there might be failure of steam generator tubes, failure of the hot leg/surge line, or a stuck open power operated relief valve. Second, there is uncertainty regarding when the failure occurs. The timing of each possible failure is hard to determine. It is not possible to specify exactly when the power operated relief valve would be stuck open under abnormal operation conditions. Third, there is uncertainty regarding severity of the failure. For example, the size of the reactor coolant pump seal leak is not known and one is unable to determine this beforehand. And fourth, there is uncertainty regarding whether further failures occur. There might be multiple failures during the accident progression.

The proposed framework for the diagnosis is a combination of an expert system and artificial neural networks. The rule-based expert system is used for the basic plant overall monitoring and diagnosis. Specific neural networks will be initiated by the expert system to determine the patterns of special events during the accident progression. The diagnosis expert system will be used to distinguish different failures, severity of the failure and further failures based on the available instrumentation reading.

The expert system will be used to monitor the progression from the start of the accidents. The initial accident conditions and major change of plant status will be recorded and displayed. This system will also determine, on detecting some sensor reading change suggesting potential failures, when the diagnostic neural networks should be initiated for failure detection and identification. The diagnostic results from neural networks will be compared, if possible, with the results from the expert system. The difference between the actual sensor reading during the accidents and the MAAP simulation will be shown in order to justify the use of neural networks and accommodate large uncertainty. MAAP simulation codes could generate the primary system pressure history and other indications, e.g., secondary side pressure and temperature, containment temperature and pressure, radiation levels. The results will also provide bounding values and timing information of the failures. Thus, MAAP run results will be used to gain qualitative, semi-qualitative, and quantitative instrument reading change patterns to form the knowledge base of the expert system. Other scientific knowledge and engineering judgment will also be incorporated into the knowledge base.

The transient data from MAAP runs can be used to train the neural networks to distinguish various failure patterns. Since the timing of the failure is uncertain, the results of use of neural networks for diagnosis purpose must be treated cautiously since the neural network training highly depends on the scenarios, even though the neural networks retain some capability of resistance to signal noise. The training of the neural networks needs to be studied in view of several uncertainties, including variability in initial conditions (timing, size of leak, etc.), differences between MAAP and actual performance, changing configuration after initiation of MAAP, misleading sensor signals, etc. These and other considerations will be examined in order to use the neural networks to best advantage. These multiple sub scenario conditions suggest that for each principal scenario, it will be useful to have a few MAAP runs appropriately selected.

Two back propagation neural networks are designed for diagnostic purpose. One is for detection of possible primary system failure (Detection Neural Network) and the other is for failure identification (Identification Neural Network). The data to be used for the training is tested progressively to maximize the best possible results. The data used for neural network training will be increased time step by time step into the accident until the test results would not be better. After the determination of that training data which is shown to be effective, the two neural networks are constructed and tested.

RESULTS

Dr. Dave Dion of PG&E has conducted a large number of MAAP simulations for station blackout accident scenarios. These results were used as the first effort to formulate the methodology or ways of diagnosis of the primary system failures before vessel breach during station blackout conditions. Other simulation runs might be needed after the analysis of these results. The plant condition before the accidents is assumed to be at the normal full power operation. The station blackout cases include cases when the auxiliary feedwater system (AFWS) is assumed to be available or unavailable at the start of the accident. For the small reactor coolant (RCP) pump seal leak, further primary failures are assumed to be possible before vessel breach.

Five out of thirty six accident scenarios are chosen to be the reference data, representing Vessel Breach, ISGTR (1 tube), ISGTR (10 tubes), Hot Line/Surge Line Failure, PORV Stuck Open cases respectively. Sensor readings of primary system pressure, steam generator pressure, steam generator temperature, containment pressure, and containment temperature were used for diagnosis. Figure 1 shows the basic structure of the diagnostic neural networks. It is a three-layer, feed-forward, backpropagation neural network. The MAAP data is used to train the neural networks which are then tested against all the other scenarios. To some extent, this would guarantee the generality of the neural networks to detect and identify the faults under various conditions.

To evaluate the data adequacy for diagnosis and determine the data for neural network training for failure detecting and identification, training data was taken from the start of the failure and was progressively increased (every 20 second step). The input neurons are determined according to the amount of data for training. There are two output neurons.

Fourteen groups of data (3x20s, 4x20s, 5x20s, 6x20s, 7x20s, 8x20s, 9x20s, 10x20s, 13x20s, 14x20s, 15x20s, 16x20s, 17x20s, and 20x20s) from AS1 (Vessel Breach), AS2a (ISGTR), AS2b(ISGTR), AS3a(H/S Failure), AS4a(PORV Failure) were used for the training. Sensor data is normalized between 0.0 and 1.0. For the network recall process, any data less than 0.25 is treated as 0, any data above 0.75 is treated as 1.0, any data between 0.25 and 0.5 is treated as likely 0, any data between 0.5 and 0.75 is treated as likely 1. The mapping scheme used for testing is shown in table 1.

Table 2 to Table 5 show the test results. Case 1 is the AFWS initially working and no RCP Seal Failure case. Case 2 is the AFWS initially working with RCP Seal Failure case. Case 3 is the AFWS initially Non-working and no RCP Seal Failure case. Case 4 is the AFWS initially Non-working with RCP Seal Failure case.

With the increasing of the data into the accident, the neural networks recall ability converges to a certain level where test results are no longer improved with more data. From the results, the converged time data for VB and Hot Leg/Surge Line is 3x20s. The converged time data for ISGTR and PORV Stuck Open is 15x20s.

Finally, the Detection Neural Networks and Identification Neural Network were constructed. For each of these two neural networks, there are 80 input neurons representing 15 time step of data of 20 second each. There are three output neurons with following mapping scheme shown in table 6.

Table 1 Output mapping scheme for testing for data evaluation

Output Neuron 1	Output Neuron 2	Mapping Case
0.0 - 0.25	0.0 - 0.25	Vessel Breach
0.25 - 0.5	0.0 - 0.25	likely Vessel Breach
0.0 - 0.25	0.25 - 0.5	likely Vessel Breach
0.25 - 0.5	0.25 - 0.5	likely Vessel Breach
0.0 - 0.25	0.75 - 1.0	ISGTR
0.25 - 0.5	0.75 - 1.0	likely ISGTR
0.0 - 0.25	0.5 - 0.75	likely ISGTR
0.25 - 0.5	0.5 - 0.75	likely ISGTR
0.75 - 1.0	0.0 - 0.25	H/S Failure
0.75 - 1.0	0.25 - 0.5	likely H/S Failure
0.5 - 0.75	0.0 - 0.25	likely H/S Failure
0.5 - 0.75	0.25 - 0.5	likely H/S Failure
0.75 - 1.0	0.75 - 1.0	PORV Failure
0.5 - 0.75	0.75 - 1.0	likely PORV Failure
0.75 - 1.0	0.5 - 0.75	likely PORV Failure
0.5 - 0.75	0.5 - 0.75	likely PORV Failure

Table 2 Test results for vessel breach identification

time data s	3x20	4x20	5x20	6x20	7x20	8x20	9x20	10x20	15x20	16x20	17x20	20x20
case1												
case2	++	++	++	++	++	++	++	++	++	++	++	++
case3	+	++	++	++	++	++	++	++	++	++	++	++
case4	+	++	++	++	++	++	++	++	++	++	++	++

Table 3 Test results for hot leg/surge line identification

time data s	3x20	4x20	5x20	6x20	7x20	8x20	9x20	10x20	15x20	16x20	17x20	20x20
case1	++	++	++	++	++	++	++	++	++	++	++	++
case2	++	++	++	++	++	++	++	++	++	++	++	++
case3	++	++	++	++	++	++	++	++	++	++	++	++
case4	++	++	++	++	++	++	++	++	++	++	++	++

Table 4 Test results for ISGTR identification

time data s	3x20	4x20	5x20	6x20	7x20	8x20	9x20	10x20	15x20	16x20	17x20	20x20
case1	++	++	++	++	++	++	++	++	++	++	++	++
case2	++	++	++	++	++	++	++	++	++	++	++	++
case3	++	+	-PORV	-PORV	-PORV	-PORV	+	+	+	+	+	+
case4	++	++	++	++	+	+	+	+	+	+	+	+

Table 5 Test results for PORV Stuck Open identification

time data s	3x20	4x20	5x20	6x20	7x20	8x20	9x20	10x20	15x20	16x20	17x20	20x20
case1	++	++	++	++	++	++	++	++	++	++	++	++
case2	-ISGTR	-ISGTR	-ISGTR	-ISGTR	-ISGTR	+	+	+	++	++	++	++
case3	+	-ISGTR	+	+	+	+	+	+	++	++	++	++
case4	+	++	++	++	++	++	++	++	++	++	++	++

Note: ++ positive + likely positive - likely negative -- negative

Table 6 Output mapping scheme for Detection and Identification Neural Networks training

CASE NAME	Output Neural 1	Output Neural 2	Output Neural 3
No Failure	0.1	0.1	0.1
Vessel Breach	0.9	0.1	0.1
ISGTR	0.9	0.1	0.9
H/S Failure	0.9	0.9	0.1
PORV Failure	0.9	0.9	0.9

The training samples included data of no failure case. The training data for Vessel Breach and H/S Failure ranged from 3x20s to 8x20s into the accident respectively. The training data for ISGTR and PORV Stuck Open ranged from 10x20s to 15x20s. Since the number of input neurons is fixed at 80 or 15 time steps of 20 seconds, most training data also covers a portion of the no failure case. Figure 2.1 and Figure 2.2 show the convergence of the training of the neural networks. The networks are tested using test data. VB and H/S Failure can be detected 20 seconds into the accident and identified 30 seconds into the accidents which would be confirmed for several more time steps. The ISGTR and PORV Stuck Open accidents can be detected 160 seconds into the accident and be identified 180 seconds into the accidents. Test of no failure cases was successful. When 30% of random noise was added to the training data, the Detection Neural Network can still correctly detect various failure. Vessel Breach or H/S Failure can be correctly identified by Identification Neural Network with 25% random noise added to the training data. PORV Stuck Open and ISGTR can be correctly identified with 10% random noise.

The Detection Neural Network and Identification Neural Network are initiated during the cycling of the PORV period and long before the start of primary system failure. Every 20 seconds, a new time step data could be fed in and the oldest time step data thrown out. If the situation is classified as No Failure by the Detection Neural Network, the process would continue. If some failure is detected by the Detection Neural Network, Identification Neural Network would be initiated to identify which failure occur. Further data input would be used for diagnosis confirmation.

SUMMARY

Neural network techniques have been successfully used to detect and identify primary system failure during station blackout. Among the things accomplished, the use of neural networks to evaluate data adequacy and sufficiency is a novel application of such technique. The same technique will be used to construct neural networks for RCP Seal Failure cases. Multiple failure cases, e.g., vessel breach after ISGTR, will also be considered. The diagnosis of ISGTR would be more effective if steam generator radiation level can be used in the neural networks. Even though we give a scale for some sort of uncertainty assessment, a more thorough uncertainty analysis would be desirable, if possible. Expert systems construction and the combination of the neural networks with such systems are the remaining tasks.

References

[1] USNRC, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plant. NUREG-1150, April, 1989

[2] Hanson, D.J. et al., Accident Management Information Needs. NUREG/CR-5513, April, 1990

[3] Brownson, D.A. et al., Intentional Depressurization Accident Management Strategy for Pressurized Water reactors. NUREG/CR-5937, April, 1993

[4] Silverman, E.B. and Klopp, G.T., " Neural Network-Based Expert System For Severe Accident management", EPRI Proceedings

[5] Guarro, S. et al., " A Decision Aid for Interpreting Instrument Information and managing Accident Conditions in Nuclear Power Plant ", OECD/CSNI Specialists Meeting On Instrumentation To Management Severe Accidents, Cologne, Germany, March 16-17, 1992

[6] Xing, L. and Okrent, D., "Application of Neural Networks in Accident Management", Proceedings of PSA International Topical Meeting, Clearwater Beach, Florida, 1993

[7] Uhrig, R.E., "Potential Use of Neural Networks in Nuclear Power Plants". IEEE International Conference on Neural Networks, 1993

[8] Basu, A. and Bartlett, E. B., "Detecting Faults in a Nuclear Power Plant by using Dynamic Node Architecture Artificial Neural Networks". Nuclear Science and Engineering, No. 116, 313-325, 1994

[9] Marseguerra, M. and Zio, E., "Fault Diagnosis via Neural Networks: the Boltzmann Machine". Nuclear Science and Engineering, No. 117, 194-200, 1994

[10] Specht, D.F., "Probabilistic Neural Networks". Neural Networks, Vol. 3, 1990

[11] Agogino, A. et al., "Integrating Neural Networks with Influence Diagrams for Power Plant Monitoring and Diagnostics". IEEE International Conference on Neural Networks, 1993

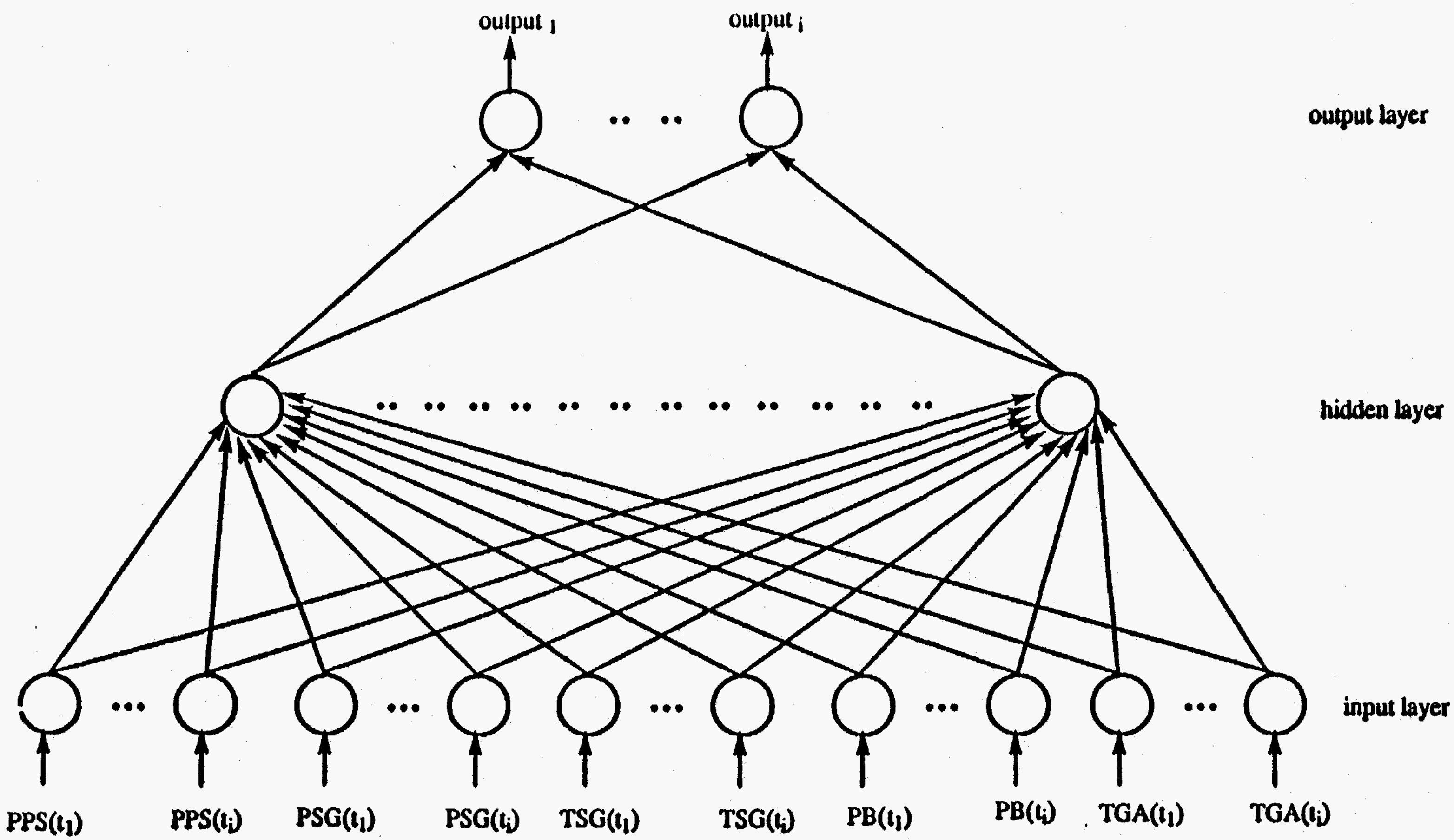
[12] Guarro, S.B., "Diagnostic Models for Engineering Process Management: A Critical Review of Objectives, Constraints and Applicable Tools". Reliability Engineering and System Safety, NO.30, 1990

[13] Kim, I.S. et al., "A Model-based Approach to On-line Process Disturbance Management: The Models". Reliability Engineering and System Safety, NO.29, 1990

[14] Yang, J.O., Chang, S.H., "A Diagnostic Expert System for the Nuclear Power Plant Based on the Hybrid Knowledge Approach". IEEE Transactions on Nuclear Science, Vol.36, NO.6, 1989

[15] Knudson, D.L., Dobbe, C.A., Assessment of the Potential for High Pressure Melt Ejection resulting from a Surry Station Blackout Transient. NUREG/CR-5949, May, 1993.

[16] Oehlberg, R.N. et al., "Practical Considerations in Accident Management". Proceedings of International Meeting on safety of Thermal Reactors, Oregon, July 21-25, 1991



PPS: Primary System Pressure
PSG: Steam Generator Pressure
TSG: Steam Generator Temperature
PB: Containment Pressure
TGA: Containment Upper Plenum Temperature

Figure 1 A three layer back propagation neural network

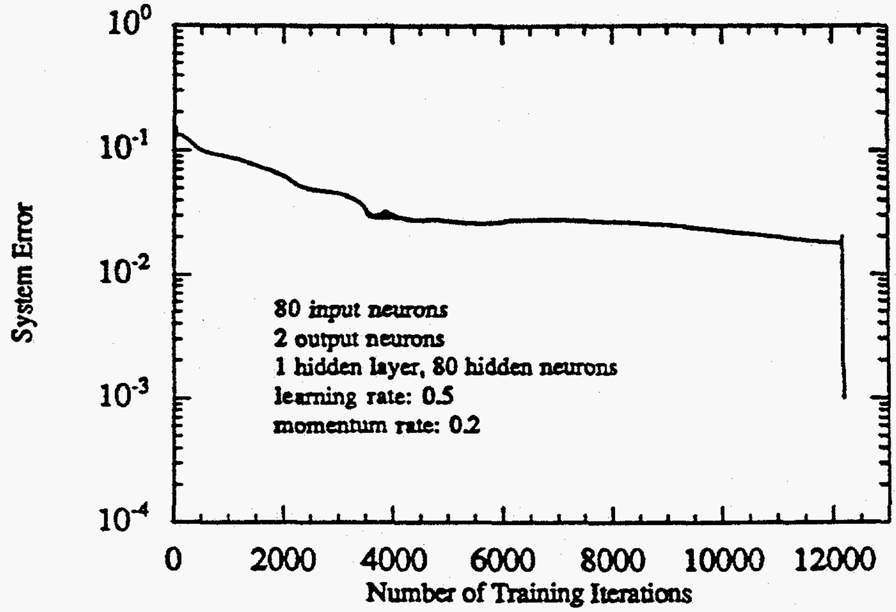


Figure 2.1 Training convergence for Detection Neural Network

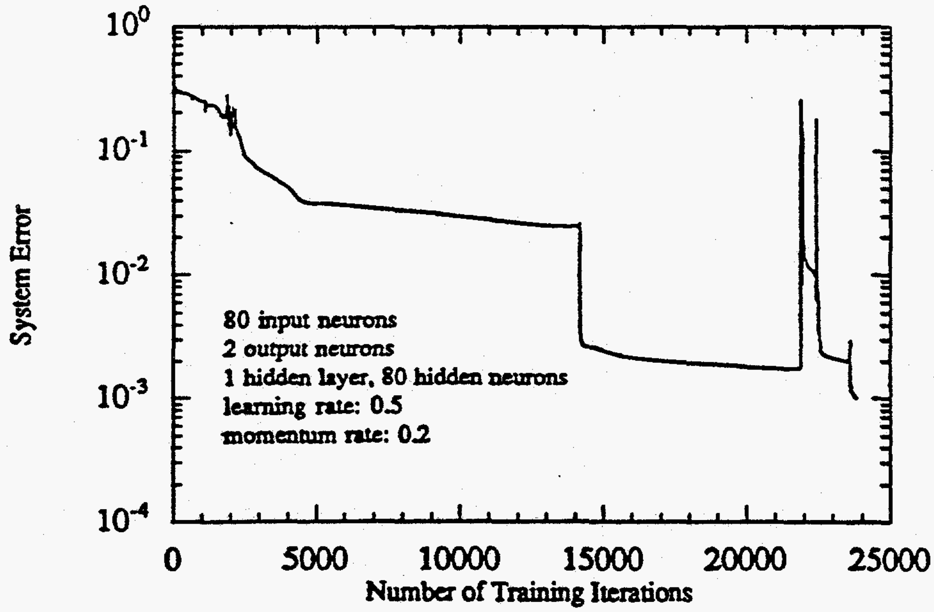


Figure 2.2 Training convergence for Identification Neural Network

APPENDIX C

Use of Artificial Intelligence in Severe Accident Diagnosis for PWRs

Use Of Artificial Intelligence In Severe Accident Diagnosis For PWRs

Zheng Wu and D. Okrent

Introduction

Severe accident management has been recognized as an essential element to enhance nuclear power plant safety, and a large effort has been devoted on related issues [1,2,3]. Silverman and Klopp used a neural network-based expert system for the purpose of severe accident management [4]. The system was used to predict parameters important for accident management during loss of coolant accidents (LOCA), e.g., the time available to core support plate and reactor vessel failure and time remaining until recovery actions were too late to prevent core damage. Guarro et al. have proposed an accident management advisor system (AMAS) as a decision aid for interpreting the instrument information and managing accident conditions in a nuclear power plant [5]. The modified logical flowgraph methodology was used to interpret the instrument readings to derive the plant parameters, and the plant status was determined through the Baysian Belief Network (BBN). Recently, artificial neural networks have been used for the BWR ATWS transients pattern recognition [6]. Core power, vessel pressure, number of open safety relief valves, and suppression pool temperature have been chosen to define the four patterns for the training of the networks. The results show that the neural networks can successfully retrieve the patterns even with large random noise and partial loss of the input information. As indicated by the authors, this kind of error resistance might be useful in severe accident situations where the instrumentation may not be available because of the harsh environment. Neural networks have also been used in many other areas of nuclear power plants, including transient diagnostics, sensor validation, plant-wide monitoring, check valve monitoring, vibration analysis [7]. In most of these applications, multi-layer, feed-forward backpropagation neural networks are used. A dynamic node architecture scheme for neural network training was proposed by Basu and Bartlett to optimize the neural network structure [8]. For a three layer backpropagation neural network, while the neuron number of input and output layers is usually determined by the diagnostic problem, the number of neurons for the hidden layer is added or deleted dynamically during the training until the optimal criteria are met with a certain number of hidden neurons. Neural networks with schemes other than backpropagation have also been applied to fault diagnosis. Specht's probabilistic neural networks [9] were modified and used to integrate with influence diagrams for power plant monitoring and diagnostics [10]. Marseguerra and Zio proposed a stochastic neural network (boltzmann machine) and used it to diagnose a pipe break in a simulated auxiliary feedwater system [11].

It is important for the personnel in charge of accident management during the accident to understand the status of the power plant and the progression trend of the accident in order to evaluate and implement effective prevention or mitigation strategies. While there are lots of efforts on diagnostic systems for accidents before core damage [12,13,14], there is a general lack of diagnosis methodologies for severe accidents where the core would undergo severe damage and accidents might progress beyond vessel breach.

The objective of the proposed work is to build a prototype severe accident diagnostic system which would monitor the progression of the severe accident and provide necessary plant status information to assist the plant staff in accident management during the accident. The station blackout type accident would be used as the case study. The current phase of research focus is on distinguishing different primary system failure modes and following the accident transient before and up to vessel breach.

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MAAP Simulation

MAAP simulation runs have been conducted by Dr. Dave Dion of PG&E for the following thirty six (36) accident scenarios (ASs). These results were used as the first effort to formulate the methodology or ways of diagnosis of the primary system failures before vessel breach during station blackout conditions. Other simulation runs might be needed to implement the prototype diagnostic system.

The plant condition before the accidents is assumed to be at the normal full power operation. For the station blackout cases AS1 to AS8, the auxiliary feedwater system (AFWS) is assumed to be available at the start of the accident. The AFWS will be in operation for four hours before the station battery is depleted. For the station blackout cases AS9 to AS16, the auxiliary feedwater system (AFWS) is assumed to be unavailable at the start of the accident. For small reactor coolant (RCP) pump seal leak, further primary failures are possible before vessel breach.

1. AS1: SBO(1) - AFWS FAILS - SGs DRYOUT - UTAF - VB - AP

This is the base case when the AFWS is initially available. After four hours, the AFWS would unavailable and the steam generators would start drying out. At the same time, the primary system would start boiloff through the cycling of the pressurizer power operated relief valves and safety relief valves. The loss of primary coolant would uncover the core. The core would then be oxidized, damaged, and relocated. The vessel would eventually fail due to the thermal attack of the relocated melting core. No other primary system failure is assumed before vessel breach. The accident would continue to progress after vessel breach. The simulation results for this case would provide the timing of the events, system pressure and temperature history, etc.

2. AS2a: SBO(1) - AFWS FAILS - SGs DRYOUT - UTAF - ISGTR (1 tube ruptured at t) - VB - AP
3. AS2b: SBO(1) - AFWS FAILS - SGs DRYOUT - UTAF - ISGTR (10 tubes ruptured at t) - VB - AP
4. AS2c: SBO(1) - AFWS FAILS - SGs DRYOUT - UTAF - ISGTR (1 tube ruptured at t') - VB - AP

For the cases AS2a,AS2b,and AS2c, the AFWS is assumed to be available for four hours. After core uncovery, the steam generator tubes would be exposed to high temperature gases under high pressure due to natural circulation. Temperature-induced steam generator tube rupture is assumed before vessel breach.

The number of ruptured tubes is uncertain under accident conditions and the number of one or ten is assumed. The results of these two simulations would be compared. The pressure and temperature change of the secondary side of the steam generator might be too slight to be noticeable.

The timing of the SGTR is uncertain. It is important to know the implication of failure at different times. Steam generator tube rupture at time t and t' is assumed. Simulation results would be compared to see if the change of system parameters would follow similar patterns.

For the purpose of diagnosis, the use of secondary side steam line radiation alarms would warrant the identification of the SGTR. Still, the simulation run results would help finding other indications for diagnosis.

5. AS3a: SBO(1) - AFWS FAILS - SGs DRYOUT - UTAF - H/S FAILURE (at t)- VB - AP
6. AS3b: SBO(1) - AFWS FAILS - SGs DRYOUT - UTAF - H/S FAILURE (at t')- VB - AP

For the cases AS3a and AS3b, the AFWS is assumed to be available for four hours. After core uncovery, the hot leg and surge line would be exposed to high temperature gases under high pressure due to natural circulation and release of gases through the pressurizer PORVs or SRVs. Temperature-induced hot leg or surge line failure is assumed before vessel breach.

The timing of the H/S failure is uncertain. Time t and t' is assumed as the H/S failure time. Simulation results would be compared to see if the change of system parameters would follow similar patterns.

