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# Human Factors Review for Severe Accident Sequence Analysis

Paul A. Krois
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This Work Performed for
U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Under
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# HUMAN FACTORS REVIEW FOR SEVERE ACCIDENT SEQUENCE ANALYSIS

Paul A. Krois
Paul M. Haas
John J. Manning
Randy Bovell*

NRC Monitor: J. P. Jenkins
Risk Analysis and Operations

Program Manager: P. M. Haas
Oak Ridge National Laboratory

Principal Investigator: P. A. Krois
Oak Ridge National Laboratory

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*General Physics Corporation
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Oak Ridge National Laboratory
Oak Ridge, Tennessee 37830
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## ABSTRACT

This report describes a human factors research project performed to: (1) support the Severe Accident Sequence Analysis (SASA) program and (2) develop a descriptive model of operator response in accident management. The first goal was accomplished by working with SASA analysts on the Browns Ferry Unit One anticipated transient without scram (ATWS) accident sequence to systematically assess critical operator actions and thereby demonstrate contributions to SASA analyses from human factors data and methods. The second goal was accomplished by developing a model called the Function Oriented Accident Management (FOAM) model, which provides both a conceptual structure linking off-normal safety functions with potential unconventional emergency responses and a method for developing technical guidance for those responses based on operations, engineering, and human factors data and expertise. The four components comprising the model are described and their use is shown through a table-top demonstration.

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## LIST OF ABBREVIATIONS

| | |
|---|---|
| ADS | automatic depressurization system |
| ALARA | as low as reasonably achievable |
| ASE | assistant shift engineer |
| ATWS | anticipated transient without scram |
| BF1 | Browns Ferry Unit One |
| BWR | boiling water reactor |
| CE | Combustion Engineering |
| CRDHS | control rod drive hydraulic system |
| ECCS | emergency core cooling system |
| EF | error factor |
| EOF | emergency operations facility |
| EOI | emergency operating instructions |
| EOP | emergency operating procedures |
| EPG | emergency procedure guidelines |
| ERF | emergency response facility |
| FEMA | Federal Emergency Management Agency |
| FOAM | function oriented accident management |
| HEP | human error probability |

| | |
|---|---|
| PSF | performance shaping factor |
| PSP | pressure suppression pool |
| PWR | pressurized water reactor |
| RCIC | reactor core isolation cooling |
| REP | radiological emergency plan |
| RHR | residual heat removal |
| RHRSW | residual heat removal service water |
| RMCS | reactor manual control system |
| RSCS | rod sequence control system |
| RWM | rod worth minimizer |
| SAINT | system analysis of integrated networks of tasks |
| SARP | severe accident research plan |
| SASA | severe accident sequence analysis |
| SE | shift engineer |
| SGT | standby gas treatment |
| SLC | standby liquid control |
| SLIM | subjective liklihood index method |
| SPDS | safety parameter display system |
| SRO | senior reactor operator |

| | |
|---|---|
| HPCI | high pressure coolant injection |
| HRA | human reliability analysis |
| IDCOR | Industry Degraded Core Program |
| INPO | Institute of Nuclear Power Operations |
| LLS | low-low set |
| LOCA | loss of coolant accident |
| LPCI | low pressure coolant injection |
| LPIS | low pressure injection systems |
| MAPPS | maintenance personnel performance simulation |
| MSIV | main steam isolation valve |
| NDL | nuclear data link |
| NPP | nuclear power plant |
| NRC | Nuclear Regulatory Commission |
| OAET | operator action event tree |
| OAT | operator action tree |
| OPPS | operator personnel performance simulation |
| ORNL | Oak Ridge National Laboratory |
| OSC | operations support center |
| PRA | probabilistic risk analysis |

| | |
|---|---|
| SROA | safety-related operator actions |
| SRV | safety relief valve |
| TDF | task data form |
| THERP | technique for human error rate prediction |
| TMI2 | Three Mile Island Unit Two |
| TSC | technical support center |
| TVA | Tennessee Valley Authority |
| UCB | uncertainty bounds |
| UER | unconventional emergency response |

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Steve Hodge and Mike Harrington of the Severe Accident Sequence Analysis (SASA) Program at Oak Ridge National Laboratory provided significant direction and technical information for the human factors analysis.

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The authors also wish to thank the Power Operations Training Center of the Tennessee Valley Authority for providing use of the Browns Ferry control room simulator, as well as providing instructors who participated in the simulator exercises of accident perturbations.

Gratitude is additionally extended to Ms. Lynn Allen for her patience, endurance, and word processing capabilities during the course of this project. Thanks also to Ms. Katie Ingersoll for her work on an earlier draft of this manuscript.

## EXECUTIVE SUMMARY

The human factors research described in this project report was undertaken to: (1) support the Severe Accident Sequence Analysis (SASA) program and (2) develop a descriptive model of operator response in accident management. The first goal was accomplished by working with SASA analysts on the Browns Ferry Unit One (BF1) anticipated transient without scram (ATWS) to systematically assess critical operator actions. These assessments demonstrate potential contributions to SASA analyses from human factors data and methods. The second goal was accomplished by developing a descriptive model called the Function Oriented Accident Management (FOAM) model, which serves both as a conceptual structure for identifying needs and deficiencies and as a method for developing technical guidance in accident management.

The human factors assessment of operator actions related to an ATWS was focused to some extent by concerns of SASA analysts. The SASA analysis considered operator actions in the context of the new symptom-based Emergency Procedure Guidelines (EPGs) developed by the BWR Owners Group. The EPGs were in the process of review by the Tennessee Valley Authority (TVA) for adaptation to BF1. Both the SASA and human factors analyses were limited to using best information available on the EPGs at the time the analyses were conducted. Based on the results of their analyses, SASA analysts made the recommendation that the emergency procedures for an ATWS be separated from the EPGs. The human factors analysis assisted in defining some of the problems operators may experience with the current structure of the EPGs. One of these problems is that certain operator actions called for in response to an ATWS are substantially different from actions appropriate to virtually all other accidents. Some of these actions are also contrary to basic operational practices on which operators are trained. One example related to an ATWS is the EPG instruction to lower and maintain the reactor vessel water level at the top of the fuel in order to reduce power. For all other accidents, low level would be an off-normal condition and the EPGs instruct operators to restore water level to within the range of the normal operating band.

SASA analysts assumed that the signature of an ATWS is so distinguishable that operators would quickly diagnose the event, and that a separate ATWS procedure would expedite operator response. From a human factors standpoint, the separation of those instructions in the EPGs relevant solely to an ATWS may or may not be entirely satisfactory. Operator performance during a transient would be affected by training, operator aids such as the safety parameter display system (not yet in use at BF1), procedures, and possibly many other variables. Such factors should be considered across a range of accidents to optimally guide operator response before deciding to again restructure the underlying approach to procedures in order to address problems related to one specific accident sequence. The question arises as to what other "special cases" or exemptions to the symptom-based EPGs might arise from further detailed analysis.

Rather than assess operator actions throughout the ATWS, the analysis was consolidated to only those operator actions prescribed by the EPGs that were judged to be most critical to the accident sequence. The identification and selection of critical operator actions were coordinated with SASA analysts in response to their concerns with certain actions required by the EPGs. Inputs to the selection process included: (1) examination of the EPGs, (2) consideration of operator actions modeled in computer codes used for accident sequence analysis, (3) review of operator actions observed during exercises conducted at the TVA Browns Ferry simulator on perturbations of an ATWS, and (4) review of an Operator Action Event Tree (OAET) developed for an ATWS and based on the EPGs. This OAET was modified based on input from SASA analysts.

Six operator actions were judged as being critical to the ATWS sequence. These actions included:

1. Manual selection and insertion of individual control rods given complete failure to scram.
2. Verification of conditions for use of the Standby Liquid Control (SLC) system and initiation of poison injection into the reactor vessel.
3. Initiation of pressure suppression pool (PSP) cooling through manual operation of the residual heat removal (RHR) system.
4. Operator control of the reactor vessel pressure by manually operating safety/relief valves (SRVs) before setpoints are reached for automatic SRV actuation.
5. Operator control of coolant injection systems in order to lower and maintain the reactor vessel water level at the top of active fuel.
6. Emergency depressurization of the reactor vessel in accordance with the PSP heat capacity temperature curve (specified in the EPGs) followed by control of low pressure injection systems.

Simulator exercises were conducted to provide data to the human factors and SASA analyses. These exercises were videotaped to provide a record of operator actions and the videotapes were subsequently used in the task analysis supporting the human reliability analysis (HRA). Exercises were held on two occasions using two BWR senior reactor operator instructors as operators.

The qualitative review was based on operators' comments and analysts' observations resulting from simulator exercises, the EPGs, and a task analysis following the format and approach of the NRC-sponsored task analysis data base. For each of the six critical operator actions, the qualitative review included: (1) an identification of problems affecting operator performance, (2) a description of actions required of the operator and constraints affecting performance, and (3) possible solutions to the problems. Potential problems include human engineering deficiencies in control room design and difficulties for operators related to unexpected system responses from changes in reactor pressure and/or water level. Suggested solutions include potential practical backfits to control room design and additional training to provide the expertise needed for the new operator actions required by the EPGs.

The purpose of the HRA was to provide quantified estimates of probable errors in operator response during an ATWS. It is noted that although primary emphasis was on operator actions prescribed by the EPGs, input to the HRA included a task analysis of certain operator actions following the event-based Emergency Operating Instructions (EOIs) currently in use at Browns Ferry. Examination of the EPGs and EOIs suggested that some operator actions required by these procedures would be performed in a closely similar manner. This similarity supports the assumption that results of the HRA, while based on the EOIs, are relevant to the EPGs. Four of the six critical operator actions were included in the HRA since there was agreement between the EPGs and EOIs on the steps comprising the actions. However, the actions of lowering and maintaining reactor vessel level at the top of the fuel and the steps necessary for emergency depressurization followed with control of the low pressure injection systems are unique to the EPGs, and this precluded their quantitative assessments in this study.

The quantitative HRA was divided into four components. First, methods for HRA reported in the literature were identified and briefly described. Second, a task analysis of the four selected critical operator actions was completed. Third, the steps in conducting the HRA using the Technique for Human Error Rate Prediction, or THERP, were completed, resulting in a listing of the derived quantitative human reliability estimates. The use of THERP was primarily relevant to estimating operator reliability for each of the individual critical tasks. Fourth, results of the analysis using the Operator Personnel Performance Simulation (OPPS) computer model were obtained. The uses of OPPS were to supplement the THERP analysis and compliment the SASA analysis by providing a time-reliability estimate across all operator actions throughout the ATWS.

For each of the four critical tasks, Task Data Forms (TDFs) were completed using the following resources: (1) BF1 emergency procedures, (2) videotapes of the simulator exercises on ATWS perturbations, (3) computer records of simulator data collected through the Performance Measurement System, including operator switch manipulations and continuous readings on selected critical plant safety parameters, and (4) expert judgement from qualified BWR plant operators and human factors personnel.

Results of the task analysis were used to guide selection of nominal HEPs from the THERP Handbook's human error data base. It is noted that the level of refined task information provided in the TDFs is typically more detailed than the level called for in the THERP Handbook. An HEP Worksheet was developed to organize and document the THERP analysis. Nominal HEPs were modified to reflect effects from performance shaping factors, such as stress, and the level of dependence among successive task elements. Modified HEPs comprising complete success paths were used to calculate final task success probabilities. Only actions for which errors would contribute to system failure were included in the calculations.

Supplementary assessment of operator actions throughout the ATWS was provided through use of the OPPS computer model. The OPPS model is programmed in the SAINT simulation language, and times the simulated control room crew branching through major phases of pre-alarm detection, event diagnosis, execution of procedure steps, and error recovery. Based on 1000 iterations of simulated ATWS events, average performance time for completion of all safety-related operator actions was 33.4 minutes. The minimum time was 23.0 minutes, and the maximum was 43.8 minutes. For comparison purposes, SASA analysts in their baseline worst case scenario of an ATWS involving no operator actions reported that containment failure would occur at 36.8 minutes into the accident. These analyses, then, suggest operators on the average should have sufficient time to complete all actions. However, not all actions would have to be completed within this time since the more critical actions would slow accident progression and extend the time remaining for the operators to complete the remaining actions.

In the area of accident management, The Function Oriented Accident Management (FOAM) model is a descriptive structure for integrating data and expertise from operations, engineering and human factors personnel. The purpose of the FOAM model is to structure general technical guidance for operators in responding to severe accidents that exceed the scope of current emergency procedures. Such technical guidance is necessary in order to extend the range of emergency procedures and training for accident management.

A review of background issues pertinent to accident management identified several human factors considerations important to development of the FOAM model. Regulatory requirements specify emergency response facilities and administrative procedures supporting accident management. The operator's cognitive behavior would be an important aspect in performance, and several qualitative approaches were examined. The scope of emergency procedures related to the safety function concept is examined, and the French accident management procedures are also reviewed. Industry and commercial training courses for mitigation of core damage are briefly described, along with a sampling of lessons for accident management learned from the Three Mile Island event. Recent guidelines for the design and evaluation of computer-based operator aids are examined as a potential resource in accident management, and use of expert systems in the control of nuclear plants is considered.

The FOAM model consists of four components. The first component is an assessment of the accident sequence with an identification of potential system failures and/or operator errors. The purpose of this assessment is to define the progression of the accident sequence and the potential end states resulting in core damage. An OAET is one method for identifying possible operator errors leading to a severe accident.

The second component involves a translation of the multiple system and operator control failures identified in the first component using a functional classification developed to identify plant safety functions and control requirements. One of the purposes of the translation is to identify potential alternate control requirements using redundant systems that could place the plant in a safe and stable condition. The functional classification is a technical guide for extending symptom-based procedures that links safety functions, control requirements, and redundant plant systems. A significant point is that when multiple failures cause an accident sequence to exceed the scope of the emergency procedures, the operators must develop one or more "unconventional emergency responses" (UERs) to either recover the failures or minimize/isolate their effects on plant safety. Assessments of UERs may use such resources as SASA analyses and recommendations, expert judgements from operations, engineering, and human factors personnel, and results of PRAs. Potential UERs would change according to the severity of core damage, i.e., operators may undertake certain actions if only limited fuel damage is evidenced, whereas other UERs would be more appropriate if massive core degradation has occurred.

The third component of the FOAM model concerns modeling the UERs for mitigating accident progression. For purposes of the FOAM model, UERs are modeled in an event tree format, which also permits an assessment of alternate end states. Each UER may be qualitatively assessed to systematically identify a range of information which, at the minimum, includes: (1) alarms and cues reflecting off-normal critical safety parameters associated with the system failures, (2) decision criteria such as identifying and weighing alternate actions, (3) an analysis of specific possible operator actions at some level of detail either to recover the failure of the reactor protection system or to isolate the effects of the failure, and (4) consequences of the UER to the plant in terms of contributions to accident mitigation or extending the timing of accident progression.

The fourth component of the descriptive FOAM model involves operator response to fuel damage and potential radiological release past plant protective barriers during the latter stages of an accident sequence. Challenges to multiple barriers would occur along liquid and gaseous streams. As part of the model, fission product pathways are identified through detailed fission product barrier diagrams tailored to the ATWS at BF1. Accompanying the barrier diagrams is a system description identifying: (1) how fission products would breach plant barriers and be subsequently released to the environment, (2) the information (alarms and recorders) available to the control room operators for assessing when a barrier has been violated, and (3) the possible actions the operator might take to mitigate a barrier breach and isolate the radiological release.

Use of the FOAM model for developing technical guidance applicable to operator training and for extending the scope of emergency procedures was shown in a table-top demonstration. A severe ATWS perturbation assessed by the SASA analysts was selected involving failure of manual rod insertion and failure of the SLC system. Five UERs were proposed to recover these failures and maintain an acceptable heat sink. Other potential UERs are discussed corresponding to more severely degraded conditions involving core slump and actions following breaching of the reactor vessel by the molten corium.

Potential applications of the FOAM model reflect regulatory, industry, and research perspectives. For each of these groups, the model provides guidance for structuring technical data and expertise and formulating potential requirements in order to improve responsiveness to degraded core conditions. The FOAM model provides a structure for developing guidance supporting such applications as extended procedures development, definition of training objectives and performance standards, specification of technical support from emergency response facilities, development of guidelines for the design and evaluation of computer aids, and assessment of control room instrumentation and layout.

Conclusions from this human factors project have fallen into two major categories. First, human factors support of the SASA program has provided some resolution of uncertainties in operator response to severe accidents. SASA analysts have emphasized the contributions from the classification and understanding of operator actions toward resolution of the possible myriad branches of the event sequence tree. Videotapes of the ATWS simulator exercises were notably useful to SASA and human factors analyses. Second, the descriptive FOAM model has suggested a structure for developing technical guidance for operator response in mitigating both core damage and radiological release.

The model provides a functional approach for standardizing procedures and training for accident management using operations, engineering, and human factors data and expertise.

Human factors support of other SASA studies is recommended to more thoroughly identify and assess operator actions affecting the accident sequence. Assessments of operator reliability, procedures, training, computer aids, and human engineering aspects of control room design are recommended. Analysis of these issues strengthens the SASA evaluation by reducing uncertainties in operator response.

Further work in accident management should attempt to provide technical support for operators to mitigate degraded core conditions. The FOAM model is one approach for standardizing and extending procedures and training. Additional work is recommended to more comprehensively apply results from SASA and PRA studies to support NRC, industry, and research needs in accident management.

# 1. INTRODUCTION

This report describes a human factors research project which had two purposes: (1) to support the Severe Accident Sequence Analysis (SASA) program with an assessment of operator actions, and (2) to develop a descriptive model of operator severe accident management. The first goal was accomplished by working with SASA analysts studying the Browns Ferry Unit One (BF1) anticipated transient without scram (ATWS) accident sequence, such that part of this human factors study demonstrates contributions to accident sequence analysis from assessments of operator performance. A second goal was accomplished by developing a new descriptive model called the Function Oriented Accident Management (FOAM) model, which serves as a function-based structure for applying human factors, engineering, and operations data and expertise in order to identify needs and deficiencies in the area of accident management.

This report describes both portions of the human factors work. Some preliminary results of the research supporting the SASA analysis were previously reported in an appendix to the SASA program ATWS report (Ref. 1). Complete results from the human factors analyses, including an assessment of recommendations by SASA analysts for operator actions during ATWS, are presented in Section 2 of this report, and the development and demonstration of the FOAM model are described in Section 3. Conclusions and recommendations associated with each portion of the human factors research are presented in Section 4.

## 2. ASSESSMENT OF OPERATOR ACTIONS DURING ATWS

The purpose of this section is to discuss the approach and results of both the qualitative and quantitative human factors assessments of operator actions during an ATWS accident sequence. Rather than assess operator actions throughout the ATWS, the analysis was consolidated to consider only those operator actions in the emergency procedures judged to be most critical to the accident sequence. A factor constraining this project was that the BF1 emergency procedures were being changed from event-based Emergency Operating Instructions (EOIs) to symptom-based Emergency Procedure Guidelines (EPGs) developed by the BWR Owners Group (Ref. 2); moreover, the EPGs were still under review by Tennessee Valley Authority (TVA) staff and SASA analysts who had particular concerns for certain controversial actions that were included in the qualitative human factors analysis. This project, then, was limited to using the best information available on the EPGs at the time the analyses were conducted. Assessment of human factors issues in the accident sequence analysis required extensive coordination with SASA analysts. For example, collection of task analysis information using the BF1 control room simulator was completed by an integrated team of human factors and SASA analysts.

Although primary emphasis was on operator actions contained in the EPGs, input to the human reliability analysis (HRA) included a task analysis of certain operator actions following the event-based EOIs. It was assumed that since these particular actions were required by both the EPGs and EOIs, they would be performed by operators in a closely similar manner. This similarity was held to support the assumption that results of the human reliability analysis, while based on the EOIs, were relevant to the EPGs.

This section begins with brief discussions providing background information on symptom- versus event-based procedures and on the human engineering analysis of control room design. This is followed by an identification of critical operator actions included in the EPGs. The qualitative analysis of critical operator actions is then presented, including the definition of problems, a description of required performance, and recommendations for problem resolution. This is followed with a description of the quantitative HRA, including a brief review of methods in HRA, the task analysis, and the approach and results corresponding to each of the two methods used.

### 2.1 Identification of Critical Operator Actions in Emergency Procedures

During familiarization with BF1 ATWS sequences juxtaposing automatic system responses with operator actions, two human factors issues identified as directly affecting operator performance were the emergency procedures and human engineering aspects in control room design. Background discussions of these human factors issues are presented first. The assessments of their effects on operator performance have been included in the qualitative analyses of critical operator actions. This is followed by the identification of critical operator actions included in the EPGs which are pertinent to an ATWS.

#### 2.1.1 Background on BF1 Emergency Procedures

As previously mentioned, the emergency procedures used at BF1 are undergoing a transition from event-based procedures to the symptom-based EPGs. Event-based procedures require operators to first diagnose the type of transient before taking corrective actions. With the symptom-based EPGs, diagnostic efforts are minimized such that operators selectively detect and attend to critical safety parameters that are off-normal. TVA is currently assessing the compatibility of the technical contents of the EPGs with BF1 system design and safety analysis.

The development of symptom-based procedures can be viewed as an attempt to reduce the cognitive workload of control room operators in diagnosing the type of transient. Through use of the EPGs during a transient, it is intended that operators verify and maintain the adequacy of critical safety functions. One advantage to an event-based procedure, however, is that operators may immediately relate causes and consequences of off-normal conditions and subsequently act directly to mitigate accident progression. Guidelines for the development of symptom-based procedures, also referred to as function-oriented emergency operating procedures (EOPs), are described in NUREG-0899 (Ref. 3).

SASA analysts have made the recommendation in Section 5.1 of the ATWS report (Ref. 1) that the emergency procedures for ATWS be separated from the EPGs. The human factors analysis assisted in defining some of the problems operators may experience with the current structure of the EPGs. One of these problems is that certain operator actions called for in response to an ATWS are substantially different from actions appropriate to other accidents. Some of these actions are also contrary to operational practices on which operators are trained. One example related to an ATWS is the instruction in the EPGs to lower and maintain vessel level at the top of the fuel in order to reduce power. For all other accidents, low vessel level would be an off-normal condition and the EPGs would instruct operators to restore vessel level to within more acceptable bounds. SASA analysts noted their assumption that the signature of an ATWS is so distinguishable that operators would quickly diagnose the event and that a separate ATWS emergency procedure would expedite operator response.

From a human factors standpoint, the structure of the EPGs presents some difficulties for operators in relation to an ATWS. However, the solution proposed by SASA analysts to separate those instructions relevant solely to an ATWS may or may not be entirely satisfactory. Operator performance during a transient would be based on several factors, including training and operator aids, such as the Safety Parameter Display System (SPDS), in addition to procedures. Currently, the SPDS at BF1 is in the design stage of development. These factors and others should be considered across a range of accidents to optimally guide operator response before targeting the restructuring of procedures to address problems related to one specific accident sequence.

An additional basis on which the EPGs may be systematically assessed follows human factors guidelines for preparation and evaluation of procedures. These guidelines are described in NUREG/CR-3177 (Ref. 4), NUREG/CR-1977 (Ref. 5), NUREG/CR-1970 (Ref. 6), and NUREG/CR-2005 (Ref. 7). These guidelines concern such issues as the general layout of procedures, and the complexity of procedure steps.

### 2.1.2 Background on Human Engineering Analysis of Control Room Design

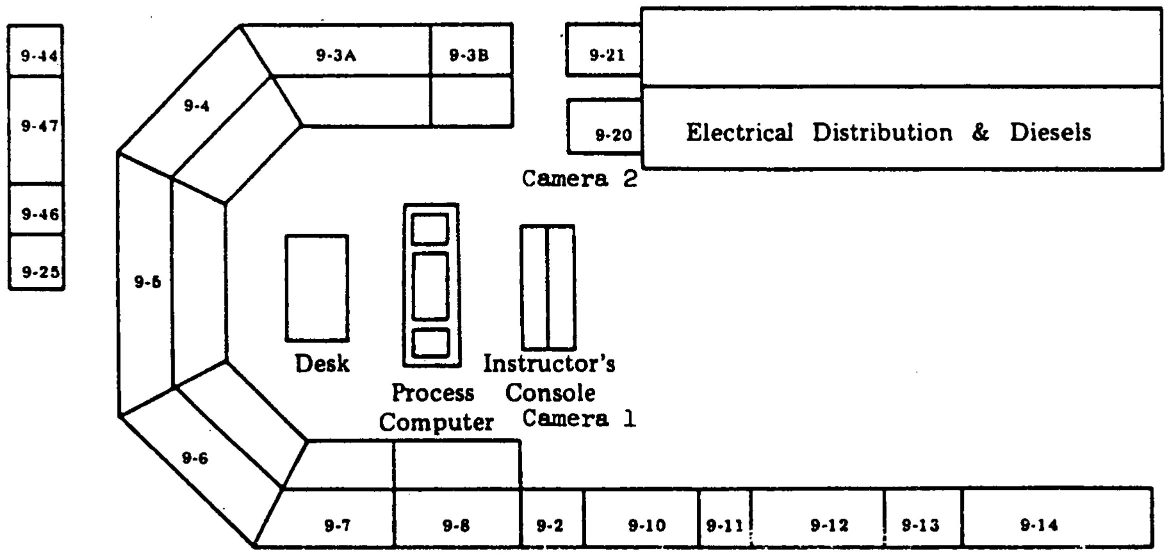
A human engineering analysis of the BF1 control room, or of any nuclear power plant (NPP) control work station, concerns the functional layout of controls and displays comprising the man-machine interface. The purpose of the analysis is to optimize operator performance. This study did not intend to undertake a comprehensive human engineering analysis of the BF1 control room. However, guidelines for a human engineering analysis are reported in NUREG-0700 (Ref. 8), NUREG-0801 (Ref. 9), and References 10, 11, and 12.

Several human engineering issues were identified during simulator exercises and an associated task analysis conducted as part of this project. The simulator exercises on ATWS sequences provided input to both the human factors analysis and the SASA analysis. Simulator exercises were videotaped to provide a record of operator actions under different ATWS perturbations. Exercises were held on two occasions using two BWR SRO-instructors as operators. On both occasions, an instructor was furnished by TVA, and the second instructor was from this human factors project. The BF1 simulator is a full-scope training simulator. The control room layout is shown in Figure 2.1. Two cameras were used to videotape the exercises, with one camera dedicated to taping the actions of each of the two instructors. This figure identifies the general panel locations of instrumentation for plant systems.

### 2.1.3 Selection of Critical Operator Actions

The identification and selection of critical operator actions was coordinated with SASA analysts in response to their need for information concerning certain actions required by the EPGs. Inputs to the selection process included: (1) examination of the EPGs, (2) consideration of operator actions included in computer code models used for accident sequence analysis, (3) critical review of operator actions observed during simulator exercises of an ATWS, and (4) review of an Operator Action Event Tree (OAET) developed for an ATWS and based on the EPGs (Ref. 13).

The OAET is a descriptive method identifying branches in the sequence of key operator actions necessary to mitigate an accident. The OAET for an ATWS initiated by closure of the main steam isolation valves (MSIVs) is shown in Figure 2.2. As part of the methodology, for each branching point the description includes the set of cues prompting operator actions, the set of actions identified in procedures, and constraints to success. Each end point is assessed in terms of the subsequent state of the plant. However, comparison of this OAET with results from SASA analyses suggests several potential modifications of this figure. The modified OAET is shown in Figure 2.3, and many of these actions are assessed in more detail later in this report. Following closure of the MSIVs, the operators should detect the failure of the reactor to scram. SASA analysts reported that within the first minute the recirculation pumps trip and the high pressure emergency core cooling systems initiate, i.e., the High Pressure Coolant Injection (HPCI) system and the Reactor Core Isolation Cooling (RCIC) system. Operators then attempt to initiate manual insertion of control rods to insert negative reactivity into the core, either through a manual scram or by selecting and inserting individual control rods. Next, when certain critical parameters exceed specified limits, operators manually initiate injection of a sodium pentaborate solution using the Standby Liquid Control (SLC) system for adding negative reactivity. Following initiation of the SLC system, the EPGs instruct operators to lower and maintain vessel level at the top of the active fuel. This temporarily inserts negative reactivity until sufficient poison, or all control rods, are inserted to bring the




Simulator Control Room Layout

| Panel | Description |
|---|---|
| 9-2 | Area and Process Radiation Monitor Recording |
| 9-3A | RHR, CS, Containment |
| 9-3B | HPCI, RCIC |
| 9-4 | Recirculation & Cleanup |
| 9-5 | Reactor Control |
| 9-6 | Feedwater, Steam & Condensate |
| 9-7 | Turbine Control |
| 9-8 | Main Generator & Auxiliary Power |
| 9-10 | Process Radiation Monitors |
| 9-11 | Area Radiation Monitors |
| 9-12 | SRM/IRM |
| 9-13 | TIP |
| 9-14 | Power Range Neutron Monitoring |
| 9-20 | Water Services |
| 9-21 | Temperature Recording |
| 9-25 | SBGT, RBCCW & Ventilation |
| 9-44 | Temperature and Radiation Monitors |
| 9-46 | Turbine Supervisory Instruments |
| 9-47 | Temperature Recording |

**Figure 2.1. BF1 simulator control room layout.**

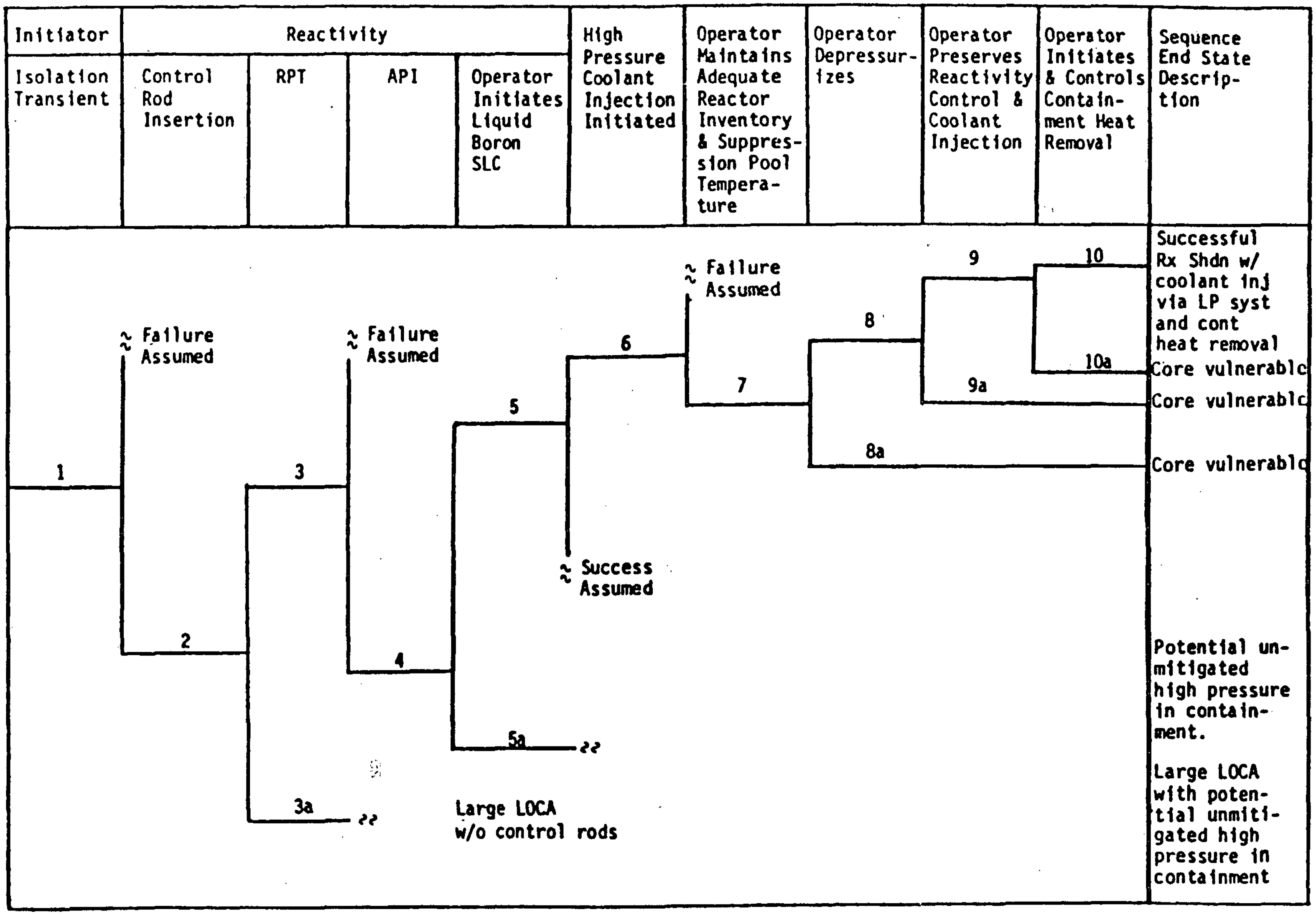



**Figure 2.2. Operator action event tree for ATWS initiated by closure of main steam isolation valves (from Ref. 13).**

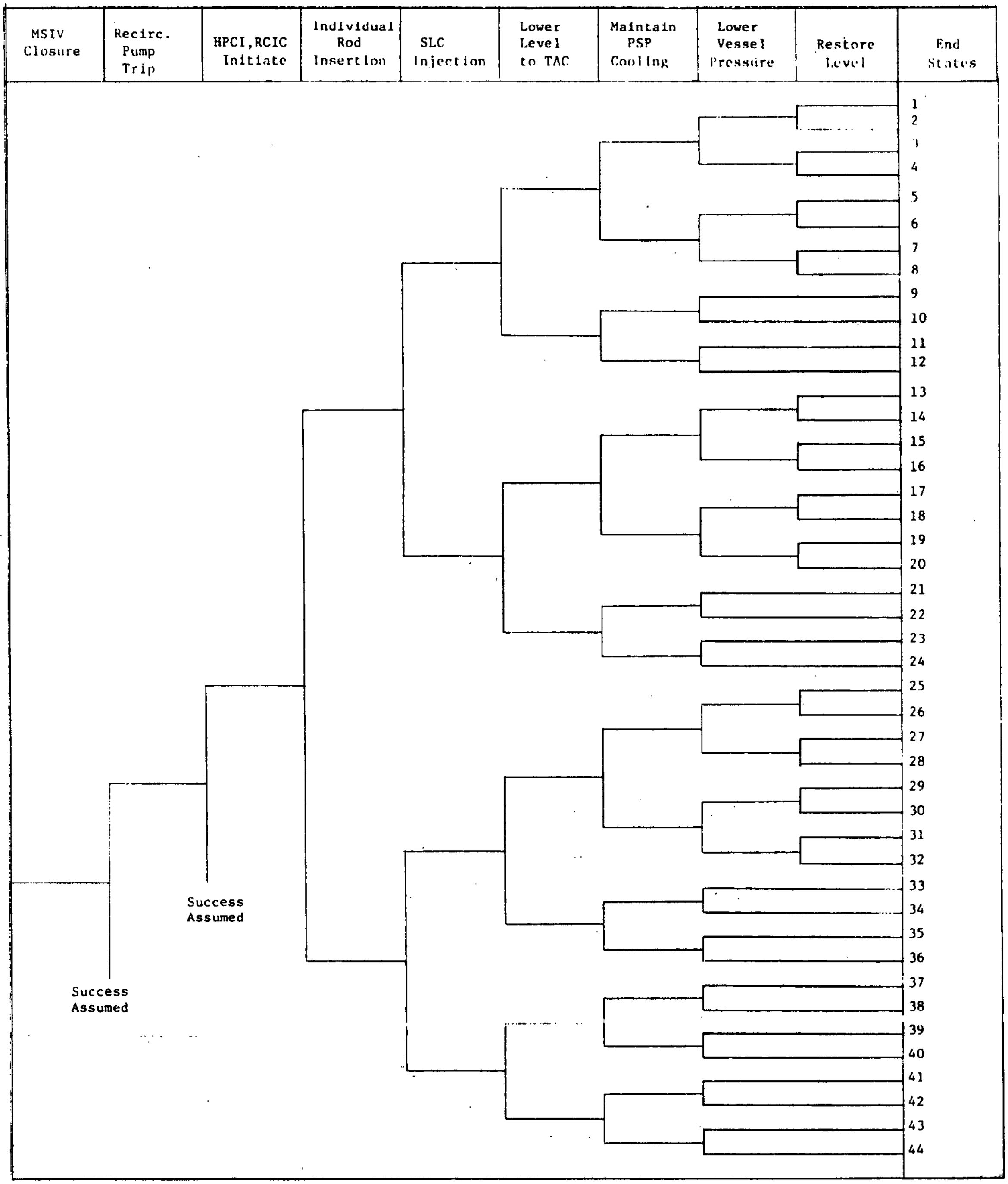



Figure 2.3. Modified OAET in accordance with SASA analysis.

reactor to hot shutdown. Operators also initiate cooling of the pressure suppression pool (PSP) which is the heat sink for reactor steam generated by decay heat following MSIV closure. SASA calculations for most ATWS perturbations show failure to maintain the PSP cooled below the limits of the EPG heat capacity temperature curve, which then requires emergency depressurization of the reactor vessel. Reactor vessel water level is then maintained near the top of the core, such as by low pressure injection, as control rods and poison continue to be inserted. Following insertion of the specified amount of poison, reactor vessel water level is restored to its normal band and the plant is at hot shutdown.

Six operator actions were judged as being critical to the ATWS sequence. These actions include:

1. Manual selection and insertion of individual control rods given complete failure to scram.
2. Verification of conditions for use of the SLC system and initiation of poison injection into the reactor vessel.
3. Initiation of PSP cooling by manual operation of the residual heat removal (RHR) system.
4. Operator control of reactor vessel pressure by manually operating safety relief valves (SRVs) before pressure setpoints are reached for automatic SRV actuation.
5. Operator control of coolant injection systems in order to lower and maintain reactor vessel water level at the top of the active fuel.
6. Emergency depressurization of the reactor vessel in accordance with the PSP heat capacity temperature curve followed by control of low pressure injection.

In the following discussions, each of these actions is assessed in a qualitative review. Subsequently, those actions for which sufficient specific documentation existed are also assessed in a quantitative HRA. Implications of SASA results and findings for these operator actions are assessed as possible training needs associated with use of the EPGs.

**2.2 Qualitative Review**

The task analysis methodology employed was that of the NRC task analysis data base (Ref. 14). The best available information was used to develop an Operating Sequence Overview based on the EPGs. The resulting description, shown in Figure 2.4, parallels the sequence of events reported in the SASA analysis of ATWS. For each of the six operator actions assessed below, the review includes: (1) a statement of problems and difficulties associated with the action, (2) a description of performance required of the operator and constraints to success, and (3) possible solutions or measures to remove the problems so as to improve performance reliability.

OPERATING SEQUENCE OVERVIEW

Plant: BFNP

Operator Function/Subfunction:
Supervise and Control Plant Operations/
Mitigate the Consequences of an Accident

NSSS/Type: GE/BWR 4

Operating Sequence ID: 7

C.R. Type: Multiple

Operating Sequence: Anticipated Transient Without Scram, Following MSIV Closure

Initial Conditions: Plant operating at 100% power and all systems in normal line-up.

Sequence Initiator: MSIV Closure

Progress of Action: The crew acknowledges the closure of the MSIVs, and recignizes that the reactor did not scram. All attempts to manually scram the reactor fail. Control rods are manually inserted using reactor manual control system. The reactor recirculation pumps trip automatically on high reactor pressure. Level rapidly decreases due to coolant loss through the safety/relief valves, and HPCI and RCIC automatically initiate on low level. The operators verify that conditions require initiation of standby liquid control and begin injection. Concurrently, coolant injection is manually throttled so that level is lowered and maintained at the top of active fuel to reduce power. Manual control rod insertion continues using RMCS.

The residual heat removal system is placed in the suppression pool cooling mode. Suppression pool temperature is monitored to maintain the torus heat capacity temperature limit. Reactor pressure is limited by automatic/ manual opening of safety/relief valves, and if SRVs are cycling or the RPV must be depressurized SRVs are manually opened until pressure drops.

Following injection of boron by SLC according to technical specifications, water level is raised using coolant injection systems to circulate poison through the core.

The Shift Supervisor declares an alert, and notifies appropriate on-site personnel.

Final Conditions: The plant is in hot shutdown with torus cooling in operation. Reactor level is being maintained using RCIC

Major Systems: Reactor Recirculation, Reactor Manual Control, Main Steam, Residual Heat Removal, RHR Service Water, Nuclear Instrumentation, HPCI, RCIC, SLC, Rod Worth Minimizer, Rod Sequence Control System, Primary Containment Isolation System, Water Level Instrumentation.

**Figure 2.4. Operating Sequence Overview with EPG-based operator actions.**

### 2.2.1 Manual Control Rod Insertion

#### 2.2.1.1 Statement of the Problem

Given the postulated failure of all 185 control rods to insert, the operator must attempt to judiciously select and insert high worth rods to maximize the rate of power reduction. This task is confounded by two human engineering problems related to control rod insertion. The switch to insert rods is a multifunction deadman lever with which errors of commission may occur. In addition, positioning errors may result while turning the rod sequence selector switch until the desired rod select pushbutton is illuminated.

#### 2.2.1.2 Performance Description

Operator actions to insert control rods are critical to shutting the reactor down in the event of failure of automatic systems to scram the reactor. Once the operators have diagnosed the failure of rods to insert, the EPGs instruct them to manually insert the control rods which can only be done one at a time. The procedure requires switching to the manual insertion mode and taking action to bypass rod sequencing and other interlocks. The process takes about one minute per control rod from the fully withdrawn position. A considerable amount of time would be required, then, to manually insert all withdrawn control rods. However, through judicious selection of high worth rods and inserting these first, the operator should reduce power more quickly. The operator reads from the rod pattern charts to select and insert high worth control rods. It is the responsibility of the nuclear engineer to designate the high worth rods.

During the simulator accident sequence exercises, the reactor operators alternately assumed responsibility for manually inserting the control rods. They reported an apparently accelerated learning curve in selecting higher worth rods and in maintaining a continuous rate of rod insertion over practice runs. However, since insertion of high worth control rods can cause fuel failure if caution is not exercised, the operators also reported some concern about introducing flux tilts in certain areas of the core when a reasonable rod pattern was not maintained.

One human engineering problem involves the switch with which operators drive in the selected control rod. This switch is a multifunction deadman lever which constrains operator mobility and may contribute to errors of commission. This lever functions to both insert and withdraw control rods. The lever is spring-loaded, so the operator must continually activate and overpressure the spring to move a rod. The operator's access to other instrumentation is limited to the reach of his arms in either direction of the switch. The instructors on the simulator were observed making commission errors in selecting the incorrect mode of the control switch. In every case, however, they recovered and placed the switch in the correct mode within approximately one second.

A second human engineering problem concerns potential errors in positioning the rod sequence selector switch to enable the desired rod select pushbutton. During the ATWS it is desirable to insert high worth control rods near the center of the core to achieve the quickest reduction in reactor power. In order to insert the high worth control rods, and to continuously insert one control rod after another, the operator must deviate from the pre-programmed rod sequence. To do this, interlocks provided by two systems to reduce the consequences of a rod drop accident must be overridden. The first system, the Rod Worth Minimizer (RWM), can be easily bypassed by activation of a keylock switch in the control room. However, the second system, the Rod Sequence Control System (RSCS), cannot be bypassed in the control room. The control room operator must communicate with an auxiliary operator in the instrument room to bypass rod groups as necessary, delaying control rod insertion. The operator must also manipulate two control room switches, the rod sequence select and mode select switches, for the RSCS to insert control rods because the Reactor Manual Control System (RMCS) imposes RSCS rod blocks when the emergency insert is used.

The RSCS switches must be positioned to permit selection and movement of the desired control rod. A problem is the need to position the rod sequence selector switch when changing from one rod group to another, which increases the time delay for rod selection and insertion. The operator manipulates the rod sequence selector switch until the desired rod select pushbutton is illuminated. The rod select pushbuttons are small and lighted from the back. This switch positioning problem is further complicated by the distant location of the switch, which makes it difficult for the operator to read the rod select pushbuttons while manipulating the switch. This may lead to a number of errors in positioning the rod sequence selector switch until the desired rod pushbutton is selected.

#### 2.2.1.3 Problem Resolution

Operator skill and knowledge for selecting and inserting high worth control rods should be developed through simulator exercises. Specific decisions and actions comprising this task should be covered in training to ensure performance proficiency.

The human engineering deficiencies associated with the multifunction deadman switch for control rod insertion could be resolved by a possible backfit. A potential solution is that, when in the emergency manual insertion mode, the switch would have a momentary block. This block would permit the operator to remove his hand from the switch and have a short period of time for other tasks.

The human engineering problem related to positioning the rod sequence selector switch requires additional engineering analysis in order to develop an acceptable solution. Whereas the rod select pushbuttons could be backlit with more powerful bulbs to facilitate discrimination, other factors involving rod blocks and system interlocks should be considered.

### 2.2.2 Checking Conditions and Initiating SLC Injection

#### 2.2.2.1 Statement of the Problem

The execution and timing of the tasks of checking conditions and initiating SLC injection are subject to some uncertainty. Initiating injection of the sodium pentaborate solution, while based on procedures, seems to be a generally controversial action. The decision may be of such magnitude that the conrol room crew will attempt all other remedial actions prior to SLC injection. However, SASA analysts have determined that, during an ATWS, power must be reduced in order to control the heat and pressure loads to the drywell.

### 2.2.2.2 Performance Description

Verifying conditions and initiating SLC injection are critical tasks insofar as poison injection satisfies the functional requirement of inserting negative reactivity to shut the reactor down. Poison injection in a BWR is also controversial with regards to lost plant availability during lengthy cleanup. One TVA manager informally estimated the removal of boron plated out on vessel internals to be about $100 million.

The new BWR Owners Group emergency procedures relieve the operator of some of the burden in this decision-making process. Initiation of the SLC system is mandatory under either of the following conditions:

1. Five or more adjacent control rods not inserted below 06 position and either the reactor water level cannot be maintained or the suppression pool water temperature limit of 110°F is reached. (The 06 position is equivalent to 18 in. (0.46 m) of rod withdrawal. Total rod travel is 144 in. (3.66 m).)
2. Thirty or more rods not inserted below 06 position and either the reactor water level cannot be maintained or the suppression pool water temperature limit of 110°F is reached.

Initiation of the SLC system is the responsibility of the Shift Engineer (SE) or Assistant Shift Engineer (ASE). The procedure, however, permits the unit operator to take this action if both the SE and ASE are not available.

Even with this procedural requirement, the operators should, if time is available, try other alternatives for manually inserting control rods before initiating SLC injection. According to emergency procedures, operators will attempt to clear any possible hydraulic lock in the control rod drives. They will also remove the scram fuses in case of a possible electrical fault. Should any of these actions result in a successful scram of the withdrawn control rods, use of the SLC system would not be required.

### 2.2.2.3 Problem Resolution

The technical specifications contain the basis for all emergency responses. This would seem to minimize any uncertainty in discretion on the operator's part. The use of SLC depends upon a rapid evaluation of the unit's state, the determination that an ATWS has occurred, and immediate action to decrease reactor power by first attempting insertion of negative reactivity other than by SLC injection. If these actions fail or the technical specifications/procedures limits are reached prior to full shutdown, then SLC injection is unavoidable. In view of the critical nature of the results from a decision to inject boron, operating crews should be trained on timely use of the SLC system. This training should bring out the potential for further problems with the PSP and drywell once limiting conditions are met and SLC injection has not been promptly initiated. Management should also reinforce the need to take prompt action whenever the procedural limits for SLC injection are reached or violated.

### 2.2.3 Initiate PSP Cooling

#### 2.2.3.1 Statement of the Problem

During an ATWS initiated by MSIV closure, the PSP becomes the primary heat sink. Preservation of this resource is time dependent to the extent that without operator action the PSP/drywell boundary may be imminently threatened. One of the goals of the operating crew is to maintain the cooling/quenching characteristics of the PSP. The response to these requirements is to initiate PSP cooling using the reactor heat removal (RHR) system. The uncertainty in operator actions is concerned with, in part, the time required to set up the RHR system in the PSP cooling mode.

#### 2.2.3.2 Performance Description

Initiation of PSP cooling is important for protecting the fuel and the primary containment integrity in the loss of availability of the main condenser following MSIV closure. Two RHR loops are available for PSP cooling involving a total of four pumps. A major contributor to the time required for task execution is whether the operator recognizes the increase of PSP temperature. Recognition of this increase should be facilitated by the PSP high temperature annunciator. The operator may be distracted from acknowledging this annunciator when he must concurrently perform other important tasks. For example, control of reactor pressure and vessel water level may compete with and delay initiation of PSP cooling.

A human engineering difficulty involves operation of the suppression pool test line valve which is used for return of the cooled water to the pool. For each of the the two RHR loops, valve motion stops when the deadman control lever for this valve is released. Cycling of the valve requires about two minutes. If during this time the operator is drawn away to perform other essential tasks, he must return to the control switch to continue and complete valve motion. The valve discontinues movement when the lever is released, which apparently is intended to prevent pump runout during testing.

#### 2.2.3.3 Problem Resolution

The event-based EOIs for an ATWS did not include a step for PSP cooling, whereas the proposed EPGs do include a specific step for initiating PSP cooling when limits are exceeded during a steam blowdown to the PSP. In using the EPGs, then, operator reliability in executing this task should be higher. The potential delay associated with cycling of certain RHR valves when respective deadman control switches are released is a problem, but is acceptable for the following reason. During testing, and especially a test on one pump, pump runout is prevented by limiting cycling of these valves in this manner.
The operator must also be aware of the potential isolation of the suppression pool cooling mode when a Low Pressure Coolant Injection (LPCI) initiation signal is received. The PSP cooling path isolates in order to ensure that cooling water is not diverted from reactor vessel injection during a LOCA. When the operator lowers the reactor water level to the top of the core, the LPCI initiation setpoint is reached at 21.5 inches above the top of the active fuel. Isolation of the PSP cooling path can be prevented at this point if the operator plans for its occurrence.

The PSP cooling valves are part of the containment spray mode of the RHR system. The valve logic that closes the torus valve will permit the valve to open or remain open with the LPCI initiation signal present if the valve select switch is in the SELECT position. The logic will also close the torus valve if the core coverage is maintained equal to or greater than two-thirds of the core height. This interlock may be defeated by use of a keylock bypass switch.

#### 2.2.4 SRV Actuation Preventing Vessel Overpressure

##### 2.2.4.1 Statement of the Problem

In attempting pressure control during an ATWS, the operator may be hindered, among other concerns, by not knowing if the SRV he is trying to open has already been activated automatically. This gap exists because no positive SRV position indication is located adjacent to the manual SRV controls.

##### 2.2.4.2 Performance Description

The BF1 unit has thirteen safety relief valves distributed among four main steam lines exiting the pressure vessel. These valves have two functions: to protect against overpressure transients and to depressurize the reactor when required during off-normal conditions. Any of the valves can be opened manually with switch action by the operators and will be automatically opened by steam pressure once their set points are exceeded. The valve set points range from 1105 to 1125 psig.

Six of the SRVs are dedicated to the automatic depressurization system (ADS). This system initiates on high drywell pressure and low reactor vessel water level. The ADS autotimer has a two-minute cycle. If the low level signal does not clear, or if the operator does not recycle the timer prior to the end of the two minutes, all six valves open. Once the ADS activates the six SRVs, the SRVs will not close until the reactor pressure drops to about 20 psi above drywell pressure or the operator manually resets the ADS timer.

The design problem is an absence of any positive individual indication of SRV activation adjacent to the SRV controls. Experienced operators may hypothesize that SRVs are automatically cycling based on pressure, flow, and other monitors. There are acoustic monitors for the SRVs, but these are displayed at the rear of one of the back panels. The only front panel indication for the operators is the switch handle mode and a light adjacent to each switch. Illumination of the light tells the operator only that the associated solenoid valve has been energized, not that the valve has actually opened. When an SRV is opened on high reactor vessel pressure, the solenoid is not energized. Therefore, the light does not inform the operator that the valve has opened. In sum, the operator is not provided timely information about valve position unless he takes several seconds to walk to the back panel to observe the acoustic monitors.

