TMI-1

SAFETY PARAMETER DISPLAY

SYSTEM SAFETY ANALYSIS

TOPICAL REPORT 018

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SUMMARY

A Critical Safety Functions (CSF) Approach was used as the basis for a Safety Parameter Display System (SPDS). The objectives of the SPDS were formulated into five (5) CSFs. Parameters were then selected for each CSF which would satisfy the objectives of that CSF. These parameters will be used to activate alarms in the existing plant computer alarm processor. The CSF were then tested on a broad range of transient scenarios to check their response. The final parameter list (Table 3.1) was sufficient to alert the user to conditions which fell outside the expected transient response. It is therefore believed that the CSFs and their parameter sets form a complete set that should meet the objectives of the SPDS during power operation and shutdown modes. While we feel the parameter set presented in this report is complete, we plan to evaluate whether any changes will be required to the set based on experience gained by developing the user guidelines, CRT displays and experience gained once the SPDS is implemented.

1.0 INTRODUCTION

The lessons learned from the Three Mile Island Unit 2 accident highlighted the inadequacies of the emergency procedure philosophy and event diagnostic techniques in place at that time. The TMI-1 procedure philosophy is changing to a symptom oriented approach rather than the previously used event oriented approach. This new approach will allow.

1. A determination as to the overall condition of the plant and identification of abnormal plant conditions.
2. Action to be taken to mitigate abnormal plant conditions without knowing the initiating cause.

The Safety Parameter Display System (SPDS) has been developed as an aid to the control room personnel in diagnosing abnormal conditions and determining if the actions taken by the operators have brought the plant to a stable and safe condition. The SPDS will provide the user with concise and unambiguous information relating to the safety status of the plant. The SPDS is not meant to be the sole or even the primary means for the control room personnel to obtain this information. The primary means of determining the safety status of the plant is by using the information provide on the operating consoles located in the control room. The SPDS does provide a secondary station where the parameters required for determining the plant safety status have been gathered in one location. Since the SPDS is a very broad based diagnostic tool, it is designed for use by the control room personnel who are trying to grasp and maintain an

overview of the plant safety status. Thus the primary users of the SPDS will be the Shift Technical Advisor (STA) and the Shift Supervisor (SS). These personnel can use the SPDS to advise or direct the operating crew to take further action or determine if the action which has been taken is appropriate. Due to the intent and design of the SPDS the control room operator at the controls should rely on his operating console indication for plant control.

The approach used in designing the SPDS was to select a set of Critical Safety Functions (CSF) which would describe the safety status of the plant. These CSF apply to all plant operating modes but will not be implemented during cold shutdown and refueling. Four methods of determining CSF were evaluated before the final set was chosen. Once the CSF where chosen then a parameter set was selected that would reflect the status of each CSF. These parameter sets were then tested by applying them to a wide variety of transient/accident scenarios.

To obtain the maximum benefit from the SPDS the users must receive proper training on SPDS philosophy, design, and use.

2.0 LITERATURE REVIEW

The available literature on this subject predates the release of NUREG-0737 Supplement 1 on December 17, 1982. The purpose of this review is to highlight the basic methodologies used and how they compare with the approach used in this report.

NSAC/55 provides information on two studies performed to develop an SPDS for a PWR. The report references a Minimum PWR Safety Panel Parameter Set (Table 2.1) developed by the Atomic Industrial Forum, Inc. The other study was performed for the Yankee Rowe Plant. The critical safety functions and parameter set for this SPDS is provided in Table 2.2.

NSAC/10 provides information on a study done to develop a generic PWR parameter set. The final reduced parameter set from this report is provided in Table 2.3.

NUREG-0737 Supplement 1, Section 4.0 requires information to be provided to the operator on:

1. Reactivity Control
2. Reactor Core Cooling and Heat Removal from the Primary System

3. Reactor Coolant System Integrity

4. Radiation Control

5. Containment Conditions.

The above NRC functions were used as guidelines in generating the CSFs for TMI-1.

TABLE 2.1

ATOMIC INDUSTRIAL FORUM INC.