7. AS4a: SBO(1) - AFWS FAILS - SGs DRYOUT - UTAF - PORV/SRV STUCK OPEN (at t)- VB- AP
8. AS4b: SBO(1) - AFWS FAILS - SGs DRYOUT - UTAF - PORV/SRV STUCK OPEN (at t')- VB-AP

For the cases AS4a and AS4b, the AFWS is assumed to be available for four hours. After core uncovery, the PORVs and SRVs would be operated at above normal conditions. PORV or SRV Stuck Open is assumed before vessel breach.

The timing of the PORV/SRV Stuck Open is uncertain. Time t and t' is assumed as the failure time. Simulation results would be compared to see if the change of system parameters would follow similar patterns.

9. AS5a: SBO(1) - RCP SEAL FAILS (PSL) - AFWS FAILS - UTAF - VB- AP
10. AS5b: SBO(1) - RCP SEAL FAILS (PSLA) - AFWS FAILS - UTAF - VB- AP
11. AS5c: SBO(1) - RCP SEAL FAILS (PSLB) - AFWS FAILS - UTAF - VB- AP
12. AS5d: SBO(1) - RCP SEAL FAILS (PSLB) - AFWS FAILS - UTAF - VB- AP

For the cases AS5a, AS5b, AS5c, and AS5d, the AFWS is assumed to be available for four hours. Since there is no cooling for the reactor coolant pump (RCP) seals from the start of the accident, RCP Seal Failure is assumed. According to DCPP IPE, four modes of seal failure are assumed, i.e., PSL, PSLA, PSLB, AND PSLC. All the failure modes have an initial leak size of 21 gpm per RCP. PSL would be keep the same leak size during the accident progression. PSLA and PSLB would develop to 250 and 480 gpm/RCP respectively after one and half hours. PSLC would develop to 155 gpm/RCP after two and half hours. PSLA is the most probable mode and PSLB is the least. Simulation results would be compared to see if these modes are distinguishable.

13. AS6a: SBO(1)-RCP SEAL FAILS(PSL)-AFWS FAILS-UTAF-ISGTR (1 tube ruptured at t)-VB- AP
14. AS6b: SBO(1)-RCP SEAL FAILS(PSL)-AFWS FAILS-UTAF-ISGTR (1 tube ruptured at t')-VB-AP

For the cases AS6a and AS6b, the AFWS is assumed to be available for four hours. RCP seal failure of PSL mode is assumed. Since the seal leak is small, the primary pressure drop is small. After core uncovery and before vessel breach, temperature-induced steam generator tube rupture is assumed. One tube is assumed to be ruptured. Since the tube rupture timing is uncertain, the rupture is assumed at t and t'.

15. AS7a: SBO(1) - RCP SEAL FAILS (PSL) - AFWS FAILS - UTAF- H/S FAILURE (at t) - VB - AP
16. AS7b: SBO(1) - RCP SEAL FAILS (PSL) - AFWS FAILS - UTAF- H/S FAILURE (at t') - VB -AP

For the cases AS7a and AS7b, the AFWS is assumed to be available for four hours. RCP seal failure of PSL mode is assumed. Temperature-induced hot leg or surge line failure is assumed. The failure timing is uncertain and is assumed at t and t'.

17. AS8a: SBO(1) - RCP SEAL FAILS (PSL) - AFWS FAILS - UTAF - PORV/SRV STUCK OPEN (at t)-VB-AP
18. AS8b: SBO(1) - RCP SEAL FAILS (PSL) - AFWS FAILS - UTAF -PORV/SRV STUCK OPEN (at t')-VB-AP

For the cases AS8a and AS8b, the AFWS is assumed to be available for four hours. RCP seal failure of PSL mode is assumed. After core uncovery, PORV or SRV Stuck Open before vessel breach is assumed. The failure timing is uncertain and is assumed at t and t'.

19. AS9: SBO(2) - SGs DRYOUT - UTAF - VB - AP

This is the base case when the AFWS is initially not available. The steam generators would start drying out. At the same time, the primary system would start boiloff through the cycling of the pressurizer power operated relief valves and safety relief valves. The loss of primary coolant would uncover the core. The core would then be oxidized, damaged, and relocated. The vessel would eventually fail due to the thermal attack of the relocated melting core. No other primary system failure is assumed before vessel breach. The accident would continue to progress after vessel breach. The simulation results for this case would provide the timing of the events, system pressure and temperature history, etc.

20. AS10a: SBO(2) - SGs DRYOUT - UTAF - ISGTR (1 tube ruptured at t) - VB - AP
21. AS10b: SBO(2) - SGs DRYOUT - UTAF - ISGTR (10 tubes ruptured at t) - VB - AP
22. AS10c: SBO(2) - SGs DRYOUT - UTAF - ISGTR (1 tube ruptured at t') - VB - AP

For the cases AS10a,AS10b,and AS10c, the AFWS is assumed to be unavailable from the start of the accident. Temperature-induced steam generator tube rupture is assumed before vessel breach. The number of ruptured tubes is uncertain under accident conditions and the number of one or ten is assumed. The timing of the SGTR is uncertain. Steam generator tube rupture at time t and t' is assumed.

23. AS11a: SBO(2) - SGs DRYOUT - UTAF - H/S FAILURE (at t) - VB - AP
24. AS11b: SBO(2) - SGs DRYOUT - UTAF - H/S FAILURE (at t') - VB - AP

For the cases AS11a and AS11b, the AFWS is assumed to be unavailable from the start of the accident. Temperature-induced hot leg or surge line failure is assumed before vessel breach. The timing of the H/S failure is uncertain. Time t and t' is assumed as the H/S failure time.

25. AS12a: SBO(2) - SGs DRYOUT - UTAF - PORV/SRV STUCK OPEN (at t) -VB - AP
26. AS12b: SBO(2) - SGs DRYOUT - UTAF - PORV/SRV STUCK OPEN (at t') - VB - AP

For the cases AS12a and AS12b, the AFWS is assumed to be unavailable from the start of the accident. PORV or SRV Stuck Open is assumed before vessel breach. The timing of the PORV/SRV Stuck Open is uncertain.Time t and t' is assumed as the failure time.

27. AS13a: SBO(2) - RCP SEAL FAILS (PSL) - UTAF - VB- AP
28. AS13b: SBO(2) - RCP SEAL FAILS (PSLA) - UTAF - VB- AP
29. AS13c: SBO(2) - RCP SEAL FAILS (PSLB) - UTAF - VB- AP
30. AS13d: SBO(2) - RCP SEAL FAILS (PSLB) - UTAF - VB- AP

For the cases AS13a, AS13b, AS13c, and AS13d, the AFWS is assumed to be unavailable from the start of the accident. RCP Seal Failure of different modes is assumed.

31. AS14a: SBO(2) - RCP SEAL FAILS(PSL) - UTAF - ISGTR (1 tube ruptured at t) - VB - AP
32. AS14b: SBO(2)- RCP SEAL FAILS(PSL) - UTAF - ISGTR (1 tube ruptured at t') - VB - AP

For the cases AS14a and AS14b, the AFWS is assumed to be unavailable from the start of the accident. RCP seal failure of PSL mode is assumed. After core uncovery and before vessel breach, temperature-induced steam generator tube rupture is assumed. One tube is assumed to be ruptured. Since the tube rupture timing is uncertain, the SGTR is assumed at t and t'.

33. AS15a: SBO(2) - RCP SEAL FAILS (PSL) - UTAF- H/S FAILURE (at t) - VB - AP
34. AS15b: SBO(2) - RCP SEAL FAILS (PSL) - UTAF- H/S FAILURE (at t') - VB -AP

For the cases AS15a and AS15b, the AFWS is assumed to be unavailable from the start of the accident. RCP seal failure of PSL mode is assumed. Temperature-induced hot leg or surge line failure is assumed. The failure timing is uncertain and is assumed at t and t'.

35. AS16a: SBO(2) - RCP SEAL FAILS (PSL) - UTAF - PORV/SRV STUCK OPEN (at t) - VB- AP
36. AS16b: SBO(2) - RCP SEAL FAILS (PSL) - UTAF - PORV/SRV STUCK OPEN (at t') -VB- AP

For the cases AS16a and AS16b, the AFWS is assumed to be unavailable from the start of the accident. RCP seal failure of PSL mode is assumed. After core uncovery, PORV or SRV Stuck Open before vessel breach is assumed. The failure timing is uncertain and is assumed at t and t'.

ABBREVIATIONS:

AFWS - Auxiliary Feedwater System
AP - Accident Progressing after Vessel Breach
H/S - Hot Leg/Surge Line
ISGTR - Induced Steam Generator Tube Rupture
PORV - pressurizer Power Operated Relief Valve
SBO(1) - Station Blackout with AFWS initially available
SBO(2) - Station Blackout with AFWS initially not available
SGs - Steam Generators
SRV - pressurizer Safety Relief Valve
UTAF - Uncovery of Top of the Active Fuel
VB - Vessel Breach

Accident scenarios AS1, AS2a, AS2b, AS3a, AS4a are chosen to be the reference data, representing Vessel Breach, ISGTR (1 tube), ISGTR (10 tubes), Hot Line/Surge Line Failure, PORV Stuck Open cases respectively. Figure 1.1 to Figure 1.25 show the MAAP simulation results of these scenarios. Sensor readings of primary system pressure, steam generator pressure, steam generator temperature, containment pressure, and containment temperature were used for diagnosis.

Artificial Neural Networks and Expert System

The human brain accomplishes very complicated tasks by using billions of simple neurons which are interconnected [17-19]. Artificial neural networks (ANN) are the computer simulations of human brain function. These networks have many artificial neurons, usually called processing elements. These processing elements are organized in layers and have similar functions as human neurons by many adding up the values with weights of the many inputs. The input layer acts as a buffer for the input data. The output layer acts as a buffer for the output results. There might be one or more hidden layers in between. A learning process is accomplished by presenting both input data and desired output results and then obtaining the weighting coefficients among layers of processing elements by some learning algorithms. During recall process, the trained neural network takes inputs and generates output results.

Figure 2 shows the basic structure of the diagnostic neural networks. It is a three-layer, feed-forward, backpropagation neural network. The MAAP data is used to train the neural networks which are then tested against all the other scenarios. To some extent, this would guarantee the generality of the neural networks to detect and identify the faults under various conditions.

The expert system will provide the general environment for monitoring the overall plant status, determination of neural networks usage, displaying necessary information. The expert system also provides independent primary system failure diagnosis, if possible. The software used for the proposed expert system will be NEXPERT OBJECT [20], which is a commercial software under the IBM PC window environment. IF-THEN rules are used for backward reasoning and forward reasoning.

Neural Network Training

To evaluate the data adequacy for diagnosis and determine the data for neural network training for failure detecting and identification, training data was taken from the starting of the failure and was progressive increased (every 20 second step). The input neurons are determined according to the amount of data for training. There are two output neurons with following mapping scheme for training shown in table 1.

Table 4.2.1 Output mapping scheme for neural network training for data evaluation

CASE NAME	Output Neuron 1 target	Output Neuron 2 target
Vessel Breach	0.1	0.1
ISGTR	0.1	0.9
Hot Leg/Surge Line Failure	0.9	0.1
PORV Stuck Open	0.9	0.9

Fourteen groups of data (3x20s, 4x20s, 5x20s, 6x20s, 7x20s, 8x20s, 9x20s, 10x20s, 13x20s, 14x20s, 15x20s, 16x20s, 17x20s, and 20x20s) from AS1 (Vessel Breach), AS2a (ISGTR), AS2b(ISGTR), AS3a(H/S Failure), AS4a(PORV Failure) was used for the training. Sensor data is normalized between 0.0 and 1.0. For network recall process, any data less than 0.25 is treated as 0, any data above 0.75 is treated as 1.0, any data between 0.25 and 0.5 is treated as likely 0, any data between 0.5 and 0.75 is treated as likely 1. The mapping scheme used for testing is shown in table 2.

Figure 3.1 to Figure 3.14 show the training convergence for these cases. Table 3 to Table 6 show the test results. Case 1 is the AFWS initially working and no RCP Seal Failure case. Case 2 is the AFWS initially working with RCP Seal Failure case. Case 3 is the AFWS initially Non-working and no RCP Seal Failure case. Case 4 is the AFWS initially Non-working with RCP Seal Failure case.

With the increasing of the data into the accident, the neural networks recall ability converge to a certain level where test results are no longer improved with more data. From the results, the converged time data for VB and Hog Leg/Surge Line is 3x20s. The converged time data for ISGTR and PORV Stuck Open is 15x20s.

Table 2 Output mapping scheme for testing for data evaluation

Output Neuron 1	Output Neuron 2	Mapping Case
0.0 - 0.25	0.0 - 0.25	Vessel Breach
0.25 - 0.5	0.0 - 0.25	likely Vessel Breach
0.0 - 0.25	0.25 - 0.5	likely Vessel Breach
0.25 - 0.5	0.25 - 0.5	likely Vessel Breach
0.0 - 0.25	0.75 - 1.0	ISGTR
0.25 - 0.5	0.75 - 1.0	likely ISGTR
0.0 - 0.25	0.5 - 0.75	likely ISGTR
0.25 - 0.5	0.5 - 0.75	likely ISGTR
0.75 - 1.0	0.0 - 0.25	H/S Failure
0.75 - 1.0	0.25 - 0.5	likely H/S Failure
0.5 - 0.75	0.0 - 0.25	likely H/S Failure
0.5 - 0.75	0.25 - 0.5	likely H/S Failure
0.75 - 1.0	0.75 - 1.0	PORV Failure
0.5 - 0.75	0.75 - 1.0	likely PORV Failure
0.75 - 1.0	0.5 - 0.75	likely PORV Failure
0.5 - 0.75	0.5 - 0.75	likely PORV Failure

Table 3 Test results for vessel breach identification

time data	s	3x20	4x20	5x20	6x20	7x20	8x20	9x20	10x20	15x20	16x20	17x20	20x20
case1													
case2		++	++	++	++	++	++	++	++	++	++	++	++
case3		+	++	++	++	++	++	++	++	++	++	++	++
case4		+	++	++	++	++	++	++	++	++	++	++	++

Table 4 Test results for hot leg/surge line identification

time data	s	3x20	4x20	5x20	6x20	7x20	8x20	9x20	10x20	15x20	16x20	17x20	20x20
case1		++	++	++	++	++	++	++	++	++	++	++	++
case2		++	++	++	++	++	++	++	++	++	++	++	++
case3		++	++	++	++	++	++	++	++	++	++	++	++
case4		++	++	++	++	++	++	++	++	++	++	++	++

Table 5 Test results for ISGTR identification

time data	s	3x20	4x20	5x20	6x20	7x20	8x20	9x20	10x20	15x20	16x20	17x20	20x20
case1		++	++	++	++	++	++	++	++	++	++	++	++
case2		++	++	++	++	++	++	++	++	++	++	++	++
case3		++	+	-PORV	-PORV	-PORV	-PORV	+	+	+	+	+	+
case4		++	++	++	++	+	+	+	+	+	+	+	+

Table 6 Test results for PORV Stuck Open identification

time data	s	3x20	4x20	5x20	6x20	7x20	8x20	9x20	10x20	15x20	16x20	17x20	20x20
case1		++	++	++	++	++	++	++	++	++	++	++	++
case2		-ISGTR	-ISGTR	-ISGTR	-ISGTR	-ISGTR	+	+	+	++	++	++	++
case3		+	-ISGTR	+	+	+	+	+	+	++	++	++	++
case4		+	++	++	++	++	++	++	++	++	++	++	++

Note: ++ positive + likely positive - likely negative -- negative

Finally, the Detection Neural Networks and Identification Neural Network were constructed. For each of these two neural networks, there are 80 input neurons representing 15 time step of data of 20 second each. There are three output neurons with following mapping scheme shown in table 7.

Table 7 Output mapping scheme for Detection and Identification Neural Networks training

CASE NAME	Output Neural 1	Output Neural 2	Output Neural 3
No Failure	0.1	0.1	0.1
Vessel Breach	0.9	0.1	0.1
ISGTR	0.9	0.1	0.9
H/S Failure	0.9	0.9	0.1
PORV Failure	0.9	0.9	0.9

The training samples included data of no failure case. The training data for Vessel Breach and H/S Failure ranged from 3x20s to 8x20s into the accident respectively. The training data for ISGTR and PORV Stuck Open ranged from 10x20s to 15x20s. Since the number of input neurons is fixed at 80 or 15 time steps of 20 seconds, most training data also covers a portion of no failure case. Figure 4.1 and Figure 4.2 show the convergence of the training of the neural networks. The networks are tested using data VB and HS can be detected 20 seconds into the accident and identified 30 seconds into the accidents which would be confirmed for several more time steps. The ISGTR and PORV Stuck Open accidents can be detected 160 seconds into the accident and be identified 180 seconds into the accidents. Test of no failure cases was successful. When 30% of random noise was added to the training data, the Detection Neural Network can still correctly detect various failure. Vessel Breach or H/S Failure can be correctly identified by Identification Neural Network with 25% random noise added to the training data. PORV Stuck Open and ISGTR can be correctly identified with 10% random noise.

The Detection Neural Network and Identification Neural Network could be initiated during the cycling of the PORV period and long before the start of primary system failure. Every 20 seconds, a new time step data could be fed in and the oldest time step data is thrown out. If the situation is classified as No Failure by Detection Neural Network, the process would continue. If some failure is detected by the Detection Neural Network, the Identification Neural Network would be initiated to identify which failure occurred. Further data input would be used for diagnosis confirmation.

SUMMARY

Neural network techniques have been successfully used to detect and identify primary system failures during station blackout. Among the things accomplished, the use of neural networks to evaluate data adequacy and sufficiency is a novel application of such a technique. The same technique will be used to construct neural networks for RCP Seal Failure cases. The diagnosis of ISGTR would be more effective if steam generator radiation level can be used in the neural networks. Even though we give a scale for some sort of uncertainty assessment, a more thorough uncertainty analysis would be desirable, if possible. Expert system knowledge base formation and the integration of the prototype severe accident diagnostic system are the remaining tasks.

References

[1] USNRC, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plant. NUREG-1150, April, 1989

[2] Hanson, D.J. et al., Accident Management Information Needs. NUREG/CR-5513, April, 1990

[3] Brownson, D.A. et al., Intentional Depressurization Accident Management Strategy for Pressurized Water reactors. NUREG/CR-5937, April, 1993

[4] Silverman, E.B. and Klopp, G.T., " Neural Network-Based Expert System For Severe Accident management", EPRI Proceedings

[5] Guarro, S. et al., " A Decision Aid for Interpreting Instrument Information and managing Accident Conditions in Nuclear Power Plant ", OECD/CSNI Specialists Meeting On Instrumentation To Management Severe Accidents, Cologne, Germany, March 16-17, 1992

[6] Xing, L. and Okrent, D., "Application of Neural Networks in AccidentManagement", Proceedings of PSA International Topical Meeting, ClearwaterBeach, Florida, 1993

[7] Uhrig, R.E., "Potential Use of Neural Networks in Nuclear Power Plants". IEEE International Conference on Neural Networks, 1993

[8] Basu, A. and Bartlett, E. B., "Detecting Faults in a Nuclear Power Plant by using Dynamic Node Architecture Artificial Neural Networks". Nuclear Science and Engineering, No. 116, 313-325, 1994

[9] Marseguerra, M. and Zio, E., "Fault Diagnosis via Neural Networks: the Boltzmann Machine". Nuclear Science and Engineering, No. 117, 194-200, 1994

[10] Specht, D.F., "Probabilistic Neural Networks". Neural Networks, Vol. 3, 1990

[11] Agogino, A. et al., "Integrating Neural Networks with Influence Diagrams for Power Plant Monitoring and Diagnostics". IEEE International Conference on Neural Networks, 1993

[12] Guarro, S.B., "Diagnostic Models for Engineering Process Management: A Critical Review of Objectives, Constraints and Applicable Tools". Reliability Engineering and System Safety, NO.30, 1990

[13] Kim, I.S. et al., "A Model-based Approach to On-line Process Disturbance Management: The Models". Reliability Engineering and System Safety, NO.29, 1990

[14] Yang, J.O., Chang, S.H., "A Diagnostic Expert System for the Nuclear Power Plant Based on the Hybrid Knowledge Approach". IEEE Transactions on Nuclear Science, Vol.36, NO.6, 1989

[15] Knudson, D.L., Dobbe, C.A., Assessment of the Potential for High Pressure Melt Ejection resulting from a Surry Station Blackout Transient. NUREG/CR-5949, May, 1993.

[16] Oehlberg, R.N. et al., "Practical Considerations in Accident Management". Proceedings of International Meeting on safety of Thermal Reactors, Oregon, July 21-25, 1991.

[17] Marvin Minsky and Seymour Papert, "Perceptrons", Cambridge, MA: MIT Press, 1969.

[18] Yoh-Han Pao, "Adaptive Pattern Recognition and Neural Networks", Copyright 1989, Addison-Wesley Publishing Company, Inc.

[19] B. Möller, J. Reinhardt, "Neural Networks, An Introduction", Copyright © Springer-Verlag Berlin Heidelberg 1990.

[20] "NEXPERT Object", Copyright 1987, by Neuron Data Inc.