The potential error from this design problem is that the operator may attempt to open a valve which is already in the blowdown mode from overpressure. As analyzed in the SASA ATWS report (Ref. 1), the pressure level will not change if the operator opens an SRV that is already open. If the SRV is indeed shut and the operator opens it, the pressure level will decrease slightly and one of the already automatically opened SRVs will close. SASA analysts' calculations show that the operator's actions would not significantly reduce the pressure level until five SRVs were manually opened. This calculation assumed the reactor was generating 29% of full steam flow in an ATWS with MSIV closure, and that each SRV has a capacity equivalent to about 6.5% of full reactor power. Given these conditions, then, four SRVs would quickly be automatically opened because of high reactor vessel pressure and not until the fifth SRV was manually opened would the pressure level decrease. The rate of pressure decrease would accelerate very rapidly because decreasing pressure serves to increase the voids in the core region. Increases in the void fraction insert negative reactivity and lower core power. In turn, lower power reduces the reactor steam generation to below the capacity of the five SRVs being manually held open, so the pressure level further decreases. SASA analysts warn that if the operator does not close some SRVs in a timely manner, the reactor vessel will depressurize to below the initiation setpoints of the low pressure injection systems (LPIS). The LPIS will then flood the core, causing severe power and pressure spikes. This particular difficulty is discussed in more detail in Section 2.2.6.

It is noted that operators are instructed to manually open SRVs when vessel pressure is at or above SRV setpoints. This is a general requirement to minimize wear and tear due to repeated automatic cycling and is not a procedure specific to ATWS mitigation. An additional problem is potentially associated with the lack of positive SRV position indication and involves a stuck-open SRV. That is, the operator may attempt to close a valve which has actually stuck open but for which he has no immediate feedback on the failure. The operator would then need to examine the acoustic monitors, along with other relevant instrumentation, to diagnose this failure.

#### 2.2.4.3 Problem Resolution

Operators are blind to actual SRV positions unless they take time to check acoustic monitors on a back panel. A status lamp, perhaps utilizing information from the acoustic monitors to generate the signals, would be sufficient to supply the necessary data to guide manual SRV actuation.

### 2.2.5 Reactor Vessel Water Level Control

#### 2.2.5.1 Statement of the Problem

For an ATWS, the EPGs instruct the operator to lower and maintain the reactor vessel water level at the top of active fuel while sodium pentaborate solution is being injected. This instruction to lower the level conflicts with intuition and training for virtually every other accident situation. Furthermore, in order to execute this instruction, operators must rely on level indicators which may be inaccurate, have insufficient range, or are located on distant panels. During execution of this instruction, operators may experience difficulty with use of high pressure injection systems.

### 2.2.5.2 Performance Description

The purpose of lowering and maintaining reactor vessel water level at the top of active fuel is, as evidenced by the EPGs, to temporarily reduce reactor power while sodium pentaborate solution is injected into the vessel. Lowered vessel level slows or stops all natural circulation through the reactor vessel downcomer region and the jet pumps. SASA analysts report that core thermal power would be about 9% with water level lowered to the top of the core and the reactor vessel fully pressurized (Ref. 1). Because of the reduced natural circulation, the sodium pentaborate injected by the SLC system would not be sufficiently swept into the core. The EPGs instruct the operator to restore vessel level to the normal operating level after the amount of sodium pentaborate required for hot shutdown has been injected. The increased coolant injection mixes the liquid poison and restores natural circulation at decay heat levels.

The instruction in the EPGs to lower the water level to the top of the core is contrary to the fundamental principle, repeatedly reinforced by training, to maintain a normal operating level under accident conditions. The event-based EOIs do not contain any instructions on monitoring the reactor vessel level and controlling coolant injection. The EPGs identify injection systems which operators may manipulate in executing this instruction.

In order to lower and maintain the vessel level, operators use level instrumentation consisting of four monitoring systems with ten total indicators in the control room. The top of the core is 360 inches above the bottom of the reactor vessel. First, narrow range GEMACs cover the range from 528 to 588 inches (scale reads from 0 to +60 inches). There are three of these sensor systems in the control room, and one of two sensor outputs is fed to a permanent recorder. These narrow range sensors are used for normal operation. The displays are located on panels close to the controls for the coolant injection systems.

Second, wide range YARWAYS comprise the emergency systems indication and cover the range from 373 to 588 inches (scale reads from −155 to +60 inches), as shown in Figure 2.5. The YARWAYS are used in off-normal conditions. There are two of these systems, and they are not fed to a recorder. The displays are located on panel 9-5 so as to be near the feedwater controls on panel 9-6, as shown in Figure 2.1.

Third, post-accident flooding indication sensors cover the range from 260 to 560 inches (scale reads from −100 to +200 inches), as shown in Figure 2.5. There are two of these systems and these sensors are used mainly in conjunction with the emergency core cooling systems (ECCS). There is a recorder indication in the range of 360 to 460 inches (scale reads from 0 to +100 inches). The displays are located with the ECCS controls on panels 9-3A and 9-3B, as shown in Figure 2.1.

Fourth, shutdown flooding range indication has one sensor which covers the range from 528 to 928 inches (scale reads from 0 to +400 inches). This instrument monitors the level when flooding of the vessel is required.

One of the design problems that would interfere with an effort to maintain the reactor vessel water level near the top of the active fuel is the lack of reliable information on the

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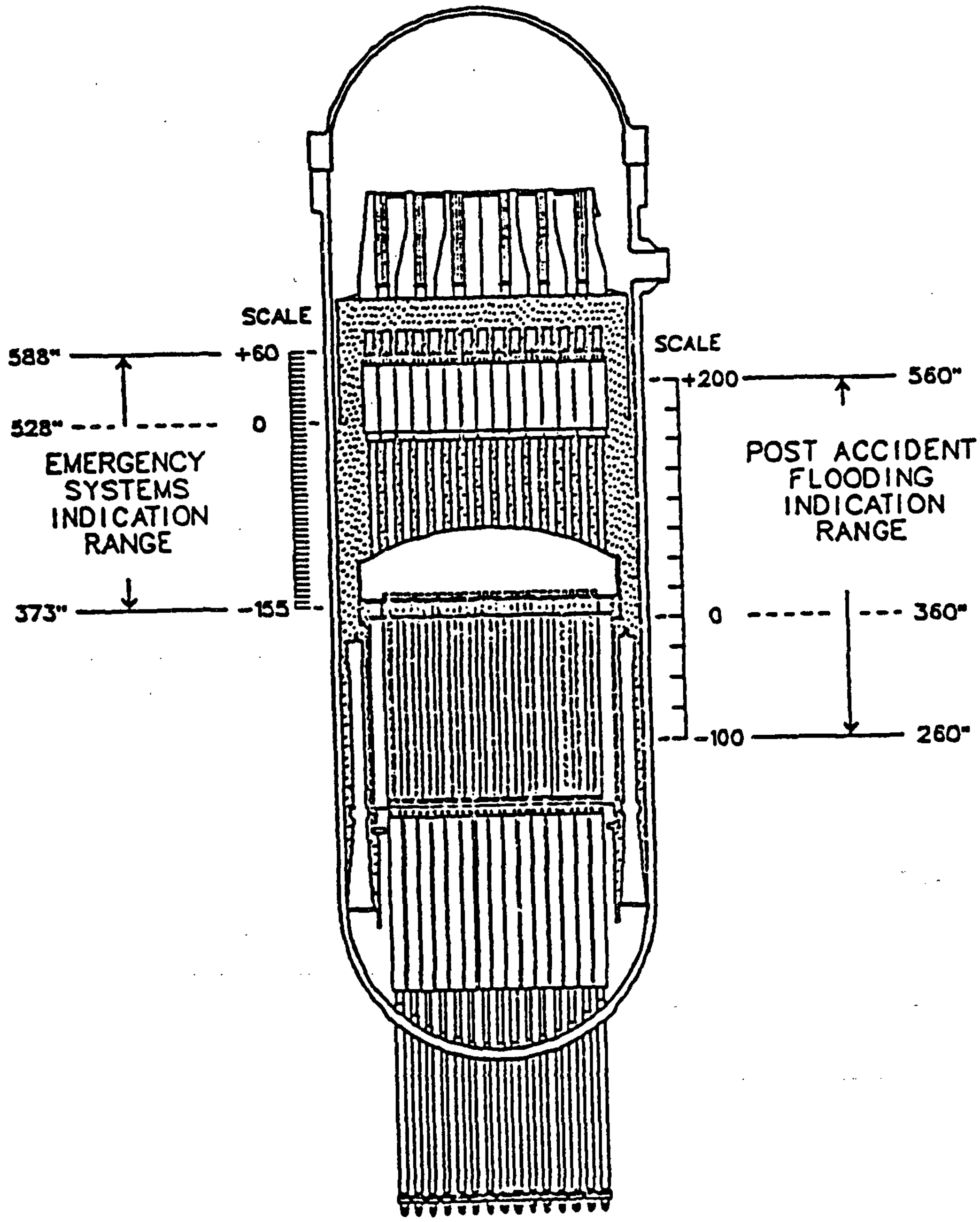


**Figure 2.5. Level instrumentation available for monitoring reactor vessel downcomer water levels near the top of the core.** (Non-scale dimensions are height in inches above the inner bottom of the reactor vessel.)

level. During the ATWS, the operators would most frequently use the emergency systems indication and the post-accident flooding indication. When the operator lowers and maintains the vessel level at the top of the core using ECCS, he would likely monitor the level by reading the nearby post-accident flooding indication which has sufficient range. However, this indication is "cold" calibrated for use in LOCA conditions, i.e., it is calibrated for the reactor at atmospheric pressure. Level information would be unreliable and would impede the operator in maintaining the vessel level at the top of the fuel. This predicates some type of variable normalization or correction factor which the operator must apply for proper level indication. Differences between the actual level compared to emergency systems indication and post-accident flooding indication are shown in Table 2.1, as reported in the SASA ATWS report (Ref. 1).

A second design problem is the adequacy of range monitored and displayed by vessel level indications. The post-accident flooding indication has a bottom end 100 inches below the top of the fuel. As shown in Table 2.1, at high vessel pressure, the indicated level will be about 43 inches below the actual level. Thus, if the operator maintains the reactor vessel water level in accordance with this indication, the actual water level would be some three and one-half feet above the top of the active fuel. The emergency systems indication, which is calibrated at operating temperatures and pressure, has a bottom end 13 inches above the top of the fuel. At low vessel pressure, the indicated level has some discrepancy compared to the actual level. If the vessel remained pressurized, the operator would monitor the emergency systems indication as long as the vessel level was at least 13 inches above the top of the fuel.

A third design problem concerns the locations of the emergency systems indication and the post-accident flooding indication. Operators are trained to use these wide range monitors in off-normal conditions. While the post-accident flooding indication displays are near the ECCS controls, the emergency system indication displays are some distance away. During the simulator exercises, the instructor acting as the lead operator (Operator #1) was attempting to control the reactivity of the core by manually inserting control rods and injecting boron through the SLC system. The second instructor (Operator #2) was controlling ECCS to lower the vessel level to the top of the fuel. Operator #2 monitored the post-accident flooding indication, but because of the associated accuracy problem, requested level information from Operator #1 reading the emergency systems indication. This interrupted the work of Operator #1 and added to his already (apparently) high workload. This elevated workload level raises the possibility of errors in display reading and communication.

An additional difficulty with reactor vessel level control during an ATWS concerns the High Pressure Coolant Injection (HPCI) system. SASA analysts report that the HPCI system will automatically and irreversibly shift its suction from the condensate storage tank to the PSP on a PSP high level signal. The difficulty is with the HPCI turbine lube oil, which is cooled by the water being pumped. As time passes and the temperature of the PSP increases, the HPCI turbine will eventually fail when the hotter, less viscous lube oil no longer protects turbine internals. SASA analysts report that in the worst case scenario, involving no operator actions, HPCI suction shift occurs at 9.5 minutes after an ATWS initiation, and HPCI fails at 16.3 minutes. Loss of HPCI results in insufficient injection to maintain the water level at the top of the fuel, forcing the operator to

**Table 2.1. Typical diferences in indicated level between the emergency systems indication and the post-accident flooding indication**

| Pressure (psia) | Actual Level (in.) 560 | Actual Level (in.) 380 |
|---|---|---|
| 1000 | ESI[a] = 560<br>PAFI[b] = 473 | ESI = 380<br>PAFI = 337 |
| 15 | ESI = 588[c]<br>PAFI = 560 | ESI = 373[d]<br>PAFI = 380 |

[a]ESI = emergency systems indication.

[b]PAFI = post accident flooding indication.

[c]Pointer pegged at upper end of scale.

[d]Pointer pegged at lower end of scale.

depressurize the vessel to a point permitting use of low pressure injection systems. Operators could prevent HPCI suction shift by racking out the breakers to the valve motor operators for the suction valves from the PSP, but this would be an extraordinary action not found in procedures. Once the suction shift occurs, operators could still manually trip the turbine to avoid its loss. The HPCI system would then be available for use when the suppression pool temperature is reduced. An anecdotal observation from the simulator sessions was that, during the first run when HPCI automatically shifted over, the operator manually shifted the Low Pressure Coolant Injection (LPCI) system so that it too took a suction on the PSP. This error of commission subsequently resulted in loss of all high pressure injection systems in that simulator run.

#### 2.2.5.3 Problem Resolution

Operator ability to maintain the water level at the top of the fuel could be supported through several means. First, the EPGs themselves identify several injection systems available for the operator to control. Second, TVA has some plans to upgrade computer software for the BF1 simulator to increase its fidelity. This effort is in response to new operator actions specified in the EPGs. Operators will then be trained on the EPGs through simulator practice on ATWS conditions, as well as on other severe accidents. Third, and in support of the previous point, the instructors involved in this study's simulator exercises reported an increase in success across trials in maintaining the level at the top of the fuel during poison injection. However, several considerations limit confidence in inferences drawn from such preliminary observations. Among these considerations are limitations within the computer software supporting the simulator and the validity of results based on only two SRO-instructors using draft procedures.

The human engineering difficulties with vessel level indications could be corrected by several different fixes. It is noted that the post-accident flooding instrumentation is designed to be used in large-break LOCA situations in which the reactor vessel would be depressurized. TVA is in the process of designing a Safety Parameter Display System (SPDS) for BF1. Other BWRs with an SPDS compensate the readings from the instruments in the computer code when the reactor is pressurized. Another advantage of the SPDS is that all available level instruments are used to calculate the reactor level value that is displayed. Installation of a large digital indicator to display the reactor water level calculated by the SPDS would give the operator a reliable indicaton of reactor water level under any vessel pressure condition.

The complexities and implications with the HPCI suction shift to the PSP and eventual failure of the HPCI turbine should be addressed in the EPGs. While intended as a response to LOCA conditions, it seems undesirable to lose HPCI during an ATWS. Should the operator choose to shut down the HPCI turbine, there are at least two potential methods that may be employed. First, the steam supply may be isolated by depressing the isolation pushbutton. This pushbutton closes the two containment isolation valves and is active only when an initiation signal is present. If the operator made the decision to return HPCI to service at some time in the future, the isolation signal must be reset and the isolation valves reopened.

Second, rather than isolate the HPCI steam supply, the operator can simply trip the turbine. The system operating instructions (BF OI-73) list the steps to perform this task. However, the procedure assumes that the initiating signal has been cleared before the turbine is tripped. With the initiation signal present, the turbine will restart as soon as the trip pushbutton is released. Also, the steam supply valve cannot be closed with the initiation signal present. The turbine can, however, be removed from service by the following steps:

1. Place the controller in manual and reduce turbine speed to 2150 RPM.
2. Place the auxiliary oil pump in "PULL TO LOCK" to prevent the pump from restarting.
3. Depress the turbine trip pushbutton until the turbine speed is zero.

Stopping the HPCI turbine by this method allows for quick restart should the operator need the pump in the future. To restart the turbine the operator simply starts the auxiliary oil pump and increases the turbine speed manually.

### 2.2.6 Emergency Depressurization

#### 2.2.6.1 Statement of the Problem

Operators may have considerable difficulty in completing instructions in the EPGs involving emergency depressurization of the reactor vessel and controlling coolant injection systems in order to maintain low vessel pressure. Large power and pressure excursions may occur as operators concurrently attempt both to keep the core covered with water and to maintain vessel pressure within certain tolerances by operating the SRVs.

#### 2.2.6.2 Performance Description

As reactor steam continues to be dumped to the PSP through the SRVs, pool temperature and level will steadily increase. Preserving the PSP's steam-quenching capabilities and preventing damage to the containment from thermal stress are responsibilities of plant operators. As engineering limits related to PSP pressure, level and load are exceeded, the EPGs require the operator to take action. Pertinent to an ATWS is the PSP heat capacity temperature curve, which essentially drives operator actions to depressurize the reactor to less than 200 psig. At low reactor pressure the demand on the PSP is greatly reduced and integrity preserved.

Reactor depressurization in the ATWS event is a difficult task to perform because of the frequent SRV operation and lack of valve position indication near the valve controls. To depressurize the reactor, one operator must observe the acoustic monitors on the back panel to determine which valves are open. A second operator at the valve controls should place the control switches for these valves in the OPEN position to hold them open with air pressure. Since the valves will no longer close when reactor pressure decreases, the reactor should begin to slowly depressurize. The depressurization rate can be increased by manually opening an additional SRV to increase the steam flow. As the desired pressure is approached, the operator should begin to close the SRVs individually until pressure stabilizes. This may be difficult because of the high capacity of the valves. At the lower pressure and steam production, one SRV may continue to depressurize the reactor. However, closing SRVs at low reactor pressure would have an undesirable effect on core reactivity and cause power and pressure spikes because of a reduction in core void fraction.

In a BWR, the formation of steam (voids) has a pronounced effect by adding negative reactivity. As voids are formed, denser moderator is displaced, and neutron thermalization is decreased. The probability of neutrons leaking from the core or undergoing resonance absorption increases. The neutron losses to leakage and resonance absorption result in fewer thermal neutrons being available to cause fission in the fuel. In general, the void coefficient becomes more negative at high void fractions and core exposure. When the reactor is depressurized, the decrease in saturation temperature causes a large increase in void formation. At the lower pressure a larger portion of the core volume is occupied by steam so that reducing reactor vessel pressure will lower reactor power under ATWS conditions. However, when the condition of high void fraction and low reactor pressure exists, a small increase in reactor pressure can cause a large decrease in void fraction. Relative change in specific volume per unit change in pressure is the reason for the large positive reactivity addition. As reactor pressure is lowered, large changes in specific volume of the voids can occur for small pressure changes. The operators should be aware of this during emergency depressurization and use caution when closing SRVs after depressurization. The SASA analysts have even recommended that the operators forego the depressurization because of the possible positive reactivity effects and the lack of SRV position indication.

Another method of taking advantage of the void coefficient of reactivity addressed in the EPGs is lowering the reactor water level to the top of the core. As the water level is lowered, the static head of water in the downcomer annulus and the carryover of liquid water from the core are decreased, reducing the natural circulation of coolant through the core. The boiling boundary (the transition point from single phase to two phase flow) in the channel is a function of coolant circulation through the fuel bundle. There are other factors involved in the location of the boiling boundary, but the coolant flow is the parameter of interest. As coolant flow through the channel decreases, the boiling boundary moves lower into the fuel bundles. The increased void formation in the channel adds negative reactivity. This method of increasing core average void fraction is more advantageous than depressurization since large changes in void fraction will not occur for small changes in pressure.

Water level control following an emergency depressurization is a crucial operator task requiring prior planning and immediate action. As the reactor is depressurized the high volume low pressure systems begin to inject water into the reactor. If the low pressure system injection is not controlled, the reactor vessel water level would quickly increase, flooding the core with cold water and a severe power transient would result.

As part of the planning prior to depressurization, the operator may consider tripping the condensate booster pumps and the low pressure ECCS pumps that are not needed to perform other functions. When the reactor level is reduced to the top of the active fuel, the core spray and LPCI systems receive an initiation signal and all pumps start. When reactor pressure decreases below 450 psig, the injection valves open. The operator can manually stop the pumps before reactor pressure decreases below 450 psig. However, if the pumps are tripped manually with the initiation signal present, they will not automatically restart if another initiation signal is received.

In the case of LPCI, the operator has a difficult decision to make. When the LPCI injection valves open, a five-minute timer starts. The injection valves cannot be closed until the timer expires. If the operator has overridden the automatic realignment of the RHR system into the LPCI mode by use of the containment spray select switch, the RHR system would remain aligned for suppression pool cooling at this time. If the operator has not overridden the automatic realignment, he must then decide if suppression pool cooling should be lost and if the RHR pumps should simultaneously inject into the reactor vessel.

During the depressurization, the operator must also be aware of the possibility of isolating the steam supply to the HPCI and RCIC turbines. The HPCI steam supply will isolate at 100 psig reactor pressure and the RCIC steam supply will isolate at 50 psig. It is desirable to prevent the isolations from occurring because of the preference of utilizing these systems over the low pressure systems. The LPCI system injects into the reactor recirculation pump discharge piping and the core spray system injects into spray spargers in the top of the core shroud. Because of the injection paths for these systems, very little heating of the coolant occurs prior to the water reaching the fuel. The cold water causes a large power excursion associated with the injection of these systems. However, the HPCI and RCIC systems inject into the feedwater system so that the cold water enters the downcomer region of the reactor. With the reactor vessel water level at the top of active fuel and the feedwater spargers uncovered, the water sprayed into the downcomer is heated by the steam and the hot vessel and vessel internals prior to circulation through the core. Therefore, the power excursions from HPCI and RCIC injection are much smaller than those resulting from core spray or LPCI injection. The operator must be aware of this during the depressurization and prevent the steam supply isolations.

#### 2.2.6.3 Problem Resolution

Several potential strategies may be identified for resolving difficulties with emergency depressurization and subsequent reactor vessel level control. First, SASA analysts recommend that operators not attempt manual control of the reactor vessel pressure under ATWS conditions (as well as prevent automatic depressurization system initiation). This recommendation is based on several considerations. These include the absence of immediate SRV auto position indication and the fact that a rapid drop in reactor vessel pressure may lead to core flooding by the low pressure injection systems and concomitant power and pressure excursions. SASA analysts concluded from their data that it is "extremely risky" to depressurize a BWR during an ATWS.

Second, SASA analysts examined effects from operators adhering to the EPGs. Operator actions included permanently opening three or more SRVs, preventing all injection (except from the CRDHS and the SLCS, if running) prior to depressurization, and restarting injection using one condensate pump, one condensate booster pump, and the main condenser hotwell. The reactor core is expected to be uncovered and critical for several minutes following reactor vessel depressurization. However, power excursions due to pressure increases are avoided since the SRVs have been manually positioned open and manually inserting the control rods adds negative reactivity to the core.

Third, TVA and SASA analysts had considered the possibility of a gradual, rather than a rapid, rate of depressurization. This would provide more time for the operators to track the reactor vessel level and to initiate low pressure injection. This strategy has apparently not been given any further attention.

Further research seems warranted to identify an optimal set of operator actions. Clearly, a fundamental question for an ATWS is whether to forego manual emergency depressurization, as SASA analysts recommended, or to adhere to the current EPGs in accordance with the PSP heat capacity temperature curve. Given the considerable complexity of controlling low pressure injection and potential subsequent pressure and power excursions, operators should be given additional guidance. Specifically, the EPGs should clarify how many SRVs should be opened and for how long, and they should prioritize use of different coolant injection systems based on such considerations as coolant flow rate and temperature. Following these recommendations, operators should receive training to address problems with reactor vessel depressurization. Whereas classroom instruction seems necessary to enhance the knowledge base of accident phenomenology, simulator practice seems essential for building skills in regulating reactor vessel depressurization and coolant injection systems.

### 2.3 Quantitative HRA

The purpose of the quantitative HRA was to provide some clarification of uncertainties in operator response during ATWS. Whereas results of HRA are useful for PRAs, they are also useful for identifying potential performance deficiencies which may be alleviated through training and simulator practice, procedures development, plant communications, and so forth. Some critical operator actions following the EPGs seemed suitable for the quantitative HRA since there seemed to be agreement between TVA and SASA analysts on the steps comprising the actions. These selected actions included (1) manual control rod insertion, (2) checking conditions and initiating SLC injection, (3) initiation of PSP cooling, and (4) SRV actuation preventing vessel overpressure. The remaining two critical actions included in the qualitative review, which were reactor vessel level control and emergency depressurization, seemed less suitable for the quantitative HRA because of the controversy over how they would be accomplished.

Presentation of the HRA is divided into four sections. First, methods for HRA reported in the literature are identified and briefly described. Second, a task analysis of the four selected critical operator actions was completed. Third, the steps in conducting the HRA using the Technique for Human Error Rate Prediction, or THERP (Ref. 15), were completed, resulting in a listing of the derived quantitative human reliability estimates. For some tasks, the analysis includes a description of assumptions reflecting how respective operator actions were incorporated into the computer code used by SASA analysts to study ATWS. The use of THERP was primarily relevant to estimating operator reliability for each of the individual critical tasks. Fourth, results of the analysis using the Operator Personnel Performance Simulation (OPPS) computer model (Ref. 16) were obtained. The uses of OPPS were to supplement the THERP analysis and complement the SASA analysis by providing a time-reliability estimate across all operator actions throughout the ATWS.

### 2.3.1 Methods for HRA

There are a range of methods proposed and developed for the quantitative analysis of operator error in HRA. These methods have previously been extensively reviewed in two NRC reports which were the front-end analysis supporting development of the Maintenance Personnel Performance Simulation (MAPPS) computer model (Ref. 17) and the Safety-Related Operator Actions (SROA) program (Ref. 16). These surveys were used in the examination of HRA methods in the context of assessing operator actions during an ATWS. The following discussion is intended only to briefly identify methods in HRA. The above two reviews, or original source documents, should be consulted if more detailed information is desired.

THERP is a recognized and accepted technique for assessing operator reliability in nuclear power plant operations (Ref. 18). It has undergone considerable development by Swain and his associates at Sandia National Laboratory (Ref. 19). THERP is a technique in which operator behaviors comprising a task are first identified through a task analysis. These discrete actions are assigned nominal human error probabilities (HEPs) which are modified by performance shaping factors (PSFs) and the level of dependence among successive task elements. Modified HEPs comprising the complete success path are used to calculate the final task success probability. Only actions for which errors would contribute to system failure are included in the calculations.

Several problems have emerged with regards to the use of THERP. First, the level of refined task information provided in the NRC's Task Data Forms, or TDFs (Ref. 14), is typically more detailed than the level called for in the THERP Handbook. Second, the matching of task analysis data using the NRC's TDFs with descriptions of operator actions listed in the THERP human error data base (Chapter 20 of Ref. 15) requires subjective judgment and has previously been found to be a source of potential error in this type of analysis (Ref. 20). The reliability of judgments between analysts in selection of HEPs for operator actions may need to be reviewed to ensure the accuracy of the analysis. Third, HEPs reported in the THERP human error data base have been subjected to some criticism dealing with their adaptation from a non-nuclear power plant operator source. However, the final version of this data base has reportedly been supplemented with HEPs from relevant sources. Other human error data bases are also available, such as those developed through simulator experiments (Ref. 16).

The previous ORNL surveys of the HRA methods have discussed a number of simulation models. Model development has substantial roots in military research and includes Siegel-Wolf network models (Ref. 21) and supervisory control models (Ref. 22). Simulation models provide analytic frameworks for systematically assessing effects from variations in input variables and process conditions on output variables. Computer models incorporate features pertinent to task performance and may include task, operator, time, and organization variables. As an example pertinent to this project, the OPPS model, developed in the SROA program (Ref. 16), simulates operator responses to transient conditions in a nuclear power plant. Results are in the form of a time reliability distribution. The OPPS model was programmed using the Systems Analysis of Integrated Networks of Tasks, or SAINT, simulation language (Ref. 23). During an OPPS iteration, the simulated control room crew is timed for completion of branches through pre-alarm detection, event diagnosis, selection of procedures, execution of operator actions following procedure steps, execution of actions outside the control room, and assessment of recovery from errors of omission and commission. An OPPS run consists of multiple iterations of the event, typically 1000. Iteration completion times are plotted by relative and cumulative frequencies. The cumulative frequency distribution may be interpreted as a time-reliability curve showing an increase in the probability of task completion following an increase in available time. Operator performance data bases used in OPPS development included results of SROA simulator experiments (Ref. 16) and the THERP human error data base (Ref. 15).

A second method utilizing the time reliability correlation is the Operator Action Tree, or OAT (Ref. 24). The OAT method bases the probability of operator failure on the amount of time available to both identify the situation and complete a set of appropriate actions. The time available is determined through an analysis of initial alerting cues and the amount of time before relevant system safety limits are exceeded. The time reliability correlation posits that the probability of operator failure decreases logarithmically with the time available for the operator to analyze the situation and complete all actions.

Another broad category of human reliability methodologies involves subjective techniques using expert opinion to systematically develop probability estimates of operator success (Ref. 25). For example, the Success Likelihood Index Methodology, or SLIM (Ref. 26), uses expert assessments of the utilities of PSFs, such as quality of training, quality of supervision, and quality of procedures, in effecting operator reliability. The SLIM technique uses a systematic procedure by which experts assign utilities to individual PSFs and combine these utilities into a consensus measure. Potential operator performance on the task under review is assessed with regards to information acquisition and goal setting, task execution, and error recovery. Operator actions are analyzed using PSFs which are assessed as being pertinent to each of these phases.

### 2.3.2 Task Analysis

An input requirement to most HRA methods including THERP is a task analysis providing systematic descriptions of operator actions. The task analysis of critical operator actions used in this review followed the standard NRC task analysis format (Ref. 14). According to this format, tasks are described at three levels of detail. As a general description, the Operating Sequence Overview identifies the general progression of actions by plant systems and operators. The ATWS overview incorporating operator actions required by the EPGs was previously shown in Figure 2.4. At an intermediate level of detail is the Task Sequence Chart (TSC). This chart identifies the normative ordering of tasks, the purpose of operator actions, cues that initiate each task, technical specifications, and plant systems involved in each task. The TSC for an ATWS is shown in Figure 2.6. The most specific level of detail is the Task Data Form (TDF). For each task, the TDF

TASK SEQUENCE CHART

Plant: BFNP

Operator Function/Subfunction:
Supervise and Control Plant Operations/Mitigate Consequences of an Accident

Operating Sequence: Anticipated Transient Without a Scram, Following MSIV Closure

Operating Sequence ID: 7

| Sequence Number | Task and Purpose | Cue | Procedure Name & Number | Plant Specific System Name |
|---|---|---|---|---|
| 1 | Recognize main steam isolation valves closed<br>To determine plant conditions | Alarms, indicator lights | GOI-100-1 Section VII Emergency Shutdown with MSIV Closure | Main steam |
| 2 | Recognize the reactor did not scram<br>To determine plant conditions | Alarms, indicator lights, digital rod position indicators | GOI-100-1 Section VII | Reactor protection system |
| 3 | Verify reactor recirculation pumps tripped<br>To verify automatic action occurred | Alarms, procedure | RC/Q-3 | Reactor recirculation |
| 4 | Monitor reactor pressure<br>To ensure pressure limits are not exceeded | Procedure | RC/P | Reactor protection vessel |
| 5 | Monitor reactor level<br>To ensure level limits are not exceeded | Procedure | RC/L | High pressure coolant injection, reactor core isolation cooling |
| 6 | Place mode switch in SHUTDOWN<br>To enforce RPS interlocks, and generate a reactor scram signal | Procedure | RC/Q-1 | Reactor protection system |
| 7 | Attempt manual scram of reactor<br>To insert control rods | Procedure | RC-1 | Reactor protection system |

**Figure 2.6. Task sequence chart for an ATWS.**

## TASK SEQUENCE CHART

Plant: BFNP

Operator Function/Subfunction: Supervise and Control Plant Operations/Mitigate Consequences of an Accident

Operating Sequence: Anticipated Transient Without a Scram, Following MSIV Closure

Operating Sequence ID: 7

| Sequence Number | Task and Purpose | Cue | Procedure Name & Number | Plant Specific System Name |
|---|---|---|---|---|
| 8 | Verify initiation of coolant injection systems<br>------------------------------<br>To verify automatic action occurred | Procedure | RC/L-2 | ECCS |
| 9 | Align main steam isolation valve switches<br>------------------------------<br>To stop venting air to the drywell | Procedure | GOI-100-1 Section VII | Main steam |
| 10 | Verify primary containment isolation<br>------------------------------<br>To verify automatic actions occurred | Procedure | RC/L-1 | Primary containment isolation system |
| 11 | Verify safety/relief valves open on overpressure<br>------------------------------<br>To verify safety/relief valves are limiting react[illegible] pressure | Procedure | RC/P | Main steam |
| 12 | Manually operate safety/relief valves if any SRV is cycling<br>------------------------------<br>To limit reactor pressure | Procedure | RC/P-1 | Main steam |
| | | | | |

Figure 2.6. Task sequence chart for an ATWS (cont.).

TASK SEQUENCE CHART

Plant: BFNP

Operator Function/Subfunction: Supervise and Control Plant Operations/Mitigate Consequences of an Accident

Operating Sequence: Anticipated Transient Without a Scram, Following MSIV Closure

Operating Sequence ID: 7

| Sequence Number | Task and Purpose | Cue | Procedure Name & Number | Plant Specific System Name |
|---|---|---|---|---|
| 13 | Monitor containment conditions<br>------------------------------<br>To ensure pressure and temperature limits are not exceeded | Procedure | PC | Primary containment |
| 14 | Monitor suppression pool temperature<br>------------------------------<br>To prevent exceeding torus heat capacity temperature limit | Procedure | RC/P | Primary Containment Instrumentation |
| 15 | Control reactor pressure and injection during depressurization<br>------------------------------<br>To maintain torus heat capacity temperature limit | Procedure | Contingency #2<br>Contingency #7 | Main steam, ECCS |
| 16 | Request auxiliary operator to scram individual control rods<br>------------------------------<br>To reduce reactor power | Procedure | RC/Q-5.4 | Reactor protection system |
| 17 | Bypass the rod worth minimizer<br>------------------------------<br>To permit control rod insertion using reactor manual control | Procedure | Caution #20 | Rod worth minimizer |
| 18 | Manually insert control rods<br>------------------------------<br>To reduce reactor power | Procedure | RC/Q-5.6 | Reactor manual control |

**Figure 2.6. Task sequence chart for an ATWS (cont.).**

# TASK SEQUENCE CHART

Plant: BRNP

Operator Function/Subfunction: Supervise and Control Plant Operations/Mitigate Consequences of an Accident

Operating Sequence: Anticipated Transient Without a Scram, Following MSIV Closure

Operating Sequence ID: 7

| Sequence Number | Task and Purpose | Cue | Procedure Name & Number | Plant Specific System Name |
|---|---|---|---|---|
| 19 | Monitor reactor power<br>------------------------------<br>To determine effectiveness of control rod insertion | Procedure | RC/Q | Neutron monitoring |
| 20 | Insert source range monitor and intermediate range monitor detectors<br>------------------------------<br>To monitor neutron flux decrease | Procedure | GOI-100-1 Section VII | Nuclear instrumentation |
| 21 | Initiate suppression pool cooling<br>------------------------------<br>To limit suppression pool temperature | Procedure | SP/T-2 | Residual heat removal |
| 22 | Declare alert status<br>------------------------------<br>To notify plant and public safety personnel | Procedure | Implementing Procedures 1 and 3 | Radiological emergency procedures |
| 23 | Verify conditions exist for initiating standby liquid control<br>------------------------------<br>To determine the need for poison injection | Procedure | RC/Q-4 | Control rod position indication, reactor vessel instrumentation, containment monitoring |
| 24 | Lower and maintain water level at Top of Active Fuel<br>------------------------------<br>To reduce power level | Procedure | Contingency #7 | Reactor Process Control Instrumentation |

Figure 2.6. Task sequence cha· ·or an ATWS (cont.).

## TASK SEQUENCE CHART

Plant: BFNP

Operator Function/Subfunction: Supervise and Control Plant Operations/Mitigate Consequences of an Accident

Operating Sequence: Anticipated Transient Without a Scram, Following MSIV Closure

Operating Sequence ID: 7

| Sequence Number | Task and Purpose | Cue | Procedure Name & Number | Plant Specific System Name |
|---|---|---|---|---|
| 25 | Initiate standby liquid control injection<br>--------------------------------<br>To reduce reactor power by poison injection | Procedure | RC/Q-4 | Standby liquid control, reactor water cleanup |
| 26 | Monitor power decrease<br>--------------------------------<br>To determine effectiveness of standby liquid control, control rod insertion and water level reduction | Procedure | RC/Q | Neutron monitoring |
| 27 | Raise water level when sufficient boron is injected<br>--------------------------------<br>To mix poison injected into bottom head | Procedure | C7-3 | Standby liquid control, Coolant Injection Systems |

**Figure 2.6. Task sequence chart for an ATWS (cont.).**

lists discrete operator actions, the object of each action, the means by which the action is conducted, and communication links. The completed TDFs for the tasks selected for HRA are contained in Appendix A.

Inputs to the task analysis leading to completion of the TDFs included the following resources.

1. BF1 procedures including EPGs, EOIs, and general operating instructions.
2. Videotapes of BWR SRO-instructors conducting exercises of ATWS perturbations on the BF1 control room simulator.
3. Computer records of operators' switch manipulations during the simulator exercises collected through the Performance Measurement System, or PMS (Ref. 27). The PMS also provides continuous data on selected critical plant safety parameters.
4. Expert judgment from operators and human factors personnel.

### 2.3.3 THERP Results

In accordance with THERP procedures, the task analysis data were used to guide selection of nominal HEPs from the THERP human error data base (Chapter 20 of Ref. 15). Selection of HEPs was coordinated between two of the authors to verify the reasonableness of HEP selection. A HEP worksheet was developed to organize and document the THERP analysis. These worksheets, which were completed for the critical tasks selected for HRA, are shown in Appendix B.

Following the THERP Handbook, nominal HEPs were modified to reflect effects from PSFs and dependence. One PSF assumed to bear on operator performance during an ATWS was stress. The effect of stress on performance was assumed to weigh more significantly on the initial determination of whether to perform the task given the abnormal condition of the plant. That is, stress was held to more likely distract the operator from executing the task, but once the task is undertaken, operator competence overrides adverse effects from stress. This assumes that the operator has correctly determined which task needs to be performed. Attributing stress effects to decision-making seems a better reflection of the complex and confusing stimuli that operators are attempting to filter. Once a course of action is selected, the relative effects of stress are reduced. This description parallels the distinction made in the THERP Handbook between dynamic decision-making tasks and step-by-step tasks. That is, HEPs are more heavily modified by stress for dynamic tasks.

HEPs were further modified to reflect effects of dependence between operator actions. Dependence may be defined as the extent to which success on one discrete action is affected by success on the previous action. Dependence was assessed for these tasks using guidelines and examples from the THERP Handbook. Assessments of the extent of dependence were based on the previously identified inputs to the task analysis.

Modified HEPs comprising complete success paths were used to calculate final task success probabilities. Only actions for which errors would contribute to system failure were included in the calculations. Uncertainty bounds (UCBs) were also factored in to reflect best case (lower UCB) and worst case (upper UCB) performance. In most cases, except as noted below, error factors (EFs) for the UCBs were used in calculations to show effects from stress on initiating execution of procedures under off-normal plant conditions. Derived estimates of failure probabilities are reported in Table 2.2 for the tasks assessed using THERP.

#### 2.3.3.1 SRV Actuation Preventing Vessel Overpressure

Control of the vessel pressure by manual operation of the SRVs has an estimated HEP of 2.72E-02. The THERP event tree is shown in Figure 2.7 with individual HEPs adjusted in accordance with the preceding discussion. The task is initiated by the operator reading vessel pressure indications at some time early in the accident sequence. There seems to be high dependence that the operator will execute the appropriate procedure given recognition of high vessel pressure causing the SRVs to cycle automatically. There also seems to be complete dependence of the operator positioning the valve open or closed given success in recognizing high vessel pressure. Success on this task, then, seems to be determined primarily by correct readings of pressure indications over the course of the accident sequence.

For comparison purposes, SASA analysts made two assumptions regarding operators controlling SRVs which were incorporated into the thermohydraulics computer code. These assumptions were that the operator checks the vessel pressure once per minute and that one SRV may be manipulated per minute depending upon the presence or absence of certain conditions. These conditions were whether the reactor vessel was at high pressure (desired vessel pressure is between 1050 and 950 psia) or low pressure (desired vessel pressure is between 300 and 0 psia after emergency depressurization). An additional condition is the deviation over time from the desired bounds, i.e., an SRV is opened or closed if the vessel pressure is outside the bounds by 60 psi compared to the check one minute earlier or by 120 psi compared to the check three minutes earlier.

#### 2.3.3.2 Manual Control Rod Insertion

The unreliability of the manual insertion of control rods by the operator has an estimated HEP of 1.82E-01 and requires careful interpretation. This HEP was calculated on the basis of selection of approximately twelve control rods inserted in such a pattern that, in combination with poison injection, power was reduced to less than one percent on the simulator computer. The selection, insertion, and position change verification of a single control rod has an estimated HEP of 9.48E-03, adjusted for dependence. The discrete operator actions necessary for driving in one control rod are shown in Figure 2.8. It was assumed that there is low dependence of success on enabling the respective master control pushbutton given success on reading the digital counter, e.g., the operator may scan many counters before selecting and enabling what is perceived to be a high worth rod. It was also assumed that there is moderate dependence of success on turning the rod insertion switch to begin rod movement given success in enabling the master control pushbutton, e.g., the operator may be scanning the counters and identify a different rod to be inserted first. Lastly, it was assumed that there is high dependence of success on observing the

**Table 2.2. Estimates of human failure probabilities for selected tasks during ATWS**

| Task Description | Nominal HEP | Uncertainty Bounds Upper | Uncertainty Bounds Lower |
|---|---|---|---|
| Manually operate SRVs before 1105 psig reactor pressure is reached | 2.72E-02 | 2.61E-01 | 1.74E-02 |
| Manual control rod insertion | 1.82E-01 | 3.71E-01 | 1.63E-01 |
| Initiate suppression pool cooling | 1.27E-01 | 3.28E-01 | 3.92E-02 |
| Verification of conditions for and initiation of SLC injection | 3.69E-02 | 2.59E-01 | 1.47E-02 |

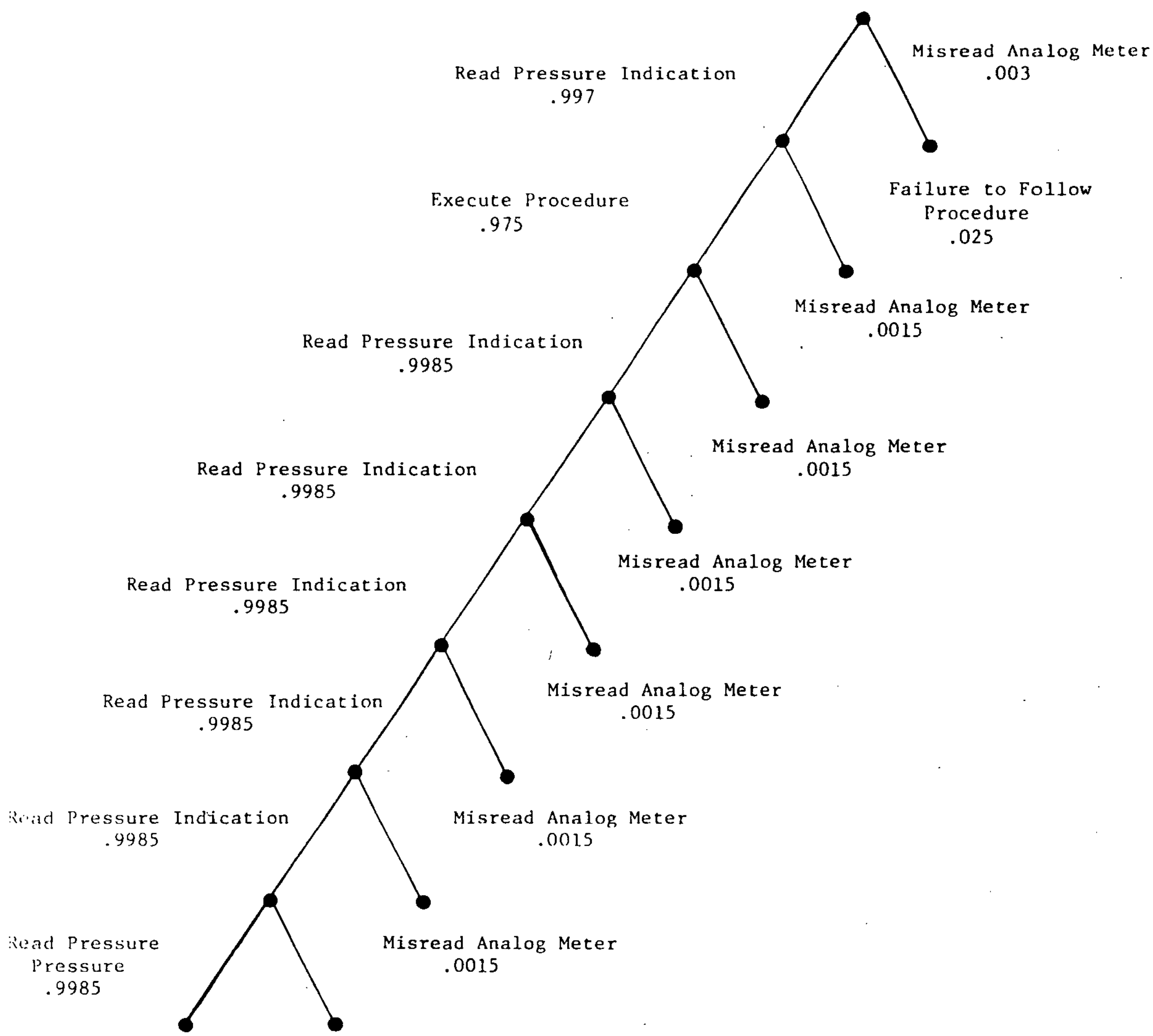


**Figure 2.7. HRA event tree for operation of safety/relief valves to prevent reactor vessel overpressure.**

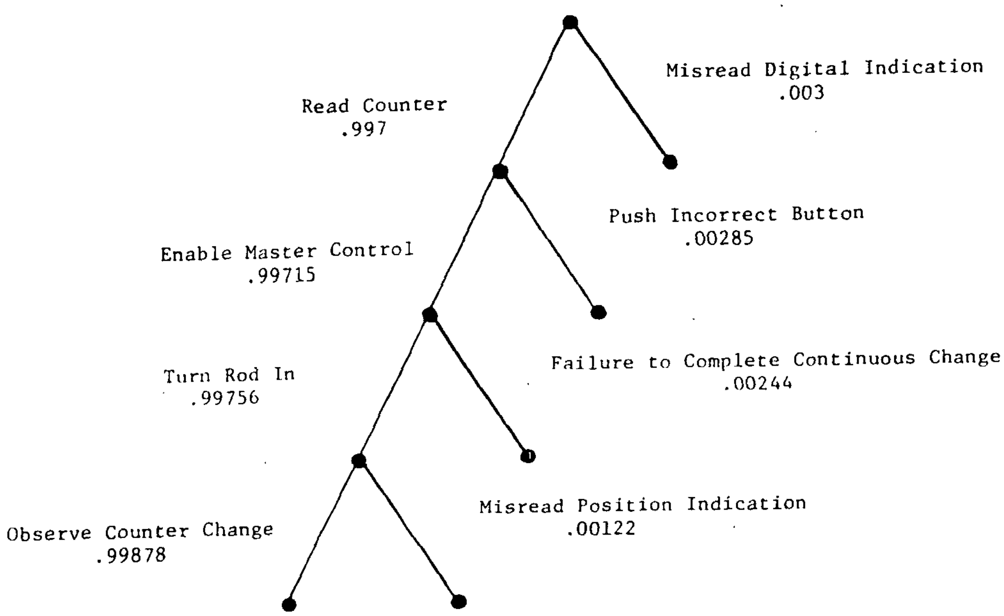


**Figure 2.8. HRA event tree for manual insertion of one control rod.**

counter change given success on turning the rod switch to the insert position, e.g., the operator may need to determine whether sufficient rod movement has occurred to warrant proceeding to selection and insertion of the next control rod.

Performance of the entire task, however, includes operation of the master group select switch when the operator shifts from one group of control rods to another according to the rod pattern being developed. The operator may also be called away from this task to perform/assist with other tasks, such as checking conditions for and initiating SLC injection. Such interruptions will likely have some effect on the reliability of successful task performance.

Interpretation of the estimated task HEP must consider that there were 85 task elements included in the task. It is important to note that the overriding significance of this task to mitigation of ATWS by bringing the reactor to a subcritical state supports an assumption that most errors would eventually, if not immediately, be recovered by the reactor operator.

The code used by SASA analysts in their assessment of ATWS reflected certain assumptions concerning manual rod insertion. The analysts assumed this effort would commence at three minutes after ATWS initiation. Based on observations of the simulator exercises, SASA analysts assessed operators to be drawn away from this task 50% of the time. This was operationalized in the computer code by doubling the effective nominal average speed of control rod movement.

**2.3.3.3 Initiate PSP Cooling**

Operator initiation of PSP cooling has an estimated HEP of 1.27E-01. The THERP event tree identifying necessary task elements is shown in Figure 2.9. The level of dependence was held to be high across these operator actions for several reasons. BF1 has two loops of the RHR system for PSP cooling, and operators are trained to simultaneously initiate these loops. RHR controls are located on the same panel. The THERP Handbook considers manipulation of paired controls to comprise complete dependence, so this THERP event tree reflects initiation of only one RHR loop.

A major contributor to operator error is a failure to recognize the increase of PSP temperature, including acknowledgment of the PSP high temperature annunciator within the first ten minutes of its initiation. THERP uses a time reliability distribution for assigning HEPs in situations involving failure to diagnose events. Within the first ten minutes of problem initiation the HEP is 0.1, which was used in calculating the nominal HEP. From ten to twenty minutes, the HEP for failure diagnosis is 0.01. This indicates that the operator is more likely to recognize the heatup of the PSP as more time passes. The upper UCB is based on a diagnosis failure during the first ten minutes and worst case high stress, whereas the lower UCB assumes less probable diagnosis failure and nominal high stress.

SASA analysts, in incorporating the actions concerning PSP cooling into their code, responded to the uncertainty in the timing of initiation of PSP cooling by assuming

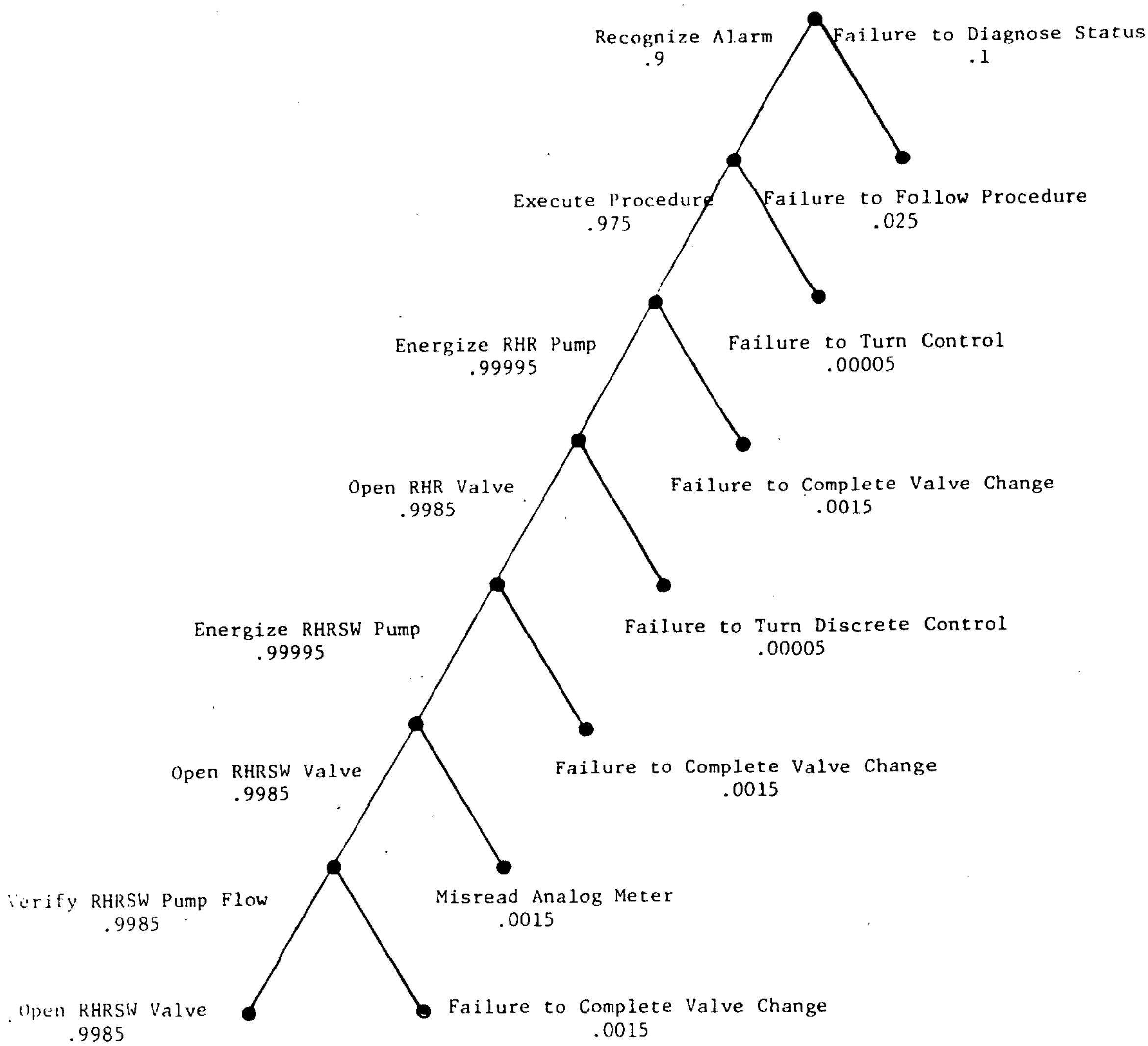


**Figure 2.9. HRA event tree for initiating suppression pool cooling.**

operators would commence this task at ten minutes. One of the results from this timing of operator response was that, for the ATWS scenario involving SLC injection and manual control rod insertion, by 17 minutes into the sequence the four RHR coolers were removing as much heat as the SRV discharge was adding from the reactor vessel.

