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TOLEDO
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Serial No. 653

September 29, 1980

Samuel J. Chilk
Secretary of the Commission
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Chilk:

Toledo Edison appreciates the solicitation of comments on NUREG-0696, Functional Criteria for Emergency Response Facilities as announced in the Federal Register of August 15, 1980 (45 F.R. 54708). We feel, by sharing the comments attached hereto, a valuable insight can be gained into developing and implementing functional criteria that will truly improve the capability of utilities and regulators in upgrading their emergency preparedness posture.

Toledo Edison has dedicated all levels of management attention to emergency preparedness, an area that we feel has been one of our most productive efforts in light of the Three Mile Island Unit 2 accident in March of 1979. As a result, a functional criteria has been developed and is currently in the process of implementation. The resolution of pitfalls and the problems of integrating such a comprehensive response capability into an organizational structure has led to the development of a truly effective emergency response capability in our company.

Due to our commitment and an accelerated schedule, our facility is to be in full operation by late Summer 1981. In light of this, we would be happy to make the facilities available to the Commissioners and their staff to observe first hand its role in our overall emergency response program.

In retrospect, the development of our functional criteria has been both difficult and exhilerating. Our conceptual development was aided greatly by the insights of a Toledo Edison Review Team that went to Three Mile Island in July of 1979. They gathered first hand emergency response experiences of the Metropolitan Edison/General Public Utilities organization.

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The emergency capabilities of the utility, industry, local, state and federal organizations have to function to direct all the resources at their command to insure the public health and safety. The purposes of emergency response facilities, some of which are described in NUREG-0696, are to support centralized management of technical assessment, radiological assessment, governmental/industry interface, public information, and recovery activity. By proper management, the available resources can be properly directed under any conditions, foreseen or not.

The centralized management philosophy of overall emergency response was formally adopted by Toledo Edison in December of 1979. The functional requirements of our organization have been defined and translated into support facility criteria. These are being implemented in an optimized fashion to insure the overall centralized management concept is intact.

One of the main elements in our response facility upgrade is a combined function facility currently in construction at the Davis-Besse site boundary. This multi-million dollar facility fulfills our functional criteria. I consider this basic approach of centralized accident management as part of my defined responsibility in insuring the health and safety of the public. Degradation of this concept is not considered justifiable.

Comments attached reflect areas that need be seriously addressed prior to the final issuance of NUREG 0696. To provide details on the logic behind our comments, Toledo Edison's approach to these support facilities is used as examples where appropriate.

Very truly yours,

R P Crouse

R. P. Crouse
Vice President, Nuclear

RPC:TJM

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cc:
Commissioner John F. Ahearne
Commissioner Victor Gilinsky
Commissioner Peter A. Bradford
Commissioner Joseph M. Hendrie
Dr. Milton S. Plesset, Chairman, ACRS (16)
Harold R. Denton, Director, NRR
Carl Walske, President, AIF

COMMENTS ON NUREG 0696 BY THE TOLEDO EDISON COMPANY

FUNCTIONAL CRITERIA FOR EMERGENCY RESPONSE FACILITIES

SEPTEMBER 1980

Contents

I. Introduction

Toledo Edison has participated jointly through representation on the Atomic Industrial Forum's Safety Parameter Integration Group Subcommittee and the utility sponsored KMC, Inc. Coordinating Group on Emergency Preparedness Implementation. Additionally direct interfacing with the NRC staff on acceptability of Toledo Edison's centralized accident management support facilities has provided dialogue with the drafters of NUREG 0696 throughout its development. Comments have been provided before and are being provided again jointly. Toledo Edison endorses comments from both the AIF and KMC groups. However, the importance of developing appropriate functional guidance for facilities to support emergency response dictates that our Company provide comments independently on several areas of major concern as well as minor comments.

In general, these comments arise from what we believe is a basic lack of appreciation that the equipment and facilities only support the response organizations. <u>It is the organization that responds to the emergency not the facilities</u>.

II. Comments on Major Items

A. Data Information Systems

1. Basis of Concern

An obvious shortcoming of the onsite and offsite response functions in the past has been the lack of accurate information. It is recognized that each activity center needs information. However, the determination of what information, its timeliness and its display format needs to be developed based on the individuals to whom the data is directed and their function in the response organization.

The only information display systems dicussed in NUREG 0696 include an SPDS in the control room, a TSC information system, an Emergency Operation Facility information system and an information system to the NRC Operations Center, the Nuclear Data Link. These systems are discussed as if there is only one type of individual in each activity center and that the need for information is instantaneous and digital.

Further both the text of the document and meetings with its authors treat the SPDS as the sole information source illustrative of the plant conditions during all modes of plant operations. If one starts from this assumption, the design criteria evolved would logically be as prescriptive and harsh as currently identified in NUREG 0696. However, it is our major contention that not only is the basic assumption in error, but the operational philosophy that spawned this approach goes directly counter to one of the root problems during the accident at TMI-2 -- the reliance on one device to interpret plant status.

No matter how available, reliable or omniscient one information source is designed to be, we have learned the absolute necessity to verify, diversely, plant data and status. This is one lesson we do not intend to rescind.

To focus on the true need of strict requirements of the SPDS, let's reflect on several activities related to the functional response located in the control room. These activities include:

a. Expanded Control Room Organization - This includes an operations team of reactor operators and senior reactor operators that have undergone intensive retraining. A Shift Technical Advisor is now assigned to provide a broader technical expertise to advise the shift operations organization. This staff is augmented during emergencies with additional senior operations staff and high level station management and communications persons.

   The important relationship here is that different information is important to different functions in the control room organization chain. An SPDS display of interest to a Shift Technical Advisor would not necessarily be the same display provided for a reactor operator. An extremely flexible computer based data acquisition and display system is important to be able to address the functional differences of the individual's needs in the control room.

b. Control Room Information Systems - The SPDS is only a small part of the control room. A control room evaluation is to be done at every nuclear power plant to ensure the man-machine interface is good enough to overcome confusion due to expected events. New instrumentation has been added with more being upgraded. The SPDS is one more operator aid that, like any other individual device, must be able to be done without. This is regardless of any high reliability and availability design goals.

c. Long Term Procedure Upgrades - A complete change in procedural philosophy is in development. It includes the conversion from event-oriented plant procedures to symptom-oriented procedures. The goal of this effort is to be able to initially respond to a plant upset in a manner that protects the core during any type event without requiring the control room organization to know the initiating cause of the condition. In our perspective this effort is the most safety effective item in the post-TMI-2 activities. These symptoms are to be recognizable with or without an SPDS operable or any other individual device.

d. Short Term Procedure Upgrades - All plant emergency procedures for upset conditions have already been modified to require verification of vital information regardless of the reliability or design pedigree of the source instrument.

2. Toledo Edison Approach

In the Area of Data Information Systems it was recognized that the functional needs of the Technical Support Center, Emergency Operations Facility, Control Room and Nuclear Data Link could be aided by a flexible system that could be used in different formats for appropriate evaluation. The development of the human factors relationship between all the different functions will evolve over a considerable time. Recognizing this, the system approach selected by Toledo Edison is computer based, flexible, powerful, reliable and proven. Figure 1 depicts the basic system. This will allow the effective development of different display formats to be optimized through an evaluation process by the functional users.

Initially, Toledo Edison is placing a high priority on the capability to assess the immediate post-reactor trip time period. Because this is essentially the first 10-15 minutes of an event this function is required to be done by the control room shift organization without the support of a TSC or EOF. The manipulation and display of key parameters has been shown to provide a significant operator aid in the initial assessment period. Appendix A attached describes one such approach. Entitled A Real Time Method For Analyzing Nuclear Power Plant Transients, Messrs. Broughton and Walsh of the GPU Service Corporation described one method that could fulfill an operator aid requirement. A computer-based display system is ideal for this approach in the man-machine interface. However, the technique is equally functional being hand plotted from information available in the control room without an SPDS.

This backup method to support such a function allows the approach of maintaining flexibility with one system to provide a wide variety of information and displays to all emergency response activity centers with off-the-shelf, highly dependable, maintainable and available equipment. Figure 1 illustrates the system selected by Toledo Edison. The basics are modeled after the data acquisition and display system at the Loss of Fluid Test Facility (LOFT).

Additionally, a special loop communications system has been devised to allow a dedicated voice system for verbal verification of information at each of the activity centers. Closed circuit television viewing of the control room will be in the Technical Support Center.

3. Proposed Changes to NUREG 0696

Page 8 - Revise Section II.F. to read:

F. Safety Parameter Display Design Criteria

The total SPDS need not be Class IE or meet the single failure criterion. The data acquisition system for the SPDS, consisting of sensors and signal conditioners, shall be designed and qualified to Class IE standards. The processing and display devices of the SPDS shall be of proven high quality and reliability.

A Limiting Condition for Operation in the Technical Specifications shall be established that is consistent with the unavailability goal of the SPDS and with the compensatory measures defined during periods when the SPDS is inoperable.

Since the function of the SPDS is to aid in the detection and monitoring of transients and accidents, the SPDS shall be capable of functioning during and following most events expected to occur during the life of the plant. Emergency operating procedures shall specify the limitations of the SPDS.

B. Facility Locations

1. Basis of Concern

Emergency planning is just that, planning organizations and support facilities established to be able to effectively respond to events that are expected. Whether it is in the commercial nuclear power industry or any other field, the best planning does not account for every possible contingency In fact, the broader the spectrum of emergencies the harder it is to define effective organizations and management support facilities. One basic reason is the nature of the required response to one particular emergency may vary to a point that a preconceived approach may need to be altered because it is not appropriate for the particular condition. The correct approach then is to optimize support for a management organization in the expected modes of response while providing for flexibility to cope with the unexpected.

Effective, flexible management depends on information availability, internal organization communication and external communication. Effective communication includes not just technical but personal interchanges. In the commercial nuclear power production arena, these interchanges are vital due to the many organizations and responsibilities involved. Each utility has to determine the optimum interrelationship it needs to support expected emergencies. Areas of potential weakness are then compensated for by strengthening other facility support capabilities.

It is Toledo Edison's contention that this evaluation could be unique to different organizations and therefore support facility types and locations can be quite varied but still be adequate in performance.

As an example, the Technical Support Center in NUREG 0696 has been prescribed to be "within approximately two minutes comfortable walking time of the control room." The functional items this is to support is face-to-face communications "between control room personnel and senior plant management working in the TSC" and information availability of items not transmitted to the Technical Support Center. By evaluating the expected conditions requiring face-to-face communication two time periods arise. The first is some unplanned event of concern. Under this condition however, such removal of activity center managers for any time period could critically impair response capability. In addition congregation of personnel in the control room could interfere with or distract the operations staff. Therefore, during unplanned events such face-to-face exchanges between control room personnel and senior plant management should be minimized

The other events of concern would be events that had been pre-planned and may require briefings or observation. By their nature of being pre-planned a two minute walking distance can certainly be expandable. The other functional aspect of the location is to acquire information not available in the TSC. Certainly, in today's era of information transmittal, addtional viable options can be provided for information gathering.

2. Toledo Edison's Approach

Figure 2 identifies the basic functional criteria utilized to determine locations of support facilities. In developing our facilities, several options were evaluated. All had shortcomings to varying degrees. The final corporate commitment was made on the ability of facilities to support a controlled, centralized accident management organization. As a result, Toledo Edison's emergency response organization is set up in three locations: the Control Room, an Emergency Plan Facility at the site boundary (containing the TSC and EOF functions) and the Toledo Edison Plaza offices twenty-seven miles away.

Figure 3 identifies the internal arrangement of the Emergency Plan Facility while Figures 6 and 7, locate the facility with respect to the site.