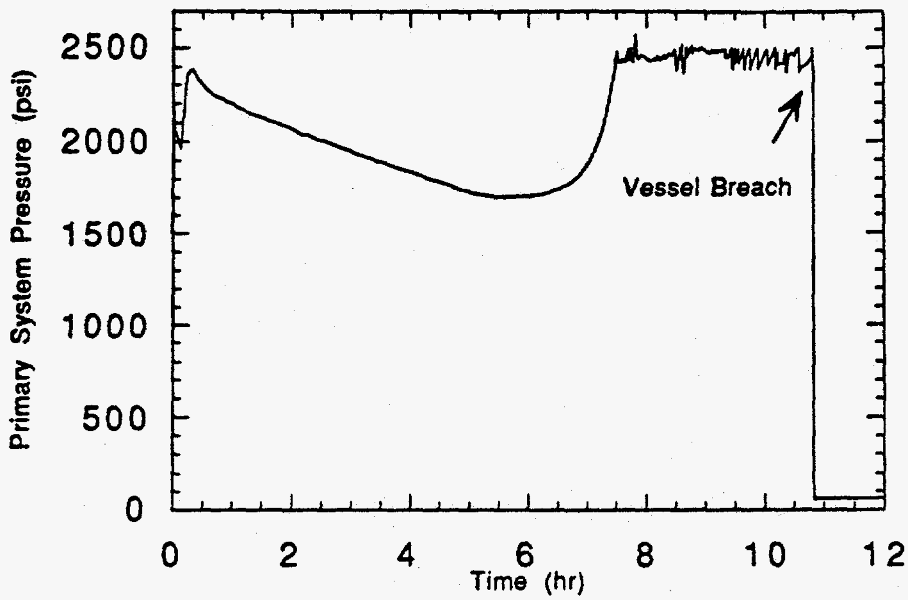

Figure 1.1 Primary System Pressure Time Series for AS1

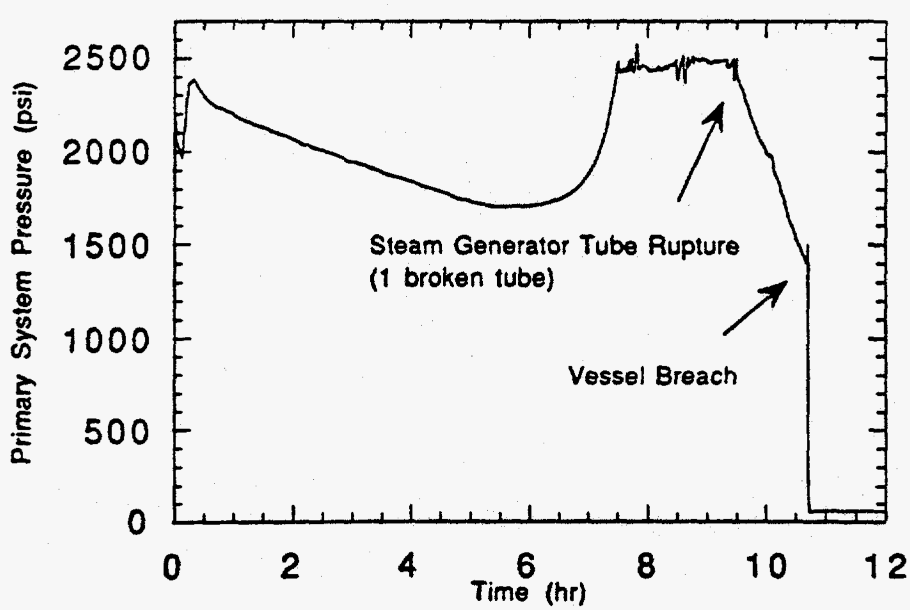

Figure 1.2 Primary System Pressure Time Series for AS2a

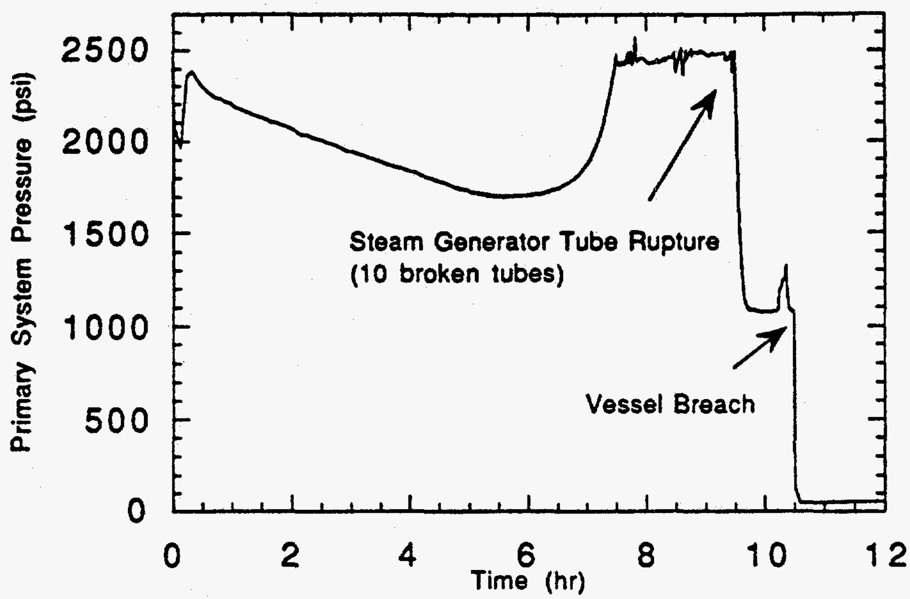

Figure 1.3 Primary System Pressure Time Series for AS2b

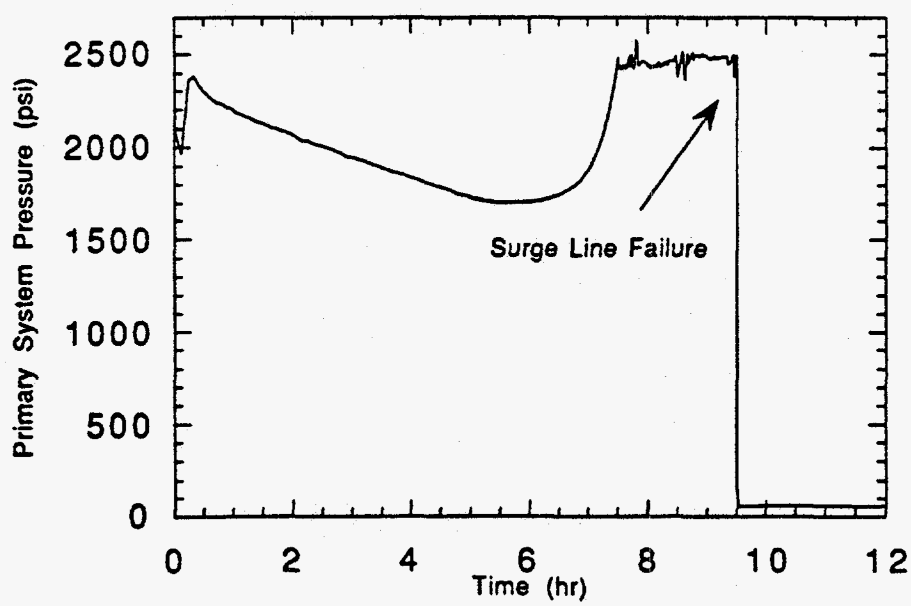

Figure 1.4 Primary System Pressure Time Series for AS3a

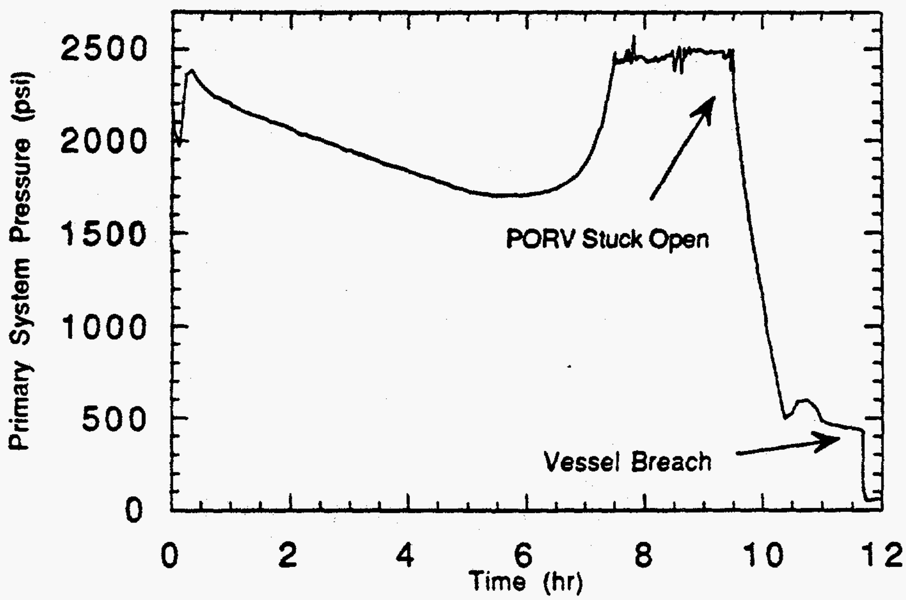

Figure 1.5 Primary System Pressure Time Series for AS4a

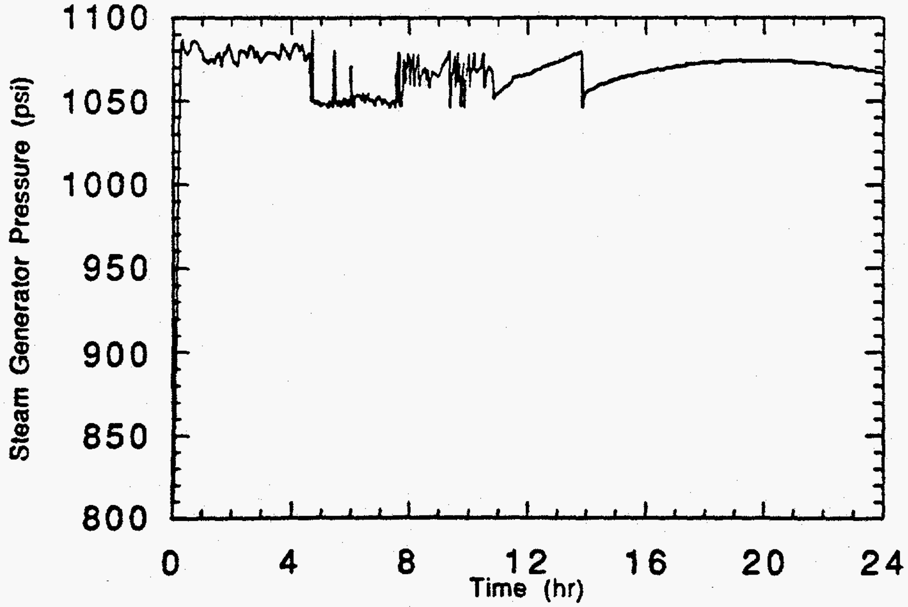

Figure 1.6 Steam Generator Pressure Time Series for AS1

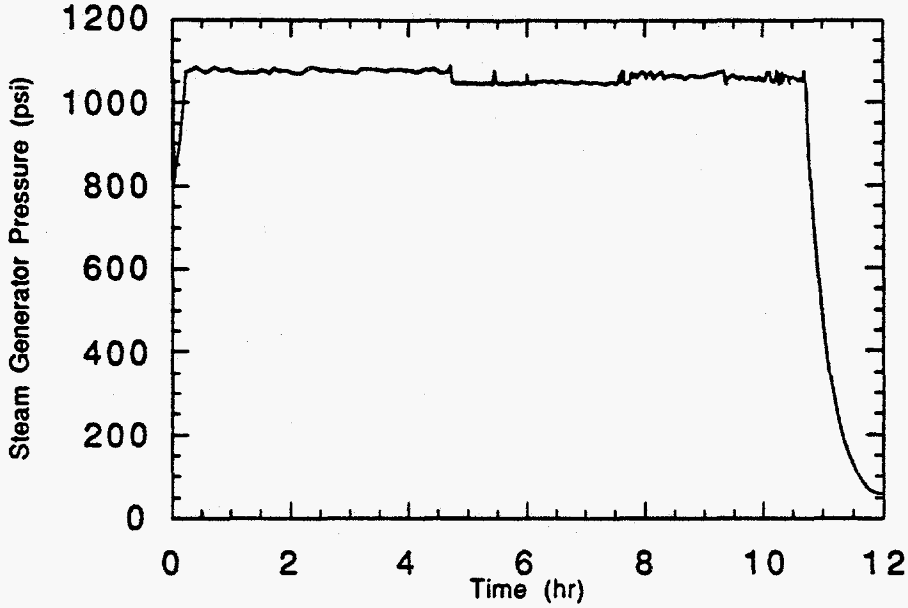

Figure 1.7 Steam Generator Pressure Time Series for AS2a

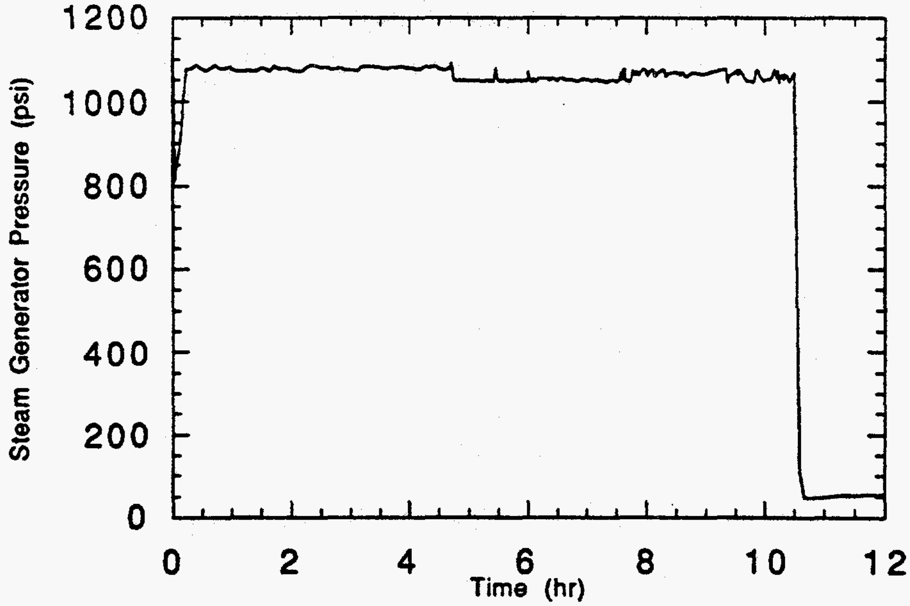

Figure 1.8 Steam Generator Pressure Time Series for AS2b

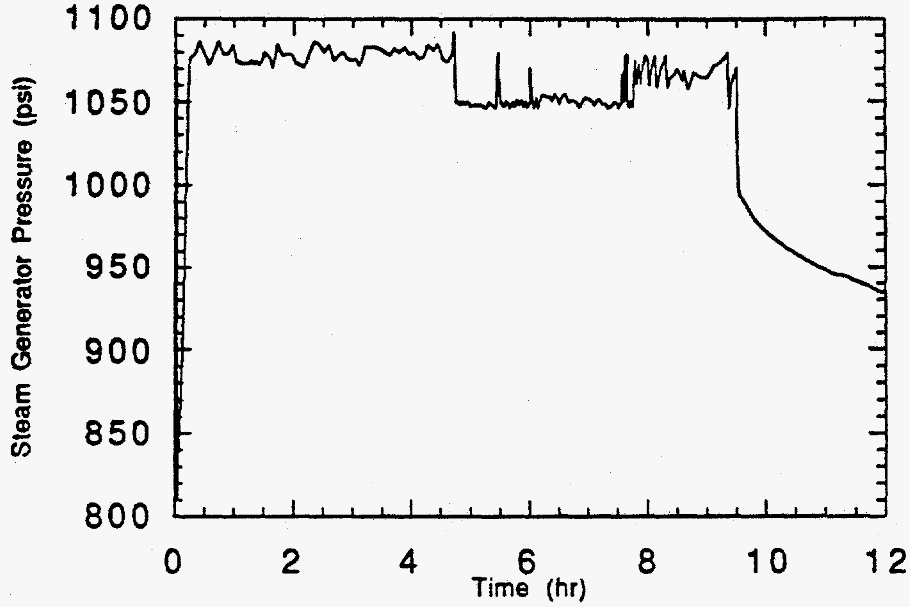

Figure 1.9 Steam Generator Pressure Time Series for AS3a

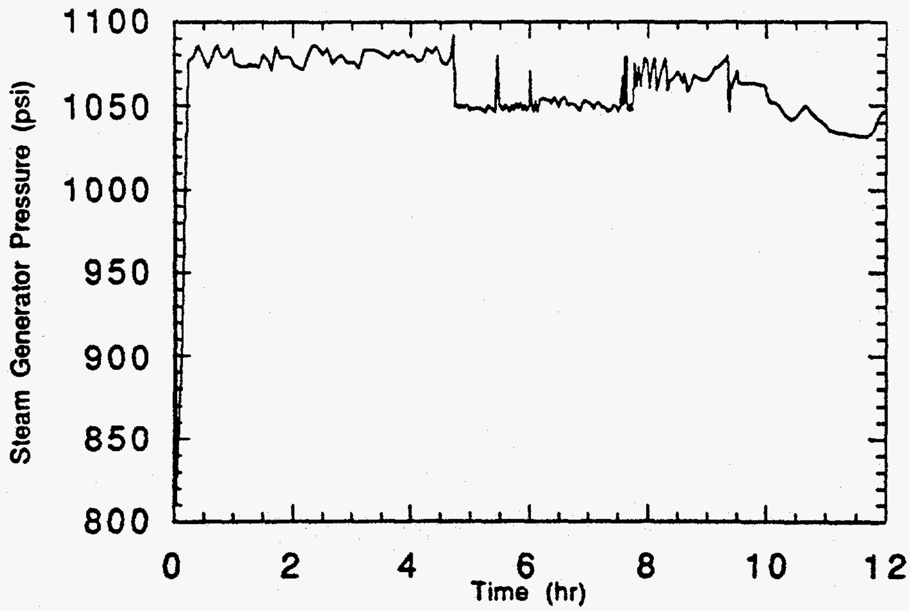

Figure 1.10 Steam Generator Pressure Time Series for AS4a

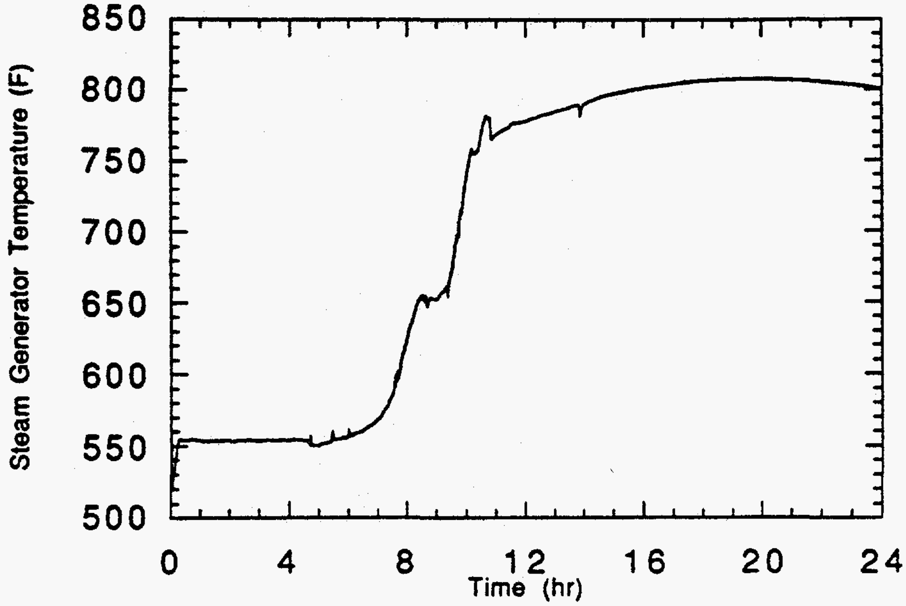

Figure 1.11 Steam Generator Temperature Time Series for AS1

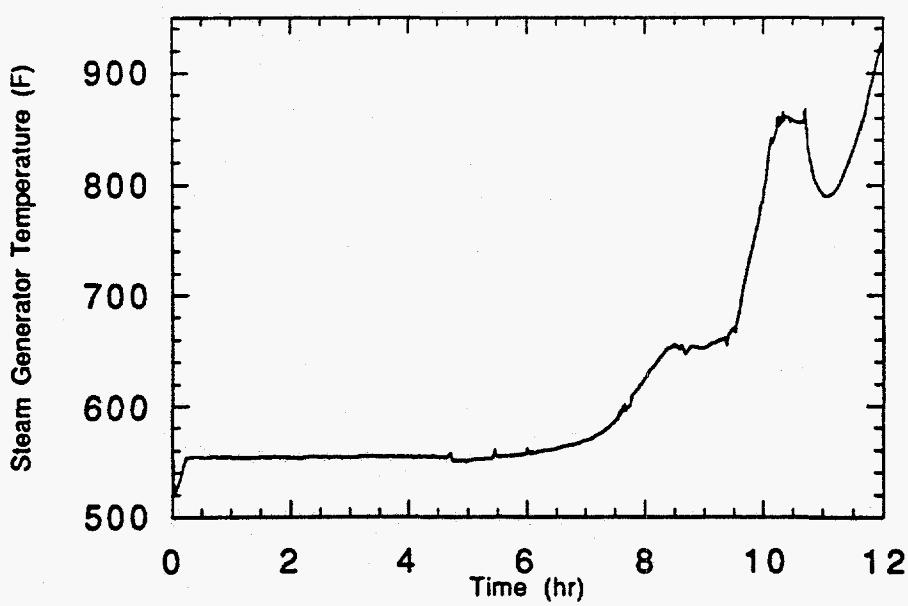

Figure 1.12 Steam Generator Temperature Time Series for AS2a

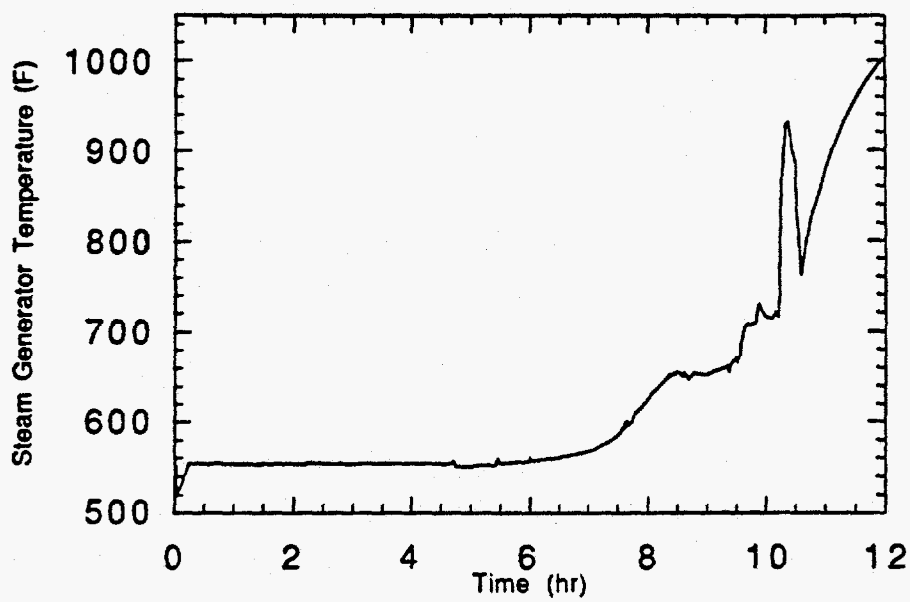

Figure 1.13 Steam Generator Temperature Time Series for AS2b

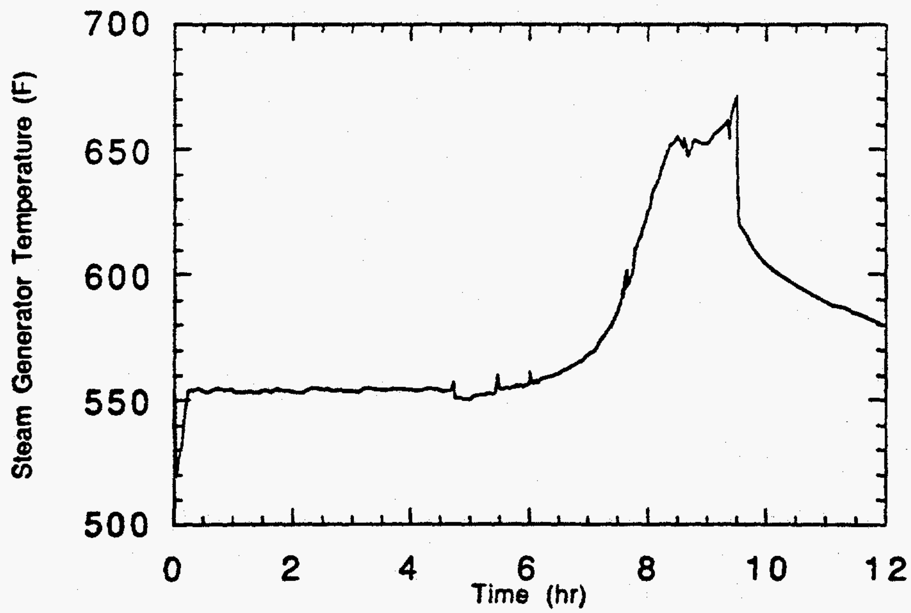

Figure 1.14 Steam Generator Temperature Time Series for AS3a

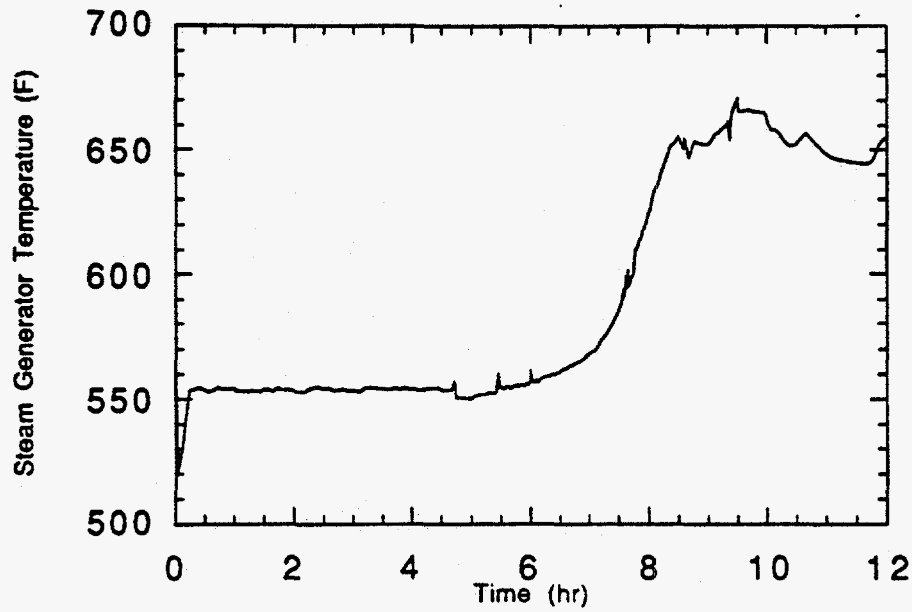

Figure 1.15 Steam Generator Temperature Time Series for AS4a

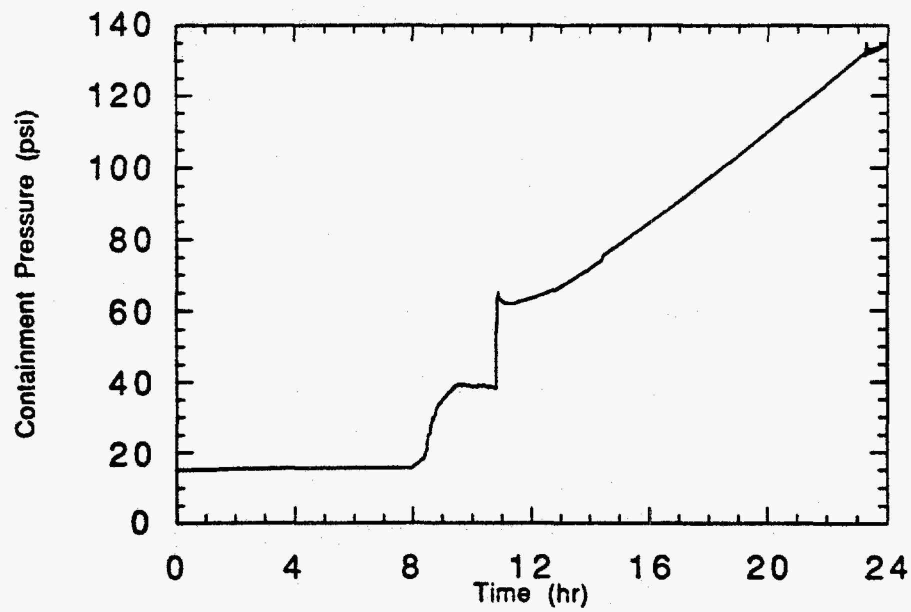

Figure 1.16 Containment Pressure Time Series for AS1

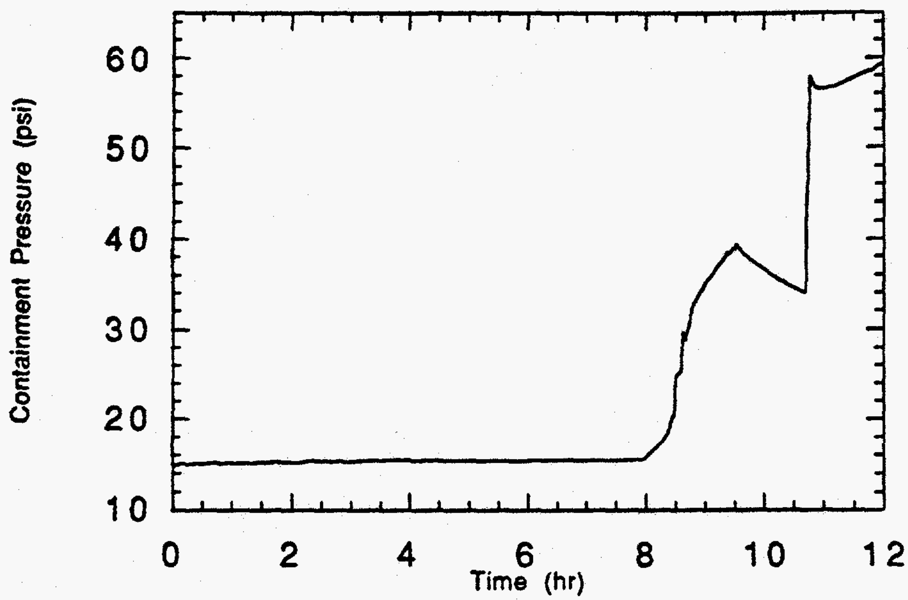

Figure 1.17 Containment Pressure Time Series for AS2a

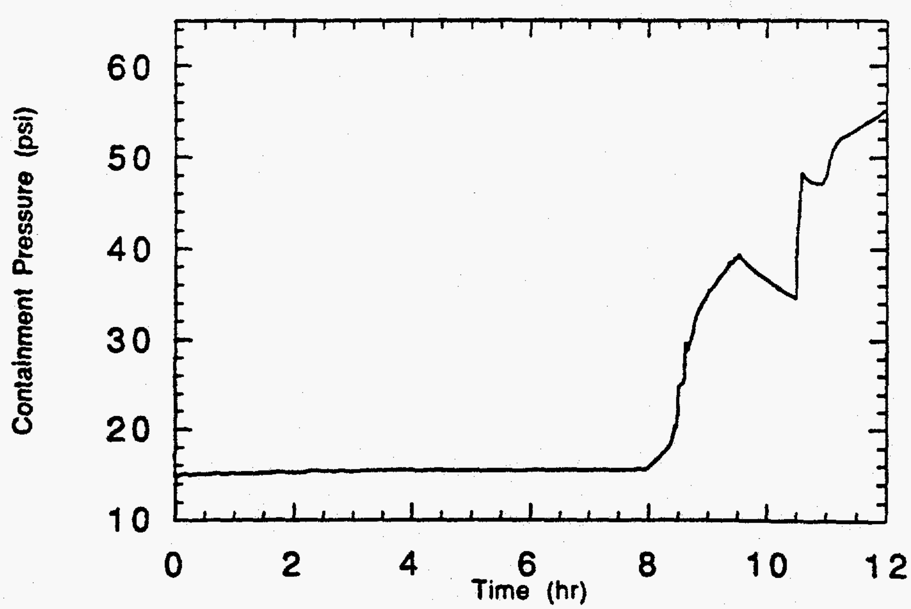

Figure 1.18 Containment Pressure Time Series for AS2b

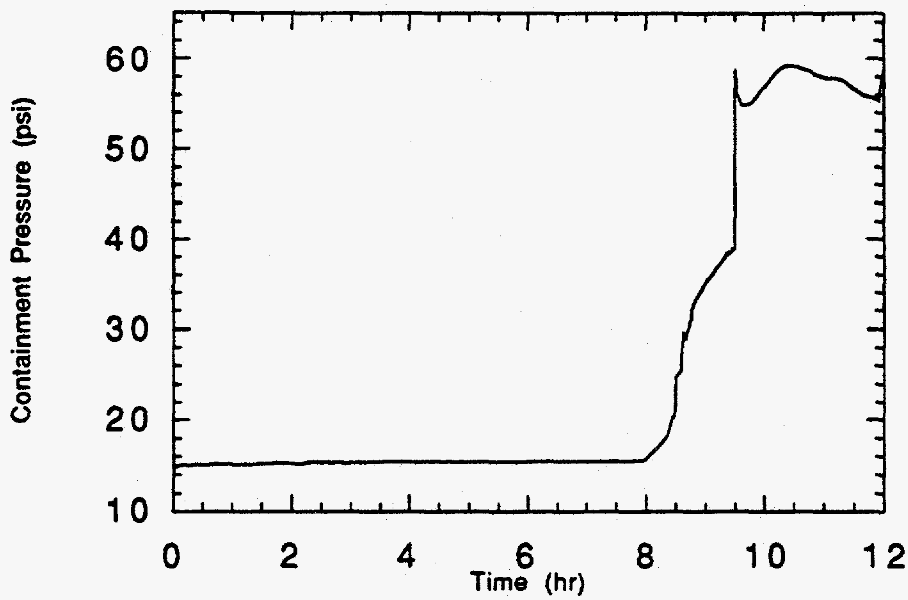

Figure 1.19 Containment Pressure Time Series for AS3a

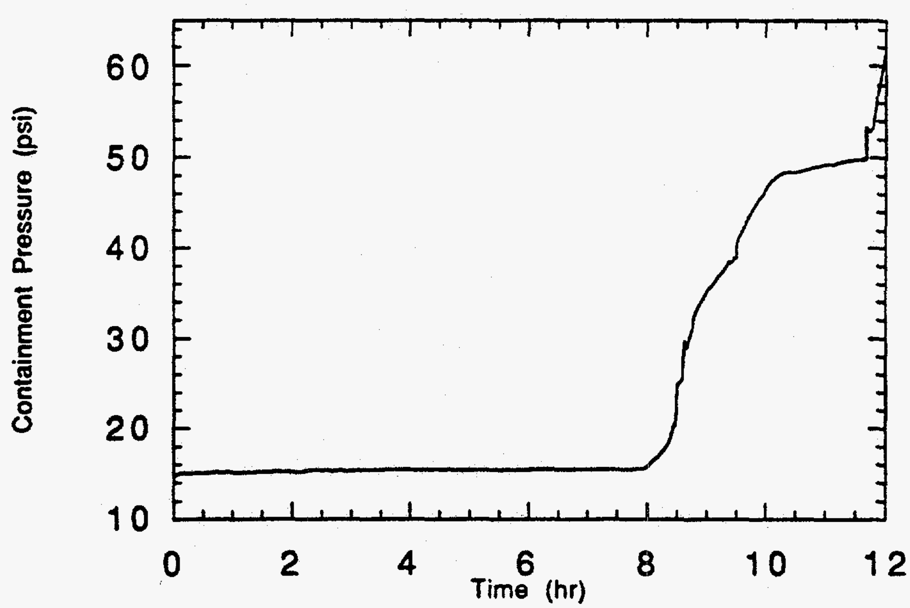

Figure 1.20 Containment Pressure Time Series for AS4a

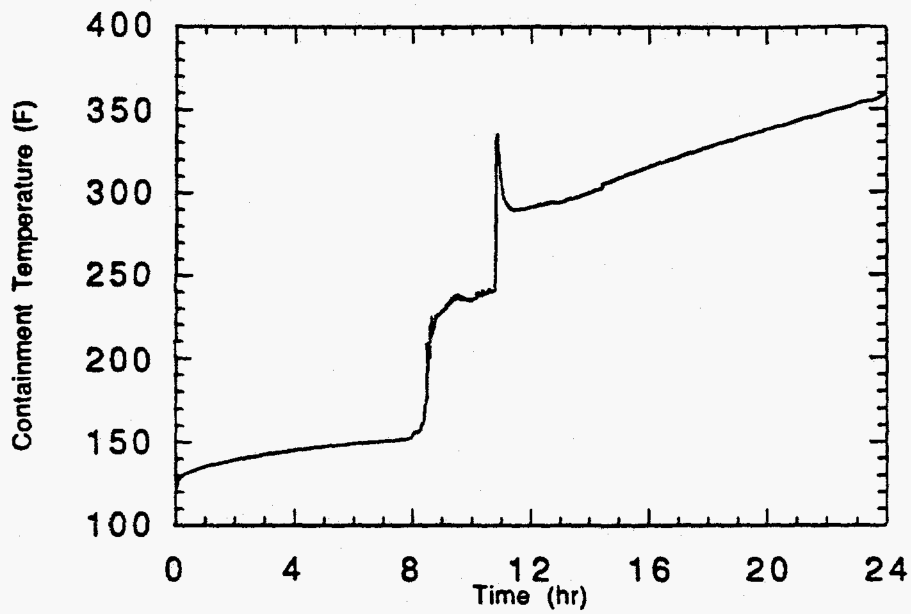

Figure 1.21 Containment Temperature Time Series for AS1

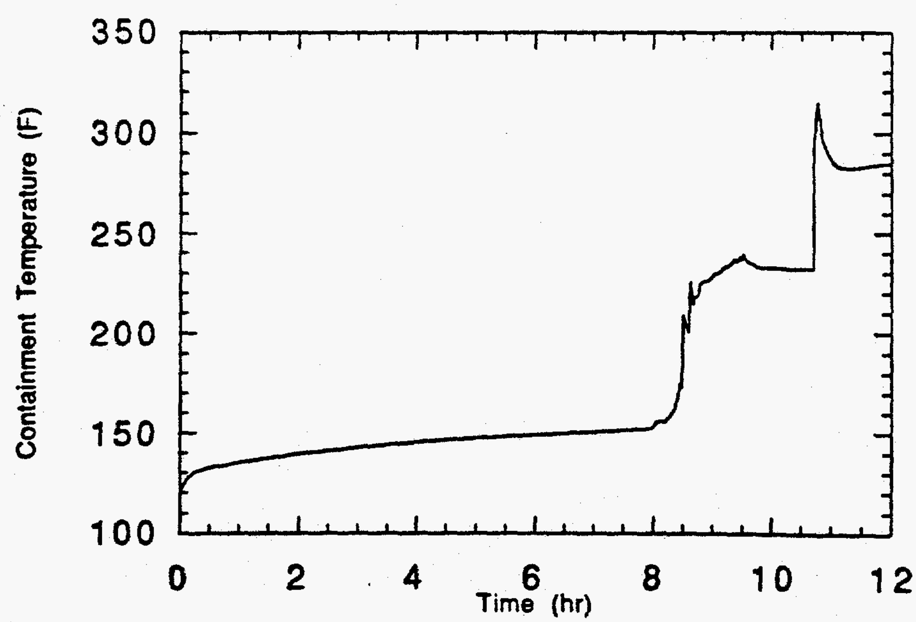

Figure 1.22 Containment Temperature Time Series for AS2a

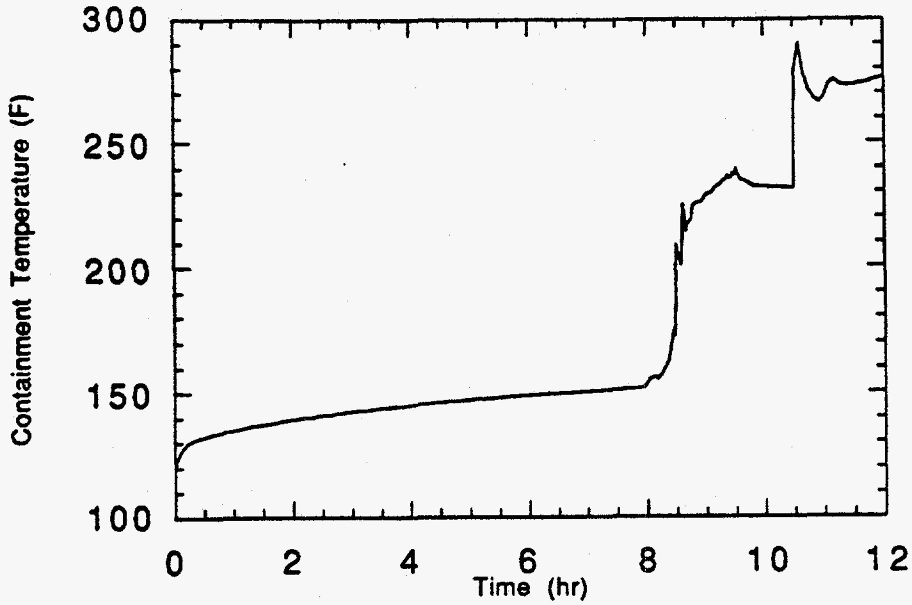

Figure 1.23 Containment Temperature Time Series for AS2b

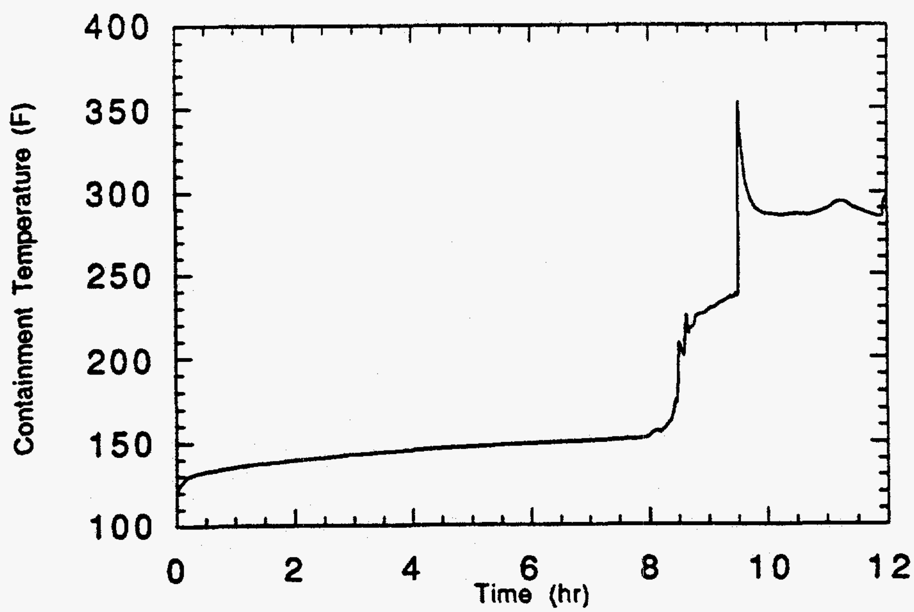

Figure 1.24 Containment Temperature Time Series for AS3a

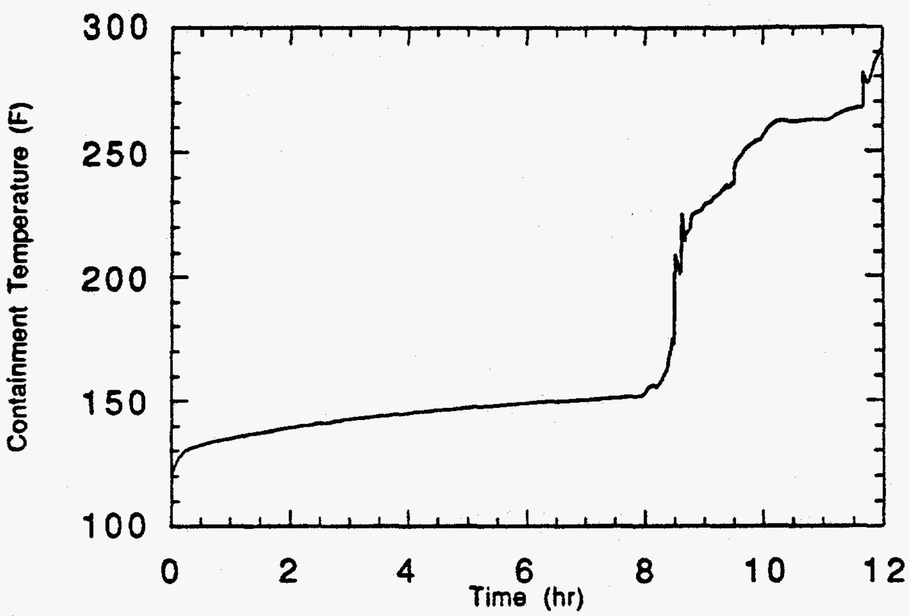

Figure 1.25 Containment Temperature Time Series for AS4a

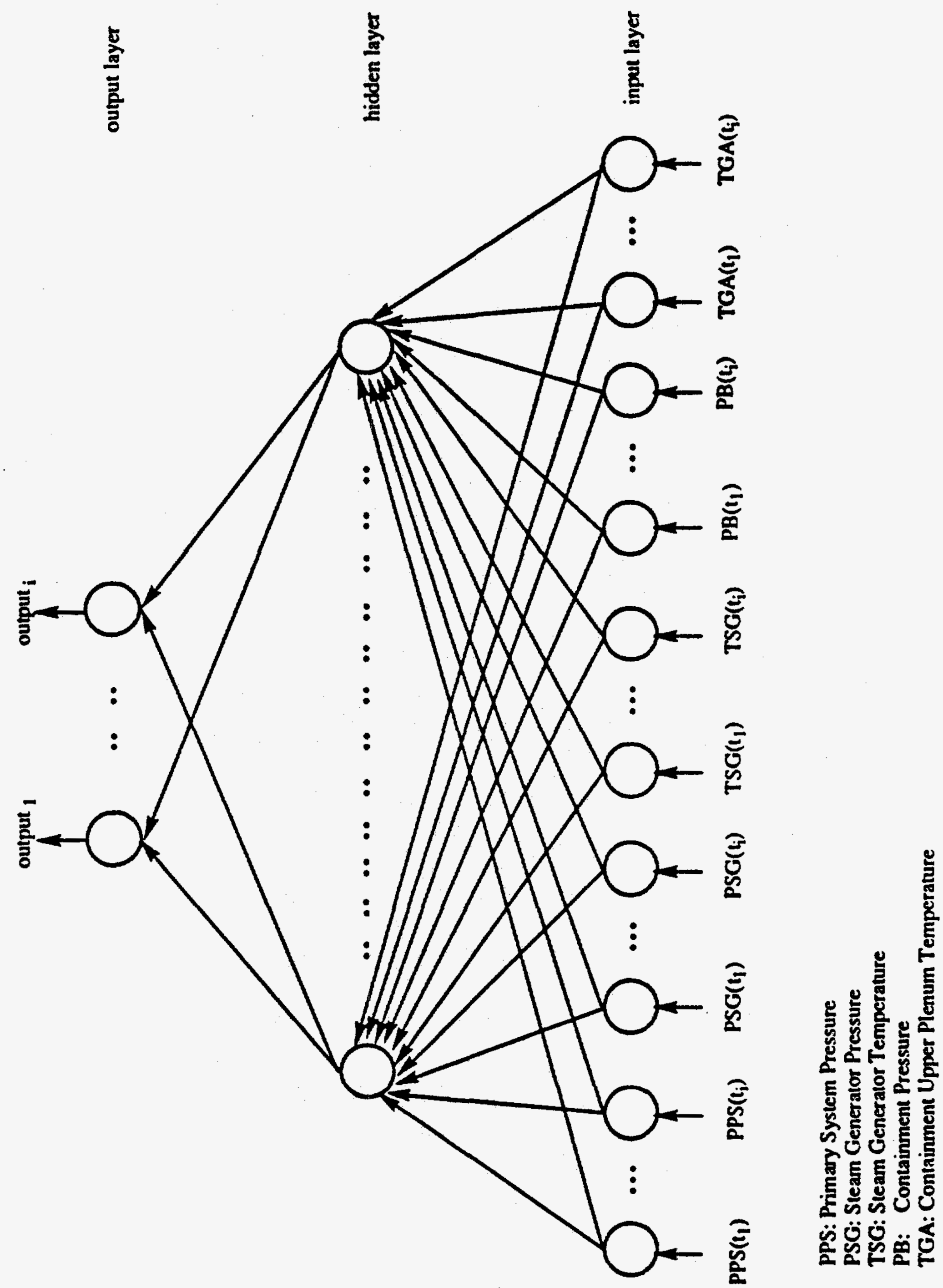

Figure 2 A three layer back propagation neural network

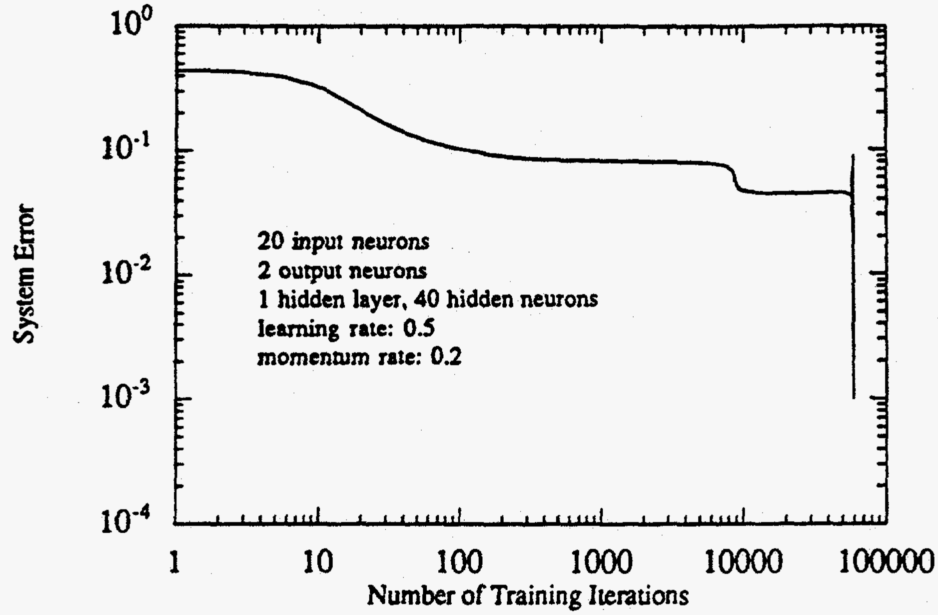

Figure 3.1 Training convergence for 3x20s data test case

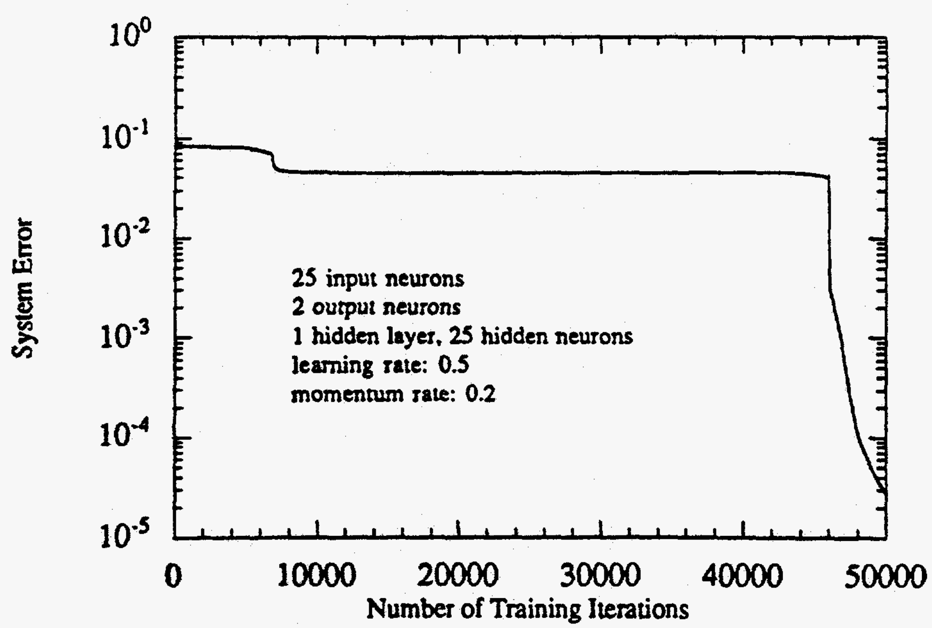

Figure 3.2 Training convergence for 4x20s data test case

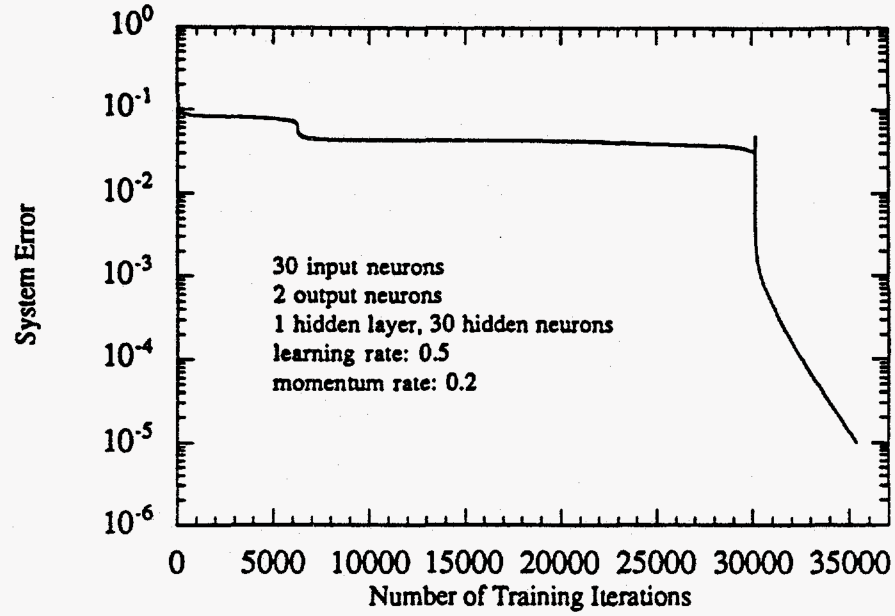

Figure 3.3 Training convergence for 5x20s data test case

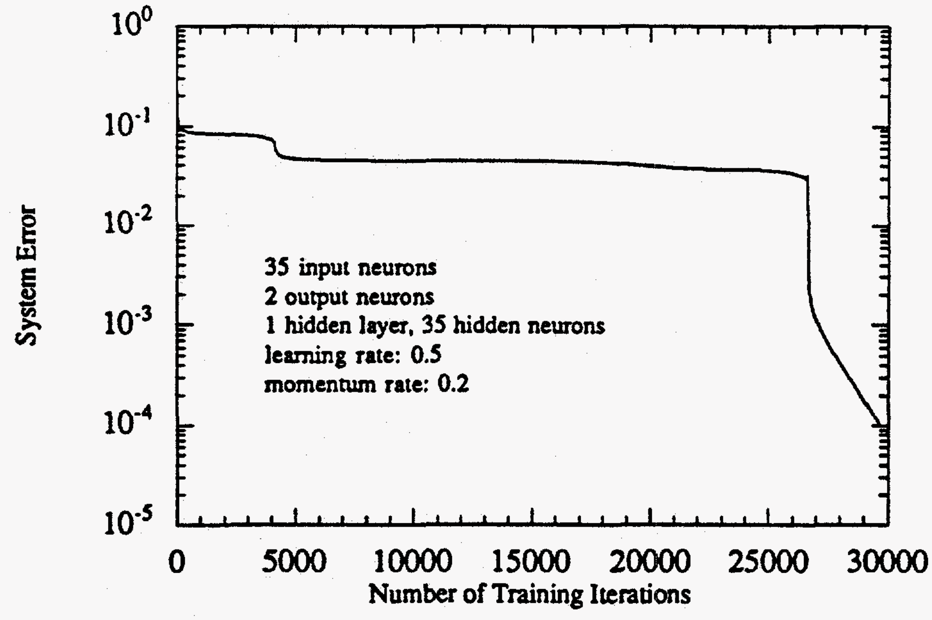

Figure 3.4 Training convergence for 6x20s data test case

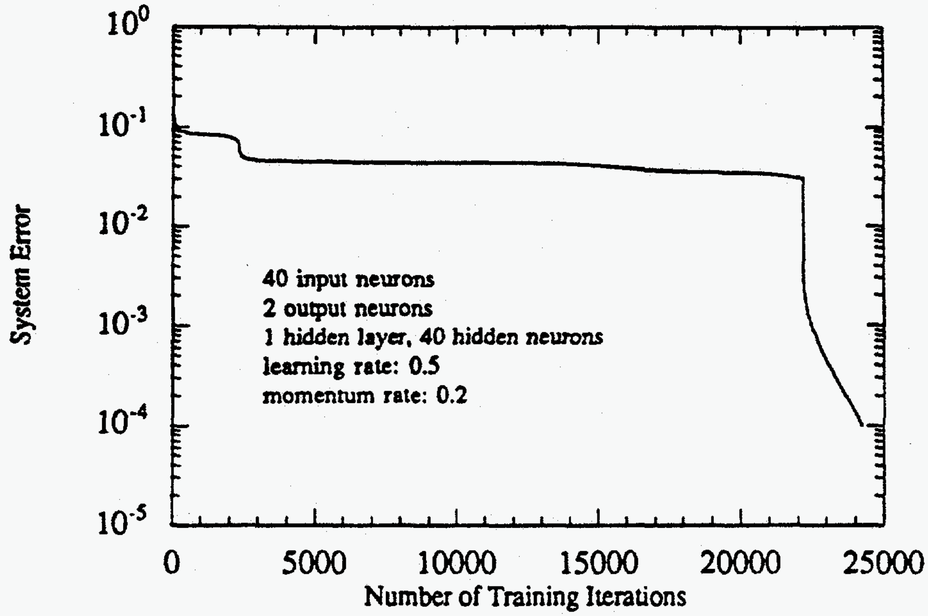

Figure 3.5 Training convergence for 7x20s data test case

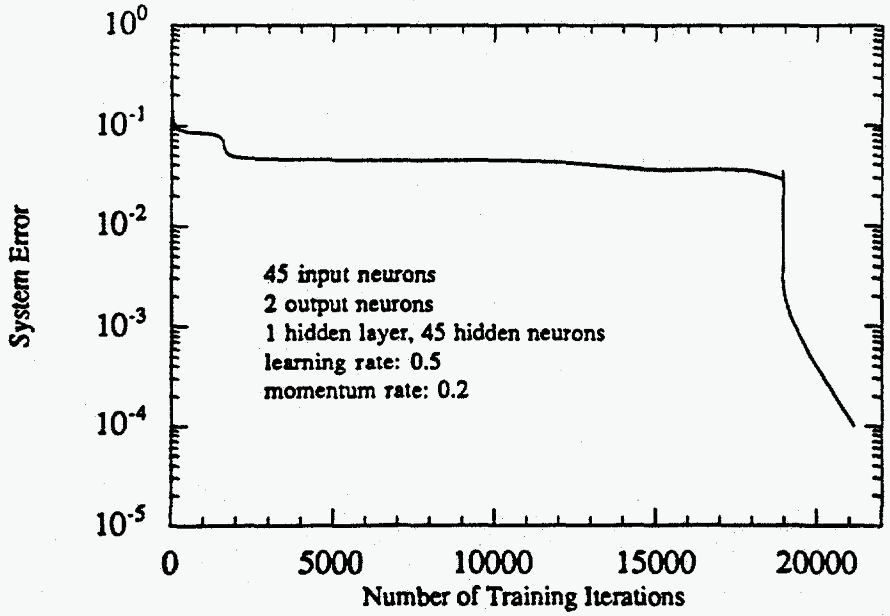

Figure 3.6 Training convergence for 8x20s data test case

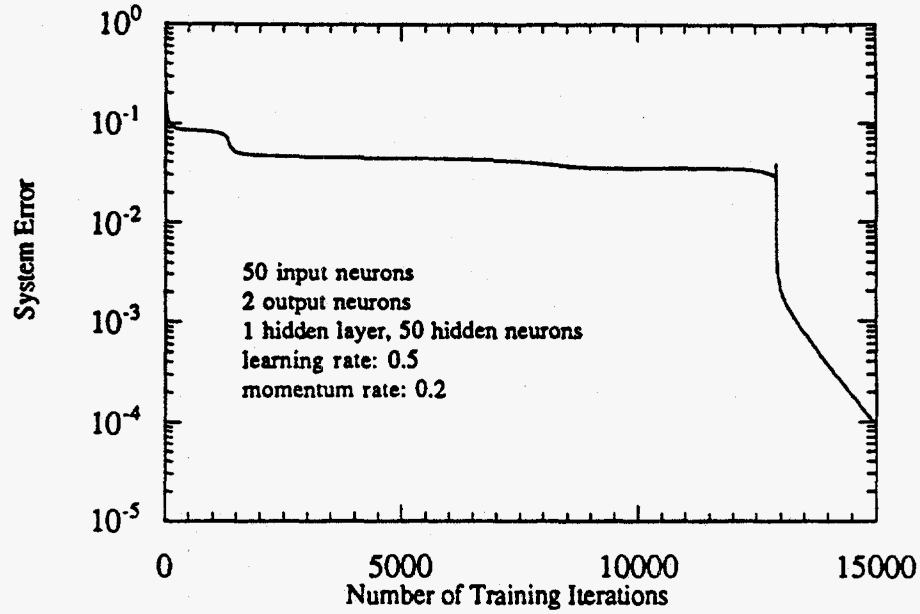

Figure 3.7 Training convergence for 9x20s data test case

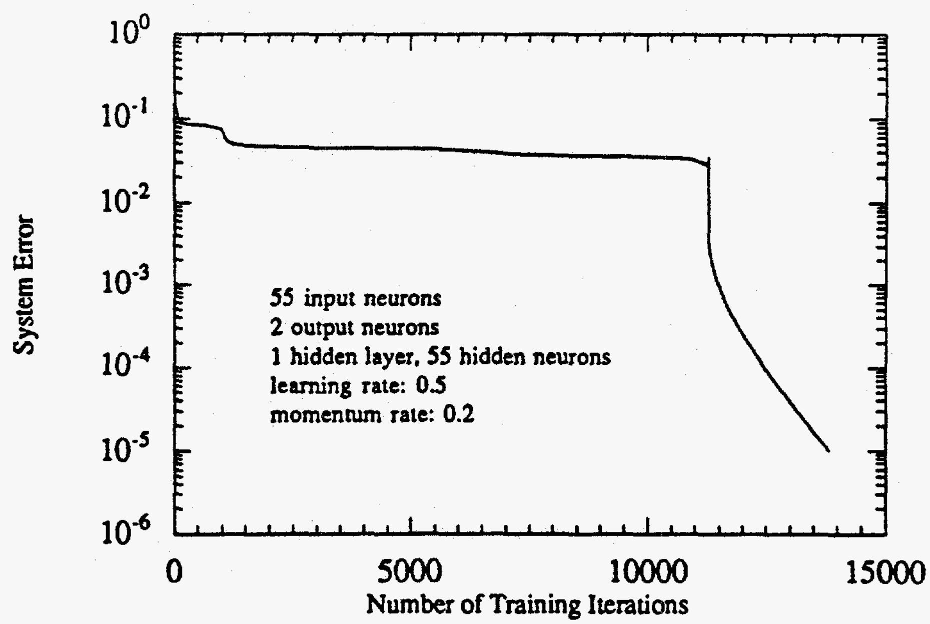

Figure 3.8 Training convergence for 10x20s data test case

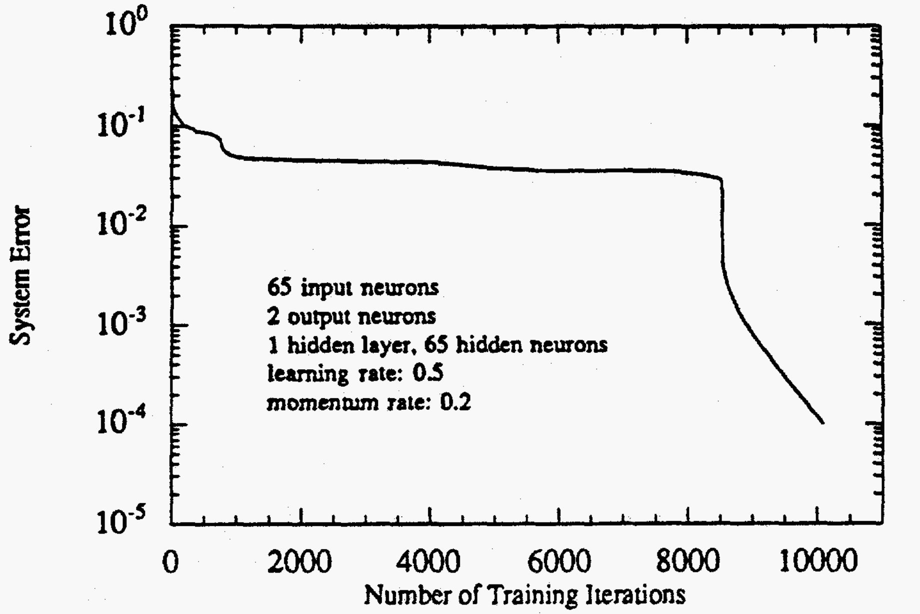

Figure 3.9 Training convergence for 12x20s data test case

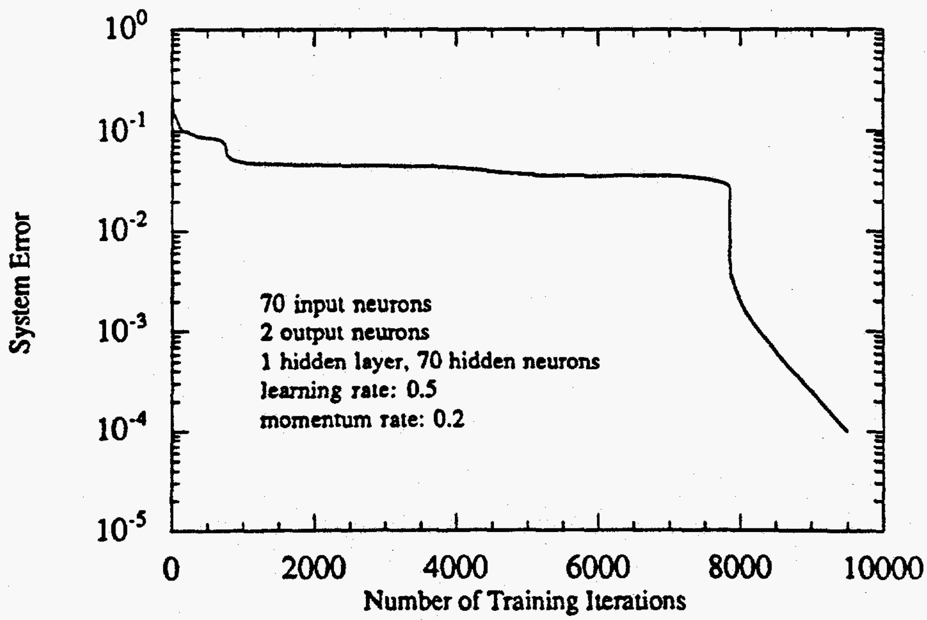

Figure 3.10 Training convergence for 13x20s data test case

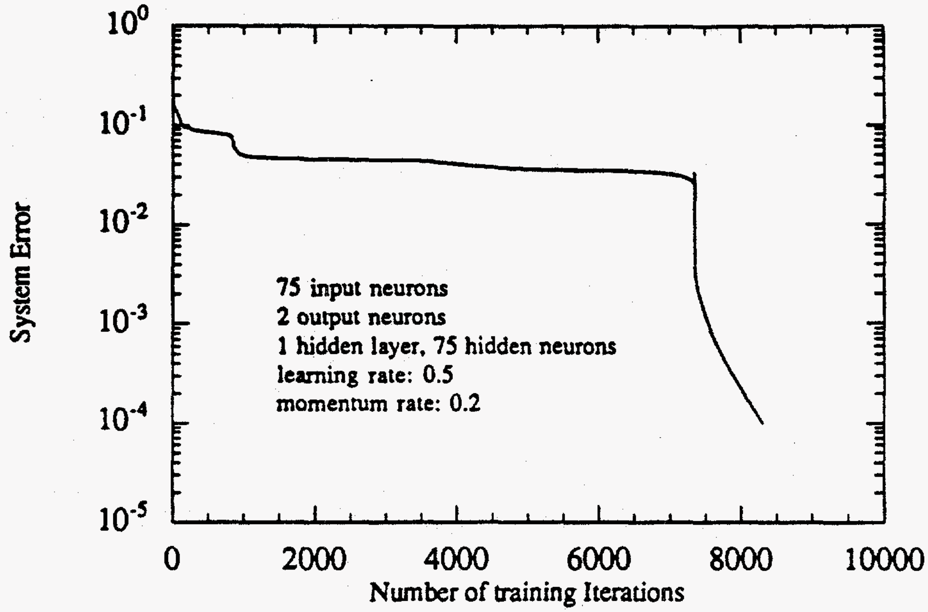

Figure 3.11 Training convergence for 14x20s data test case

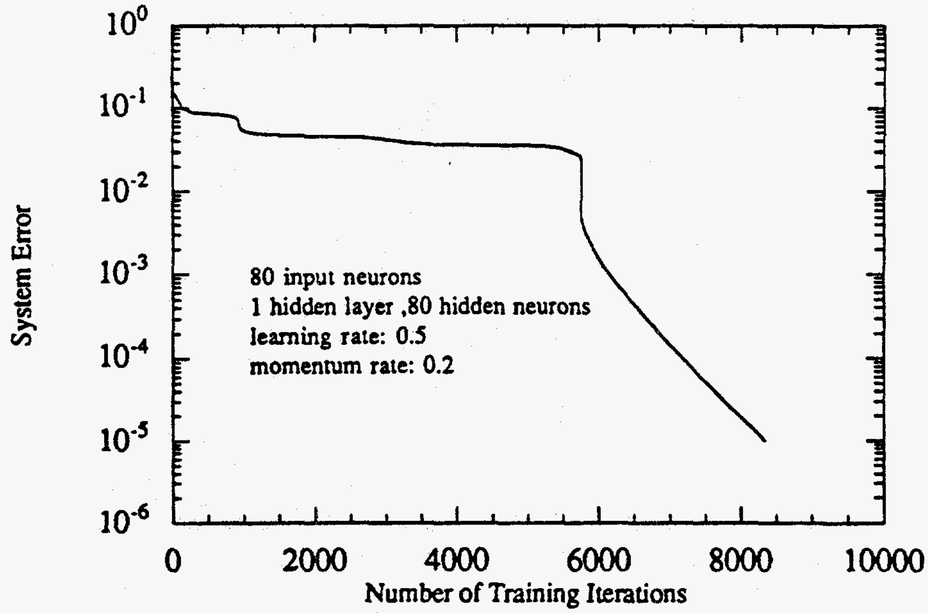

Figure 3.12 Training convergence for 15x20s data test case

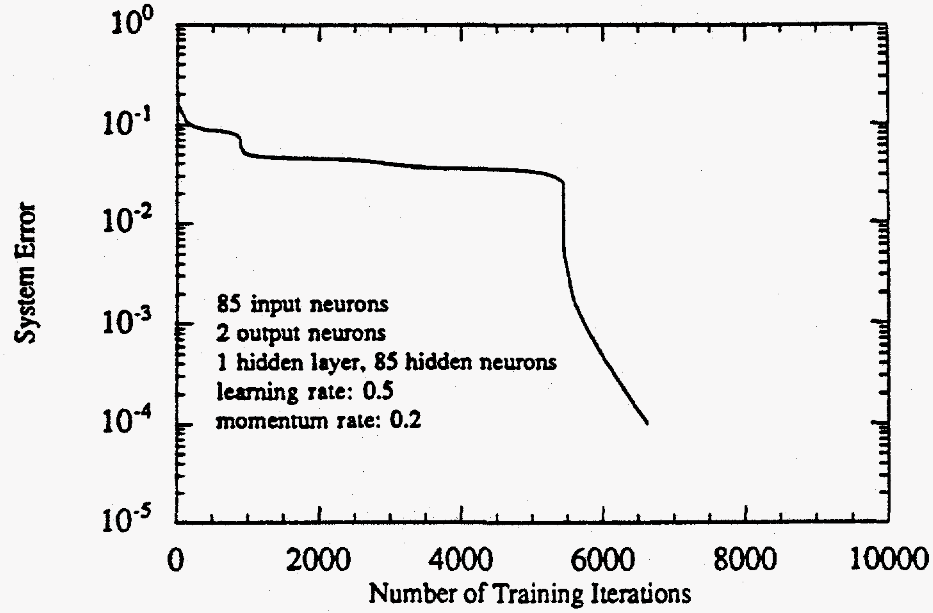

Figure 3.13 Training convergence for 16x20s data test case

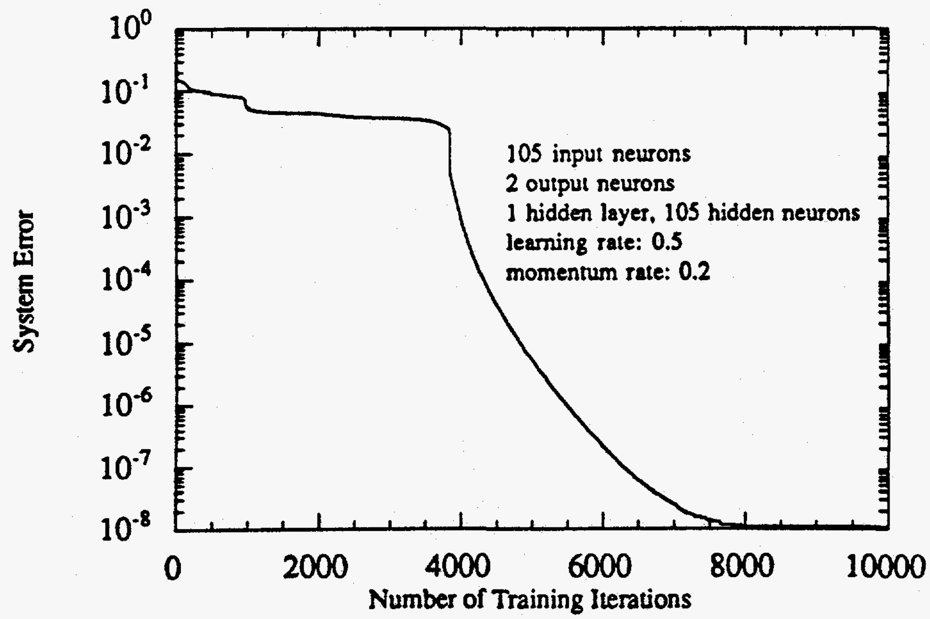

Figure 3.14 Training convergence for 20x20s data test case

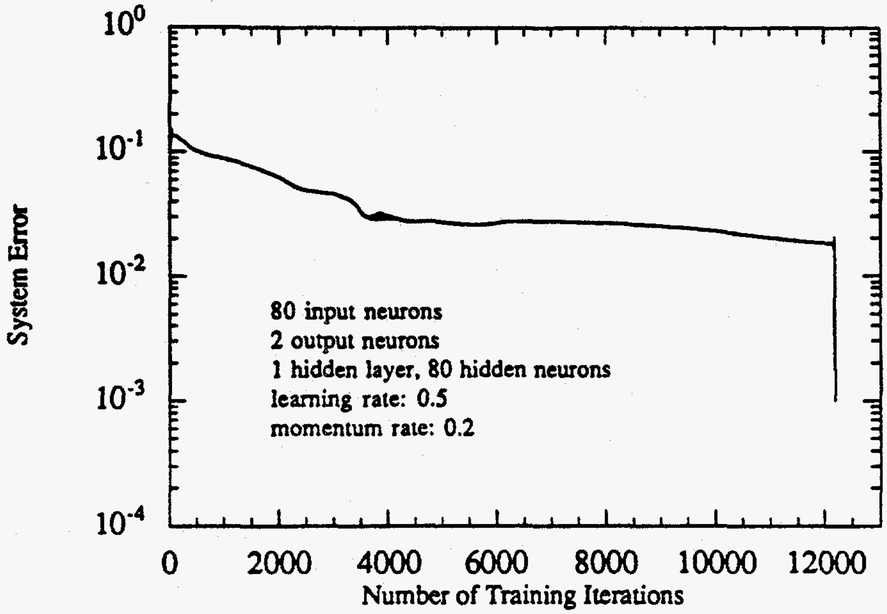

Figure 4.1 Training convergence for Detection Neural Network

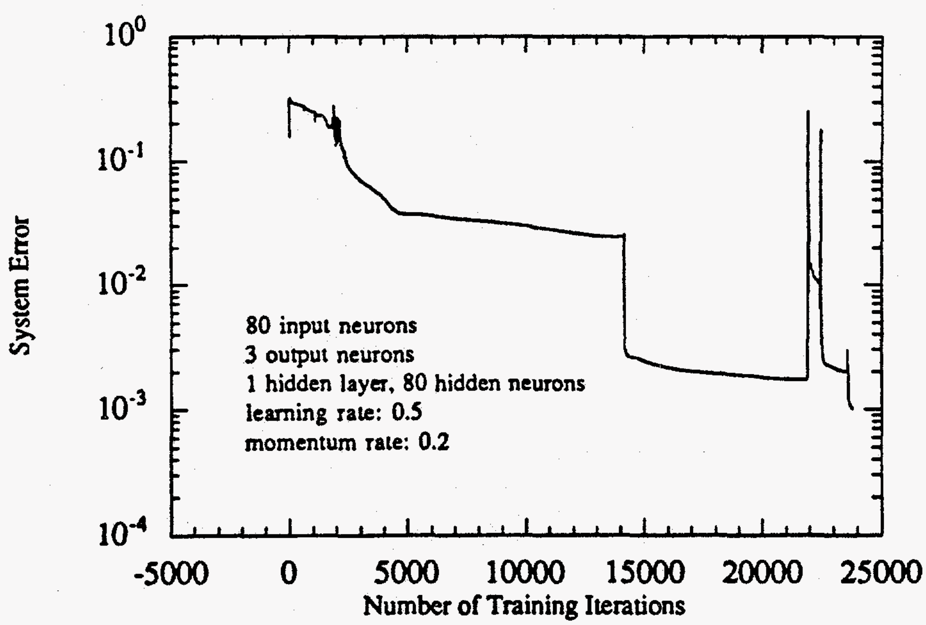

Figure 4.2 Training convergence for Identification Neural Network

APPENDIX C-1

Published Papers on the Use of Artificial Intelligence
in Severe Accident Diagnosis for PWR's

CONF papers removed for separate cycling

APPENDIX D

Influence of Organizational and Management Quality on Risk

I. INTRODUCTION

The effect of organizational factors on the risk of a nuclear power plant has received more and more attention in recent decades. The ultimate goal is to incorporate the influence of organizational factors into risk analysis. To achieve this goal, the following three questions must be answered:

(1) what are the organizational factors and what are their characteristics;
(2) how to measure these factors; and
(3) how to quantitatively include the impact of these factors into risk analysis.

