INDIANA & MICHIGAN ELECTRIC COMPANY
DONALD C. COOK NUCLEAR PLANT

FACILITY CONCEPTUAL DESIGN DESCRIPTION

FOR THE

TECHNICAL SUPPORT CENTER AND THE EOF.

<u>ATTACHMENT TO AEP:NRC:0531A</u>

Revised: September 15, 1982 and incorporated as Attachment 2 to AEP:NRC:0531E

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1. INTRODUCTION

1.1 SYSTEM FUNCTIONS:

The D.C. Cook Plant Technical Support Center Data System is being developed and designed using the guidelines of NUREG 0696 to provide the plant operating and technical support personnel with the pertinent plant information to facilitate the emergency response to an accident. This System, which utilizes the Westinghouse P2500 TSC Computer Systems, can also be used during normal plant operation for other functions such as plant performance analysis, personnel training etc.

This system consists of two similar computerized data acquisition, processing and display systems, one for each D.C. Cook Unit. The four major functions provided by this computer system are:

1.1.1 TECHNICAL SUPPORT CENTER (TSC):

The computer system will receive, store, process and display on color CRT terminals and/or on hard-copy terminals the real time data acquired from various plant systems. Pre-trip and post-trip data are also collected and can be processed and displayed by the computer. This system will facilitate the assessment of the plant's condition by plant operating and technical support personnel. The data displays of the Technical Support Center function will provide sufficient information to determine:

- Plant steady state operating conditions prior to the unit trip.
- Transient conditions producing the initiating event and system behavior during the course of the accident.
- Present conditions of the plant.

The TSC data display system may be used for:

- Reviewing the accident sequence.
- Determining appropriate mitigating actions.
- Evaluating the extent of any damage.
- Determining plant status during recovery operations.

This function will be described in details in Section 3.

1.1.2 PLANT SAFETY STATUS DISPLAY (PSSD):

This PSSD system was designed in accordance with the guidelines for the Safety Parameter Display System (SPDS) of NUREG 0696. This PSSD system, which displays the safety status of the plant in a format that can be easily recognized by the control room operators, will help the operators to detect any abnormal condition in a timely manner. Additional features of this PSSD system will help the operators and technical support personnel to obtain detailed information on the safety systems of the plant. Detailed descriptions of this system are provided in Section 4.

1.1.3 NUCLEAR DATA LINK (NDL)

The TSC computer system has a built-in off-site data transmission capability which can be used for interfacing with a future Nuclear Data Link (NDL) Sub-System.

1.1.4 BYPASS & INOPERABLE STATUS INDICATION SYSTEM (BISI):

The BISI system provides the operators and technical support personnel with a clear indication of the availability of the plant safety systems (ESF Systems). Detailed descriptions of this system are provided in Section 5.

1.2 REPORT BASIS:

This report is based on the proprietary Westinghouse WCAP Report 9725 "Westinghouse Technical Support Complex" which was submitted to the NRC. Appropriate modifications were made to reflect the specific design of D.C. Cook Units 1 and 2.

2. THE DATA ACQUISITION & DISPLAY SYSTEM

2.1 THE COMPUTER SYSTEM:

Figure 2.1 shows the computer system hardware for each Cook Unit. Multiple 16-bit high speed minicomputer and memory devices are used to process plant data, generate displays and perform other man-machine interface functions. The system is configured in a fault tolerant design. If a central processing unit (CPU) or a portion of memory fails, the system will automatically reconfigure itself to perform its designated functions.

2.2 INPUT SYSTEM

Figure 2.2 shows the schematic diagram for the TSC computer System. Input signals from the control room and other plant locations are taken to the remote Input/Output (I/O) cabinets. Signal isolation is provided in the I/O cabinets so that no failure on the output side of the I/O cabinets will affect the input signals. In addition to these isolators, all signals coming from the safety systems are taken after the existing qualified isolators on these systems. The input signals, after going through the isolators, will be converted to binary information on the input cards and then are multiplexed to the computer. Each analog signal channel has its own Analog/Digital Converter, thus providing a high degree of reliability for the input system.

2.3 DATA DISPLAY SYSTEM

2.3.1 Technical Support Center Room

Each D.C. Cook Unit has a dedicated command console located in the Onsite Technical Support Center. Each command console is equipped with two color CRT displays and a video hard copier (which can be used to obtain a hard copy of the screen image). One CRT is dedicated to the PSSD function and the second CRT is a general purpose display. Three satellite stations, each with a color CRT display, are also provided. The satellite stations can be connected to either Cook Unit 1 or Unit 2 TSC Computer System. A shared video hard copier is provided for the three satellite CRTs.The satellite stations are arranged so that visual access from the command station can be maintained while still providing sufficient room to minimize noise and distrubance. For printing lengthy reports, a line printer is provided.

2.3.2 Control Room:

Two redundant PSSD display CRTs and two redundant BISI CRTs are provided in each control room. A video hard copier is also provided to obtain had copy output from the CRT screen image.

2.3.3 Emergency Operating Facilities (EOF):

A color CRT terminal, which can be connected to either Cook unit TSC computer, is provided in the Emergency Operating Facilities. The remote CRT can be used to display all of the displays available on

the PSSD, TSC and BISI functions except for the top level iconic display of the PSSD function. This iconic display was designed for early recognition of an event by the control room operators and therefore is not included in the EOF.