There are key areas of the response organization, equipment and facilities that were upgraded to address initial shortcomings. They include:

a. Need for control room access for information - an elaborate data acquisition and display system was selected to make available not only a minimum number of plant parameters to the TSC and EOF but to essentially access the total library of the plant computer. A video link from the Control Room to the TSC is being provided as well as a dedicated loop communications system that will place the Control Room on line with the technical assessment area and the radiological assessment area.

b. Need to have face-to-face interaction between senior station management personnel in the TSC and Control Room personnel - the response organization at Toledo Edison supplements the Control Room staff with senior station operations personnel and the Assistant Station Superintendent. The Technical Support Center response organization includes the Station Superintendent and the Manager of Nuclear Engineering. The detailed day-to-day interactions of these persons provide valuable experience that will aid in the capability of handling verbal discussions and directions.

   Additionally, to allow rapid transportation between the TSC and Control Room, a dedicated road is being constructed totally within Toledo Edison's control. Transport time utilizing a dedicated vehicle from the TSC to the Control Room is easily under five minutes.

c. Emergency Operation Facility Survivability - Toledo Edison located the EOF functions at the Emergency Plan Facility. The entire first floor is designed to be habitable for the same radiological event as the Control Room (General Design Criteria 19). Location of this facility here is considered to aid greatly the more critical face-to-face communications between different governmental/utility/industry/ media response organizations and allow timely briefings within or among any of these organizations.

To arrange for flexibility to cope with unexpected events the Emergency Plan Facility provides for:

a. An entire second floor of offices that normally will house selected elements of the station staff.

b. An extensive communication system to ensure independence from local phone overload conditions.

c. An internal security and badging system to aid in the recognition of organizational authorities.

d. A power supply system that provides diesel generator backup as well as a battery backed uninterruptable power supply (UPS) system for critical data and communications functions.

Beyond the site response organization is a support engineering function that reports to the Toledo Edison corporate offices at the Edison Plaza. A data link provides all information available to the SPDS, TSC and EOF such that this organization can support, monitor and assume a direct response function if required due to extreme unforeseen site conditions.

The accident management organization is well supported by the locations of its facilities. Additionally, it has the flexibility important when trying to design facilities for all possible events, foreseen or not.

3. Proposed Changes to NUREG 0696

Page 10 - Revise Section III.B. to read:

B. Technical Support Center Location

The requirement for an inplant TSC was established to provide facilities for detailed analysis of plant conditions and to alleviate the problem of control room overcrowding during an accident. The TSC shall be the emergency operations work area for designated senior plant management personnel, designated licensee engineering and technical personnel, a small staff of NRC personnel, and any other licensee designated personnel needed to provide the required technical support.

The TSC shall be located to readily allow face-to-face interaction between control room personnel and the senior plant management working in the TSC.

The TSC shall normally be in a location that is within approximately five minutes of the control room.

Provisions shall be made for the safe and timely movement of personnel between the TSC and the control room under emergency conditions.

III. OTHER COMMENTS

A. Nuclear Data Link (NDL)

The Nuclear Data Link is still in its formative stages, but is being based on a consideration that has not yet matured. Fundamental to the establishment of the NDL is a clear determination by the Commission and an understanding by the staff of the function and role of the NRC in an emergency. This is identified as Action Plan Task III.A.3.1, and is still ongoing. Resolution of this concern is an important prerequisite to development of the NDL (which is Action Plan Task III.A.3.4). As such, that part of NUREG-0696 that relates to the NDL, if retained in the final version, should be considered as information only, with no implementation inference at this time.

B. NRC Regulatory Guide 1.97

All references to Regulatory Guide 1.97 (Instrumentation to assess and follow the control of an accident, etc.) should be qualified. Reg Guide 1.97 is not final yet. The parameter sets for the emergency facilities should be based on the function of each facility. The Atomic Industrial Forum's Safety Parameter has recommended to the Advisory Committee on Reactor Safeguards (ACRS) that a systematic approach be used to establish the data requirements for emergency facilities. This approach, is contrast to Reg Guide 1.97, integrated the consideration of human factor engineering, the need for and importance of the information, and the function for which the information is going to be used. As a result, the ACRS did not endorse Reg Guide 1.97 in its present form and recommended that additional effort be made to resolve some of the rather major differences in the approach between NRC staff and industry.

C. Availability

Availabilty of information and power supply requirements need to be defined with respect to function (purpose) of the SPDS, TSC, etc. Unavailability should not mean loss of a single input parameter but loss of the function of each Emergency Response Facility.

Design availability (or unavailability) should be defined using standard manufacturers data such as Mean Time Between Failures and Mean Time To Repair and should be based upon actual historical or generic data.

Commercially available computers typically have an advertised availability of 99.5% and when used in conjunction with available input/output devices and power supplies overall availability of 99.0% in achievable. To meet the 99.9% availability requirement would require redundant computer systems, input/output devices and power sources. To statistically demonstrate an availability of 99.9% with a confidence level of 95% would require a test period of approximately 400,000 hours.

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Davis Besse
Data Acquisition and Display System

Basic System
Input Parameters
Parameter Processing

Technical Assessment Area
Radiological Assessment Area
Control Room
Offsite Data Transmission

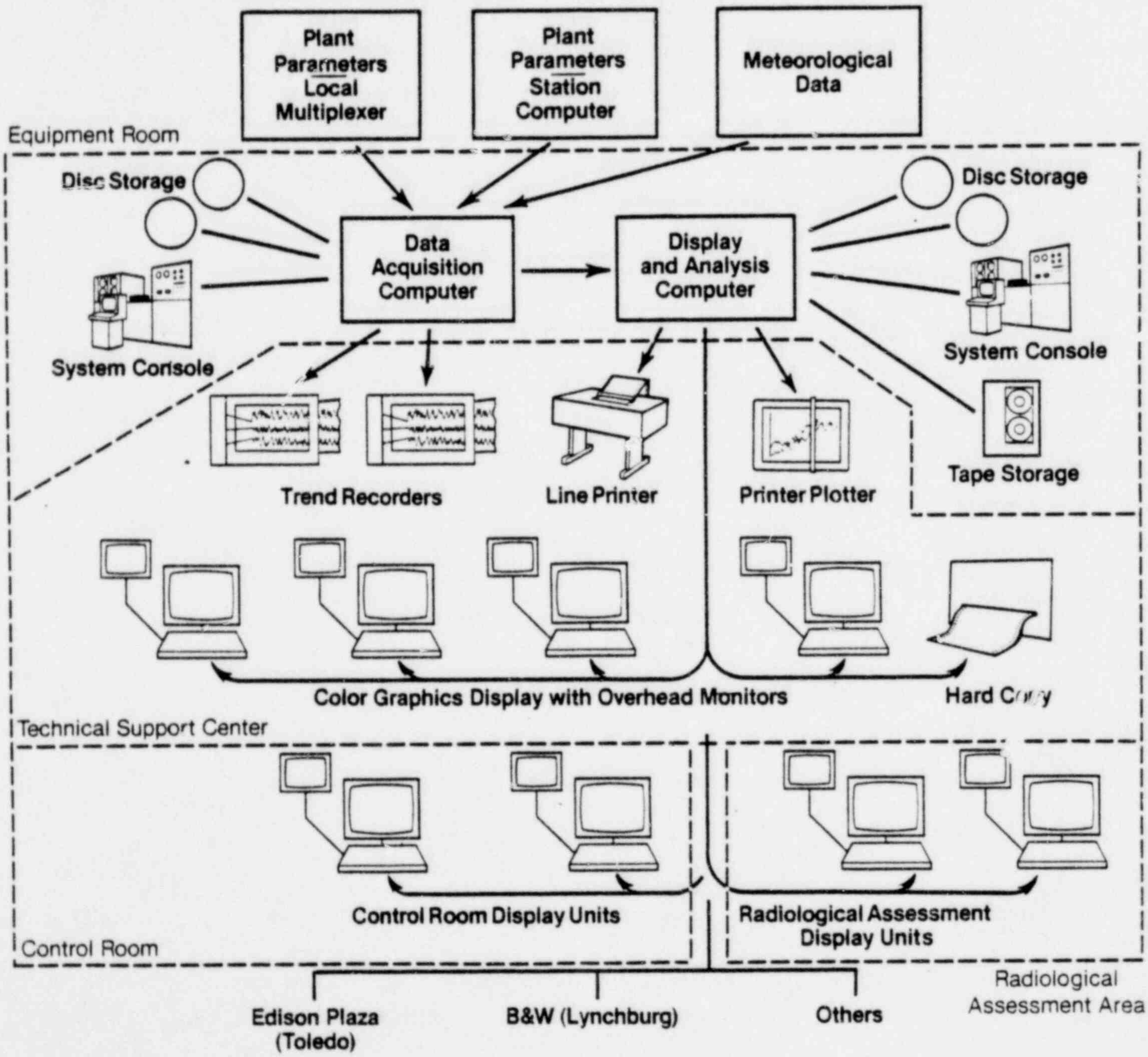

Figure 1

Toledo Edison
Emergency Plan Facility
Functional Criteria

- Provide Assessment Data and Communications
- Optimize Overall Accident Management
- Isolate Technical Assessment from Public Interference
- Provide For Controlled Public Interactions
- Minimize Transport Time for TECo Functional Managers
- Minimize Transport Time for Joint TECo/Governmental Agency Interface
- Minimize Functional Impacts During Protective Action
- Address Known NRC Guidance and Investigative Group Recommendations