MINIMUM PWR SAFETY PANEL PARAMETER SET

1. Reactivity Control
    - A. Power Range
    - B. Intermediate Range
    - C. Source Range

2. Reactor Core Heat Removal
    - A. Pressurizer Level
    - B. Primary System Pressure
    - C. Hot Leg or Core Exit Thermocouple Temperature
    - D. Cold Leg Temperature

3. Secondary Heat Removal
    - A. Steam Generator Level
    - B. Steam Generator Pressure
    - C. Feedwater Flow

4. Primary Coolant Inventory
    - A. Pressurizer Level
    - B. Core Exit Thermocouple Temperature
    - C. Containment Sump Level
    - D. Containment Temperature

4. Containment Integrity

   A. Containment Pressure

   B. Containment Temperature

   C. Containment Dome Radiation Monitor

   D. Condensor Air Ejector Radiation Mointor

TABLE 2.2

YANKEE ATOMIC ELECTRIC COMPANY

SPDS PARAMETER SET FOR THE YANKEE ROWE PLANT

1. Reactivity Control
   - A. Source Range
   - B. Intermediate Range
   - C. Power Range

2. Core Heat Removal
   - A. Pressurizer Level
   - B. Main Coolant Sy,tem Pressure
   - C. Temperature Core Exit
   - D. Coldleg Temperature (lowest)
   - E. Coldleg Temperature (highest)

3. Secondary Heat Removal
   - A. Steam Generator Pressure
   - B. Steam Generator Level
   - C. Steam Flow
   - D. Feedwater Flow
   - E. Emergency Feedwater Flow
   - F. Steam Line Radiation

4. Main Coolant Inventory

   A. Main Coolant System Pressure

   B. Pressurizer Level

   C. Condenser Air Ejector Radiation

   D. Vapor Containment Flood Level

   E. Vapor Containment Pressure

   F. Vapor Containment Air Particulate

5. Containment Integrity

   A. Vapor Containment Pressure

   B. Vapor Containment Temperature

   C. Vapor Containment H_2 Concentration

   D. Vapor Containment High Range Radiation

   E. Primary Vent Stack Radiation

TABLE 2.3

PROPOSED FUNDAMENTAL VARIABLE LIST FOR A PWR SAFETY CONSOLE

1. Reactivity
   - A. Neutron Flux
   - B. Boron Concentration
   - C. Control Rod Position

2. Heat Removal
   - A. Core Exit Temperature
   - B. Hot Leg Temperature
   - C. Cold Leg Temperature
   - D. Reactor Coolant Pressure
   - E. Coolant Subcooling
   - F. Reactor Vessel Liquid Level
   - G. Primary Coolant Flow
   - H. RHR Flow
   - I. Neutron Flux
   - J. Steam Generator Level
   - K. Steam Generator Pressure
   - L. Main/Aux Feedwater Flow

3. Coolant Inventory

   A. Reactor Vessel Liquid Level

   B. Pressurizer Level

   C. Liquid Volume Control Tank Level

   D. Letdown Flow

   E. Pump Seal Return Flow

   F. Quench Tank Level

   G. Emergency Sump Level

   H. ESF Status

   I. PORV Flow Rate

4. Primary Pressure Boundary Integrity

   A. Pressurizer Level

   B. Letdown Flow

   C. Pump Seal Return Flow

   D. Quench Tank Level

   E. Quench Tank Pressure

   F. Pressurizer Pressure

   G. Emergency Sump Level

   H. ESF Status

   I. Containment Pressure

   J. Containment Temperature

   K. Containment Radioactivity

   L. Primary Coolant Radioactivity

   M. RHR Radioactivity

   N. Condenser Evacuator Radioactivity

5. Containment Integrity

   A. Containment Pressure

   B. Containment Temperature

   C. Containment Radioactivity

   D. Main Stack Gross Gamma

   E. Containment Hydrogen Concentration

   F. RHR Radioactivity

   G. Condenser Evacuator Radioactivity

3.0 SPDS PARAMETER IDENTIFICATION

The first step in the identification process was to define the objectives of the SPDS. Once the objectives were defined the users who could best accomplish these objectives were selected from a list of potential users.

The next step was to select a set of Critical Safety Functions (CSF) which satisfy the SPDS objectives and the requirements of NUREG-0737 Supplement 1. A parameter set was then selected for each CSF which reflected the status of that function. Users' guidelines will be developed to aid the user in evaluating and dealing with normal and abnormal conditions. These guidelines will also serve as the basis for developing CSF displays and for training the SPDS users. The first four steps of this process are described in this report.