To address these three questions, this year we have carried on work in three aspects. First, operational experiences expressed in the ASSET reports are examined in terms of twenty organizational dimensions (factors) proposed by the Brookhaven National Laboratory and Pennsylvania State University. Second, the Behaviorally Anchored Rating Scales (BARS) method has been used to develop the measurement scales for some of the categories of one important organizational factor, deep technical knowledge. BARS for seven subcategories of deep technical knowledge have been finished. This study provides a measurement method for an organizational factor and can be extended to some other organizational factors. Third, the most important step is how to use these measurement scales to include the influence of organizational factors into Probabilistic Safety Assessment (PSA). The Work Process Analysis Model (WPAM) has been proposed. It is an analysis tool to quantitatively include the impact of organizational factors on nuclear safety through the key work processes in a nuclear power plant. The algorithm for the design change work process has been developed and applied to modify one dominant sequence of Plant A.

II. OPERATIONAL EXPERIENCE STUDY

In the past, we studied the LERs of Plant A (LERs of 1988-1991) and the LERs in 1985's Precursor analysis , hoping to obtain some statistical data on organizational and management factors. But information in LERs is usually very brief and does not go deeply into organizational factor's root cause analysis in many cases. For this reason, we then searched other literature. Accident analysis reports and the ASSET reports are very useful in this sense. Three accident analysis reports and three ASSET report were chosen for detailed study initially. Recently, as a result of a series of research activities, a collection of twenty organizational dimensions (factors) has been identified [1, 2]. Brookhaven National Laboratory (BNL) has worked out the definitions for these dimensions (Appendix). These dimensions represent a comprehensive although overlapping taxonomy of organizational elements that relate to the safe operations of nuclear power plants. In the study of operational reports, we use these twenty organizational dimensions.

For several years the IAEA has offered its help to assist nuclear power plant operating organizations by means of the Assessment of Safety Significant Events Team (ASSET) program. More than twenty nuclear power installations have invited the IAEA to send ASSET teams of experts to perform reviews of operating experiences. The team prepares a written report to the nuclear installation, a report which focuses on the effectiveness of the organization in correcting problems, and, in particular, the depth and adequacy of root cause analysis. The purpose of ASSET is to review the operating organization and provide conclusions on the appropriateness and completeness of the planned and implemented corrective action. Generic lessons are drawn and suggestions are offered when necessary to improve plant management control on prevention of incidents and thus to enhance the overall level of operational safety.

The team provides a report (ASSET report) [3] which follows the same structure, and more importantly, it perform root cause analysis deep into the organization and management level; thus, the reports provide a very useful database for our study. The path we have chosen is to analyze each of a large number of ASSET reports in terms of twenty organizational dimensions, identifying those organizational dimensions which appear to play a significant role in the operating events chosen for detailed analysis by the ASSET teams. Other significant occurrences discussed in the ASSET reports are also examined. We look for patterns for each nuclear power plant (NPP) and correlations among them. We found a considerable correlation between these dimensions and the management related recommendations made by the ASSET teams.

2.1 ASSET Root Cause Analysis

Thirteen plants and a total of thirty events are analyzed. The results are listed in Table 1. In the table the numbers are relevant times of occurrences for each event. In the following, Leningrad (Russia) nuclear power plant event analysis is given in detail as an example. The words in italics are quoted from the ASSET reports. Our choice of dimensions for each occurrence follows immediately below.

Two notes need be made here. First, Safety Culture is only one of the twenty dimensions in BNL's taxonomy. Second, ASSET reports frequently suggest surveillance control or program deficiencies as a root cause in many events. We interpret this to mean that the plant's activities are not clearly defined and carried out, and attribute this to the dimension Roles-Responsibilities.

Event: Fuel damage followed by release of unfiltered gases outside the plant on March 24, 1992

Brief Description of the Event: On March 24, 1992, Unit 3 was operating at full power. The flow of water to one channel decreased sharply suddenly. The channel tube ruptured and in approximately 5 seconds high core cavity pressure initiated a fast emergency reactor trip and turbine generator shutdown, and closing of the flow of helium and nitrogen to the core cavity. In addition, the fuel channel integrity monitoring system showed high moisture levels in the core cavity.