Figure 2

Toledo Edison Emergency Plan Facility
First Floor

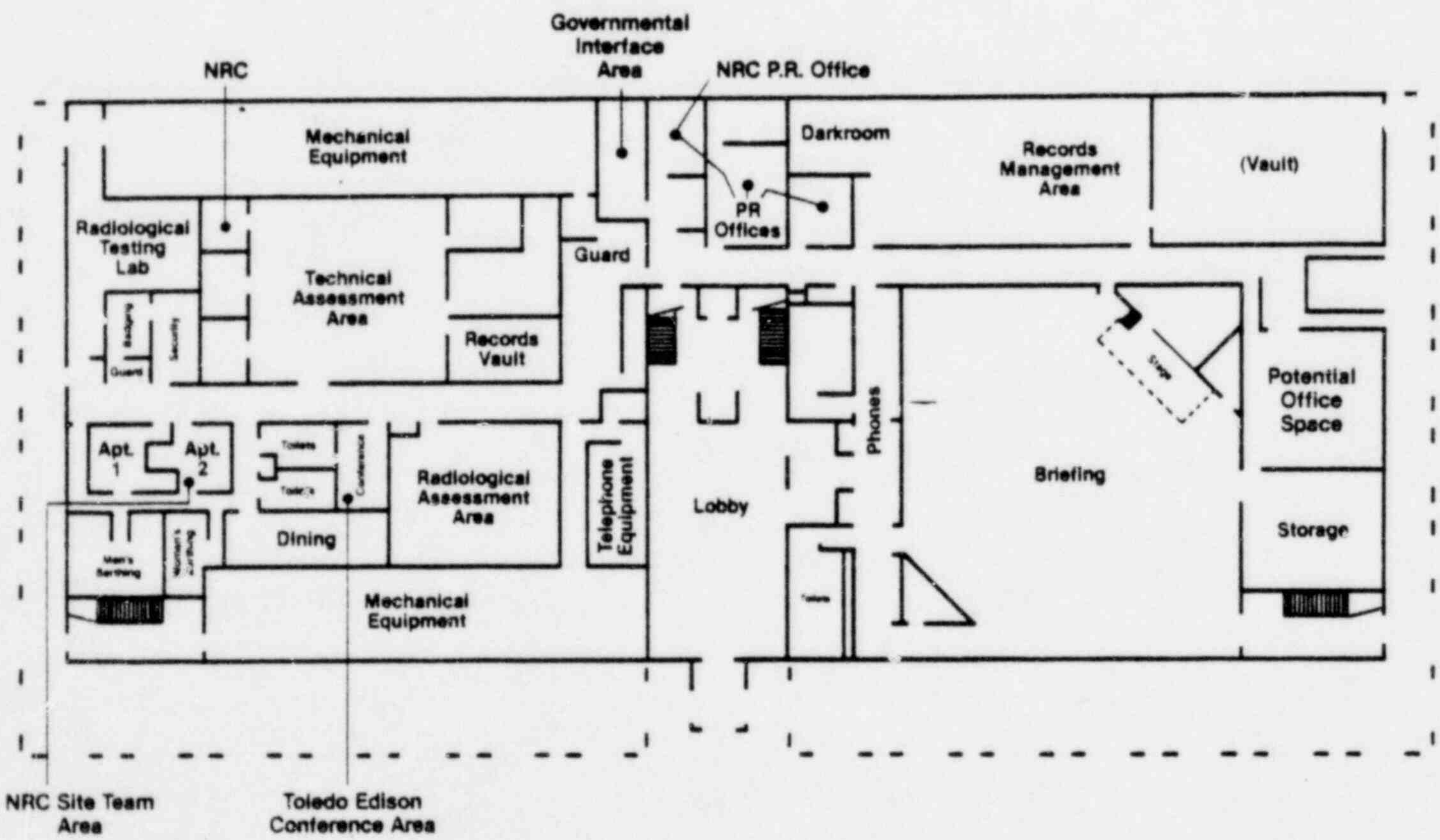

Figure 3

Technical Assessment Area

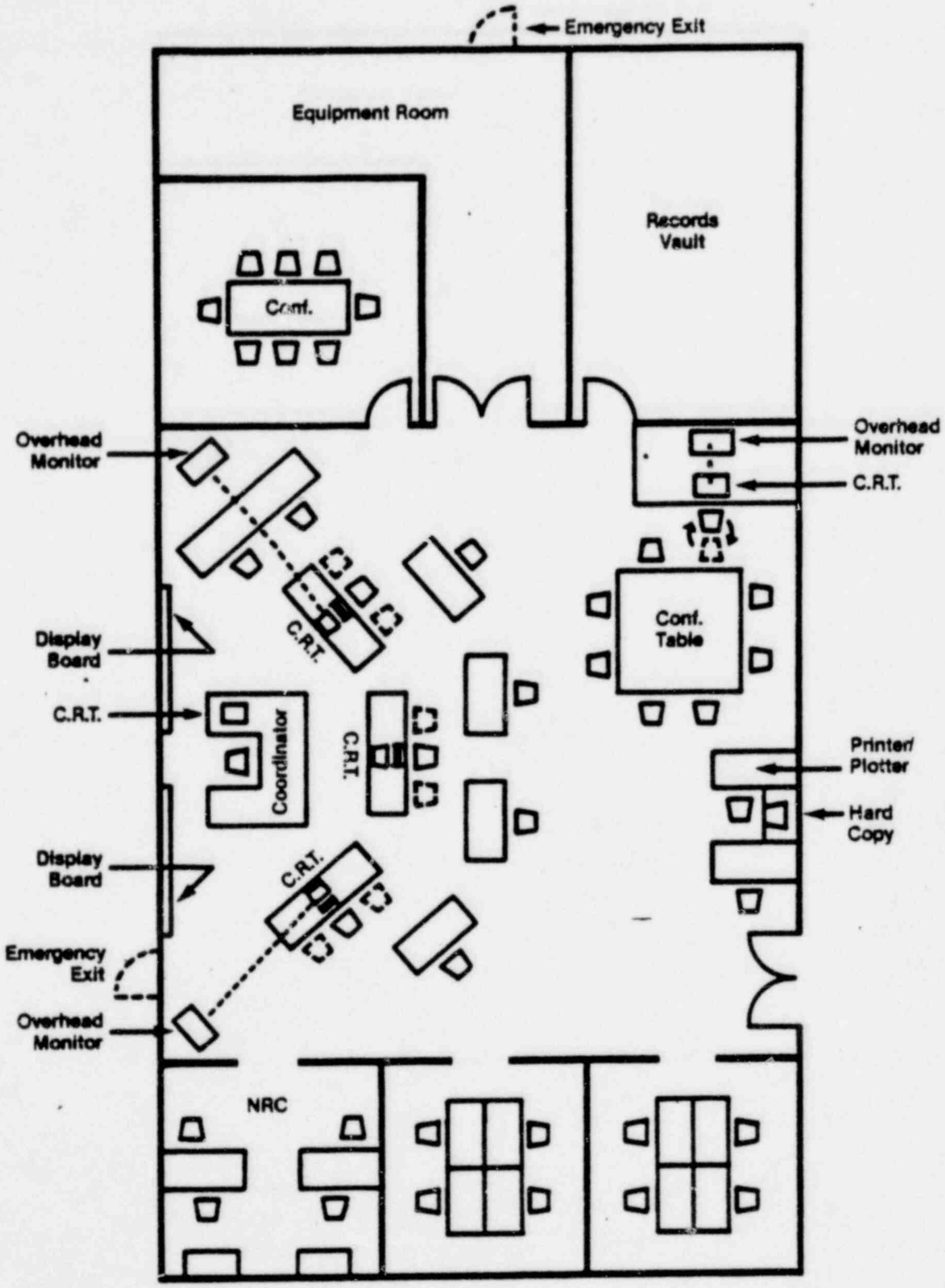

Figure 4

Radiological Assessment Area

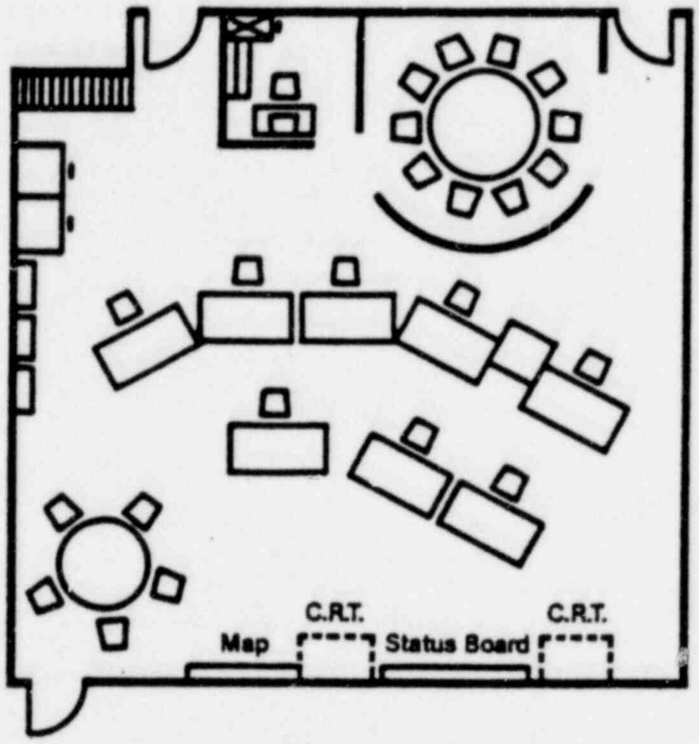

Figure 5

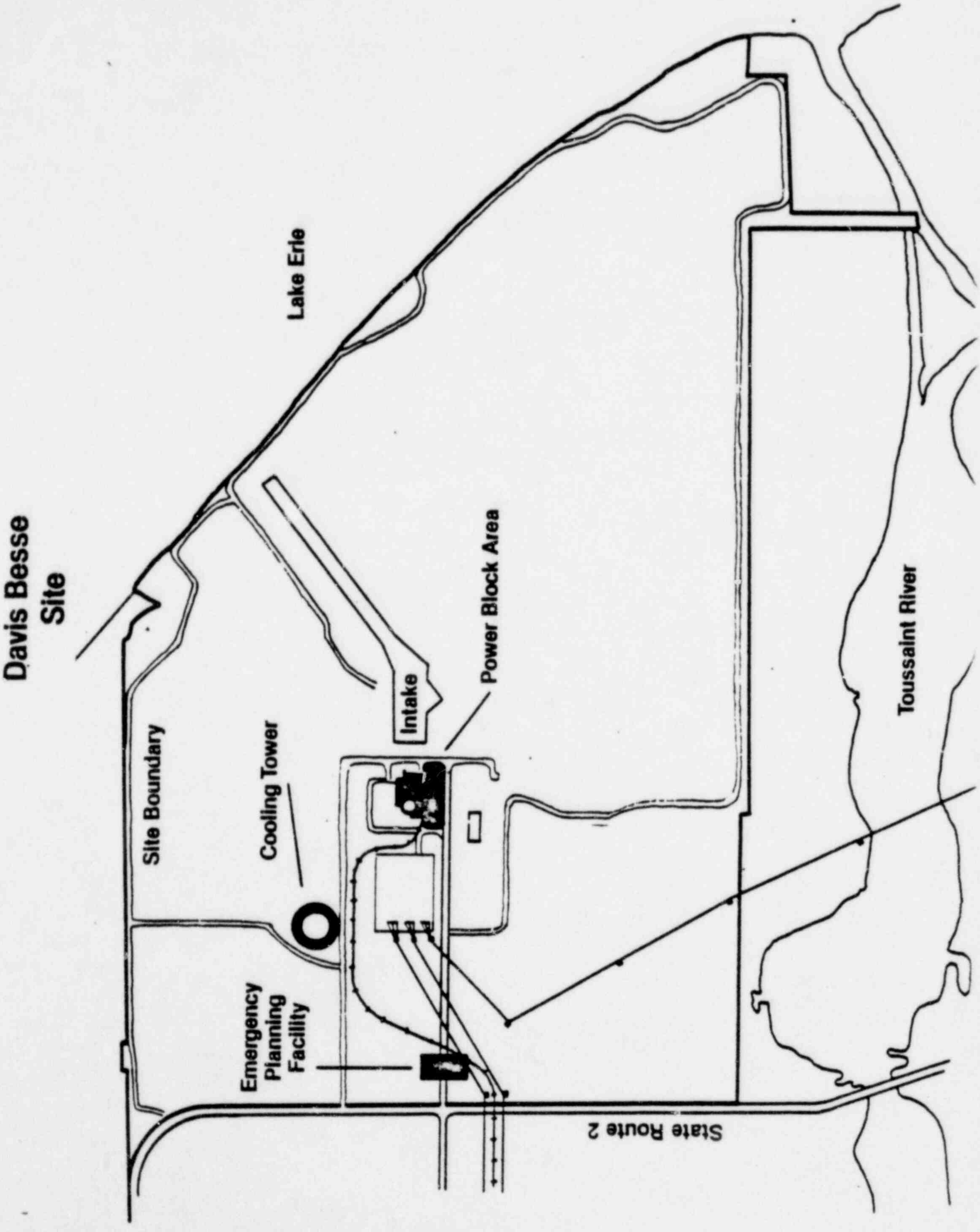
Davis Besse Site
Lake Erie
Site Boundary
Cooling Tower
Intake
Power Block Area
Emergency Planning Facility
Toussaint River
State Route 2