3.1 SPDS Objectives

1. The SPDS should provide concise and unambiguous information which can be interpreted easily and quickly during stressful situations.

2. At a minimum, the information provided by the SPDS should reflect the plant safety status during power operation and post trip modes.

3. The level of detail provided in the SPDS should be for overview purposes.

4. The SPDS will be a diagnostic tool for short term safety concerns not long range safety concerns.

5. A minimum set of parameters must be chosen that would uniquely describe each critical safety function.

6. User guidelines should be developed for purposes of parameter selection, display logic generation, user training, and procedure development.

7. The SPDS should be tested against a variety of transient/accident scenarios to evaluate the parameters chosen for each critical safety function.

3.2 User Selection

The following groups were considered when trying to select the users of the SPDS.

1. Control Room Operators (CRO)
2. Shift Foreman (SF)
3. Shift Supervisor (SS)
4. Shift Technical Advisor (STA)

5. Technical Support Center Engineers (TSCE)

6. Parsippany Technical Support Center Engineers (PTSCE)

7. Emergency Offsite Facility Engineers (EOFE)

During the selection of the SPDS users, consideration was given to the number of personnel normally in the control room, their normal and emergency work stations, the shift structure, individual responsibilities, and responsibility of emergency support centers.

The two groups who can best utilize the SPDS in both day to day operation and during transients/accidents are the Shift Technical Advisors and Shift Supervisors. These two groups need to maintain an overview of the plant safety status and can use the SPDS to evaluate the plant response to control actions. The emergency response centers may be able to use portions of the SPDS but since their main goal is to analyze the long term safety concerns they are not considered primary users. These secondary user centers will have access to all the parameters used by the SPDS but the parameters will not be presented in the identical manner as in the control room.

3.3 Critical Safety Functions

After investigating four different approaches to selecting CSF, a combination of the "systems" and "barrier" approach was selected. This is the same approach selected for NUREG-0737 Supplement 1. This approach led to the following CSFs:

1. Reactivity/Power Distribution
2. Primary Side Heat Removal
3. Reactor Coolant System Integrity
4. Radiation Control
5. Containment Conditions

3.4 Parameter Selection

Each CSF will be discussed separately. The parameters will be selected to provide useful and unambiguous information to the user. The selection process will be an iterative one as the parameter sets are compared against the transient/accident scenarios, preliminary user guidelines, and the Abnormal Transient Operator Guidelines (ATOG).

3.4.1 Reactivity Control

When the plant is at steady state power operation the Reactivity Control CSF will monitor for power peaking and DNBR. Since power peaking and DNBR are not directly measurable, they will be monitored by using other parameters that will then bound the power peaking and DNBR concerns.

During all modes the core startup rate will be monitored to determine gross reactivity concerns.

Reactor power will also be monitored during power operation and post trip to determine if an ATWS or power excursion has occurred.

The required parameters are:

1. RC Total Flow
2. Reactor Trip Signal
3. Core Power (Heat Balance)
4. Source Range Startup Rate
5. Intermediate Range Startup Rate
6. Power Range Power
7. Power Range Imbalance

3.4.2 Primary Side Heat Removal

This CSF will monitor the heat removal characteristics from the core to a heat sink. Normally the heat sink will be the secondary side of the steam generators. Under certain conditions other heat removal methods may be necessary such as HPI cooling.

During the power operation and post trip modes, monitoring will be performed using a plot of pressure versus temperature. This will allow a determination of excessive or inadequate cooling along with pressure and inventory control problems.

The main feedwater flow and reactor power will also be compared to monitor the heat removal versus heat generation and the heat removal relationship between each steam generator.

The required parameters are:

1. Core Power (Heat Balance)
2. RCS Wide Range Pressure
3. RCS Wide Range Hot Leg Temperature
4. Main Feedwater Flow
5. Steam Generator Pressure
6. RCS Wide Range Cold Leg Temperature
7. Reactor Vessel Water Level
8. Incore Thermocouple Temperature
9. Reactor Trip Signal
10. Reactor Coolant Pump Status
11. Heatup/Cooldown Rate
12. RC Saturation Temperature Margin

3.4.3 Reactor Coolant System Integrity

This CSF will use diverse means to monitor for steam generator tube leakage, leakage into the reactor building, and leakage into the Intermediate and Decay Heat Closed Cooling Water Systems. The reactor coolant water level is also monitored when in the Natural Circulation mode of cooling, and void fraction is monitored when in the forced flow mode of cooling.