#### 2.3.3.4 Checking Conditions and Initiating SLC Injection

Injection of sodium pentaborate solution from the SLC system has an estimated HEP of 3.69E-02. This action is actually a combination of two tasks: verify that conditions exist for initiating SLC and initiate SLC injection. Clearly, the composite HEP is of primary interest because both are essential if the functional requirement of adding negative reactivity is to be satisfied using the SLC system. The THERP event tree identifying necessary operator actions is shown in Figure 2.10. The level of dependence was held to be moderate across all these actions.

The complexities of this task include the considerable difficulty operators would have in deciding to execute the task and the high level of stress accompanying the decision. Based on these considerations, it may be more appropriate to take the worst case scenario and use the upper UCB (2.59E-01) as a more conservative estimate.

Through their computer code SASA analysts estimated that SLC injection would commence at five minutes after initiation of an ATWS. Conditions necessary to proceed with poison injection would be estimated to exist about two minutes into the accident sequence. Additional time was provided within the operator action model to reflect attempts to obtain an alternate scram of the control rods.

### 2.3.4 OPPS Results

Supplementary assessment of operator actions throughout the ATWS was provided through use of the OPPS computer model (Ref. 16). Results of the OPPS analysis include the relative and cumulative frequency distributions shown in Figure 2.11. The time-reliability curve is taken as the cumulative frequency distribution showing an increase in successful completion of operator actions with an increase in time. This OPPS analysis was based on 1000 iterations of simulated operator response to an ATWS. Performance time for completion of all operator actions averaged 2005 seconds (33.42 minutes) with a minimum of 1382 seconds (23.03 minutes) and a maximum of 2629 seconds (43.82 minutes). The average number of errors of omission was 3.68, or about 3.5% of all actions.

The input and assumptions to this OPPS analysis are that 105 control room switch manipulations are necessary (based on the task analysis) to mitigate an ATWS and that equipment delay time was embedded in the procedures. Regarding diagnosis of an ATWS, branches selected were that annunciators indicate specific conditions rather than general alarms for identifying an ATWS, that five indications are sufficient to diagnose the type of disturbance, and that operator diagnosis is terminated at the symptom level using the EPGs rather than extending to the root cause of rod failure to insert. Additional branches concerning planning and procedures were selected to reflect that procedures are written and indexed, that immediate actions are memorized by the operator, and that the ATWS scenario is used in training.

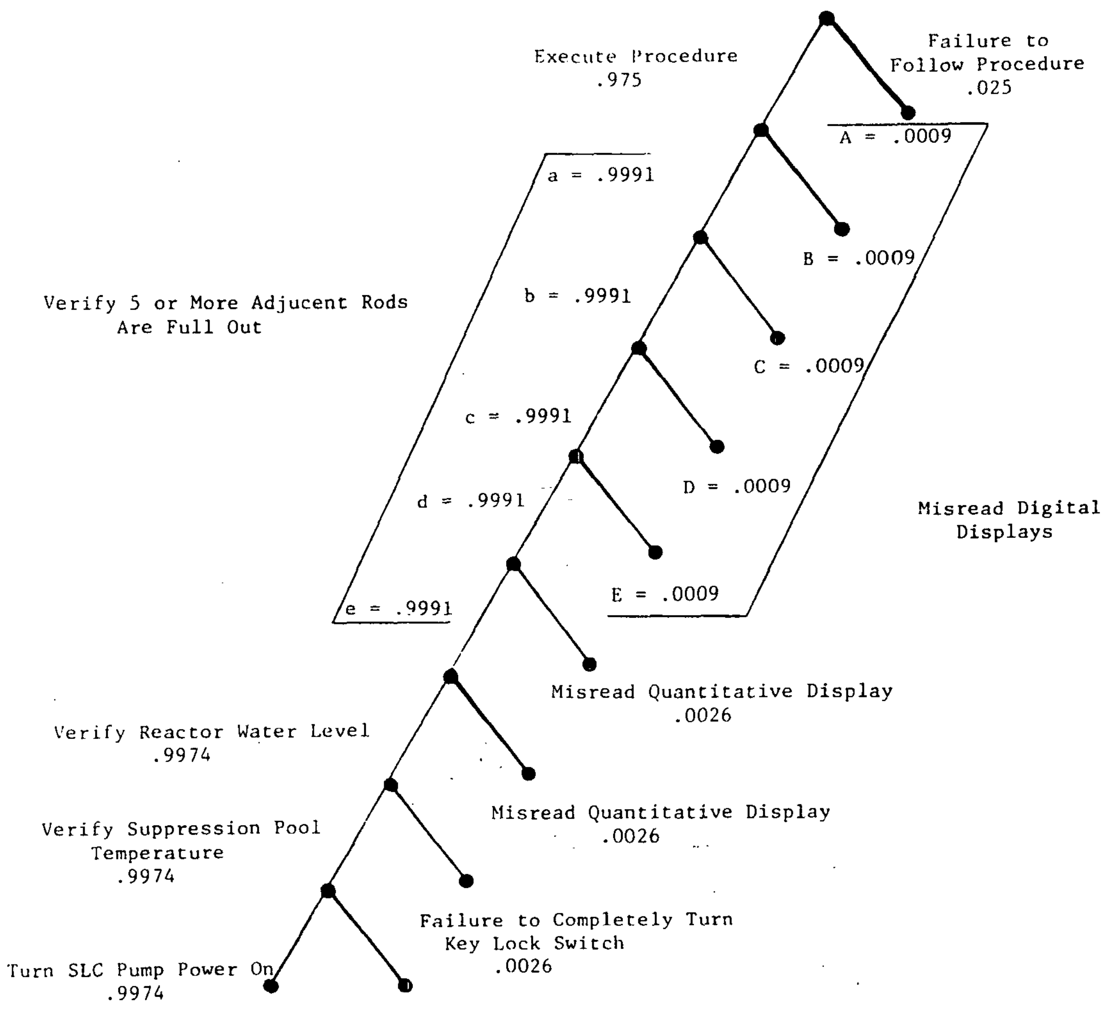


**Figure 2.10. HRA event tree for verifying conditions and initiating injection of SLC tank.**

***HISTOGRAM OF THE AVERAGE FIR STA STATISTIC FOR TASK 41 (STOP )***

| OBSV FREQ | RELA FREQ | CUML FREQ | UPPER CELL LIMIT |
|---|---|---|---|
| 0 | 0.0 | 0.0 | 0.1200E+04 |
| 0 | 0.0 | 0.0 | 0.1260E+04 |
| 0 | 0.0 | 0.0 | 0.1320E+04 |
| 0 | 0.0 | 0.0 | 0.1380E+04 |
| 3 | 0.003 | 0.003 | 0.1440E+04 |
| 0 | 0.0 | 0.003 | 0.1500E+04 |
| 6 | 0.006 | 0.009 | 0.1560E+04 |
| 8 | 0.008 | 0.017 | 0.1620E+04 |
| 29 | 0.029 | 0.046 | 0.1680E+04 |
| 33 | 0.033 | 0.079 | 0.1740E+04 |
| 66 | 0.066 | 0.145 | 0.1800E+04 |
| 96 | 0.096 | 0.241 | 0.1860E+04 |
| 102 | 0.102 | 0.343 | 0.1920E+04 |
| 130 | 0.130 | 0.473 | 0.1980E+04 |
| 130 | 0.130 | 0.603 | 0.2040E+04 |
| 93 | 0.093 | 0.696 | 0.2100E+04 |
| 93 | 0.093 | 0.789 | 0.2160E+04 |
| 67 | 0.067 | 0.856 | 0.2220E+04 |
| 51 | 0.051 | 0.907 | 0.2280E+04 |
| 33 | 0.033 | 0.940 | 0.2340E+04 |
| 22 | 0.022 | 0.962 | 0.2400E+04 |
| 17 | 0.017 | 0.979 | 0.2460E+04 |
| 16 | 0.016 | 0.995 | 0.2520E+04 |
| 3 | 0.003 | 0.998 | 0.2580E+04 |
| 2 | 0.002 | 1.000 | 0.2640E+04 |
| 0 | 0.0 | 1.000 | 0.2700E+04 |
| 0 | 0.0 | 1.000 | 0.2760E+04 |
| 0 | 0.0 | 1.000 | 0.2820E+04 |
| 0 | 0.0 | 1.000 | 0.2880E+04 |
| 0 | 0.0 | 1.000 | 0.2940E+04 |
| 0 | 0.0 | 1.000 | 0.3000E+04 |
| 0 | 0.0 | 1.000 | INF |
| --- | | | |
| 1000 | | | |

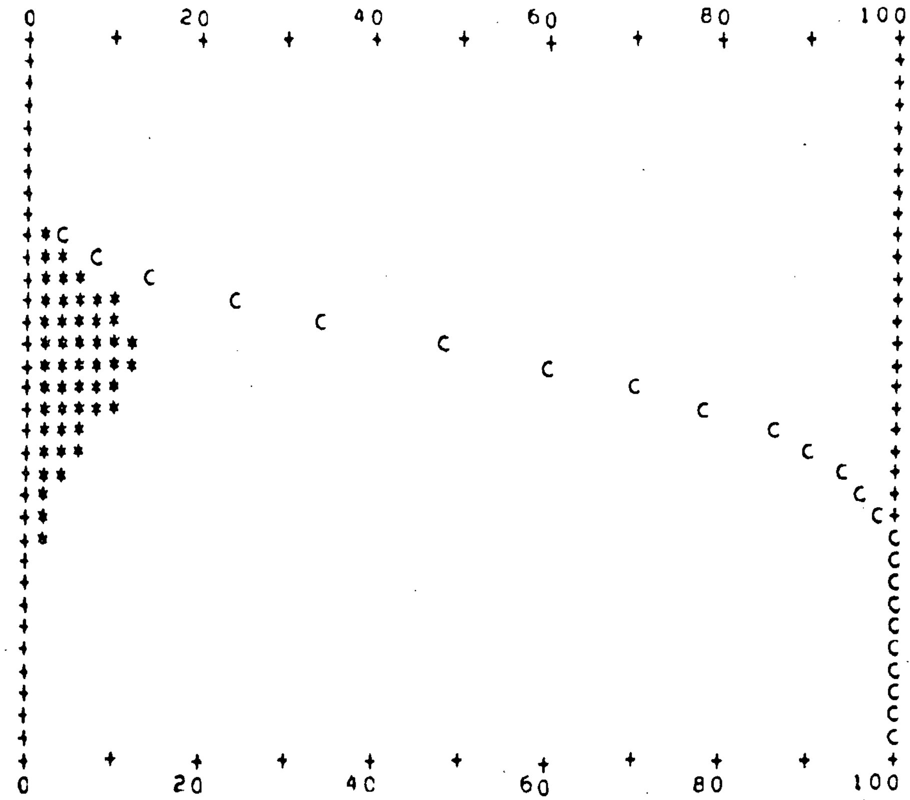


**Figure 2.11. OPPS time reliability distribution with relative cumulative frequencies.**

For comparison purposes, SASA analysts in their baseline worst case scenario of an ATWS involving no operator actions reported that containment failure would occur at 36.8 minutes into the accident. With the OPPS model calculating an average simulated performance time of 33.42 minutes, these analyses suggest operators should have sufficient time to complete all actions necessary to shut the reactor down. Moreover, not all safety-related actions would have to be completed within this time since the more critical actions would likely be performed early during the accident and would slow accident progression. Performance of these critical actions would also extend the time remaining for the operators to complete the other necessary actions.

The advantages and disadvantages of the OPPS model can be summarized as follows. A major disadvantage is that it is a simplified model of the control room operator. For example, it assumes that all errors are eventually recovered, regardless of the amount of time required. Among its advantages is that fact that the OPPS model provides results which can be compared to the results of the SASA analysis. Results of the THERP analysis do not readily lend themselves to comparisons with SASA results. Also, time-reliability distributions provided by OPPS seem superior to OAT curves in that time distributions used in the OPPS model were collected through the SROA program at ORNL in which operators were actually timed performing various actions.

# 3. HUMAN FACTORS RESEARCH IN ACCIDENT MANAGEMENT

The objectives of operator response in accident management, as identified in the Severe Accident Research Plan, or SARP (Ref. 28), have been defined in "... a broad context that includes preventing accidents, arresting the course of an accident, and mitigating the consequences of an accident." Due to the paucity of human factors research in accident management and the amorphous boundaries of the area, this project undertook development of an exploratory, conceptual approach for describing operator responses in accident management. This conceptual approach may support needs analysis and development of human factors guidance for enhancing operator response toward accident mitigation. The approach selected employs a functional analysis to systematically conceptualize linkages among safety functions, control requirements, unconventional emergency responses, and control of radiological releases.

Results of this research led to the development of the Function Oriented Accident Management (FOAM) model. The FOAM model provides both a conceptual structure systematically describing operator response in accident management and a method for standardized development of technical guidance supporting operator decision-making and response. Guidance is necessary because accident conditions are assumed to exceed the scope of existing emergency procedures. Technical guidance may be developed through integration of data from SASA and PRA studies and expertise compiled from operations, engineering and human factors personnel. This guidance is central for extending emergency procedures and systematic operator training, as well as other important human factors issues in operator performance.

In this section, selected background literature is reviewed for the purpose of developing a baseline of information on factors shaping operator response in accident management. The FOAM model is subsequently described with a brief overview of its structure and a detailed discussion of each of the four components comprising the model. A table-top demonstration of the FOAM model is presented to illustrate its potential use and expected results. Several potential applications of the FOAM model are then identified related to regulatory, industry and research perspectives, and are oriented toward the enhancement of accident management practices.

## 3.1 Review of Selected Research Literature Relevant to Accident Management

The purpose of this section is to develop a baseline of information on factors expected to effect operator performance during severe emergency accident conditions. First, NRC regulatory requirements which define nuclear power plant emergency administrative planning and preparedness, as well as emergency response facilities requirements, are identified. Second, in recognition of the importance of operator decision making and problem solving for accident mitigation, several theoretical models of operator cognitive behavior are reviewed. Third, approaches to the development of plant emergency procedures are examined, including French accident management procedures which represent a unique approach to this problem area. Fourth, several operator training programs for accident management and mitigation of core damage are reviewed. Fifth, developments in computer-based operator aids supporting operator response are sampled.

It is noted that these reviews were intended to be representative samplings of the subject-matter domains and not comprehensive literature surveys.

### 3.1.1 Human Factors-Related Regulatory Requirements in Accident Management

A set of regulatory requirements specifically addresses human factors issues pertinent to severe accident management. These requirements involve emergency response facilities (ERFs) and administrative radiological emergency plans (REPs).

The ERFs consist of three types of facilities and two types of data systems. The objective of all ERFs is to support and improve operator detection, diagnosis and response to emergency situations. Functional criteria for the development of ERFs are described in Reference 29, and a draft methodology for the evaluation of ERFs is reported in Reference 30.

The three types of emergency facilities and their functions are:

1. Technical Support Center (TSC) — provide on-site plant management and technical support to the control room crew during an emergency, including technical (plant data and records) and administrative (health physics and communications) support.
2. Operational Support Center (OSC) — provide an on-site assembly area for coordinating logistic support and restrict control room access.
3. Emergency Operations Facility (EOF) — provide a near-site support facility for overall emergency management and coordination of radiological assessments and public protective actions.

The two data systems are the Safety Parameter Display System (SPDS) and the Nuclear Data Link (NDL). The SPDS employs a computer-based, human engineered display of plant safety parameters to facilitate operator detection of abnormal operating conditions. Computer-based operator aids are described in more detail in Section 3.1.5. The NDL acquires and relays information on certain plant safety parameters, as well as radiological and site meteorological data, from the NPP to the NRC Operation Center.

Through coordination with the Federal Emergency Management Agency (FEMA), each plant is required to prepare an REP meeting certain design and evaluation criteria (Ref. 31). The scope of the REP includes 16 different planning standards. For example, one standard involves classification of events according to four emergency action levels, which are, in order of increasing severity: (1) notification of unusual event, (2) alert, (3) site area emergency, and (4) general emergency. For each plant, threshold values of critical parameters for equipment state, electrical power, radiological conditions, security, medical emergency, natural phenomena, fire or explosion, and other conditions are specified for entering emergency action levels. Another part of the REP consists of implementing procedures addressing actuation of emergency facilities, alerting plant personnel to report to the plant, and emergency notification of off-site authorities.

These regulatory requirements establish certain technical resources and administrative procedures that have been judged to be important in accident management. However, these administrative procedures do not address the actions that emergency facilities staffs should take to mitigate accident progression as would extended emergency operations procedures. The difficulty in extending emergency operating procedures to deal with severe accidents is the absence of any technical guidance that integrates data and hypotheses of accident phenomenology. The need for such technical guidance is unequivocably emphasized in NUREG-0899 (Ref. 3) as required input for procedure development. Moreover, a standardized structure for organizing this technical guidance would enhance integration of varied methodologies and data.

### 3.1.2 Operator Cognitive Behavior

The purpose in reviewing literature on operator cognitive behavior is to identify theories and approaches to assessing operator decision making and problem solving applicable to accident management. A rich source of information is the proceedings of a workshop concerned with cognitive modeling of nuclear plant control room operators (Ref. 32). The following is intended to be only a sampling of theories and approaches.

Rasmussen proposed a three-level hierarchical conceptualization of operator behavior (Ref. 33). At the lowest level, operator behavior is held to be skill-based with actions having a reflexive quality. During a plant event operator behavior will typically be rule-based in that emergency procedures provide rules or guidance which operators should follow. Symptom-based emergency procedures represent a set of rules linking the symptoms of off-normal safety functions with requisite operator actions for correcting these symptoms. For an event exceeding the scope of procedures, operator behavior would become increasingly knowledge-based. Operators would be expected to use their personal expertise to detect, identify and solve problems. Knowledge-based behavior becomes more important for some events such as those which involve multiple system failures. A combination of such problems may affect plant operations in such a manner that operator actions in following emergency procedures do not result in the desired change. Whereas Rasmussen's approach provides general descriptions of behavior which appear to have face validity, his conceptualization does not readily lend itself to a methodology for the systematic detailed analysis of operator behavior.

Pew developed a task analysis approach for the assessment of cognitive behavior (Ref. 34). A cognitive task analysis data collection system was developed linking inputs, cognitive processes, decisions, and feedback. Operator decisions during an event are to be assessed across time for a series of cognitive elements comprising the task analysis structure. These elements include informational inputs, the event these inputs reflect, the resultant knowledge and/or belief state associated with the event, and the operator's intentions, expectations, and decisions. The approach also considers the procedures or specific knowledge supporting each decision, and the specific feedback necessary to verify or correct the decision. Pew's approach provides a systematic method for cognitive task analysis. A weakness of this approach, however, is the level of detail to which the task in question must be defined in terms of the specific types of information required across the various cognitive elements. Because some of the types of tasks which may be necessary in accident management are difficult to define, such detailed analysis may be difficult to achieve.

Rouse has proposed a pattern recognition model for assessing operator response to a problem situation involving system failure (Ref. 35). The model keys on a series of decisions involving: (a) whether a frame or pattern of symptoms in control room instrumentation is recognized and classified, (b) whether procedures are available or a response must be planned, and (c) whether to execute symptomatic rules/procedures or to branch to other modes of responding. Alternate modes of responding to an event for which symptoms do not fit particular frames and for which procedures are not available or appropriate require the operators to develop a structure-oriented response. These alternate modes attempt to transcend surface symptoms of the problem in order to identify the underlying cause(s). One strength of Rouse's model is its consideration of accident symptoms not fitting preconceived frames even though procedures should attempt to encompass all possible accident conditions. This model suggests that some systematic approach be used to guide exploratory problem solving. Potential methods useful for problem solving in unusual accident conditions were not discussed.

Sage (Ref. 36) has applied the Janis and Mann decision making process model to the systems engineering context. The resulting model provides a standardized structure supporting rational problem solving in systems engineering. The decision maker is held initially to assess the situation by relating needs and problems with potential solutions, then to analyze these solutions as to their acceptability, and lastly to interpret the set of remaining potential solutions in terms of consequences, priorities, and satisfaction of requirements. An implied requirement of the model is that sufficient time is available to complete the analysis before committing to a particular solution. Availability of time is important so that alternatives can be assessed in terms of costs and consequences. However, in the context of accident management, operator actions are likely to be time-driven such that operators make decisions based on best current hypotheses of plant state.

Several insights have been provided by the preceding discussion. First, in a severe accident, operator actions will tend to become more knowledge-based as procedures become less useful. It would seem advantageous to extend existing emergency operating procedures, as would be suggested by Rouse, to guide operator actions in accident management so that operators are less dependent on their troubleshooting abilities. Second, operators will need to consider a range of potential courses of action, as discussed by Sage, in responding to a severe accident for which they have not been specifically trained. Operators should not resort to tunnel vision in diagnosis and response. Third, time will be a critical factor driving operator response in terms of these assessing alternative solutions..

### 3.1.3 Emergency Procedures and Safety Functions

Safety functions are the cornerstone in the design and development of symptom-based emergency procedures. A safety function, as defined in NUREG-0899 (Ref. 3), is "... a function specifically required to keep the plant in a safe condition so that public health and safety will not be endangered." Corcoran *et al.*, in the development of symptom-based procedures for PWRs designed by Combustion Engineering, Inc. (C-E), defined safety functions as "... a group of actions that prevent melting of the reactor core or minimize radiation releases to the general public. They can be used to provide a hierarchy of practical plant protection that an operator should use" (Ref. 37). Safety functions, then, provide a functional basis for the defense-in-depth philosophy which is operationalized through symptom-based emergency procedures. The following discussion reviews a few of the typical approaches to classifying safety functions. These include Corcoran's approach to symptom-based emergency procedures, the emergency procedures developed by the French nuclear power industry for mitigating core damage which represent a unique approach to accident management, and the use of safety functions in a functional classification of plant operations.

Corcoran *et al.,* identified ten safety functions supporting symptom-based emergency procedures for PWRs designed by C-E. These functions were divided into four classes or higher level safety functions which were anti-core-melt, containment integrity, control of indirect radioactive releases, and maintenance of vital auxiliaries. Each safety function was described as having two or more possible success paths for maintaining or restoring the critical parameters associated with the function to safe acceptable levels. A success path was considered to be a method for accomplishing a safety function in terms of relevant systems given their availability and operability at the time when they are needed. The design of multiple success paths supporting each safety function is another aspect of the defense-in-depth concept. Corcoran noted that the safety functions concept is relevant not only to procedures but also to the development of computer-based operator aids, training, human engineering design of control room instrumentation, analysis of operating experience, and evaluation of roles of operators in relation to ERFs. Corcoran raised the possibility that some potential accident situations could exceed the scope of emergency procedures or the plant may not respond as expected. Such difficulties could be handled with a procedure which would provide a set of guidelines or checklists to identify the extent to which safety functions are being challenged and availability/operability of respective success paths. It was recognized that, given potential failure of all success paths for a safety function, operators may need technical guidance supporting restoration of success paths and technical guidance on planning unconventional success paths. The intent of this technical guidance is to provide resource support to operators for all accident eventualities.

The French nuclear power industry has developed a set of PWR emergency procedures for accident management (Ref. 38). Called the U procedures (U stands for ultimate), they represent both a "final" step in preventing fuel damage and mitigating potential radioactive release. To place U procedures in perspective, operators at French NPPs have two types or levels of emergency procedures. The first type is a set of event-based H procedures for loss of different systems. Specific events include loss of heat sink, loss of feedwater, and loss of all electrical supply. The second type consists of the set of U procedures. These procedures are entered by the shift safety engineer or shift technical advisor who monitors critical parameters relative to a predetermined decision logic involving primary water inventory, heat removal, and containment integrity.

The first U procedure, or U1, gives guidance for operator actions depending upon the state of the primary coolant system, steam generator availability, and ECCS availability. Other French U procedures are concerned with the consequences of a molten core. Major emphasis is on maintaining the integrity of containment to prevent or reduce, with reasonable probability, radioactive releases to the environment.

Certain assumptions are made regarding the progression of severe accidents.

1. A steam explosion which would rupture the containment within a few hours after reactor shutdown is very unlikely if not impossible.

2. Hydrogen explosions will not introduce large breaks in containment.

3. Containment isolation failures resulting from leakages at penetrations or failures of isolation systems must be corrected. A separate U2 procedure guides detection and repair of such failures.

4a. A base-mat melt-through is the most probable failure mode of containment, but it is also the one leading to the lowest radioactive release to the environment. A separate U4 procedure guides control of gaseous and volatile fission products.

4b. Gas production from concrete erosion may increase containment pressure and threaten its integrity. A separate U5 procedure guides controlled containment depressurization using coarse filters. (The U3 procedure involves the use of mobile units to support safeguard systems.)

Safety functions have also been used in the analysis of plant operations, in terms of economical, reliable, safe nuclear power production (Ref. 39). An "integrated approach" concept was developed to organize and classify levels of functions necessary for the design, construction, operation and maintenance of an NPP. As part of the general functional classification, four objectives were identified: normal operations, core and plant protection, containment integrity, and emergency preparedness. Under each objective, different plant conditions were identified, e.g., two conditions identified under core and plant protection were core damage controlled and plant damage controlled. The functional classification continued in its delineation for each plant condition, the functions, subfunctions and success paths (systems, components, and human actions) associated with that condition.

Four types of success paths were described: (1) a normal success path that maintains the function in normal operations, (2) a principal success path that maintains the function when challenged by an event, (3) an alternate success path when the principal path is ineffective or unavailable, and (4) an extraordinary success path which is not described in procedures and is an unconventional response by the operator in using components. An analysis of a nuclear plant based on the functional classification has provided some feedback on its usability.

As a general observation, it appears that symptom-based procedures implicitly assume that operator actions as guided by procedures will always successfully ameliorate plant events. It appears necessary that procedures be extended and broadened to handle a greater variety of accident phenomenology. The functional classification associated with the integrated approach to NPP operations provides some guidance in terms of the need to identify and structure additional operator responses in accident management. Consideration of alternate and extraordinary success paths may be important for events involving multiple system failures, as well as events compounded by operator errors.

### 3.1.4. Operator Training for Accident Mitigation

Operator training in accident management and mitigation of core damage appears to be in its infancy. A review of operator training was directed in three areas. First, the Institute of Nuclear Power Operations (INPO) has a set of guidelines for training operators in recognizing and mitigating consequences from core damage. Second, some commercial courses have also been developed on mitigating core damage. Third, a cursory unstructured telephone survey of nuclear industry training organizations was conducted to assess the scope of courses on mitigation of core damage.

INPO guidelines are based on the goal that operators should possess knowledges and skills to recognize potentially hazardous plant conditions and make effective decisions concerning accident mitigation (Ref. 40). Training should be plant specific to reflect design modifications and existing procedures. Personnel to receive training include management, operations, engineering, and support (e.g., health physics technicians) staff. The recommended topic areas and approximate classroom contact hours were:

1. Core Cooling Mechanics (12 hours)
2. Potentially Damaging Operating Conditions (16 hours)
3. Gas/Steam Binding Affecting Core Cooling (10 hours)
4. Recognizing Core Damage (20 hours)
5. Core Recriticality (12 hours)
6. Hydrogen Hazards During Accidents (8 hours)
7. Monitoring Critical Parameters During Accident Conditions (20 hours)
8. Radiation Hazards and Radiation Monitor Response (10 hours)
9. Criteria for Operation and Cooling Mode Selection (20 hours)

Total classroom hours are approximately 128. There are currently no guidelines that address the potential role of control room simulators to practice skills in recognizing accident conditions or assessing operator performance.

A commercial course developed by General Physics Corporation (Ref. 41) is a week-long classroom course for either BWR or PWR plants. The curricula included in this course are representative of topics covered in other commercial courses, as indicated by the telephone survey discussed later in this section. Major topics covered for the BWR class include:

1. Three Mile Island Unit 2 (TMI-2) Incident
2. Core Cooling Mechanics
3. Potentially Damaging Operating Conditions
4. Recognizing Core Damage/Critical Plant Parameters

5. Hydrogen Hazards During Severe Accidents
6. Neutron Monitoring/Core Recriticality
7. Radiation Hazards/Radiation Monitoring
8. Lessons Learned

Two points in this course are particularly notable. First, a method used to track radiation hazards following fuel damage consists of cursory diagrams showing pathways of radioactive releases through the plant. Pathways are constructed for gaseous and liquid streams. Because these diagrams are rather generic for BWRs, they do not provide a lot of detailed data. A strength of this method, however, is that it depicts the depth of barriers that radionuclides must breach in order to be released to the environment. More detailed diagrams may provide technical guidance useful for training in managing radiation hazards during severe accidents. This concept was integrated into the development of the FOAM model as fission product barrier diagrams, and this is further discussed in Section 3.2.5.

Second, the discussion of lessons learned from the TMI-2 incident provides insights for accident management and training operators for mitigation of core damage. These lessons have been extensively reported elsewhere, but some summation of major human factors issues may be noted.

Some of the major lessons learned from TMI-2 relevant to operator training in managing accidents may be grouped into four categories as follows.

1. Alarm Systems
   a. Operators should not discount recurring alarms by assuming that the initiation is always the same.

2. Control Room Displays
   a. Operators should check multiple instrumentation to verify plant conditions rather than concentrate on one display.
   b. Operators should, in reading reactor vessel level from different displays, consider different reference zero points and manner of calibration.
   c. Operators should not discount data that deviates from other data unless certain of the cause of the discrepancy.

3. Control of Systems
   a. Operators should not prevent the automatic response of a safety system unless certain of its imminent/actual failure or certain of unsafe consequences to the plant.

4. Technical Specifications
   a. Operators should use technical data, procedures, etc. as necessary.

An unstructured telephone survey was conducted of nuclear industry training facilities. The sampling of contacts included BWR and PWR training centers, commercial service firms, and nuclear engineering programs at universities. Only some of the training centers provide training programs on mitigation of core damage. Some of this training is conducted as part of requalification training. Formal training involves a combination of classroom instruction and simulator practice. One survey respondent stated that their classroom curricula are similar to the topics covered in the previously discussed commercial course. Another survey respondent from a different A-E firm characterized the simulator scenarios used in some of their training as involving transients which force the operators to "throw the book away" and combat the transients based on plant knowledge and operations expertise.

In general, the INPO guidelines provide some structure for standardizing training for mitigation of core damage. While based on a limited sample size, commercial courses seem to parallel the INPO guidelines. While classroom instruction appears to be a dominant training mode, simulator practice is also necessary to ensure operator skills in event detection and diagnosis. Some systematic approach to developing relevant scenarios followed with appropriate measures of operator performance also seems necessary.

### 3.1.5 Computer-Based Operator Aids

One of the major purposes of computer-based operator aids, such as SPDS, is to support and enhance operator monitoring of plant states and detection and diagnosis of plant events. State-of-the-art technology in the design of operator aids is rapidly advancing. The following discussion is not intended to be a comprehensive review but rather a representative sampling to illustrate the scope of development having the most apparent application to accident management. A fully automated control system has also been researched using an expert systems approach for proposed control of a nuclear research reactor.

Rouse, Kisner, Frey, and Rouse proposed an analytic method for the two-stage process of classification and evaluation of operator aids (Ref. 42). First, a taxonomy of thirteen common decision making tasks were identified for assessing decision support. These tasks were classified according to a taxonomy of operator tasks as follows.

A. Execution and Monitoring

1. Implementation of plan.
2. Observation of consequences.
3. Evaluation of deviations from expectations.
4. Selection between acceptance and rejection.

B. Situation Assessment: Information Seeking

1. Generation/identification of alternative information sources.
2. Evaluation of alternative information sources.
3. Selection among alternative information sources.

C. Situation Assessment: Explanation

   1. Generation of alternative explanations.
   2. Evaluation of alternative explanations.
   3. Selection among alternative explanations.

D. Planning and Commitment

   1. Generation of alternative courses of action.
   2. Evaluation of alternative courses of action.
   3. Selection among alternative courses of action.

The second stage of evaluation seeks to assess the types of tasks for which the aid is intended to provide support. The results of this evaluation contribute to an overall assessment of the aid's "understandability," such as in terms of display (messages) and training (knowledge) requirements. In the evaluation of an aid, three types of event situations were conceived as potentially confronting operators. These were as follows:

1. Familiar and frequent situations — anticipated events for which considerable experience is compiled as operators frequently deal with them. The course of action is typically straightforward.
2. Familiar and infrequent situations — anticipated events for which there is only limited experience due to their infrequent occurrence. Operators may need to collect more information to verify their diagnosis before executing an appropriate response.
3. Unfamiliar and infrequent situations — events which are unanticipated and operators have little or no previous experience with them. The course of action is not at all obvious. The event at Three Mile Island Unit 2 fits this type of situation.

Rouse *et al.*, proposed an integration of decision making tasks with the three types of event situations. First, executing and monitoring decision making tasks were held to be relevant to all three types of situations. Second, situation assessment decision making tasks were held to be relevant to both familiar and infrequent types of situations and unfamiliar and infrequent types of situations, but not to familiar and frequent types of situations. Third, planning and commitment decision making tasks were held to be relevant to only unfamiliar and infrequent tasks.

Additional research has comprehensively addressed a spectrum of issues in the design and evaluation of operator aids. For example, Kisner and Frey assessed the functions of nuclear power plant crews such as in terms of the sequence of emergency operations included in symptom-based procedures (Ref. 43). Frey and Kisner also studied operator acceptance of computer-based aids (Ref. 44). Rouse and Frey have developed guidelines for computer-generated display design and evaluation (Ref. 45,46). Additional work is necessary to assess and validate these design guides.

An automated control system was proposed by Nelson who applied an expert systems approach to a nuclear test reactor (Ref. 47). Response trees were developed explicitly structuring linkages between plant symptoms and operator responses for a range of plant, system, and component states. Decision criteria identify threshold values for critical plant parameters as necessary antecedents for control actions. The manageability of a plant, and of a severe accident, would depend to some degree upon the comprehensiveness of the expert system control model in addressing, among other factors, the spectrum of potential failures, the adequacy and validity of decision criteria, and the effectiveness of designated responses.

In summary, the functions and requirements of computer-based operator aids are being identified and defined based, in part, according to the types of situations with which operators may be potentially confronted. Rouse *et al.*, proposed that planning and commitment decision making tasks were most pertinent to unfamiliar and infrequently performed tasks. Such tasks seem closely related to severe accident scenarios. However, an assessment still needs to be made regarding what types of information are needed by operators under degraded core conditions and whether current operator aids are likely to provide that information. For example, some sensors may fail to function, give incorrect readings, or have insufficient range in certain severe accidents.

### 3.2 The Function Oriented Accident Management (FOAM) Model

The Function Oriented Accident Management (FOAM) model represents both a conceptual structure for systematically describing operator response in accident management and a method for developing and organizing technical guidance supporting operators in mitigating severe accidents. The FOAM model was developed in response to a perceived absence of any systematic guidance for standardizing those factors which may shape operator performance in accident management, such as training and emergency procedures. The model is a functionally oriented approach for translating into an operational context engineering data from severe accident studies such as SASA or the Industry Degraded Core (IDCOR) program, from PRA studies; and from operations, engineering and human factors subject-matter experts. Although this research in accident management is exploratory in nature, the FOAM model appears to be useful in supporting related regulatory, industry and research efforts.

Some of the major assumptions upon which the FOAM model is based include the following items.

1. The severe accident exceeds the scope of emergency procedures such that the plant does not respond as expected.
2. Operators will have to increasingly rely on their knowledge of the plant and its processes, and less so on emergency procedures.
3. Operator actions will be goal-oriented and time-driven.
4. The two goals operators will be most concerned with are mitigating damage to the core and preventing/limiting radioactive releases to the environment.

This section is organized into two parts. First, a brief overview of the FOAM model is presented in order to identify its four components and describe their purposes and interrelationships. Second, each component is sequentially described in more detail.

### 3.2.1 Overview of the FOAM Model

The FOAM model consists of four components. Integration of the relationships among these components is shown in Figure 3.1. The first component is an assessment of the accident sequence with an identification of system and/or operator responses and potential failures and/or errors. The purpose of this assessment is to define the progression of the event from its initiation through the branches in the event tree and on to the end states. Some end states may potentially result in core damage. Depending upon several factors, some single failures/errors may lead to core damage, whereas in other cases multiple failures/errors may be requisite for potential core damage. An Operator Action Event Tree (OAET) is one method for identifying relevant operator responses and possible errors.

The second component uses a functional classification of plant safety functions and control requirements to "translate" the failure(s) or error(s) identified in the first component. The purpose of the translation is to attempt to identify potential alternate or redundant control requirements in the functional classification which would support recovery of the off-normal safety function. Depending upon the particular failures or errors associated with the event, certain safety functions may be off-normal. For the case in which the functional classification does not identify redundant viable control requirements (the respective systems are still available to support the particular control requirement), the operators must plan and execute one or more potential "unconventional emergency responses" (UERs) to either recover the failures or minimize/isolate their effects to plant safety. Potential UERs may be identified using data from SASA and PRA studies and expertise from operations, engineering, and human factors personnel. At a high level, the functional classification also has a branch involving the administration of emergency plans.

The third component concerns modeling the UERs through systematic assessments which result in technical guidance for mitigating severe accident progression. Two methods for modeling UERs are: (1) an event tree format which also permits an assessment of alternate end states, and (2) qualitative assessments for standardizing technical guidance on planning and executing UERs. Assessments of UERs should include: (1) alarms and areas associated with system failure, (2) decision criteria for initiating actions, (3) analysis of operator performance requirements, and (4) consequences from the UERs to plant safety and accident mitigation.

The fourth component involves operator response to fuel damage and potential subsequent radiological release past plant protective barriers. The greatest hazard to the health and safety of plant personnel and the public is the release of fission products. Human factors guidance on mitigating radiological releases is provided using fully detailed "fission product barrier diagrams." These diagrams show potential pathways of fission products along gaseous and liquid streams. Potential breach points along multiple plant barriers are identified and systematically assessed. This assessment includes: (1) a description of how breaches of barriers may occur, (2) control room instrumentation operators may use to detect and diagnose breaches, and (3) potential actions to isolate the breaches.

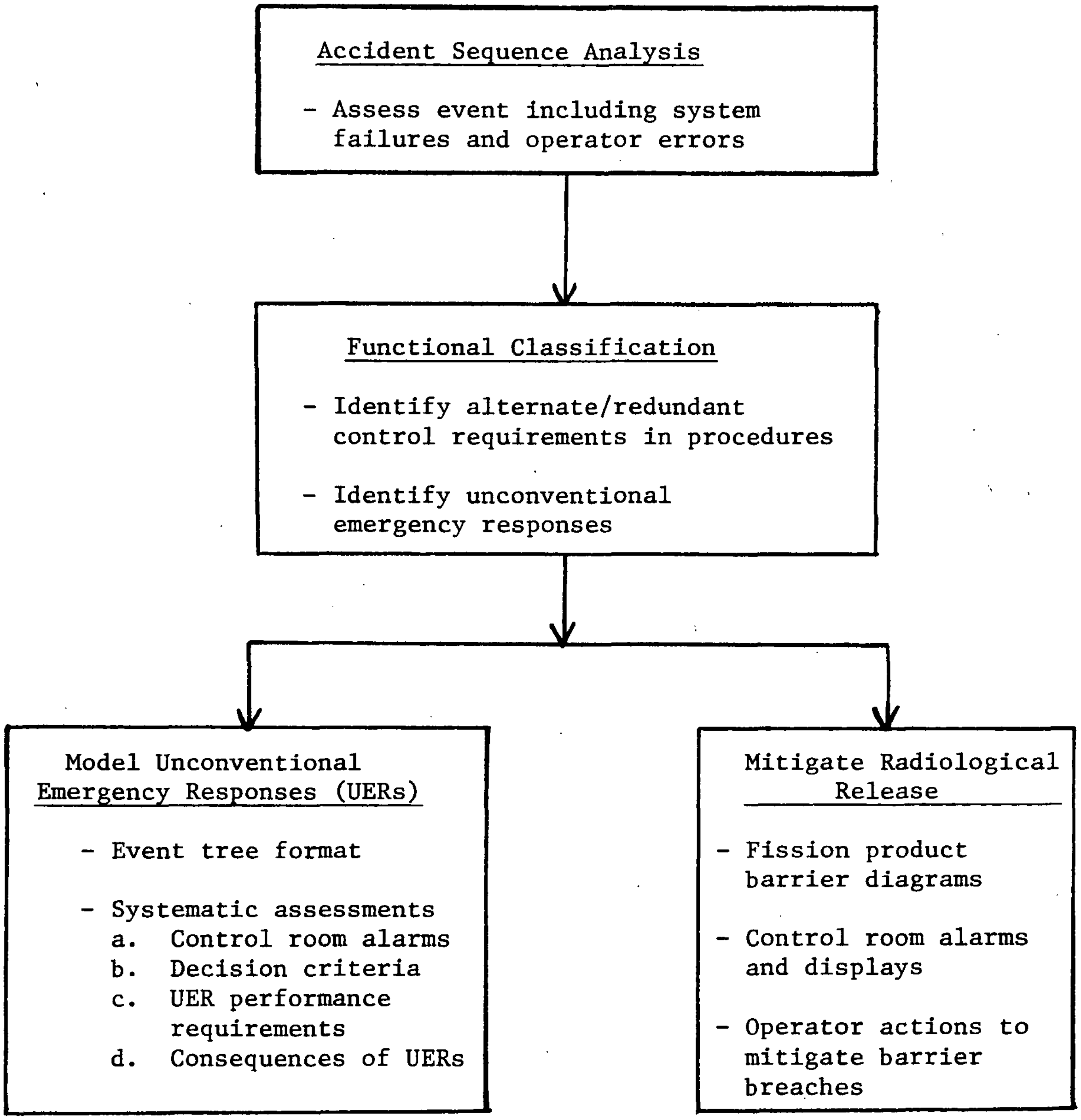


**Figure 3.1 Components of the Function Oriented Accident Management (FOAM) model.**

Each of these components is sequentially discussed in more detail in the following sections. A table-top demonstration is also reported in a later section to illustrate the model's potential use and expected results.

### 3.2.2 Assessment of the Accident Sequence

The purpose in assessing the accident sequence is to describe the progression of the event from its initiation through critical system and/or operator responses to some end state. It is important to identify potential system failures and/or operator errors that may significantly affect the course of the event. Several existing methods provide assessments of the accident sequence, such as a system event tree identifying appropriate systems responses to the event, and the OAET which was previously described in Section 2.2.

The expected result from the assessment of the event sequence is an identification of system failures and/or operator errors which may, as either single or multiple failures/errors, lead to fuel damage or a threat to containment. Operator errors and system failures may be assessed qualitatively or probabilistically, and system failures may also be assessed deterministically. The product from this first component is an identification of system and/or operator responses somewhat generic to a range of events. To some extent, then, the FOAM model takes an event-based perspective to describing accident management. In order to at least conceptualize the goals and time constraints affecting operator response, an initial event orientation seems necessary to assess the scope of the operator's emergency response. Eventually, it may be desired that technical guidance for accident management be system-based to provide the greatest flexibility such that operator response is function-oriented rather than event oriented.

### 3.2.3 Functional Classification

As part of the defense-in-depth design philosophy, a hierarchy of plant protection may be conceptualized to include safety functions, control requirements, and plant systems. Each safety function is supported by two or more control requirements and their associated plant systems. Symptom-based emergency procedures provide operators with guidance on responding to off-normal safety functions through control requirements using available systems. One of the purposes of the functional classification is to translate single or multiple failures/errors in order to identify potential alternate control actions that might restore the off-normal function(s).

Depending upon the nature of the event and the potential system failures and/or operator errors that may occur, one or more safety functions may be off-normal over some period of time. Operator response to system failures or errors may be characterized as follows. On the one hand, a single failure or error may not result in serious consequences if a redundant control requirement can be identified from the functional classification and satisfied through the emergency procedures. On the other hand, the nature of multiple failures or errors may force all control requirements linked with a particular safety function to be unsatisfied. This candidate severe accident scenario would likely exceed the scope of emergency procedures, partly because procedures assume operators will successfully restore the off-normal function. Other candidate accidents scenarios are possible, such as failures/errors affecting more than one safety function over the course of the accident, and these would place a different workload on operator problem solving and planning unconventional actions.

Given the severity and timing of the failures/errors and the difficulties in recovering the off-normal safety function(s), operators must identify, plan, and execute one or more "unconventional emergency responses" (UERs). Sources of UERs include data and recommendations from SASA analyses, results of PRAs, and expertise compiled from operations, engineering, human factors, and other related subject matter experts. UERs may be proposed for the purposes of either recovering one or more system failures or operator errors, minimizing/isolating/circumventing the effects to plant safety, or modifying reactor processes to extend the timing of the event.

At the highest level of the functional classification, the goal of accident management is divided into two first level functions, as shown in Figure 3.2. These high level functions show that the major issues in accident management are protecting plant systems and processes and administering emergency plans involving the protection of plant personnel and the general public. In the following two sections, each of these first level functions is further discussed to identify lower level functions and control requirements. It is noted that the identification and development of UERs are directly tied to operator emergency responses for lower level functions and their respective control requirements under the first level function of Protect Systems and Processes. Lower level functions and control requirements classified under the first level function of Administer Emergency Plans are distinct from UERs. The purpose of this branch in the functional classification is to provide comprehensiveness in scoping the domain of accident management. The functional classification is represented in a fault tree format to identify relationships among functions and control requirements.

#### 3.2.3.1 Protect Systems and Processes

The branch of the functional classification dealing with the protection of systems and processes involves safety functions important to preserving the integrity of plant systems, structures, and operation processes. One of the purposes of this branch of the functional classification is to identify lower level functions and their respective control requirements which are necessary to maintain and preserve reactor operations within engineering specifications. Control requirements are classified according to higher and lower level requirements. The lower level requirements may be further linked to plant systems, but this involves an additional level of detail which was not deemed essential to the higher level of analysis expected of the FOAM model.

The second and third level functions associated with Protect Systems and Processes are shown in Figure 3.2. Control requirements associated with each third level function are shown in cross section in Figure 3.3 (A-H). Control requirements have been divided into higher and lower level requirements in a generic BWR representation.