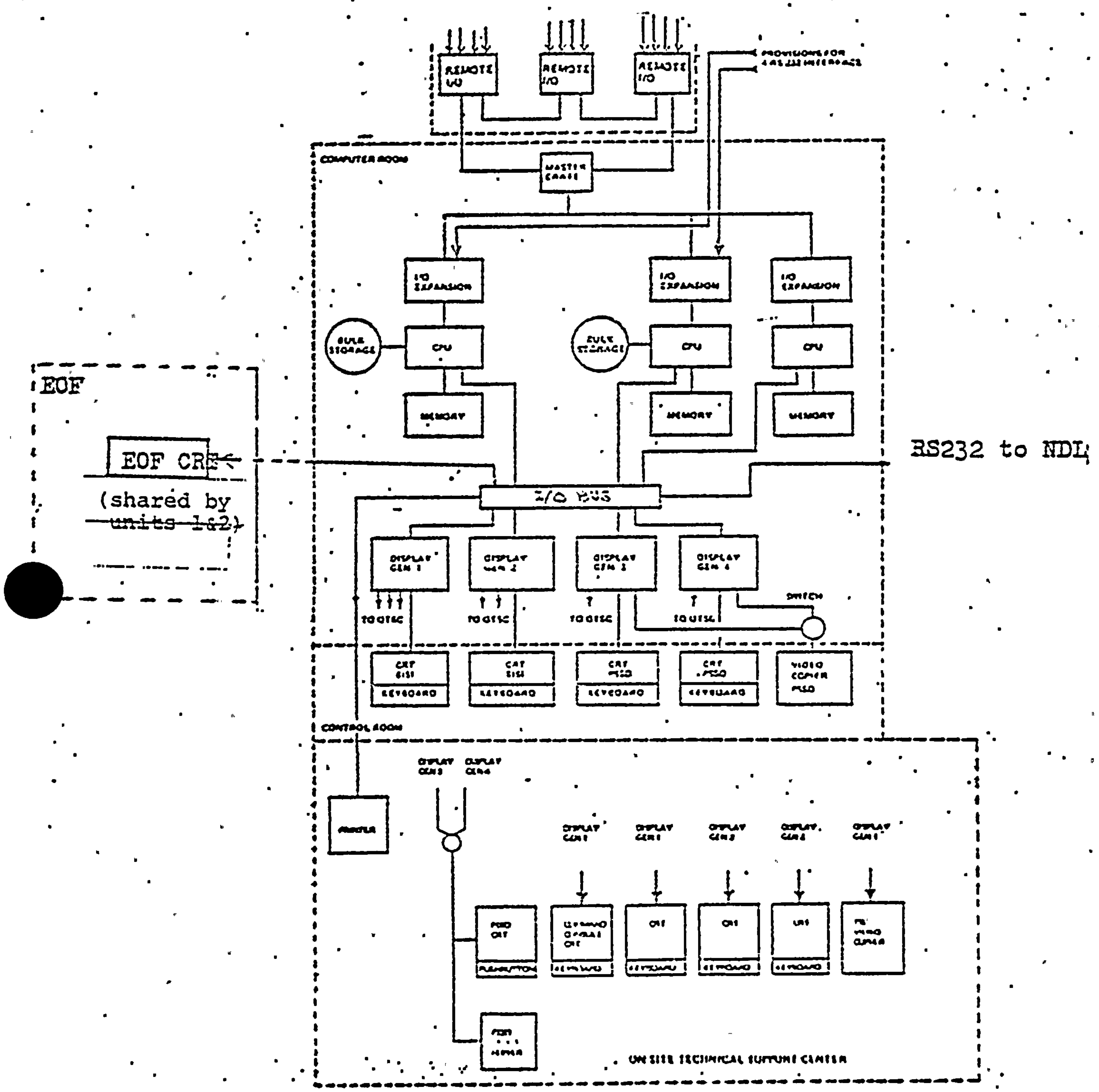

Figure 2.1. Technical Support Complex System Configuration

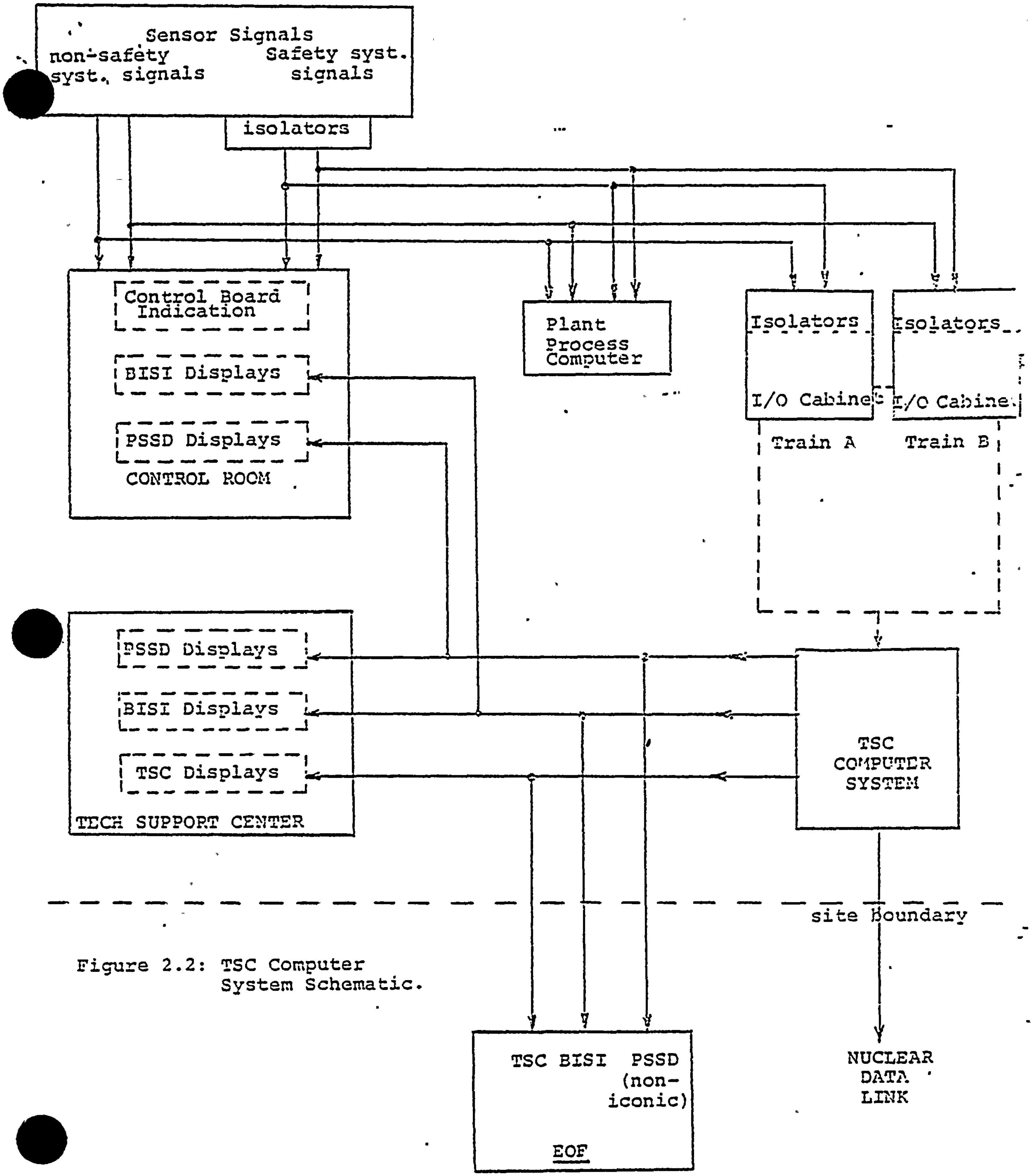

Figure 2.2: TSC Computer System Schematic.

3. ONSITE TECHNICAL SUPPORT CENTER

3.1 DESIGN BASIS:

The Onsite Technical Support Center (OTSC) serves as the focal point for post-accident recovery management. As such, it must have the capability to access, display and transmit pertinent plant status information independent of actions in the control room. The Technical Support Center function of the TSC Computer System was designed to satisfy the following requirements:

1. Personnel in the OTSC must have access to the real time information defining the current status of critical plant systems and functions.
2. The TSC function must have the capability to store historical pre-event and post-event data in order to enable a diagnosis and evaluation of the event to determine the extent of any possible plant system damage.
3. The TSC function must have the capability to access and display plant parameters independent of actions in the control room.
4. The interface of the TSC system equipment with exisiting plant instrumentation must not result in any degradation of the plant protection system, control room or other functions.
5. Parameters to the extent possible should be from the same source that is used for control room indications to ensure data consistency.
6. The TSC system must have the capability of interfacing with communication equipment for the off-site transmission of pertinent plant data.