The required parameters are:

1. RM-A5 (Condenser Exhaust)
2. RM-G26 (A Loop Steam Relief)
3. RM-G27 (B loop Steam Relief)
4. RM-A2 Gas (Reactor Building Atmosphere)
5. RM-G8 (Reactor Building Dome)
6. Reactor Building Sump Level
7. RM-G25 (Condenser Exhaust)
8. Letdown Flow
9. Pressurizer Level
10. Makeup Flow
11. High Pressure Injection Flow
12. RM-L9 (Intermediate Closed Cooling Water System)
13. RM-L2 (A Loop Decay Heat Closed Cooling Water System)
14. RM-L3 (B Loop Decay Heat Closed Cooling Water System)
15. Hot Leg Water Level
16. Reactor Vessel Water Level
17. Incore Thermocouple Temperature
18. Void Fraction
19. Reactor Trip
20. Reactor Coolant Pump Status
21. Heatup/Cooldown Rate

3.4.4 Radiation Control

This CSF is used to monitor for concerns regarding the potential safety of plant workers and the public. Fuel failure is monitored to alert for the potential increase in levels of radioactivity in the plant. The Air monitors are used to alert for the unplanned release of radioactivity to the environment.

The required parameters are:

1. RM-L1 Lo (Letdown)
2. RM-A6 Gas (Auxiliary Building Atmosphere)
3. RM-A4 Gas (Fuel Handling Building Atmosphere)
4. RM-A8 Gas (Plant Stack)
5. RM-A9 Gas (Reactor Building Stack)

3.4.5 Containment Conditions

This CSF will monitor the reactor building to minimize the formation of an adverse environment. This will decrease the probability of the reactor building integrity from being breached. It will also alert the user to a potential increase in instrument error caused by the adverse environment.

The required parameters are:

1. Reactor Building Pressure
2. Reactor Building Temperature
3. RM-A2 Gas (Reactor Building Atmosphere)
4. RM-G8 (Reactor Building Dome)
5. Reactor Building Hydrogen Concentration
6. Reactor Building Flood Level
7. Reactor Building Sump Level

3.5 Use of SPDS with Abnormal Transient and Emergency Procedures

Since the plant procedures are used by the operator at the controls to control the plant they are more detailed than the SPDS. The SPDS displays and user guidelines will be designed to guide the user to specific Abnormal Transient or Emergency Procedures for the detailed response. While developing the SPDS, areas may be found which are not adequately covered by the plant procedures. If this happens, guidelines will be developed and incorporated into the appropriate area (i.e. procedures, training, user guidelines).

3.6 Comparison with Other Parameter Sets

In order to compare the TMI-1 CSF parameter set with those discussed in Section 2, Table 3.1 has been re-arranged in terms of the CSF and placed in the same format as Tables 2.1, 2.2, and 2.3. The following is a comparison of the TMI-1 parameter set against the three sets given in Section 2.0.

1. Reactivity Control:

   The TMI-1 parameter set is more detailed to incorporate its use during power operation. When below the power range, the start up rate is used to monitor the core criticality conditions. Start up rate allows for easier and quicker interpretation of the core conditions as does power level. When in the power range, both start up rate and power level are used.

2. Heat Removal:

   All the parameter sets for this CSF are very similar. TMI-1 has added a few parameters to allow for monitoring during steady state power operation. We also choose not to use the pressurizer level and steam generator level in this CSF. Pressurizer level is not used since it does not provide useful unambiguous information on the heat removal

characteristics of the system. The core can continue to be cooled even if the pressurizer is empty (i.e. heat removal via break) and likewise the pressurizer may have adequate level but not adequate heat removal from the core (i.e. TMI-2 accident). In the same manner the steam generator level does not provide useful unambiguous information concerning heat removal. Adequate heat removal can be accomplished using auxiliary feedwater with no steam generator level and the fact that water level exists does not mean heat removal is adequate (i.e. primary and secondary are uncoupled). The TMI-1 CSF provides adequate parameters to determine heat removal conditions without using pressurizer and steam generator level.