Occurrence 1. Unexpected closing of regulating valve.
Problem Identification, Organizational Learning, Technical Knowledge
Direct cause of Occurrence 1:
(1) Deviation in the thermal treatment (manufacturing weakness). Cracks initiated.
Technical Knowledge
(2) Extra force applied to close valve (operating practices). Formalization, Technical Knowledge, Organizational Learning, Training
Root cause of Occurrence 1:
(1) Inadequate detection of manufacturing weaknesses. Incorrect quality control.
Roles-Responsibilities, Problem Identification, Safety Culture
(2) Inadequate detection of valve cracks in service (surveillance program).
Roles-Responsibilities, Resource Allocation

Occurrence 2. Lack of logic system for this situation.
Technical Knowledge, Problem Identification, Communication-Interdepartmental, Training
Direct cause of Occurrence 2:
(1) The automatic switch over from air release to filtered vent was not actuated by the core cavity overpressure signal (design).
Technical Knowledge
(2) The lack of logic system was not recognized by plant staff since start of operation.
Safety Culture, Problem Identification, Technical Knowledge

Root cause of Occurrence 2:

(1) The commissioning tests failed to check all the safety actuations that should work with the core cavity overpressure signal.
Technical Knowledge

(2) (Inadequate policy of Surveillance) The operating experience feed back had no opportunity to discover the lack of system.
Roles-Responsibilities, Communication-Interdepartmental

Occurrence 3. Procedures fail to give guidance for manual switch over to filtered vent.
Technical Knowledge, Formalization

Direct cause of Occurrence 3:

(1) Technological Procedures were not written to cope with failure or non-operation of Localization System, thus resulting in delay of air release to filtered vent.
Formalization, Technical Knowledge

(2) The complex shift organization and divided responsibilities made it difficult to recognize and act on problem.
Communication-Intradepartmental, Roles-Responsibilities

Root cause of Occurrence 3:

(1) Operations staff were not aware that Localization System logic would not operate for a channel rupture.
Technical Knowledge

(2) Lack of clear Design Basis document to clarify the systems operation, specifically for channel rupture.
Technical Knowledge, Roles-Responsibilities

2.2 Results and Conclusions

Though the number of samples is too small to give statistical data and to draw conclusions, the study still sheds some light. First, The twenty organizational factors in general are an adequate taxonomy. We are relatively satisfied with the definitions of the twenty organizational dimensions. We usually can relate the organizational weakness in the events to one or more dimensions. Second, for essentially every plant studied, Formalization, Roles-Responsibilities, Safety Culture, and Technical Knowledge played a role in each event. This is suggestive of a pattern which might warrant: (1) greater emphasis on these dimensions, and (2) a spot check at other aspects of the plant to see if these deficiencies are common. For a few plants (Kozloduy, Fessenheim) training was a contributor to each event, which is possibly suggestive of a deficiency. Problem Identification appeared frequently for Balakovo, Kozloduy, and Novoronezh.

III. ORGANIZATION FACTORS MEASUREMENT

After identifying the organizational factors which will impact the plant safety, the next question is how to measure these factors. Based on the previous operational experiences study, one important organizational factor, adequate or deep technical knowledge, has caught our attention and received detailed study as an example of developing measurement methodology. In an effort to provide an initial basis for further examination of deep technical knowledge, technical knowledge was divided into six broad categories, some of which are subdivided into two or three subcategories as follows:

(1) PRA: Level 1, Level 2, Level 3
(2) Details of Plant: Structures, Systems, Components
(3) Transient Behavior: Reactor Physics, Thermal Hydraulic
(4) Severe Accident Management
(5) Physical Science: Health Physics, Chemistry, Materials

(6) Safety Basis: Design Basis, Technical Specification, Regulation and Industry Standards

Currently, Structured Interview Protocol, Behavioral Checklist and Behavioral Anchored Rating Scales (BARS) are methods used in organizational factors measurement. BARS has been identified as a potentially valuable instrument in the measurement of various attributes important to an evaluation of the quality of organization. BARS is a performance evaluation device that incorporates behavioral examples with general performance dimensions. Specifically, each scale represents an area of performance (in this case one of the subcategories of deep technical knowledge). The behaviors are designed to facilitate the user's interpretation of poor, average, and high on each of the scales. In the development of BARS, experts are brought together to define the dimension and provide behavioral examples.