7. The users must be able to create or modify displays to meet the needs as conditions may dictate.

3.2 INPUT DETERMINATION

In order to define the information which must be available in the OTSC, a generic study of critical plant systems and key safety functions (as listed in Table 3.1) was conducted by Westinghouse. This study resulted in a list of parameters to be monitored by the computer for the Technical Support Center function. This Westinghouse parameter list was reviewed and made Cook Plant specific by AEP. Table 3.2 lists the principal parameters and Table 3.3 lists the basis for input selection. Redundancy and diversity of process indications are utilized to satisfy concerns associated with unavailable signals due to sensor failure. Some refinement of the input parameters list may be made after the submittal of this conceptual design report

3.3 OTSC OPERATOR INTERFACE

The ability of the OTSC to be an effective tool in post-accident recovery management is a function of the inputs provided and the ability to present information in a meaningful and organized manner. As stated previously, the man-machine interface is through the use of interactive graphic color CRT displays. The interface functions in the OTSC consist of displays and console functions.

The display types available for OTSC personnel use consist of graphic and alphanumeric displays which are both preformatted and user constructible. Examples of the types of displays available are shown in Figures 3.1, 3.2 and 3.3. Figure 3.1 is an example of a preformatted system status display, gathering important system and loop parameters onto a single page of display. Figure 3.2 shows more detailed information on individual parameters such as information on sensor status, current value, and high and low limits.. Figure 3.3 is an example of a graphic trend display showing a time history of related parameters. Highlighting techniques for indicating parameters or conditions of interest utilize both color and achromatic means.

By providing a combination of both preformatted and user constructible displays the OTSC personnel are provided with prearranged quickly accessible system information and the flexibility to permit the tailoring of information presentation to meet specific needs as conditions dictate. The specific content of preformatted displays will be determined by analyzing post-accident data requirements in terms of event evaluation, the safety status of the plant, and long-term recovery planning. Displays will also be designed to reflect plant specific design details.

Display access is provided both by dedicated functional console pushbuttons and standard keyboard entries. Dedicated keys provide access to the most frequently used displays or functions. For other functions access can be either direct by entering short codes or by utilizing an instruction function to determine the identification code for a display if it is unknown.

Other types of information is available through the console keyboard. These consist of functions such as point review, logs, post-trip historical data review, and offsite data transmission.

The point review functions enable the console operator to review plant sensor information. The types of review functions available are:

1. Values of individual points.

2. Points removed from scan.

3. Points removed from limit checking.

4. Points failed under quality checking routines.

5. Points whose scan frequencies have been changed from the normal scan frequencies.

There are log functions available to the OTSC personnel which can be displayed on CRTs with periodic updates or output onto a hard copy device such as a line printer. These functions can be preprogrammed and automatically initiated or specified and initiated by console operator input.

The post-trip review function provides the capability to review historical data to aid in an event evaluation. This function continuously stores in memory an updated table of preassigned sensor values for a predefined period. Upon the occurrence of a disturbance (e.g., plant trip) the system continues to store data for a defined time period. After this period, the entire data record can be reviewed by the OTSC personnel on CRTs and/or output to hard copy devices for permanent record storage purposes.

The offsite data transmission function enables OTSC personnel to transmit plant data to offsite locations via owner supplied communications systems. The OTSC operator can initiate transmission of data either on a "one-shot" or periodic basis. The transmitted data can be arranged into four edited versions for the specific needs of separate offsite communications receivers such as the NRC.

TABLE 3.1

CRITICAL PLANT SYSTEMS/FUNCTIONS

Reactivity Control

Primary System Inventory

Core Heat Removal Capabilities

Availability and Capacity of Heat Sinks

Containment Integrity

Primary System Pressure and Temperature

Availability and Capacity of Alternate Water Sources

Availability and Operability of Critical Support Systems

Radioactivity Control

Table 3.2

TSC Parameters List

Variables	Min. No of Signals	Range
-RCS hot leg temp	4	0-700 deg F
-RCS cold leg temp	4	0-700 deg F
-RCS pressure	2	0-3000 psig
-Reactor water level	6	0-100 %
-RCS boron concentration	1	0-5000 ppm
-Pressurizer water level	2	0-100 %
-Steam generator level		
Wide range	4	0-100 %
Narrow range	8	0-100 %
-Steam line pressure	8	0-1400 psig
-Containment pressure	2	-5-+36 psig
-Containment water level		
Low range	2	589'-599' elev.
high range	2	599'-614' elev.
-RWST water level	2	0-100 %
-Condensate storage tank level	2	0-100 %
-Boric acid tank level	3	0-100 %
-Aux feed water flow	4	0-250 Klbs/hr
-Main feed water flow	4	0-5000 Klbs/hr
-High head injection flow	4	0-200 gpm

Table 3.2

TSC Parameters List

Variables	Min. No of Signals	Range
-Low head injection flow	4	0-5500 gpm
-Core exit temperature	16	0-2500 deg F
-Component cooling water flow	2	0-10000 gpm
-Component cooling water temp.	2	32-200 deg F
-Containment hydrogen concent.	2	0-30 %
-Containment temperature	8	0-100 deg F
-Neutron flux	2	0-120 % power
-Control rod position	53	Full in or not
-Primary system relief & safety valves	4	Closed-not closed
-Sec. syst. relief valves	4	Closed-not closed
-Containment isolation valves	139	Closed-not closed
-PZR relief tank pressure	1	0-100 psig
-PZR relief tank level	1	0-100 %
-PZR relief tank temp.	1	50-350 deg F
-RCS degree of subcooling	N/A[1]	200 sub-5 super
-Accumulator level	8	0-100 %
-Accumulator pressure	8	0-700 psig
-Accumulator isolation valves	4	Closed-not closed
-Aux building sump level	1	0-flood level
-RHR system flow	2	0-7000 gpm

Table 3.2

TSC Parameters List

Variables	Min. No of Signals	Range
-RHR heat ex. outlet temp.	2	0-400 deg F
-Boric acid charging flow	1	0-10 gpm
-RCS let-down flow	1	0-200 gpm
-RCS make-up flow	1	0-200 gpm
-Emerg. ventilation damper	4	closed-not closed
-Status of standby power	8	Energized or not
-High radioactivity liquid tank level	1	0-100 %
-Radioactive gas decay tk press	4	0-150 psig
-Reactor Coolant Pumps status	4	0-1200 amps
-PZR heater bank status	2	0-200 amps
-Meteorology		
Wind direction	1	0-360 deg
Wind speed	1	0-100 miles/hr
Atm. delta temp.	1	0-50 deg F
-Radiation[2]		
Containment area radiation	1	.1-10E4 mR/hr
Containment radio gas	1	10-10E6 cpm
Containment air particulate	1	10-10E6 cpm
Unit Vent radio gas	1	10-10E6 cpm
Unit Vent iodine	1	10-10E6 cpm

Table 3.2

TSC Parameters List

Variables	Min. No. of Signals	Range
- Radiation (continued)		
Steam gen. blow down	1	10-10E6 cpm
Condenser air ejector	1	.1-10E4 mR/hr
Cooling water East	1	10-10E6 cpm
Cooling water West	1	10-10E6 cpm
Service water East	1	10-10E6 cpm
Service water West	1	10-10E6 cpm
Waste Liquid off-gas	1	10-10E6 cpm
Waste gas decay tank	1	10-10E6 cpm
Control room area	1	.1-10E4 mR/hr
Spent fuel area	1	.1-10E4 mR/hr
Charing pp. room area	1	.1-10E4 mR/hr

Note 1: Degree of subcooling will be independently calculated by the TSC computer.