Figure 6

Toledo Edison Davis-Besse Nuclear Power Station

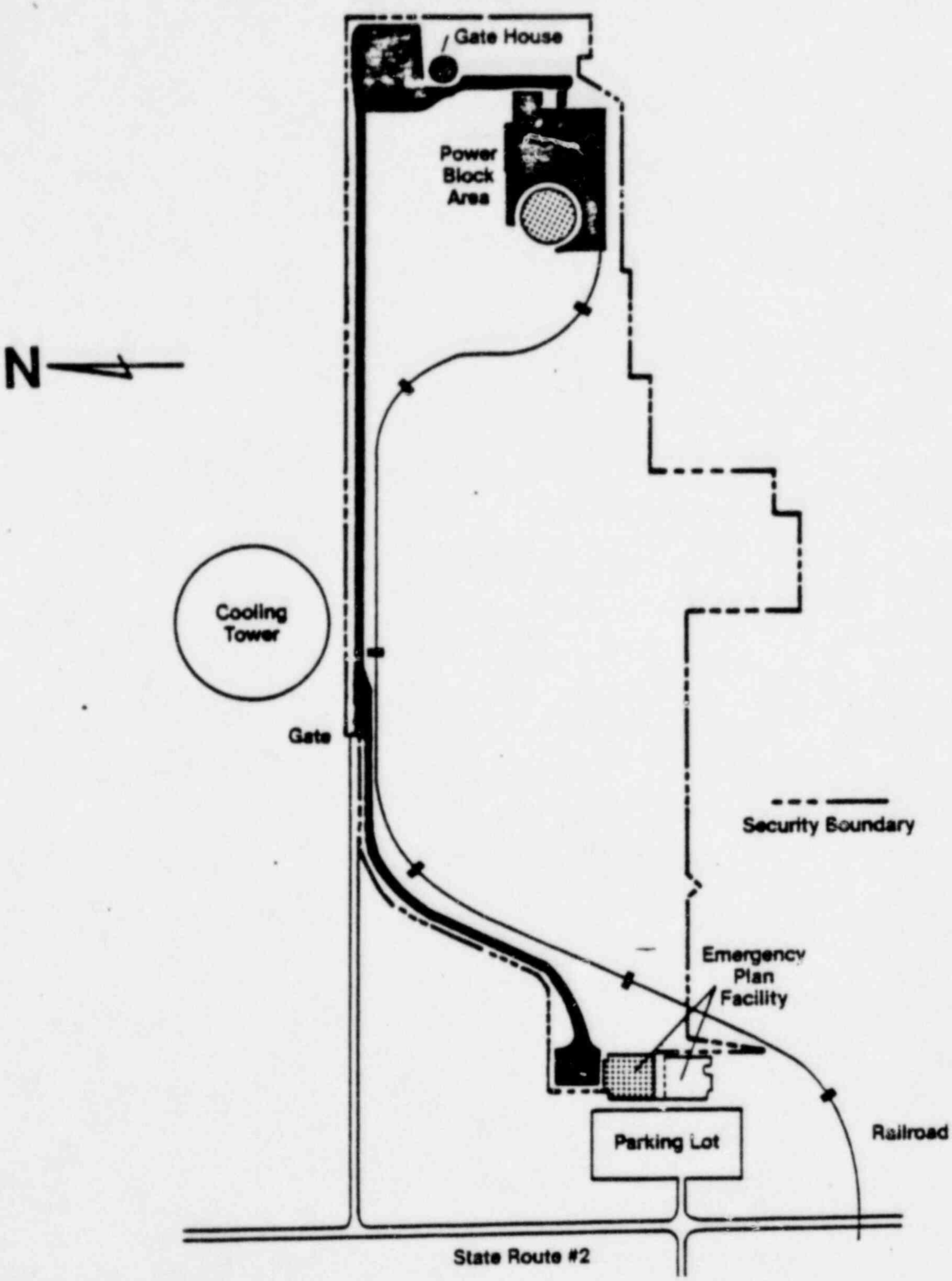

Figure 7

APPENDIX A

A REAL-TIME METHOD FOR ANALYZING NUCLEAR POWER PLANT TRANSIENTS *

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* Discussed in ANS Transactions, Volume 34, TANSAD 34 1-899 (1980) pages 723-724

A REAL-TIME METHOD FOR ANALYZING NUCLEAR POWER PLANT TRANSIENTS

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Introduction

Monitoring power plant performance during transients and detecting abnormal performance requires comparing many parameters to limiting values, determining correlations among parameters, evaluating trends and verifying the status of key systems and components. This task may be difficult to perform in real time during unexpected transients. After-the-fact analysis of the TMI-2 accident, for example, revealed that the multiple malfunctions which occurred could have been diagnosed from the values of and relationships between a few key parameters. The presence of additional data of lesser importance, however, resulted in confusion and inhibited mitigating actions. This accident and other plant transients demonstrate a clear need for a real-time analysis method useable by operators.

Efforts to develop a diagnostic method concentrated on heat transfer and pressure control following a reactor trip. A method of determining the effectiveness of steam generator heat removal was particularly desirable. Many alternate approaches were evaluated and potential methods were refined or eliminated in an attempt to develop a tool which was based on sound principles, could be consistently applied over a wide range of expected events and could be easily learned and used by operators.

An analysis method using primary and secondary pressures and primary temperatures proved to provide operations personnel with the desired tool to diagnose power plant status in real time following a reactor trip. This method has been specifically applied to pressurized water reactors with once through steam generators. However, the principles of this method are applicable to other light water reactor types.

As a test of this method, actual plant data from more than twenty transients as well as data from computer simulations of transients have been plotted to evaluate the effectiveness of the technique. All the examples used in this paper are based on data from actual plant transients.

Method

During the transition from power operation to post trip decay heat removal four key parameters (primary and secondary pressure and hot and cold leg temperatures) are monitored using pressure/temperature plots (Figure 1). If normal post trip decay heat removal conditions are attained, temperature and pressure will be within their expected range, as indicated by the box on the plot. Normal operation at power is outside the expected post trip range. The normal transition to stable post trip conditions requires 5 to 10 minutes. At any point during the transition, the plot indicates the proximity to the expected value and any limiting values. In addition, the trend allows the "analyst" to anticipate the course of the transient.

A single plot which includes primary and secondary conditions has proven most useful (Figure 2). The expected post trip ranges are indicated for the primary and secondary systems. The saturation curve is also included. The primary plot of primary pressure versus hot leg temperature directly indicates primary system subcooling (saturation margin). The secondary plot of steam generator pressure versus cold leg temperature indicates the effects of the steam generator on the primary system. Effective steam generator heat transfer is reflected by cold leg temperature (steam generator outlet temperature) nearly equal to the saturation temperature for steam generator pressure. This relationship should exist following trip since at decay heat levels the relatively low heat transfer in the steam generator results in a small temperature difference across the steam generator tubes.

Assumptions and Limitations

Since this monitoring method only considers four key parameters, it cannot diagnose all abnormal power plant conditions nor can it determine the specific cause of the abnormal conditions which exist. For example, a small steam generator tube leak (leak rate less than makeup capacity) would not affect these thermal hydraulic parameters but would be indicated by secondary system radioactivity. The cause of overcooling events cannot be determined without additional steam and feedwater system data. Therefore, this method should be considered a supplement to the normal monitoring of system and component status and other parameters important to safety.

Other assumptions are implicit in the use of this method. The most important is that the reactor is shutdown since the expected values and trends which are the basis for diagnosis are valid only at decay heat levels. Of course, instrumentation accurately displaying the monitored parameters must be available to permit analysis. Also, it is necessary to know whether forced primary flow or natural circulation exists since the expected values and abnormal conditions limits are dependent on the flow rate.

Normal Post Trip Performance

Figure 3 is an example of normal post trip performance following a loss of feedwater. The conditions just prior to the trip (time = 0 minutes) are normal for full power. The loss of feedwater causes the reactor and turbine to trip. During the first minute steam pressure rises to the safety valve setpoint due to reduced steam generator heat removal following turbine trip, then decreases to the post trip turbine bypass valve control setpoint. Hot leg temperature decreases and cold leg temperature increases as a result of reduced reactor power generation. From these conditions the transition to stable post trip decay heat removal begins. The cooldown causes a primary system contraction which reduces primary pressure. As the cooldown rate decreases, primary pressure recovers to the expected range. As the secondary system pressure is controlled at a constant value by turbine bypass valves, the cold leg temperature decreases to the corresponding saturation temperature.

These pressure/temperature traces are typical of normal transitions. However, the conditions at the time of reactor trip, the exact transition path and the time to reach the expected values vary with the specific transient.

Detecting Abnormal Performance

Several categories of abnormal performance can be diagnosed using this plot. Certain regions on the plot which indicate abnormal performance can be defined using limit lines. However, the trend of the data provides the earliest indication of abnormal performance.

1. LOCA (Figure 4) - Loss of coolant accidents and loss of pressure events are reflected by a continuous decrease of primary pressure to the saturation curve or below (superheat region) without significant temperature decrease and no effect on steam generator conditions. Pressures below the high pressure injection setpoint or below a predefined subcooling margin may be considered indication of a LOCA.

2. Loss of Heat Sink (Figure 5) - Loss of steam generator heat sink events are reflected by the secondary plot trending away from the saturation curve. This indicates an increasing temperature difference between cold leg temperature and steam generator saturation temperature indicative of degraded heat transfer conditions between the primary and secondary. This plot defines a loss of steam generator heat sink region as cold leg temperatures greater than 35°F from the saturation curve or cold leg temperature more than 25°F above the expected value. The total loss of primary heat sink is reflected by primary temperature increasing above the expected range. The limit for this zone has been set at 25°F above the expected value.

3. Overcooling (Figure 6) - Overcooling events are reflected by temperature decreases below the expected range. On this plot the overcooling region is considered to be temperatures more than 10°F below the expected value. The temperature decreases can cause or be caused by a steam generator pressure decrease. A rapid cooldown will produce a primary pressure decrease while a slower cooldown may not affect primary pressure.

Other limits can be indicated on the plot, for example, primary system relief valve settings and steam line rupture isolation system setpoint.

One of the key advantages of this technique is the ability to distinguish between a LOCA depressurization and a rapid overcooling which results in low primary pressure. The LOCA is reflected only in the primary trend but the major indicator of overcooling is the secondary trend.

It is also possible to determine if a low steam generator pressure is due to loss of heat transfer or overcooling. This distinction is important since the correct action for loss of heat transfer (addition of feedwater) is incorrect for the overcooling case.

Asymmetric and Multiple Casualties

Plotting each loop individually on the same graph enhances early recognition of asymmetric transients. Figure 7 illustrates a stuck open turbine bypass valve in the A loop steam system. At 2 minutes the overcooling trend on A is becoming visible on the secondary plot. After 3 minutes the A loop continues to overcool, but the B loop steam generator acts as a heat source to the primary (steam generator saturation temperature is greater than cold leg temperature indicating heat transfer from secondary to primary).

Multiple casualties can also be diagnosed by this method. Many combinations are possible including loss of heat sink in one steam generator with overcooling in another, LOCA combined with loss of heat sink or overcooling, or sequentially occurring faults.

Figure 8 illustrates the multiple effects of the TMI-2 accident. LOCA indications develop at 2 minutes. High pressure injection (HPI) has no effect on the primary pressure decrease but does remove decay heat between 2 and 4 1/2 minutes during the loss of steam generator heat sink. When HPI flow is reduced at 4 1/2 minutes a loss of primary heat sink results. The primary reaches saturation conditions at 6 minutes and pressure increases as the system heats up. The steam generators are restored as a heat sink when emergency feedwater is initiated at 8 minutes. By 20 minutes the secondary system has been restored to the expected range but the primary system still reflects the LOCA.

Natural Circulation

Monitoring and diagnostic methods used during natural circulation are identical to forced circulation except that the expected value of hot leg temperature is higher, the loss of primary heat sink limit is higher, and the increased loop transport time presents additional challenges in diagnosis.

The expected temperature difference between hot and cold legs during natural circulation is 20 to 40°F versus the 2 to 3°F expected during the forced flow. The box indicating the expected range following reactor trip during natural circulation reflects these differences (Figure 9). The expanded range of hot leg temperature results in a higher limit for definition of loss of heat sink, 10°F higher than the maximum expected hot leg temperature (Figure 9).