For each subcategory, a generic set of performance measures has been prepared, each representing a differing combination of aspects of deep technical knowledge applicable to the particular performance dimension. For the dimension "reactor physics", which is a subcategory of transients, ten performance measures, which take the place of the behavior examples usually formulated in a BARS application, were developed to make a list from which selections have been made to provide preliminary five-point BARS for ten different positions at the plant. The draft ten performance measures for subcategory "reactor physics" are the following:

(1) Good working knowledge (quantitative) of steady state and transient neutronics (kinetics and dynamics), including all relevant reactivity contributors, deep familiarity with all plant specific reactivity control features, reactivity accident potential (e.g. phenomenological course of ATWS) and criticality considerations under severe accident conditions, and quantitative grasp of the interaction between thermal-hydraulic and neutronic phenomena.

(2) Phenomenological understanding (semi-quantitative) of all important reactivity related effects in steady state, start-up, shutdown and accident conditions, including severe accident re-criticality, details of plant specific reactivity control features including indirect reactivity control effects. Capable of understanding the interaction between neutronic and thermal hydraulic phenomena.

(3) Capable of recognizing abnormal reactivity conditions, performing an estimated critical position, estimating the magnitude of changes in power associated with anticipated transients, (e.g. drop rods, loss of feedwater, etc.).

(4) Continuing familiarity with major relevant reactor physics concepts, (e.g. multiplication, burnup, fission product poisons, reactivity feedbacks), familiarity with reactor physics role in safety for specific plant.

(5) Capable of visualizing the plant response to change in reactivity due to plant activities (e.g. startup, shutdown), anticipate abnormal reactor states (e.g. high flux tilt, inoperable control rods, etc.) and thermal hydraulic effects on power (e.g. cool-water accident, loss of feedwater heating).

(6) Some familiarity with concepts of criticality, shutdown, reactivity feedback, reactivity transients, influence of system failure on ability to shutdown, understand safety function of critical components in systems important to reactivity control.

(7) Familiar with the concept of reactivity control (rods, boron, etc.), understands the important systems and components in controlling reactivity during normal plant operation and accident conditions.

(8) Some familiarity with the concept of fission, reactor control and systems for controlling the fission process, understands the importance of maintenance on reactivity control system, especially maintenance on redundant trains.

(9) Understands the basic fission process and concept of criticality. Can name the major systems related to shutdown of the reactor.

(10)Knows the plant uses nuclear energy as a heat source, can find his way through the plant, understands the concepts of safety (similar to a general employee).

The draft performance measures for the positions of shift technical advisor and maintenance foreman follow. (Note that an excellent rating is not appropriate for each position for each dimension.)

Shift Technical Advisor:

Excellent	1	Generic Measure(2)
	2	Generic Measure(3)
Good	3	Generic Measure(4)
	4	Generic Measure(5)
Poor	5	Generic Measure(5)

Maintenance Foreman:

Excellent	1	Generic Measure(5)
	2	Generic Measure(6)
Good	3	Generic Measure(7)
	4	Generic Measure(8)
Poor	5	Generic Measure(9)

The method has been applied thus far in draft form for seven dimensions (or subcategories) of deep technical knowledge: PRA level 1, PRA level 2, plant structures, plant systems, plant components, reactor physics and thermal hydraulic. Ten or twelve generic measures appeared to suffice for ten positions; however, it is anticipated several more generic measures would be useful to cover twenty different plant positions. The study on deep technical knowledge provides a feasible measurement method. This method can be extended to some other organizational factors.

IV. WORK PROCESS ANALYSIS MODEL FOR DESIGN CHANGE WORK PROCESS

Industrial experience and research have found that some nuclear power plants (NPP) with lower than average core melt frequencies had a poor regulatory performance as indicated by Nuclear Regulatory Commission (NRC) reports, such as Systematic Assessment of Licensee Performance ratings and Licensee Event Reports (LERs). The issue of the influence of organizational and management factors on risk has received more and more attention in recent decades. Work Process Analysis Model (WPAM) is an analysis tool to quantitatively include the impact of organizational factors on nuclear safety by building the link between these factors and probabilistic risk assessment (PRA) through the key work processes in nuclear power plants. While Davoudian et al's work 1, 2 gave the analysis for the corrective maintenance work process, this report focuses on the analysis of the design change work process. The organizational factors used in WPAM are the twenty organizational factors mentioned above.

As one of the safety related work processes in nuclear power plants, the design change work process is receiving increased emphasis in the nuclear industry. The purpose of this design change work process study is to identify generic problems relating to design control and modification activities, to determine how organizational and management factors influence these problems through the design change work process, and to develop an algorithm that qualitatively and quantitatively includes the organizational factors into the PRA of nuclear power plants.

4.1. The design change work process

Regulatory inspections and utility experience show that modification and design control are similar from plant to plant and repeat over time. In our study, the design change work process refers to the design control and modification activities of a nuclear power plant including: (1) field change: procedures and other documents modifications which do not alter plant functions, or design basis; (2) minor modification: minor design change activities which involve simple changes or small scopes of work; conceptual and preliminary engineering packages, formal cost estimating and design alternative consideration are not required; (3) design change: change other than above two. Figure 1 gives an actual plant design change work process flow chart. The design change work process typically involves five major steps: design change request initiation, review and scope assessment, package generation and approval, field implementation and document close out.

4.1.1 Task Initiation

Design change work process initiation generally occurs in one of four ways: (1) As a work request initiated, reviewed and approved in accordance with station engineering procedure and site work request procedure; (2) As the result of a licensing or other regulatory issue; (3) As a dispositioned nonconformance report, or station problem report assigned to the design organization for full or partial implementation, or a request for problem resolution; (4) Informal requests from station management for design engineering support.

Design change tasks are reviewed by the nuclear engineering design organization management/discipline manager, and assigned to the appropriate Group Supervisor. He/she assigns the task to a System Design Engineer, who is responsible for the conception, design and implementation of the assigned design task. The System Design Engineer is the designated system design engineer for the affected system, but may be a discipline specialist if a task is non-system specific.

4.1.2 Review And Scope Assessment

The System Design Engineer and Group Supervisor review the scope and complexity of the design activity and determine the appropriate package format. The package alternatives are the Field Notice Package for field change, the Minor Modification Package for minor modification or the Design Change Package for design change.

4.1.3 Package Generation And Approval

A. Field Change Notice Package Generation and Approval.

The generated package content typically includes: (1) Cover sheet form; (2) The supplement page, which is used to document the "BEFORE" and "AFTER" conditions of each affected drawing; (3) Configuration document check list and forms, which are used to identify other documents that are affected by the change; (4) Determination that the proposed change is bounded by the safety evaluation; and (5) Review of the proposed change against the design criteria. The package is reviewed, approved and closed in accordance with the relevant plant quality procedure.

B. Minor Modification Package Generation and Approval.

The content of generated Minor Modification Package mainly includes: (1) Cover sheet forms; (2) Description of change and an engineering evaluation; (3) A 10CFR50.59 safety evaluation; (4) License document impact; (5) Design criteria evaluation; (6) Interim Design Change Notice forms which are used to document the "BEFORE" and "AFTER" condition of each affected drawing; and (7) Identification of affected design calculations and special testing requirements. The package is reviewed and approved in accordance with the relevant plant quality procedure.

C. Design Change Package Generation and Approval.

Design Change Package usually includes Conceptual Engineering Package and Preliminary Engineering Package. They are generated, reviewed and approved in accordance with the relevant plant quality procedure.

A Conceptual Engineering Package is generated by the System Design Engineer which clearly describes the objective and reason for the change. A field walkdown should be performed by the System Design Engineer to identify any special field considerations such as accessibility, obstructions and interferences, and local environmental conditions. Alternate solutions should be developed and evaluated. The Conceptual Engineering Package should address the basis for the alternative, give sufficient explanation of the task for plant modification review and approval, and provide technical guidance for the generation of a Preliminary Engineering Package. The Conceptual Engineering Package should be numbered and routed by Technical Service-Document Control for review and approval by the Discipline Manager, other discipline Responsible Engineers as applicable, and the Station Technical Supervisor. Any comments generated in the course of this review should be resolved by the System Design Engineer. Based on the estimation contained in the Conceptual Engineering Package, the Cost and Schedule Organization will evaluate the task for expense/capital funding considerations.

The System Design Engineer will coordinate the generation of a Preliminary Engineering Package. When selecting materials or components, he/she should give preferential consideration to existing site warehouse stock. He/she is also responsible for generating the safety evaluation and design basis impact sections of the Package. All significant feasibility studies, analyses and assessment of design optimization alternatives should be completed prior to the issuance of the Package. The completed Preliminary Engineering Package should be numbered and routed for review by the affected discipline Responsible Engineers, the System Design Engineer, Discipline Manager and Station Technical Supervisor. After the Package has been distributed for review and comment, a station review meeting is then conducted. The System Design Engineer and system technical cognizant engineer should review and address any questions raised at that time.

After the Preliminary Engineering Package is completed, the Design Change Package is assembled. The Originator should conduct interim station review meetings to provide the opportunity for representatives from interfacing organizations and disciplines to have input to and provide status on design development activities. Coordination of a site walkdown of the change with Construction and Station Technical Engineering personnel is also required. Site Configuration control is responsible for distribution of the approved Design Change Package to site organizations which will be potentially impacted by the change. The following site organizations are included in this distribution: Operation, Maintenance, Training and Site Procedures Group.

4.1.4 Field Implementation

The following are required to support field implementation of the design change (as applicable): (1) approved Design Change Package, or Minor Modification Package, or Field Change Notice; (2) approved Technical Specification Changes License Amendments as required by the engineering evaluation; (3) field and engineered material received, inspected and staged; (4) Testing Procedures issued; and (5) plant conditions established. The Nuclear Construction (or Maintenance) department is responsible for coordination of construction and testing activities associated with the change. The System Design Engineer is responsible for the coordination and issuance of any field design changes required in the course of design change implementation. He/she is also responsible for the coordination of design support for as-built, routing, installation, detail or isometric drawings as required.

4.1.5 Document Closeout

Document closeout refers to that portion of the design change work process during which the documentation generated by package generation, installation, verification and testing processes is packaged, closed and filed. It also refers to the revision of related documentation to achieve consistency with as-built plant configurations.

4.2 WPAM for Design Change Work Process

4.2.1 Task Analysis

The predictable nature of the work processes suggests that a systematic analysis can be conducted to identify the characteristics of a given process and develop performance measures with respect to the strengths and weaknesses in the process. Furthermore, since work processes are closely related to plant performance, it is possible to conduct the analysis to facilitate the integration of organizational factors and PRA methodology. The first step of WPAM consists of qualitative analysis of a given work process: Task Analysis. It focuses on understanding: (1) Tasks that are involved in the work process and the plant personnel involved in each task; (2) Actions involved in each task and their failure modes; (3) The defenses or barriers involved in each task and their failure modes. The result of the task analysis is a cross-reference table. For the design change work process it is listed in Table 2.

Based on the task analysis, the organizational factors matrix for the studied work process is defined. The matrix shows the organizational factors that might influence the performance of each task in the work process. It is an assessment of the importance of organizational factors in the overall quality and efficiency of the work process. The organizational factors matrix for the design change work process is given in Table 3.

4.2.2 Candidate Parameter Groups

The goal of WPAM is to qualitatively include organizational factors into existing PRA. This can be achieved by either adding organizational factors into existing fault trees, or modifying existing fault tree entries. WPAM uses the latter. It is argued that the organizational factors are already in PRA because, first, human error analyses are already in PRA, and second, failure data used in PRA are plant specific with organizational factors already considered. For this reason, our study mainly focuses on the organizational dependent failures, which include direct organizational dependent failures and orga-nizational factor common cause failures. Direct organizational dependent failures are defined as the hardware failures caused by organizational factors and those human errors caused by organizational factors but not covered by existing human error study.

Organizational factors common cause failures are defined as the failures of two or more components (either identical or not identical) caused by same organizational factors.

Typically, PRA results include a set of dominant accident sequences presented in logical combinations of minimal cut sets (MCSs), which contain basic events, such as hardware failures and human errors. Different organizational factors influence a basic event in different ways and play different roles, i.e., the organizational factors have different influence weights for different basic events. For this reason, a group of generic parameters, called the Candidate Parameter Groups, is defined so that the influence weights of organizational factors for these parameters are not changed. For the design change work process, the candidate parameter groups are defined as those parameters which are associated with the design change work process, and to which failure modes in minimal cut sets are susceptible. A list of these parameters are:

(1) Failure due to hardware change (FHC);
(2) Failure due to hardware modification (FHM);
(3) Calibration procedures deficiency (CPD);
(4) Maintenance procedures deficiency (MPD);
(5) Operating procedures deficiency (OPD);
(6) Testing procedures deficiency (TPD).

These six candidate parameter groups are only preliminary. Some of them can be subdivided. For example, "failure due to hardware change" can be divided as "wrong material", "wrong system interaction", etc.

4.2.3 Analytic Hierarchy Process (AHP) Application

To find the influence weights of organizational factors for each candidate parameter group, the Analytic Hierarchy Process is used. AHP is a decision model which provides a method for multi-attributes to obtain priorities. The first step of AHP is to build a hierarchy for the system to study the functional interactions of its components and their impacts on the entire system. The second step is to take measurement and make judgment. A meaningful scale for the pairwise comparisons is adopted. Then a matrix calculation technique is used to obtain priorities. The details of this method can be found in Reference 7.

Based on the task analysis, the hierarchy for the design change work process can be developed (Figure 2). Our goal is to obtain the influence weights of organizational factors for each candidate parameter group. The first level of the hierarchy is the six candidate parameter groups of the design change work process. The second level is the tasks of the design change work process because the failure modes influenced by the organizational factors occur while performing these tasks. The third level is the plant personnel/positions involved in these tasks because their behaviors decide the quality of the tasks conducted. The last level is organizational factors, which influence the behavior of the personnel in the organization structure.

After the hierarchy is developed, experts are asked to assign the pairwise comparisons for the hierarchy. A one to nine scale is used in WPAM. Then, a computer code is developed to calculate the influence weights of the organizational factors for each candidate parameter group. The AHP results for the design change work process are given in Table 4.

4.2.4 Modification of Probabilities of MCSs

To modify the probability of minimal cut sets, we need to build a connection between minimal cut sets and the candidate parameter groups and convert expert judgments into probabilities. In general the core damage frequency contributed by a MCS can be expressed as

$$f_{MCS} f_{IE} \prod_{i}^{n} p_i \qquad (1)$$

where

f_{MCS} = the core damage frequency contributed by a MCS,
f_{IE} = the initiating event frequency,
p_i = the probabilities of basic events,
n = the number of basic events in a MCS.

Since our study focuses on the organizational dependent failures, p_1 is left alone. WPAM modifies the probabilities of the second or third events, given the first event has occurred, by considering the influence of organizational dependent failures. For a MCS with two events, for example, equation (1) is changed to

$$f_{MCS} = F_{IE} \cdot p_1 \cdot p_{2/1}$$

Here, $p_{2|1}$ is the conditional probability considering organizational dependent failure.

To find $p_{2|1}$, a modified Success Likelihood Index Model [8] (SLIM) is adopted. The SLIM is developed in human factor studies to convert expert judgments into probabilities. Experts evaluate the influenced performance factors with ratings and set the importance weights of these factors. The basic assumption of SLIM is that if the experts are correct, then the weighted average of the ratings is related to the probability of the success that would be observed in the long run in the situation of interest.

Before conducting the SLIM process, some definitions are given first as follows:

Influence weight $W_{CPG_{ij}}$ = the influence weight of ith organizational factor for ith candidate parameter group;
Independent weight $EW_{k,i}$ = the independent influence weight of jth organizational factor for jth event;
Common cause weight $CCW_{k,i}$ = the relative importance weight of the ith organizational factor for the kth event considering organizational factor dependent failures.

To conduct the SLIM process, the first step is to decide the independent weights for each event in a MCS, which are functions of the influence weights:

$$EW_{k,i} = f(W_{CPG_{i,j}}, \ldots, W_{CPG_{7,j}}) \qquad \text{for } i = 1-20, \quad k = 1-n \qquad (3)$$

The best way to obtain $EW_{k,i}$ is based on the percentage contribution data of all kinds of design changes in the studied plant. If the data are not sufficient, expert judgements can be used. The independent weights are equal to the weighted averages of the influence weights.

Considering the organizational dependent failures, the independent weights for the event 2 or 3 should change to common cause weights. Since the common cause weights mean the influence weights of organizational factors when both events occur, the similar concept of probability theory is used. This suggests multiplying the independent weights of two events to calculate the common cause weights, but here the normalization is needed:

$$CCW_{2|1,j} = \frac{EW_{i,j} \cdot EW_{2,j}}{\prod_{i}^{n} (EW_{i,j} \cdot EW_{2,j})} \qquad (4)$$

For a three event MCS, the common cause weight for the third event is calculated as

$$CCW_{3|1,2,j} = EW_{1j} \cdot EW_{3j} + EW_{2j} \cdot EW_{3j} + EW_{1j} \cdot EW_{2j} \cdot EW_{3j} \tag{5}$$

then normalized. Another way suggested in Reference 6 is to calculate $CCW_{3|1,\,i}$ and $CCW_{3|2,\,i}$ using equation (4). The higher value is used. It is not clear which way is better at present.

The second step of the SLIM process is to find the organizational factors performance ratings for the design change work process. The ratings are obtained using the organizational factor measurement methods developed in References 1 and 9 (method of this report Section III). The ratings given in Section III are the organizational performance ratings for the positions in the plant. Using the AHP hierarchy developed for the design change work process in Figure 2, the weights of the positions connected with the design change work process versus the candidate parameter groups of the design change work process can be obtained. The effective ratings for the design change work process can be calculated as:

$$R_{CPGj,i} = \sum_{Position} W_{CPGj,Position} \cdot R_{Position,i} \tag{6}$$

The ratings obtained from equation (6) are then averaged to obtain the final used ratings

$$R_i = \sum_{CPG} \frac{R_{CPG\,i}}{6}, \qquad \text{for} \quad i = 1-20 \tag{7}$$

The scale for R_i is from one to five. Five means the plant's organizational performance is perfect, while 1 means the worst.

In the next step we need to convert the common cause weights and the plant organizational performance ratings into the probability modification for the events in the MCS. For the second event of a MCS, let

$$p_{2|1} = \alpha p_2 \tag{8}$$

where α is a modifying coefficient, which is a function of the plant organizational performance ratings and the common cause weights

$$\alpha = F(CCW_{2|1,i}, R_i) \qquad \text{for} \quad j = 1-20 \tag{9}$$

The function F should satisfy the following conditions:

1) if $CCW_{2|1,\,i} = 1$, and $R_i = 5$, to simplify the problem, it is assumed in this case there is no organizational dependent failure occurring for perfect organizational performance, i.e., no modification for the original probability, $p_{2|1} = p_2$ and $\alpha = 1$;
2) if $CCW_{2|1,\,i} = 1$, and $R_i = 1$, that means the two events have largest organizational dependency and the plant performance is worst, in this case, the modification caused by this factor should be the largest, i.e., $p_{2|1} \rightarrow p_{max}$, $\alpha = \alpha_{max}$;
3) if $CCW_{2|1,\,i} \rightarrow 0$, that means the two events have little organizational dependency, then the probability would not change much, i.e., $p_{2|1} \rightarrow p_2$, $\alpha \rightarrow 1$. The first order Taylor approximation of the function F which satisfies these conditions is

$$\alpha=\prod_i\left[1+\frac{5-R_i}{4}(\alpha_{max}-1)\cdot CCW_{2IIj}\right] \tag{10}$$

where α_{max} is the maximum contribution of organizational dependency for the design change work process.

4.3 Algorithm Application and Results

The methodology developed in Section 4.2 has been used to analyze one of the dominant accident sequences in Plant A (a BWR). The sequence is initiated by Loss of Offsite Power and the reactor is scrammed, and subsequently onsite power is lost also. The safety relief valves open and reclose to relieve the pressure from the power imbalance caused when the turbine trips. The station blackout renders all core cooling systems inoperable except high pressure coolant injection (HPCI) system, reactor core isolation cooling (RCIC) system and fire protection system. Since the feedwater system cannot provide reactor make-up, reactor water level falls. At a certain reactor water level, HPCI and RCIC are automatically initiated. HPCI injects water to control core water level. Automatic switchover of HPCI suction from the condensate storage tank to the torus on high torus water level is bypassed. After the initial reflooding with water provided by HPCI, the operator may use HPCI or RCIC to provide reactor level control. HPCI is expected to fail after 8 hours because of battery depletion, with core damage after about 13 hours. This sequence results in late core damage and a vulnerable containment.

The reason for choosing this sequence to study is that 91% of the total internal core melt frequent (CDF) in Plant A is attributed to station blackout. The studied sequence has the highest frequency of 6.17E-07/year. It contributes approximately 37.2% the total CDF. The mean CDF of the plant internal events is 1.92E-06/year.

The sequence comprises 4384 minimal cut sets. The IPE (Individual Plant Examination) lists the top 150 minimal cut sets, which contribute 90.6% of the sequence frequency. WPAM is used to modify the probabilities of these 150 MCSs. There are a total of 54 different basic events in the 150 MCSs. Four of them are not associated with the design change work process. This leads to some MCSs which are not relevant to modification considering the influence of the organizational factors. After screening, 101 MCSs remained for further analysis.

Since the plant design change history data is not available from the IPE, expert judgment is used in determining the percentage contributions of the candidate parameter groups of the design change work process for each kind of basic event. Preliminary weights of these basic events versus the candidate parameter groups are given in Table 5. Each basic event is expressed with a code composed of four parts and sixteen characters. The parts are: three-character system identifier, two-character event or component type identifier, two-character failure mode identifier, and five-character unique event identifier. Table 6 lists the descriptions of these codes.

Using the weights in Table 4 and 5, the independent weights are calculated for each basic event in each MCS with Eqn. (3). The common cause weights are calculated using Eqn. (4). When modifying the second or third event in a MCS, there are two kinds of parameters in Eqn. (10): plant organizational performance ratings for the studied work process and the anchoring point α_{max}. The value of α_{max} is not clear at present because of a lack of data on organizational dependent failures. To see how the sequence frequency or core damage frequency changes with α_{max}, it is assumed that the plant has an average organizational performance rating of 3 for the all organizational factors. The new sequence frequencies and corresponding core damage frequencies (increase caused by this sequence), considering the organizational dependent failures in this case, are shown in Figure 3. $\alpha_{max} = 1$ means that there is no organizational dependent failure modification, and the values of sequence frequency and the core damage frequency are the same as the values in the IPE. As can be predicted, both sequence frequency and CDF increase almost linearly with the increase of α_{max}. This is because the modification approximation of Eqn. (10) is a linear function of α_{max} and ratings. From Figure 3 it can be seen that when $\alpha_{max} > 10$, the sequence frequency curve is almost the same as the CDF curve, that is, the organizational dependent failures are the dominant contributors to the CDF.

Sensitivity study is conducted for α_{max} ranging from one thousandth to 100 times of the original probabilities of the basic event. Figure 4 shows how the sequence frequency (top 150 MCSs) changes with α_{max}, given different ratings as a parameter. Figure 5 shows how the sequence frequency changes with ratings given different anchoring point α_{max} as a parameter. From the figure, if $\alpha_{max} < 1.1$, i.e., the probabilities of basic events increase one tenth considering organizational dependent failure, while the influence for the sequence frequency and core damage frequency can almost be ignored for the plant studied. This might be true, or the linear approximation may be too coarse to correctly reflect the influence of organizational factors in this range. Which conclusion is true requires more research on this issue and the collection of statistical organizational dependent failure data in the future.

Results in Figures 3 to 5 assume that all the modified event probabilities have the same anchoring point. This may be too conservative when α is large. The common cause weights are relative importance influence weights are and not absolute values. Some events may be loosely connected. For example, the MCS No. 14 in the sequence is:

T1 * ESW-MDP-FR-P2A * AC4-XHE-MC-UVRLB * NR-LOSP-13HR-TB1

The first basic event is failure to continue running of the emergency service water system pumps P2A, and the second basic event is the miscalibration of a relay on the AC electric power system bus. The failures involve different components, different failure modes, and the maintenance work belongs to different departments and follows different procedures. The chance is rare for these two events to occur because of design change and modification activities, and to be caused by the same organizational factors. For this reason, sensitivity study is performed to focus on individual components. There are two kinds of mechanical components in the 150 MCSs: pumps and valves (check valve and motor driven valve). There is no MCS in which at least two basic events are related to valves in the 150 MCSs. Therefore pump is chosen as individual component, and α_{max} is increased for pump relevant events while keeping $\alpha_{max} = 1$ for all other events. The increase of sequence frequency in this case is shown in Figures 6 and 7. For comparison, the results of increasing α_{max} for all events are also shown in these figures, that is, marked with "all". Since for the "pump" curve in the figures, only the dependent failures of one component is considered, the actual sequence frequency should drop between the "pump" curve and the "all" curve.

One point needs to be made here. To simplify the problem, the algorithm used in this report assumes that the basic event probability stays unchanged when the plant has the highest organizational performance ratings 5. This assumption may be too conservative. If generic data are for the average plants, the assumption should change to the basic event probability remaining unchanged when the organizational performance ratings of the plant equal to an average of 3. In this case, upper and lower anchoring points are needed.

V. CONCLUSION

The result of this report offers some new insights on the influence of organizational factors on safety. Organizational factors play a very important role in nuclear plant safety. The work process approach gives in-depth understanding of how organizational factors affect human behavior in an organization and finally impact safety. The algorithm proposed here provides a method to quantitatively include the influence of organizational factors on safety into PRA.

REFERENCES

[1] JACOBS R., HABER S., "Organizational Processes and Nuclear Power Plant Safety", Reliability Engineering and System Safety, 45, (1994) 75-83.

[2] SHURBERG D.A., HABER S., Techniques to Assess Organizational factors: Progress to date, BNL-A-3956-1-7/94, 1994.

[3] ASSET guideline Revised 1991 Edition, IAEA-TECDOC-632, December 1991.

[4] Report of ASSET mission to the Leningrad nuclear power plant, IAEA-NEWS/ASSET/93/R/03, 1993.

[5] K. Davoudian, J. S. Wu and G. Apostolakis, "Incorporating organizational factors into risk assessment through the analysis of work processes," Reliability Engineering and System Safety, 45 85 (1994).

[6] K. Davoudian, J. S. Wu, and G. Apostolakis, "The work process analysis model (WPAM)," Reliability Engineering and System Safety, 45 107 (1994).

[7] T. L. Saaty, The Analytic Hierarchy Process: Planning, Priority Setting, Resource Allocation, McGraw-Hill, New York, (1980).

[9] D. E. Embrey, P. C. Humphreys, E. A. Rosa, B. Kirwan and K. Rea, SLIM-MAUD: An approach to assessing human error probabilities using structured expert judgment, NUREG/CR-3518, US Nuclear Regulatory Commission, Washington, D.C., USA, (1984).

[10]D. Okrent, E. C. Abbott, D. Leonard and Y. Xiong, "Use of behaviorally anchored rating scales (BARS) for deep technical knowledge", In Proceedings of the Probabilistic Safety assessment International Topical Meeting, PSA'93, Clearwater FL, American Nuclear Society, La Grange Park, IL, USA, pp. 234-239, (1993).

Appendix Definitions of the Organizational and Management Factors

Centralization: Centralization refers to the extent to which decision-making and/or authority is localized in one area or among certain people or groups.

Communication-External: External communication refers to the exchange of information, both formal and informal, between the plants, its parent organization, and external organizations (e.g., NRC, state, and public).

Communication-Interdepartmental: Interdepartmental communication refers to the exchange of information, both formal and informal, between the different departments or units within the plant. It includes both the top-down and bottom-up communication networks.

Communication-Intradepartmental: communication refers to the exchange of information, both formal and informal, within a given department or unit in the plant. It includes both the top-down and bottom-up communication networks.