Note 2: The radiation signals listed above are signals from the existing radiation detectors. AEP is in the process of implementing a new Radiation Monitor System at Cook Units 1 and 2, and will provide a separate Radiation Data Display System for the TSC and EOF.

TABLE 2-3

TSC INSTRUMENT BASIS

(b,c)

PARAMETER	INITIAL EVENT DIAGNOSIS*	BASIS
Containment Pressure	- Determine if break is inside or outside of containment	- Monitor containment conditions
Steamline Pressure	- Determine if high energy secondary line rupture occurred	- Maintain an adequate reactor heat sink - Monitor secondary side pressure to: - verify operation of pressure control steam dump system - monitor RCS cooldown rate
Narrow Range Steam Generator Water Level	- Determine if malfunction of secondary side system has occurred	- Monitor heat sink - Maintain steam generator water level
Wide Range Steam Generator Water Level	- None	- Determine if heat sink is being maintained
Boric Acid Tank Level	- None	- Verify RCS boration system functions for adequate reactivity control
Condensate Storage Tank Level	- None	- Maintain adequate water supply for auxiliary feedwater pumps
Refueling Water Storage Tank Level	- None	- Verify adequate supply of emergency core cooling water - Verify ECCS and containment spray system are functioning

*Certain indications on this table are used as secondary diagnoses as the operator proceeds through Post-Incident Recovery.

TABLE 2-3 (Continued)

TSC INSTRUMENT BASIS

(b,c)

PARAMETER	INITIAL EVENT DIAGNOSIS*	BASIS

Wide Range T_h and T_c	- None	- Maintain adequate reactor heat sink
		- Maintain the proper relationship between RCS pressure and temperature
		  - verify vessel NDTT criteria
		  - maintain primary inventory subcooled
		  - maintain safe shutdown condition
		  - maintain RHR considerations for cooldown
		  - monitor RCS heatup and cooldown rate

Pressurizer Water Level	- None	- Confirm if plant is in a safe shutdown condition - Determine ability to control RCS pressure - Monitor RCS inventory - Maintain pressurizer water level

*Certain indications on this table are used as secondary diagnoses as the operator proceeds through Post-Incident Recovery.

TABLE 2-3 (Continued)

TSC INSTRUMENT BASIS

(b,c)

PARAMETER	INITIAL EVENT DIAGNOSIS*	BASIS
System Wide Range Pressure	- None	- Determine if plant is in a safe shutdown condition - Maintain the proper relationship between RCS pressure and temperature - verify vessel NDTT criteria - maintain primary inventory subcooled (particularly with loss of offsite power) - maintain RHR considerations for cooldown
Containment Building Water Level	- Determine whether high energy line rupture has occurred inside or outside containment	- Determine NPSH for recirculation mode cooling - Determine which equipment in containment is submerged
Condenser Air Ejector Radiation	- Determine if steam generator tube leak has occurred	- Monitor radioactivity release path to environment
Steam Generator Blowdown Radiation	- Determine if steam generator tube leak has occurred	- Monitor radioactivity release path to environment
Containment Radiation	- Determine if high energy line break or fuel mishandling accident	- Monitor radioactivity release path to environment - Determine accessibility to containment building

*Certain indications on this table are used as secondary diagnoses as the operator proceeds through Post-Incident Recovery.

TABLE 2-3 (Continued)

TSC INSTRUMENT BASIS

(b,c)

PARAMETER	INITIAL EVENT DIAGNOSIS*	BASIS
		- Determine if significant fuel damage has occurred - Monitor environmental conditions around equipment in containment
Auxiliary Feedwater Flow	- None	- Determine if sufficient flow exists to maintain heat sink
High Head Safety Injection Flow	- None	- Determine that ECCS is delivering flow - Monitor ability to keep core covered
Low Head Safety Injection Flow	- None	- Determine that ECCS is delivering flow - Monitor ability to keep core covered - Infer spray operation
Area Radiation Monitoring in Auxiliary Building and Control Room	- Determine if source of accident is outside containment building	- Monitor accessibility to plant zones/equipment - Monitor radioactivity release path to environment - Monitor effectiveness of cleanup/holdup systems - Monitor integrity of long-term cooling system

*Certain indications on this table are used as secondary diagnoses as the operator proceeds through Post-Incident Recovery.

TABLE 2-3 (Continued)

TSC INSTRUMENT BASIS

(b,c)

PARAMETER	INITIAL EVENT DIAGNOSIS*	BASIS
		- Monitor habitability of the control room
Core Exit Thermocouples	- None	- Determine if core is being cooled
Neutron Flux	- None	- Monitor ability of reactivity control systems to keep the core subcritical
		- Determine if plant is in a safe shutdown condition
Degree of Subcooling of Primary Coolant	- None	- Maintain adequate reactor heat sink
		- Maintain safe shutdown conditions
Primary System Safety and Relief Valve Position	- None	- Maintain primary system inventory
		- Monitor radioactivity release paths into the containment
Pressurizer Relief Tank Pressure, Temperature, and Level	- None	- Monitor capacity to relieve primary coolant pressure
		- Monitor radioactivity release paths into the containment
Containment Isolation Valve Position	- None	- Monitor radioactivity release paths to the environment
		- Monitor status of containment isolation

*Certain indications on this table are used as secondary diagnoses as the operator proceeds through Post-Incident Recovery.

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TABLE 3.3 (Continued)

TSC INSTRUMENT BASIS

(b,c)

PARAMETER	INITIAL EVENT DIAGNOSIS*	BASIS
Secondary Safety, Reliefs, and Atmospheric Dump Valves	- None	- Monitor radioactivity release paths to the environment - Monitor secondary system integrity
Accumulator Tank Level	- None	- Monitor primary system inventory - Determine whether the accumulator tanks have injected into the RCS
Accumulator Isolation Valve Position	- None	- Determine system operation
RHR System Flow	- None	- Monitor primary system inventory - Monitor core heat removal capabilities
RHR Heat Exchanger Outlet Temperature	- None	- Monitor core heat removal capabilities
Component Cooling Water Flow and Temperature	- None	- Monitor system operation of a critical support system

*Certain indications on this table are used as secondary diagnoses as the operator proceeds through Post-Incident Recovery.