The scope of procedures may become critical as an event becomes increasingly severe. An event may be exceeding the scope of procedures when the set of operator actions associated with control requirements for an off-normal safety function fail in some manner, such as

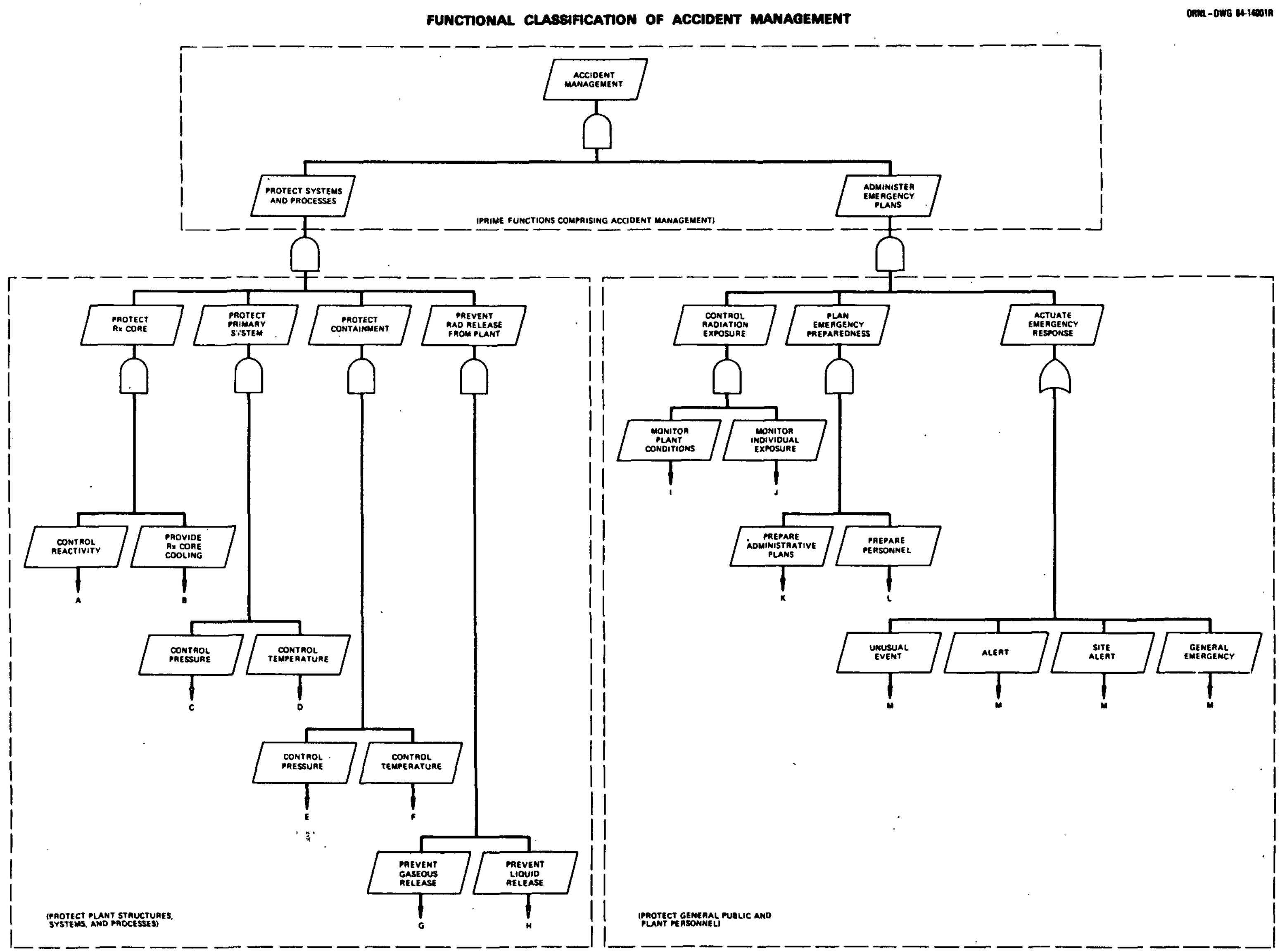


**Figure 3.2. Hierarchical functional classification of accident management.**

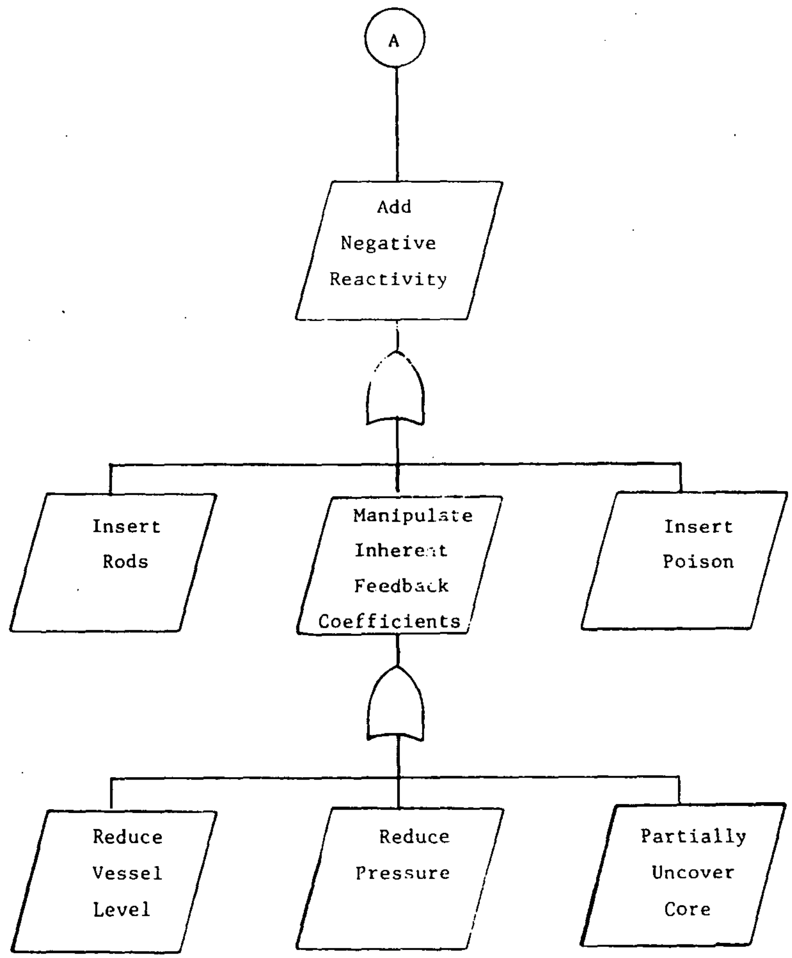


**Figure 3.3(A). Reactivity control requirements.**

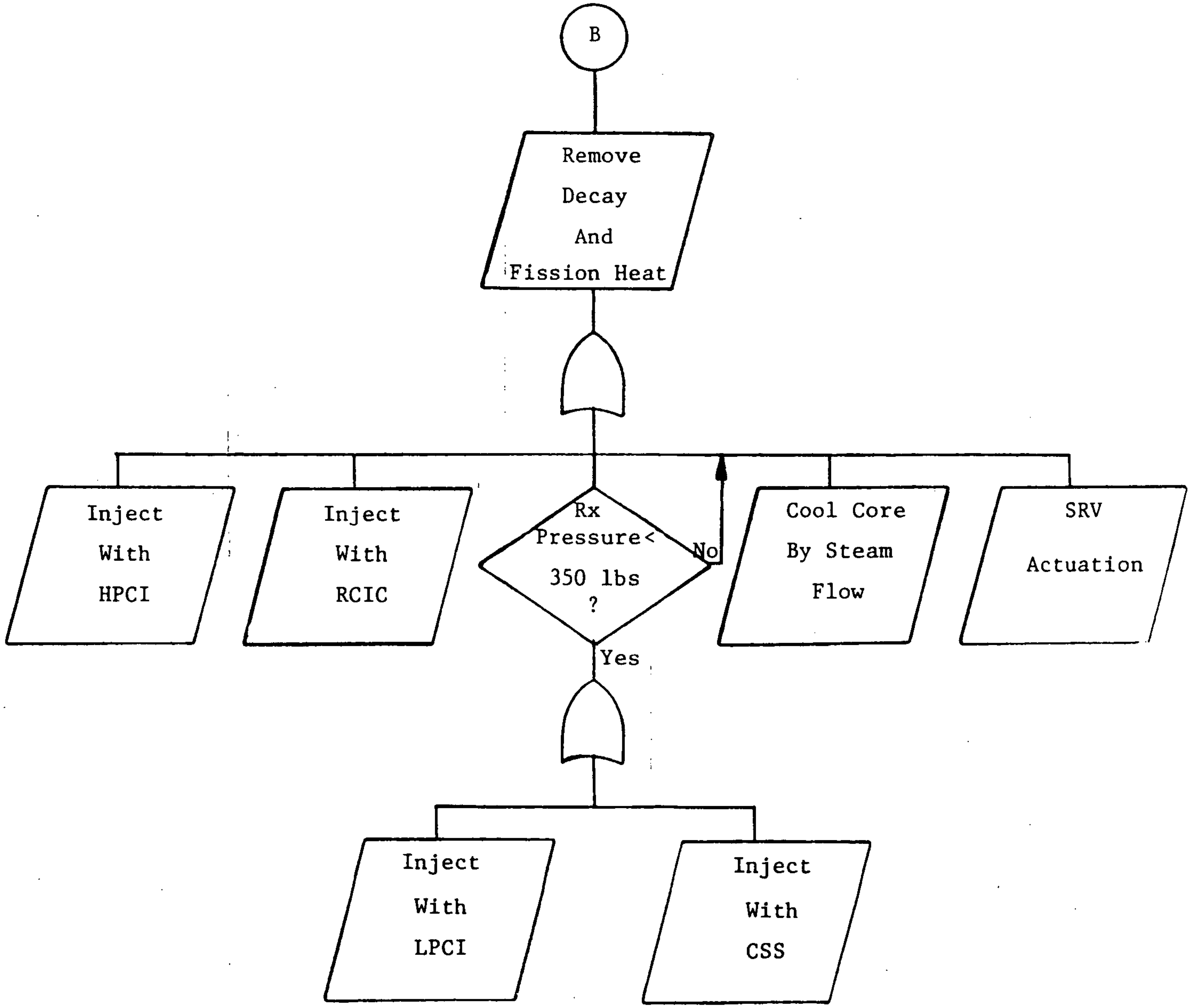


**Figure 3.3(B). Reactor core cooling requirements.**

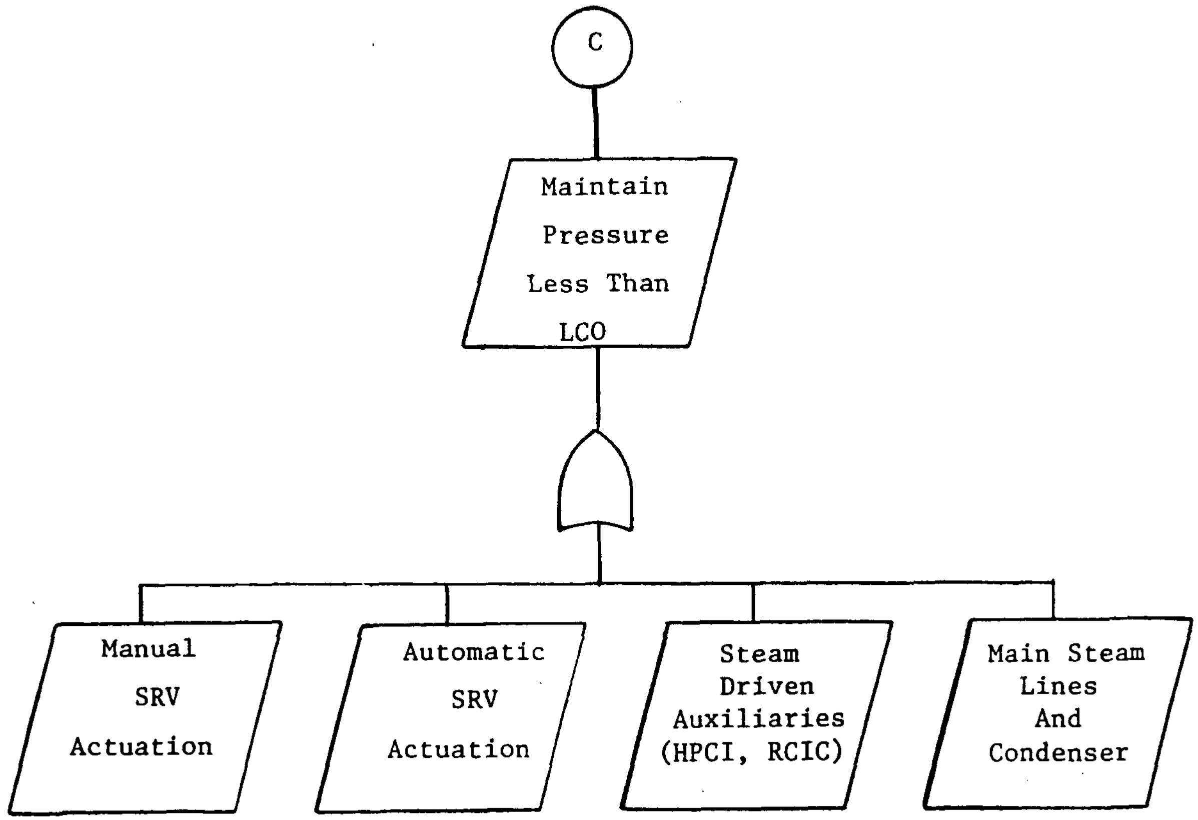


**Figure 3.3(C). Primary system pressure control requirements.**

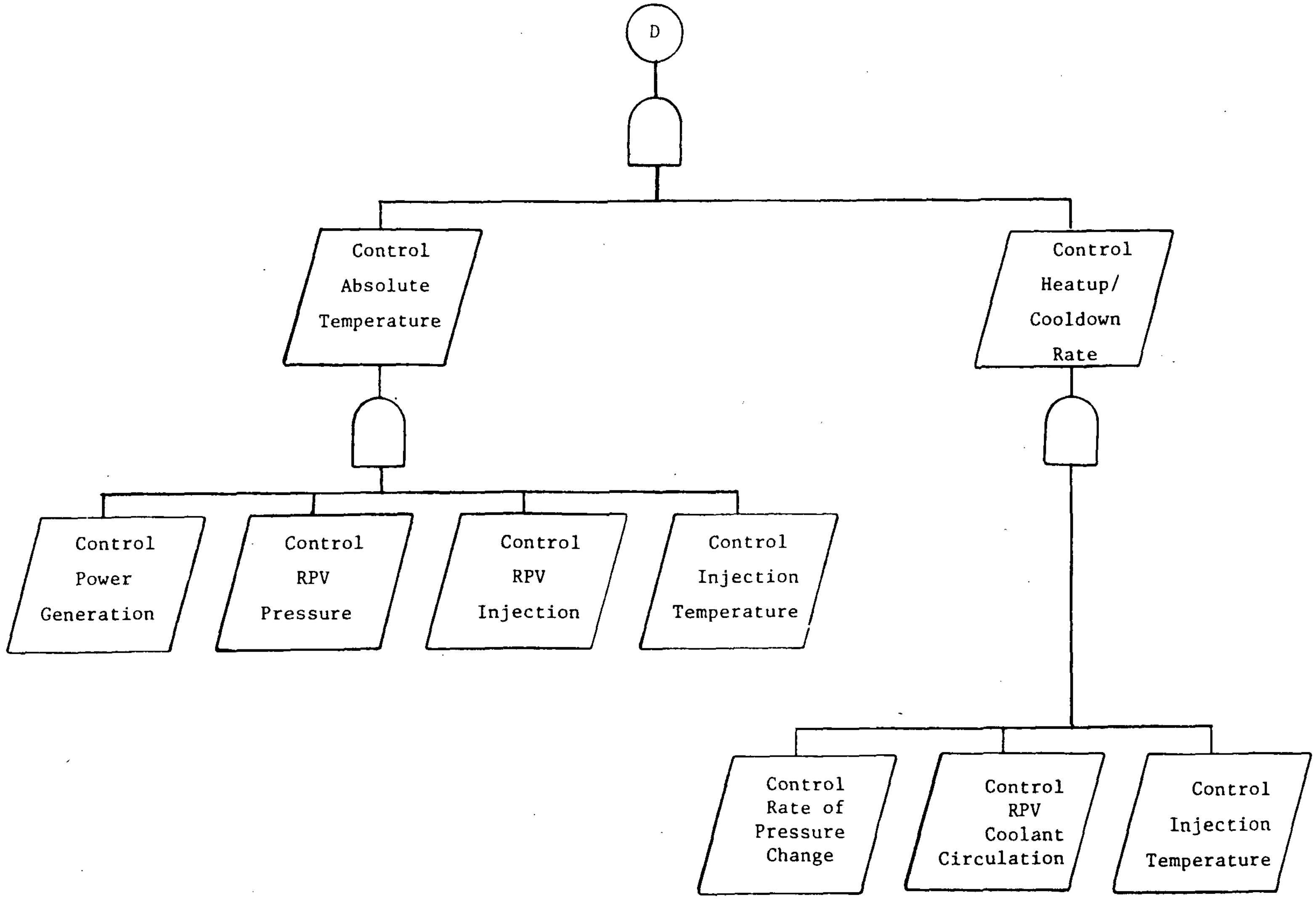


**Figure 3.3(D). Primary system temperature control requirements.**

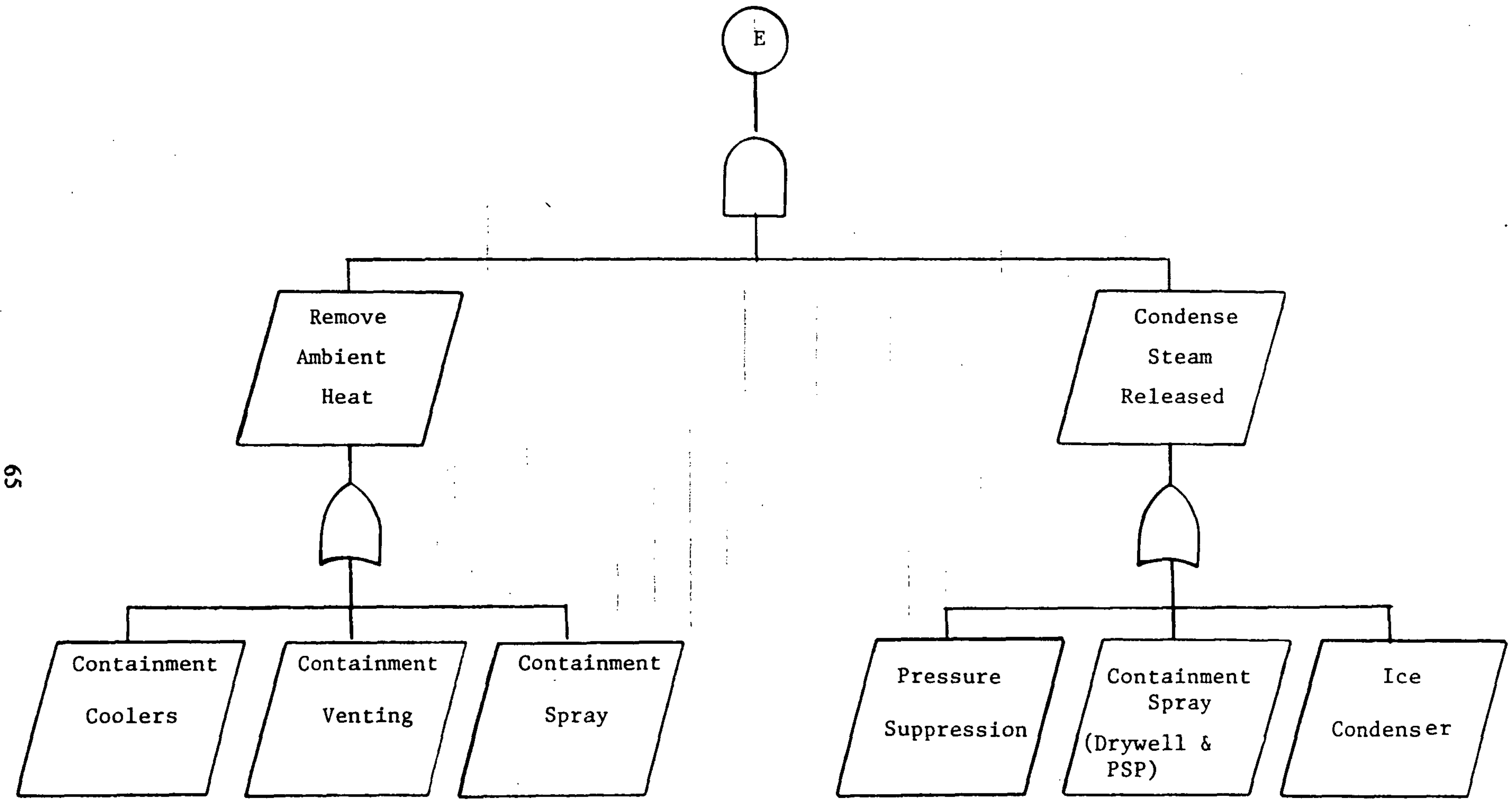


**Figure 3.3(E). Containment temperature control requirements.**

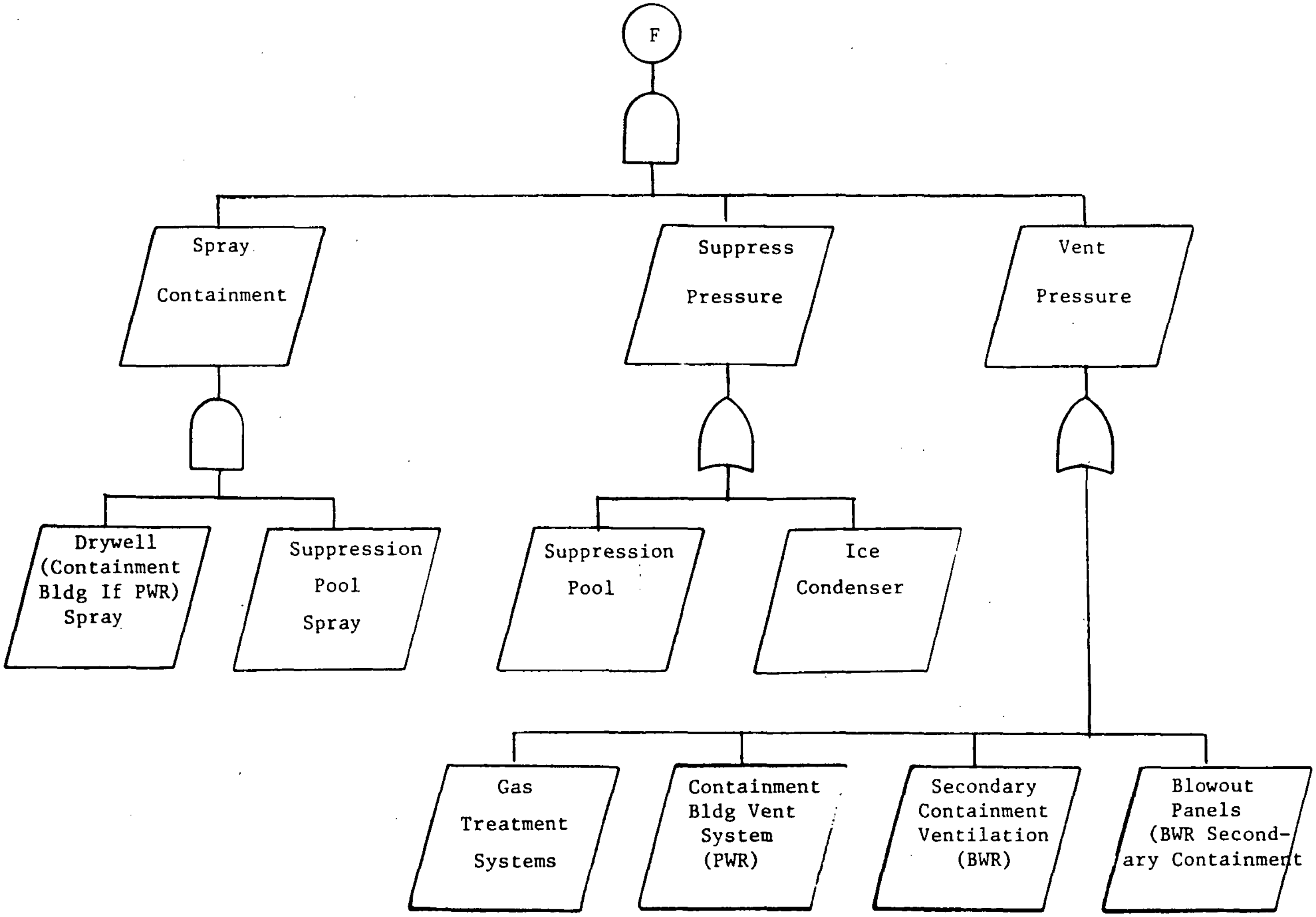


**Figure 3.3(F). Containment pressure control requirements.**

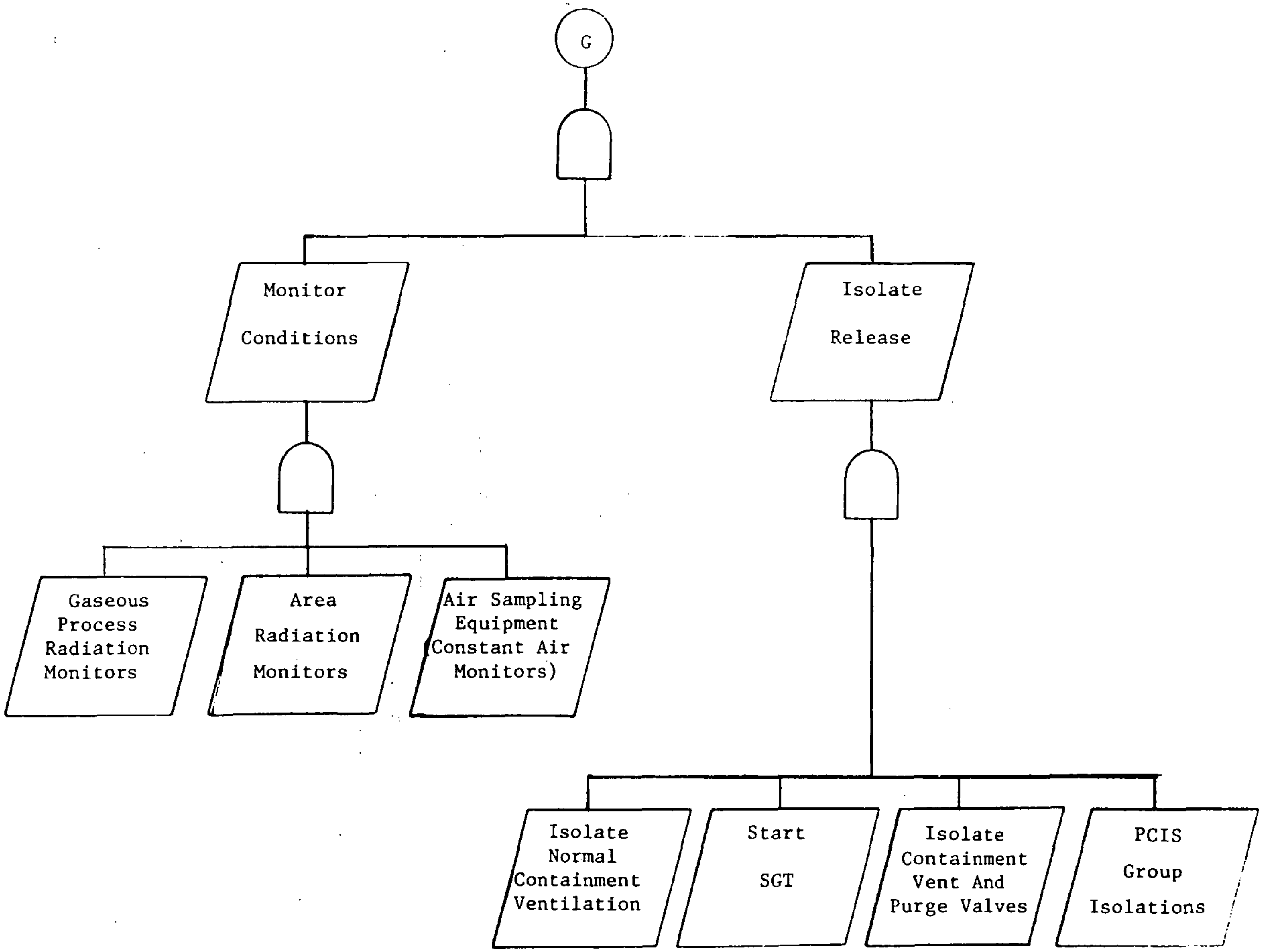


**Figure 3.3(G). Gaseous release control requirements.**

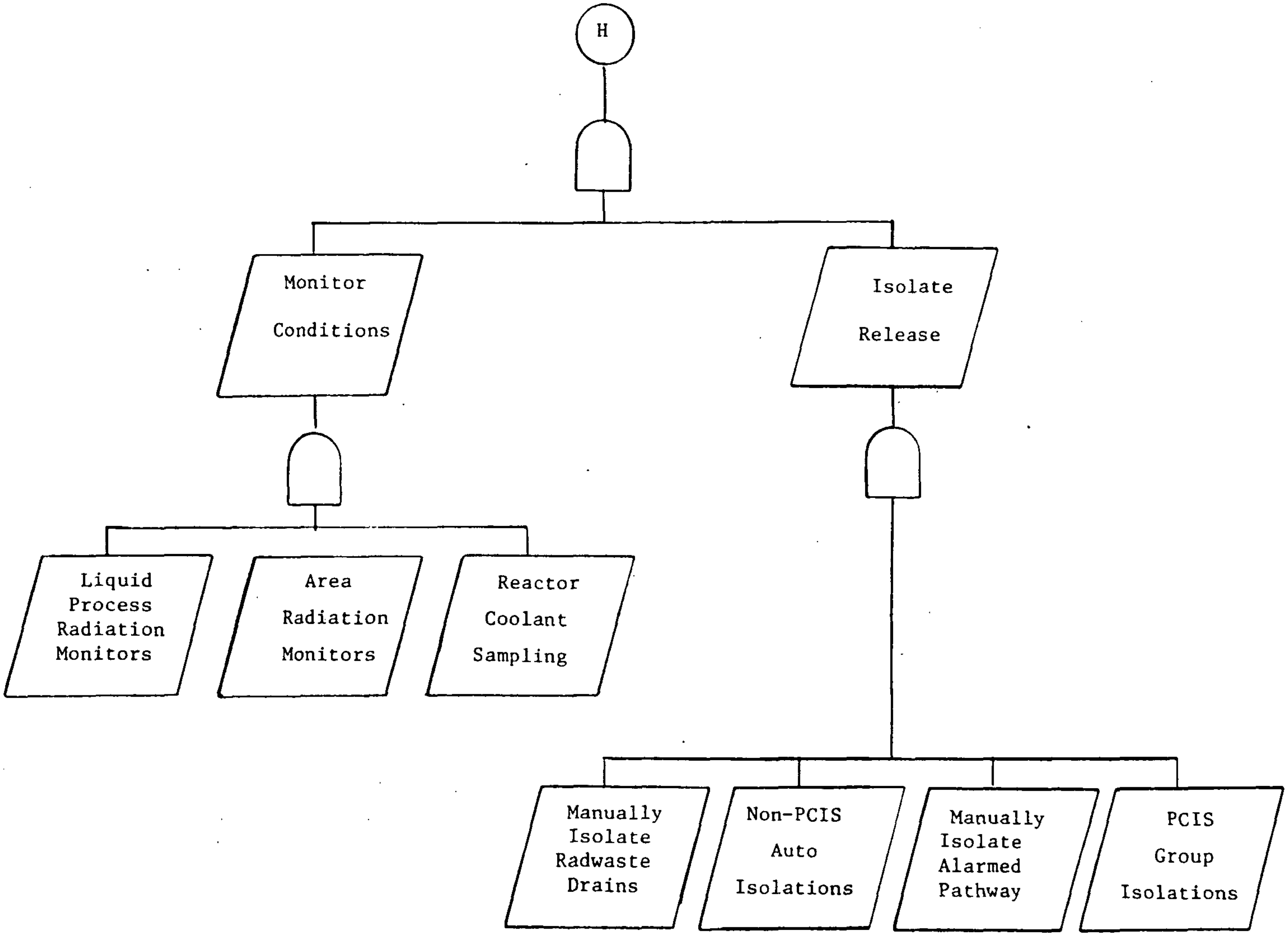


**Figure 3.3(H). Liquid release control requirements.**

due to a system/component failure or to operator errors. Examples of potential operator errors are tripping the automatic operation of an engineered safety feature, or failing to anticipate an automatic response which, for the particular event, results in undesirable consequences to the plant.

Given that operators are confronted with a severe accident that exceeds the scope of emergency procedures, it is likely that their response will evolve through the collective expertise of operators in the control room, and, later, the individuals manning the emergency response facilities. This operator response would seem to correspond with Rasmussen's concept of knowledge-based behavior. Furthermore, these actions are unconventional because of either the absence of procedures to guide them, or the actions are based on an unusual application of an already existing procedure which was developed for another purpose. For accident management, these operator actions may be termed Unconventional Emergency Responses (UERs). UERs are actions which may directly mitigate accident progression or indirectly influence the event by prolonging the timing of accident progression.

The purposes of UERs are, among other factors, to: (1) seek to directly recover failed systems/components, (2) apply systems/components in ways different from their original design objectives, or (3) circumvent automatic system responses considering the consequences to the plant in its current condition. UERs may be identified through use of several different sources. These sources include, but may not be limited to, the following: (1) data, findings and recommendations from severe accident studies such as SASA and IDCOR, as well as from PRA studies, and (2) expertise from operations, engineering, and human factors personnel.

#### 3.2.3.2 Administer Emergency Plans

The branch of the functional classification dealing with the administration of emergency plans identifies functions important to emergency preparedness, accident classification, and protection of plant personnel and the general public from radiation exposure.

The lower level functions supporting the first level function of Administer Emergency Plans were shown in Figure 3.2. The cross sections of control requirements supporting third level functions are shown in Figure 3.3 (I-M). One general comment and one specific comment should be noted. The general comment is that this branch of the classification reflects activities which already are standard practices at NPPs. This branch of the functional classification was developed to systematically organize elements that currently comprise administrative functions and required controls in accident management. The specific comment concerns the control requirement of controlling individual radiation exposure to a level "as low as reasonably achievable" (ALARA). Three factors are recognized as contributing to the ALARA requirement: (1) the individual's distance from the radiation source, (2) the amount of exposure time, and (3) the protection provided by radiation shielding. These factors are controlled continuously in order to minimize individual radiation exposure.

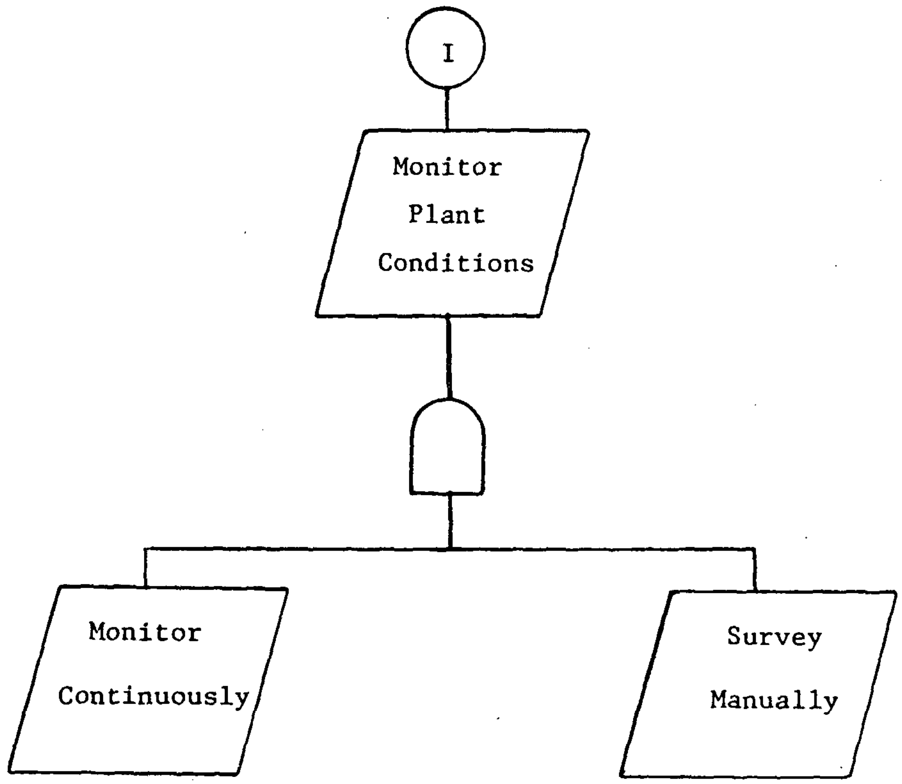


**Figure 3.3(I). Radiation control plant monitoring requirements.**

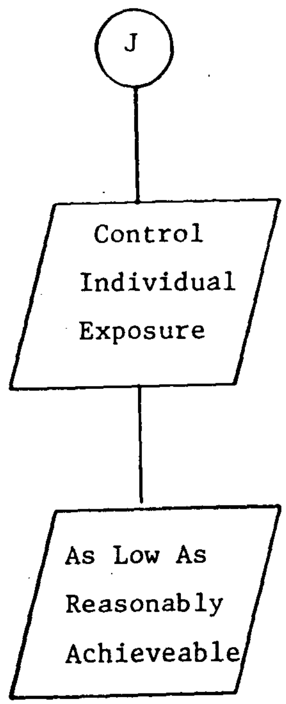


**Figure 3.3(J). Radiation control personnel exposure requirements.**

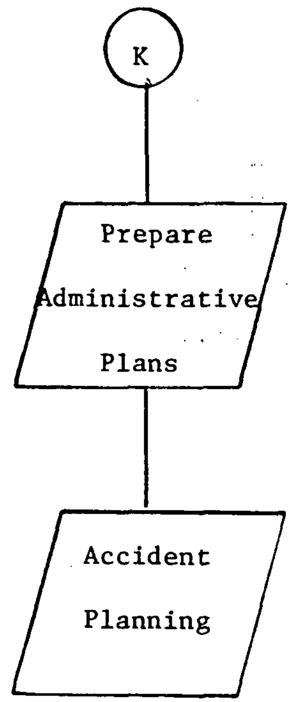


**Figure 3.3(K). Administrative planning control requirements.**

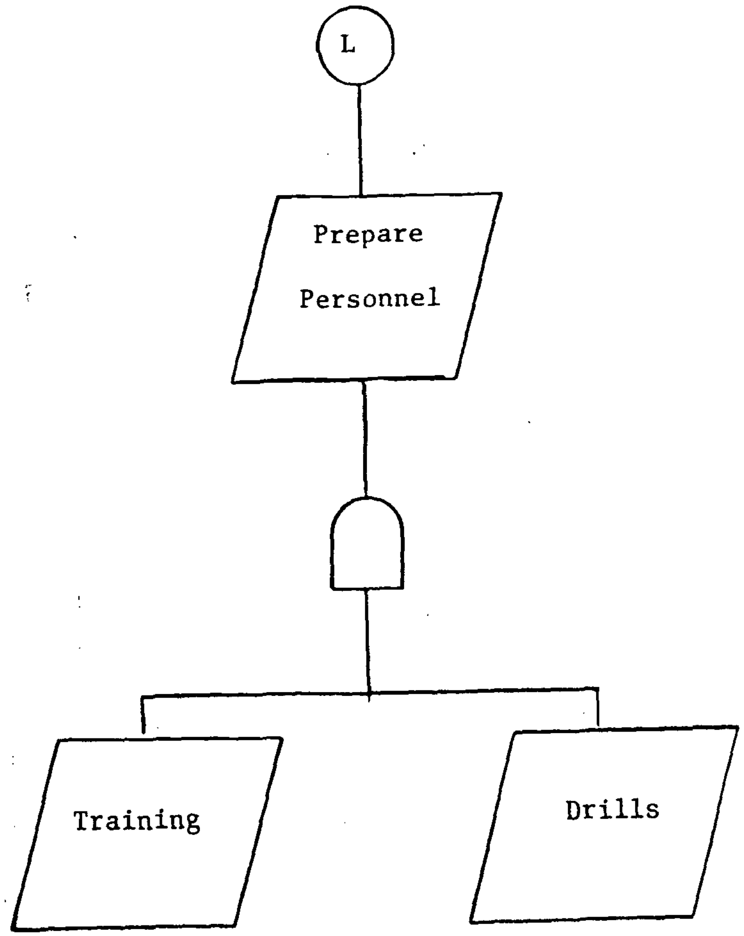


**Figure 3.3(L). Personnel emergency preparedness control requirements.**

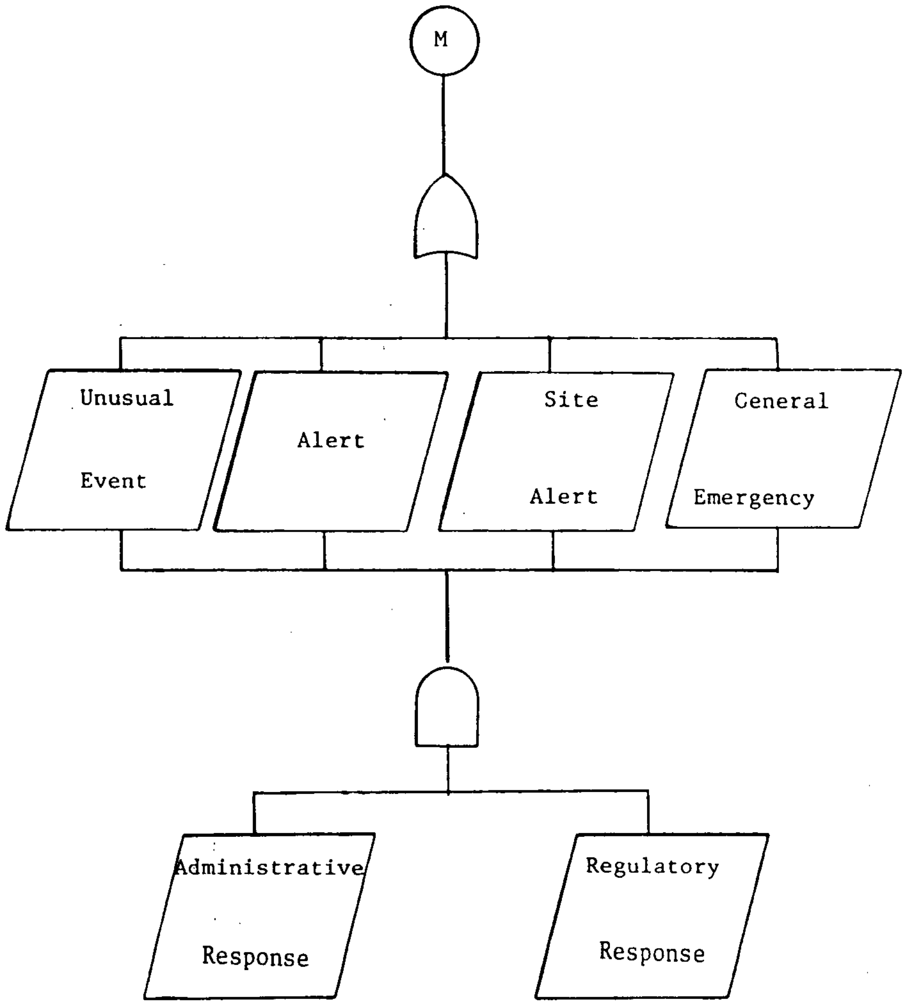


**Figure 3.3(M). Emergency response control requirements.**

### 3.2.4 Modeling Unconventional Emergency Responses

Proposed UERs are potential unusual operator actions for mitigating a severe accident that has exceeded the scope of emergency procedures. The nature of these responses should correspond with the severity of fuel damage, i.e., operators may undertake certain responses given fuel damage short of fuel slump and perform other responses following massive fuel failure with corium melting through the bottom of the reactor vessel. The proposed UERs identified through the translation of accident conditions using the functional classification must be systematically assessed in order to develop technical guidance supporting accident management. The inputs to, methods for, and results from this assessment are described in this section.

The inputs to the assessment of UERs were described in Section 3.2.3.1. A data base of technical information and analyses would be compiled using SASA and PRA reports, as well as other technical literature, such as technical drawings and safety analysis reports. UERs may also be assessed based on expertise of subject-matter experts having relevant backgrounds. Because research data on accident phenomenology are being continually generated, the technical data base would need to be continually updated.

Two methods are suggested to assess UERs as part of the FOAM model. First, proposed UERs are modeled in an event tree format to identify potential branches across a set of UERs relevant to a particular severe accident and the resultant end states identified in the event tree. Second, a systematic and somewhat general qualitative structure for standardizing the assessment of UERs has been developed.

An event tree of proposed UERs identifies potential operator responses to a fairly specific accident perturbation. Different perturbations of the same accident may have different event trees, depending upon the relevance or uniqueness of UERs to any one perturbation. Some particular features of the event tree are that, first, the event tree is entered through a transition from the accident sequence assessment (the first component of the FOAM model) such that the preceding accident conditions, system failures, and operator errors are understood. Second, the UERs should be ordered according to criteria reflecting their priority such as availability of time, consequences to the plant, and so forth. Third, the end states across the set of proposed UERs should be examined to assess possible accident outcomes.

The systematic, qualitative structure for assessing UERs consists of four elements. The description of each UER should include the following elements:

1. Alarms, annunciators and displays in the control room associated with critical parameters reflecting off-normal safety functions.
2. Decision criteria operators should assess and verify in order to implement each proposed UER.
3. An analysis of operator actions including a description of control manipulations of systems/components and the time available for completing the actions.
4. The consequences from the UER in terms of expected changes to critical parameters.

Results from the assessments of UERs are technical guidelines supporting operator response in accident management. The development and validation of technical guidelines supports, among other human factors issues, the extension of emergency procedures for accident management and the identification of training objectives and performance requirements.

### 3.2.5 Fission Product Barrier Diagrams

The potential hazard to the health and safety of plant personnel and the general public is the release of fission products. The identification of fission product pathways and engineered barriers to their release are represented in detailed fission product barrier diagrams. This section presents a general description of the characteristics of these diagrams. Detailed diagrams were prepared for the ATWS accident scenario as part of the FOAM model demonstration, and the diagrams and the accompanying systems description are discussed in a later section.

There are three major characteristics of the fission product barrier diagrams. First, pathways of release are separately shown for gaseous and liquid streams. Second, breaches of protective barriers are identified down to the component level and may include leakage in sumps, pumps, valves, and containment penetrations. Third, the detailed diagrams are supported with a systems description assessing human factors issues related to fission product transport. The systematic assessment includes:

1. A description of fission products formed as a result from fuel damage.
2. A description of how breaches of barriers may occur.
3. A description of control room instrumentation such as radiation monitors, recorders, and annunciators, which operators use to detect barrier breaches and diagnose the severity of radionuclide release.
4. A description of potential operator actions to isolate radiation releases and preserve the integrity of available protective barriers.

This systems description provides additional unique technical guidance in accident management. Technical guidance supports the development of emergency procedures for mitigating fission product transport and release, as well as training operators in diagnosing and responding to breaches of plant protective barriers. This guidance corresponds with some of the French U2 accident management procedure for preserving containment isolation.

## 3.3 FOAM Model Demonstration

In order to show the usability of the FOAM model, a cursory table-top demonstration was developed. It was desired that the demonstration event be an ATWS perturbation which was included in the SASA analysis of the BWR ATWS (Ref. 1) so as to have a reasonable starting point. The selected ATWS perturbation involves multiple system failures of manual control rod insertion and SLC injection. Operator actions contained in the EPGs were factored in by SASA analysts as previously discussed in Section 2. The sequence of events as reported by SASA analysts is shown in Table 3.1. It is important to note that at the time of this analysis the ORNL SASA analysts were in the preliminary stages of their degraded core analysis and had not yet performed calculations for any specific event sequence.

The following four sections step through each of the components of the FOAM model in order to demonstrate its use and potential results. This demonstration goes beyond the planned scope of this initial study. The analysis was based only on data readily available and represents only an initial assessment of potential operator responses in managing a severe ATWS accident.

### 3.3.1 Assessment of the ATWS Sequence

The modified OAET for ATWS was previously shown in Figure 2.3. Based on the sequence of events in the SASA analysis, following the multiple system failures of manual control rod insertion and SLC injection, the EPGs instruct the operator to lower the reactor vessel water level to the top of active fuel, to maintain PSP cooling, and to begin emergency depressurization. Because of the assumed failures, the operators are unable to add negative reactivity into the core by inserting control rods or poison neutron absorber. It is assumed that they would not restore the reactor vessel level because increased flow would add positive reactivity. For this ATWS perturbation, the SASA analysis indicates that vessel injection would be cycling and pressure and power spikes would be occurring about every 13 minutes after some 40 minutes into the event. In the modified OAET, previously shown in Figure 2.3, the most likely end state for this scenario would be #38, although #37, 39 and 40 may also be relevant depending upon operator actions in reactor vessel pressure control and level control. The complexities and consequences of those instructions in the EPGs involving pressure and level control were previously described in Section 2.

### 3.3.2 Translation of Failures in ATWS Using the Functional Classification

In the functional classification, the third level safety function of control reactivity may be satisfied by three different first level control requirements. The control requirements involving insertion of control rods and injection of poison are unsatisfied in this demonstration ATWS perturbation. The control requirement of manipulating inherent feedback coefficients so as to reduce reactor power is satisfied, in accordance with the EPGs, by lowering the reactor vessel water level and emergency depressurization. However, only insertion of control rods or poison injection results in complete shutdown of the reactor. Manipulating inherent feedback coefficients provides only an interim reduction in steam generation and does not shut the reactor down, but this action is important in prolonging the timing of accident progression before core degradation.

The translation of these particular failures, given that the functional classification identifies no other redundant control requirements that would shut the reactor down, suggests that two UERs may be appropriate. These UERs are: (1) recover control rod insertion, and (2) recover SLCS injection. Additional translation comes from the SASA finding that eventual drywell failure results from an increase in PSP temperature. In

**Table 3.1. Sequence of events reported by SASA analysts for case without manual rod insertion of SLC injection, but with pool cooling.**

| Time (min) | Event | Comment |
|---|---|---|
| 0 | MSIVs begin to close | Anticipated transient |
| 0.1 | No reactor scram | |
| 0.1 | Recirculation pumps trip | |
| 1.5 | HPCI and RCIC start | Automatic actuation, total injections 5600 gpm (353 l/s) |
| 2 | Operator control of vessel pressure begins | To prevent SRV cycling on automatic actuation |
| 7 | Operator trips HPCI and RCIC | Per EPG level/power control guideline |
| 8 | Core spray and RHR pumps start | At vessel water level <413.5 in. (12.5 m) |
| 8.4 | Vessel water level below TAF | Operator restarts HPCI at 1800 gpm (113 l/s) |
| 8.5 | Reactor power below 10% | |
| 9 | Vessel pressure dropping | Operator shuts all but one SRV |
| 10 | Operators initiate suppression pool cooling with all four coolers | "Containment Spray Select" switch actuated |
| 14.8 | Vessel water level above TAF | Not back on scale of emergency systems indication |
| 16.8 | Power spike | Core thermal power to 35% |
| 16.8 | Automatic SRV actuations | |
| 17 | Operators decrease HPCI flow | Vessel water level too high |
| 18.7 | Operators begin emergency depressurization of reactor vessel | Suppression pool in violation of EPG heat capacity temperature limit |
| 18.7 | Operators trip HPCI and RCIC turbines and the core spray condensate, condensate booster, and RHR pumps | Interrupts suppression pool cooling |
| 19.5 | Drywell pressure exceeds 2.45 psig (118 kPa) | |
| 19.6 | Core completely uncovered | Subcritical and producing only decay heat |
| 20.1 | Vessel pressure below 450 psig (3.21 MPa) | Core spray and LPCI valves open (LPCI valves interlocked open for 5 min) |
| 20.6 | Operators resume vessel injection | Using condensate booster pumps, flow controlled by startup bypass valve |
| 27 | Operators restart suppression pool cooling | After overriding 2/3 core coverage interlock |
| 27.8 | All SRVs shut | Vessel-to-drywell pressure difference <20 psi |
| 31.8 | Vessel water level recovered to >TAF | Level not back on scale of emergency systems indication |
| 33.3 | Operators discontinue injection flow | Emergency systems indication on scale but increasing too fast |
| 33.8 | SRVs reopen | Vessel-to-drywell pressure difference >50 psi |
| 34.6 | Vessel power and pressure spike | Maximum core thermal power = 81% |
| 34.8 | Automatic SRV actuations | At 1105 psig (7.72 MPa) |
| 36.5 | Vessel pressure below 450 psig (3.1 MPa) | Depressurizing with five open SRVs |
| 40—end | Additional power/pressure spikes | Occurring about every 13 min |
| 120 | Suppression pool temperature at 232 F (384 K) | Still increasing |
| 720 | Suppression pool temperature at 345 F (447 K) | Drywell overpressure failure imminent |

order to provide additional time for recovery of systems failures, two UERs are proposed to slow the rate of PSP temperature increase. These additional UERs are: (3) initiate PSP spray, and (4) replenish PSP volume. Another translation that addresses the effort to slow the rate of PSP temperature increase is to use the main condenser as a heat sink, given that no fission products are detected by radiation sensors or by testing a coolant sample. This extra UER is: (5) open one MSIV. An open MSIV would handle all steam generated by the reactor with reactor power less than 25% and would permit utilization of the feedwater system for coolant injection.

Additional UERs could be considered given the absence of success in recovering the ability to insert rods or inject poison. Operators are confronted with the following three problems which they must successfully address to prevent core uncovery and subsequent melting of the fuel: (1) injecting water to cool the fuel, or corium if fuel melt has occurred, (2) minimizing containment pressure which threatens integrity of the containment, and (3) minimizing containment temperature. Several UERs could be proposed to attempt to mitigate each of these problems. For example, containment flooding following reactor vessel failure may be a likely response. However, it was beyond the scope of this limited demonstration to assess such additional specific UERs.

### 3.3.3 Modeling the UERs for the ATWS Perturbation

Inputs to the assessments of the five UERs identified in the previous section were expertise from two of the authors having reactor operations experience, BF1 procedures and technical data, and previous SASA reports on accident sequences at BF1. A more thorough analysis should result from the work of SASA analysts on core degradation scenarios.

The event tree of proposed UERs is shown in Figure 3.4. Recovery of failures involving control rod insertion and poison injection are ordered first because of their importance to shutting the reactor down. Initiating PSP spray and replenishing PSP volume are listed next to prolong the amount of time, if needed, to recover the above failures. The UER of opening one MSIV is ordered last because it was judged to be, of the five UERs, the least desirable due to the concern over fission products potentially entering the reactor coolant and being directly transported outside the pimary containment to the main condenser.

The assessments of the five UERs systematically covers four elements. These elements were: (1) alarms and cues associated with failure conditions, (2) decision criteria regarding initiation of operator actions comprising the UER, (3) a description of expected performance, and (4) consequences from operator actions to the plant. Detailed assessments of the five UERs in accordance with these elements are presented in Appendix C.

The end states shown in Figure 3.4 have been classified according to expected possible plant conditions. A description of these end state conditions is shown in Table 3.2 which includes the qualitative classification logic. Accident conditions associated with several of these end states necessitate identification and assessment of additional UERs. Accident phenomenology involving a molten core and drywell overpressure and overtemperature may continue to challenge operators.