Effects of Low Flow Rate

The long loop transport times during natural circulation alter the trends slightly from the forced flow case but, more importantly, introduce time differences between primary and secondary plots which give them the appearance of independence. The loop transport time increases from about 15 seconds during forced flow to about 8 minutes during natural circulation. The effects on plotted data are seen in Figure 10. Pressure changes are reflected in the data immediately. In the secondary system, where pressure controls cold leg temperature, the time delay associated with cold primary water flowing from the steam generator to the cold leg temperature detector is about 2 minutes. Therefore, to evaluate steam generator performance, pressure at time 0 should be plotted with temperature at time 2 minutes. Although this correction improves the presentation of the data, the direction of the trend remains unchanged, and therefore applying the corrections in real time is of little value and not recommended. The close proximity of hot leg temperature to the pressure sensor renders a correction to the primary system plot unnecessary.

This delayed response of indication during natural circulation makes it important to provide the operator with a method of detecting abnormal conditions in time to control the plant.

Detecting Abnormal Conditions

Figure 11 illustrates an overcooling event during natural circulation. The cause of overcooling is overfeeding which would be indicated by increasing steam generator level. The secondary plot provides early indication of this abnormal condition (4 minutes). The primary plot begins to reflect overcooling 2 or 3 minutes later.

The existence of adequate natural circulation flow must be inferred from other parameters in cases where low range flow instrumentation is not provided. Loss of natural circulation flow prevents heat transfer from the primary to the secondary and hot leg temperature can be expected to rise indicating a loss of heat sink on the primary plot. If steaming continues in the secondary, steam generator pressure will begin to drop without reducing cold leg temperature, also indicating a loss of heat sink.

This will also result in an increasing differential between hot and cold leg temperatures. However, other events may also produce large primary temperature differentials even though adequate flow exists. A severe overcooling event, such as a stuck open turbine bypass valve or steam system relief valve, may result in primary temperature differentials in excess of 100°F. This diagnostic method allows the overcooling trend to be easily distinguished from the loss of natural circulation flow (loss of heat sink) event by the response of both the primary and secondary plots.

Implementation

Implementation of this analysis method involves several areas. An automated data display is desirable to enable operators to devote full time to analysis. However, manual plotting capability should be considered as a backup and for its use as a training method. Procedures should be consistent with the diagnostic method. For example, if an overcooling event is diagnosed, there should be procedural guidance to mitigate overcooling. Introductory and proficiency training is required in both the analysis method and the use of related procedures for operators and appropriate staff members.

This analysis method has been applied to post trip data recorded during normal and abnormal plant transients and to transient data generated by computer simulation. Operator training using this data base has shown that the method can be quickly learned and effectively applied. In the classroom environment operators are able to manually plot and evaluate trends in real time. Operators who analyzed plant transients without this method seldom completed the analysis in real time, were less confident of their conclusions and made more errors in diagnosis.

Additional uses for this method include basic training in plant dynamic response. The historical trend aids in communicating the actual or simulated dynamic plant response and may be used to supplement the narrative reports of events of interest to the industry. This analysis method has been developed for use following reactor trips but the principles may be applied to other power plant operating conditions.

An example of the method applied to a PWR with a U-tube steam generator is shown in Figure 12. The transient was a reactor trip test in which a turbine bypass valve malfunctioned and stuck open. The pressure-temperature plot shows an abnormal cooldown below the expected range in 1-1/2 minutes. The primary pressure drop was rapid but the steam generator response and the large saturation margin clearly indicate an overcooling problem and not a LOCA. Although an extensive amount of data has not been plotted for this type of plant, this example demonstrates that the method has potential applicability.

Summary

A method for analyzing power plant performance in real time following a reactor trip has been developed. Using trends of primary temperatures and primary and secondary pressures on a pressure/temperature plot, plant response can be evaluated as either normal or abnormal with respect to one or more of the following categories: Loss of Coolant Accident, Loss of Heat Sink or Overcooling. This method facilitates distinguising LOCA from overcooling, highlights asymmetric effects and diagnoses multiple casualties. The analysis and diagnostic capability is applicable during forced flow and natural circulation. Implemen- tation in an operating environment involves training, procedures and methods to provide plots. Other uses include communicating dynamic plant response to operations personnel. The potential exists to extend this method to other plant conditions and to pressurized water reactors with U-tube steam generators.