3. Reactor Coolant System Integrity:

All of the parameter sets reviewed had different ideas of what parameters were required for this CSF. The set discussed in Table 2.3 even broke it into two CSFs. Table 2.1 is mainly monitoring for leaks into the reactor building and core uncovery. Table 2.2 looks for leaks in the reactor building and steam generators. Table 2.3 provides parameters to try and do a detailed mass balance on the reactor coolant system. The TMI-1 parameters use a

simplified mass balance and radiation monitors to determine leakage into the reactor building, steam generators, and auxiliary systems. Parameters are also provided to monitor the reactor coolant system water level from the top to bottom of the hot legs. Incore thermocouples are used to determine core uncovery. Thus the TMI-1 parameter set is more comprehensive than the others reviewed.

4. Radiation Control:

   The TMI-1 parameter set incorporates a Radiation Control CSF, which is not found in any of the other three. This CSF does meet the NUREG-0737 Supplement 1 requirements.

5. Containment Conditions:

   All of the parameter sets are similar in the selection of parameters for this CSF. The concerns covered by all the sets reviewed include pressure, temperature, explosive gases, water and radiation. The TMI-1 set also addresses all of these concerns.

3.7 Alarm Criteria

Each Critical Safety Function will have a Priority 1 and 2 alarm added to the existing alarm processor in the TMI plant computer. If a parameter on the CSF reaches a predetermined warning (Priority 2) or alarm (Priority 1) setpoint, it will activate the CSF alarm which will be displayed on a CRT in the control room. The parameter in alarm will also be identified as alarming on the SPDS display. This will allow the computer to continuously monitor the SPDS and alert the user when a setpoint has been reached. This also allows for the SPDS to be integrated with the existing plant computer functions increasing the reliability of the user machine interface.

TABLE 3.1

SAFETY PARAMETER DISPLAY SYSTEM

PARAMETERS	NOTES	Critical Safety Functions 1	2	3	4	5
REACTOR TRIP SIGNAL	3	X	X	X		X
CORE POWER (HEAT BALANCE)	3	X	X			
POWER RANGE POWER	3	X				
POWER RANGE IMBALANCE	3	X				
SOURCE RANGE START UP RATE	3	X				
INTERMEDIATE RANGE START UP RATE	3	X				
RCS WIDE RANGE PRESSURE	3		X			
STEAM GENERATOR PRESSURE	3		X			
REACTOR BUILDING PRESSURE	3					X
RC SATURATION TEMPERATURE MARGIN	3		X			
RCS WIDE RANGE COLD LEG TEMPERATURE	3	X	X	X		
RCS WIDE RANGE HOT LEG TEMPERATURE	3		X			
INCORE THERMOCOUPLE TEMPERATURE	3		X			
REACTOR BUILDING TEMPERATURE	1					X
HEATUP/COOLDOWN RATE	3		X	X		
REACTOR BUILDING SUMP LEVEL	3			X		X
REACTOR BUILDING FLOOD LEVEL	3					X
REACTOR VESSEL WATER LEVEL	2		X	X		
HOT LEG WATER LEVEL	2			X		
PRESSURIZER LEVEL	3			X		
RCS TOTAL FLOW	3	X				
RCS LETDOWN FLOW	3			X		
RCS MAKEUP FLOW	3			X		
HIGH PRESSURE INJECTION FLOW	1			X		
MAIN FEEDWATER FLOW	3		X			
REACTOR COOLANT PUMP STATUS	3	X	X	X		
VOID FRACTION	2			X		
REACTOR BUILDING HYDROGEN MONITOR	3					X
RM-L1 Lo (Letdown)	1				X	
RM-L2 (A Loop Decay Heat Closed)	1			X		
RM-L3 (B Loop Decay Heat Closed)	1			X		
RM-L9 (Intermediate Closed)	1			X		
RM-A2 GAS (RB Atmosphere)	1			X		X
RM-A4 GAS (FHB Atmosphere)	1				X	
RM-A5 (Condenser Exhaust)	1			X		
RM-A6 GAS (Aux Bldg Atm)	1				X	
RM-A8 GAS (Plant Stack)	1				X	
RM-A9 GAS (RB Stack)	1				X	
RM-G8 (Reactor Building Dome)	1			X		X
RM-G25 (Condenser Exhaust)	1			X		
RM-G26 (A Loop Steam Relief)	1			X		
RM-G27 (B Loop Steam Relief)	1			X		