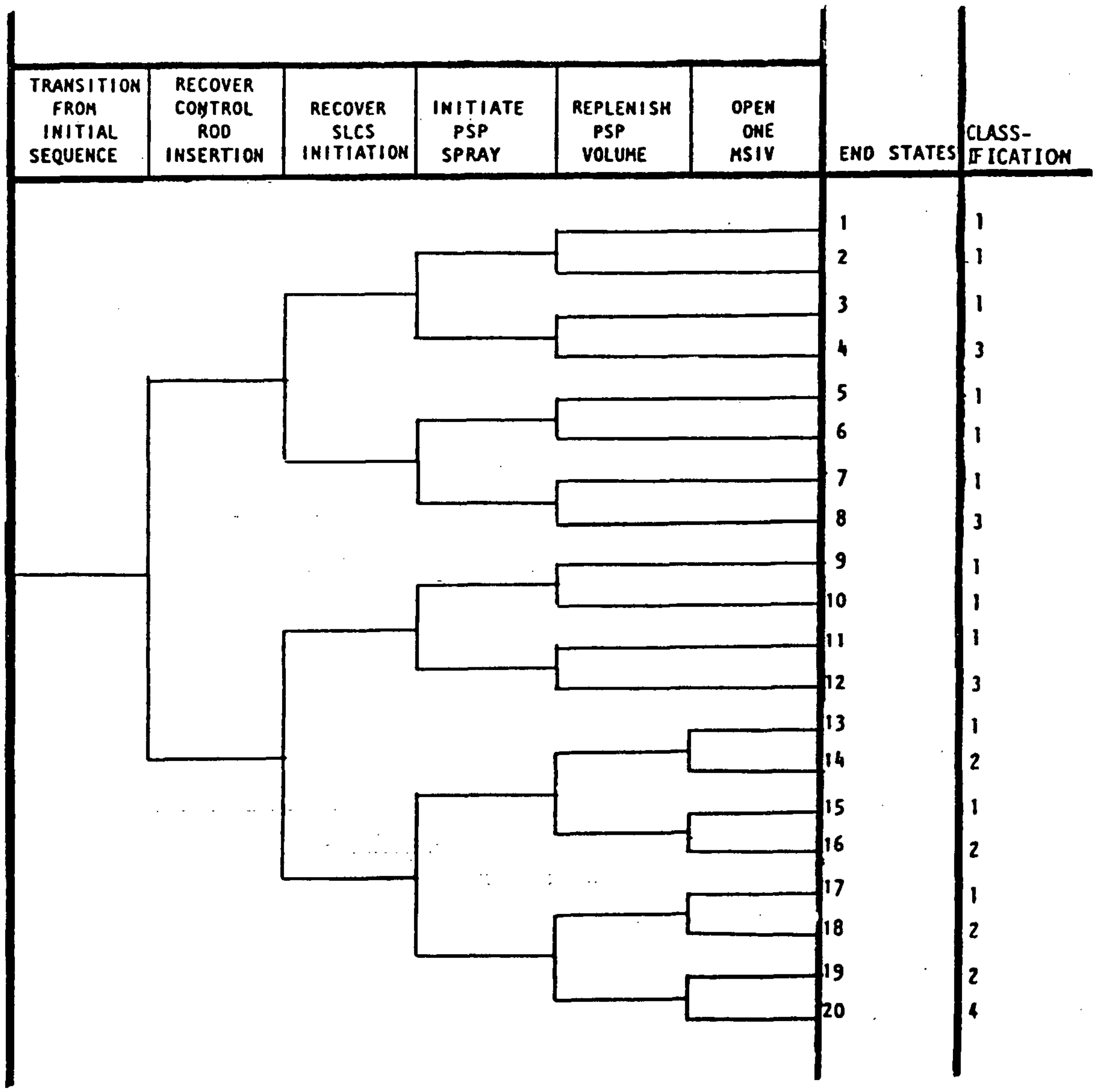


**Figure 3.4. Unconventional emergency response event tree for ATWS following failures of manual control rod insertion and SLCS initiation.**

**Table 3.2. Classification of end states for proposed UERs to an ATWS**

| Condition Number | Possible Plant Condition | Classification Logic |
|---|---|---|
| 1 | No fuel or containment failure | Requires either or both tasks associated with reactivity control to be successful, and at least one task for containment control to be successful including opening one MSIV since the fuel and the containment can withstand a higher power level. |
| 2 | Fuel failure, no containment failure | Requires failure of both tasks associated with reactivity control, but success with at least one containment control task. Necessitates identification and assessment of additional UERs for dealing with potential molten core. |
| 3 | Containment failure, no fuel failure | Requires success with one or both reactivity control tasks, but failure of all containment control tasks. May necessitate identification and assessment of additional UERs for dealing with decay heat removal given the deteriorating state of the PSPAs a heat sink. |
| 4 | Both fuel and containment failure | Requires failures with reactivity control tasks and failures with opening one MSIV and one other of the containment control tasks. Necessitates identification and assessment of additional UERs for dealing with catastrophic conditions. |

### 3.3.4 Fission Product Barrier Diagrams for BF1 ATWS

Detailed fission product barrier diagrams were prepared showing pathways of radionuclide release along gaseous and liquid streams. Breaches of protective barriers were identified down to the component level. These diagrams were developed for the BF1 ATWS as a case example using the best plant data available and are intended to be a comprehensive identification of potential pathways and barrier breaches. The diagrams and the supporting systems description are included in Appendix D. The diagrams may be applicable to other severe accidents at BF1 and other BWRs. A detailed evaluation would be necessary to validate and verify the comprehensiveness of the diagrams for these applications.

## 3.4 Potential Applications of the FOAM Model

Potential applications of the FOAM model may be identified pertinent to regulatory, industry, and research perspectives. In considering these potential applications, it is noted that the FOAM model was developed as part of an exploratory assessment of human factors issues in accident management. The FOAM model is both a conceptual approach for describing the scope of accident management and a method for standardizing technical guidance based on best available data and expertise.

From the regulatory perspective, the FOAM model has at least two major potential applications. First, the model provides a method for standardizing the development of technical guidance supporting accident management. In the development of emergency operating procedures, NUREG-0899 (Ref. 3) identified the development of technical guidelines as a prerequisite to the writing of procedures, but did not address the format of these guidelines. As a result, there has been no standardization in the structures of symptom-based procedures developed by the different Owners Groups. An advantage of standardized technical guidance unique to the accident management area is the increased possibility of transfer of data and expertise across different NPPs. Of course, such guidance would have to be adapted to specific reactor features. Second, the fission product barrier diagrams represent a method for identifying and assessing human factors issues in the control of radiological release from the plant. The French U procedures, discussed in Section 3.1, provide some perspective on regulatory guidelines for monitoring and controlling radiological release. The FOAM model provides an approach for developing regulatory guidelines supporting human factors issues in fission product transport.

From the industry perspective, the FOAM model provides a means for developing technical guidelines supporting operator response in accident management. Operator response to severe accidents would be shaped by such factors as procedures, training, computer-based operator aids, control room staffing, and emergency facilities. Technical guidance developed through the FOAM model would support all of these considerations. Technical guidelines would support extending emergency procedures by identifying and systematically assessing potential site-specific UERs. Guidelines would support operator training by assisting in the identification of training objectives, the assessment of performance requirements, and task analysis. Technical guidelines would support identification of design requirements and evaluation criteria associated with computer-based operator aids. Guidelines may provide implications for control room staffing requirements, such as manning requirements to accomplish UERs along parallel paths. Technical guidelines would support operation of emergency facilities, such as administration of emergency plans pertinent to planning and execution of UERs.

From the research perspective, at least two potential applications of the FOAM model may be identified. First, the model provides a method by which SASA and PRA results can be compiled as a data base supporting operator response in accident management. As UERs are developed in response to problems analyzed under different studies, they may be potentially classified and catalogued using the functional classification structure. Second, the FOAM model is a proposed exploratory human factors method for developing technical guidelines supporting operator response in accident management, and the model itself needs additional development. Other methods may also be appropriate, and researchers should investigate their strengths and weaknesses, such as in a trade-off analysis, prior to making a commitment toward one method over the others.

Preliminary indications from SASA analysts suggest that application of the FOAM model may be useful for identifying a large number of operator actions for mitigating a variety of severe accidents. As with the FOAM model demonstration, the analysis of modeling UERs identified a range of potential end states. For example, the end states that result in fuel failure, containment failure, or both would require further development of additional UERs involving subsequent actions the operator may take in response to the accident. End states 4, 8, 12, 14, 16, 18, 19, and 20 in Figure 3.4 were described as requiring development of additional UERs. The unconventional responses identified in these UERs may include such actions as poison injection using alternate systems, containment and vessel flooding using RHRSW, and containment venting through the SGT system.

## 4. CONCLUSIONS AND RECOMMENDATIONS

Major findings, needs, and directions for further human factors research have been identified over the course of this project. Conclusions are presented from the work with SASA analysts on the BWR ATWS, followed by the findings from the development of the high-level descriptive FOAM model. Recommendations are suggested regarding continued human factors support of SASA investigations and further research in accident management based on an assessment of problems and needs.

### 4.1 Conclusions from the Human Factors Assessment of the BWR ATWS

The purpose of the human factors assessment of the BWR ATWS was to support SASA analysts by qualitatively and quantitatively analyzing operator actions, and thereby demonstrate the utility of human factors methods and data to accident sequence analysis. The human factors assessment of operator actions was largely motivated by the questions and concerns of SASA analysts about operator performance. However, the assessment was constrained by multiple project objectives, so that the human factors assessment was consolidated to only six operator actions judged to be most critical to ATWS.

The approach to and results from the human factors analysis were generally helpful to SASA analysts. Conducting and videotaping BF1 control room simulator exercises provided data having several uses, such as some benchmarking of the SASA computer code, the task analysis supporting the HRA, and identification and qualitative assessment of potential operator errors. Preliminary results from the human factors assessment were included as Appendix C to the SASA ATWS report (Ref. 1)

Several findings were obtained concerning operator performance during ATWS. In general, the recommendation of SASA analysts that emergency procedures pertinent to ATWS be separated from the EPGs may be reasonable considering the complexity of required actions. However, revamping the EPGs for one event sets a precedent which contradicts the intent of symptom-based procedures. The question arises as to what other "special cases" or exemptions to the symptom-based EPGs might arise from further detailed analysis. An important point is that the symptom-based procedures, and the training that operators receive on their use, should emphasize the distinct symptoms of an ATWS and the special actions required. One should also ensure that these symptoms are well annunciated.

In the qualitative analysis, potential problems were identified with certain instructions in the EPGs for operator actions, and with human engineering deficiencies in control room design. Of the six operator actions systematically assessed in the qualitative review, two actions unique to the EPGs seemed especially problematic. First, the instruction to lower reactor vessel water level to the top of the fuel is constrained by control room instrumentation deficiencies and the potential loss of HPCI. Reactor vessel level indications may be inaccurate, have insufficient range, and be located on different console panels. The potential loss of HPCI increases the difficulty in completing this task. Because lowering the water level to the top of the fuel is a new response to ATWS developed by the BWR Owners Group, operators must be trained on proper use of coolant injection systems and interpretation of vessel level indications. Skills and applications of relevant knowledges should be demonstrated on the control room simulator to ensure proficiency. Second, the instruction regarding emergency depressurization may result in power and pressure excursions that complicate efforts at keeping the core covered with water. SASA analysts recommend that operators not attempt manual control of reactor vessel pressure under ATWS conditions, concluding that it is "extremely risky" to depressurize a BWR during an ATWS. SASA, TVA, and other members of the BWR Owners Group are considering strategies for completing this instruction, and it appears that the jury is still out. However, given the complexities and consequences of this task, operator training and simulator practice appear to be absolutely necessary for developing and ensuring performance proficiency.

In the quantitative analysis, two methods in human reliability analysis (HRA) were used. The THERP analysis of four critical tasks, while of limited direct use by SASA analysts, provided probabilistic information reflecting complexities of operator actions. Several problems may be noted with this analysis. The task analysis data developed using the NRC task analysis procedures were at a different and finer level of detail than required for input to the THERP analysis. Task HEPs resulting from the THERP analysis tend to be high. Consideration of recovery factors may have reduced these failure probabilities. Furthermore, considering that these particular tasks were selected for their criticality to ATWS mitigation, as well as the amount of time available to operators to respond as evidenced in the simulator exercises, it may be argued that operator reliability on these tasks would have a success probability approaching 1.00. The OPPS analysis provided a time-reliability curve having some comparison with SASA results. SASA analysts, in their no-operator-actions scenario, reported that containment failure would occur at 36.8 minutes into the ATWS. The average time in the OPPS analysis for completion of all operator actions was 33.4 minutes. This suggests that operators should have sufficient time to complete most required actions, and certainly those actions most critical to mitigating the event.

## 4.2 Conclusions from Research in Accident Management

The purpose of the research in accident management was to develop a high-level descriptive model for assessing operator response in accident mitigation. Accident management was found to be a multidimensional problem having somewhat amorphous boundaries. Assuming that operator response is goal-oriented and time-driven, a four component Function Oriented Accident Management (FOAM) model was developed. The intent of the FOAM model is to provide both a conceptual structure for scoping major issues in accident management and a method for identifying and organizing technical guidance supporting the operator's emergency responses. Technical guidance is a prerequisite to the extension of emergency procedures, the specification of training objectives and performance requirements, and other factors comprising good accident management practices. Technical guidance is developed by compiling findings and recommendations from SASA and PRA studies, as well as documenting the expertise from engineering, operations, and human factors subject matter experts.

The FOAM model is one method for the development of technical guidelines specifically suited to accident management. It is responsive to different types of events and supports a general identification of unconventional emergency responses (UERs) using best available data and knowledge of accident phenomenology. The model also provides directions for development of UERs in response to severely degraded cores wherein molten corium threatens reactor vessel integrity and containment integrity.

Detailed fission product barrier diagrams represent an initial assessment of human factors issues paralleling fission product transport engineering analyses. The diagrams and their accompanying qualitative systems descriptions represent technical guidance addressing, among other factors, the adequacy of plant wide radiation instrumentation, operator actions appropriate to isolating radiation releases, and information needs that may be potentially supported by computer-based operator aids. The importance of controlling fission product transport in accident management somewhat exceeds the scope of mitigating core damage because of the necessity of protecting plant personnel and the general public from radiation exposure even after the reactor is shut down.

The utility of the FOAM model for developing technical guidance in accident management was demonstrated in a table-top assessment of a severe ATWS perturbation. Given assumed systems failures and threats to containment integrity, certain data from SASA and PRA studies combined with operations expertise were compiled resulting in a set of five potential UERs. The assessment of these UERs identified cues, decision criteria, performance requirements, and consequences to the plant. Detailed engineering, operations, and human factors analyses seem necessary to ensure the acceptability of these potential UERs due to the seriousness of some consequences. Other UERs might well be proposed to deal with more advanced stages of core degradation and movement of molten corium through the reactor vessel and onto the containment floor. These UERs may address injection of water to cool the corium, controlling containment overpressure and overtemperature, and minimizing radiological release through containment penetrations. The FOAM model is intended to be a generalized method for development of technical guidance in accident management and not constrained to the conditions associated with ATWS.

Applications of the FOAM model were identified for regulatory, industry, and research perspectives. These applications are linked by the apparent central need for technical guidelines to support emergency procedures and operator training. Regulatory guidelines for accident management practices could identify a standardized method for structuring, at a high general level, the scope of operator response. At a site-specific level, the method would be modified to tailor technical guidelines according to plant specifications. Procedures and training would take advantage of general data and knowledge on accident phenomenology as applied to plant operations. The advantage of a qualitative method is the flexibility provided in structuring and tailoring technical guidance for specific plants.

### 4.3 Recommendations for Accident Sequence Analysis

From the limited base of experience in supporting SASA analysts accrued in the BWR ATWS investigation, several recommendations have emerged. These recommendations concern problems and deficiencies identified during the assessment of ATWS and the application of methods supporting further SASA investigations.

Human factors problems and deficiencies identified in the assessment of ATWS include design and training problems with the EPGs and control room instrumentation deficiencies. Further evaluation of the SASA recommendation to separate ATWS-related instructions from the EPGs seems necessary. For example, symptom-based emergency procedures for PWRs should be examined to assess alternate approaches to the design of instructions pertinent to ATWS. These comparisons may result in potential improvements to the emergency procedures. The EPGs should also be assessed along other types of events to identify other potential difficulties in their use before revamping them solely to accommodate the complexities of ATWS. Two operator actions were identified in the EPGs as presenting considerable difficulty for operators. Instructions for lowering the reactor vessel water level to the top of the fuel and emergency depressurization under the conditions imposed by an ATWS need to be addressed in operator training. Training and procedures should address the potential loss of HPCI and also review the negative consequences from manually shifting the suction of LPCI to the PSP. Simulator practice seems important for ensuring performance proficiency on all these tasks. Furthermore, a program of training may be beneficial for general severe accident mitigation. A goal of such a program would be to review symptom-based procedures by working through perturbations of severe accidents to facilitate operator skills in event detection, diagnosis, and mitigation. The motivation for this program assumes that, while operator response is initially guided by symptom-based procedures, operators will be attempting to classify the transient and may tend to make some decisions on the basis of that classification. Walkthroughs and simulator practice would be relevant elements to a severe accident related training program.

Previous research in the Safety-Related Operator Actions (SROA) program suggests that operator response is a combination of symptomatic and event-based actions (Ref. 48). The finding that the EPGs present some problems for operators in responding to ATWS tends to confirm this initial observation as to how operators actually do respond.

Several human engineering deficiencies in control room instrumentation design were identified. Reactor vessel water level indications should be assessed so as to provide operators with a clearer display of actual level. It is noted that reactor vessel level instrumentation has been a controversial issue, primarily for PWRs, since the TMI accident. A large digital readout mounted atop a central panel and referenced to the top of the fuel is one option. This digital indicator may be designed to receive its input from the SPDS. Some indications are needed reflecting the positions of SRVs in their automatic cycle mode. A status lamp that would immediately illuminate when an SRV would automatically open on high vessel pressure may be useful to operators. Even more appropriate to reactor vessel pressure control would be an automatic or computer-based system using a low-low set (LLS) logic that would stagger the auto opening and closing of SRVs, but also permit manual control. This auto control might systematically open a set of SRVs over a range of 1000-1040 psi and close them over a range of 850-890 psi. This would lessen operator workload during an ATWS, and in other transients, in monitoring and attempting to control reactor vessel pressure. The LLS arrangement also reduces the challenges to containment integrity.

Certain human factors methods seem more useful than others to SASA analysts. The planning, conducting, and videotaping of control room simulator exercises involving accident perturbations provide essential plant and human factors data for other SASA studies. Simulator performance measurement data and observations of operator performance provides an operational perspective different from that provided by computer thermohydraulics codes. The qualitative and quantitative assessments of performance requirements and performance shaping factors provide additional understanding of accident progression and mitigation. Thus, it is recommended that future SASA studies receive ongoing and comprehensive human factors support. This support should seek the integrated assessment of operator performance requirements, procedures, training, control room staffing, computer-based operator aids, and other factors deemed important to particular SASA investigations. In addition, part of the codes used by SASA analysts model selected important operator actions in a deterministic manner. Considering recent progress in human factors computer simulation modeling, some SASA code enhancements could be made through integration of more state-of-the-art operator codes. Such an enhanced code could well provide a much more realistic representation of operator and system response. It is strongly recommended that development of an enhanced model be undertaken combining thermohydraulics and operator/crew codes. Integration of these codes seems to be the next logical step given the state-of-the-art modeling work already in use by SASA and human factors analysts.

### 4.4 Recommendations for Accident Management

The high level descriptive FOAM model provides a conceptual method for the assessment of issues, needs, and directions for accident management research and practices pertinent to regulatory, industry, and research perspectives. A major issue needing to be addressed in accident management concerns the design and development of technical guidance supporting operator response in mitigating core damage and controlling radiological release from the plant. Sources for such guidance include findings and recommendations from SASA and PRA studies, and expertise from engineering, operations, and human factors personnel. Such data would support an effort at validating the fission product barrier diagrams, both in terms of pathways of radiological release and the human factors considerations accompanying potential breaches. Additional work is needed in further developing the FOAM model, or some other appropriate method, for compiling existing data into technical guidance. This guidance should then be used to support extending emergency procedures, for identifying training objectives and task analysis, for developing design and evaluation guidelines for computer-based operator aids, and so forth.

Further research should support formulation of regulatory guidelines addressing human factors issues important to accident management practices. Requirements should address methods for the development of technical guidelines, such as the method enclosed in the FOAM model. The scope of regulatory guidelines as suggested in the functional classification for accident management should encompass operational functions and administrative functions. Administrative functions have been retroactively identified in existing regulatory guidelines. However, some direction is needed for integrating emergency resources into emergency control room operations.

Further research is necessary for extended development of the FOAM model as well as examination of other methods suited to the development of technical guidelines. The nuclear industry needs to take a closer look at accident management in anticipation of new regulatory guidelines. Operator training appears to be one area requiring increased attention, and the INPO guidelines provide some focus for training on mitigating core damage. Identification and assessment of potential unconventional emergency responses should be included in training. Research should also be undertaken in support of operator training that considers knowledges and cognitive abilities pertinent to successful operator response during severe accidents. Such a study might attempt to model skilled performance by distinguishing between expert and novice operators with regard to the manner of cognitive response.

Some potential criteria which these methods should meet include decision criteria for initiation of unconventional actions, and criteria for assessing alternate actions in terms of their consequences from either executing or not executing the actions along serial or parallel paths, and the expected time required to complete actions compared to expected time available. Technical guidelines for accident management should be as standardized as possible using methods reviewed and selected to provide the best support for development of emergency procedures, operator training, and other human factors issues important to accident management practices.

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# APPENDIX A:

## TASK DATA FORMS (TDFs)

# TASK DATA FORM (DESCRIPTIVE)

Page No. 1 of 4

**PLANT IDENTIFICATION**

Plant Name: Browns Ferry

Unit Number: 1

NSSS Vendor: General Electric

A-E: Utility

TG Vendor: General Electric

CR Type: Multiple

OL Date:

**TASK IDENTIFICATION**

Operating Sequence: Anticipated transient without scram

Operating Sequence ID: 7

Operator Function: Supervise and control plant operation

Operator Sub-function: Mitigate consequences of an accident

Comments:

CUE: Procedure

Task Statement: Manually operate safety/relief valves before 1105 psig reactor pressure is reached

Task Purpose: To limit reactor pressure

INPO Task Code:

Task Sequence No.: 13

Task Duration: 5 minutes 44 seconds

Procedures: GOI-100-1 Section VII Emergency shutdown with MSIV closure

Data Collected at: Simulator

| | | Behavior | | Object of Action | | | | | | | Communication Link | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| JOBCAT | JLOC | TIME | VERB | COMPONENT | PARAMETER | STATE | OTHER OBJECT | PLANT SYSTEM | INPO EQUIV | MEANS | RJC | RLOC | CONTENT |
| RO1 | 29 | 00:24 00:25 | Monitors | Reactor | Pressure | High | | Main Steam | | Meter | | | |
| RO1 | 29 | 00:26 00:27 | Informs | | | | | | | Verbal | RO2 | 10 | Pressure is 1080 |
| RO2 | 10 | 00:28 00:29 | Informs | | | | | | | Verbal | RO1 | 29 | I am going to reduce pressure |
| RO2 | 10 | 00:30 00:31 | Positions | Valve | Position | Open | | Main Steam | | Discrete Control | | | |
| RO2 | 10 | 00:31 00:47 | Monitors | Reactor | Pressure | Decreasing | | Main Steam | | Meter | | | |
| RO2 | 10 | 00:47 00:48 | Positions | Valve | Position | Closed | | Main Steam | | Discrete Control | | | |
| RO2 | 10 | 1:52 1:53 | Positions | Valve | Position | Open | | Main Steam | | Discrete Control | | | |
| RO2 | 10 | 1:52 1:53 | Positions | Valve | Position | Open | | Main Steam | | Discrete Control | | | |

TX-5197

## TASK DATA FORM (DESCRIPTIVE)



**PLANT IDENTIFICATION**

Plant Name ______

Unit Number ______

NSSS Vendor ______

A-E ______

TG Vendor ______

CR Type ______

OL Date ______

**TASK IDENTIFICATION**

Operating Sequence ______

Operating Sequence ID ______

Operator Function ______

Operator Sub-function ______

Comments ______

CUE ______

Task Statement ______

Task Purpose ______

INPO Task Code ______

Task Sequence No. ______

Task Duration ______

Procedures ______

Data Collected at: ______

| | | Behavior | | Object of Action | | | | | | | Communication Link | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| JOBCAT | JLOC | TIME | VERB | COMPONENT | PARAMETER | STATE | OTHER OBJECT | PLANT SYSTEM | INPO EQUIV | MEANS | RJC | RLOC | CONTENT |
| RO2 | 10 | 1:53 2:02 | Monitors | Reactor | Pressure | Decreasing | | Main Steam | | Meter | | | |
| RO2 | 10 | 2:02 2:03 | Positions | Valve | Position | Closed | | Main Steam | | Discrete Control | | | |
| RO2 | 10 | 2:02 2:03 | Positions | Valve | Position | Closed | | Main Steam | | Discrete Control | | | |
| RO1 | 29 | 5:17 5:18 | Informs | | | | | | | Verbal | RO2 | 10 | Pressure is high |
| RO2 | 10 | 5:20 5:21 | Positions | Valve | Position | Open | | Main Steam | | Discrete Control | | | |
| RO2 | 10 | 5:20 5:21 | Positions | Valve | Position | Open | | Main Steam | | Discrete Control | | | |
| RO2 | 10 | 5:20 5:49 | Monitors | Reactor | Pressure | Decreasing | | Main Steam | | Meter | | | |

TX-5197

# TASK DATA FORM (DESCRIPTIVE)



**PLANT IDENTIFICATION**

Plant Name ______

Unit Number ______

NSSS Vendor ______

A-E ______

TG Vendor ______

CR Type ______

OL Date ______

**TASK IDENTIFICATION**

Operating Sequence ______

Operating Sequence ID ______

Operator Function ______

Operator Sub-function ______

Comments ______

CUE ______

Task Statement ______

Task Purpose ______

INPO Task Code ______

Task Sequence No. ______

Task Duration ______

Procedures ______

Data Collected at: ______

| | | Behavior | | Object of Action | | | | | | | Communication Link | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| JOBCAT | JLOC | TIME | VERB | COMPONENT | PARAMETER | STATE | OTHER OBJECT | PLANT SYSTEM | INPO EQUIV | MEANS | RJC | RLOC | CONTENT |
| R02 | 10 | 5:27<br>5:28 | Positions | Valve | Position | Open | | Main Steam | | Discrete Control | | | |
| R02 | 10 | 5:27<br>5:28 | Positions | Valve | Position | Open | | Main Steam | | Discrete Control | | | |
| R02 | 10 | 5:27<br>5:28 | Informs | | | | | | | Verbal | R01 | Control Room | I am decreasing reactor pressure |
| R02 | 10 | 5:54<br>5:55 | Positions | Valve | Position | Closed | | Main Steam | | Discrete Control | | | |
| R02 | 10 | 5:54<br>5:55 | Positions | Valve | Position | Closed | | Main Steam | | Discrete Control | | | |
| R02 | 10 | 5:54<br>6:10 | Monitors | Reactor | Pressure | Decreasing | | Main Steam | | Meter | | | |
| R02 | 10 | 5:55<br>5:56 | Positions | Valve | Position | Closed | | Main Steam | | Discrete Control | | | |

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# TASK DATA FORM (DESCRIPTIVE)



**PLANT IDENTIFICATION**

Plant Name ______

Unit Number ______

NSSS Vendor ______

A-E ______

TG Vendor ______

CR Type ______

OL Date ______

**TASK IDENTIFICATION**

Operating Sequence ______

Operating Sequence ID ______

Operator Function ______

Operator Sub-function ______

Comments ______

CUE ______

Task Statement ______

Task Purpose ______

INPO Task Code ______

Task Sequence No. ______

Task Duration ______

Procedures ______

Data Collected at: ______

| | | Behavior | | Object of Action | | | | | | | Communication Link | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| JOBCAT | JLOC | TIME | VERB | COMPONENT | PARAMETER | STATE | OTHER OBJECT | PLANT SYSTEM | INPO EQUIV | MEANS | RJC | RLOC | CONTENT |
| RO2 | 10 | 5:55<br>5:56 | Positions | Valve | Position | Closed | | Main Steam | | Discrete Control | | | |
| RO2 | 10 | 6:02<br>6:04 | Informs | | | | | | | Verbal | RO1 | 29 | Pressure decreased to 800 but is increasing fast |
| RO2 | 10 | 6:09<br>6:10 | Informs | | | | | | | Verbal | RO1 | 29 | Pressure is now 1100 |

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# TASK DATA FORM (DESCRIPTIVE)



**PLANT IDENTIFICATION**

- Plant Name: Browns Ferry
- Unit Number: 1
- NSSS Vendor: General Electric
- A-E: Utility
- TG Vendor: General Electric
- CR Type: Multiple
- OL Date:

**TASK IDENTIFICATION**

- Operating Sequence: ATWS with MSIV closure
- Operating Sequence ID: 7
- Operator Function: Supervise and control plant operation
- Operator Sub-function: Mitigate consequences of an accident
- Comments:
- CUE: Procedure

- Task Statement: Manual insertion of control rods
- Task Purpose: To reduce reactor power
- INPO Task Code:
- Task Sequence No. 17
- Task Duration: 13 minutes 30 seconds
- Procedures: EOI-47 "Failure of reactor to scram when required"
- Data Collected at: Simulator

| | | Behavior | | Object of Action | | | | | | | Communication Link | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| JOBCAT | JLOC | TIME | VERB | COMPONENT | PARAMETER | STATE | OTHER OBJECT | PLANT SYSTEM | INPO EQUIV | MEANS | RJC | RLOC | CONTENT |
| R01 | 29 | 0:29<br>0:30 | Positions | Control Rod | Rate | Override | | $R_x$ Manual Control | | Discrete Control | | | |
| R01 | 29 | 0:31<br>0:33 | Positions | Control Rod | Position | Out | | RMCS | | Discrete Control | | | |
| R01 | 29 | 0:33<br>0:41 | Scans | Control Rod | Position | Steady | | $R_x$ Control | | Counters | | | |
| R01 | 29 | 0:33<br>0:41 | Positions | Control Rod | Position | In | | RMCS | | Discrete Control | | | |
| R01 | 29 | 0:34<br>0:35 | Informs | | | | | | | Verbal | R02 | 16 | Emergency rod in and selecting a control rod |
| R01 | 29 | 0:37<br>0:37 | Pushes | Master Controller | Reactivity | Enable | | RMCS | | Discrete Control | | | |
| R01 | 29 | 0:45<br>0:53 | Positions | Control Rod | Position | In | | RMCS | | Discrete Control | | | |

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# TASK DATA FORM (DESCRIPTIVE)



**PLANT IDENTIFICATION**

Plant Name ______

Unit Number ______

NSSS Vendor ______

A-E ______

TG Vendor ______

CR Type ______

OL Date ______

**TASK IDENTIFICATION**

Operating Sequence ______

Operating Sequence ID ______

Operator Function ______

Operator Sub-function ______

Comments ______

CUE ______

Task Statement ______

Task Purpose ______

INPO Task Code ______

Task Sequence No. ______

Task Duration ______

Procedures ______

Data Collected at: ______

| | | Behavior | | Object of Action | | | | | | | Communication Link | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| JOBCAT | JLOC | TIME | VERB | COMPONENT | PARAMETER | STATE | OTHER OBJECT | PLANT SYSTEM | INPO EQUIV | MEANS | RJC | RLOC | CONTENT |
| R01 | 29 | 0:54<br>0:58 | Positions | Control Rods | Position | In | | RMCS | | Discrete Control | | | |
| R01 | 33 | 1:28<br>1:32 | Locates | | | | Rod Sequence Procedure | RMCS | | Printed Material | | | |
| R01 | 29 | 1:34<br>1:36 | Reads | | | | Rod Sequence Procedure | RMCS | | Printed Material | | | |
| R01 | 29 | 1:38<br>1:38 | Pushes | Master Controller | Reactivity | Enable | | RMCS | | Discrete Control | | | |
| R01 | 29 | 1:40<br>1:40 | Pushes | Master Controller | Reactivity | Enable | | RMCS | | Discrete Control | | | |
| R01 | 29 | 1:40<br>1:41 | Observes | Control Rods | Position | Steady | | RMCS | | Counter | | | |
| R01 | 29 | 2:14<br>2:58 | Positions | Control Rods | In | Continuous | | RMCS | | Discrete Control | | | |

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# TASK DATA FORM (DESCRIPTIVE)

Page No. 3 of 14

**PLANT IDENTIFICATION**

Plant Name ______
Unit Number ______
NSSS Vendor ______
A-E ______
TG Vendor ______
CR Type ______
OL Date ______

**TASK IDENTIFICATION**

Operating Sequence ______
Operating Sequence ID ______
Operator Function ______
Operator Sub-function ______
Comments ______

CUE ______

Task Statement ______
Task Purpose ______
INPO Task Code ______
Task Sequence No. ______
Task Duration ______
Procedures ______

Data Collected at: ______

| | | Behavior | | Object of Action | | | | | | | Communication Link | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| JOBCAT | JLOC | TIME | VERB | COMPONENT | PARAMETER | STATE | OTHER OBJECT | PLANT SYSTEM | INPO EQUIV | MEANS | RJC | RLOC | CONTENT |
| RO1 | 29 | 2:26<br>2:31 | Informs | | | | | | | | SRO1 | 33 | I am deviating from the rod pattern |
| RO1 | 29 | 2:26<br>2:32 | Scans | Control Rods | Position | Steady | | RMCS | | Counter | | | |
| RO1 | 29 | 2:33<br>2:33 | Pushes | Master Controller | Reactivity | Enable | | RMCS | | Discrete Control | | | |
| RO1 | 29 | 3:00<br>3:02 | Scans | Control Rods | Position | Steady | | RMCS | | Counter | | | |
| RO1 | 29 | 3:03<br>3:03 | Pushes | Master Controller | Reactivity | Enable | | RMCS | | Discrete Control | | | |
| RO1 | 29 | 3:03<br>3:19 | Positions | Control Rods | Position | In | | RMCS | | Discrete Control | | | |
| RO1 | 29 | 3:30<br>4:01 | Positions | Control Rods | Position | In | | RMCS | | Discrete Control | | | |

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# TASK DATA FORM (DESCRIPTIVE)



**PLANT IDENTIFICATION**

Plant Name ______

Unit Number ______

NSSS Vendor ______

A-E ______

TG Vendor ______

CR Type ______

OL Date ______

**TASK IDENTIFICATION**

Operating Sequence ______

Operating Sequence ID ______

Operator Function ______

Operator Sub-function ______

Comments ______

CUE ______

Task Statement ______

Task Purpose ______

INPO Task Code ______

Task Sequence No. ______

Task Duration ______

Procedures ______

Data Collected at: ______

| | | Behavior | | Object of Action | | | | | | | Communication Link | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| JOBCAT | JLOC | TIME | VERB | COMPONENT | PARAMETER | STATE | OTHER OBJECT | PLANT SYSTEM | INPO EQUIV | MEANS | RJC | RLOC | CONTENT |
| RO1 | 29 | 3:34<br>3:34 | Pushes | Master Controller | Reactivity | Enable | | RMCS | | Discrete Control | | | |
| RO1 | 29 | 3:35<br>3:41 | Scans | Control Rod | Position | Steady | | RMCS | | Counter | | | |
| SRO1 | 33 | 3:40<br>3:42 | Directs | | | | | RMCS | | Verbal | RO1 | 19 | Manually insert rod as allowed by RSCS |
| RO1 | 29 | 3:42<br>3:42 | Pushes | Master Controller | Reactivity | Enable | | RMCS | | Discrete Control | | | |
| RO1 | 29 | 3:45<br>3:48 | Informs | | | | | RMCS | | | SRO1 | 33 | Rods are being inserted optimizing rod group selection |
| SRO1 | 33 | 3:55<br>4:01 | Directs | | | | | RMCS | | Verbal | RO1 | 19 | Place rod sequence mode select switch to insert and select rod group |
| RO1 | 26 | 4:00<br>4:01 | Positions | Master Controller | Reactivity | Manual Group Select | | RMCS | | Discrete Control | | | |

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# TASK DATA FORM (DESCRIPTIVE)



**PLANT IDENTIFICATION**

Plant Name ______________________

Unit Number ______________________

NSSS Vendor ______________________

A-E ______________________

TG Vendor ______________________

CR Type ______________________

OL Date ______________________

**TASK IDENTIFICATION**

Operating Sequence ______________________

Operating Sequence ID ______________________

Operator Function ______________________

Operator Sub-function ______________________

Comments ______________________

CUE ______________________

Task Statement ______________________

Task Purpose ______________________

INPO Task Code ______________________

Task Sequence No. ______________________

Task Duration ______________________

Procedures ______________________

Data Collected at: ______________________

| | | Behavior | | Object of Action | | | | | | | Communication Link | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| JOBCAT | JLOC | TIME | VERB | COMPONENT | PARAMETER | STATE | OTHER OBJECT | PLANT SYSTEM | INPO EQUIV | MEANS | RJC | RLOC | CONTENT |
| RO1 | 29 | 4:03<br>4:16 | Scans | Control Rod | Position | Steady | | RMCS | | Counter | | | |
| RO1 | 29 | 4:17<br>4:17 | Pushes | Master Controller | Reactivity | Enable | | RMCS | | Discrete Control | | | |
| RO1 | 29 | 4:17<br>4:46 | Positions | Control Rod | Position | In | | RMCS | | Discrete Control | | | |
| RO1 | 29 | 4:59<br>5:26 | Positions | Control Rod | Position | In | | | | Discrete Control | | | |
| RO1 | 29 | 5:25<br>5:34 | Scans | Control Rod | Position | Steady | | RMCS | | Counter | | | |
| RO1 | 29 | 5:34<br>5:34 | Pushes | Master Controller | Reactivity | Enable | | RMCS | | Discrete Control | | | |
| RO1 | 29 | 5:35<br>5:55 | Positions | Control Rod | Position | In | | RMCS | | Discrete Control | | | |

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# TASK DATA FORM (DESCRIPTIVE)



**PLANT IDENTIFICATION**

Plant Name ______
Unit Number ______
NSSS Vendor ______
A-E ______
TG Vendor ______
CR Type ______
OL Date ______

**TASK IDENTIFICATION**

Operating Sequence ______
Operating Sequence ID ______
Operator Function ______
Operator Sub-function ______
Comments ______

CUE ______

Task Statement ______
Task Purpose ______
INPO Task Code ______
Task Sequence No. ______
Task Duration ______
Procedures ______

Data Collected at: ______

| | | Behavior | | Object of Action | | | | | | | Communication Link | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| JOBCAT | JLOC | TIME | VERB | COMPONENT | PARAMETER | STATE | OTHER OBJECT | PLANT SYSTEM | INPO EQUIV | MEANS | RJC | RLOC | CONTENT |
| R01 | 29 | 5:51<br>5:55 | Scans | Control Rod | Position | Steady | | RMCS | | Counter | | | |
| R01 | 29 | 5:58<br>5:58 | Pushes | Master Controller | Reactivity | Enable | | RMCS | | Discrete Control | | | |
| R01 | 29 | 6:00<br>6:22 | Positions | Control Rod | Position | In | | RMCS | | Discrete Control | | | |
| R01 | 29 | 6:26<br>6:26 | Pushes | Master Controller | Reactivity | Enable | | RMCS | | Discrete Control | | | |
| R01 | 29 | 6:27<br>6:29 | Scans | Control Rod | Position | Steady | | RMCS | | Counter | | | |
| R01 | 29 | 6:29<br>6:29 | Pushes | Master Controller | Reactivity | Enable | | RMCS | | Discrete Control | | | |
| R01 | 29 | 6:25<br>6:25 | Positions | Master Controller | Reactivity | Manual Group Select | | RMCS | | Discrete Control | | | |

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# TASK DATA FORM (DESCRIPTIVE)



**PLANT IDENTIFICATION**

Plant Name ______
Unit Number ______
NSSS Vendor ______
A-E ______
TG Vendor ______
CR Type ______
OL Date ______

**TASK IDENTIFICATION**

Operating Sequence ______
Operating Sequence ID ______
Operator Function ______
Operator Sub-function ______
Comments ______

CUE ______

Task Statement ______
Task Purpose ______
INPO Task Code ______
Task Sequence No. ______
Task Duration ______
Procedures ______

Data Collected at: ______

| | | Behavior | | Object of Action | | | | | | | Communication Link | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| JOBCAT | JLOC | TIME | VERB | COMPONENT | PARAMETER | STATE | OTHER OBJECT | PLANT SYSTEM | INPO EQUIV | MEANS | RJC | RLOC | CONTENT |
| RO1 | 29 | 6:28<br>6:28 | Pushes | Master Controller | Reactivity | Enable | | RMCS | | Discrete Control | | | |
| RO1 | 29 | 6:31<br>6:31 | Positions | Master Controller | Reactivity | Manual Group Select | | RMCS | | Discrete Control | | | |
| RO1 | 29 | 6:32<br>6:32 | Positions | Master Controller | Reactivity | Manual Group Select | | RMCS | | Discrete Control | | | |
| RO1 | 29 | 6:33<br>6:33 | Positions | Master Controller | Reactivity | Manual Group Select | | RMCS | | Discrete Control | | | |
| RO1 | 29 | 6:35<br>6:35 | Pushes | Master Controller | Reactivity | Enable | | RMCS | | Discrete Control | | | |
| RO1 | 29 | 6:35<br>6:35 | Pushes | Master Controller | Reactivity | Enable | | RMCS | | Discrete Control | | | |
| RO1 | 29 | 6:36<br>7:42 | Positions | Control Rod | Position | In | | RMCS | | Discrete Control | | | |

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# TASK DATA FORM (DESCRIPTIVE)



**PLANT IDENTIFICATION**

Plant Name ______

Unit Number ______

NSSS Vendor ______

A-E ______

TG Vendor ______

CR Type ______

OL Date ______

**TASK IDENTIFICATION**

Operating Sequence ______

Operating Sequence ID ______

Operator Function ______

Operator Sub-function ______

Comments ______

CUE ______

Task Statement ______

Task Purpose ______

INPO Task Code ______

Task Sequence No. ______

Task Duration ______

Procedures ______

Data Collected at: ______

| | | Behavior | | Object of Action | | | | | | | Communication Link | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| JOBCAT | JLOC | TIME | VERB | COMPONENT | PARAMETER | STATE | OTHER OBJECT | PLANT SYSTEM | INPO EQUIV | MEANS | RJC | RLOC | CONTENT |
| RO1 | 29 | 6:59<br>7:06 | Scans | Control Rod | Position | Steady | | RMCS | | Counter | | | |
| RO1 | 29 | 7:06<br>7:06 | Pushes | Master Controller | Reactivity | Enable | | RMCS | | Discrete Control | | | |
| RO1 | 29 | 7:07<br>7:07 | Scans | Control Rod | Position | Steady | | RMCS | | Counter | | | |
| RO1 | 29 | 7:16<br>7:16 | Pushes | Master Controller | Reactivity | Enable | | RMCS | | Discrete Control | | | |
| RO1 | 29 | 7:17<br>7:23 | Scans | Control Rod | Position | Steady | | RMCS | | Counter | | | |
| RO1 | 29 | 7:47<br>8:09 | Positions | Control Rod | Position | In | | RMCS | | Discrete Control | | | |
| RO1 | 29 | 8:08<br>8:10 | Scans | Control Rod | Position | Steady | | RMCS | | Counter | | | |

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# TASK DATA FORM (DESCRIPTIVE)



**PLANT IDENTIFICATION**

Plant Name ____________________

Unit Number ____________________

NSSS Vendor ____________________

A-E ____________________

TG Vendor ____________________

CR Type ____________________

OL Date ____________________

**TASK IDENTIFICATION**

Operating Sequence ____________________

Operating Sequence ID ____________________

Operator Function ____________________

Operator Sub-function ____________________

Comments ____________________

CUE ____________________

Task Statement ____________________

Task Purpose ____________________

INPO Task Code ____________________

Task Sequence No. ____________________

Task Duration ____________________

Procedures ____________________

Data Collected at: ____________________

| | | Behavior | | Object of Action | | | | | | | Communication Link | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| JOBCAT | JLOC | TIME | VERB | COMPONENT | PARAMETER | STATE | OTHER OBJECT | PLANT SYSTEM | INPO EQUIV | MEANS | RJC | RLOC | CONTENT |
| R01 | 29 | 8:11<br>8:11 | Pushes | Master Controller | Reactivity | Enable | | RMCS | | Discrete Control | | | |
| R01 | 29 | 8:16<br>8:40 | Positions | Control Rod | Position | In | | RMCS | | Discrete Control | | | |
| R01 | 29 | 8:41<br>10:00 | Positions | Control Rod | Position | In | | RMCS | | Discrete Control | | | |
| R01 | 29 | 9:01<br>9:19 | Scans | Control Rod | Position | Steady | | RMCS | | Counter | | | |
| R01 | 29 | 9:20<br>9:20 | Pushes | Master Controller | Reactivity | Enable | | RMCS | | Discrete Control | | | |
| R01 | 29 | 9:25<br>9:31 | Scans | Control Rod | Position | Steady | | RMCS | | Counter | | | |
| R01 | 29 | 10:06<br>10:08 | Positions | Control Rod | Position | In | | RMCS | | Discrete Control | | | |

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# TASK DATA FORM (DESCRIPTIVE)

Page No. 10 of 14

**PLANT IDENTIFICATION**

Plant Name ____
Unit Number ____
NSSS Vendor ____
A-E ____
TG Vendor ____
CR Type ____
OL Date ____

**TASK IDENTIFICATION**

Operating Sequence ____
Operating Sequence ID ____
Operator Function ____
Operator Sub-function ____
Comments ____
CUE ____

Task Statement ____
Task Purpose ____
INPO Task Code ____
Task Sequence No. ____
Task Duration ____
Procedures ____
Data Collected at: ____

| | | Behavior | | Object of Action | | | | | | | Communication Link | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| JOBCAT | JLOC | TIME | VERB | COMPONENT | PARAMETER | STATE | OTHER OBJECT | PLANT SYSTEM | INPO EQUIV | MEANS | RJC | RLOC | CONTENT |
| R01 | 29 | 10:11 10:11 | Positions | Master Controller | Reactivity | Manual Group Select | | RMCS | | Discrete Control | | | |
| R01 | 29 | 10:11 10:19 | Observes | Master Controller | Reactivity | Steady | | RMCS | | Indicator Lights | | | |
| R01 | 29 | 10:14 10:14 | Positions | Master Controller | Reactivity | Manual Group Select | | RMCS | | Discrete Control | | | |
| R01 | 29 | 10:15 10:15 | Positions | Master Controller | Reactivity | Manual Group Select | | RMCS | | Discrete Control | | | |
| R01 | 29 | 10:16 10:16 | Positions | Master Controller | Reactivity | Manual Group Select | | RMCS | | Discrete Control | | | |
| R01 | 29 | 10:17 10:17 | Positions | Master Controller | Reactivity | Manual Group Select | | RMCS | | Discrete Control | | | |
| R01 | 29 | 10:21 10:21 | Positions | Control Rod | Rate | Override | | RMCS | | Discrete Control | | | |

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# TASK DATA FORM (DESCRIPTIVE)



**PLANT IDENTIFICATION**

Plant Name ______

Unit Number ______

NSSS Vendor ______

A-E ______

TG Vendor ______

CR Type ______

OL Date ______

**TASK IDENTIFICATION**

Operating Sequence ______

Operating Sequence ID ______

Operator Function ______

Operator Sub-function ______

Comments ______

CUE ______

Task Statement ______

Task Purpose ______

INPO Task Code ______

Task Sequence No. ______

Task Duration ______

Procedures ______

Data Collected at: ______

| | | Behavior | | Object of Action | | | | | | | Communication Link | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| JOBCAT | JLOC | TIME | VERB | COMPONENT | PARAMETER | STATE | OTHER OBJECT | PLANT SYSTEM | INPO EQUIV | MEANS | RJC | RLOC | CONTENT |
| RO1 | 29 | 10:21<br>10:21 | Pushes | Master Controller | Reactivity | Enable | | RMCS | | Discrete Control | | | |
| RO1 | 29 | 10:22<br>11:15 | Positions | Control Rod | Position | In | | RMCS | | Discrete Control | | | |
| RO1 | 29 | 10:51<br>11:15 | Scans | Control Rod | Position | Steady | | RMCS | | Counter | | | |
| RO1 | 29 | 11:04<br>11:04 | Pushes | Master Controller | Reactivity | Enable | | RMCS | | Discrete Control | | | |
| RO1 | 29 | 11:21<br>11:43 | Positions | Control Rod | Position | In | | RMCS | | Discrete Control | | | |
| RO1 | 29 | 11:48<br>13:44 | Positions | Control Rod | Position | In | | RMCS | | Discrete Control | | | |
| RO1 | 29 | 12:01<br>12:25 | Scans | Control Rod | Position | Steady | | RMCS | | Counter | | | |

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# TASK DATA FORM (DESCRIPTIVE)



**PLANT IDENTIFICATION**

Plant Name ____________________

Unit Number ____________________

NSSS Vendor ____________________

A-E ____________________

TG Vendor ____________________

CR Type ____________________

OL Date ____________________

**TASK IDENTIFICATION**

Operating Sequence ____________________

Operating Sequence ID ____________________

Operator Function ____________________

Operator Sub-function ____________________

Comments ____________________

____________________

____________________

CUE ____________________

Task Statement ____________________

Task Purpose ____________________

INPO Task Code ____________________

Task Sequence No. ____________________

Task Duration ____________________

Procedures ____________________

____________________

Data Collected at: ____________________

| | | Behavior | | Object of Action | | | | | | | Communication Link | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| JOBCAT | JLOC | TIME | VERB | COMPONENT | PARAMETER | STATE | OTHER OBJECT | PLANT SYSTEM | INPO EQUIV | MEANS | RJC | RLOC | CONTENT |
| RO1 | 29 | 12:06<br>12:06 | Pushes | Master Controller | Reactivity | Enable | | RMCS | | Discrete Control | | | |
| RO1 | 29 | 12:42<br>12:56 | Scans | Control Rod | Position | Steady | | RMCS | | Counter | | | |
| RO1 | 29 | 12:52<br>12:52 | Pushes | Master Controller | Reactivity | Enable | | RMCS | | Discrete Control | | | |
| RO1 | 29 | 13:23<br>13:44 | Scans | Control Rod | Position | Steady | | RMCS | | Counter | | | |
| RO1 | 29 | 13:48<br>14:05 | Scans | Control Rod | Position | Steady | | RMCS | | Counter | | | |
| RO1 | 29 | 13:51<br>13:51 | Positions | Master Controller | Reactivity | Manual Group Select | | RMCS | | Discrete Control | | | |
| RO1 | 29 | 13:54<br>13:54 | Positions | Master Controller | Reactivity | Manual Group Select | | RMCS | | Discrete Control | | | |

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# TASK DATA FORM (DESCRIPTIVE)



**PLANT IDENTIFICATION**

Plant Name ______

Unit Number ______

NSSS Vendor ______

A-E ______

TG Vendor ______

CR Type ______

OL Date ______

**TASK IDENTIFICATION**

Operating Sequence ______

Operating Sequence ID ______

Operator Function ______

Operator Sub-function ______

Comments ______

CUE ______

Task Statement ______

Task Purpose ______

INPO Task Code ______

Task Sequence No. ______

Task Duration ______

Procedures ______

Data Collected at: ______

| | | Behavior | | Object of Action | | | | | | | | Communication Link | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| JOBCAT | JLOC | TIME | VERB | COMPONENT | PARAMETER | STATE | OTHER OBJECT | PLANT SYSTEM | INPO EQUIV | MEANS | RJC | RLOC | CONTENT |
| R01 | 29 | 13:56<br>13:56 | Positions | Master Controller | Reactivity | Manual Group Select | | RMCS | | Discrete Control | | | |
| R01 | 29 | 13:59<br>13:59 | Positions | Control Rod | Rate | Override | | RMCS | | Discrete Control | | | |
| R01 | 29 | 13:59<br>15:01 | Positions | Control Rod | Position | In | | RMCS | | Discrete Control | | | |
| R01 | 29 | 14:00<br>14:00 | Pushes | Master Controller | Reactivity | Enable | | RMCS | | Discrete Control | | | |
| R01 | 29 | 14:31<br>14:46 | Scans | Control Rod | Position | Steady | | RMCS | | Counter | | | |
| R01 | 29 | 14:34<br>14:34 | Pushes | Master Controller | Reactivity | Enable | | RMCS | | Discrete Control | | | |
| R01 | 29 | 15:21<br>15:52 | Positions | Control Rod | Position | In | | RMGS | | Discrete Control | | | |

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# TASK DATA FORM (DESCRIPTIVE)



**PLANT IDENTIFICATION**

Plant Name ______
Unit Number ______
NSSS Vendor ______
A-E ______
TG Vendor ______
CR Type ______
OL Date ______

**TASK IDENTIFICATION**

Operating Sequence ______
Operating Sequence ID ______
Operator Function ______
Operator Sub-function ______
Comments ______