Figure 1

Pressure Temperature Plot

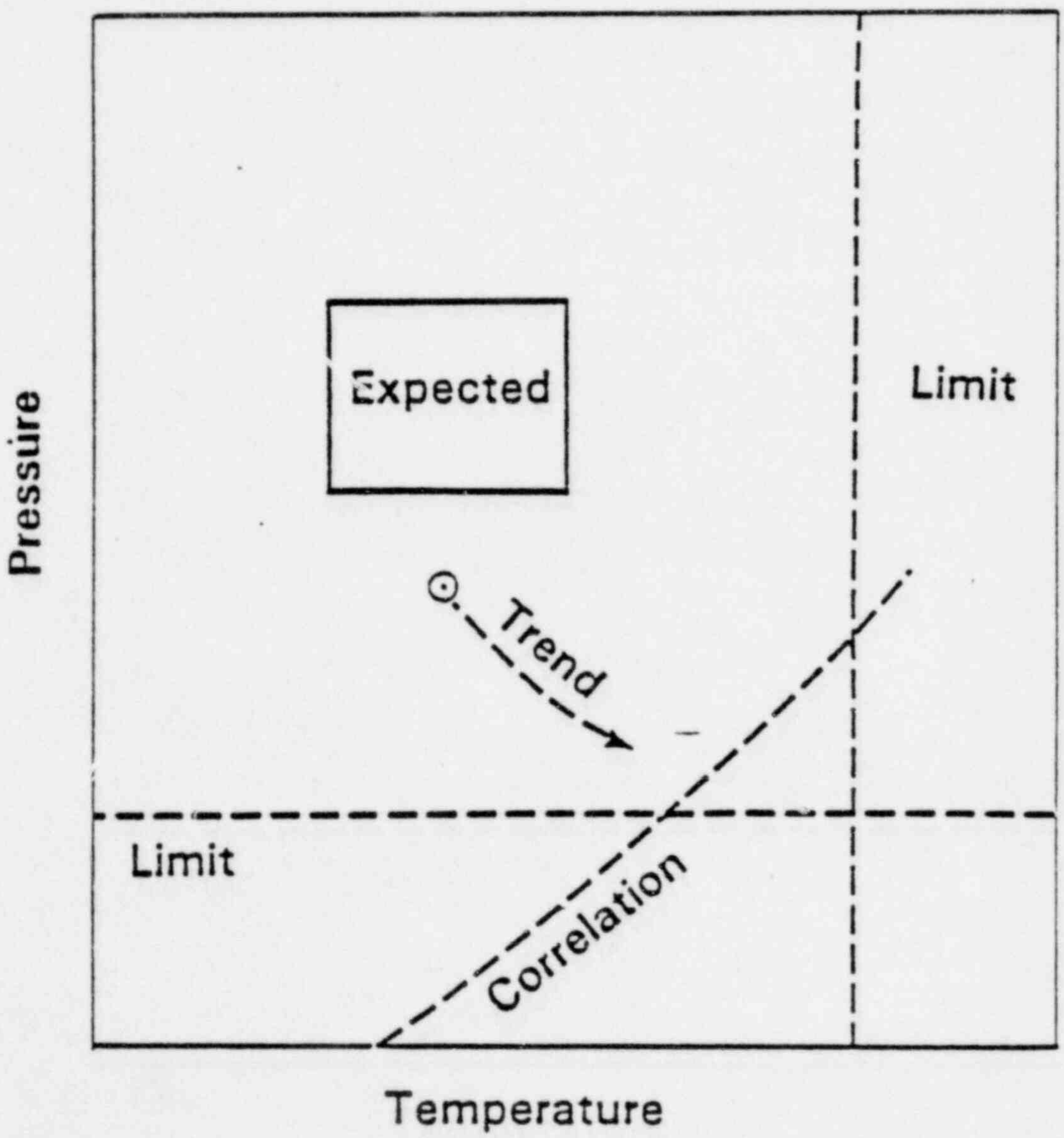


Figure 2
Expected Post Trip Conditions

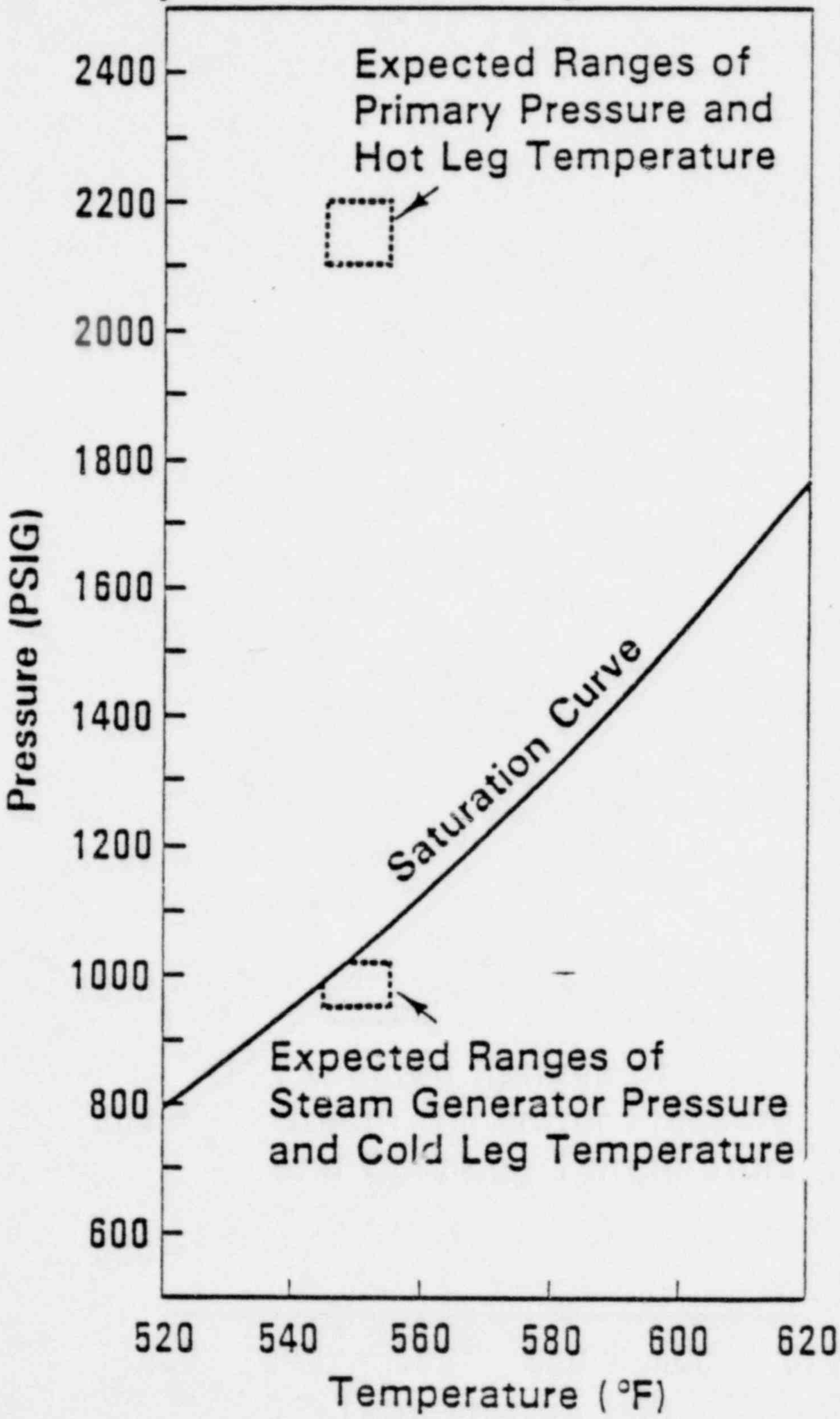


Figure 3
Normal Post Trip Response

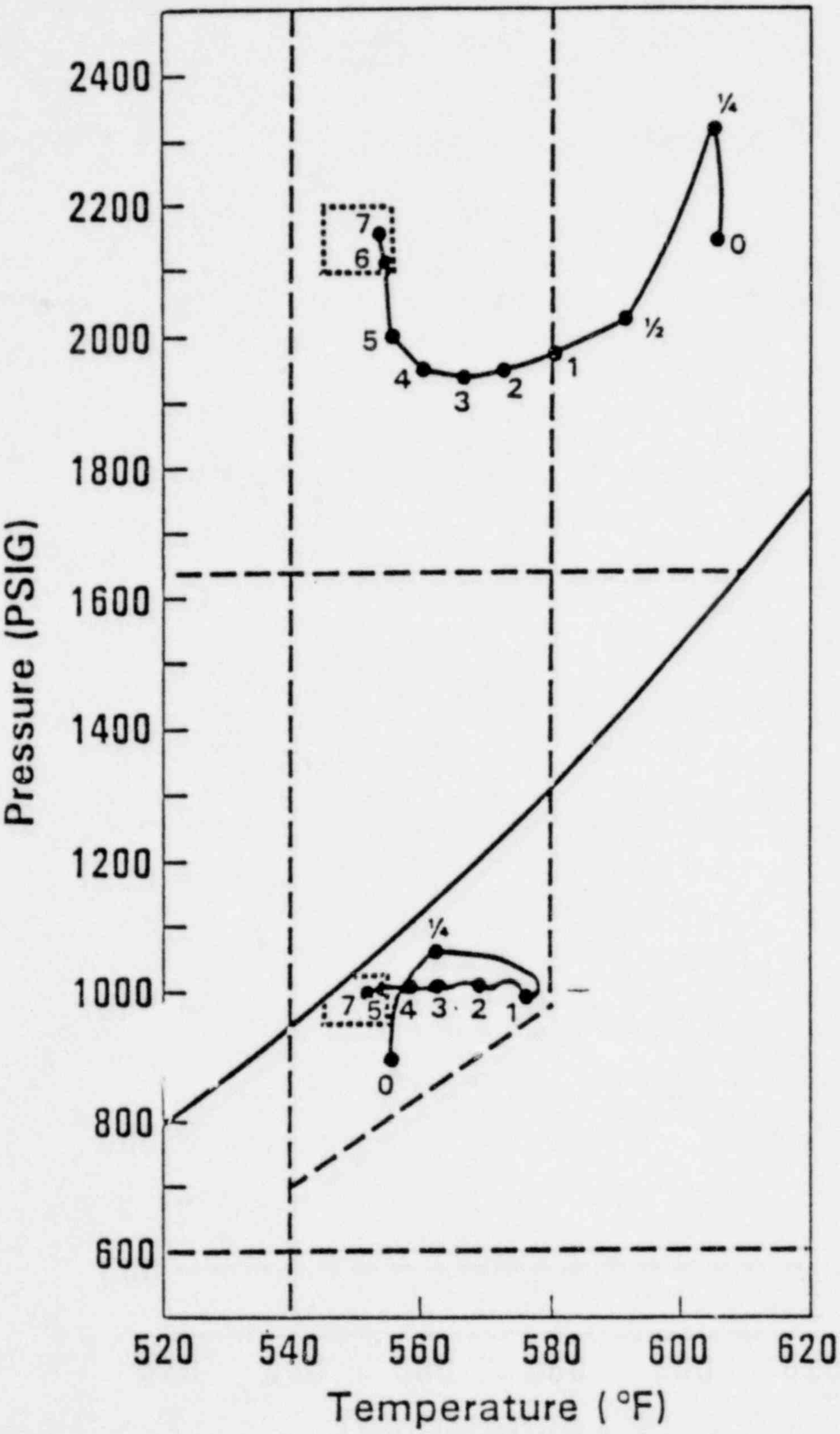


Figure 4
Loss of Coolant Accident

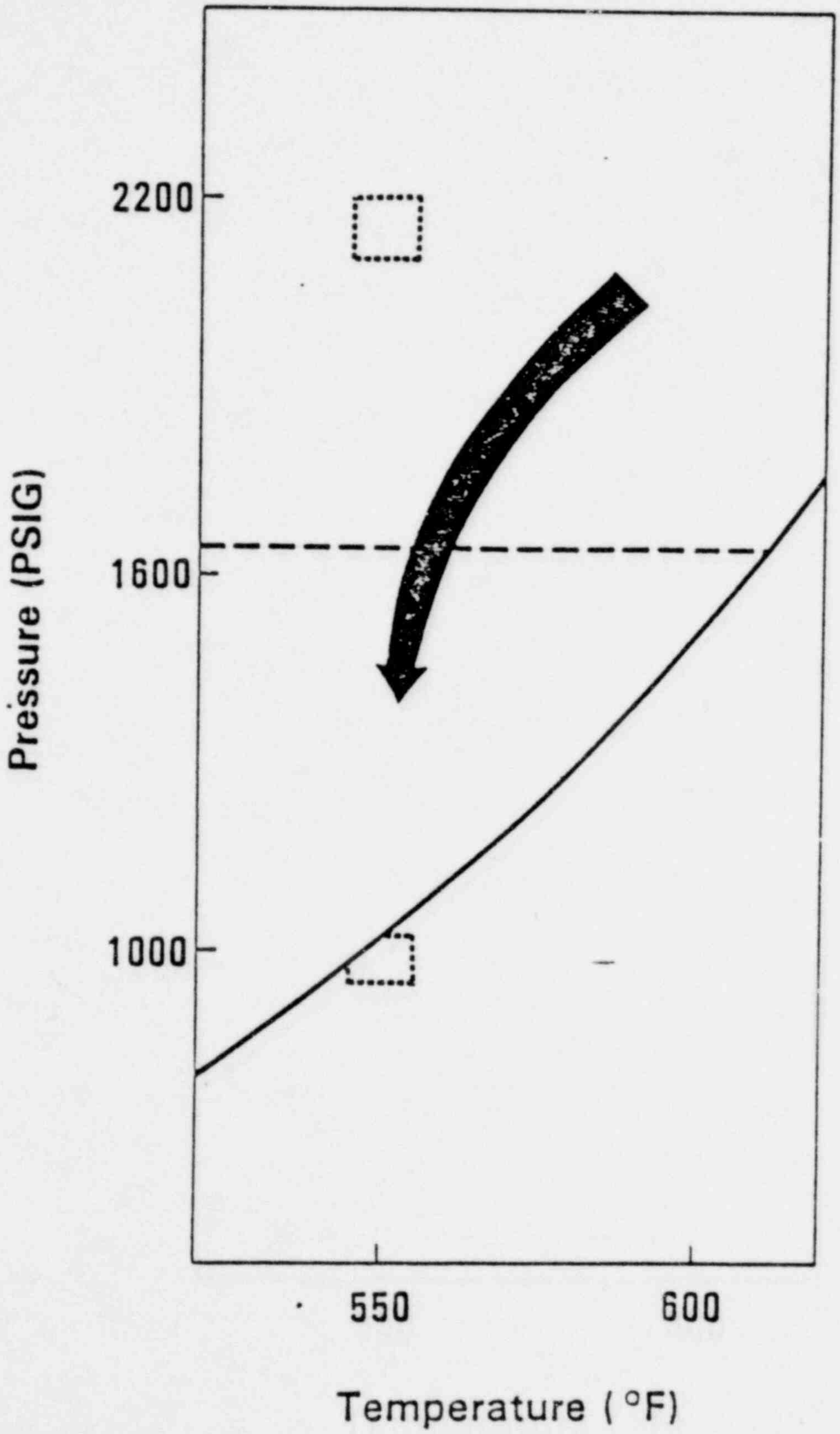


Figure 5
Loss of Heat Sink

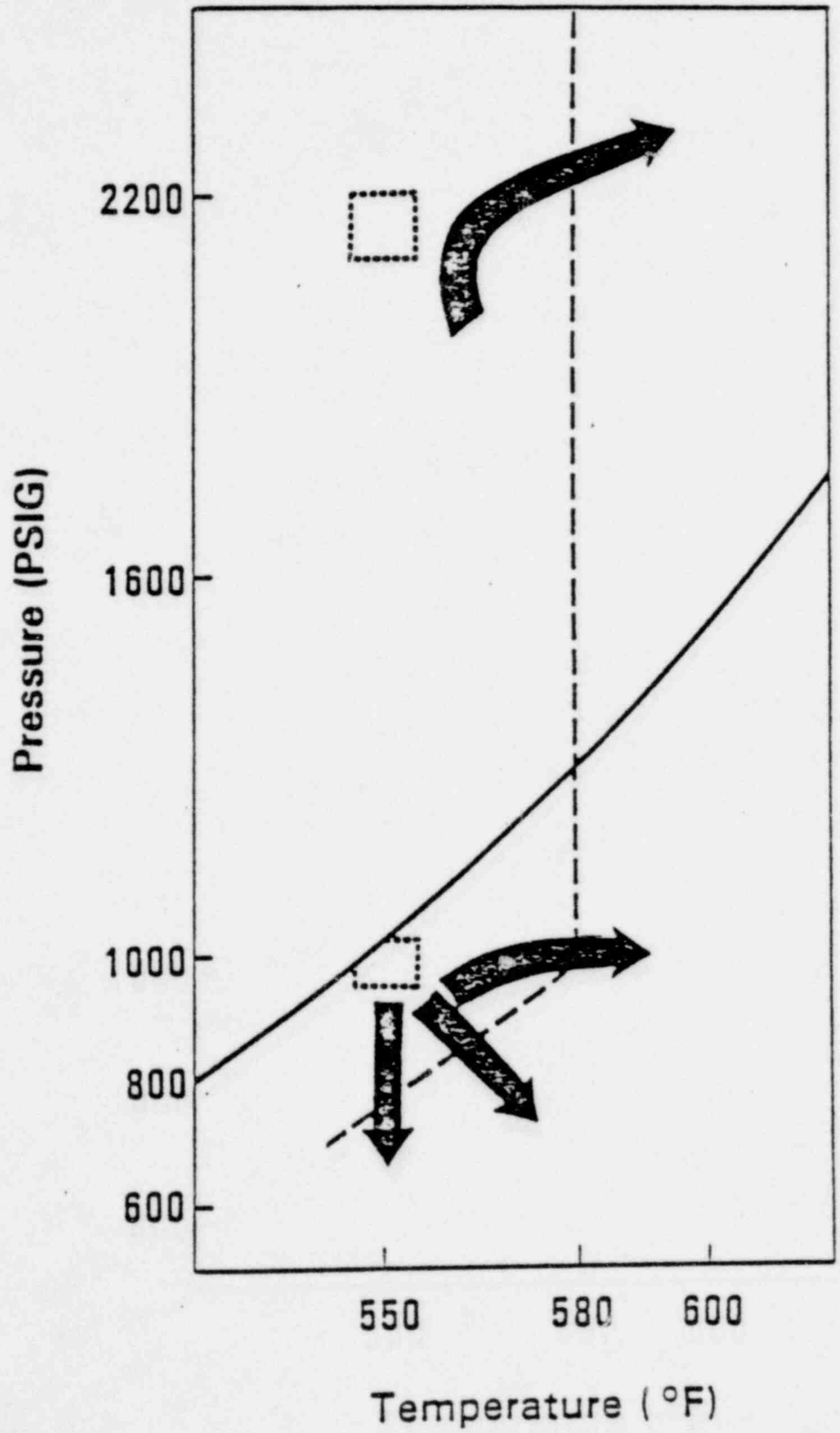