TABLE 3.1 (Cont'd)

CRITICAL SAFETY FUNCTIONS

1. Reactivity/Power Distribution
2. Primary Side Heat Removal
3. Reactor Coolant System Integrity
4. Radiation Control
5. Containment Conditions

NOTES

1. Parameters presently available in the control room but not available on the plant process computer.
2. Parameters which are not presently installed at TMI #1.
3. Parameters which are presently available in the Plant Process Computer Data Base.

TABLE 3.2

TMI-1 SAFETY PARAMETER SET

1. REACTIVITY/POWER DISTRIBUTION

   A. Source Range Startup Rate
   B. Intermediate Range Startup Rate
   C. Core Power (Heat Balance)
   D. Power Range Power
   E. Power Range Imbalance
   F. Reactor Trip Signal
   G. RC Total Flow
   H. Reactor Coolant Pump Status
   I. RCS Wide Range Cold Leg Temperature

2. PRIMARY SIDE HEAT REMOVAL

   A. Core Power (Heat Balance)
   B. RC Saturation Temperature Margin
   C. RCS Wide Range Pressure
   D. RCS Wide Range Hot Leg Temperature
   E. RCS Wide Range Cold Leg Temperature
   F. Main Feedwater Flow
   G. Steam Generator Pressure
   H. Reactor Vessel Water Level
   I. Incore Thermocouple Temperature

J. Reactor Trip Signal

K. Reactor Coolant Pump Status

L. Heatup/Cooldown Rate

3. REACTOR COOLANT SYSTEM INTEGRITY

A. RM-A5 (Condenser Exhaust)

B. RM-G26 (A Loop Steam Relief)

C. RM-G27 (B Loop Steam Relief)

D. RM-A2 Gas (Reactor Building Atmosphere)

E. RM-G8 (Reactor Building Dome)

F. Reactor Building Sump Level

G. RM-G25 (Condenser Exhaust)

H. Letdown Flow

I. Pressurizer Level

J. Makeup Flow

K. High Pressure Injection Flow

L. RM-L9 (Intermediate Closed Cooling Water System)

M. RM-L2 (A Loop Decay Heat Closed Cooling Water System)

N. RM-L3 (B Loop Decay Heat Closed Cooling Water System)

O. Hot Leg Water Level

P. Reactor Vessel Water Level

Q. Void Fraction

R. Reactor Trip

S. Reactor Coolant Pump Status

T. Heatup/Cooldown Rate

U. RCS Wide Range Cold Leg Temperature

4. RADIATION CONTROL

A. RM-L1 Lo (Letdown)

B. RM-A4 Gas (Fuel Handling Building Atmosphere)

C. RM-A6 Gas (Auxiliary Building Atmosphere)

D. RM-A8 Gas (Plant Stack)

E. RM-A9 Gas (Reactor Building Stack)

5. CONTAINMENT CONDITIONS

A. Reactor Building Pressure

B. Reactor Building Temperature

C. RM-A2 Gas (Reactor Building Atmosphere)

D. RM-G8 (Reactor Building Dome)

E. Reactor Building Hydrogen Concentration

F. Reactor Building Flood Level

G. Reactor Building Sump Level

H. Reactor Trip Signal

4.0 SPDS RESPONSE TO TRANSIENTS/ACCIDENTS

Once the parameter identification process was complete, 9 transient/accident scenarios were selected to validate the parameter selection process. Some of these scenarios were developed for the TMI-1 Abnormal Transient Operator Guidelines (ATOG). These guidelines provide the basis from which the symptom oriented emergency procedures were developed. The rest of the scenarios are taken from actual data gathered during the events at TMI-2. The following is a list of the 9 transient/accident scenarios.