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TABLE 3.3 (Continued)

TSC INSTRUMENT BASIS

(b,c)

PARAMETER	INITIAL EVENT DIAGNOSIS*	BASIS
Boric Acid Charging Flow	- None	- Monitor primary system inventory - Determine boron concentration for reactivity control - Monitor ability to control RCS pressure
Letdown Flow	- None	- Monitor primary system inventory - Monitor ability to control RCS pressure - Monitor core heat removal capability - Determine boron concentration for reactivity control
Water Level in Closed Spaces Around Safety Equipment in Auxiliary Building	- None	- Monitor environmental conditions around required safety equipment outside of containment
Emergency Ventilation Damper Position	- None	- Ensure proper ventilation to vital areas under post-accident conditions
High Level Radioactive Liquid Tank Level	- None	- Monitor capacity to contain and store radioactive liquids

*Certain indications on this table are used as secondary diagnoses as the operator proceeds through Post-Incident Recovery.

TABLE 2-3 (Continued)

TSC INSTRUMENT BASIS

(b,c)

PARAMETER	INITIAL EVENT DIAGNOSIS*	BASIS
Radioactive Gas Holdup Tank Pressure	- None	- Monitor capacity to contain and store radioactive gases
Status of All Electric Power Supplies and Systems	- None	- Ensure adequate electric power to safety and support systems
Effluent Radioactivity Noble Gases, Radiohalogens, and Particulates	- None	- Monitor radioactivity release paths to the environment
Plant and Environs Radioactivity (Permanent and Portable Instruments)	- None	- Monitor release of radioactive materials not covered by effluent monitors
Sampling System	- None	- Determine RCS chemistry for reactivity control and extent of fuel clad damage
Meteorology (wind speed and direction temperature profile, and precipitation)	- None	- Monitor radioactive effluent transportation for emergency planning, dose assessments, and source estimates
Containment Atmosphere Temperature and Hydrogen Concentration	- None	- Monitor containment integrity - Monitor environmental conditions around equipment in containment

*Certain indications on this table are used as secondary diagnoses as the operator proceeds through Post-Incident Recovery.

WESTINGHOUSE PROPRIETARY CLASS 2

Systems Status - Reactor Coolant System				
	Loop 1	Loop 2	Loop 3	Loop 4
T average (°F)	595.2	595.2	595.2	595.2
Overpower ΔT (%PWR)	110.0	110.0	110.0	110.0
Overtemp. ΔT (%PWR)	110.0	110.0	110.0	110.0
Cold leg temp. (narrow range) (°F)	559.8	559.8	559.8	559.8
Hot leg temp. (narrow range))°F)	624.0	624.0	624.0	624.0
Reactor coolant flow (%)	100.0	100.0	100.0	100.0
Reactor coolant pressure - WR (PSIG)	2250.0	2250.0	2250.0	2250.0
Pressurizer pressure (PSIA)	2250.0			
Pressurizer vapor temp. (°F)	563.8			
Pressurizer liquid temp. (°F)	565.2			
Pressurizer relief tank pressure (PSIG)	1.5			
Pressurizer relief tank level (%)	77.6			
Pressurizer relief tank temp. (°F)	110.3			
Pressurizer safety relief temp. (°F)	120.0			

Figure 3.1 System Status Display at Onsite Technical Support Center (Example)

Parameter Summary					
Point	Description	Value	Range	Units	Status
TO400	RCS Loop 1 Hot Leg T	593.4	0:700	DEGF	Normal
TO406	RCS Loop 1 Cold Leg T	547.2	0:700	DEGF	Normal
PO480	RCS Pressure	2234.1	0:3000	PSIG	Normal
LO421	Stm Gen 2 Narrow Range Level	39.1	0:100	PC	Low
PO549	Steamline Pressure	893.0	0:1100	PSIG	Normal
LO103	RWST Level	100.0	0:100	PC	Normal
LO114	Boric Acid Tank Level	98.8	0:100	PC	Normal
LO119	Condensate Storage Tank Level	56.4	0:100	PC	Normal
LO947	Containment Bldg. Water Level	3.3	0:100	PC	High

Figure 3.2: Parameter Information Display at Onsite Technical Support Center (Example)