Coordination of Work: Coordination of work refers to the planning, integration, and implementation of the work activities of individuals and groups.

Formalization: Formalization refers to the extent to which there are well-identified rules, procedures, and/or standardized methods for routine activities as well as unusual occurrences.

Goal Prioritization: Goal prioritization refers to the extent to which plant personnel understand, accept, and agree with the purpose and relevance of goals.

Organizational Culture: Organizational culture refers to plant personnel's shared perceptions of the organization. It includes the traditions, values, customs, practices, goals, and socialization processes that endure over time and that distinguish an organization from others. It defined the "personality" of the organization.

Organizational Learning: Organizational learning refers to the degree to which plant personnel and the organization use knowledge gained from past experiences to improve future performance.

Organizational Knowledge: Organizational knowledge refers to the understanding plant personnel have regarding the interactions of organizational subsystems and the way in which work is actually accomplished within the plant.

Ownership: Ownership refers to the degree to which plant personnel take personal responsibility for their actions and the consequences of the actions. It also includes commitment to and pride in the organization.

Performance Evaluation: Performance evaluation refers to the degree to which plant personnel are provided with fair assessments of their work-related behaviors. It includes regular feedback with an emphasis on improvement of future performance.

Personnel Selection: Personnel selection refers to the degree to which the plant personnel are identified with the requisite knowledge, experiences, skills, and abilities to perform a given job.

Problem Identification: Problem identification refers to the extent to which the organization encourages plant personnel to draw upon knowledge, experience, and current information to identify problems.

Resource Allocation: Resource allocation refers to the manner in which the plant distributes its financial resource. It includes both the actual distribution of resource as well as individual perceptions of this distribution.

Roles-Responsibilities: Roles and responsibilities refer to the degree to which plant personnel and departmental work activities are clearly defined and carried out.

Safety Culture: Safety culture refers to the characteristics of the work environment, such as the norms, rules, and common understandings, that influence plant personnel's perceptions of the importance that the organization places on safety. It includes the degree to which a critical, questioning attitude exists that is directed towards plant improvement.

Technical Knowledge: Technical knowledge refers to the depth and breath of requisite understanding plant personnel have regarding plant design and systems, and of phenomena and events that bear on plant safety.

Time Urgency: Time urgency refers to the degree to which plant personnel perceive schedule pressure while completing various tasks.

Training: Training refers to the degree to which plant personnel are provided with the requisite knowledge and skills to perform tasks safely and effectively. It also refers to plant personnel perceptions regarding the general usefulness of the training programs.

Table 1 Organizational Weaknesses of the Plants

PLANT	Angra	Balakovo		Bohunice	Dungeness			Fessenheim			Ignalina		Khmelnitsky		
EVENT		A	B		A	B	C	A	B	C	A	B	A	B	C
Centralization															
Communication-External				1											
Communication-Interdepartmental		1										2			
Communication-Intradepartmental															
Coordination of Work															
Formalization	1	1	2	2	2	2	2	1	1	4	3	1	2	3	2
Goal Prioritization															
Organizational Culture															
Organizational Knowledge												1			1
Organizational Learning								1		1			1	2	
Ownership		1		1	2		1	1		1		2		2	2
Performance Evaluation				1						1		1			1
Personnel Selection									1			1			
Problem Identification		1	4	1		2				1			1	1	
Resource Allocation			1					2							
Roles-Responsibilities	1		2	2	2	1	1	1	1	3	2	2	1	2	3
Safety Culture	1	1	3	2	2	2	2	2	1	3	4	3	2	2	1
Technical Knowledge		1	2	1	2	1	2	3	1	3	2	3	1	3	2
Time Urgency															
Training		1		1	1			1	1	1				1	1

Task	Action/Barrier	Department	Personnel
Design Change Task Initiation	Document Assembly	Nuclear Eng. Design Organization (NEDO)	System Design Engineer (SDE)
	Task Initiation	Various Dept.	Variable
	Review	NEDO	NEDO Manager
Review and scope assessment	Review	NEDO	Group Supervisor (GS) SDE
Package Generation and Approval	Conceptual Engineering Package (CEP) Generation	NEDO and various Dept.	SDE and variable
	CEP Review and Approval	NEDO, Station Operation (SO), NES&L* Dept. Nuclear Generating Site (NGS) Dept., Design Review Committee (DRC), PMRC**	SDE, GSs, Discipline Manager (DM) Discipline Responsible Engineer (DRE) Technical Supervisor Engineer (TSE) Independent Review Engineer (IRE)
	Preliminary Engineering Package (PEP) Generation	NEDO and various Dept.	SDE and variable
	PEP Review and Approval	NEDO, SO, PMRC, NES&L Dept. NGS Dept	SDE, GSs, DM, DRE, TSE, IRE
	Meetings	Representative of Review Org.	SDE, DREs
	Design Change Package or Minor Modification Package	NEDO Construction Dept. Operation Maintenance	SDE, IRE, Integrated Plant Review Engineer and variable
Field Implementation	Document Assembly		
	Execution	Nuclear Construction Maintenance	Variable

* NES&L: Nuclear Engineering Safety and Licensee
**PMRC: Plant Modification Review Committee

Table 2 The Cross Reference Table for the Design Change Work Process

	Design Change Initiation				Package Generation and Approval						Imple-menta-tion		
	Document Assembly	Task Initiation	Review	Review and Scope Assessment	CEP Generation	CEP Review and Approval	PEP Generation	PEP Review and Approval	Meetings	DCP G, R & A	Document Assembly	Execution	Documentation
Centralization		X	X	X	X	X	X	X	X	X			
Communication-External													
Communication-Interdepartmental	X	X		X							X	X	
Communication-Intradepartmental		X			X		X			X	X		
Coordination of Work		X			X		X			X	X	X	
Formalization	X		X	X	X		X			X	X	X	
Goal Prioritization			X	X									
Organizational Culture	X	X	X	X	X	X	X	X	X	X	X	X	X
Organizational Knowledge		X			X		X			X		X	
Organizational Learning					X		X			X		X	
Ownship		X	X	X	X		X			X		X	
Performance Evaluation		X											
Personnel Selection			X	X	X		X			X		X	
Problem Identification		X	X	X		X		X	X	X			
Resource Allocation					X	X	X	X	X	X			
Roles-Responsibilities	X	X	X	X	X	X	X	X	X	X	X	X	X
Safety Culture		X	X	X	X	X	X	X	X	X		X	X
Technical Knowledge	X	X	X	X	X	X	X	X	X	X	X	X	X
Time Urgency		X	X	X	X		X			X	X	X	
Training		X	X		X		X			X		X	

Table 3 The Organizational Factors Matrix for the Design Change Work Process

Table 4 AHP Results: Final Candidate-Parameter-Group Weights for Design Change Work Process

	FHC	FHM	CPD	MPD	OPD	TPD
Centralization						
Communication-External						
Communication-Interdepartmental	.0634	.0634	.0676	.0676	.0676	.0676
Communication-Intradepartmental	.0856	.0856	.0907	.0907	.0907	.0907
Coordination of Work	.0697	.0697	.0693	.0693	.0693	.0693
Formalization	.1711	.1711	.1640	.1640	.1640	.1640
Goal Prioritization						
Organizational Culture						
Organizational Learning	.0891	.0891	.0856	.0856	.0856	.0856
Organizational Knowledge	.0916	.0916	.0826	.0826	.0826	.0826
Ownership						
Performance Evaluation						
Personnel Selection						
Problem Identification						
Resource Allocation						
Roles-Responsibilities						
Safety Culture	.0030	.0030	.0034	.0034	.0034	.0034
Technical Knowledge	.3454	.3454	.3483	.3483	.3483	.3483
Time Urgency						
Training	.0811	.0811	.0885	.0885	.0885	.0885

Table 5 Weights of Basic Events Versus CPGs

BASIC EVENTS	FHC	FHM	CPD	MPD	OPD	TPD
ESW-CCF-FR-PUMPS	0.1	0.4	0.1	0.3		0.1
ESW-CCF-FS-PUMPS	0.1	0.4	0.1	0.3		0.1
ESW-CCF-OO-102AB	0.1	0.4	0.1	0.3		0.1
EDG-CCF-HW-4EDGS	0.1	0.4	0.1	0.3		0.1
EDG-CCF-HW-EDGAC	0.1	0.4	0.1	0.3		0.1
AC4-XHE-MC-UVRLA AC4-XHE-MC-UVRLB			1.0			
ESW-XHE-RE-ESW3A ESW-XHE-RE-ESW3B ESW-XHE-RE-P2A ESW-XHE-RE-P2B						1.0
AC6-SBR-DN-EP2A AC6-SBR-DN-EP2B		0.2		0.8		
ESW-CKV-CC-ESW1A ESW-CKV-CC-ESW1B ESW-CKV-CC-ESW6A ESW-CKV-CC-ESW6B	0.1	0.3	0.1	0.4		0.1
ESW-MDP-FR-P2A ESW-MDP-FR-P2B	0.1	0.3	0.1	0.4		0.1
ESW-MDP-FS-P2A ESW-MDP-FS-P2B	0.1	0.3	0.1	0.4		0.1
ESW-MOV-OO-102A ESW-MOV-OO-102B	0.1	0.3	0.1	0.4		0.1
AC4-RCI-FE-94EA3 AC4-RCI-FE-94EB3		0.2		0.7		0.1
ESW-RCI-FE-A42C ESW-RCI-FE-B42C ESW-RCI-FE-A63A ESW-RCI-FE-B63A		0.2		0.7		0.1
ESW-RCS-OO-A63A9 ESW-RCS-OO-B63A9		0.2		0.7		0.1

Table 6 Descriptions of Basic Events Codes

Code	Description
ESW-CCF-FR-PUMPS	Common cause failure of ESW pumps to run
ESW-CCF-FS-PUMPS	Common cause failure of ESW pumps to starts
ESW-CCF-OO-102AB	Common cause failure of 46MOV-102A/B to close on demand
EDG-CCF-HW-4EDGS	Common cause failure of EDGS A, B, C and D
EDG-CCF-HW-EDGAC	Common cause failure of EDGS A and C
ESW-XHE-RE	Failure to restore valve 46ESW-3A/B after test
ESW-XHE-MC	Miscalibration of bus 10500/10600 UV relays
AC6-SBR-DN	Circuit breaker does not operate
ESW-CKV-CC	Check valve normal close does not open
ESW-MDP-FR	Motor drive pumps 46P-2A/B fail to continue running
ESW-MDP-FS	Motor drive pumps fail to start
ESW-MOV-OO	Motor operate valve normal open fails to close
AC4-RCI-FE	Electric (relay) coil does not energize
ESW-RCI-FE	Electric (relay) coil does not energize
ESW-RCK-NO	Control circuit no output
AC4-RCS-OO	Contacts, normal open fail to close
ESW-RCS-OO	Contacts, normal open fail to close
AC4-RLY-NO	Relay no output
DC1-BAT-HW	Battery failure
DC1-BDC-ST	Panel faults at any load
AC4	AC Electric power systems: 4.16KVac
AC6	AC Electric power systems: 600Vac
DC1	DC Electric power systems: 125Vdc
EDG	Emergency diesel generators
ESW	Emergency service water system

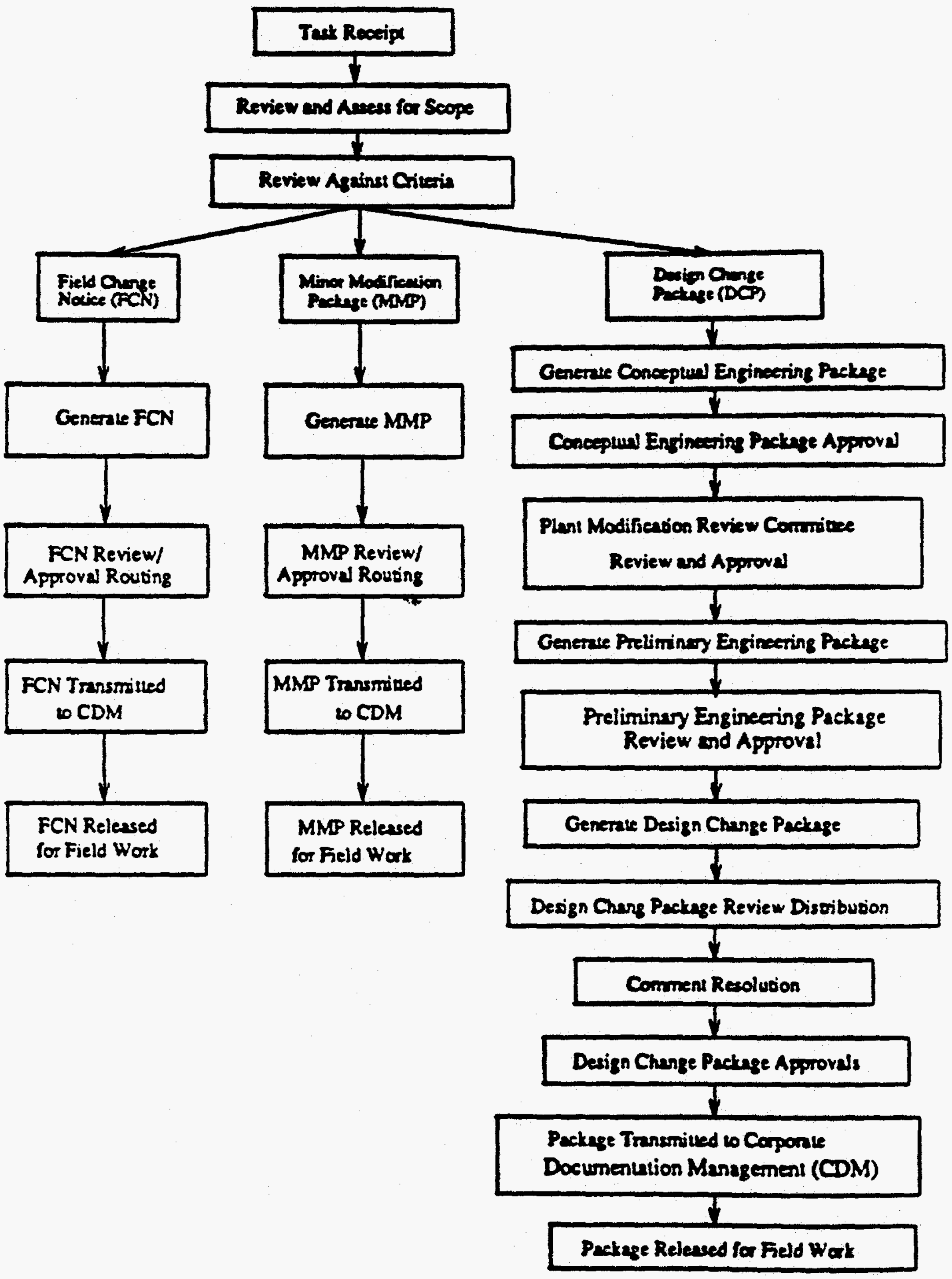

Figure 1 Design Change Work Process Flowchart

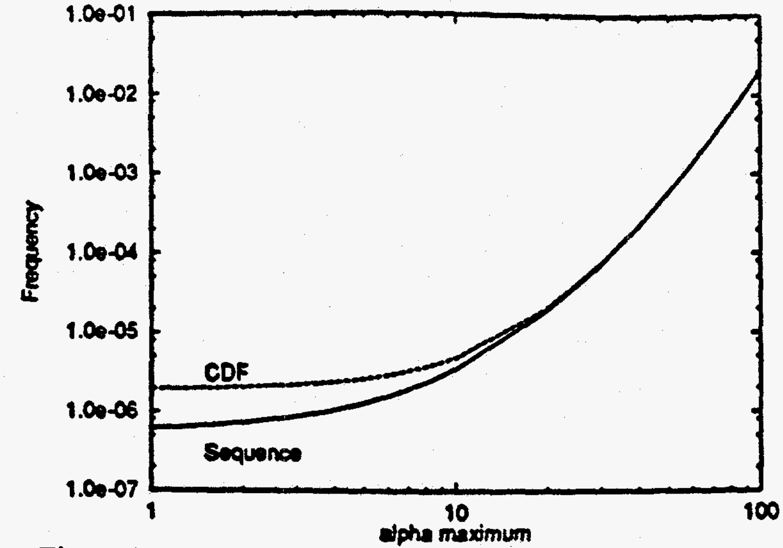

Figure 3 Sequence Frequency and CDF Versus α_{max} with Ratings = 3

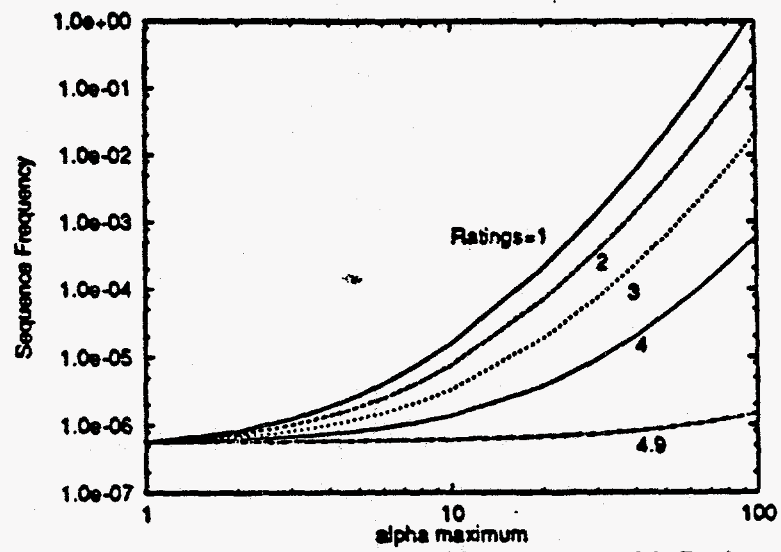

Figure 4 Sequence Frequency Versus α_{max} with Ratings as Parameter

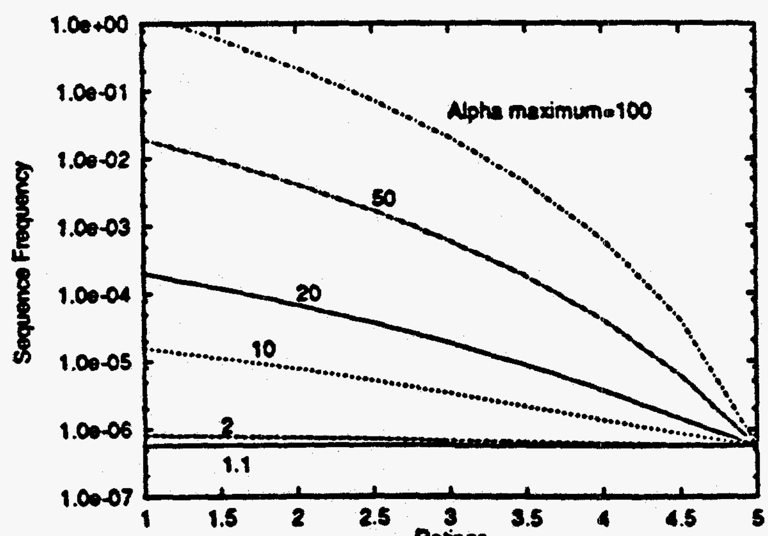

Figure 5 Sequence Frequency Versus Ratings with α_{max} as Parameter

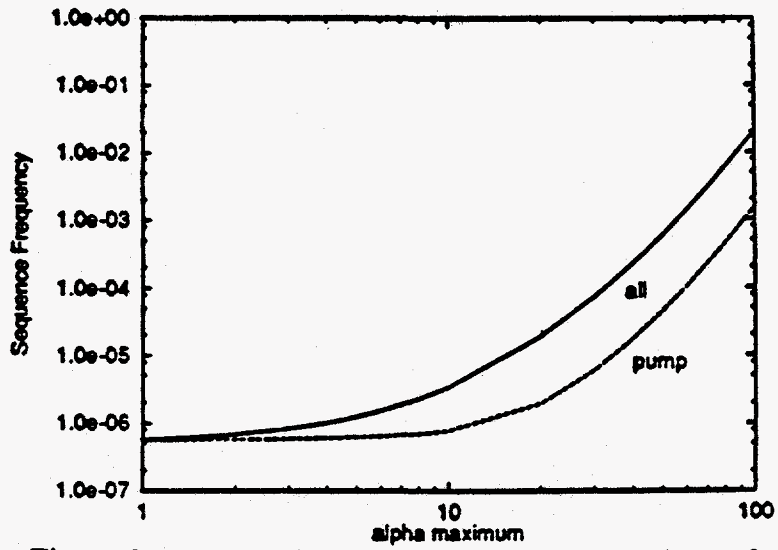

Figure 6 Sequence Frequency Versus α_{max} (Ratings =3)

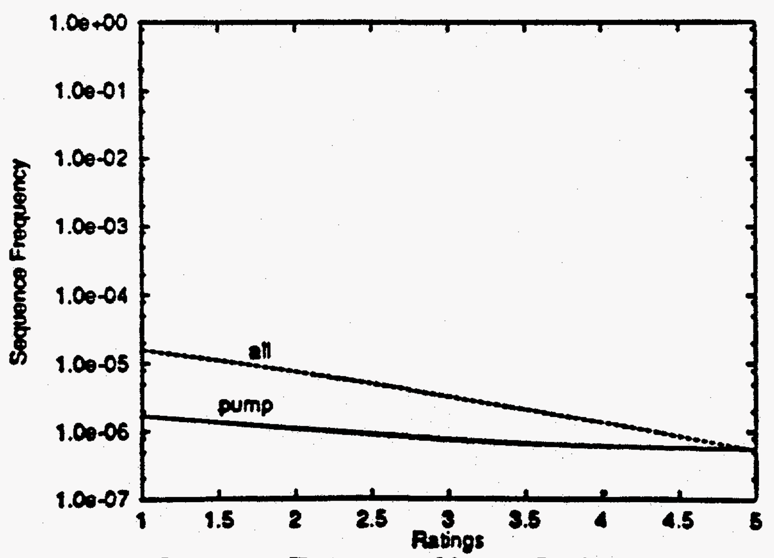

Figure 7 Sequence Frequency Versus Ratings (α_{max} = 10)

Table 1. (Continued) Organizational Weaknesses of the Plants

PLANT	Kola	Kozloduy					Leningrad	Novovoronezh				Paluel	Smolensk		
EVENT		A	B	C	D	E		A	B	C	D		A	B	C
Centralization				1											
Communication-External															
Communication-Interdepartmental	1						1								
Communication-Intradepartmental	1						1								
Coordination of Work	1														
Formalization		2	1	1	1	1	2	1	2	1	1		2	2	2
Goal Prioritization														1	
Organizational Culture										1	1				
Organizational Knowledge								2		1	1	1			
Organizational Learning							1							2	
Ownership				2		1		3		1	1	1	1	1	
Performance Evaluation	1		1	2		1		2						1	
Personnel Selection	1			1		1		1				1			
Problem Identification		1		3	1	1	2	1	2	1		1	3	1	
Resource Allocation						1	1								
Roles-Responsibilities	1	3	1	3	1	2	2	2		2	1	1	2	2	
Safety Culture		3	1	3	1	1	2	3	3	1	2	1	2	4	1
Technical Knowledge	2	3	1	3	1	2	3	2	1	2	2	4	2	4	2
Time Urgency				1										1	
Training	1	1	1	2	1	1	2	1		1		1		1	1

Table 1. Organizational Weaknesses of the Plants

PLANT	Angra	Balakovo		Bohunice	Dungeness			Fessenheim			Ignalina		Khmelnitsky		
EVENT		A	B		A	B	C	A	B	C	A	B	A	B	C
Centralization															
Communication-External				1											
Communication-Interdepartmental		1		1								2			
Communication-Intradepartmental															
Coordination of Work															
Formalization	1	1	3	2	2	2	2	1	1	4	3	1	2	3	2
Goal Prioritization															
Organizational Culture															
Organizational Knowledge											2	1			2
Organizational Learning								1		1			2	2	
Ownership		1		2	2		1	1	1	3		2		3	2
Performance Evaluation				1						1		1			2
Personnel Selection									1			1			
Problem Identification	1	1	3	1		2			1	2	1	1	2	1	
Resource Allocation			1					2							
Roles-Responsibilities	1	1	3	2	2	1	2	1	1	3	2	3	1	2	3
Safety Culture	1	1	2	4	2	3	1	2	1	3	3	3	2	3	1
Technical Knowledge		2	4	1	2	2	1	3	4	3	2	4	1	3	2
Time Urgency															
Training			1	1	1			1	2	1		1		2	2

Task	Action/Barrier	Department	Personnel
Design Change Task Initiation	Document Assembly	Nuclear Eng. Design Organization (NEDO)	System Design Engineer (SDE)
	Task Initiation	Various Dept.	Variable
	Review	NEDO	NEDO Manager
Review and scope assessment	Review	NEDO	Group Supervisor (GS) SDE
Package Generation and Approval	Conceptual Engineering Package (CEP) Generation	NEDO and various Dept.	SDE and variable
	CEP Review and Approval	NEDO, Station Operation (SO), NES&L* Dept. Nuclear Generating Site (NGS) Dept., Design Review Committee (DRC), PMRC**	SDE, GSs, Discipline Manager (DM) Discipline Responsible Engineer (DRE) Technical Supervisor Engineer (TSE) Independent Review Engineer (IRE)
	Preliminary Engineering Package (PEP) Generation	NEDO and various Dept.	SDE and variable
	PEP Review and Approval	NEDO, SO, PMRC, NES&L Dept. NGS Dept	SDE, GSs, DM, DRE, TSE, IRE
	Meetings	Representative of Review Org.	SDE, DREs
	Design Change Package or Minor Modification Package	NEDO Construction Dept. Operation Maintenance	SDE, IRE, Integrated Plant Review Engineer and variable
Field Implementation	Document Assembly		
	Execution	Nuclear Construction Maintenance	Variable

* NES&L: Nuclear Engineering Safety and Licensee
**PMRC: Plant Modification Review Committee

Table 2. The Cross Reference Table for the Design Change Work Process

	Design Change Initiation				Package Generation and Approval						Implementation		
	Document Assembly	Task Initiation	Review	Review and Scope Assessment	CEP Generation	CEP Review and Approval	PEP Generation	PEP Review and Approval	Meetings	DCP G, R & A	Document Assembly	Execution	Documentation
Centralization		X	X	X	X	X	X	X	X	X			
Communication-External													
Communication-Interdepartmental	X	X		X							X	X	
Communication-Intradepartmental		X			X		X			X	X		
Coordination of Work		X			X		X			X	X	X	
Formalization	X		X	X	X		X			X	X	X	
Goal Prioritization			X	X									
Organizational Culture	X	X	X	X	X	X	X	X	X	X	X	X	X
Organizational Knowledge		X			X		X			X		X	
Organizational Learning					X		X			X		X	
Ownship		X	X	X	X		X			X		X	
Performance Evaluation		X											
Personnel Selection			X	X	X		X			X		X	
Problem Identification		X	X	X		X		X	X	X			
Resource Allocation					X	X	X	X	X	X			
Roles-Responsibilities	X	X	X	X	X	X	X	X	X	X	X	X	X
Safety Culture		X	X	X	X	X	X	X	X	X		X	X
Technical Knowledge	X	X	X	X	X	X	X	X	X	X	X	X	X
Time Urgency		X	X	X	X		X			X	X	X	
Training		X	X		X		X			X		X	

Table 3. The Organizational Factors Matrix for the Design Change Work Process

Table 4. AHP Results: Final Candidate-Parameter-Group Weights for Design Change Work Process