CUE ______

Task Statement ______
Task Purpose ______
INPO Task Code ______
Task Sequence No. ______
Task Duration ______
Procedures ______

Data Collected at: ______

| | | Behavior | | Object of Action | | | | | | | Communication Link | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| JOBCAT | JLOC | TIME | VERB | COMPONENT | PARAMETER | STATE | OTHER OBJECT | PLANT SYSTEM | INPO EQUIV | MEANS | RJC | RLOC | CONTENT |
| R01 | 29 | 15:58<br>15:58 | Pushes | Master Controller | Reactivity | Enable | | RMCS | | Discrete Control | | | |
| R01 | 29 | 15:59<br>16:26 | Positions | Control Rod | Position | In | | RMCS | | Discrete Control | | | |
| R01 | 29 | 17:27<br>17:33 | Positions | Control Rod | Position | In | | RMCS | | Discrete Control | | | |
| R01 | ENDS | TASK #17 | | | | | | | | | | | |

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# TASK DATA FORM (DESCRIPTIVE)

Page No. 1 of 3

**PLANT IDENTIFICATION**

Plant Name: Browns Ferry

Unit Number: 1

NSSS Vendor: General Electric

A-E: Utility

TG Vendor: General Electric

CR Type: Multiple

OL Date:

**TASK IDENTIFICATION**

Operating Sequence: Anticipated transient without scram

Operating Sequence ID: 7

Operator Function: Supervise and control plant operation

Operator Sub-function: Mitigate consequences of an accident

Comments:

CUE: Procedure

Task Statement: Initiate suppression pool cooling

Task Purpose: To limit suppression pool temperature

INPO Task Code:

Task Sequence No.: 20

Task Duration: 2 minutes 41 seconds

Procedures: GOI-100-1 Section VII Emergency shutdown with MSIV closure

Data Collected at: Simulator

| | | Behavior | | Object of Action | | | | | | | Communication Link | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| JOBCAT | JLOC | TIME | VERB | COMPONENT | PARAMETER | STATE | OTHER OBJECT | PLANT SYSTEM | INPO EQUIV | MEANS | RJC | RLOC | CONTENT |
| RO2 | 12 | 12:40<br>12:42 | Monitors | Torus | Temperature | High | | RHR | | Meter | | | |
| RO2 | 12 | 12:43<br>12:44 | Positions | Pump | Power | On | | RHR | | Discrete Control | | | |
| RO2 | 12 | 12:43<br>12:44 | Positions | Pump | Power | On | | RHR | | Discrete Control | | | |
| RO2 | 12 | 12:43<br>12:44 | Observes | Pump | Power | On | | RHR | | Indicator Light | | | |
| RO2 | 12 | 12:43<br>12:44 | Observes | Pump | Power | On | | RHR | | Indicator Light | | | |
| RO2 | 12 | 12:47<br>12:51 | Positions | Valve | Position | Open | | RHR | | Discrete Control | | | |
| RO2 | 12 | 12:49<br>14:38 | Positions | Valve | Position | Open | | RHR | | Discrete Control | | | |
| RO2 | 12 | 12:54<br>12:54 | Positions | Pump | Power | On | | RHRSW | | Discrete Control | | | |

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# TASK DATA FORM (DESCRIPTIVE)



**PLANT IDENTIFICATION**

Plant Name ______

Unit Number ______

NSSS Vendor ______

A-E ______

TG Vendor ______

CR Type ______

OL Date ______

**TASK IDENTIFICATION**

Operating Sequence ______

Operating Sequence ID ______

Operator Function ______

Operator Sub-function ______

Comments ______

CUE ______

Task Statement ______

Task Purpose ______

INPO Task Code ______

Task Sequence No. ______

Task Duration ______

Procedures ______

Data Collected at: ______

| | | Behavior | | Object of Action | | | | | | | Communication Link | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| JOBCAT | JLOC | TIME | VERB | COMPONENT | PARAMETER | STATE | OTHER OBJECT | PLANT SYSTEM | INPO EQUIV | MEANS | RJC | RLOC | CONTENT |
| RO2 | 12 | 12:54<br>12:54 | Positions | Pump | Power | On | | RHRSW | | Discrete Control | | | |
| RO2 | 12 | 12:55<br>12:55 | Observes | Pump | Power | On | | RHRSW | | Indicator Light | | | |
| RO2 | 12 | 12:55<br>12:55 | Observes | Pump | Power | On | | RHRSW | | Indicator Light | | | |
| RO2 | 12 | 12:57<br>14:10 | Positions | Valve | Position | Open | | RHRSW | | Discrete Control | | | |
| RO2 | 12 | 12:57<br>12:58 | Observes | Valve | Position | Open | | RHRSW | | Indicator Light | | | |
| RO2 | 12 | 13:12<br>15:24 | Monitors | Pump | Flow | Increasing | | RHRSW | | Meter | | | |
| RO2 | 12 | 13:13<br>14:39 | Monitors | Pump | Flow | Increasing | | RHR | | Meter | | | |

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# TASK DATA FORM (DESCRIPTIVE)



**PLANT IDENTIFICATION**

Plant Name ______
Unit Number ______
NSSS Vendor ______
A-E ______
TG Vendor ______
CR Type ______
OL Date ______

**TASK IDENTIFICATION**

Operating Sequence ______
Operating Sequence ID ______
Operator Function ______
Operator Sub-function ______
Comments ______
CUE ______

Task Statement ______
Task Purpose ______
INPO Task Code ______
Task Sequence No. ______
Task Duration ______
Procedures ______
Data Collected at: ______

| | | Behavior | | Object of Action | | | | | | | Communication Link | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| JOBCAT | JLOC | TIME | VERB | COMPONENT | PARAMETER | STATE | OTHER OBJECT | PLANT SYSTEM | INPO EQUIV | MEANS | RJC | RLOC | CONTENT |
| RO2 | 12 | 13:33<br>13:35 | Informs | | | | | | | Verbal | Crew | Control Room | Loop 1 RHR is in Torus cooling |
| RO2 | 12 | 14:12<br>15:24 | Positions | Valve | Position | Open | | RHRSW | | Discrete Control | | | |
| RO2 | 12 | 15:20<br>15:24 | Informs | | | | | | | Verbal | Crew | Control Room | Loop 1 RHR is full flow in Torus cooling |

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# TASK DATA FORM (DESCRIPTIVE)

Page No. 1 of 1

**PLANT IDENTIFICATION**

Plant Name: Browns Ferry

Unit Number: 1

NSSS Vendor: General Electric

A-E: Utility

TG Vendor: General Electric

CR Type: Multiple

OL Date:

**TASK IDENTIFICATION**

Operating Sequence: Anticipated transient without scram

Operating Sequence ID: 7

Operator Function: Supervise and control plant operation

Operator Sub-function: Mitigate consequences of an accident

Comments:

CUE: Procedure

Task Statement: Verify conditions exist for initiating standby liquid control

Task Purpose: To determine the need for poison injection

INPO Task Code:

Task Sequence No. 22

Task Duration: 0 minutes 52 seconds

Procedures: EOI-47 "Failure of reactor to scram when required"

Data Collected at: Simulator

| | | Behavior | | Object of Action | | | | | | | Communication Link | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| JOBCAT | JLOC | TIME | VERB | COMPONENT | PARAMETER | STATE | OTHER OBJECT | PLANT SYSTEM | INPO EQUIV | MEANS | RJC | RLOC | CONTENT |
| SR01 | 33 | 4:19<br>4:24 | Requests | | | | | | | Verbal | R01 | 29 | Are five or more adjacent rods not inserted past 06 position? |
| R01 | 29 | 4:25<br>4:27 | Informs | | | | | | | Verbal | SR01 | 33 | More than five adjacent rods are full out |
| SR01 | 33 | 4:29<br>4:31 | Requests | | | | | | | Verbal | R01 | 29 | Can reactor water level be maintained? |
| R01 | 29 | 4:32<br>4:36 | Informs | | | | | | | Verbal | SR01 | 33 | No, we are minus 100 inches reactor level |
| SR01 | 33 | 4:37<br>4:39 | Requests | | | | | | | Verbal | R01 | 29 | Is the suppression pool temperature greater than 110°? |
| R01 | 29 | 4:39<br>4:40 | Informs | | | | | | | Verbal | SR01 | 33 | Suppression pool temperature is high |
| R02 | 12 | 4:41<br>4:43 | Informs | | | | | | | Verbal | SR01 | 33 | Suppression pool temperature is 160° |

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# TASK DATA FORM (DESCRIPTIVE)



**PLANT IDENTIFICATION**

Plant Name: Browns Ferry

Unit Number: 1

NSSS Vendor: General Electric

A-E: Utility

TG Vendor: General Electric

CR Type: Multiple

OL Date:

**TASK IDENTIFICATION**

Operating Sequence: Anticipated transient without scram

Operating Sequence ID: 7

Operator Function: Supervise and control plant operation

Operator Sub-function: Mitigate consequences of an accident

Comments:

CUE: Procedure

Task Statement: Initiate standby liquid control in section

Task Purpose: To reduce reactor power by poison injection

INPO Task Code:

Task Sequence No.: 23

Task Duration:

Procedures: EOI-47 "Failure of reactor to scram when required"

Data Collected at: Simulator

| | | Behavior | | Object of Action | | | | | | | Communication Link | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| JOBCAT | JLOC | TIME | VERB | COMPONENT | PARAMETER | STATE | OTHER OBJECT | PLANT SYSTEM | INPO EQUIV | MEANS | RJC | RLOC | CONTENT |
| SR01 | 33 | 4:43<br>4:47 | Directs | | | | | | | Verbal | R01 | 29 | Initiate SLC |
| R01 | 29 | 4:48<br>4:53 | Positions | Pump | Power | On | | Standby Liquid Control | | Discrete Control | | | |
| R01 | 29 | 4:53<br>4:54 | Informs | | | | | | | Verbal | SR01 | 33 | SLC initiated |
| R01 | 29 | 4:55<br>4:57 | Observes | Pump | Power | On | | Standby Liquid Control | | Indicator Light | | | |
| R01 | 29 | 4:55<br>4:56 | Informs | | | | | | | Verbal | SR01 | 33 | Pump A running |
| SR01 | 33 | 4:59<br>5:02 | Directs | | | | | | | Verbal | R01 | 29 | Confirm that RWCU isolates |
| R02 | 12 | 5:01<br>5:02 | Informs | | | | | | | Verbal | SR01 | 33 | RWCU has previously isolated |

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# TASK DATA FORM (DESCRIPTIVE)



**PLANT IDENTIFICATION**

Plant Name ______

Unit Number ______

NSSS Vendor ______

A-E ______

TG Vendor ______

CR Type ______

OL Date ______

**TASK IDENTIFICATION**

Operating Sequence ______

Operating Sequence ID ______

Operator Function ______

Operator Sub-function ______

Comments ______

CUE ______

Task Statement ______

Task Purpose ______

INPO Task Code ______

Task Sequence No. ______

Task Duration ______

Procedures ______

Data Collected at: ______

| | | Behavior | | Object of Action | | | | | | | Communication Link | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| JOBCAT | JLOC | TIME | VERB | COMPONENT | PARAMETER | STATE | OTHER OBJECT | PLANT SYSTEM | INPO EQUIV | MEANS | RJC | RLOC | CONTENT |
| RO1 | 29 | 5:02 5:04 | Informs | | | | | | | Verbal | SRO1 | 33 | RWCU isolated on low reactor level |
| RO2 | 30 | 5:04 5:05 | Verifies | Valves | Position | Closed | | Reactor Water Cleanup | | Indicator Light | | | |
| RO2 | 30 | 5:05 5:06 | Verifies | Pump | Power | Off | | Reactor Water Cleanup | | Indicator Light | | | |
| SRO1 | 33 | 5:06 5:07 | Directs | | | | | | | Verbal | RO2 | 30 | Restart recirculation pumps minimum flow |
| RO1 | 29 | 5:08 5:11 | Informs | | | | | | | Verbal | SRO1 | 33 | Low level prohibits restart of recirculation pumps |

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# APPENDIX B:

# HEP WORKSHEETS (THERP)

## HEP WORKSHEET (THERP)

**Task Number:** 13
**Task Name:** Manually operate safety/relief valves before 1105 psig reactor pressure is reached.

| Task Description | Potential Error | Table No. (Item No.) | Tabled HEP | Dependence | Stress (Ref.) | Adjusted HEP | UCB-EF (Ref.) |
|---|---|---|---|---|---|---|---|
| Read pressure indication (*) | Misread analog meter | 20-10(1) | .003 | | | .003 | |
| Execute operating procedure under abnormal conditions (*) | Failure to follow procedure | 20-6(4) | .005 | High | 5 20-16(5) | .025 | 10 20-20(8) |
| Manually position valve open | Failure to turn discrete control | 20-12(8;5) | .0001 | Complete | | 0 | |
| Read pressure indication (*) | Misread analog meter | 20-10(1) | .003 | High | | .0015 | |
| Manually position valve closed | Failure to turn discrete control | 20-12(8;5) | .0001 | Complete | | 0 | |
| Read pressure indication (*) | Misread analog meter | 20-10(1) | .003 | High | | .0015 | |
| Manually position valve open | Failure to turn discrete control | 20-12(8;5) | .0001 | Complete | | 0 | |
| Manually position valve open | Failure to turn discrete control | 20-12(8;5) | .0001 | Complete | | 0 | |
| Manually position valve closed | Failure to turn discrete control | 20-12(8;5) | .0001 | Complete | | 0 | |
| Manually position valve closed | Failure to turn discrete control | 20-12(8;5) | .0001 | Complete | | 0 | |

*Omission of these actions was judged to contribute to system failure.

HEP WORKSHEET (THERP)

**Task Number:** 13 (cont.)
**Task Name:**

| Task Description | Potential Error | Table No. (Item No.) | Tabled HEP | Dependence | Stress (Ref.) | Adjusted HEP | UCB-EF (Ref.) |
|---|---|---|---|---|---|---|---|
| Read pressure indication (*) | Misread analog meter | 20-10(1) | .003 | High | | .0015 | |
| Manually position valve open | Failure to turn discrete control | 20-12(8;5) | .0001 | Complete | | 0 | |
| Manually position valve open | Failure to turn discrete control | 20-12(8;5) | .0001 | Complete | | 0 | |
| Manually position valve open | Failure to turn discrete control | 20-12(8;5) | .0001 | Complete | | 0 | |
| Manually position valve open | Failure to turn discrete control | 20-12(8;5) | .0001 | Complete | | 0 | |
| Read pressure indication(*) | Misread analog meter | 20-10(1) | .003 | High | | .0015 | |
| Manually position valve closed | Failure to turn discrete control | 20-12(8;5) | .0001 | Complete | | 0 | |
| Manually position valve closed | Failure to turn discrete control | 20-12(8;5) | .0001 | Complete | | 0 | |
| Manually position valve closed | Failure to turn discrete control | 20-12(8;5) | .0001 | Complete | | 0 | |
| Manually position valve closed | Failure to turn discrete control | 20-12(8;5) | .0001 | Complete | | 0 | |

*Omission of these actions was judged to contribute to system failure.

HEP WORKSHEET (THERP)

**Task Number:** 13 (cont.)
**Task Name:**

| Task Description | Potential Error | Table No. (Item No.) | Tabled HEP | Dependence | Stress (Ref.) | Adjusted HEP | UCB-EF (Ref.) |
|---|---|---|---|---|---|---|---|
| Read pressure indication (*) | Misread analog meter | 20-20(1) | .003 | High | | .0015 | |
| Read pressure indication (*) | Misread analog meter | 20-20(1) | .003 | High | | .0015 | |

*Omission of these actions was judged to contribute to system failure

HEP WORKSHEET (THERP)

**Task Number:** 17
**Task Name:** Manual insertion of control rods.

| Task Description | Potential Error | Table No. (Item No.) | Tabled HEP | Dependence | Stress (Ref.) | Adjusted HEP | UCB-EF (Ref.) |
|---|---|---|---|---|---|---|---|
| Execute operating procedure under abnormal conditions | Failure to follow procedure | 20-6(4) | .005 | | 5 20-16(5) | .025 | 10 20-20(8) |
| Position control rod rate in override | Turn switch in wrong direction | 20-12(5) | .0005 | High | | .0002 | |
| Read control rod position counters | Misread position indication | 20-9(4) | .003 | High | | .0015 | |
| Enable master controller | Push incorrect button | 20-12(2) | .003 | Low | | .00285 | |
| Turn control rod position in | Failure to complete continuous change | 20-12(10) | .003 | Moderate | | .0026 | |
| Execute rod sequence procedure | Failure to follow procedure | 20-6(4) | .005 | High | | .0025 | |
| Read control rod position counters | Misread position indication | 20-9(4) | .003 | Moderate | | .0026 | |
| Enable master controller | Push incorrect button | 20-12(2) | .003 | Low | | .0029 | |
| Turn control rod position in | Failure to complete continuous change | 20-12(10) | .003 | Moderate | | .0026 | |
| Read control rod position counters | Misread position indication | 20-9(4) | .003 | Moderate | | .0026 | |

HEP WORKSHEET (THERP)

**Task Number:** 17 (cont.)
**Task Name:**

| Task Description | Potential Error | Table No. (Item No.) | Tabled HEP | Dependence | Stress (Ref.) | Adjusted HEP | UCB-EF (Ref.) |
|---|---|---|---|---|---|---|---|
| Enable master controller | Push incorrect button | 20-12(2) | .003 | Low | | .0029 | |
| Read control rod position counters | Misread position indication | 20-9(4) | .003 | Moderate | | .0026 | |
| Enable master controller | Push incorrect button | 20-12(2) | .003 | Low | | .0029 | |
| Turn control rod position in | Failure to complete continuous change | 20-12(10) | .003 | Moderate | | .0026 | |
| Enable master controller | Push incorrect button | 20-12(2) | .003 | Low | | .0029 | |
| Read control rod position counters | Misread position indication | 20-9(4) | .003 | Moderate | | .0026 | |
| Enable master controller | Push incorrect button | 20-12(2) | .003 | Low | | .0029 | |
| Execute procedure step regarding mode select switch | Failure to follow procedure | 20-7(3) | .003 | High | | .0015 | |
| Position master controller to manual group select | Turn to incorrect setting | 20-12(9) | .001 | High | | .0005 | |
| Read control rod position counters | Misread position indication | 20-9(4) | .003 | Moderate | | .0026 | |

HEP WORKSHEET (THERP)

**Task Number:** 17 (cont.)
**Task Name:**

| Task Description | Potential Error | Table No. (Item No.) | Tabled HEP | Dependence | Stress (Ref.) | Adjusted HEP | UCB-EF (Ref.) |
|---|---|---|---|---|---|---|---|
| Enable master controller | Push incorrect button | 20-12(2) | .003 | Low | | .0029 | |
| Turn control rod position in | Failure to complete continuous change | 20-12(10) | .003 | Moderate | | .0026 | |
| Turn control rod position in | Failure to complete continuous change | 20-12(10) | .003 | Moderate | | .0026 | |
| Read control rod position counters | Misread position indication | 20-9(4) | .003 | High | | .0015 | |
| Enable master controller | Push incorrect button | 20-12(2) | .003 | Low | | .0029 | |
| Turn control rod position in | Failure to complete continuous change | 20-12(10) | .003 | Moderate | | .0026 | |
| Read control rod position counters | Misread position indication | 20-9(4) | .003 | High | | .0015 | |
| Enable master controller | Push incorrect button | 20-12(2) | .003 | Low | | .0029 | |
| Turn control rod position in | Failure to complete continuous change | 20-12(10) | .003 | Moderate | | .0026 | |
| Enable master controller | Push incorrect button | 20-12(2) | .003 | LOw | | .0029 | |

**Task Number:** 17 (cont.)
**Task Name:**

| Task Description | Potential Error | Table No. (Item No.) | Tabled HEP | Dependence | Stress (Ref.) | Adjusted HEP | UCB-EF (Ref.) |
|---|---|---|---|---|---|---|---|
| Position master controller to manual group select | Turn to incorrect setting | 20-12(9) | .001 | High | | .0005 | |
| Enable master controller | Push incorrect button | 20-12(2) | .003 | Low | | .0029 | |
| Position master controller to manual group select | Turn to incorrect setting | 20-12(9) | .001 | High | | .0005 | |
| Position master controller to manual group select | Turn to incorrect setting | 20-12(9) | .001 | High | | .0005 | |
| Position master controller to manual group select | Turn to incorrect setting | 20-12(9) | .001 | High | | .0005 | |
| Enable master controller | Push incorrect button | 20-12(2) | .003 | Low | | .0029 | |
| Turn control rod position in | Failure to complete continuous change | 20-12(10) | .003 | Moderate | | .0026 | |
| Read control rod position counters | Misread position indication | 20-9(4) | .003 | High | | .0015 | |
| Enable master controller | Push incorrect button | 20-12(2) | .003 | Low | | .0029 | |

**Task Number:** 17 (cont.)
**Task Name:**

| Task Description | Potential Error | Table No. (Item No.) | Tabled HEP | Dependence | Stress (Ref.) | Adjusted HEP | UCB-EF (Ref.) |
|---|---|---|---|---|---|---|---|
| Read control rod position counters | Misread position indication | 20-9(4) | .003 | Moderate | | .0026 | |
| Enable master controller | Push incorrect button | 20-12(2) | .003 | Low | | .0029 | |
| Read control rod position counters | Misread position indication | 20-9(4) | .003 | Moderate | | .0026 | |
| Turn control rod position in | Failure to complete continuous change | 20-12(10) | .003 | Moderate | | .0026 | |
| Read control rod position counters | Misread position indication | 20-9(4) | .003 | High | | .0015 | |
| Enable master controller | Push incorrect button | 20-12(2) | .003 | Low | | .0029 | |
| Turn control rod position in | Failure to complete continuous change | 20-12(10) | .003 | Moderate | | .0026 | |
| Turn control rod position in | Failure to complete continuous change | 20-12(10) | .003 | Moderate | | .0026 | |
| Read control rod position counters | Misread position indication | 20-9(4) | .003 | High | | .0015 | |
| Enable master controller | Push incorrect button | 20-12(2) | .003 | Low | | .0029 | |

**Task Number:** 17 (cont.)
**Task Name:**

| Task Description | Potential Error | Table No. (Item No.) | Tabled HEP | Dependence | Stress (Ref.) | Adjusted HEP | UCB-EF (Ref.) |
|---|---|---|---|---|---|---|---|
| Read control rod position counters | Misread position indication | 20-9(4) | .003 | High | | .0015 | |
| Turn control rod position in | Failure to complete continuous change | 20-12(10) | .003 | Moderate | | .0026 | |
| Position master controller to manual group select | Turn to incorrect setting | 20-12(9) | .001 | Moderate | | .0017 | |
| Observe master controller indicator lights | Confirm status change | 20-11(7) | Negligible | High | | 0 | |
| Position master controller to manual group select | Turn to incorrect setting | 20-12(9) | .001 | Moderate | | .0017 | |
| Position master controller to manual group select | Turn to incorrect setting | 20-12(9) | .001 | Moderate | | .0017 | |
| Position master controller to manual group select | Turn to incorrect setting | 20-12(9) | .001 | Moderate | | .0017 | |
| Position master controller to manual group select | Turn to incorrect setting | 20-12(9) | .001 | Moderate | | .0017 | |

HEP WORKSHEET (THERP)

**Task Number:** 17 (cont.)
**Task Name:**

| Task Description | Potential Error | Table No. (Item No.) | Tabled HEP | Dependence | Stress (Ref.) | Adjusted HEP | UCB-EF (Ref.) |
|---|---|---|---|---|---|---|---|
| Position control rod rate in over-ride | Turn in wrong direction | 20-12(5) | .0005 | High | | .00025 | |
| Enable master controller | Push incorrect button | 20-12(2) | .003 | Low | | .0029 | |
| Turn control rod position in | Failure to com-plete continuous change | 20-12(10) | .003 | Moderate | | .0026 | |
| Read control rod position counters | Misread position indication | 20-9(4) | .003 | High | | .0015 | |
| Enable master controller | Push incorrect button | 20-12(2) | .003 | Low | | .0029 | |
| Turn control rod position in | Failure to com-plete continuous change | 20-12(10) | .003 | Moderate | | .0026 | |
| Turn control rod position in | Failure to com-plete continuous change | 20-12(10) | .003 | Moderate | | .0026 | |
| Read control rod position counters | Push incorrect button | 20-9(4) | .003 | High | | .0015 | |
| Enable master controller | Push incorrect button | 20-12(2) | .003 | Low | | .0029 | |
| Read control rod position counters | Misread position indication | 20-9(4) | .003 | Moderate | | .0026 | |

HEP WORKSHEET (THERP)

**Task Number:** 17 (cont.)
**Task Name:**

| Task Description | Potential Error | Table No. (Item No.) | Tabled HEP | Dependence | Stress (Ref.) | Adjusted HEP | UCB-EF (Ref.) |
|---|---|---|---|---|---|---|---|
| Enable master controller | Push incorrect button | 20-12(2) | .003 | Low | | .0029 | |
| Read control rod position counters | Misread position indication | 20-9(4) | .003 | Moderate | | .0026 | |
| Read control rod position counters | Misread position indication | 20-9(4) | .003 | Moderate | | .0026 | |
| Position master controller to manual group select | Turn to incorrect setting | 20-12(9) | .001 | Moderate | | .0017 | |
| Position master controller to manual group select | Turn to incorrect setting | 20-12(9) | .001 | Moderate | | .0017 | |
| Position master controller to manual group select | Turn to incorrect setting | 20-12(9) | .001 | Moderate | | .0017 | |
| Position control rod rate in over-ride | Turn in wrong direction | 20-12(5) | .0005 | Moderate | | .0004 | |
| Turn control rod position in | Failure to com-plete continuous change | 20-12(10) | .003 | Moderate | | .0026 | |
| Enable master controller | Push incorrect button | 20-12(2) | .003 | Low | | .0029 | |

**Task Number:** 17 (cont.)
**Task Name:**

| Task Description | Potential Error | Table No. (Item No.) | Tabled HEP | Dependence | Stress (Ref.) | Adjusted HEP | UCB-EF (Ref.) |
|---|---|---|---|---|---|---|---|
| Read control rod position counters | Misread position indication | 20-9(4) | .003 | Moderate | | .0026 | |
| Enable master controller | Push incorrect button | 20-12(10) | .003 | Low | | .0029 | |
| Turn control rod position in | Failure to complete continuous change | 20-12(10) | .003 | Moderate | | .0026 | |
| Enable master controller | Push incorrect button | 20-12(2) | .003 | Low | | .0029 | |
| Turn control rod position in | Failure to complete continuous change | 20-12(10) | .003 | Moderate | | .0026 | |
| Turn control rod position in | Failure to complete continuous change | 20-12(10) | .003 | Moderate | | .0026 | |

## HEP WORKSHEET (THERP)

**Task Number:** 20
**Task Name:** Initiate Suppression Pool Cooling

| Task Description | Potential Error | Table No. (Item No.) | Tabled HEP | Dependence | Stress (Ref.) | Adjusted HEP | UCB-EF (Ref.) |
|---|---|---|---|---|---|---|---|
| Recognize torus high temperature alarm (*) | Failure to diagnose status | 20-3(2) | .1 | | | | |
| Execute operating procedure under abnormal conditions (*) | Failure to follow procedure | 20-6(4) | .005 | | 5 20-16(5) | .025 | 10 20-20(8) |
| Energize RHR pump (*) | Failure to turn discrete control | 20-12(8;5) | .0001 | High | | .00005 | |
| Energize RHR pump | Failure to turn discrete control | 20-12(8;5) | .0001 | Complete | | 0 | |
| Verify pump power on | Failure to confirm change in status lamp | 20-11(7) | Negligible | High | | 0 | |
| Verify pump power on | Failure to confirm change in status lamp | 20-11(7) | Negligible | Complete | | 0 | |
| Open RHR valve(*) | Failure to complete valve change | 20-12(10) | .003 | High | | .0015 | |
| Open RHR valve | Failure to complete valve change | 20-12(10) | .003 | Complete | | 0 | |

*Omission of these actions was judged to contribute to system failure.

HEP WORKSHEET (THERP)

**Task Number:** 20 (cont.)
**Task Name:**

| Task Description | Potential Error | Table No. (Item No.) | Tabled HEP | Dependence | Stress (Ref.) | Adjusted HEP | UCB-EF (Ref.) |
|---|---|---|---|---|---|---|---|
| Energize RHRSW pump (*) | Failure to turn discrete control | 20-12(8;5) | .0001 | High | | .00005 | |
| Energize RHRSW pump | Failure to turn discrete control | 20-12(8;5) | .0001 | Complete | | 0 | |
| Verify pump power on | Failure to confirm change in status lamp | 20-11(7) | Negligible | High | | 0 | |
| Verify pump power on | Failure to confirm change in status lamp | 20-11(7) | Negligible | Complete | | 0 | |
| Open RHRSW valve (*) | Failure to complete valve change | 20-12(10) | .003 | High | | .0015 | |
| Verify valve open | Failure to confirm change in status lamp | 20-11(7) | Negligible | Complete | | 0 | |
| Verify RHRSW pump flow (*) | Misread analog meter | 20-10(1) | .003 | High | | .0015 | |
| Verify RHR pump flow | Misread analog meter | 20-10(1) | .003 | Complete | | 0 | |
| Open RHRSW valve (*) | Failure to complete valve change | 20-12(10) | .003 | High | | .0015 | |

*Omission of these actions was judged to contribute to system failure.

HEP WORKSHEET (THERP)

**Task Number:** 22
**Task Name:** Verify Conditions Exist for Initiation of SLC Injection.

| Task Description | Potential Error | Table No. (Item No.) | Tabled HEP | Dependence | Stress (Ref.) | Adjusted HEP | UCB-EF (Ref.) |
|---|---|---|---|---|---|---|---|
| Execute operating procedure under abnormal conditions (*) | Failure to follow procedure | 20-6(4) | .005 | | 5<br>20-16(5) | .025 | 10<br>20-20(8) |
| Read procedure step to verify five or more adjacent rods are not inserted past 06 position | Omission of step procedures | 20-7(3) | .003 | Moderate | | .0026 | |
| Verify five or more adjacent rods are full out (*) | Misread digital displays | 20-11(1) | .001<br>.001<br>.001<br>.001<br>.001 | Moderate | | .0009<br>.0009<br>.0009<br>.0009<br>.0009 | |
| Read procedure step to verify whether reactor water level can be maintained | Omission of step in procedures | 20-7(3) | .003 | Moderate | | .0026 | |
| Verify reactor water level (*) | Misread quantitative display | 20-10(1) | .003 | Moderate | | .0026 | |
| Read procedure step to verify suppression pool temperature greater than 110°F | Omission of step in procedures | 20-7(3) | .003 | Moderate | | .0026 | |

*Omission of these actions was judged to contribute to system failure.

HEP WORKSHEET (THERP)

**Task Number:** 22(cont.)
**Task Name:**

| Task Description | Potential Error | Table No. (Item No.) | Tabled HEP | Dependence | Stress (Ref.) | Adjusted HEP | UCB-EF (Ref.) |
|---|---|---|---|---|---|---|---|
| Verify suppression pool temperature (*) | Misread quantitative display | 20-10(1) | .003 | Moderate | | .0026 | |

HEP WORKSHEET (THERP)

**Task Number:** 23
**Task Name:** Initiate Injection of SLC Tank

| Task Description | Potential Error | Table No. (Item No.) | Tabled HEP | Dependence | Stress (Ref.) | Adjusted HEP | UCB-EF (Ref.) |
|---|---|---|---|---|---|---|---|
| Read procedure step to initiate SLC injection | Omission of step in procedures | 20-7(3) | .003 | Moderate | | .0026 | |
| Turn SLC pump power on (*) | Failure to completely turn key lock switch | 20-12(10) | .003 | Moderate | | .0026 | |
| Verify pump power on | Misread indicator light | 20-11(7) | .003 | Moderate | | .0026 | |
| Read procedure step to verify RWCU isolation | Omission of step in procedures | 20-7(3) | .003 | Moderate | | .0026 | |
| Verify RWCU isolation | Misread valve indicator lights | 20-11(8) | Negligible | | | - | |

# APPENDIX C:

# QUALITATIVE ASSESSMENT OF UNCONVENTIONAL EMERGENCY RESPONSE (UERs)

## C.1 Introduction

This appendix provides qualitative assessments of the five Unconventional Emergency Responses (UERs) identified in the table-top demonstration of a severe ATWS perturbation. The presentation follows the format described in Section 3.2.4 concerning the modeling of UERs. The following assessments are intended to be representative examples based on the operation and engineering expertise of some of the authors. The actions associated with the UERs are potential responses to the multiple failures postulated in the demonstration ATWS scenario, and other responses might be more appropriate depending upon accident conditions, especially for more severely degraded conditions. These responses are only used to demonstrate the utility of the FOAM model in accident management and should not be interpreted as a recommendation for emergency response to an ATWS.

The problems confronting the operator in the ATWS scenario are that he must mitigate fuel damage before extensive failure occurs and he must protect the primary containment from damage. Once fuel damage occurs, the containment will begin to show an increase in radioactive contaminants from the release of reactor coolant through the SRVs. The containment, therefore, becomes a vital barrier to the release of radioactive contaminants to the environment following fuel damage. The five tasks that the operator might perform to prevent further fuel damage and preserve containment integrity are:

1. Recover Control Rod Insertion
2. Recover SLCS Initiation
3. Initiate PSP Spray
4. Replenish PSP Volume
5. Open One MSIV

Each of these tasks will be examined for the effect that the response has on the state of the plant should the response succeed or fail. Assessments will include the following:

1. Alarms and Cues
2. Decision Criteria
3. Actions
4. Consequences of Actions

## C.2 Recover Control Rod Insertion

The most dominate failure modes for control rod insertion based on the Tennessee Valley Authority's (TVA's) Probabilistic Risk Assessment (PRA) for Browns Ferry Unit One (BF1) are possible faults in the venting of the hydraulic control rod drive mechanisms and mechanical binding of the mechanisms. The validity of the "hydraulic lock" problem was demonstrated by an event on BF3 during a planned shutdown. In this event, the scram discharge volume (SDV) on one bank of hydraulic control units was not completely drained when the reactor was manually scrammed. The result was that the control rods in the affected bank did not completely insert and the reactor remained critical at approximately 5% thermal power. There have also been cases of mechanical binding of control rods in BWRs. However, the occurrence of multiple control rod failures has not been a problem in the past. The large number of control rods (185 in BF1) minimizes the effect of stuck control rods for achieving subcriticality. Another possible source of failure is the link between the reactor protection system (RPS) and the control rod drive hydraulic system. A common fault in the scram relays and breakers could prevent venting of the valve operators for the scram valves and the SDV vent and drain valves. Although this has not been a problem in BWRs, such a failure has occurred in a PWR due to improper maintenance of scram breakers.

C.2.1 Alarms and Cues

a. Control rods withdrawn.

b. Scram condition alarmed.

c. Failure of automatic and manual scram.

d. Some individual rod scram lights may not be lit.

e. Failure of rods to insert by signal from reactor manual control.

f. Power level indicated on power range neutron monitors.

C.2.2 Decision Criteria

a. Possible failure of RPS to deenergize.

b. Possible hydraulic lock in the control rod drive hydraulics system.

c. Possible rod binding.

d. Possible binding of scram pilots and backup scram valves.

C.2.3 Actions

a. Operator places reactor mode switch to SHUTDOWN (scram signal).

b. Operator depresses scram buttons.

c. Operator requests auxiliary operator to scram control rods individually at scram time test panel.

d. Operator requests auxiliary operator to isolate and vent the scram air header.

e. Operator requests auxiliary operator to place the RPS test switches in TEST.

f. Operator begins manual control rod insertion using RMCS.

g. Operator attempts to reset the scram to drain the SDV.

C.2.4 Consequences of Actions

a. If scram failure was due to automatic scram failure, placing the mode switch in SHUTDOWN deenergizes the manual scram channel in each trip system.

b. Depressing the scram buttons will have the same effect as C.2.4.a.

c. If the scram failure is the result of RPS relay failure, the individual rod scram test switches may insert the rods since the test switch is independent of the automatic trip circuit.

d. If the scram actions above fail and the air supply to the scram valves is not interrupted, the action of isolating and venting the scram air header will cause the scram valves to open and the SDV vent and drain valves will close (rods scram).

e. Placing the RPS test switches in TEST is another attempt to deenergize RPS.

f. If the scram inlet and outlet valves cannot be opened, the operator may begin inserting control rods using the reactor manual control system. If the scram failure is the result of hydraulic lock or rod binding, this may not be successful.

g. If the rods do not insert because of hydraulic lock, resetting the scram will open the SDV vent and drain valves so that the SDV will drain to the equipment drain pump. Once the draining is complete, the operator may attempt to scram the reactor again. Resetting the scram may require extraordinary actions such as bypassing all scram signals.

**C.3 Recover Standby Liquid Control (SLC) System Injection**

The most dominant failure modes for actuation of the standby liquid control (SLC) system, as based on TVAs PRA for BF1, are valve misalignment or a stuck open relief valve. The valve alignment is changed for the required surveillance testing that is frequently performed during normal reactor power operation. If the valve alignment is not returned to the normal configuration following testing, the system will not perform its intended function when initiated. In addition, the discharge piping of each SLC pump is equipped with a relief valve to prevent piping damage from overpressure. The SLC pumps are positive displacement pumps that could burst the piping if started without a complete discharge path. The relief valves discharge back to the storage tank when open to relieve the system pressure. Should one of these relief valves stick open, the SLC pump discharge will simply recirculate back to the tank.

C.3.1 Alarms and Cues

a. Criteria listed in EPGs for initiation of the SLC system are met:

   1. Five or more adjacent control rods not inserted past position 06 and reactor water level cannot be maintained or suppression pool temperature cannot be maintained below 110 degrees F.

   2. Thirty or more control rods not inserted past position 06 and reactor water level cannot be maintained or suppression pool temperature cannot be maintained below 110 degrees F.

b. Operator has attempted to start the SLC system.

c. The white flow indicating light on the 9-5 panel is not lit.

d. SLC tank level does not decrease.

e. SLC pump runlight not lit.

C.3.2 Decision Criteria

a. Check valve alignment.

b. Check relief valve position.

c. Check squib valve continuity.

d. Check pump motor energized.

e. Attempt to align some other injection system to inject poison solution.

C.3.3 Actions

a. Realign valves to open the injection path.

b. Close relief valve.

c. Start other SLC pump.

d. Attempt to fire squib valves by turning the control switch to the other position.

e. Mix sodium-pentaborate solution in the condensate storage tank and inject with HPCI or RCIC.

C.3.4 Consequences of Actions

a. If the test path was aligned, placing the system in the correct valve alignment will restore the injection capabilities of the system. If the manual isolation valve inside containment is closed, the system will be lost for the duration of the accident.

b. If the relief valve was stuck open and the auxiliary operator succeeded in closing the valve, the injection capability of the system will be restored.

c. Starting the other SLC pump will restore the system's capabilities if the failure is a faulty pump, pump motor or stuck open relief valve.

d. If the cause of the failure is a failure of the squib valves to fire, turning the control switch to the other position will send another activation signal to the valves and start the other pump.

**C.4 Initiate Pressure Suppression Pool Spray**

The challenge to the integrity of the containment during an ATWS is from the frequent SRV operation because of high reactor pressure. The containment pressure will rapidly increase if the operator does not take action to lower the suppression pool temperature and pressure. During the early stages of the accident the operator will start suppression pool cooling using the test path for the RHR system. However, the RHR system can only remove enough heat to sustain pool temperature when the reactor power level is approximately 5% or less. If the operator cannot maintain reactor thermal power below 5%, the pool temperature and pressure will steadily increase to the failure point. In addition, as the suppression chamber pressure increases, the drywell pressure will also increase as the containment vacuum breakers between the torus and drywell open to equalize the pressure. If the pressure increase is not mitigated, the entire containment is in danger of failure on overpressure. More efficient cooling of the suppression pool may be achieved by using the RHR system to spray suppression pool water into the drywell and torus atmosphere. This has the effect of condensing the steam in the containment atmosphere and lowering the containment temperature and pressure.

C.4.1 Alarms and Cues

a. Suppression pool temperature greater than 110 degrees F.

b. Torus high temperature alarm.

c. Containment high pressure alarm.

d. Torus pressure above spray initiation pressure.

e. Drywell temperature high.

C.4.2 Decision Criteria

a. Suppression pool cooling in operation.

b. Reactor cannot be scrammed.

c. Suppression pool temperature cannot be maintained below the Heat Capacity Temperature Limit and Pressure Suppression Limit.

d. Reactor cannot be depressurized.

e. Drywell temperature cannot be maintained below design temperature.

f. Suppression chamber pressure approaching 17.4 psig and suppression pool level is below the spray nozzles.

C.4.3 Actions

a. Place containment spray valve select switch in SELECT position.

b. If reactor water level is below 2/3 core coverage, place containment spray in OVERRIDE.

c. Shut down (or verify already shut down) reactor recirculation pumps and drywell cooling fans.

d. Start (or verify already running) RHR pumps in loop to be used for spray.

e. Open inboard and outboard drywell spray valves.

f. Open torus valve and torus spray valves.

C.4.4 Consequences of Actions

a. Primary containment temperature decreases.

b. Primary containment pressure decreases.

c. Drywell electrical equipment damaged.

### C.5 Replenish PSP Volume

The failure to achieve proper suppression pool cooling through the torus cooling mode of RHR or torus spray will require extraordinary action to preserve containment integrity. The need for additional cooling water may be satisfied by pumping the hot torus water to the main condenser or radwaste system and replenishing the pool inventory either by pumping water from the condensate storage tank (the normal method) or by a more drastic measure of pumping river water to the torus using the RHR service water system. The steps for reducing and increasing the PSP level under normal conditions are outlined in the BFNP Operating Instruction 01-74.

An alternate method of replenishing the PSP inventory that is more desirable than the method described above would be to crosstie the RHR system in Unit 1 with the RHR system in Unit 2. The RHR crosstie capability at Browns Ferry is shown in Figure C.1.

C.5.1 Alarms and Cues

a. Torus high level alarm.

b. Torus level increasing from SRV operation.

C.5.2 Decision Criteria

a. Suppression pool level cannot be maintained below load limit.

b. Reactor cannot be depressurized.

c. RCIC injection cannot be stopped.

d. Water from a source inside containment cannot be injected.

C.5.3 Actions

a. To lower torus level by normal method:

1. Shut down RHR pumps in one loop.
2. Dispatch a Health Physics technician to measure radiation levels and determine stay time in RHR pump room.
3. Dispatch auxiliary operators to the corner room to perform the following:
    - i. Align RHR flush pump to take a suction from the RHR loop upstream of the pump torus suction valve (Note 1).
    - ii. Align manual valves on discharge of the flush pump to discharge to the main condensor (Note 2).
    - iii. Start the flush pump.
4. The control room operator throttles open the motor operated blowdown valve.
5. The control room operator monitors torus level decrease.
6. When torus level reaches -6", the auxiliary operator closes the motor operated valve and stops the flush pump.

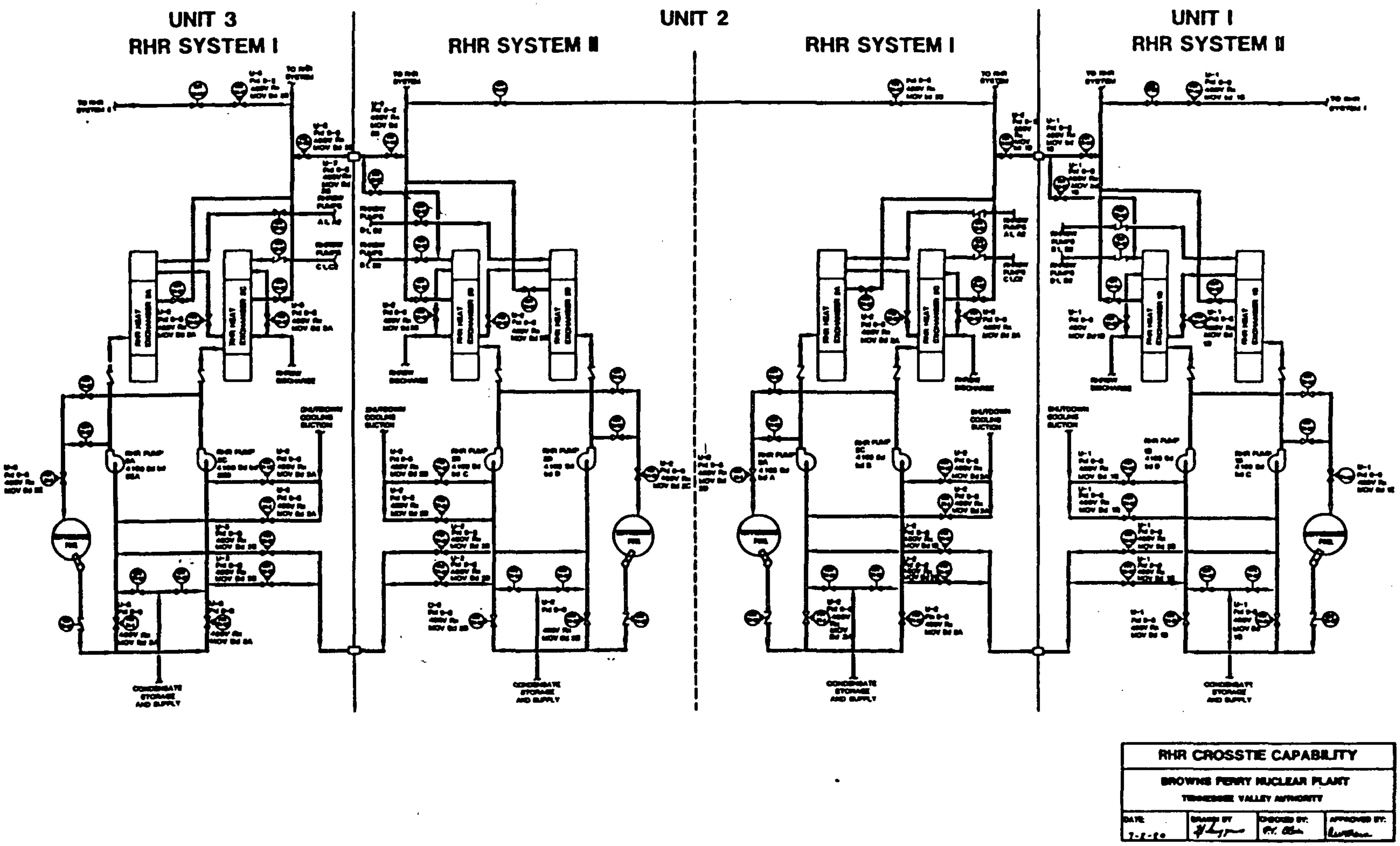


**Figure C.1. RHR crosstie capability.**

b. To lower torus level by unit cross-tie:

1. Place Unit 2 RHR System II in suppression pool cooling.
2. Close Unit 1 RHR System II torus valve.
3. Open Unit 1 unit crosstie valve FCV-74-101.
4. Open Unit 2 unit crosstie valve FCV-74-100.
5. Open Unit 2 RHR System I torus valve and test valve.
6. When Unit 1 torus level reaches -6", close Unit 2 torus and test valves on RHR System I.
   NOTE: It is not necessary to run the RHR pumps on Unit 2 torus and test valves on RHR System I.

c. To raise torus level by normal method:

1. Shut down core spray pumps if running.
2. Open core spray system I or II test valve.
3. Monitor torus level increase.
4. Close test valve when torus level reaches -1".

d. To raise torus level by the alternate method:

1. Shut down the pumps in RHR System II.
2. Start (or verify running) RHRSW pump D2.
3. Open the two Standby Coolant Supply cross-connect valves from RHRSW to RHR.
4. Open (or verify open) the torus valve.
5. Open (or verify open) the RHR test valve.
6. Monitor torus level increase.

e. To raise torus level by unit crosstie:

1. Shut down the RHR pumps in Unit 1 RHR System II.
2. Start Unit 2 RHR pumps in System I.
3. Open torus and test valves on Unit I RHR System II.
4. Stop Unit 2 RHR pumps when Unit I torus level reaches -1".

C.5.4 Consequences of Actions

a. Half of the torus cooling capability is lost during torus level reduction.

b. Pressure suppression and steam quenching capabilities are diminished during torus level reduction.

c. The large volume of water discharge from the torus precludes the use of the radwaste system. The torus water must be discharged to the main condenser affecting the water quality of the condensate/feedwater system and the CST.

d. Replacing heated torus water with cooler water from the keep fill system, the Unit 2 torus, or RHRSW improves the suppression pool's steam quenching capabilities.

e. Lowering the water level in the torus reduces the torus load.

f. Raising the pool level with river water makes the torus water unsuitable for injection except as a last resort.

g. The river water increases the quantity of radioactive waste that must be processed following the accident.

Note 1: Aligning the flush pump suction requires the auxiliary operator to verify that five valves are closed and to open one valve manually.

Note 2: Aligning the flush pump discharge requires the auxiliary operator to verify that two valves are closed and to open one valve manually.

**C.6 One Main Steam Isolation Valve (MSIV)**

When it becomes apparent that the reactor cannot be scrammed and that a loss of the suppression pool is inevitable, the operators must take extraordinary action to prevent further fuel failure and/or containment failure. The restoration of the normal heat sink, the main condensor, will provide pressure control for a reactor power level of 25% or less. An additional 5% steam flow may be consumed if steam driven auxiliaries are used. If fuel damage is severe, the operator must be cognizant of increasing radioactive release rates to the environment. Therefore, the operator will only open the MSIV as a last resort and only if limited fuel damage has occurred.

One additional benefit to opening the MSIV is that the feedwater system may be placed in service. This will eliminate the need of using the condensate storage tank to maintain coolant inventory in the reactor. The HPCI and RCIC systems may be shut down, thereby stopping the accumulation of water in the containment. Also, using the feedwater system will provide coolant at a higher temperature, limiting the power spikes that accompany cold water injection.

C.6.1 Alarms and Cues

a. Reactor power greater than 5% (boron injection required)

b. Reactor pressure high alarm

c. Reactor pressure greater than 1105 psig

### C.6.2 Decision Criteria

a. Reactor cannot be scrammed.

b. Suppression pool load limit and heat capacity limit are being exceeded.

c. Reactor power above RHR torus cooling capabilities.

d. Main condenser available.

e. No fuel damage is suspected.

f. No main steam line break is suspected.

### C.6.3 Actions

a. Control room operator aligns the condensate system for recirculation mode.

b. Start turbine seals on auxiliary steam.

c. Start steam packing exhauster.

d. Start mechanical vacuum pump(s) and establish greater than 7" Hg vacuum in the main condenser.

e. Request instrument mechanic to jumper all GROUP I isolation signals and GROUP I seal-in.

f. Open main steam line drain outboard restricting orifice bypass valves.

g. Open outboard MSIV on one steam line.

h. Open main steam line drain containment isolation valves.

i. Close and hold closed the MSL drain to the main condenser.

j. Pressurize the MSL down stream of the MSIV until d/p across the MSIV is less than 150 psig.

k. Open inboard MSIV.

l. Raise EHC pressure setpoint to 920 psig.

m. Swap turbine seals over to nuclear steam.

n. Align off-gas system for service.

o. Start steam jet air rejectors on nuclear steam.

p. Shut down the mechanical vacuum pump(s).

q. Warm a reactor feedwater pump turbine.

r. Once the RFPT is warmed, use the RFP to feed the reactor.

s. Shut down the RCIC system when the feedwater system is feeding the reactor.

t. Augment pressure control using SRVs.

C.6.4 Consequences of Actions

a. Load on the torus and drywell is reduced.

b. Stable pressure control is achieved.

c. More stable level control is achieved.

d. A higher power level can be maintained for an indefinite period of time.

e. Injection of CST water is not necessary; the coolant inventory inside the containment does not continue to increase.

f. MSIV closure on high MSL radiation or MS break will not occur with the jumpers in place.

g. Possible release of radioactive material from the plant stack.

h. Possible damage to the main condenser with full bypass valve flow.

i. Possible release of radioactive material into the turbine building atmosphere from the main condenser or turbine seals.