Figure 6
Overcooling

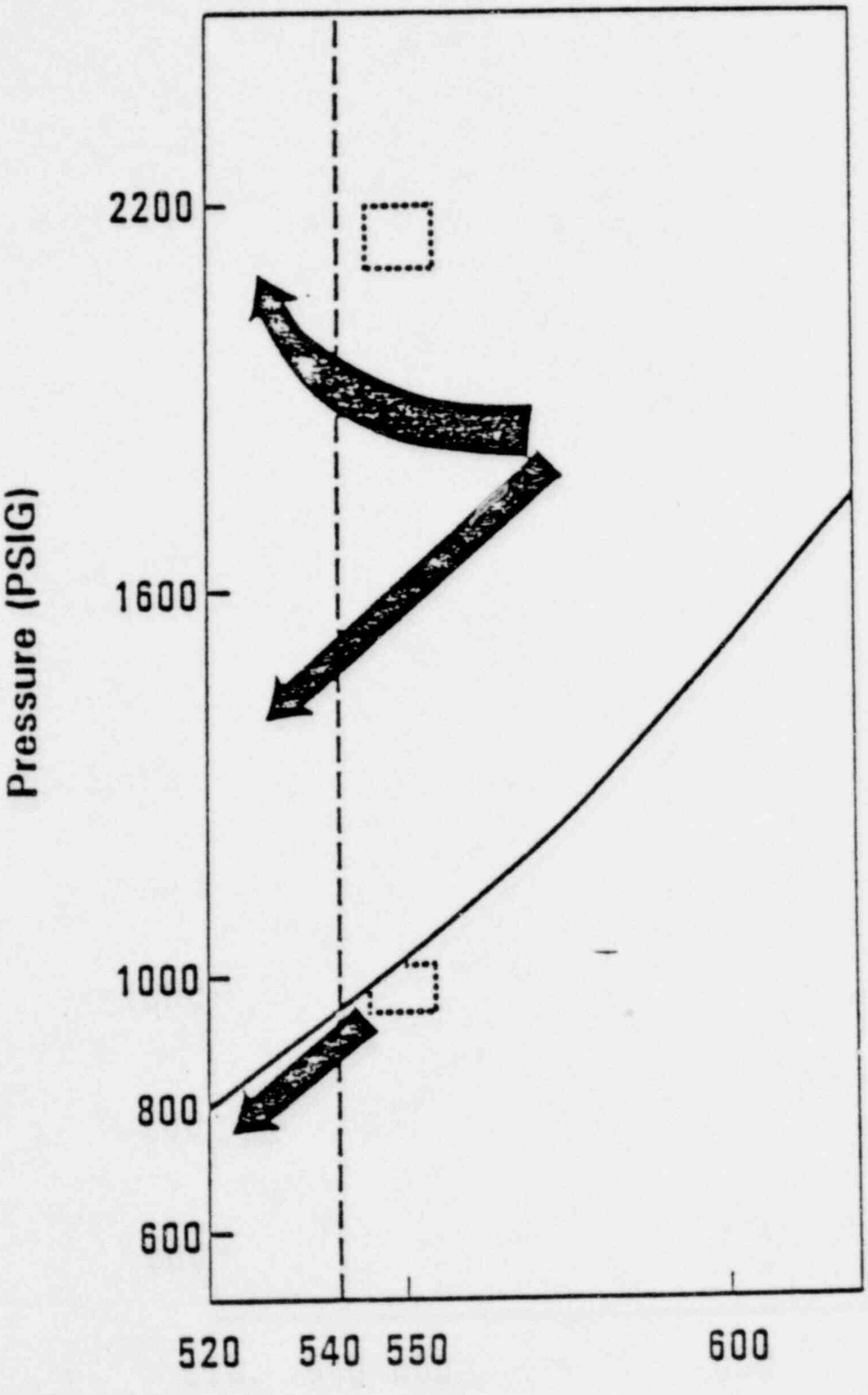


Figure 7
Stuck Open Turbine Bypass Valve-"A" Loo

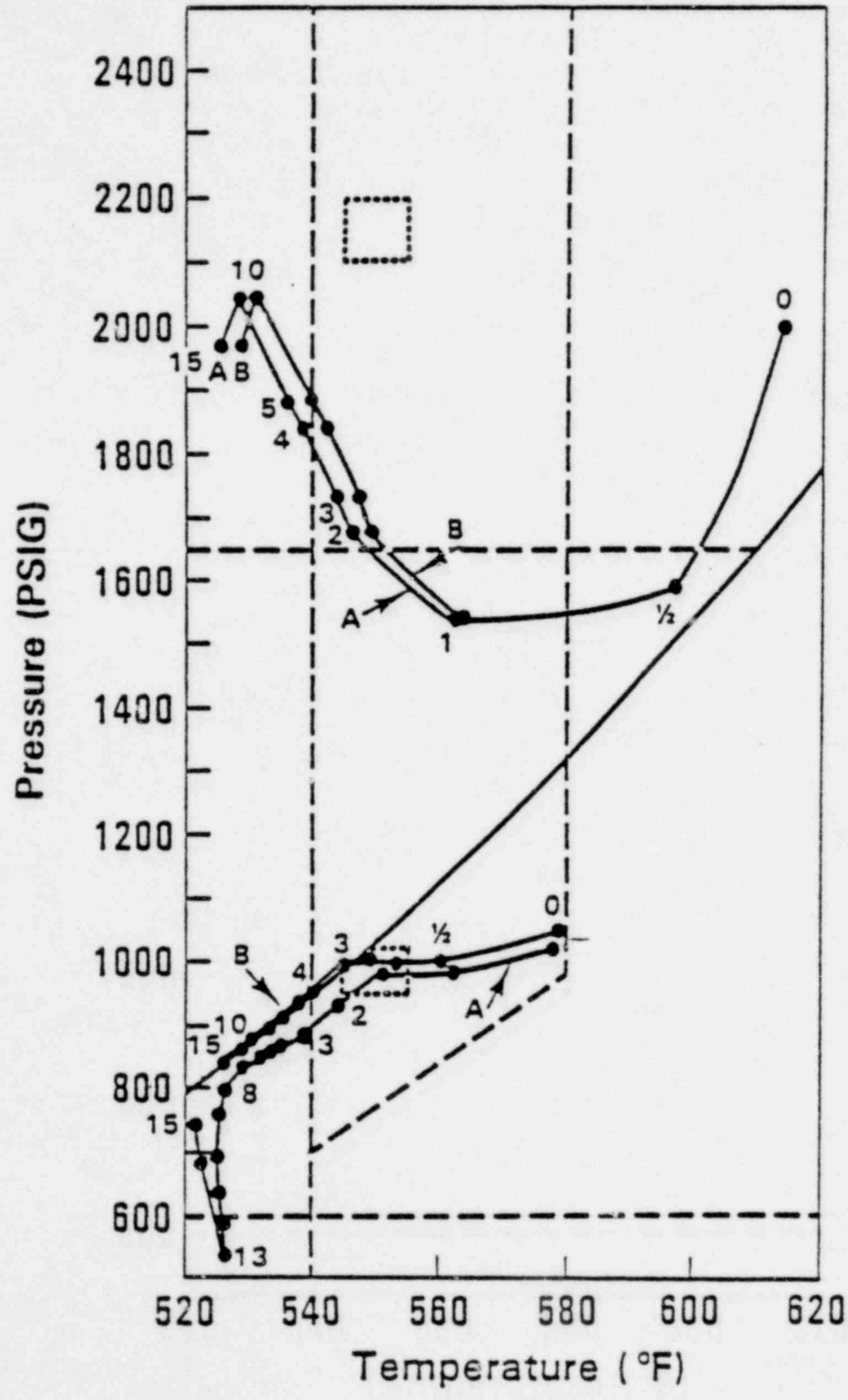


Figure 8
TMI-2 Accident

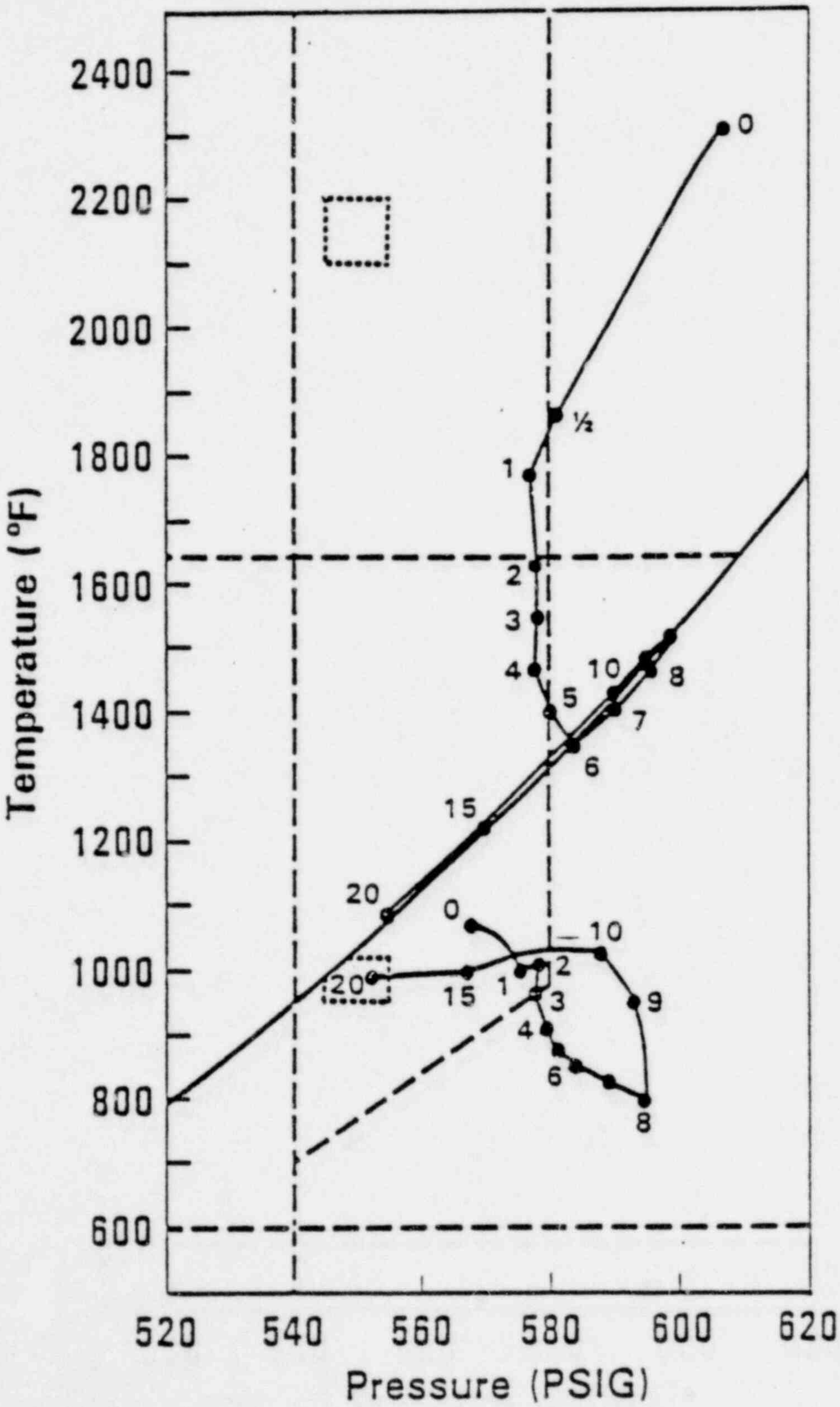


Figure 9
Natural Circulation Expected Post Trip Ranges

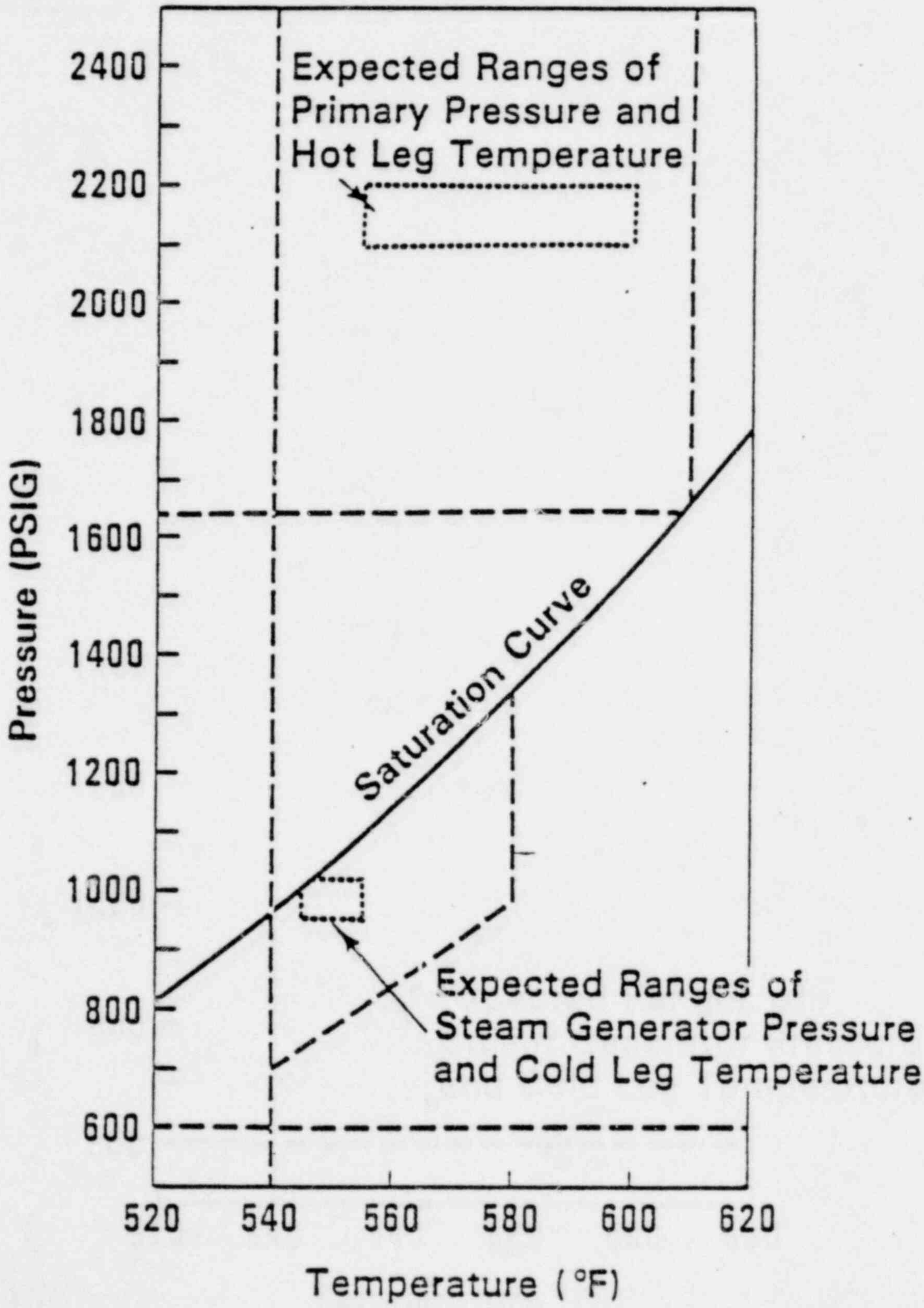
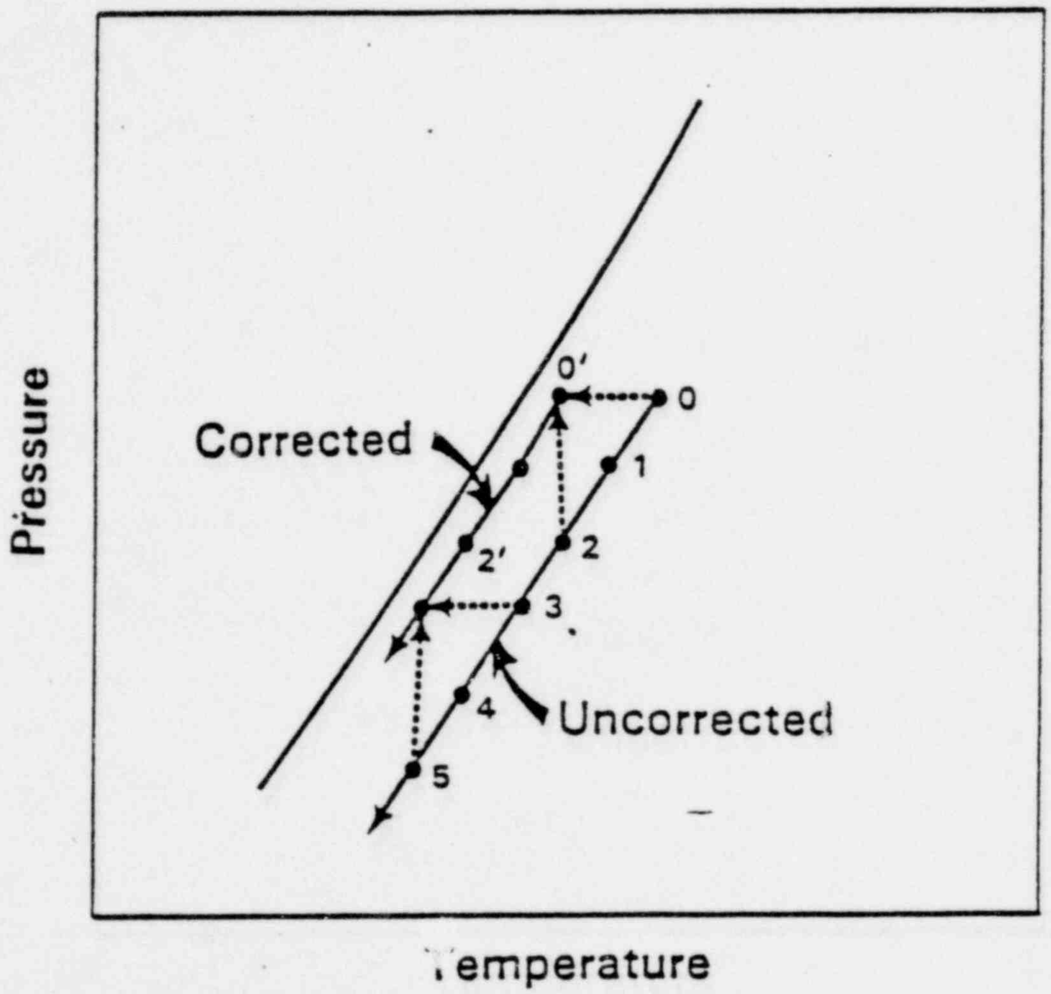


Figure 10