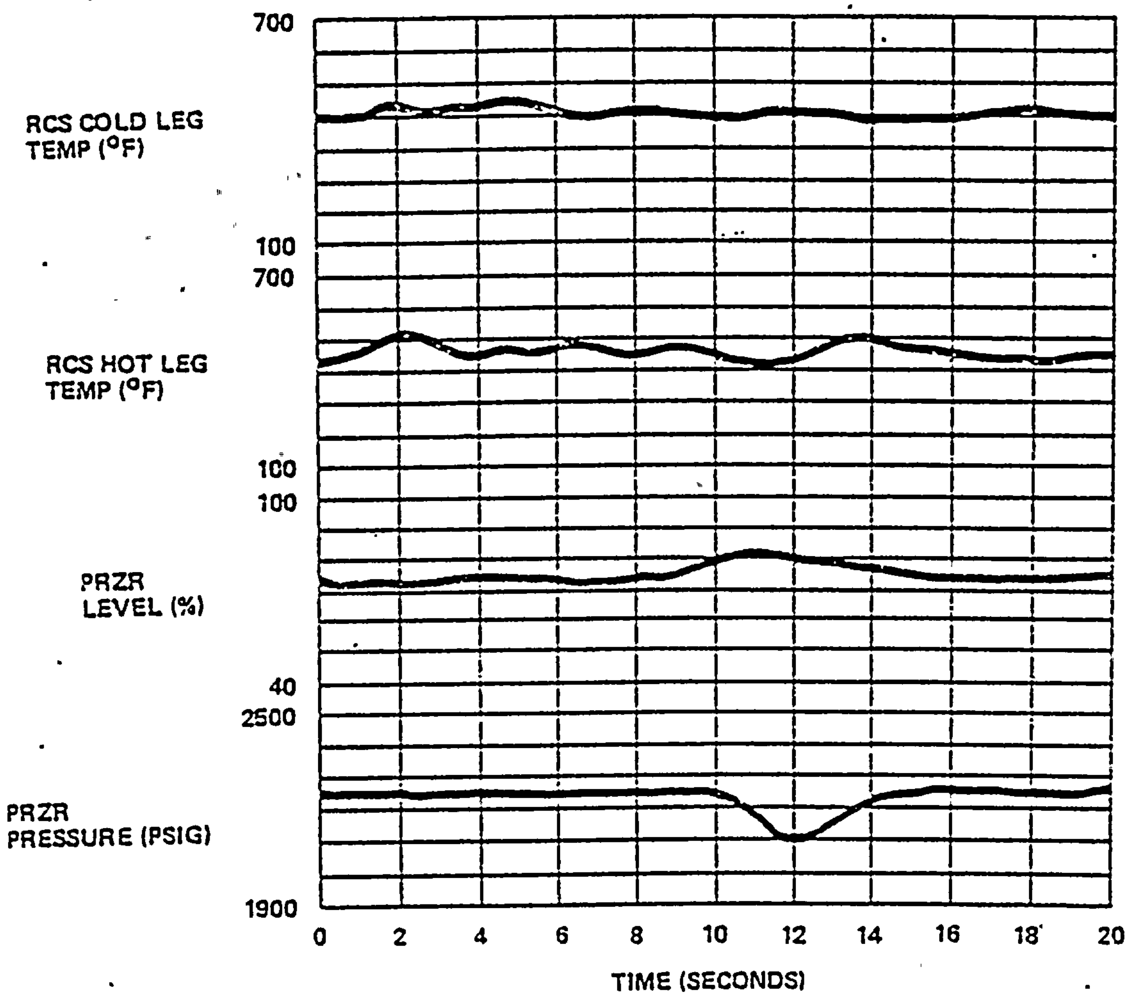


Figure 3.3 Graphic Display at Onsite Technical Support Canter (Example)

4.0 PLANT SAFETY STATUS DISPLAY

4.1 PURPOSE

The function of the Plant Safety Status Display (PSSD) is to present a succinct account of the overall plant safety status to the control room operator (or supervisor). The entire data base should be available to the operator arranged in a format that will enhance his response to events and the diagnoses of the cause of the event. Because the PSSD serves as an important interface between the plant process and the operator, the information presentation should be defined in terms of parameters and logic supportive of defined operating procedures for dealing with abnormal events.

4.2 INPUT DETERMINATION

In order to determine the required operational modes for the PSSD one (b,c,e)
must first consider the various types of transients which may occur. A
review of postulated plant transients (events) indicated that they can
be divided into two basic categories:

1. Slow transients which do not result in immediate protection systems actuation and for which the control room operator has an opportunity to react to possibly terminate the event before safety systems are required to function.

2. Fast transients which result in almost immediate reactor trip and possibly safeguards actuation and for which the control room operator's response is to react to ensure that appropriate safety measures have been taken and to diagnose the event.

Because of the fact that different parameters and signal ranges are (b,c,e)
associated with the two potential event types, the PSSD incorporates two (b,c,e)
operating modes. The first mode (TERMINATE MODE) is active while the

(b,c,e) [plant is in a normal operating condition and the second mode (MITIGATE MODE) is active following a reactor trip]. The parameters available for
(b,c,e) [each mode were chosen to maximize the useful amount of information to be
(b,c,e) displayed to the operator]. The role for which the PSSD provides [support for each of the operating modes] is as follows:

(b,c,e) [TERMINATE MODE

1. Monitor the plant process for abnormalities indicative of slow transients that do not result in immediate reactor trips and for which the control room operator might take corrective or protective action.

2. Monitor the integrity of the various boundaries to radioactive release.

MITIGATE MODE

1. Monitor the safety status of the as tripped condition.

2. Monitor for conditions which might lead to a breach of any of the levels of defense against radioactive release.

3. Monitor the condition of the barriers to radioactive release.

For any event, the safety status of the plant can be evaluated in terms of six basic safety concerns. These concerns can be stated as follows:

1. Saturation of Reactor Coolant

2. Reactivity Excursion

3. Loss of Primary Coolant Inventory

4. Loss of Pressure and Temperature Control]

5. Radioactive Release (b,c,e)

6. Containment Environment

By addressing key safety concerns, the consequences of abnormal events can be limited or mitigated. (b,c,e)

The key safety concerns can be related to specific abnormal occurrences. Tables 4-1 and 4-2 indicate key safety goals for some typical postulated events in terms of the PSSD operating mode. It must be noted that these events are typical and it is conceivable for multiple events to occur in undefinable sequences. For these reasons, the PSSD must be designed on the basis of key safety concerns rather than specific scenarios. (b,c,e)

In defining the inputs for the PSSD, two requirements have to be met as follows: (b,c,e)

1. The inputs selected must represent a minimum set sufficient for monitoring all possible events including those which might not have been anticipated. (b,c,e)

2. The selection of inputs must address conditions with potentially erroneous signals, conflicting indications, and parameters out of range (i.e., redundancy and diversity).

In response to the first requirement, the function of the PSSD has been considered in two ways. The primary function is to monitor the plant process in terms of satisfying the key safety concerns. As stated above, by guaranteeing that these concerns are addressed, the conditions of unanticipated events or event sequences can be satisfied. The second function of the PSSD is to support the monitoring function of the plant for postulated events and to provide a man-machine interface design that supports a defined evaluation process and procedures for responding to abnormal events. (b,c,e)

(b,c,e) In order to satisfy the [second consideration of evaluating erroneous signals and the need for redundancy and diversity, the PSSD must perform operations upon multi-sensor inputs to evaluate erroneous signals and be able to provide the operator with a diverse method of indicating the plant process. The inputs to the PSSD are chosen upon the basis of their direct relevance to the key safety concerns. Tables 4-3 and 4-4 list some specific inputs related to key safety concerns for several events].

4.3 MAN-MACHINE INTERFACE

(a,b,c) (a,c) The PSSD system will process the defined input data set of plant parameters at [two second intervals] and generate displays for redundant PSSD dedicated CRTs located in the control room. [A dedicated CRT will also be located in the Onsite Technical Support Center].

In order to achieve an effective man-machine interface, the display system must be designed to provide a logical and human engineered display structure and selection process in a manner which supports defined roles in which the operator is expected to perform during an abnormal occurrence.

(b,c) The role of the control room operator in [detecting and reacting to an abnormal occurrence is expected to follow the four basic activities] depicted in Figure 4-1. The display system structure should be defined
(b,c) such that it [supports an identifiable goal for each of the general activities shown in the figure. These goals] are defined as follows:

(b,c) [Activity: Detection

Goal: The control room operator should be in a state of readiness to make a rapid detection of incipient threats or actual events which may affect plant safety. The response of the operator would be based upon his knowledge of expected plant performance and his skill in controlling the plant process].

[Activity: Reaction (b,c)

Goal: The control room operator must immediately react to the detection of an event. His first objective is to assure that appropriate safety system responses have been taken and that key safety concerns are being addressed by observing critical plant parameters.

Activity: Diagnosis

Goal: Following the control room operator's immediate reaction it is then necessary to diagnose the cause(s) of the event and determine if any damage to the various barriers to radioactive release has occurred. The operational mode at this time would be based on the operator's knowledge supported by references to various abnormal and emergency operating procedures.

Activity: Terminate/Mitigate

Goal: At the later stages of the event the control room operator will need to implement the rules or strategies that have been identified as a result of the diagnosis activity. The operator's goal is to verify that corrective actions are satisfying the key safety concerns].