# APPENDIX D:

# PATHWAYS FOR THE RELEASE OF RADIONUCLIDES FROM A BWR DURING AN ANTICIPATED TRANSIENT WITHOUT SCRAM

## D.1. Introduction

Large scale nuclear generating plants are designed and operated to minimize the effects of an accident on the health and safety of the public should an accident occur during operation. The safety analysis should prove that sufficient barriers exist to limit radioactive releases to within 10CFR100 limits for any credible accident. Boiling Water Reactor (BWR) power plants use multiple barriers that are designed to meet the 10CFR100 specifications for radioactive release during a large Loss of Coolant Accident (LOCA). The most commonly used barriers in BWR's are:

1. Nuclear Fuel
2. Fuel Cladding
3. Reactor Coolant
4. Primary System Boundary
5. Primary Containment
6. Secondary Containment
7. Secondary Containment Ventilation System

Maintenance of these seven barriers is required by plant's Technical Specifications during power operation to minimize the consequences of an accident, should one occur. These barriers to radioactive release will be examined to determine possible pathways for release during an ATWS. The scenario for the ATWS event assumes that the main steam isolation valves (MSIVs) close and the reactor fails to scram. Assuming core damage occurs, the release of radioactive materials to the environment requires breaching or penetrating each of the seven barriers. The accompanying diagrams illustrate the possible pathways for a radioactive release. This discussion will describe these barriers and pathways, and the information available to the control room operators for determining if a barrier has been violated. In addition, the possible actions the operator may take to mitigate the breach of each of these barriers will be discussed briefly.

There are two classes of radionuclides that are present in a nuclear power plant; activation products and fission products. The activation products are made in the reactor by exposing materials that normally are not radioactive to neutrons. The materials absorb the neutrons to form radioactive isotopes of the same element. Fission products are made in the reactor by the fission of the uranium and plutonium fuels. A majority of the fission products are radioactive and decay by beta emission.

There are two groups of activation products; corrosion activation products and coolant activation products. Table D.1 lists the major corrosion activation products and Table D.2 lists the coolant activation products. The following tables were taken from the General Electric Station Nuclear Engineer's Manual.

**TABLE D.1**

**ACTIVATED CORROSION PRODUCTS**

| Nuclide | Half-Life |
|---|---|
| Cr-51 | 27.8 days |
| Mn-54 | 312 days |
| Mn-56 | 2.58 hours |
| Fe-59 | 45 days |
| Co-58 | 71 days |
| Co-60 | 5.24 years |
| Co-64 | 12.9 hours |
| Zn-65 | 234 days |
| W-187 | 24.0 hours |

TABLE D.2

ACTIVATION PRODUCTS OF WATER

| Nuclide | Half-Life |
|---|---|
| N-16 | 7.1 seconds |
| 0-19 | 29 seconds |
| N-13 | 10 minutes |
| F-18 | 110 minutes |
| H-3 (tritium) | 12.33 years |

Fission products are divided into three classes; fission gases, iodines, and soluble and insoluble fission products (often referred to as particulates). The fission gases are listed in Table D.3, the iodines are listed in Table D.4 and the particulates are listed in Table D.5.

TABLE D.3

FISSION GASES

| Nuclide | Half-Life |
|---|---|
| Xe-138 | 14.2 minutes |
| Kr-87 | 76 minutes |
| Kr-88 | 2.79 hours |
| Kr-85m | 4.4 hours |
| Xe-135 | 9.16 hours |
| Xe-133 | 5.27 days |
| Xe-135m | 15.7 minutes |
| Kr-85 | 10.76 years |

TABLE D.4

FIVE IODINE ISOTOPES WITH HALF-LIVES GREATER THAN 85 SECONDS

| Nuclide | Half-Life |
|---|---|
| I-134 | 52.3 minutes |
| I-132 | 2.28 hours |
| I-135 | 6.7 hours |
| I-133 | 20.8 hours |
| I-131 | 8.06 days |

TABLE D.5

FISSION GAS DAUGHTER PARTICULATES

| Nuclides | Half-Life |
|---|---|
| Rb-88 | 17.7 minutes |
| Cs-138 | 32.2 minutes |
| Sr-89 | 50.8 days |
| Sr-90 | 30 years |
| Sr-91 | 9.67 hours |
| Sr-92 | 2.69 hours |
| Ba-139 | 83.2 minutes |
| Ba-140 | 12.8 days |
| Ce-141 | 32.5 days |
| Ce-144 | 284 days |

The discussion of the release pathways will concentrate on the fission products only. The activation products are always present and comprise only a small fraction of the radioactivity in the reactor. The greatest hazard in accident mitigation is, by far, from the release of fission products. Also, the reactor operator has no direct means of controlling the presence of activation products. Activation products are controlled by maintaining reactor coolant chemistry within certain specifications.

## D.2. Description of Fission Product Barriers

### D.2.1. Reactor Fuel

The reactor fuel is the first barrier to the release of radioactive material, even though it is itself the source of radioactivity. A majority of the fission products are retained in the fuel lattice under normal conditions. However, fission product gases can escape from the fuel with relative ease.

The fuel used in commercial light water reactors is slightly enriched (with 2% to 4% U-235) uranium oxide ($UO_2$). During operation some of the U-238 is converted to plutonium. The ceramic $UO_2$ is fabricated into cylindrical pellets, and then stacked in zirconium alloy tubes to form fuel rods. Each pellet has a diameter of 0.416 inch and a length of 0.5 inch. The active fuel length of a fuel rod is 12 feet or 12.5 feet, depending on the fuel type. The fuel pellets are held in place by a plenum spring located in the top of each fuel rod. The plenum of the top of each fuel rod allows for the collection of fission gases released from the pellets during operation.

### D.2.2. Fuel Cladding

The fuel cladding is tubing made of zircaloy-2 (approximately 92% zirconium with traces of tin, iron, chromium, and nickel). The tubing has a wall thickness 0.032 inch, which is thick enough to make the tubing free standing and able to withstand reactor pressures without collapsing. The fuel pellets are stacked inside the cladding, and the end plugs are welded to seal the cladding against leakage of fission products.

The distinction between cladding defects and cladding failure is the origin of the breach point. A cladding defect is a manufacturer's flaw, and a failure is induced during operation. Fabrication standards limit the size of defects that are acceptable for use in the reactor. Microscopic defects are present in the metal regardless of the quality standards of the manufacturing process. Cladding failures, however, should not occur with proper operation of fuel. The onset of cladding failure is the result of overheating, oxidation, or corrosion, or any combination of the three.

### D.2.3. Reactor Coolant

Although the reactor coolant is not a physical barrier to fission product release (in fact it is a transport medium) it is considered an indirect barrier because it is necessary for maintenance of other barriers. The presence of coolant is necessary to preserve the integrity of the fuel, the fuel cladding, and the primary system boundary.

In a BWR, the coolant is ordinary water of high purity. The coolant must be free of dissolved solids and gases to limit corrosion and activation. Water for the reactor coolant system is taken from the river (or well water at some plants) and purified by the demineralized water system. The demineralized water is held in a clean storage tank until makeup in the reactor coolant system is required. Surplus water that has been used in the reactor coolant system is stored in a separate tank called the Condensate Storage Tank (CST). Two standby systems (RCIC and HPCI) use the CST for their supply of reactor makeup coolant. In addition to the standby systems, the Control Rod Drive (CRD) hydraulic system takes a suction from the CST to supply high presure water to the control rod drive mechanisms.

The pressure suppression pool also provides a large source of standby coolant to the reactor. The low pressure ECCS are normally lined up to take suction from the torus. Also, the high pressure systems that take a suction from the CST can be lined up to take a suction from the torus when the CST is unavailable.

In the unlikely event that both the CST and suppression pool are unable to supply sufficient reactor coolant, the Residual Heat Removal Service Water (RHRSW) system can be cross-connected to the RHR system to flood the reactor or cool the containment. To inject RHRSW into the reactor, however, the reactor pressure must be low (less than 150 psig). This capability to inject river water into the reactor provides an unlimited supply of reactor coolant during an emergency.

### D.2.4. Primary System Boundary

The primary system boundary is the reactor pressure vessel, the recirculation system piping, the main steam system piping, and various interconnecting piping inside primary containment. The reactor vessel and all connecting piping is made of low carbon alloy steel. To limit corrosion of the steel, the cylindrical portion of the vessel and the bottom head are clad with 0.125 inch stainless steel overlay. All penetrations into the vessel and primary system piping use full penetration welds (with the exception of the control rod drive housing stub tubes and pipes less than 2 inches in diameter). Penetrations into the vessel in the active core region are avoided due to neutron embrittlement of the metal in this region.

The reactor vessel has a design pressure of 1250 psig (safety limit 1375 psig). The normal operating pressure at rated power is 1005 psig. To ensure the safety limit is not exceeded, a sufficient number of Safety/Relief Valves (S/RV) are installed on the main steam lines to provide greater than 80% steam flow following a main steam line isolation.

The technical specifications for the primary system boundary require the operator to limit the rate of temperature change of the vessel and piping to less than 100°F/hour. This requirement is established to minimize thermal stresses on the vessel and piping induced by temperature gradients between the inner and outer surfaces of the metal.

### D.2.5. Primary Containment

The primary containment is a low leakage barrier to fission product release that is designed to absorb the energy released from the Primary System during a Design Basis LOCA. The containment is a steel pressure vessel enclosed in reinforced concrete. Mark I containments are constructed in a drywell and torus shaped suppression pool arrangement.

The drywell contains the reactor and the reactor recirculation system. Piping for the main steam system, reactor feedwater system, ECCS, and reactor auxiliaries penetrate the containment wall. Most pipes that penetrate the containment wall have double isolation valves that close automatically in an emergency. There are also a number of pipes and ducts that penetrate the contanment wall that do not connect to the primary system boundary, i.e., component cooling water and ventilation systems. These systems may not be required to have double isolation valves, and the isolation valves that are installed may be outside containment.

The torus is connected to the drywell by eight large diameter vent pipes. The primary purpose of the torus is to provide pressure suppression during the Design Basis LOCA preventing the containment internal pressure from exceeding the design pressure for the steel pressure vessel. The suppression pool contains a sufficient volume of water to condense the steam released from the reactor during a LOCA. The suppression pool water also provides a source of standby reactor coolant for operation of the ECCS, and provides a heat sink for condensing steam discharged through the main steam safety/relief valves.

## D.2.6. Secondary Containment

The Secondary Containment is a barrier surrounding the Primary Containment that limits ground level release of radioactive material during a Design Basis LOCA. While Primary Containment is inaccessible during normal operation, the Secondary Containment is accessible.

There are basically two reasons for using a primary and secondary containment instead of a large primary containment. For normal operations, certain reactor auxiliary equipment must be accessible for maintenance. For accident conditions, the secondary containment provides an additional barrier to fission product release.

Secondary Containment is divided into zones, one reactor zone for each unit and a common refuel zone. Each zone is equipped with a ventilation system to control internal pressure and supply outside air to the building. Each ventilation system is composed of two supply fans with a modulating inlet damper, and two exhaust fans. The supply fans, the modulating damper, and the exhaust fans work together to regulate the internal pressure of secondary containment at 0.25 inch water less than the external pressure. This ensures that any air leakage is into the building.

## D.2.7. Secondary Containment Ventilation Systems

The secondary containment ventilation systems provide a means of elevated release of the reactor building atmosphere. The normal ventilation systems do not filter the air prior to discharge since the building atmosphere is usually clean. To prevent the release of fission products through the ventilation exhaust, each exhaust duct is equipped with Geiger-Muller detectors to monitor for airborne radioactivity.

When high radiation is detected in the ventilation exhaust, the normal ventilation system in the effected zone shuts down, and the Standby Gas Treatment System (SGT) starts to filter the building atmosphere prior to release through the plant stack. These same automatic actions will occur if high drywell pressure or low reactor water level is sensed.

The SGT system provides filtered elevated release of the secondary containment atmosphere during emergency conditions. The building atmosphere is filtered for particulates and halogens using two high efficiency (HEPA) filters, and a charcoal filter. The filter train exhaust is then discharged through the plant stack for mixing in the atmosphere. By discharging through the plant stack there is more dilution and better distribution of the plume than can be achieved using the ventilation exhaust stack. There are three SGT trains for the BF plant, each train rated at 9,000 SCFM.

## D.3. Pathways for Release of Gaseous and Airborne Particulate Radionuclides in a BWR During ATWS

Gaseous and airborne particulate radionuclides are retained through a series of plant barriers in a BWR. These barriers and their potential breach points associated with ATWS are represented in Figure D.1.

### D.3.1. Reactor Fuel

Fission products escaping from the fuel pellet are normally contained within the fuel cladding. There is a 0.0045 inch gap between the pellets and cladding. If the cladding is not intact, this gas gap will fill with coolant and fission products will be released from the fuel pellets directly into the reactor vessel.

The release rate of fission products from the fuel pellets is accelerated by overpowering or overheating of the fuel. Since the fuel is a ceramic material, high temperatures will cause the ceramic to crack. Excessive cracking of the pellets provides a larger than normal cross sectional area for the escape of the fission gases.

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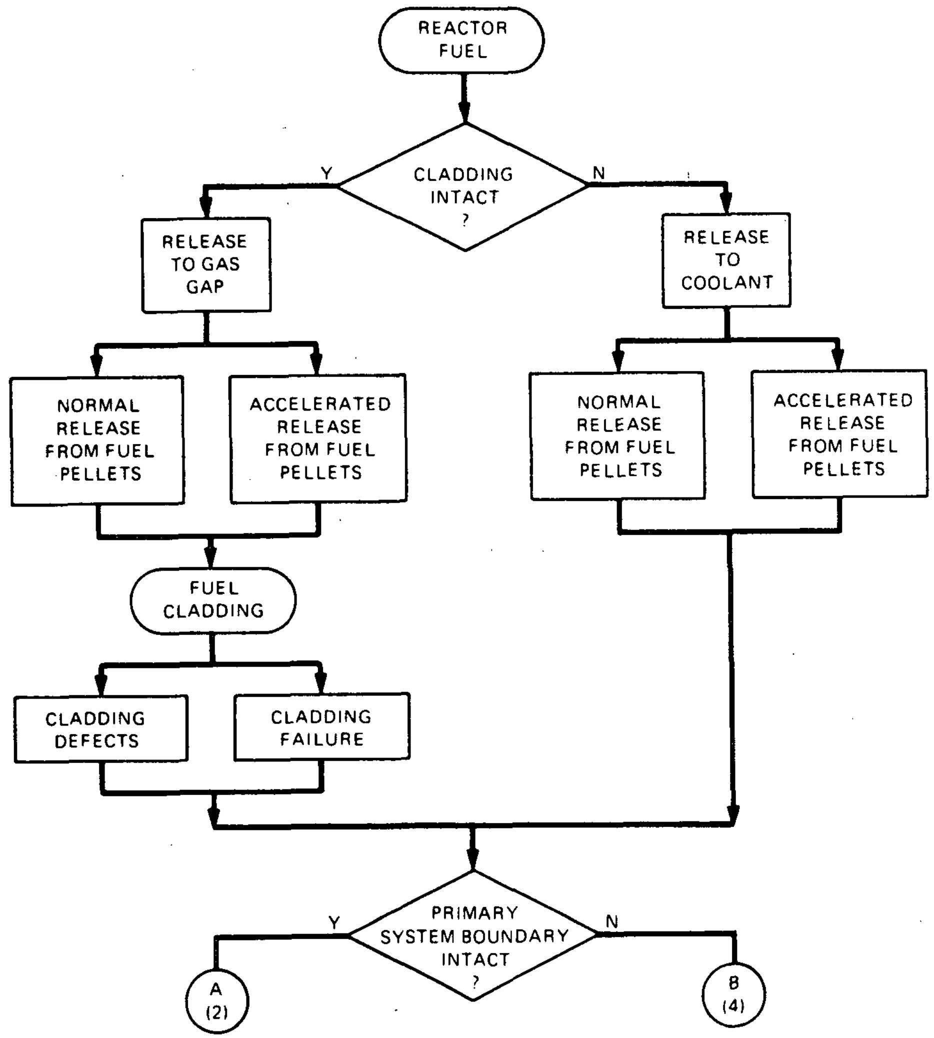


**Figure D.1. Pathways for the release of gaseous and airborne particulate radionuclides from a BWR during ATWS.**

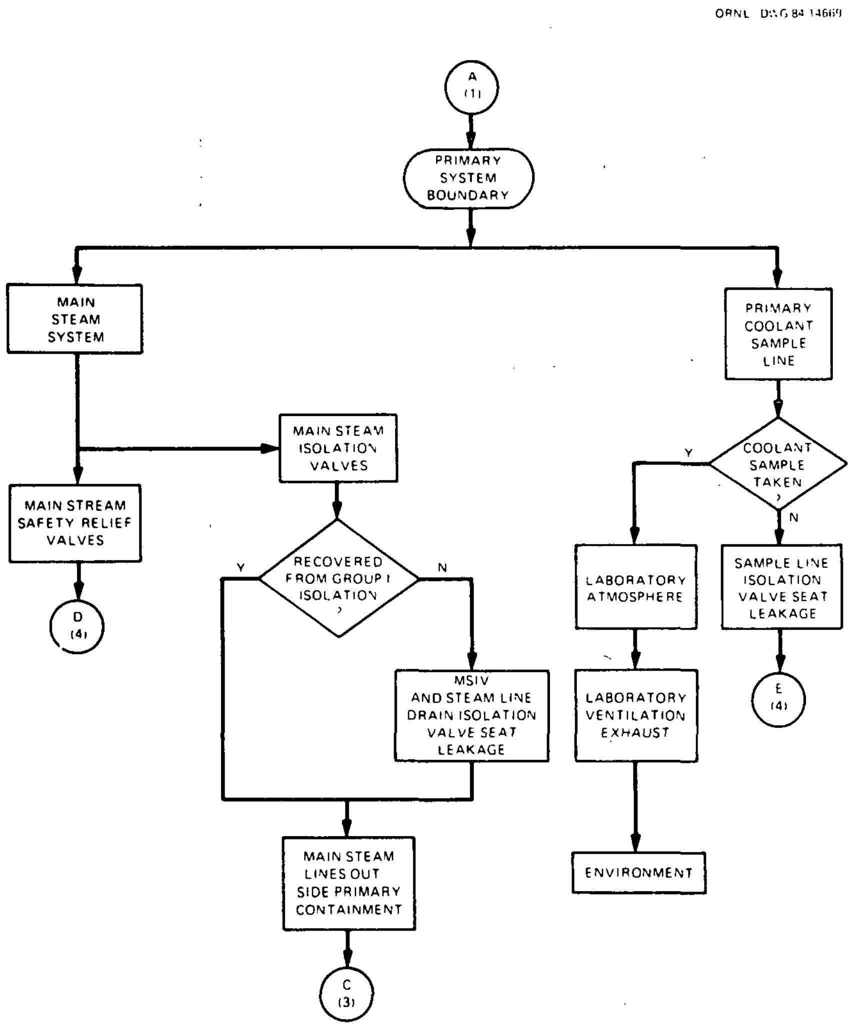


**Figure D.1 (cont.).**

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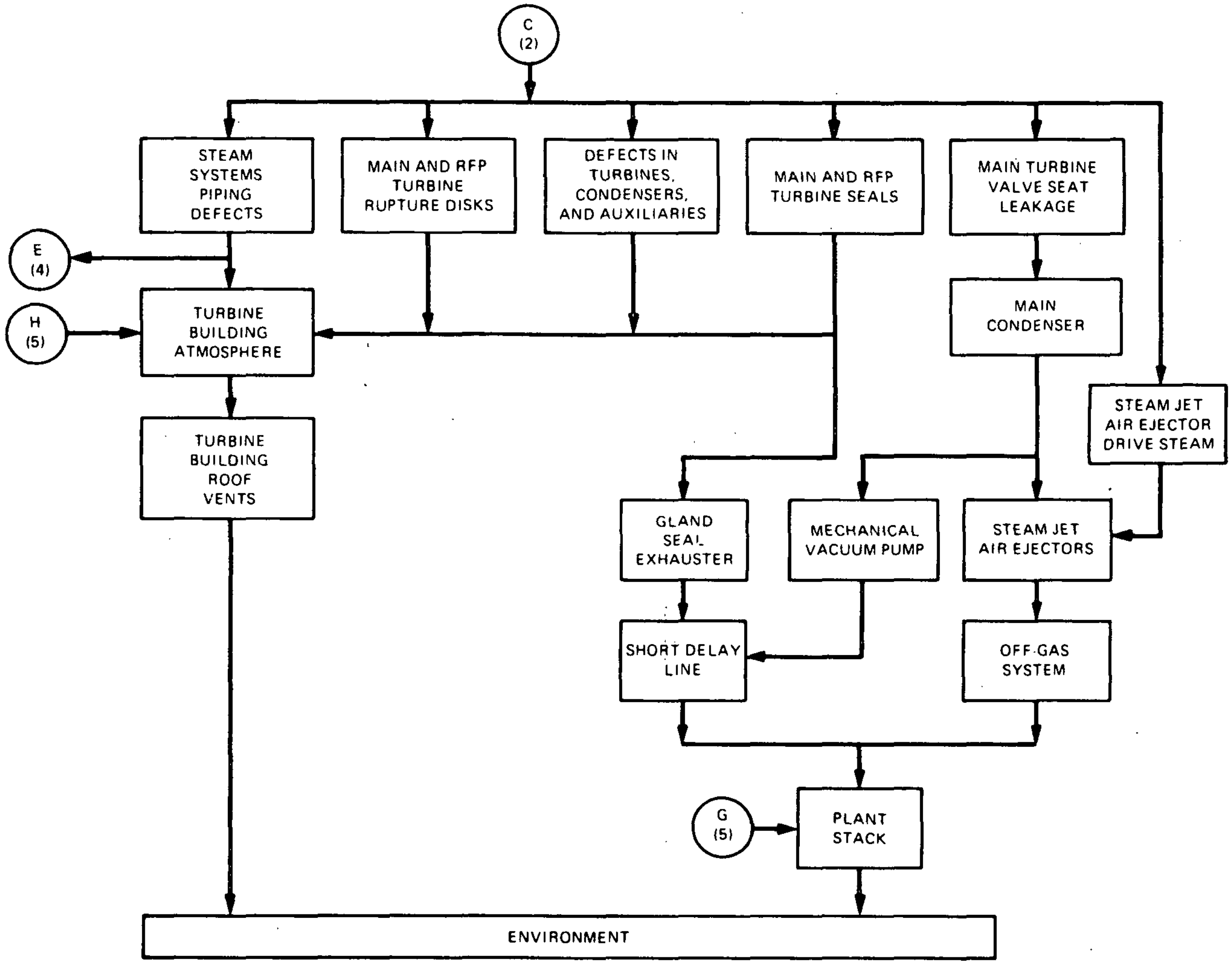


**Figure D.1 (cont.).**

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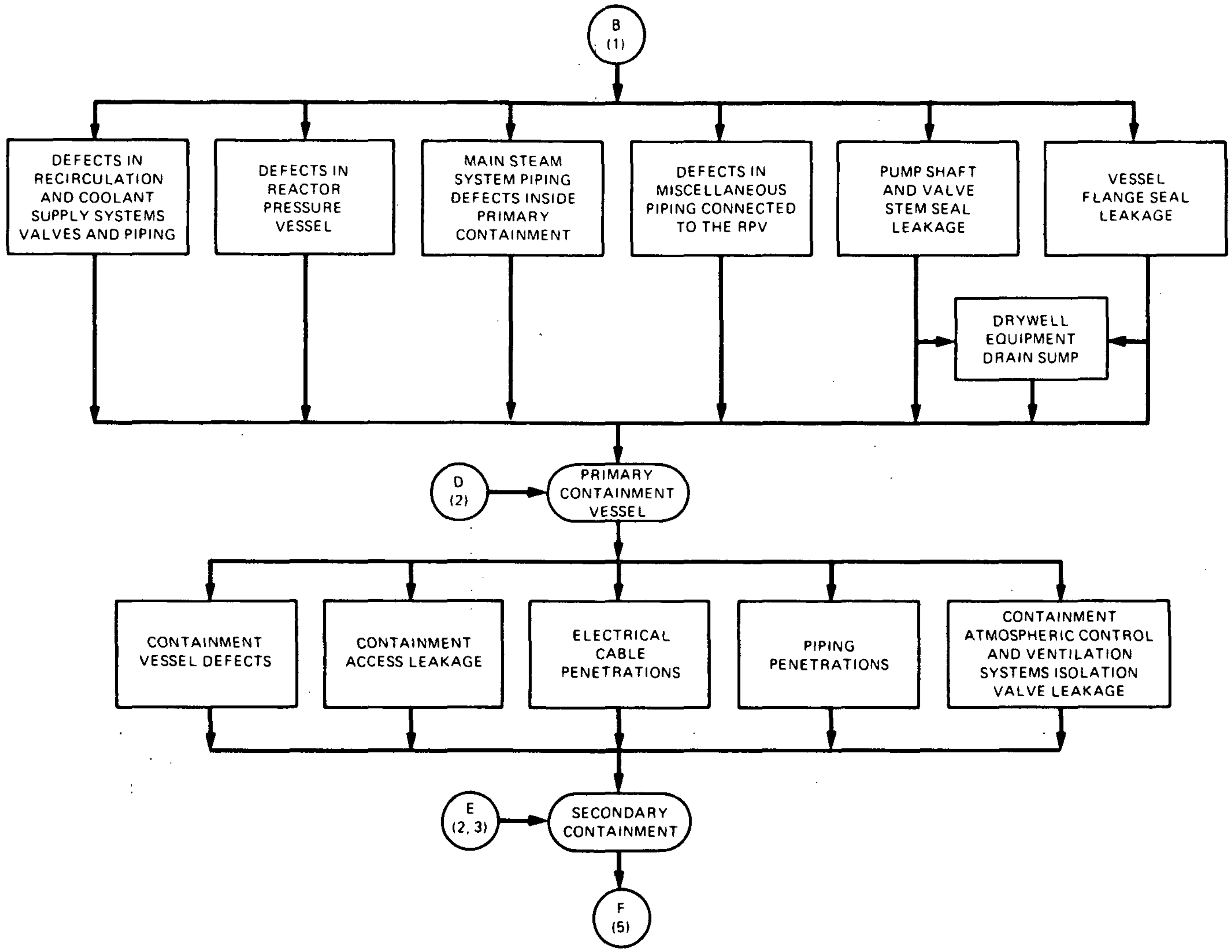


**Figure D.1 (cont.).**

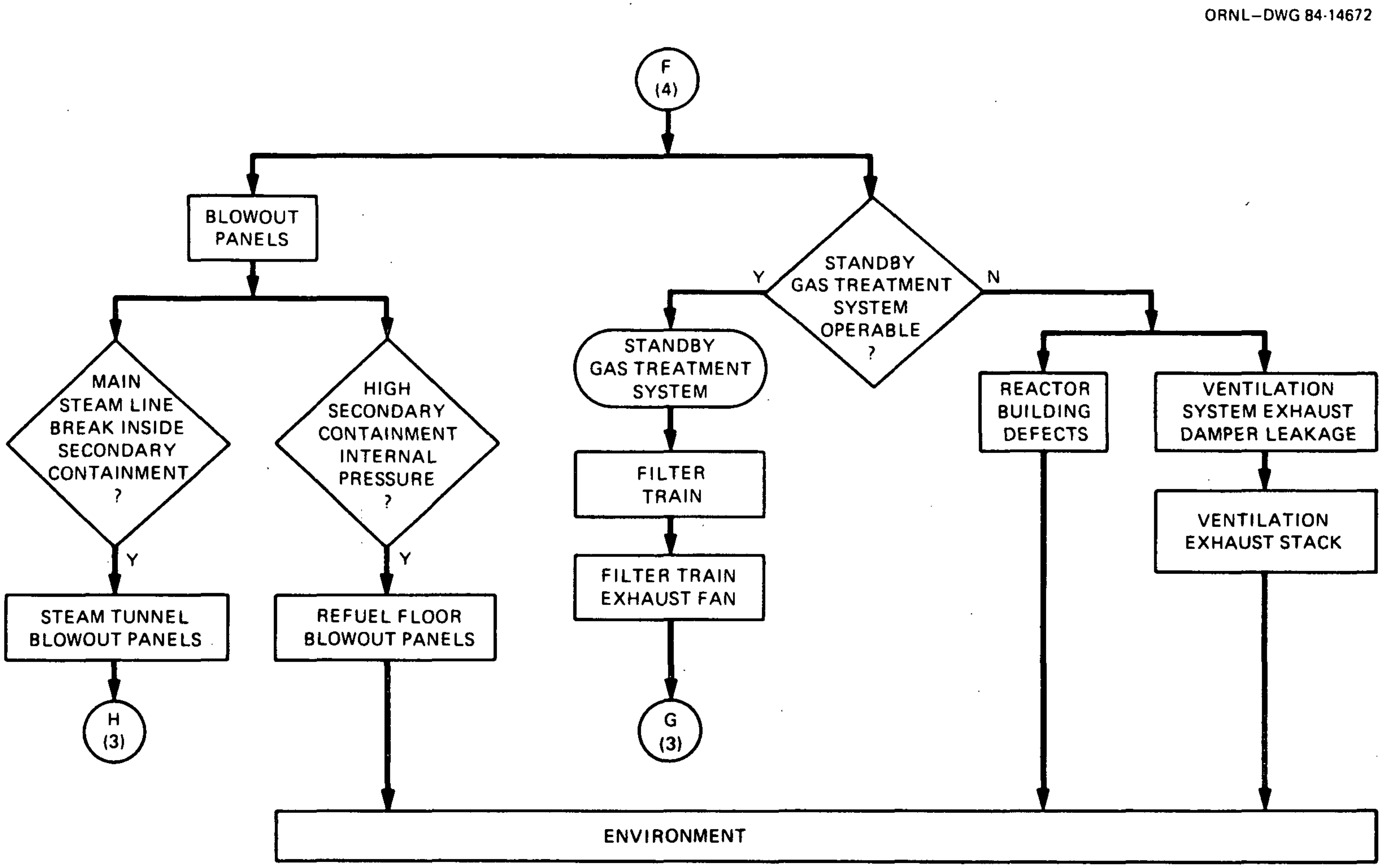


**Figure D.1 (cont.)**

### D.3.2. Fuel Cladding

During normal operation certain gaseous and volitile fission products migrate through the cladding and reach the coolant. This is due to microscopic defects in the cladding that permit the release of these fission products. In addition, fission products escape through larger cladding defects which may go undetected until the fuel is operated. This will cause a higher than normal coolant activity, and increased airborne radioactivity outside the reactor.

Cladding failure can occur due to thermal stresses, corrosion or oxidation. Thermal stresses occur when the cladding temperature changes rapidly due to partial film (transition) boiling. Localized regions of the cladding become vapor blanketed and the clad temperature rises rapidly. When the vapor eventually is displaced by liquid coolant, the clad temperature decreases rapidly. These rapid temperature oscillations will quickly cause the cladding to fail, releasing fission products to the coolant.

Cladding oxidation occurs at high temperatures in the presence of water or steam. The zirconium reacts with water molecules to form zirconium oxide and free hydrogen. The zirconium oxide is brittle and easily breaks under stress.

Under extreme conditions of fuel overheating or insufficient coolant the cladding will melt. The melting point of the zirconium alloy (zircaloy 2) is 3365°F, while the melting point of zirconium oxides is 4920°F. Since the melting point of uranium oxide is 5200°F cladding melt will occur before complete fuel melt although there will probably be centerline fuel melt at the time of cladding melt due to higher temperatures at the centerline. At temperatures above the zircaloy melting point, the $UO_2$ and Zr form a eutectic mixture with a significantly lower melting point (approximately 4250°F) than that of pure $UO_2$. The molten cladding theoretically will run down the surface of the fuel rods like hot candle wax and refreeze in the cooler regions of the lower core. The unconfined fuel pellets will fall into the lower portion of the core and vessel forming a rubble bed. Although the geometry of the rubble bed would prevent criticality, the temperature of the fuel material will rapidly increase because of poor heat transfer from the surface of the rubble bed.

### D.3.3. Primary System Boundary

Confinement of gaseous fission products within the primary system boundary can be a difficult task during an ATWS. Under the conditions of high temperature and pressure encountered during the ATWS, the escape of gaseous contaminants through the primary system increases proportionally. The primary system also permits the release of fission products without the benefits of the primary containment barrier. Several systems connected directly to the reactor vessel penetrate the containment through dual isolation valves. If these isolation valves experience any seat leakage, or they are opened during the accident, the gaseous fission products can bypass the primary containment barrier. The main steam lines are the most likely path for such a release from the primary system.

Since part of the primary system boundary penetrates primary containment, the escape of radioactive material from the primary system must be examined for two cases; the primary system intact and the primary system not intact.

With the primary system boundary intact, radioactive material can be released by two systems, the main steam system and the primary coolant sample line. During an ATWS, a significant amount of coolant will leave the reactor through the main steam S/RVs. Any non-condensible gases in the coolant and the reactor steam dome will leave with the steam. Discharge from the S/RVs is piped to the suppression pool which is part of primary containment. Gaseous fission products can also be transported through containment via the main steam lines. The main steam isolation valves (MSIV) close at the initiation of the ATWS event to establish a barrier at primary containment. With the MSIVs closed a small amount of leakage may occur at the valve seat. At some point in the accident the operator may choose to open the MSIVs to use the main condenser as a heat sink. In either case, radioactive material can be transported outside primary containment and eventually reach the environment by one or more of the following paths:

1. Steam System Piping Defects - Gaseous fission products may be released into the Secondary Containment atmosphere, or the turbine building atmosphere. If the leakage occurs in the turbine building, the radioactive material will exit the building through the roof vents.

2. Main and Reactor Feedwater Pump (RFP) Turbine Rupture Disks - Rupture disks are installed on the turbines to prevent damage to the turbine casing from high pressure. If one of these rupture disks has been cut, gases may be released from the turbine casing into the turbine building atmosphere.

3. Defects in Turbines, Condensers, and Auxiliaries - The turbine building contains a large number of components that handle radioactive materials. Defects in turbine support equipment or the condensate and feedwater systems has the potential for releasing radioactive materials into the turbine building atmosphere.

4. Main and RFP Turbine Seals - The main steam system supplies low pressure steam to seal the main and RFP turbine shafts where they penetrate the turbine casing. Leakoff from the turbine seals is removed by an exhauster. The non-condensible gases are discharged to the plant stack via a short (1.75 minute) holdup line.

5. Main Turbine Valve Seat Leakage - Any leakage through the main turbine stop and control valves will drain to the main condenser (valve above and below seat drains, and turbine casing drains). The non-condensible gases are removed from the main condenser by one of two methods; steam jet air ejectors (SJAE) or mechanical vacuum pumps.

    a. Steam Jet Air Ejectors - The steam jet air ejectors are normally used to remove non-condensible gases from the main condenser during power operations. Driving steam for the air ejectors is taken from the main steam lines before the turbine stop valves. The driving steam removes the condenser gases by a venturi effect and discharges the gases to the off-gas system for treatment. However, the SJAEs cannot be used when the main steam lines are isolated since the driving steam is taken from the main steam lines outside primary containment.

b. Mechanical Vacuum Pumps - The mechanical vacuum pumps remove the non-condensible gases from the main condenser during startup and discharge to the short holdup line on the discharge of the gland seal exhauster. During the ATWS the mechanical vacuum pump cannot be operated because of a high reactor pressure interlock. However, if the suction and discharge valves are open the vacuum pump is a possible release pathway.

The primary coolant sample line is connected to the reactor recirculation system discharge piping. The sample line penetrates the primary containment, via two isolation valves, to a sample station in secondary containment. Leakage of gaseous fission products through the isolation valves is released into the secondary containment atmosphere. Periodically during normal operation, and when requested during accident operations, samples of the primary coolant are taken for laboratory analysis. Gases dissolved in the coolant may be released from the coolant sample in the laboratory. The airborne contamination may then be released to the environment through the laboratory ventilation system.

If the primary system boundary is not intact, the fission product gases will be released into the primary containment. The release from the primary system may be from one or more of the following sources:

1. Defects in Recirculation and Coolant Supply Systems Valves and Piping - Several systems that connect to the reactor pressure vessel (RPV) may develop small cracks due to the unusually high pressures that could occur during an ATWS. It is more likely, however, that valve steam seals would fail before the piping cracked. The systems included in this

category are the reactor recirculation system, the feedwater system, the Reactor Core Isolation Cooling System (RCIC), and three Emergency Core Cooling Systems (ECCS).

2. Defects in the Reactor Pressure Vessel - Although it is unlikely the reactor vessel will fail due to high presusres, there is the possibility of severe vessel failure in the event of core melt.

3. Main Steam System Piping Defects Inside Primary Containment - In the case of main steam piping defects, a larger quantity of gaseous fission products will be released than in the case of liquid line defects. The main steam lines connect to the RPV well above the normal reactor water level where non-condensible gases would tend to collect.

4. Defects in Miscellaneous Piping Connected to the RPV - There are a large number of connections to the reactor pressure vessel for instrument lines, vent lines, head spray, and control rod drive hydraulics. A failure of one of these small diameter lines would release fission products directly into the drywell atmosphere.

5. Pump Shaft and Valve Stem Seal Leakage - The recirculation pumps are designed for controlled shaft seal leakage. This leakage and the stem seal leakage from certain valves inside the drywell are collected in the equipment drain sump. Gaseous fission products escaping from the coolant will be released into the drywell atmosphere or into the sump. From the sump, the gases can enter the drywell atmosphere through the sump vent.

6. Vessel Flange Seal Leakage - Two concentric seal rings in the vessel flange prevent leakage between the flange faces at operating pressure. Leakage past the inner seal ring is piped to the drywell equipment drain sump, while any leakage past the outer seal ring is releaed into the drywell atmosphere.

## D.3.4. Primary Containment

The primary containment serves as the major barrier to the release of gaseous fission products under accident conditions. The steel pressure vessel is designed to confine steam, water, and non-condensible gases during a loss of coolant accident with a maximum leakage of 1% of the containment volume per day at design pressure.

Leakage from the primary containment during ATWS may occur through one or more of the following pathways:

1. Containment Vessel Defects - Periodic testing for leakage ensures that the containment vessel is free of defects. However, in the ATWS scenario defects may develop due to excessive internal or external pressure, or in an extreme case severe failure due to contact with molten fuel.

2. Containment Access Leakage - All openings in the containment for personnel and equipment access are sealed during operation. High internal pressure could increase the leakage rate through the access door seals.

3. Electrical Cable Penetrations - All cable penetrations into containment are sealed against leakage. High internal pressure could cause an increase in leakage through the cable penetrations.

4. Piping Penetrations - With hot piping penetrations some pipes are fitted with a guard pipe which vents back into the drywell if a line break should occur inside the penetration. A bellows assembly is welded between the process line and the guard pipe to seal the penetration. Any defects or rupture of the bellows would release fission products from the primary containment into secondary containment.

5. Containment Atmospheric Control and Ventilation Systems Isolation Valve Leakage - During normal operation the containment is inerted with nitrogen. The pressure inside containment is controlled by regulating the nitrogen make-up to containment and the operation of cooling units. There is also a fresh air ventilation system that is operated when personnel are working inside the drywell.

   During accident conditions, containment pressure is controlled by a combination of pressure suppression and vacuum relief valves. The vacuum breakers prevent the containment from exceeding the design external pressure.

   Each line connected to the drywell or torus air space is equipped with double isolation valves located outside Primary Containment. Any valve seat leakage through these isolation valves would release fission products into the Secondary Containment.

## D.3.5. Secondary Containment

The gaseous fission products escaping from a breach in primary containment will collect in the secondary containment. The radiation levels in the reactor building will increase in proportion to the severity of the primary containment failure and the degree of fuel damage. The release rate from the secondary containment depends on the status of the next barrier, the standby gas treatment system.

If the SGT system should fail, ground level releases would occur through reactor building defects and ventilation damper leakage. The untreated contaminants that escape from the secondary containment will then be dispersed to the atmosphere.

In extreme cases of Primary Containment and Primary System Boundary failure, the internal pressure of Secondary Containment could reach the relief pressure of the blowout panels. The steam tunnel blowout panels would relieve the pressure into the turbine building and then to the environment through the turbine building roof vents. The refuel floor blowout panels relieve the pressure directly to the environment.

### D.3.6. Secondary Containment Ventilation System

During the ATWS event, the primary release path from Secondary Containment would be through the SGT system. The normal ventilation will shut down and the SGT will start soon after initiation of the event due to low reactor water level. The SGT will also start automatically on high drywell pressure. However, the first initiation signal will be the low reactor water level signal. This will limit ground level radioactive releases, but the elevated release rate will exceed regulatory limits if Primary Containment failure occurs. Extremely high concentrations of particulate fission products or high humidity would reduce the filter trains effectiveness for removing fission products from the atmosphere prior to release.

Although it is undesirable to release any contaminants to the environment, the release through the SGT is more desirable than the ground level release. By diluting the gases with outside air and releasing the mixture at a higher elevation (600 feet above ground level), the radiation doses to the general public are greatly reduced. Another advantage of using the SGT is the removal of nearly all of the particulate fission products. Many of the particulates have long radiological half-lives and/or long biological half-lives.

## D.4. Pathways for the Release of Radionuclides in Liquid Streams from a BWR During ATWS

Unlike PWRs which confine most contaminated systems inside the containment building, BWRs have contaminated systems throughout the plant. Effective management of these systems is essential in both normal and emergency operations to prevent the release of radioactive contaminants to the environment. For this reason, each possible release pathway is continuously monitored. The potential pathways for release of radionuclides in liquid streams are represented in Figure D.2.

During the ATWS event, many of the potential release pathways isolate because of low reactor water level or high drywell pressure. The operator should use caution when defeating or resetting these isolations if fuel damage is suspected. Higher than normal radiation levels in liquid streams can result in high

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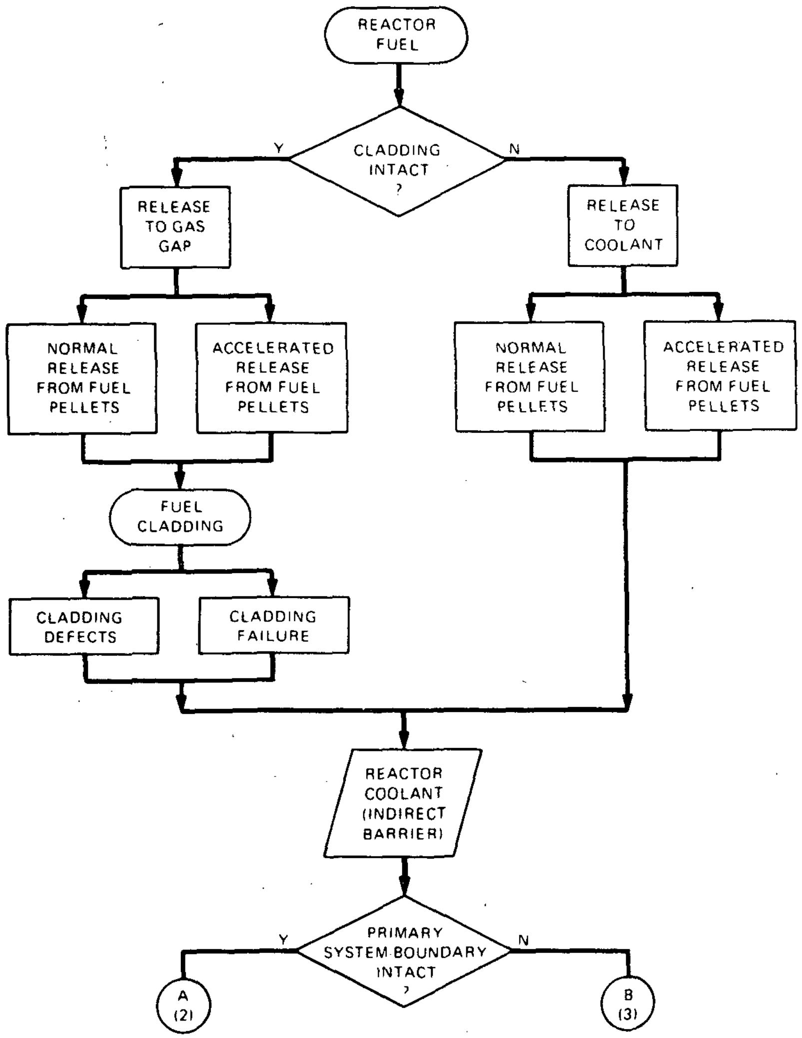


**Figure D.2. Pathways for release of radionuclides in liquid stream from a BWR during ATWS.**

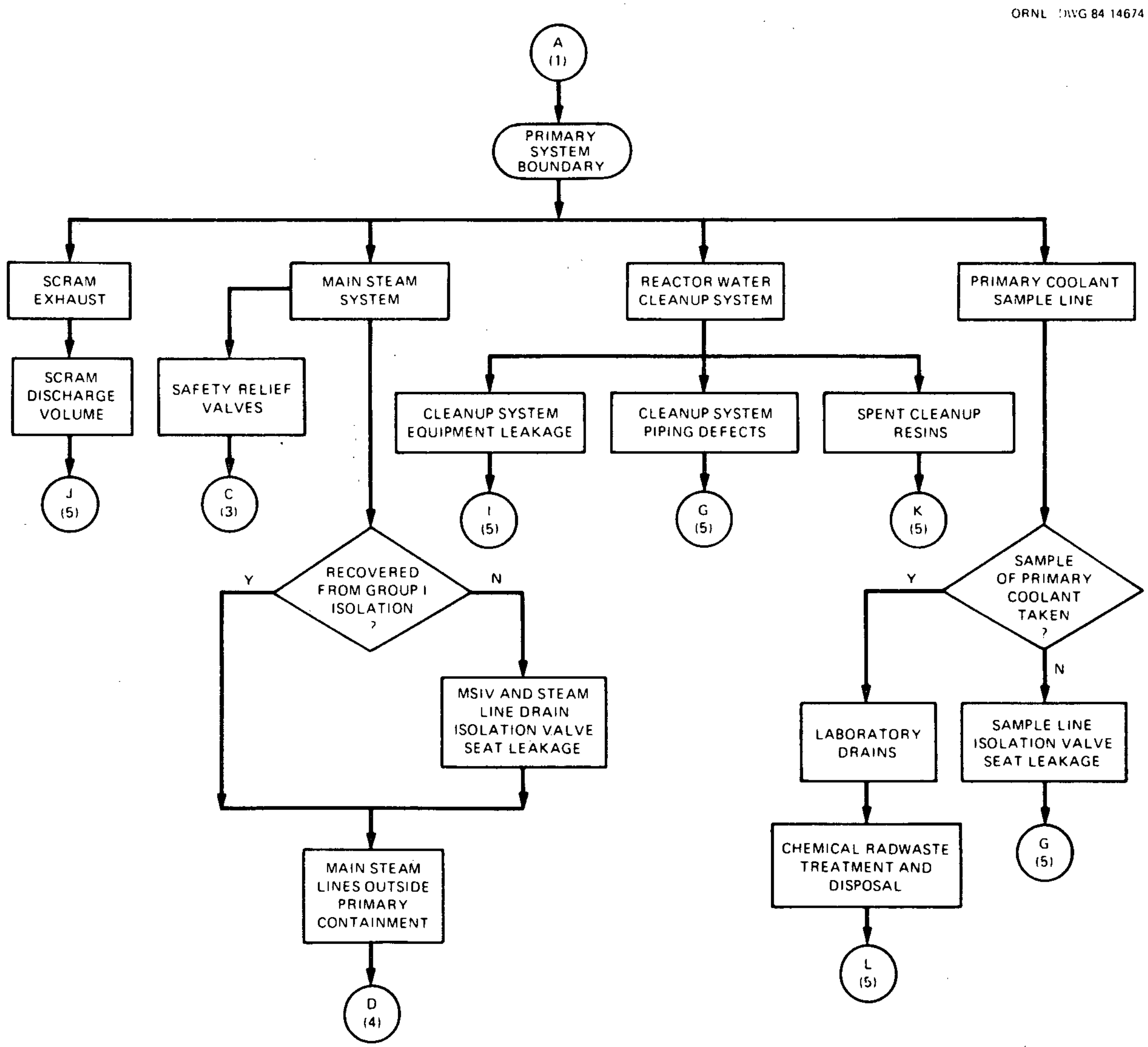


**Figure D.2 (cont.).**

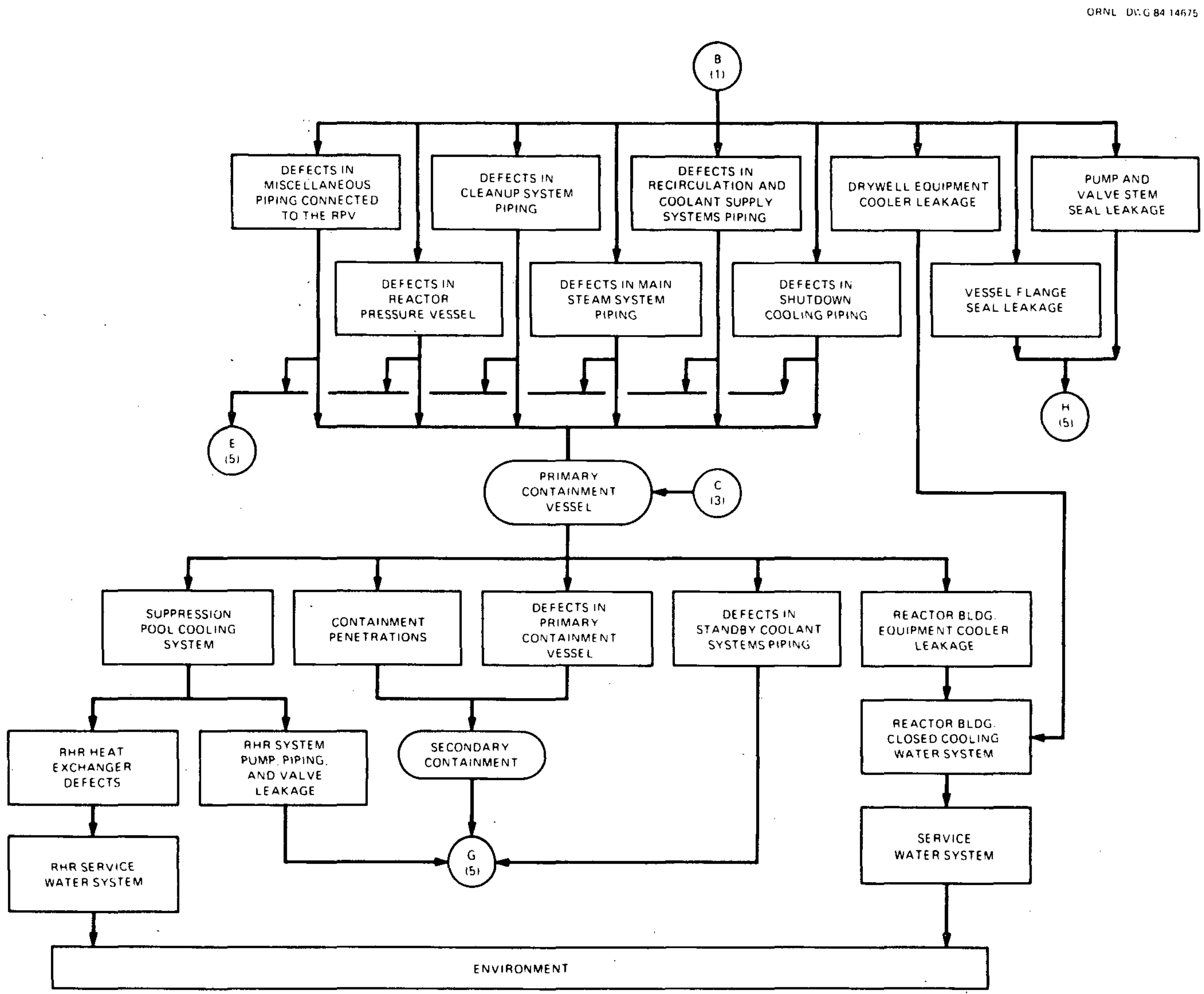


**Figure D.2 (cont.).**

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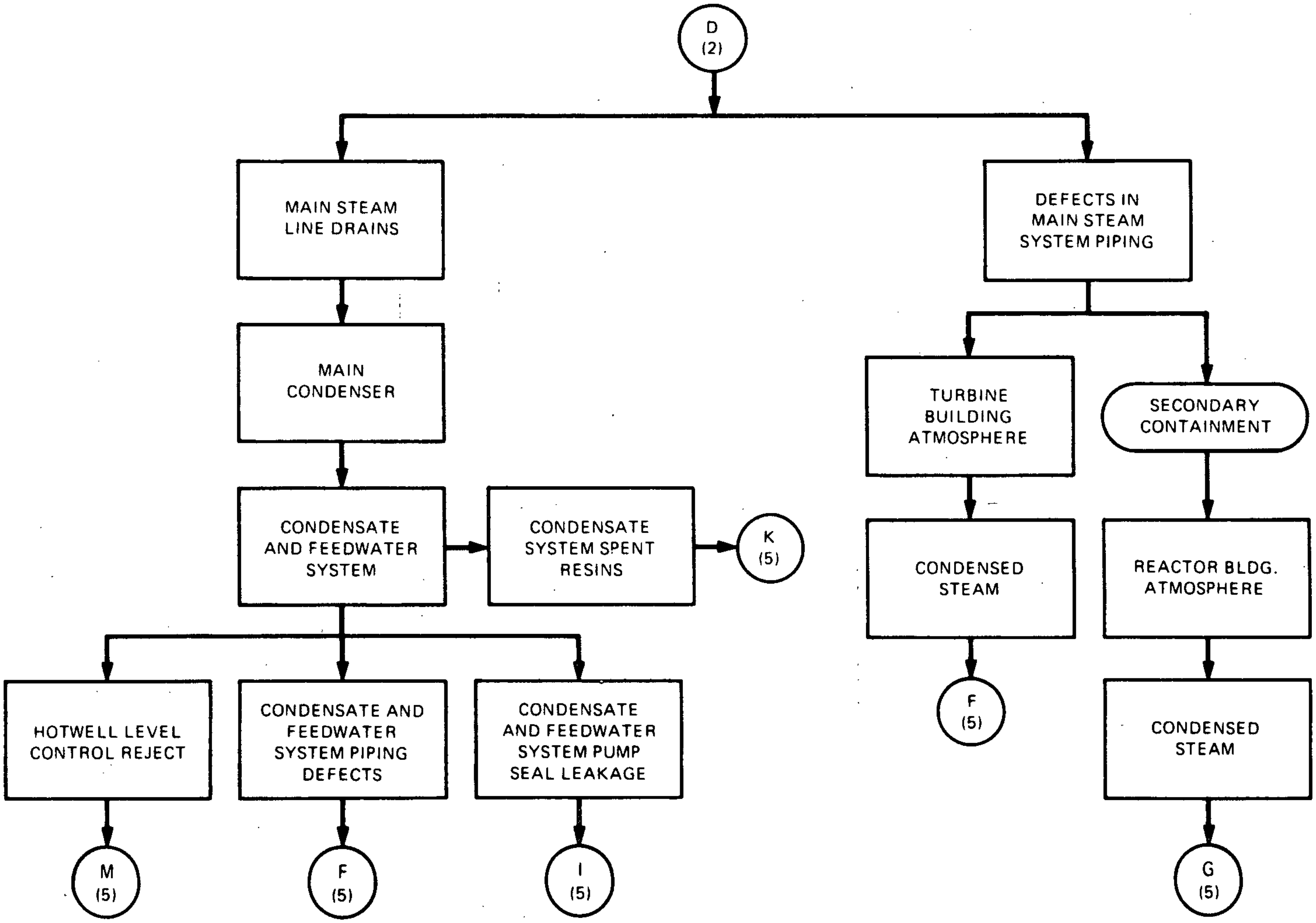


Figure D.2. (cont.).