1. Excessive Main Feedwater
2. Loss of Main Feedwater
3. Steam Generator Tube Leak/Rupture
4. Loss of Offsite Power
5. Small Steam Leak
6. Loss of Coolant Accidents
7. TMI-2 Accident
8. TMI-2 Loss of One Main Feedwater Pump
9. TMI-2 Turbine Trip Test (No Reactor Trip)

The SPDS critical safety functions and their associated parameters were applied to the above listed transients/accidents. The objectives of this review were:

1. To determine if the SPDS selected parameters were adequate to detect and categorize each transient/accident.

2. To determine if the SPDS selected parameters would alert the user to any deviations from the expected plant response.

3. To determine if the SPDS selected parameters would allow the user to assess the adequacy and correctness of the operators response to the transient/accident.

4. To determine if additional points should be added or existing points should be deleted from the parameter list and still achieve the first 3 objectives.

The SPDS could not identify the initiating event (i.e. Loss of Offsite Power) under all circumstances, but it was able to identify all the symptoms of that event (i.e. Inadequate Heat Removal). This is the same philosophy used by the symptom oriented emergency procedures. The user response is symptom oriented and independent of the initiating event. Since the TMI-1 SPDS was able to identify all the symptoms of each initiating event for the transient/accident scenarios reviewed, the first objective was satisfied.

The TMI-2 accident is used as an example of the symptom oriented event identification method used during this investigation. The combination of steam generator pressure, reactor inlet temperature, outlet temperature, and reactor coolant pressure can be used to identify the loss of heat sink which occurred when the main feedwater

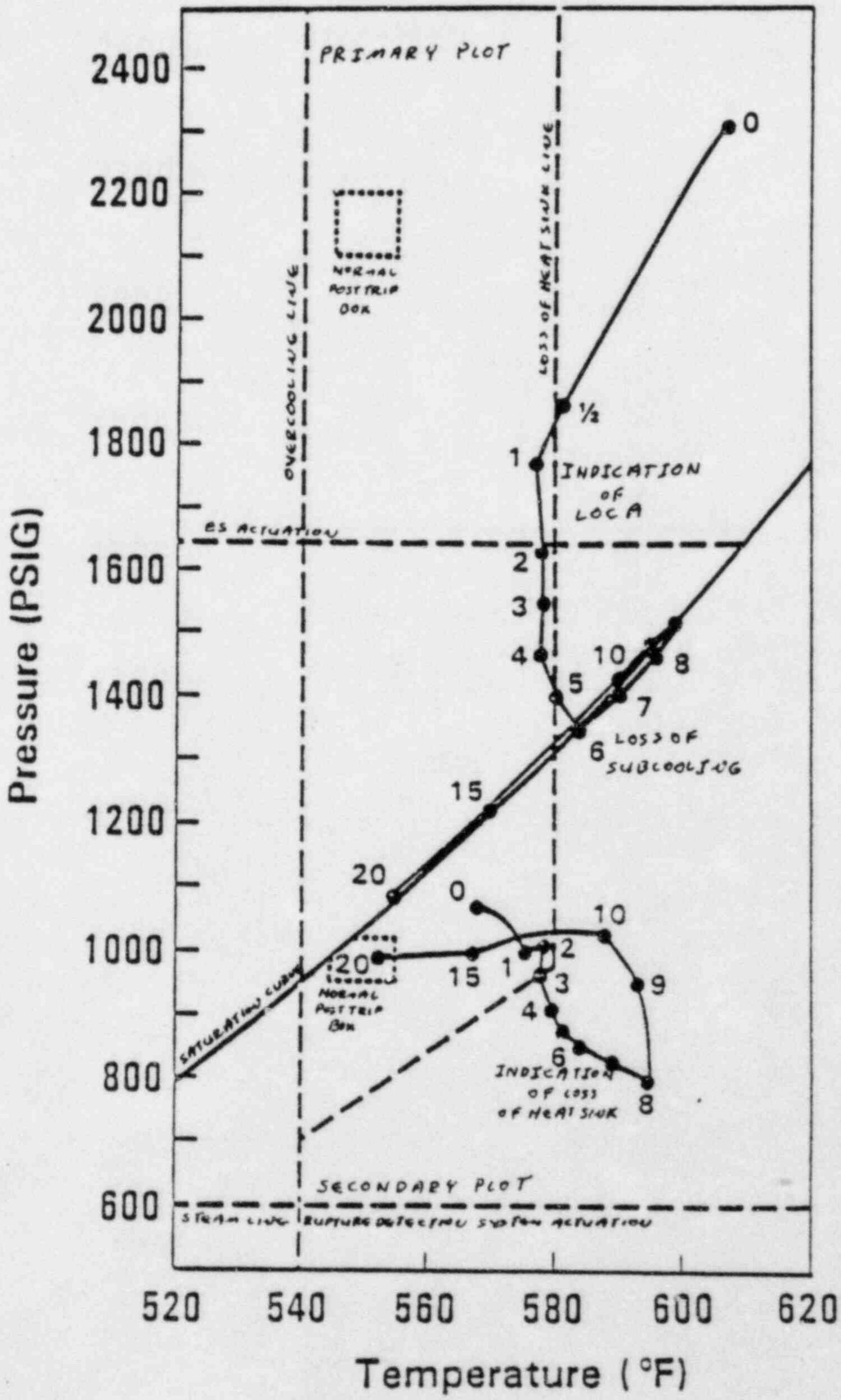
pumps tripped and flow from the emergency feedwater pumps was blocked by closed isolation valves. The same parameters can be used to identify the LOCA and Loss of Subcooling which were caused by the stuck open PORV. Figure 4.1 shows the TMI-2 accident response on the post trip primary side heat removal critical safety function display. This display has been implemented in the TMI-1 control room for over a year. The LOCA would also have been identified by the Reactor Coolant System Integrity CSF by using the makeup flow, HPI flow, letdown flow, heatup cooldown rate, pressurizer level, and reactor building sump level indications.