The display structure shown in Figure 4-2 [supports the specified control room operator activities and goals. The displays are structured into three levels of information ranging from general plant system summary information with a broad field of attention, secondly to a level of information with a narrower field of attention and more definitive information on subsystems and functions, and finally to a level of information containing individual sensor values and status]. (a,c,f)

(a,c,f) [Level 1 would contain information in the form of a continuous graphic display for each of the two operating modes of the PSSD. Information contained in the display would support the detection activity].

A major problem associated with the man-machine interface is the (a,c,f) [requirement that the plant operator sample and process a large number of plant parameters and perform what are termed multi-parameter decision processes. An advanced concept in graphic CRT display designed to aid the operator, is employed for Level 1 information in the PSSD]. Figure (a,c,f) 4-3 is an illustration of the display. [Each ray in the figure represents the scale for a process parameter. When the normal operating values for the parameters are plotted on the scales and lines are drawn connecting the points, a geometric pattern is developed. Positive deviations from the normal values result in points further away from the center of the figure while negative deviations result in points closer to the center of the figure. When the actual values of parameters are different from the normal or reference values, the result is a geometric pattern different from the original pattern].

(a,c,f) Figures 4-4 and 4-5 are preliminary versions of [Level 1 displays for each of the PSSD operational modes] for two sample events: Primary to Secondary Coolant System Leak and Primary Coolant System Leak to Containment. The parameters chosen for the displays were chosen to (a,c,f) [permit an evaluation of the key safety concerns].

(a,c,f) [This advanced graphic display provides two distinct advantages over conventional control room indicators: a concise, systems level oriented, integration of parameters and secondly, a graphic display format. The detection of an abnormal condition is enhanced as the operator task is now based upon the discrimination of two geometric figures. Multi-parameter decisions and event evaluation is facilitated by the integrated nature of the display and the fact that only differences in parameters are highlighted by the display. The operator upon detecting abnormalities is then able to seek more specific information at other information levels to support the reaction, diagnosis, and terminate/mitigate activities].

The information at Level 2 is an expansion of each of the key safety concerns and systems. More detailed information is provided on the status of the process. For example, the values of pressures and water levels in individual steam generators could be provided at this level. In addition, trend displays for the previous 5 minutes of operation of Level 1 primary display parameters are provided. Diversity in process indications at this level will be employed to enable the operator to verify conclusions. At Level 3, the data is detailed further to provide information on the status of individual sensors, multiple measurement points, and data anomalies. The sensor values are annotated to include such things as data-out-of-range and process limits. Information on suspect data quality is carried into upper display levels. (a,c,f)

TABLE 4-1

PLANT SAFETY STATUS DISPLAY - SAFETY
GOALS - TERMINATE MODE TRANSIENTS

(b,c,e)

Reactor Control Systems Malfunction

- Stop rod motion
- Maintain core thermal and nuclear parameters within limits

Reactor Coolant System Makeup Control

- Prevent core thermal and nuclear parameters from exceeding limits
- Maintain pressurizer pressure and level

Inadvertent Depressurization (Slow)

- Terminate depressurization
- Restore system pressure

Reactor Coolant System Leak

- Limit radioactive release
- Maintain pressurizer pressure and level

TABLE 4-2

PLANT SAFETY STATUS DISPLAY - SAFETY
GOALS - MITIGATE MODE TRANSIENTS

(b,c,e)

Reactor Trip

- Maintain heat sink via steam generators
- Maintain subcooling by controlling steam pressure
- Maintain pressurizer level

Station Blackout

- Provide secondary heat sink
- Maintain subcooling
- Maintain pressurizer level

Emergency Boration

- Prevent return to criticality

Operation with Natural Circulation

- Provide heat sink
- Control subcooling
- Maintain pressurizer level

Spurious Safety Injection

- Determine safety injection is not required and terminate action

Loss of Reactor Coolant

- Verify and establish short term core cooling
- Maintain long term shutdown and cooling

TABLE 4-2. (Continued)

PLANT SAFETY STATUS DISPLAY - SAFETY
GOALS - MITIGATE MODE TRANSIENTS

(b,c,e)

Loss of Secondary Coolant

- Establish stabilized reactor coolant system and steam generator conditions
- Minimize energy release
- Prevent lifting of pressurizer safety valves
- Isolate auxiliary feed to affected steam generator
- Borate to maintain reactor shutdown margin

Steam Generator Tube Rupture

- Minimize radioactive material release
- Establish feedwater to unaffected steam generators and isolate faulted unit
- Maintain residual heat removal capability
- Maintain RCS subcooling
- Prevent over-flooding of faulty steam generator

TABLE 4-3

PLANT SAFETY STATUS DISPLAY TERMINATE MODE PARAMETERS (b,c,e)

Variable	Transient			
	Reactor Control System Malfunction	Reactor Coolant Makeup Control System Malfunction	Inadvertent Depressurization	Reactor Coolant System Leak
T_{avg}	X	X		
T_{ref}	X	X		
Rod position	X	X		
Delta T	X			
Startup rate		X		
Count rate		X		
Pzr. pressure			X	
Charging flow				X
Pzr. level				X
Comp. cool H_2O rad.				X
Containment rad				X
Air eject rad.				X
Blowdown rad.				X
Cont. humidity				X
Cont. temperatures				X
Cont. pressure				X
Prz. discharge piping temps			X	X
PRT pressure			X	X
PRT level			X	X
PRT temps			X	X
RCP seal temperature				X
RCP seal flow				X
RCP seal level				X
VCT flow				X

TABLE 4-4

PLANT SAFETY STATUS DISPLAY - MITIGATE MODE PARAMETERS

b,c,e)

Variable	Transient						
	Reactor Trip	Station Blackout	Emergency Boration	Operation with Natural Circulation	Loss of Coolant Accident	Loss of Secondary Coolant	Steam Generator Tube Rupture
Reactor trip breaker	X			X			
Startup rate	X						
Neutron flux	X		X				
Rod position	X		X				
Turbine trip		X					
Blackout signal		X					
T_{avg} (thermocouples)			X	X			
Rod bottom ind.			X				
Primary pressure				X	X	X	
T_{ref}			X				
Steam flow				X	X		
Feed flow				X	X		
Pressurizer level				X	X	X	X
Core thermocouples				X	X	X	
Cont. radiation						X	
Air ejector radiation					X		X
Blowdown radiation							X
Cont. pressure					X	X	
Pri. W.R. temp.					X	X	
Steam pressure					X	X	
Cont. sump level					X		
Cont. temperature						X	
Cont. humidity							
Charging flow							X
S.G. level					X	X	X
B.A. tank level					X		
Aux. feed flow					X	X	
SI flow					X		
RWST level					X	X	
CST level					X	X	