Effects of Steam Generator Pressure to Cold Leg Temperature Delay



Data gathered at time 0 is pressure at 0, temperature at -2

Plotting pressure at 0 against temperature at 2 shows direct effect of pressure on temperature.

Figure 11
Overcooling During Natural Circulation

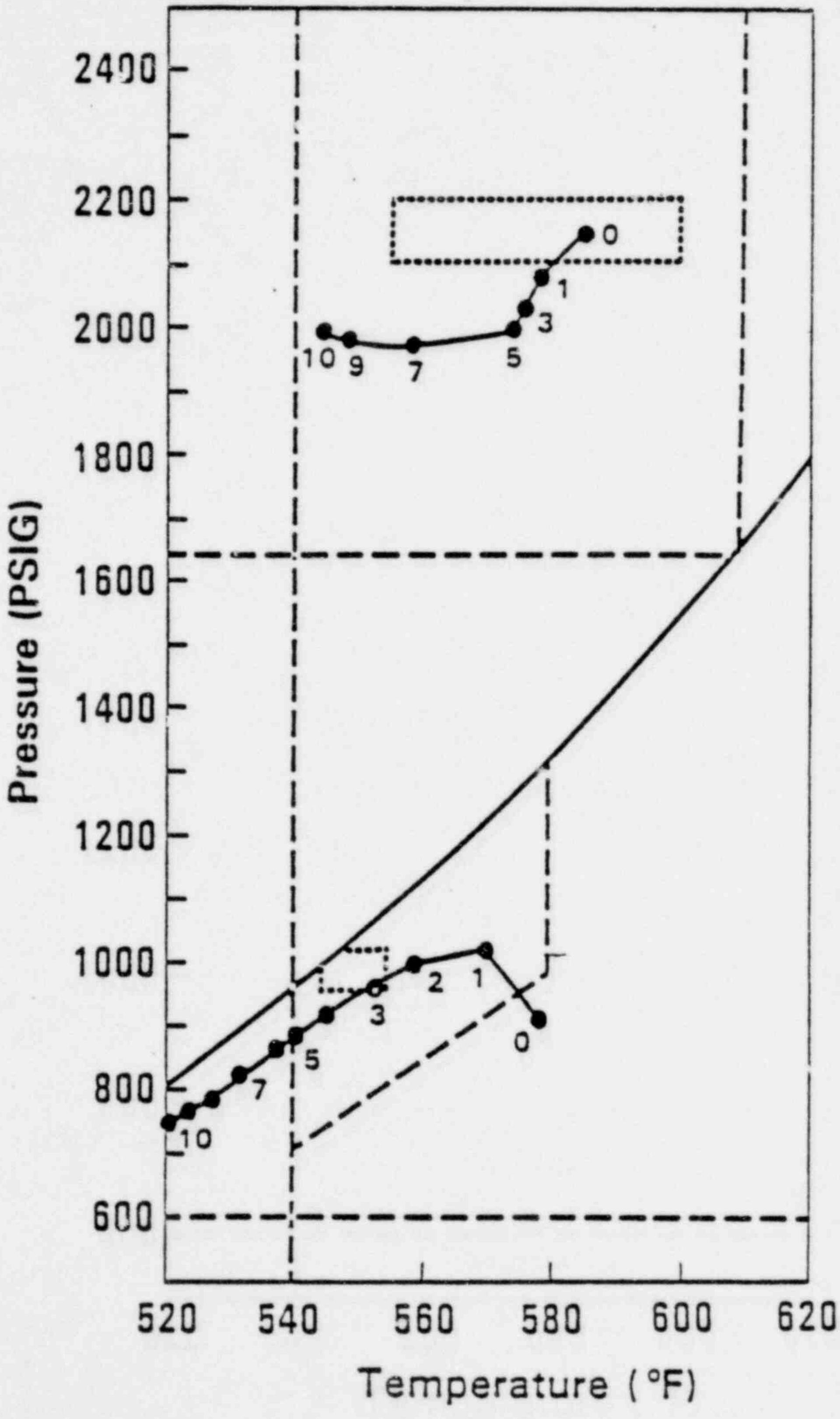


Figure 12
Reactor Trip
Stuck Open TBV
U-Tube Steam Generator