	FHC	FHM	CPD	MPD	OPD	TPD
Centralization						
Communication-External						
Communication-Interdepartmental	.0634	.0634	.0676	.0676	.0676	.0676
Communication-Intradepartmental	.0856	.0856	.0907	.0907	.0907	.0907
Coordination of Work	.0697	.0697	.0693	.0693	.0693	.0693
Formalization	.1711	.1711	.1640	.1640	.1640	.1640
Goal Prioritization						
Organizational Culture						
Organizational Learning	.0891	.0891	.0856	.0856	.0856	.0856
Organizational Knowledge	.0916	.0916	.0826	.0826	.0826	.0826
Ownership						
Performance Evaluation						
Personnel Selection						
Problem Identification						
Resource Allocation						
Roles-Responsibilities						
Safety Culture	.0030	.0030	.0034	.0034	.0034	.0034
Technical Knowledge	.3454	.3454	.3483	.3483	.3483	.3483
Time Urgency						
Training	.0811	.0811	.0885	.0885	.0885	.0885

Table 5. Weights of Basic Events Versus CPGs

BASIC EVENTS	FHC	FHM	CPD	MPD	OPD	TPD
ESW-CCF-FR-PUMPS	0.1	0.4	0.1	0.3		0.1
ESW-CCF-FS-PUMPS	0.1	0.4	0.1	0.3		0.1
ESW-CCF-OO-102AB	0.1	0.4	0.1	0.3		0.1
EDG-CCF-HW-4EDGS	0.1	0.4	0.1	0.3		0.1
EDG-CCF-HW-EDGAC	0.1	0.4	0.1	0.3		0.1
AC4-XHE-MC-UVRLA AC4-XHE-MC-UVRLB			1.0			
ESW-XHE-RE-ESW3A ESW-XHE-RE-ESW3B ESW-XHE-RE-P2A ESW-XHE-RE-P2B						1.0
AC6-SBR-DN-EP2A AC6-SBR-DN-EP2B		0.2		0.8		
ESW-CKV-CC-ESW1A ESW-CKV-CC-ESW1B ESW-CKV-CC-ESW6A ESW-CKV-CC-ESW6B	0.1	0.3	0.1	0.4		0.1
ESW-MDP-FR-P2A ESW-MDP-FR-P2B	0.1	0.3	0.1	0.4		0.1
ESW-MDP-FS-P2A ESW-MDP-FS-P2B	0.1	0.3	0.1	0.4		0.1
ESW-MOV-OO-102A ESW-MOV-OO-102B	0.1	0.3	0.1	0.4		0.1
AC4-RCI-FE-94EA3 AC4-RCI-FE-94EB3		0.2		0.7		0.1
ESW-RCI-FE-A42C ESW-RCI-FE-B42C ESW-RCI-FE-A63A ESW-RCI-FE-B63A		0.2		0.7		0.1
ESW-RCS-OO-A63A9 ESW-RCS-OO-B63A9		0.2		0.7		0.1

Table 5. (Continued) Weights of Basic Events Versus CPGs

BASIC EVENTS	FHC	FHM	CPD	MPD	OPD	TPD
ESW-RCK-NO-P2A ESW-RCK-NO-P2B ESW-RCK-NO-102A ESW-RCK-NO-102B		0.2	0.2	0.3		0.3
AC4-RCS-OO-A63A9 AC4-RCS-OO-B63A9 AC4-RCS-OO-94EA3 AC4-RCS-OO-94EB3		0.2		0.5		0.3
AC4-RLY-NO-HOEA1 AC4-RLY-NO-HOEB1 AC4-RLY-NO-HOEA3 AC4-RLY-NO-HOEB3		0.2		0.7		0.1
DC1-BAT-HW-BATTA		0.2		0.6		0.2
DC1-BDC-ST-BCB2A DC1-BDC-ST-DC-A2 DC1-BDC-ST-DC-A3 DC1-BDC-ST-DC-B3 DC1-BDC-ST-DC-A4 DC1-BDC-ST-DC-B4		0.1	0.4	0.2		0.3

Table 6. Descriptions of Basic Events Codes

Code	Description
ESW-CCF-FR-PUMPS	Common cause failure of ESW pumps to run
ESW-CCF-FS-PUMPS	Common cause failure of ESW pumps to starts
ESW-CCF-OO-102AB	Common cause failure of 46MOV-102A/B to close on demand
EDG-CCF-HW-4EDGS	Common cause failure of EDGS A, B, C and D
EDG-CCF-HW-EDGAC	Common cause failure of EDGS A and C
ESW-XHE-RE	Failure to restore valve 46ESW-3A/B after test
ESW-XHE-MC	Miscalibration of bus 10500/10600 UV relays
AC6-SBR-DN	Circuit breaker does not operate
ESW-CKV-CC	Check valve normal close does not open
ESW-MDP-FR	Motor drive pumps 46P-2A/B fail to continue running
ESW-MDP-FS	Motor drive pumps fail to start
ESW-MOV-OO	Motor operate valve normal open fails to close
AC4-RCI-FE	Electric (relay) coil does not energize
ESW-RCI-FE	Electric (relay) coil does not energize
ESW-RCK-NO	Control circuit no output
AC4-RCS-OO	Contacts, normal open fail to close
ESW-RCS-OO	Contacts, normal open fail to close
AC4-RLY-NO	Relay no output
DC1-BAT-HW	Battery failure
DC1-BDC-ST	Panel faults at any load
AC4	AC Electric power systems: 4.16KVac
AC6	AC Electric power systems: 600Vac
DC1	DC Electric power systems: 125Vdc
EDG	Emergency diesel generators
ESW	Emergency service water system

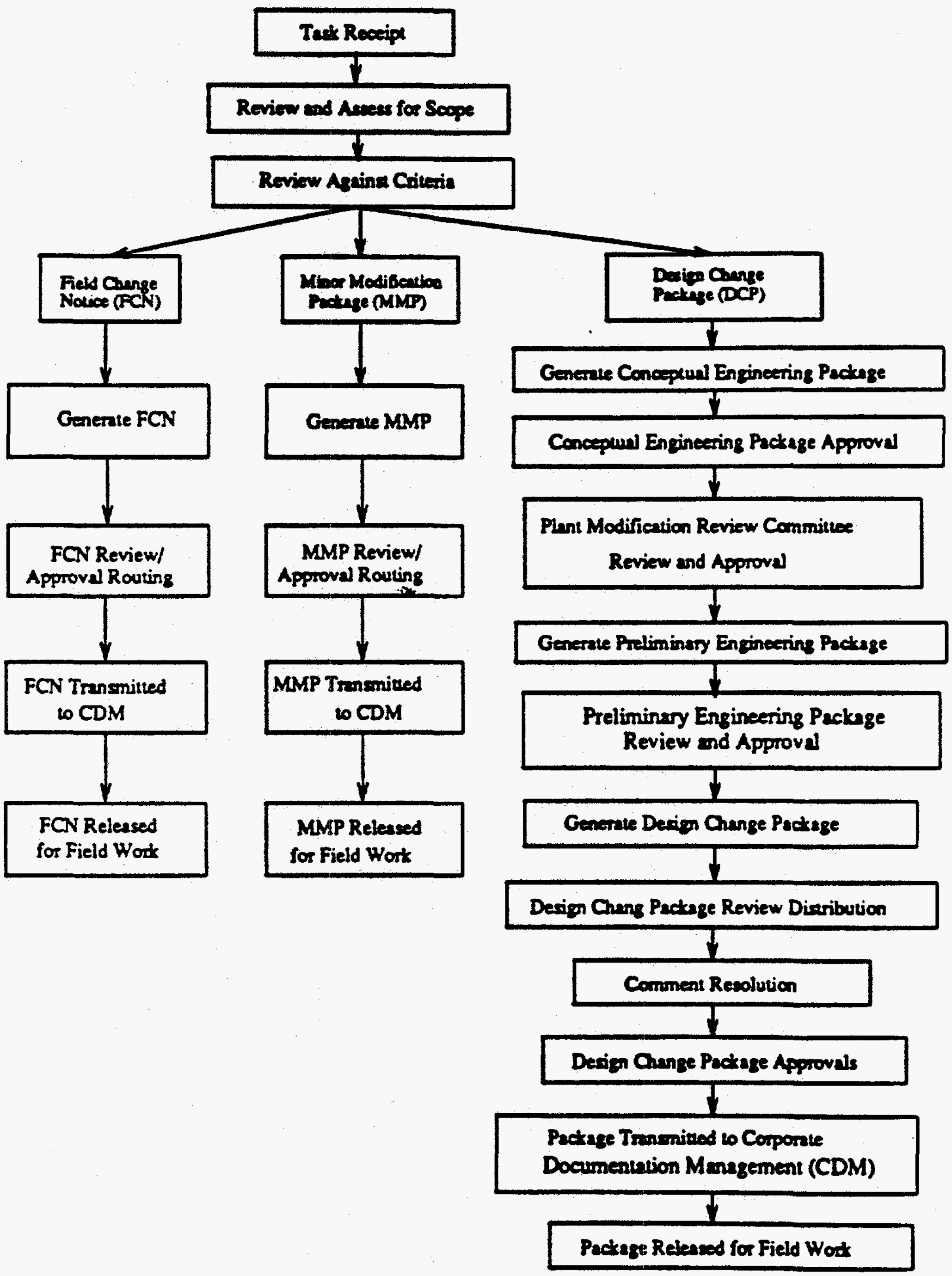

Figure 1. Design Change Work Process Flowchart

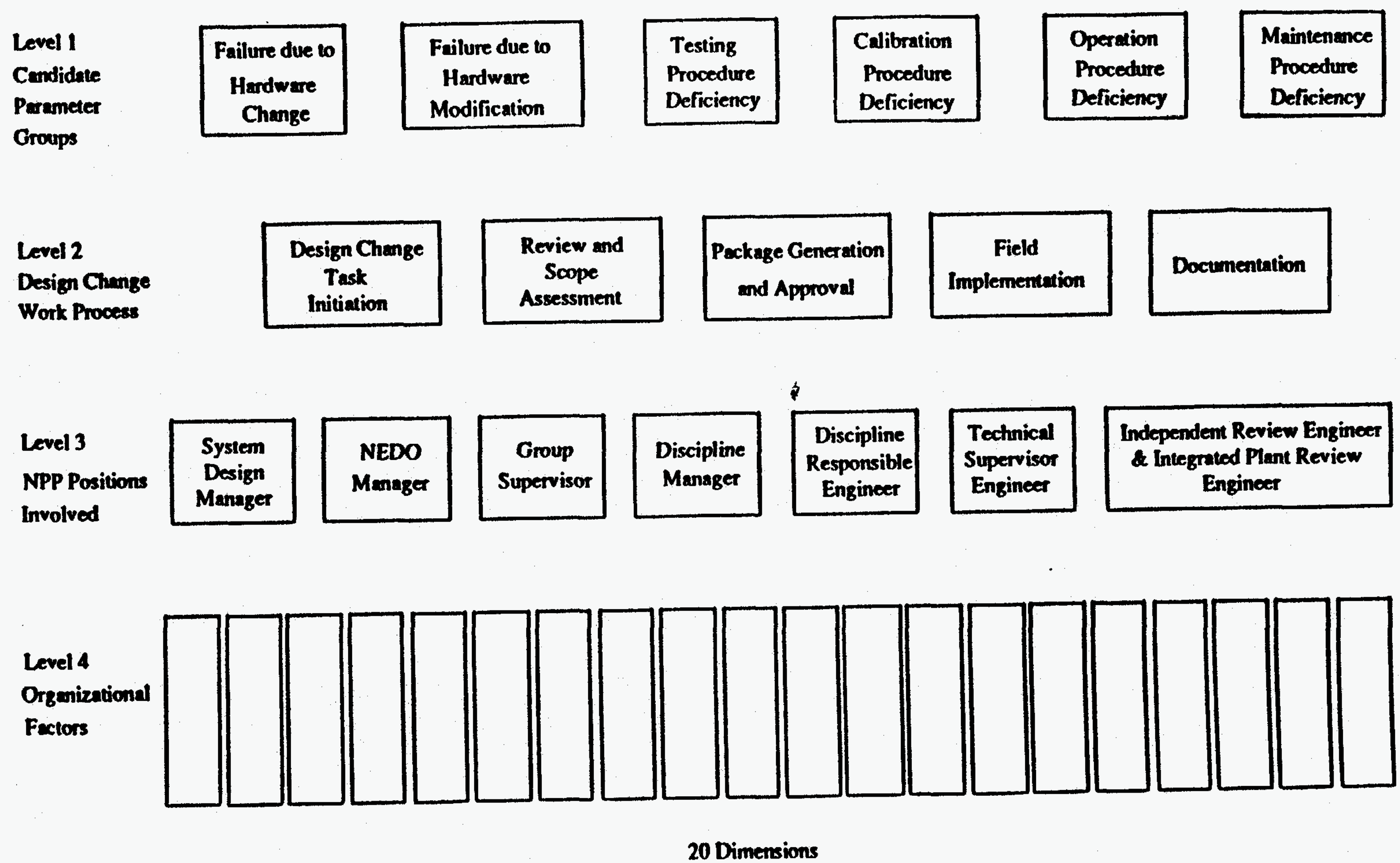

Figure 2. AHP Model for Design Change Work Process

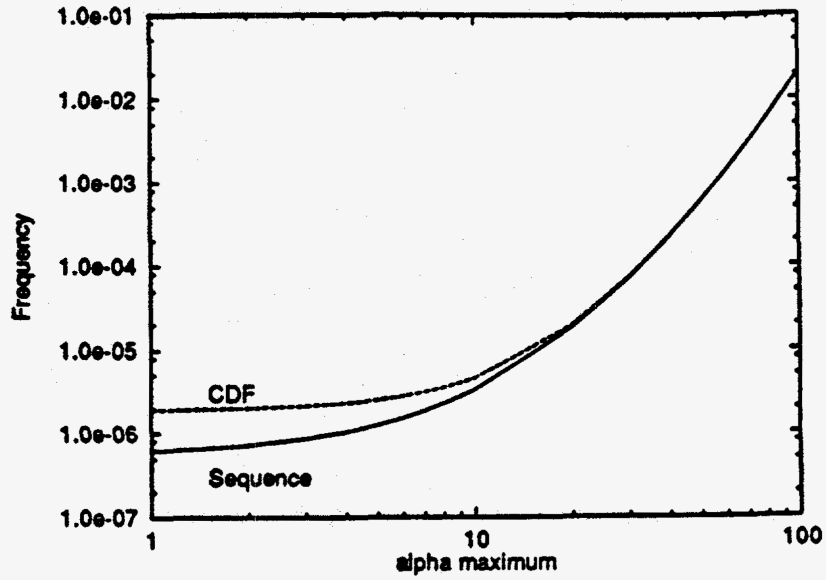

Figure 3: Sequence Frequency and CDF Versus α_{max} with Ratings =3

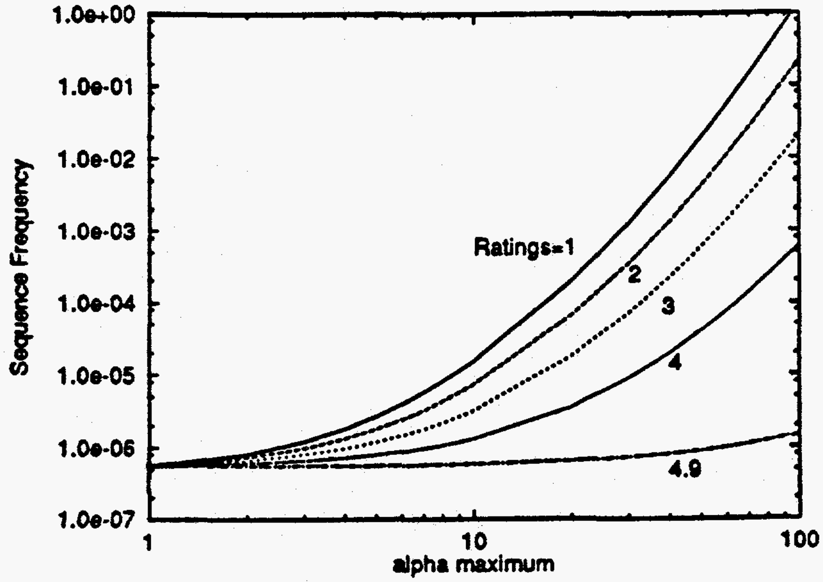

Figure 4: Sequence Frequency Versus α_{max} with Ratings as Parameter

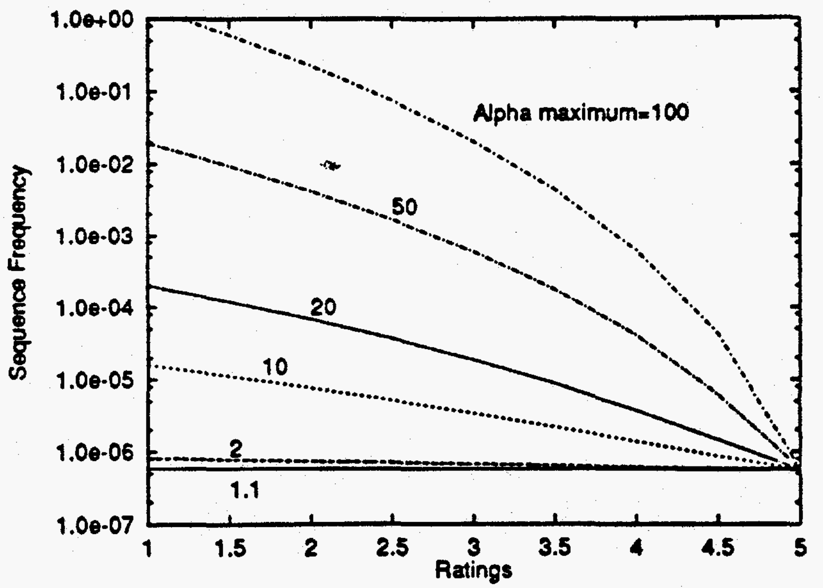

Figure 5: Sequence Frequency Versus Ratings with α_{max} as Parameter

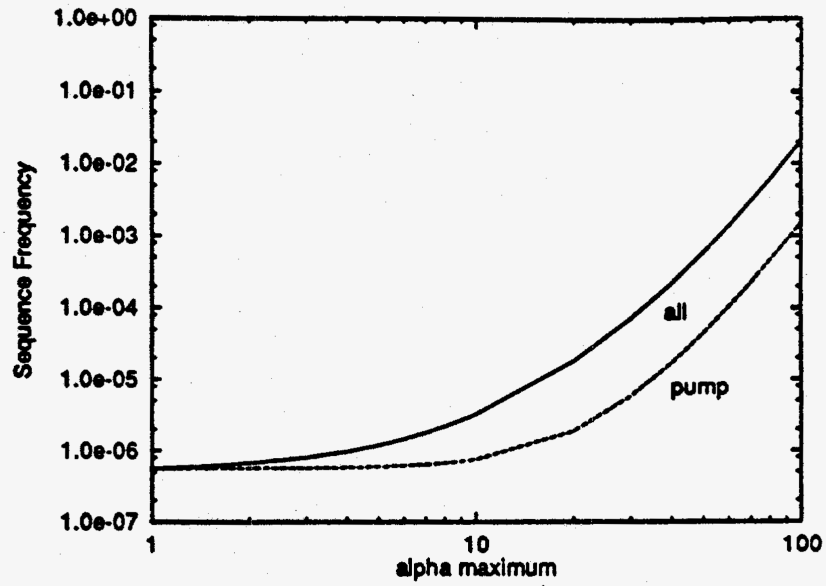

Figure 6: Sequence Frequency Versus α_{max} (Ratings = 3)

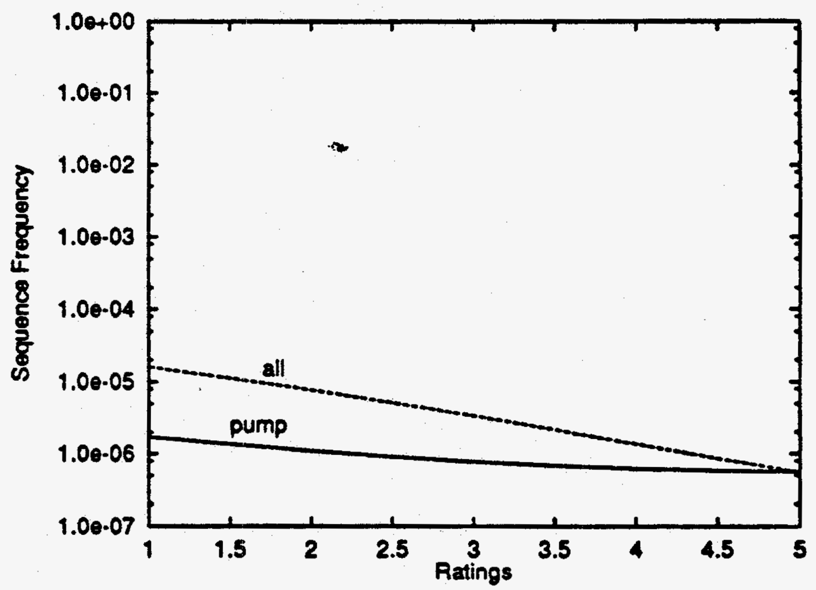

Figure 7: Sequence Frequency Versus Ratings ($\alpha_{max} = 10$)

APPENDIX D-1

Published Papers on the Influence of
Organizational and Management Quality on Risk

CONF papers removed and cycled separately

APPENDIX E

Considerations of Long Term Risk in the Disposal of Hazardous Wastes

Considerations of Long and Very Long Term Risk from the Disposal of Hazardous Wastes

In recent years it has become recognized that there is a need for a general philosophic policy to guide the regulation of waste disposal involving long term and very long term risks. In the past this has seemed to be a problem which belonged to the disposal of high-level radioactive wastes. However,there has been international recognition that large quantities of non-radioactive carcinogens are being disposed of,and that these materials will never decay, e.g.,arsenic, nickel,etc. Countries like the Netherlands, are examining this issue as a matter of national policy, and officials in the Nordic countries have also identified this as a matter requiring the development of a consistent policy.

One of the difficult aspects of regulating very long term risks arises from the issue of intergenerational transfer of risk or intergenerational equity. Some translate this into the question of whether future health effects can be discounted. However, this is an over-simplification of the problem. Furthermore, the balance between intergenerational equity and intragenerational equity, as well as issues such as the legacy left by an activity and the future state of society, must be considered, among others.

Most electricity generation sources, indeed most energy sources, including efficiency, have the possibility of introducing long-term risk in waste disposal. Fossil fuels contain heavy metal contaminants. Photovoltaic devices may contain arsenic. Shale development would pose still greater waste problems. In many of these cases the issue of very long term risk from waste disposal has not studied.

Similarly, chemical factories, petroleum refineries, and electroplating plants, among others, are sources of hazardous waste, either directly from the plant or from the end use of the product.

In this country, thus far, the Environmental Protection Agency (EPA) has treated the disposal of radioactive wastes far more stringently than non-radioactive wastes. The time horizon for which very low individual or societal risks must be predicted with high confidence is 10,000 years and possibly longer. Institutional controls must be assumed to be ineffective after 100 years, and all knowledge of the existence of the geologic repository is lost after that period. Society is assumed to be like it is today technologically, that is, there are no advancements in medicine. Furthermore, it must be assumed that individuals will not have the benefit of routine testing of water and food for radioactivity, and cleaning it up or substituting for it, if appropriate.

On the other hand, hazardous chemical waste disposal sites operating under the Resource Conservation and Recovery Act (RCRA) are generally regulated by provisions that are effective for 100 years or less. The burial of such wastes is relatively shallow and can lie above an aquifer.

Carcinogenic Metals, Should a Loss of Societal Memory Occur, J. of Hazardous Materials, 38 (1993) 363-384.

Okrent D., "Issues Related to the USEPA Probabilistic Standard for Geologic Disposal of High Level Radioactive Waste", Proc. Int. Conf. SAFE WASTE 93, Avignon, June 1993.

D. Okrent, "On Intergenerational Equity and Policies to Guide the Regulation of Disposal of Wastes Posing Very long Term Risks", UCLA Engineering Report UCLA-ENG-22-94, January 1994.

Sikora R.I. and B. Barry, Editors, "Obligations to Future Generations", Temple University Press, 1978.

Weiss E. B., "In Fairness to Future Generations". Transnational Publishers, 1989.

APPENDIX F

Probabilistic Treatment of Fuel Motion and Relocation
within the Reactor Vessel during a Severe Core Damage Accident

Probabilistic Treatment of the Uncertainties in Fuel Motion Within the Reactor Vessel During a Severe Core Damage Accident

The prediction of the motion within the reactor vessel of core fuel, control rods, and structural material during a severe core damage accident is extremely complicated. Even after the fact, analysts have found it difficult to match accurately the final fuel configuration found in the reactor vessel during the post mortem examination of the accident at Three Mile Island 2.

The accuracy of such predictions has taken on a growing importance in recent years. With the increased attention to the course of postulated severe accidents, and the efforts to develop accident management methods to ameliorate the consequences of such accidents, the potential for reactor vessel breach, or failure elsewhere in the primary system, becomes very important. Both the advanced light water reactors and some of those currently in operation are considering flooding of the containment to a high enough level to cool the lower part of the reactor vessel, in order to help retain the hot fuel.

However, there are considerable uncertainties in the modeling of a core meltdown accident. SCDAP/RELAP 5 has been the tool most frequently used, but it is known to be inadequate in its formulation, and many adjustable, empirical parameters are needed to match the bulk of the post mortem results at TMI 2.

In this research it is planned to examine SCDAP/RELAP 5 for its good and weak points, and to try to assign uncertainties to various facets of the analysis.

In addition, a lack of precise knowledge exists of the reactor conditions at the time of initiation of an actual accident, as well as of the interventions which may be made by man, and of the continued operability of various systems and components.

These all contribute to uncertainty in the actual fuel motion, and a treatment of where and when the fuel is likely to go, with what probability, could be of considerable value to an assessment of the likelihood of retaining the bulk of the fuel and radioactivity within the primary system.

It is the objective of this research to develop such a probabilistic methodology. The research is currently in the review stage. Mr. Xuegao An, a PhD candidate, has begun this effort.

References

1. U.S. Nuclear Regulatory Commission,"Reactor Safety Study, Wash-1400, October 1975.

2. U.S. Nuclear Regulatory commission, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, NUREG-1150, December 1990.

Release Management Strategies for a BWR Mark II Containment", NUREG/CR-5805, BNL-NUREG-52306, June 1992.

18. Hanson, D.J., et al, "A Systematic Process for Developing and Assessing Accident Plans", NUREG/CR-5543, EG&G-2595, 1990.

19. "Identification and Assessment of BWR (In-Vessel) Severe Accident Mitigation Strategies", NUREG/CR-5869, ORNL/TM-12080, 1992.

20. Asfia, F.J., and V.K. Dhir, "Experimental Investigation of Natural Convection Heat Transfer in Volumetrically Heated Spherical Segments", To appear in Journal of Heat Transfer.

Appendix C:

Use of Artificial Intelligence in Severe Accident Diagnosis for PWRs

APPENDIX G

Reports Sent to DOE Headquarters, Washington, DC
in Connection with DOE Award No. DE-FG03-92-ER75838

Progress Reports

1st Progress Report, May 1, 1993
2nd Progress Report, March 18, 1994
3rd Progress Report, March 8, 1995

Annual Reports

1st Annual Report, February 9, 1994
2nd Annual Report, January 3, 1995
3rd Annual Report, February 1, 1996