The second objective was also met for all the transient/accident scenarios reviewed. Again, we will use the TMI-2 accident as an example of how the SPDS could alert the user to deviations from the expected response. When the operators discovered the blocked emergency feedwater flow, they reinitiated flow to the steam generators and restored cooling to the primary side. This is shown in Figure 4.1 by the secondary plot moving in to the normal post trip box. The user would then expect the primary plot to also move toward the normal post trip box. This does not occur indicating that the plant is still deviating from the normal post trip response. Later in the accident two other deviations would have been identified by the TMI-1 SPDS. The Radiation Control CSF would have identified the increasing radioactivity levels and the Containment Condition CSF would have identified the increasing hydrogen concentration levels in the reactor building.

The third objective was met by all the transient/accident scenarios reviewed. The re-establishing of feedwater to the steam generators in the TMI-2 accident was identified by the SPDS to be the correct action (secondary plot went to the normal box) but the operators did not take correct actions to eliminate the LOCA and restore the primary plot to the normal box. Once the reactor coolant pumps were tripped, the Reactor Coolant System Integrity CSF would have warned the user that inadequate steps were being taken to keep the core covered with water. This would have been indicated by the incore thermocouple temperatures, hot leg water level and reactor vessel head water level parameters.

After applying the TMI-1 SPDS to the listed transient/accident scenarios, no new parameters were identified which should be added to the parameter list. Also no parameters were identified which should be deleted from the list. It is believed that the TMI-1 critical safety functions and their associated parameters represent a complete list of parameters required to adequately and correctly display the safety status of TMI-1 to the user.

Figure 4.1
TMI-2 Accident



NOTE: NUMBERED DOTS INDICATE TIME SINCE REACTOR TRIP

5.0 REFERENCES

5.1 A Review of the Three Mile Island Unit 1 Control Room from a Human Factors Viewpoint.

5.2 Guidelines for an Effective Safety Parameter Display System Implementation Program. NUTAC, January 1983.

5.3 Requirements for Emergency Response Capability. NUREG-0737, Supplement 1.

5.4 A Parameter Set for a Nuclear Plant Safety Console. NSAC/10, November 1980.

5.5 Safety Parameter Display System for the Yankee Atomic Electric Company. NSAC/55, August 1982.

5.6 STA Duties and Responsibilities - GPUN Procedure ES 005.

5.7 A Real Time Method for Analyzing Nuclear Power Plant Transients. Broughton, T. G. and Walsh, P. S.

5.8 Abnormal Transient Operating Guidelines (ATOG) Three Mile Island Unit 1.

5.9 Analysis of Three Mile Island Unit 2 Accident. NSAC-1, March 1980.

5.10 Three Mile Island Unit 2 Reactor Trip Report #13. November 7, 1978.

5.11 Annotated Sequence of Events. March 28, 1979. GPUN TDR 044. June 1981.

5.12 Three Mile Island Unit 2 Initial Startup Report.