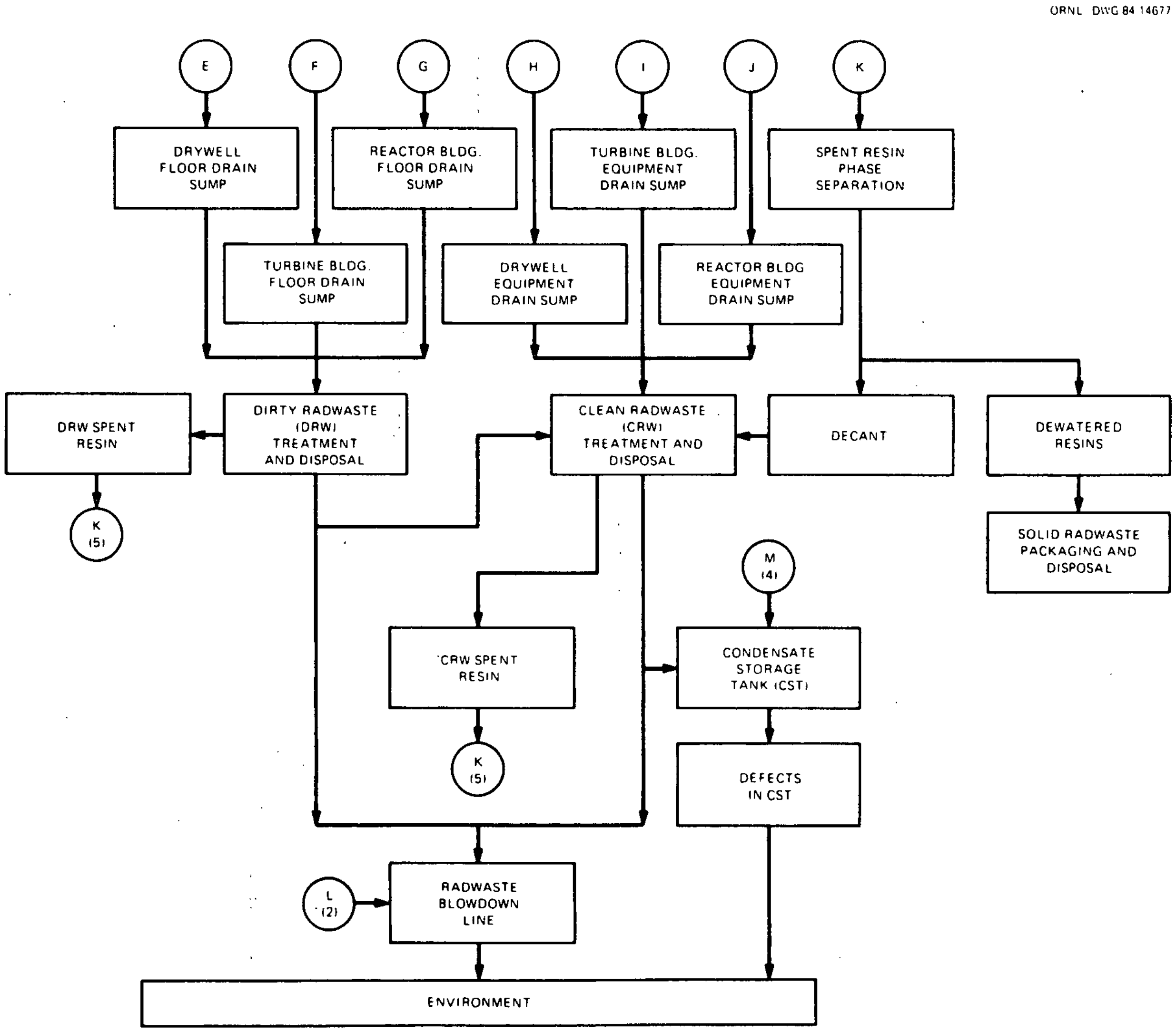


Figure D.2 (cont.).

doses to plant personnel in accessible areas of the plant when contaminated liquids are not confined. Also, the potential for off-site release increases when isolation valves are opened.

### D.4.1. Reactor Fuel

To minimize contamination of liquid streams, preservation of fuel integrity is essential. Internal fuel damage has the potential for releasing large quantities of fission products to the gas gap. From the gas gap the fission products can reach the reactor coolant through cladding defects or cladding failure points. When the fuel is overpowered or overheated the release rate of halogen and particulate fission products increases. Fuel pellet cracking at high temperatures provides a release path from the interior of the pellet where these fission products would otherwise remain. Radial fuel expansion at high temperatures can narrow or close the gap between the pellets and the cladding increasing the possibility of clad failure.

### D.4.2. Fuel Cladding

The fuel cladding is, perhaps, more important for limiting the contamination of liquid streams than for airborne contamination. As previously noted, the gaseous and iodine fission products can escape through defects in the cladding. However, those fission products that are likely to remain in the liquid streams cannot easily escape the cladding, although small quantities do reach the coolant through microscopic cladding defects.

An important reason for preserving cladding integrity is to prevent the coolant from coming into contact with the fuel. If this occurs, the coolant will leach particulate fission products that would otherwise remain in the pellets.

The presence of iodine in the coolant is a normal occurance during operation due to its volitile nature. However, the presence of large quantities of particulates in the coolant is an abnormal condition. A transport medium is necessary for the particulates to escape from the fuel pellets. This is the reason that particulates in the coolant are a good indicator of fuel and/or cladding failure.

### D.4.3. Primary System Boundary

During an ATWS, the reactor vessel and the interconnecting piping comprise the principle barrier for liquid streams. Maintenance of the primary system boundary will ensure that contaminated liquids are contained and manageable. The operators are responsible for controlling three parameters for protection of the primary system boundary; system pressure, temperature, and rate of temperature change.

The release of fission products in liquid streams from the primary system boundary can occur whether the boundary is intact or not. If the boundary is intact, the release may occur by one or more of the following pathways:

1. Scram Exhaust - If the operator should succeed in scraming the reactor, the water displaced by the control rod drive pistons is exhausted to a Scram Discharge Volume (SDV) that is external to primary containment. The scram exhaust water is held in the SDV until the scram is reset by the operator. After resetting the scram, the SDV drain valves open and drain the water to the Reactor Building Equipment Drain Tank (RBEDT). The sump is pumped to the radwaste system when full. (The radwaste system will be discussed in Section D.4.6).

2. Main Steam System - In the ATWS event, the greatest loss of coolant from the primary system is through the main steam system. After the main steam lines isolate, the reactor pressure is controlled using the safety/relief valves. A typical S/RV can pass 810,000 lb/hr of steam at 1100 psig. The steam is discharged into the suppression pool for quenching. The high pressures that occur during ATWS with MSIV closure will cause several relief valves to cycle until actions are taken to reduce reactor power.

The main steam system can transport fission products through primary containment. Outside containment the fission products can be released through the steam cycle or through defects in the main steam system piping.

If steam escapes from the main steam lines through piping defects, external to primary containment, the steam will condense in the building atmosphere. Depending on the location of the leak, either in the reactor building or in the turbine building, the steam will drain to one of the floor drain sumps. Leakage collected in the sumps will then be pumped to the radwaste system.

If the steam does not leak from piping defects, it will be routed to the main condenser through the steam line drains or through the turbine. The condensate system takes a suction on the main condenser for supplying water to the reactor feedwater system. At various points in the condensate and feedwater system, fission products may be released from the system through:

a) spent demineralizer resins

b) hotwell level control reject to the CST

c) piping defects in the condensate and feedwater system

d) pump shaft seal leakage

With the exception of hotwell level control reject, the leakage is collected by the radwaste system.

3. Reactor Water Cleanup System - The reactor water cleanup system continuously filters and demineralizes the reactor water during normal operation. During ATWS, however, the system will automatically isolate on low

reactor water level. At some point in the recovery phase the cleanup system would be restarted, especially if fuel damage is suspected. Fission product release from the cleanup system may occur through:

a) RWCU equipment leakage

b) piping defects

c) spent demineralizer resins

4. Primary Coolant Sample Line - The coolant sample line is a possible source of fission product release into Secondary Containment or the chemical radwaste system. With the sample line isolated, any leakage through the valves will be collected in the sample station, and drained to the reactor building floor drain sump. The effluent is then sent to the dirty radwaste system for treatment.

   Samples drawn for analysis are taken to the laboratory. Once the sample has been analyzed it is discarded into the laboratory drains, which are processed by the chemical radwaste system.

In the event that the Primary System boundary is not intact, the release of fission products will occur inside the primary containment. The reactor coolant that escapes the primary system inside containment may be contained by the Primary Containment Vessel and collected by the drywell equipment drain sump or collected by the drywell floor drain sump. If the source of leakage is large, the sumps will fill quickly. In this case, reactor coolant will drain to the suppression pool via the eight vent pipes.

The possible breach points in the Primary System Boundary inside Primary Containment are:

1. Defects in Miscellaneous Piping Connected to the RPV - This possible source of leakage includes the many instrument lines and support systems connected to the reactor vessel.

2. Defects in Reactor Pressure Vessel - Although failure of the reactor vessel is unlikely, severe fuel failure could result in vessel failure if molten fuel comes in contact with the vessel. Control rod drive stub tubes are likely to fail in the event of core melt. The molten fuel will then fall to the containment floor releasing large quantities of fission products to the containment.

3. Defects in Cleanup System Piping - The cleanup system piping and valves inside primary containment form part of the Primary System Boundary. A failure of the system's piping or valves would drain reactor coolant into the drywell, without the possibility of isolating the leakage.

4. Defects in Main Steam System Piping - The main steam piping inside Primary Containment is a possible source of coolant leakage from the Primary System. Steam escaping from the main steam lines will cause the drywell pressure to increase. When drywell pressure exceeds torus pressure, the steam will be forced into the suppression pool for quenching.

   A steam line break is less severe than a liquid line break where core cooling is a concern. Less coolant mass is lost through a steam break than through a comparable size liquid line break. The steam break also has less effect on vessel flooding capability. The reactor vessel will, however, depressurize more quickly on a steam line break than a liquid line break.

5. Defects in Recirculation and Coolant Supply Systems Piping - The consequences of a failure of the recirculation system piping, or the piping for one of the coolant supply systems, can have a devastating effect during an ATWS. The DBA LOCA is a complete break of the recirculation pump suction piping. However, the analysis assumes that the reactor scrams when the LOCA occurs. If the reactor remains critical (or partially critical) more coolant mass will be lost through the break than the FSAR analysis indicates because of the higher energy of the fluid. Core reflooding may be difficult or impossible under these circumstances.

In the event of a coolant supply system (e.g. HPCI, RCIC, feedwater) piping failure, two factors affect the ability to reflood the reactor. In addition to losing reactor coolant through the break, coolant is lost from the system that would otherwise provide core cooling.

The coolant lost through these piping defects would be contained by the Primary Containment. The coolant that is released into the drywell will drain into the drywell floor drain sump. Once the sump is filled, the water will drain to the suppression pool via the eight vent pipes.

6. Defects in Shutdown Cooling Piping - The portion of the shutdown cooling suction piping inside Containment is another possible large source of leakage. Since the shutdown cooling suction lines are connected to the recirculation system, a break of this pipe in analogous to a recirculation piping break.

7. Drywell Equipment Cooler Leakage - Several drywell components are cooled by the Reactor Building Closed Cooling Water System (RBCCW). The single drywell component that could leak into the RBCCW system is the drywell equipment drain tank cooler. The RBCCW system is normally operated at a pressure higher than the drain cooler so that leakage into RBCCW is unlikely. However, the possibility of leakage into RBCCW exists if the system pressure falls below sump cooler pressure.

8. Vessel Flange Seal Leakage - Any leakage past the inner flange seal ring is drained to the drywell equipment drain sump.

9. Pump and Valve Steam Seal Leakage - Drywell components that normally have controlled leakage are equipped with drain lines to the drywell equipment drain sump.

## D.4.4. Primary Containment Vessel

The ATWS event will severely challenge the Primary Containment even without a failure of the preceeding barriers. The operation of relief valves due to high reactor pressure will cause a steady increase in suppression pool water temperature and level, and drywell pressure, temperature, and humidity. High suppression pool water temperature and level reduces the pool's capacity for steam quenching. This, in turn, affects the integrity of the fuel, since insufficient core cooling may result. High drywell pressure, temperature, and humidity have a less drastic effect on the plant, but may cause electrical equipment failures and increased containment leakage rates. However, if corrective action is not taken the containment may fail due to excessive internal pressure.

The possible primary containment breach points are:

1. Suppression Pool Cooling System - The suppression pool cooling mode of the RHR system is used during ATWS to transfer heat from the suppression pool to the ultimate heat sink (the environment). In the torus cooling mode, the RHR pumps take a suction from the suppression pool and circulate the water through the shell side of the RHR heat exchanger. The water is then returned to the torus through a full flow test line. RHR Service Water is circulated through the tube side of the heat exchanger to remove the heat from the torus water.

   Fission products may be released from the system in either the heat exchanger or the numerous system pumps, piping, and valves. Heat exchanger defects have the potential for releasing fission products into the service water, which will transport the contamination to the environment. Normally, the service water system is operated at a higher pressure than the RHR system to prevent leakage into the service water.

Defects in the pumps, piping, and valves of the RHR system will release the fission products into Secondary Containment. The reactor building floor drain sump collects the leakage for processing in the radwaste system.

2. Containment Penetrations - The containment penetrations for piping and electrical cables could be a source of leakage from Primary Containment. Leakage from the containment penetrations is collected in the reactor building floor drain sump for processing in the radwaste system. Most leakage from these penetrations, however, would be gaseous rather than liquid.

3. Primary Containment Vessel Defects - Defects in the containment vessel could develop from missiles, pipe whip, overpressurization, or corrosion. Fission products escaping from the Primary Containment will be collected in the Secondary Containment. The primary containment may also be breached in the event of core melt and reactor vessel failure. The molten fuel that falls to the drywell floor would attack the concrete floor of the containment.

4. Defects in Standby Coolant System Piping - The standby core cooling systems (RCIC and ECCS) use the suppression pool for pump suction. Any leakage from these systems will be collected in the Secondary Containment.

### D.4.5. Secondary Containment

The Secondary Containment is designed to prevent the escape of fission products in liquid streams to the environment. The reactor building is constructed of reinforced concrete resting on bedrock. The weakest points in the structure (expansion joints, dampers, and blowout panels) are located such that a failure will not permit the release of liquid streams.

In the event of a large pipe break inside Secondary Containment, the water spilled on the floor will drain to the lowest level of the building through access openings and equipment hatches in the floor. Floor drains on each level of the building drain the spillage to the Reactor Building Floor Drain Sump. The sump pump will automatically start on high sump level to pump the water to the radwaste Floor Drain Collector Tank (Dirty Radwaste) for storage until processing. Because of the limited tank capacity, the time required to drain the spillage in Secondary Containment is restricted by the capacity of the radwaste system for processing the waste.

Small liquid leaks in Secondary Containment are more easily confined to local areas of the building. Curbs are constructed around equipment that has a potential for leakage. Floor drains inside the curbed area will drain the leakage to the sump.

### D.4.6. Liquid Radwaste System

In a sense, the radwaste system acts as an additional barrier to the release of fission products to the environment. The liquid radwaste system collects, treats, and returns processed radioactive liquid radwaste to the plant for reuse. Treated wastes that are unsuitable for reuse are discharged from the plant through the circulating water discharge canal or solidified for offsite burial.

The liquid radwaste system is divided into three subsystems; dirty radwaste, clean radwaste, and chemical radwaste. The distinction between clean and dirty waste is the conductivity and source of the waste. Floor drains collect liquids from unknown sources that are usually of high conductivity. Equipment drains are from known sources of low conductivity. Chemical wastes are from sources where the pH value requires special treatment of the waste. Chemical waste is neutralized and then treated in the dirty radwaste system.

In addition to the three liquid radwaste systems, the spent resin handling system handles both liquid and solid radwaste. The demineralizer resins used in the condensate system, the reactor water cleanup system, the fuel pool cleanup system and the liquid radwaste system are treated by the spent resin handling system. The spent resins are dewatered and packaged for off-site burial. The decant is sent to clean radwaste for further processing.

One objective of the liquid radwaste system is to recycle as much waste as possible. The recycled water is pumped to the condensate storage tank until needed. Since the CST is located outside, any defects in the tank would leak slightly contaminated water to the environment.

When the CST is full or the liquid radwaste system is overburdened, the operator may choose to discharge the processed fluids from the plant through a blowdown line. Prior to discharging effluents, however, the waste is pumped to a holding tank and sampled before discharging the waste to the environment. As a backup to laboratory analysis the blowdown line is equipped with a radiation monitor that will shut the blowdown line isolation valve on high radiation levels. The blowdown line discharges into the circulating water system discharge canal to dilute the effluents.

## D.5. Control Room Information for Detecting Barrier Degradation

### D.5.1. Reactor Fuel

Information concerning the fuel is provided by the neutron monitoring system inputs into the process computer. The Local Power Range Monitors (LPRM) are used by the process computer to calculate the following thermal parameters:

1. Linear Heat Generation Rate (LHGR)
2. Critical Power Ratio (CPR)
3. Average Planer Linear Heat Generation Rate (APLHGR)

CPR is a thermal parameter that is monitored to ensure cladding integrity. LHGR is the thermal parameter that is monitored for determining internal stress of the fuel pellets. APLHGR is monitored to ensure peak clad temperature does not exceed 2200°F following a Loss of Coolant Accident (LOCA) when a portion of the core is uncovered. CPR will be discussed in more detail with the fuel cladding.

APLHGR is monitored during normal operation to ensure peak cladding temperature does not exceed 2200°F on a design basis LOCA. The significance of 2200°F peak clad temperature is the onset of the self-sustaining metal water reaction. The limit on APLHGR ensures the peak clad temperature does not exceed 2200°F after the reactor is subcritical. Interpretation of APLHGR during the ATWS could be difficult or confusing to the operator, so it is of limited use during this accident.

The LHGR thermal parameter is used to indicate fuel enthalpy. Fuel enthalpy will indicate the extent of fuel damage due to excess heat generation within the pellets. Fuel enthalpies above 170 cal/g should be avoided to ensure fuel pellet to cladding interaction does not accelerate the release of fission products from the fuel. At 170 cal/g the internal heat generation induces cladding perforation. This is the design LHGR on the fuel. At 220 cal/g the pellets begin melting at the centerline, and fuel melt is complete at 280 cal/g. When the fuel melt is complete, the heat from the fuel causes plastic deformation of the cladding. At 425 cal/g fuel enthalpy, the fuel vaporizes and bursts the cladding. Finely divided fuel particles are then dispersed throughout the coolant causing a steam explosion. However, this is unlikely since the cladding will probably melt before fuel vaporization can occur.

Reactor thermal power is measured by the neutron monitoring system. The neutron monitoring system is divided into three ranges to provide neutron flux indication over the entire operating range of the reactor. The three neutron monitoring subsytems are:

1. Source Range Monitors (SRM) - The SRMs are four miniature fission chambers operated in the proportional region. The sensitivity of the detectors provides flux indication during shutdown and startup when neutron flux is low. SRM count rate and period indicators are located on Panel 9-5. Two of the SRM channels can be selected for recording, also on Panel 9-5.

2. Intermediate Range Monitors (IRM) - The eight IRMs are similar to the SRMs, but are less sensitive (ionization chambers). The IRMs provide flux indication from criticality to about 24% reactor power. The SRM and IRM detectors are retractable from the core to extend their useful life. During an ATWS neither of these subsystems are useful until reactor power is reduced below the power range. IRM readings are recorded by four dual pen strip chart recorders on Panel 9-5.

3. Power Range Monitors - The power range neutron monitoring subsystems use 172 miniature fission chambers (43 detector strings with four detectors per string) distributed throughout the core. These detectors provide flux information to the following subsystems:

    a. Local Power Range Monitors (LPRM) - The output of any LPRM detector can be displayed by selection of the detectors on Panel 9-14.

    b. Average Power Range Monitors (APRM) - APRM output can be read on Panel 9-14 indicators and strip chart recorders on Panel 9-5.

    c. Rod Block Monitor (RBM) - The RBM provides indication of average neutron flux around the selected control rod.

    d. Process Computer - The computer can print the readings for all LPRMs and display or print any APRM reading.

The APRMs and RBMs provide front panel indications of core average and local average power respectively.

Although the neutron flux information available to the operator is more than adequate for most normal and accident operations, during an ATWS this information can be confusing and unreliable. The effects of high temperature and core voiding reduce the reliability of these instruments. Also, plant parameter changes during an ATWS are rapid and cyclical (i.e. pressure, temperature, water level, etc.). The effects of plant parameters on reactor power are reflected by the nuclear instrumentation. The erratic behavior of the nuclear instrumentation also effects the process computer calculations of the thermal parameters, since the LPRM inputs are used in these calculations.

### D.5.2. Fuel Cladding

Information on the fuel cladding in the control room is limited. The only source of information available to the operator is the process computer.

The thermal parameter Critical Power Ratio (CPR) provides the operator with information concerning the boiling state inside the core. CPR is the ratio of actual power to the power which is calculated to cause transition (partial film) boiling in the core.

CPR = Actual Power/Critical Power

When this ratio is less than 1.0, transition boiling is occuring at some point in the core. Transition boiling leads to cladding damage as portions of the cladding experience film boiling. Localized regions of the cladding become steam blanketed and cladding temperature rises because of poor heat transfer. When the vapor is displaced by liquid the cladding temperature decreases rapidly. The rapid temperature changes experienced during transition boiling can stress the cladding to the failure point.

Cladding oxidation is another hazard during an ATWS. If the cladding temperature reaches 2200$^{O}$F in the presence of water or steam, a zirconium - water reaction occurs liberating free hydrogen. Detection of the hydrogen occurs in the containment by a hydrogen analyzer. For this reason, detection of cladding damage due to oxidation would be delayed until the hydrogen reaches the containment.

Positive detection of cladding damage is primarily through radiation levels. The increase in radiation levels in the main steam lines (only if MSIVs are open) or the primary containment are positive indication of cladding failure. The steam line radiation monitors are on Panel 9-10. The operator can confirm cladding damage has occurred by requesting laboratory analysis of a coolant sample. The sample analysis will also reveal the degree of cladding damage, but this is time consuming and requires personnel to enter potentially hazardous areas of secondary containment.

### D.5.3. Reactor Coolant

There is an abundance of information available in the control room regarding reactor coolant in the control room. For reactor water level alone there are ten indications covering four overlapping ranges. The level indications are:

1. Normal Range - three indicators and one recorder (Panel 9-5)

2. Emergency Range - two indicators (Panel 9-5)

3. Post Accident Flooding Range (Fuel Zone) - two indicators and one recorder (Panel 9-3)

4. Floodup Zone - one indicator (Panel 9-3)

In addition to reactor water level indication, every system that injects into the reactor has control room indication of flow rate. For ATWS, the important flow indicators are:

1. HPCI pump discharge flow (Panel 9-3)

2. RCIC pump discharge flow (Panel 9-3)

3. CRD Hydraulic System flow (Panel 9-5)

4. RHR System flow (Panel 9-3)

5. Core Spray System flow (Panel 9-3)

6. Feedwater System Flow (Panel 9-5)

The control room operator also has indication of coolant reserve in the CST (Panel 9-20) and suppression pool (Panel 9-3).

### D.5.4. Primary System Boundary

The information available to the operator for the primary system boundary is extensive due to the importance and complexity of this boundary. Obtaining a clear and concise picture of how the barrier is being breached may require the operator to assimilate the information from a large number of instruments and alarms. This information may deal with parameters internal and external to the primary system boundary. The following alarms and indications are available to the operator to detect and evaluate a breach of the primary system boundary:

(* denotes high level indications for diagnosis)

1. Reactor Pressure
    a. Wide Range Recorder (1) (Panel 9-5)
    b. Narrow Range Recorder (1) (Panel 9-5)

c. Narrow Range Indicators (3) (Panel 9-5)*

d. High Pressure Alarm (Panel 9-5)*

e. High Pressure Scram

f. ATWS Recirculation Pump Trip*

2. Reactor Recirculation System

a. Jet Pump Flow (Differential Pressure) Indicators (20) (Panel 9-4)

b. Recirculation Pump Discharge Flow Indicators (1/pump) (Panel 9-4)

c. Total Core Flow Recorder (1) (Panel 9-5)*

d. Recirculation Pump Seal Cavity Pressure Indicators (2/pump) (Panel 9-4)

e. Pump Seal Failure Alarms (2) (Panel 9-4)

f. Recirculation Pump Differential Pressure Indicators (1/pump) (Panel 9-4)

3. Residual Heat Removal System

a. Low Pressure Coolant Injection (LPCI) Initiation Signal*

1) Low-Low-Low Reactor Level (-143.5"), or

2) High Drywell Pressure (2.5 psig) and Low Reactor Pressure (450 psig)

4. Core Spray System

a. Core Spray System Initiation Signal (Same as LPCI)*

b. Core Spray Line Break Detection Alarm (on instrument rack)

5. Reactor Core Isolation Cooling

a. System Isolations and Alarms*

1) Low Reactor Pressure

2) High RCIC Steam Line Flow

3) High Exhaust Rupture Diaphragm Pressure

4) High Steam Space Temperature

b. Steam Line Pressure Indicator (Panel 9-3)*

c. Turbine Exhaust Pressure Indicator (Panel 9-3)

6. High Pressure Coolant Injection
   a. System Isolations and Alarms*
      1) Low Reactor Pressure
      2) High HPCI Steam Line Flow
      3) High Exhaust Rupture Diaphragm Pressure
      4) High Steam Space Temperature
   b. Steam Line Pressure Indicator (Panel 9-3)*
   c. Turbine Exhaust Pressure Indicator (Panel 9-3)

7. Primary Containment
   a. Drywell and Suppression Chamber Pressure Recorder (Panel 9-3)
   b. Drywell Pressure Indicator (Panel 9-3 and Panel 9-6)*
   c. Drywell Air Temperature Indicator (Panel 9-3)
   d. Suppression Pool Water Temperature Indicator (Panel 9-3)*
   e. High Drywell Pressure Alarm (Panel 9-3)*
   f. LOCA Signal to ECCS*

8. Main Steam System
   a. Main Steam Tunnel High Temperature Alarm and Isolation
   b. Main Steam Line High Flow Alarm and Isolation
   c. High Steam Line Radiation Alarm, Isolation, and Scram Signal
   d. Main Steam Line Flow Indicators (4)
   e. Safety/Relief Valve High Temperature Alarm (Panel 9-3)*
   f. Safety/Relief Valve Temperature Recorder (Panel 9-44)
   g. Safety/Relief Valve Acoustic Monitors (Panel 9-44)

9. Containment Radiation Monitor
   a. Noble Gas Monitor (Panel 9-2)
   b. Particulate Monitor (Panel 9-2)
   c. Halogen Monitor (Panel 9-2)

Note: Control room indication for these three radiation monitors is a multi-pen strip chart recorder shared by all three monitors.

10. Gaseous Radwaste Monitors

    a. Off-Gas Pre-treatment rad monitors (1 linear, 1 log) (Panel 9-2)
    b. Off-Gas Post-treatment rad monitors (2) (Panel 9-2)
    c. Stack Gas Rad Monitor (Inside stack)
    d. Off-Gas Post-treatment Hi-Hi-Hi rad isolation and alarms (Panel 9-3)*
    e. Stack Gas Monitor High Rad Alarm (Panel 9-3)*

It is obvious from the number of indications listed that diagnosis and planning for primary system failures can depend heavily on the conditions. During the ATWS a large number of alarms are received in the control room and the operator's attention is focused on a few key parameters. Detecting changes in the status of the primary system requires the operator to continuosly track and reappraise the situation as the conditions change.

### D.5.5. Primary Containment

Monitoring for a breach of the primary containment is by the use of radiation detectors in the Secondary Containment. There are 23 area radiation monitors located in accessible areas of the plant where there is a potential for changing radiological conditions. These area radiation monitors have control room indications (Panel 9-2) of the gamma radiation levels in the monitored areas.

### D.5.6. Secondary Containment

Gaseous and airborne particulate fission products escaping from primary containment may also be detected in the secondary containment ventilation system. Inside the exhaust ducts for each Secondary Containment zone are two radiation monitors that provide control room indication of the radiation levels in the exhaust gases. When high radiation is detected, these radiation monitors provide an automatic trip signal to the ventilation system in the effected zone, and start the Standby Gas Treatment System.

There is very little information available to the operator for detecting secondary containment failure. There are no control room indications that would inform the operator of a secondary containment failure. The only indications of secondary containment failure are loss of the building differential pressure and off-site radiation monitors. The loss of negative pressure indicates the building air in-leakage is greater than the ventilation system capacity. This would be the case for blowout panels opening. However, the differential pressure loss may be the result of ventilation system failure rather than secondary containment failure.

### D.5.7. Secondary Containment Ventilation System Failure

Control room indications are provided for both the normal ventilation system and the SGT. Included in the instrumentation are indicating lights for dampers and fans, and flow indication for the normal ventilation system. A control room alarm "Reactor Building Ventilation Abnormal" annunciates when the normal ventilation system trips and the standby gas treatment system initiates.

Fission product release from the SGT is detected by the stack gas radiation monitor. High radiation levels in the stack gas, however, could originate in the off-gas system, the steam seal system, the mechanical vacuum pump, or the SGT. The operator can determine the source of the stack gas contamination by monitoring the off-gas radiation monitors, and ensuring the steam packing exhauster and mechanical vacuum pumps are tripped.

### D.5.8. Miscellaneous

Auxiliary systems that have the potential for transporting contaminants to the environment are monitored for radiation. The four systems in this category are the Reactor Building Closed Cooling Water (RBCCW) system, Raw Cooling Water (RCW) system, the RHRSW system, and the Radwaste system.

The RBCCW system is equipped with a scintillation detector on the return piping. Control room indication is provided on Panel 9-2 and an alarm is located on Panel 9-3.

The RCW system discharge piping is equipped with a scintillation detector. Control room indication is provided on Panel 9-2 and an alarm is located on Panel 9-3.

The RHRSW system discharge piping is equipped with scintillation detectors, also. One detector monitors the discharge from heat exchangers A and C, and a second detector monitors the discharge from heat exchangers B and D. Control room indication is provided on Panel 9-2 and an alarm is located on Panel 9-3.

The radwaste blowdown line is monitored by a scintillation detector. The indication for this detector is located in the radwaste contorl room. The radwaste blowdown radiation monitor will alarm (in radwaste) and isolate the blowdown line on high radiation.

## D.6. Mitigating Operator Actions

The operator actions listed below assume that each barrier is breached in sequence from the fuel to the secondary containment ventilation system. It should be noted, however, that the analysis of the ATWS indicates the barriers would not fail in sequence. However, for the purposes of studying operator actions during a severe accident, this description is useful for examining the variety of alternatives the operator may use to mitigate the failure of a major fission product barrier.

### D.6.1. Reactor Fuel

The immediate concern of the operator in the ATWS is to gain control of reactor power. At the beginning of the accident when the MSIVs close reactor power "spikes" to a point where internal fuel damage is possible. The operator will monitor the APRMs for his principle source of information concerning the fuel. Additional information will be printed by the process computer concerning the thermal parameters LHGR, CPR, and APLHGR

(APLHGR is not very useful for this event). During the early stages of the ATWS the reactor power will oscillate wildly due to rapid changes in reactor pressure and temperature. The operator will monitor the nuclear instrumentation for trends rather than specific values because of these oscillations. Oscillations in the LHGR will obviously follow reactor power. The operator will undertake every possible attempt to scram the reactor. The operator will depress the scram pushbuttons, place the mode switch in SHUTDOWN, deenergize the reactor protection system, and shut off and vent the air supply to the scram valves. If the reactor does not scram by any of these actions (as the scenario requires), alternate methods of reducing reactor power are required.

There are several methods of reducing reactor power without scraming the reactor. None of these methods are particularly quick, and not all of them will take the reactor subcritical. The best method for reducing and controlling reactor power during an ATWS is a controversial issue, so the operator should use current procedures to dictate his selection.

The BWR Owners Group Emergency Procedure Guidelines (EPG) require the operator to initiate the Standby Liquid Control (SLC) system under the following conditions:

1. Five or more adjacent control rods are not inserted past position 06 (three notches from full in) and reactor water level cannot be maintained, or suppression pool temperature cannot be maintained less than 110°F, OR

2. Thirty or more control rods are not inserted past position 06 and reactor water level cannot be maintained, or suppression pool temperature cannot be maintained less than 110°F.

The SLC system will inject the sodium pentaborate solution in a period of 50 to 120 minutes (by technical specifications) to take the reactor to a cold, xenon free shutdown. However, during the long time delay before shutdown (subcriticality is reached in approximately 20 minutes) fuel damage can occur without supplemental actions to help reduce reactor power.

Another method of reducing reactor power is to lower the reactor water level down to the top of the active fuel. This method takes advantage of the reactor physics of coolant voids. A BWR operates with a large fraction (35% to 40%) of the core occupied by steam voids. The steam voids add negative reactivity to the reactor by reducing neutron thermatization. When the reactor water level is reduced the boiling boundary is lower in the core, increasing the core average void fraction. This action is intended to reduce the reactor power to a manageable level.

To lower the reactor water level to the top of the core the operator reduces or stops injection from HPCI and/or RCIC. When the desired level is reached, the injection flow is stabilized. Stabilizing the reactor water level is difficult because of the pressure oscillations and the absence of S/RV steam flow indication. Adequate core cooling can be achieved if the water level can be maintained above the core midplane, however. The lower portion of the core is cooled by submersion, while the upper portion of the core is cooled by steam flow through the fuel assemblies.

In the course of performing the above mentioned tasks, the plant operators will be inserting control rods using the Reactor Manual Control System (RMCS). The RMCS generates electrical signals for positioning control rods during normal operation. Inserting control rods with RMCS, however, is time consuming and much too slow to be effective by itself. Rods inserted using RMCS takes one second for each notch the rod is inserted. Therefore, to fully insert a rod that is fully withdrawn would take 48 seconds. If the operator uses the notch insert switch to insert rods there is an additional six seconds for the "settle function" before another rod can be selected. The time delay for the settle function can be avoided by using the emergency rod insert which does not energize the settle bus upon completion of rod movement.

To insert control rods using RMCS the Rod Worth Minimizer (RWM) must be bypassed. The RWM prevents control rod movement outside a predetermined pattern. It is more effective to insert high worth rods in the center of the core, but this requires the operator to deviate from the pattern. Bypassing the RWM will prevent the RWM from imposing the rod blocks so that the operator can insert the high worth rods in the center of the core. Deviating from the rod pattern requires close supervision by the nuclear engineer to ensure the rod insertion does not cause fuel damage.

The operator must also take action to defeat the Rod Sequence Control System which is a backup system to the RWM. This may be done by one of two methods. The operator can communicate with an instrument technician at the RSCS electronics cabinet to bypass rod groups as necessary. The operator may also defeat RSCS rod block by two control room switches; the rod mode select switch and the rod sequence select switch. By proper positioning of these switches the operator can defeat the RSCS rod blocks.

### D.6.2. Fuel Cladding

Detecting and evaluating cladding damage is difficult and time consuming when the MSIVs are closed. The operator can, however, determine that cladding damage is possible by monitoring the thermal parameters on the process computer. The critical power ratio is the most important, since it will indicate the transition from nucleate to partial film boiling.

If fuel cladding failure has occurred the fission products can be detected once they reach the containment. The containment atmospheric monitors will alert the operator that fuel damage has occurred if there is a marked increase in airborne radiation levels. The hydrogen analyzer may also support the diagnosis if the failure is the result of cladding oxidation.

If gross cladding failure occurs, coolant sample analysis may be necessary to determine the extent of the failure. The hazards involved in sampling the coolant mandate extreme caution on the part of the laboratory technician.

The operators immediate actions when fuel failure has occurred are to ensure adequate core cooling, to ensure that automatic isolations have occurred, and to prevent the spread of fission products.

Adequate core cooling to preserve cladding integrity is analogous to maintaining fuel integrity. The operator must take actions to reduce reactor power, and ensure sufficient coolant level.

If fuel damage is suspected the operator should verify isolation of (or manually isolate) the following systems:

1. Main steam

2. Off-gas

3. Mechanical vacuum pump

4. Drywell equipment and floor drain sumps

5. Drywell and torus vent lines

6. Reactor building and refuel floor ventilation

Some of these systems can be opened if it is determined that opening the valves or dampers will not spread fission products to other areas of the plant.

If the operator succeeds in scraming the reactor after cladding damage has occurred, the scram should not be reset. This will drain reactor coolant from the scram discharge volume to the reactor building floor drain sump. From the sump, the water will be pumped to the radwaste system. In the process of pumping the sump, plant personnel could receive high radiation doses as the contents of the sump pass through areas of the plant with inadequate shielding.

As soon as the plant is taken subcritical and reactor water level is returned to normal, the operator should return the reactor water cleanup system to operation. The cleanup system will remove fission products from the coolant. This may require the operator to wait until the completion of poison injection because of an interlock between SLC and the cleanup systems which causes a cleanup system isolation during poison injection.

### D.6.3. Reactor Coolant

Since the coolant is not a physical barrier in the same sense as the other barriers, it is difficult to discuss mitigating actions concerning the coolant. However, the importance of the coolant in preserving other barriers and the potential for transporting fission products via the coolant should not be neglected.

The importance of the coolant for controlling fuel, cladding, and reactor vessel temperatures is obvious. Insufficient cooling can lead to catastrophic failure of these vital barriers. To achieve sufficient cooling the operator must not only be concerned simply with maintaining reactor water level but also with controlling reactor pressure, since temperature and pressure are linked in a saturated steam system. In the ATWS with Group I isolation scenario pressure control is achieved by S/RV operation and the operation of HPCI and RCIC. These systems, in turn, depend on the suppression pool for quenching the steam. Because of the interrelationships of all of these systems, multiple equipment failure can have a drastic affect on the operators ability to maintain barriers.

To ensure adequate coolant supply the operator should verify automatic initiation of HPCI and RCIC or manually initiate these systems. Reactor water level should be monitored continuously to ensure adequate core coverage is maintained at all times. Stable reactor water level may be difficult to achieve because of pressure "spikes" and the potential failure of HPCI. The HPCI system fails because of an automatic suction swap over to the suppression pool on high suppression pool level. When the suppression pool temperature reaches 190$^{o}$F (at about 28 minutes) the HPCI turbine fails because of high lubricating oil temperature. Once the HPCI system fails, the only source of makeup coolant to the reactor is from the control rod drive hydraulic pump and the RCIC pump. The lower injection flow will cause a decrease in reactor water level until approximately five feet of the upper core is uncovered. Injection by other systems will require reactor depressurization.

The importance of coolant in controlling power level has already been discussed. Void fraction is an effective means of reducing reactor power. In addition to the effects of water level and pressure on void fraction discussed earlier, coolant flow rate through the core is also an important consideration. The reactor recirculation system is used in normal operation to control reactor power by sweeping the voids higher in the core. Because of the effect that forced circulation has on reactor power, the recirculation pumps are automatically tripped at the beginning of the accident to permit void formation in the core. (The pumps trip on 1120 psig reactor pressure.) This has the immediate effect of reducing reactor power to less than 50%, without any operator action. The operator should verify the recirculation pumps have tripped or manually trip them immediately should the automatic action fail to occur.

A final note on the subject of the coolant concerns the use of SLC. The poison solution is injected into the vessel bottom head region. Proper mixing and circulation of the poison through the core requires the operation of the reactor recirculation pumps. Instead of starting the recirculation pump the operator should raise the reactor water level to increase natural circulation in the reactor. Natural circulation is, however, less effective for mixing the poison.

### D.6.4. Primary System Boundary

During the ATWS or any other accident the operator is concerned with two parameters to ensure primary system integrity; vessel pressure and vessel heatup or cooldown rate. The design pressure on the vessel is 1250 psig. The relief valves are set to open well below this pressure to limit the reactor pressure. Vessel failure, however, should not occur until the safety limit of 1375 psig is exceeded.

The operator has a direct influence on reactor vessel temperature changes through manual operation of S/RV's. The rate of temperature change of the vessel should be limited to 100$^{o}$F/hour. Excessive heatup or cooldown can lead to thermal stresses that may cause vessel failure. Therefore, the operator should use caution when depressurizing the reactor or injecting cold water into the vessel.

Failure of the primary system poses some difficult decisions to the operator during an ATWS. The automatic actions that occur following a loss of coolant in the drywell can accelerate fuel and cladding failure, but defeating these automatic actions could also have the same effect. In the event the system failure is outside primary containment, the potential for off-site release increases significantly. Therefore, whenever there is a failure of the primary system the evacuation of the general population in the vicinity of the plant should be implemented.

Consider first the case of a failure outside primary containment. If the break cannot be isolated, fission products will be released directly into the atmosphere of the reactor building or the turbine building. Gaseous fission products released in the reactor building will be treated by the SGT while liquid leakage will be collected in the reactor building floor drain sump. The effectiveness of the SGT filter train could be diminished if the building atmosphere has high humidity.

In the event the failure is in the turbine building, there is no provision for holding or treating gaseous fission products. The fission products would exit the building through the roof vents. If the gaseous fission products can be confined to the normal path through the main steam system and the main condenser, the condenser off-gas system will provide protection from significant off-site release.

The only course of action the operator can take for a failure outside containment is to verify the SGT is running and to attempt to isolate the break. The high radiation levels will cause the area radiation monitors to alarm. When an alarm is received the operator should announce the condition over the page system. Personnel in the affected area should leave the area until health physics conducts a survey to determine radiological conditions. If personnel must reenter an affected area to perform essential tasks, protective clothing and respiratory protection may be necessary.

Should the primary system failure occur inside primary containment the ECCS will automatically initiate on high drywell pressure or low reactor water level. During an ATWS both these conditions will probably already exist. The reactor will depressurize at a rate proportional to the size of the break. If a rapid depressurization occurs, the low pressure pumps in the RHR, core spray, and condensate systems will inject a high volume of cold water to reflood the reactor. The cold water will cause a rapid increase in power that accelerates fuel and cladding failure. The operator should prevent injection by more pumps than is necessary to hold water level at the top of the active fuel.

The low pressure ECCS and condensate system pumps will also inject into the reactor if automatic depressurization occurs. As S/RVs are opened on overpressure or manual action, the drywell pressure will increase. When the operator lowers the water level to the top of the core the conditions for Automatic Depressurization System (ADS) are satisfied and upon expiration of the 120 second timer the six S/RVs assigned to ADS open. The operator should either stop the low pressure pumps or reset the 120 second timer before expiration to prevent the ADS valves from opening.

When the operator determines that an actual break in the primary system boundary has occurred, the reactor vessel instrumentation should be checked in an effort to determine the location of the break. If the break is in a location that can be isolated, the operator should carefully isolate the break. Isolating a break could cause a sharp increase in reactor pressure. The operator should ensure there is adequate reactor water level before isolating the break to avoid core uncovery.

In the event of severe core melt, the reactor vessel could fail. Analysis indicates that the molten core and vessel internals (e.g. fuel channels, cladding, and fuel supports) would attack the stub tube welds in the lower plenum region of the vessel. The stub tube welds are not the full penetration type, so the presence of molten material can quickly cause a failure of these welds. This will be indicated by rapid increases in drywell pressure, temperature, and radiation levels. The operator should attempt to reflood the reactor using any means available. In addition, measures must be taken to protect the containment vessel from failure.

### D.6.5. Primary Containment

The primary containment is the last high integrity barrier before the environment. The secondary containment is not designed to withstand extremely high concentrations of fission products. Every effort possible should be made to protect the primary containment from excessive temperatures and pressures. The containment is designed for an internal pressure of 56 psig and an internal temperature of 281°F. The design external pressure is 2 psig. The maximum allowable suppression pool water temperature is 110°F when the reactor is critical.

The suppression pool water temperature is the immediate concern of the operator at the beginning of the accident. The frequent S/RV operation will cause the pool temperature to increase rapidly. The operator will immediately place the RHR system in the suppression pool cooling mode in an attempt to hold the pool temperature down. This may be difficult, however, if reactor power remaining higher than the RHR systems capacity to remove the heat from the suppression pool. To avoid violation of the torus heat capacity curve contained in the Emergency Procedure Guidelines, emergency reactor depressurization will be necessary.

The need for emergency depressurization is determined by comparing reactor pressure with suppression pool temperature. The ability of the suppression pool to quench the steam from S/RV discharge or LOCA blowdown decreases as the pool temperature increases. At high suppression pool temperatures which would occur during an ATWS, the operator must depressurize the reactor below 135 psig. At this low reactor pressure the steam discharged through the S/RV's transfers less energy to the suppression pool. The suppression pool cooling mode of RHR can maintain the pool temperature when this action is taken, thereby preventing containment failure from overpressurization.

However, the SASA analysis indicates that emergency depressurization is not recommended while the reactor remains critical during the ATWS. Several relief valves cycle open and closed on overpressure. If the operator opened a valve that was already open there would be no effect on reactor pressure. If a closed S/RV is opened, there will be a negligible change in reactor pressure when one of the valves already open on overpressure closes. Manually opening another valve could cause a rapid decrease in reactor pressure permitting the low pressure ECCS to inject into the reactor.

Emergency depressurization of the reactor also effects the reactor physics. At low pressure small changes in steam vapor specific volume for a given change in pressure can add positive reactivity to the reactor. Controlling reactor pressure under these conditions would be difficult.

Procedures require the operator to spray the containment using the RHR system when containment pressure reaches 22.5 psig or drywell atmosphere temperature is greater than 281$^{o}$F for 30 minutes. Prior to spraying the drywell, all electrical equipment in the drywell (recirculation pumps and drywell cooler fans) should be deenergized to prevent electrical arcing. If the RHR system is not available for containment spray, the RHR service water system may be cross-connected to spray the RHR system for containment spray. Any time the containment is sprayed both the drywell and suppression pool must be sprayed to minimize the internal to external pressure differential to less than 2 psig.

If core melt and reactor vessel failure is suspected, containment flooding should be considered. Containment spray alone may not be sufficient to prevent damage to the containment vessel. Containment flooding presents some special problems that make this a last resort effort. If the water level in the containment is raised too high, the low point in the containment could exceed the safety limit on pressure and subsequently rupture due to the static head of the water. The operator should consult with engineering personnel before flooding the containment. The containment vessel may also be overpressurized due to compression of the drywell atmosphere as the containment is flooded. Gases must be vented from the upper portion of the drywell to prevent this from occurring. Venting the drywell through the SGT, however, will require defeating isolation interlocks so that the vent valve can be opened. This is not recommended unless absolutely necessary. If containment venting is required, the atmosphere should be sampled and analyzed prior to venting.

During an accident where fuel damage is suspected or confirmed, the operators must be aware of the possibility of a hydrogen explosion in the containment. The containment is filled with nitrogen during normal operation to keep the oxygen concentration below the combustible mixture. During the accident, if the oxygen concentration approaches 4% by volume, the operator should place the Containment Atmospheric Dilution (CAD) system in operation. The CAD system will supply a large volume of nitrogen to the drywell and suppression pool to dilute the concentration of oxygen inside primary containment. Some plants have installed hydrogen recombiners to remove the hydrogen should the metal-water reaction occur.

### D.6.6. Secondary Containment

If the accident progresses to the point of primary containment failure, the operator cannot prevent the release of fission products to the environment. However, maintaining secondary containment integrity will minimize the ground level release and provide a means of elevated release through the standby gas treatment system.

The failure of primary containment will be indicated by rapidly increasing radiation levels in the reactor building. The area radiation monitors in secondary containment will alarm in the control room to alert the operator of the failure. When this occurs a site evacuation will be initiated. The control room will remain staffed, however, by placing the control bay ventilation system in the pressurization mode.

### D.6.7. Secondary Containment Ventilation System

The operators will closely monitor the reactor building differential pressure to ensure the SGT is maintaining the -0.25" $H_20$ pressure in the building. If this pressure differential is lost, this would indicate either a failure of the SGT or excessive leakage into secondary containment. In either case the emergency response team should be notified to expedite the off-site evacuation.

If the SGT has failed, every effort must be made to restore operation. If another SGT is in standby it should be started to minimize ground level releases until full capacity can be restored.

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Engineering Physics and Mathematics Division
Oak Ridge National Laboratory
P. O. Box X
Oak Ridge, Tennessee 37830
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General Physics Corp.
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13. ABSTRACT (200 words or less)

This report describes a human factors research project performed to: (1) support the Severe Accident Sequence Analysis (SASA) program and (2) develop a descriptive model of operator response in accident management. The first goal was accomplished by working with SASA analysts on the Browns Ferry Unit One anticipated transient without scram (ATWS) accident sequence to systematically assess critical operator actions and thereby demonstrate contributions to SASA analyses from human factors data and methods. The second goal was accomplished by developing a model called the Function Oriented Accident Management (FOAM) model, which provides both a conceptual structure linking off-normal safety functions with potential unconventional emergency responses and a method for developing technical guidance for those responses based on operations, engineering, and human factors data and expertise. The four components comprising the model are described and their use is shown through a table-top demonstration.

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