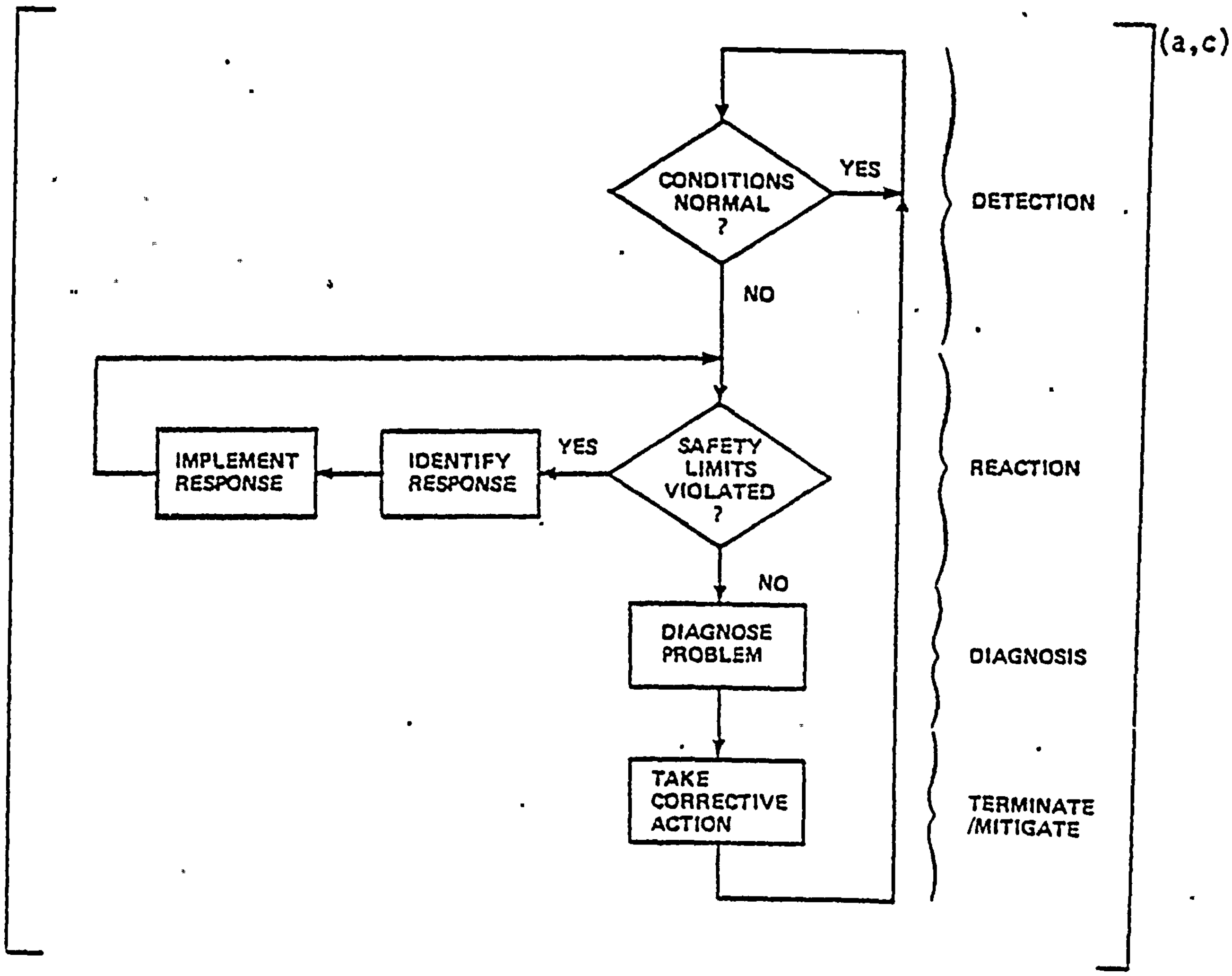


Figure 4-1. Operator Response Model

(a,c,f)

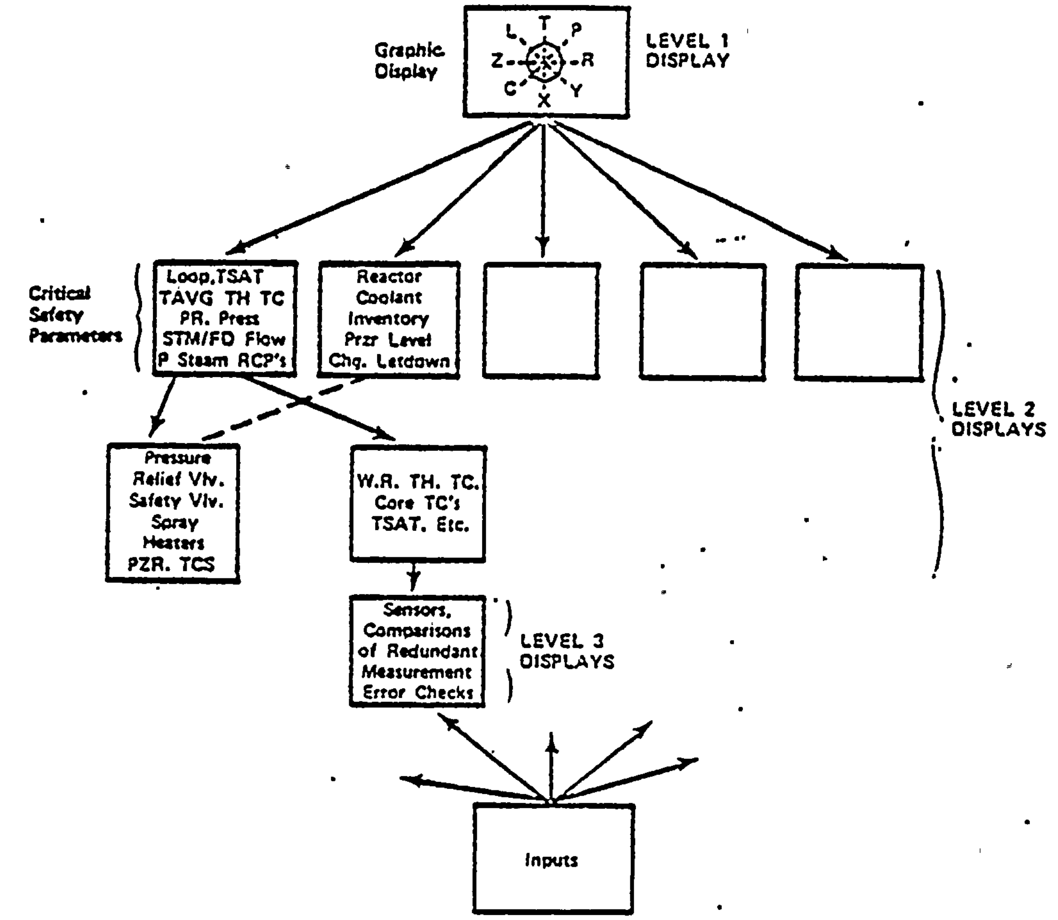


Figure 4-2. Display Structure of Plant Status Display

(a,c,f)

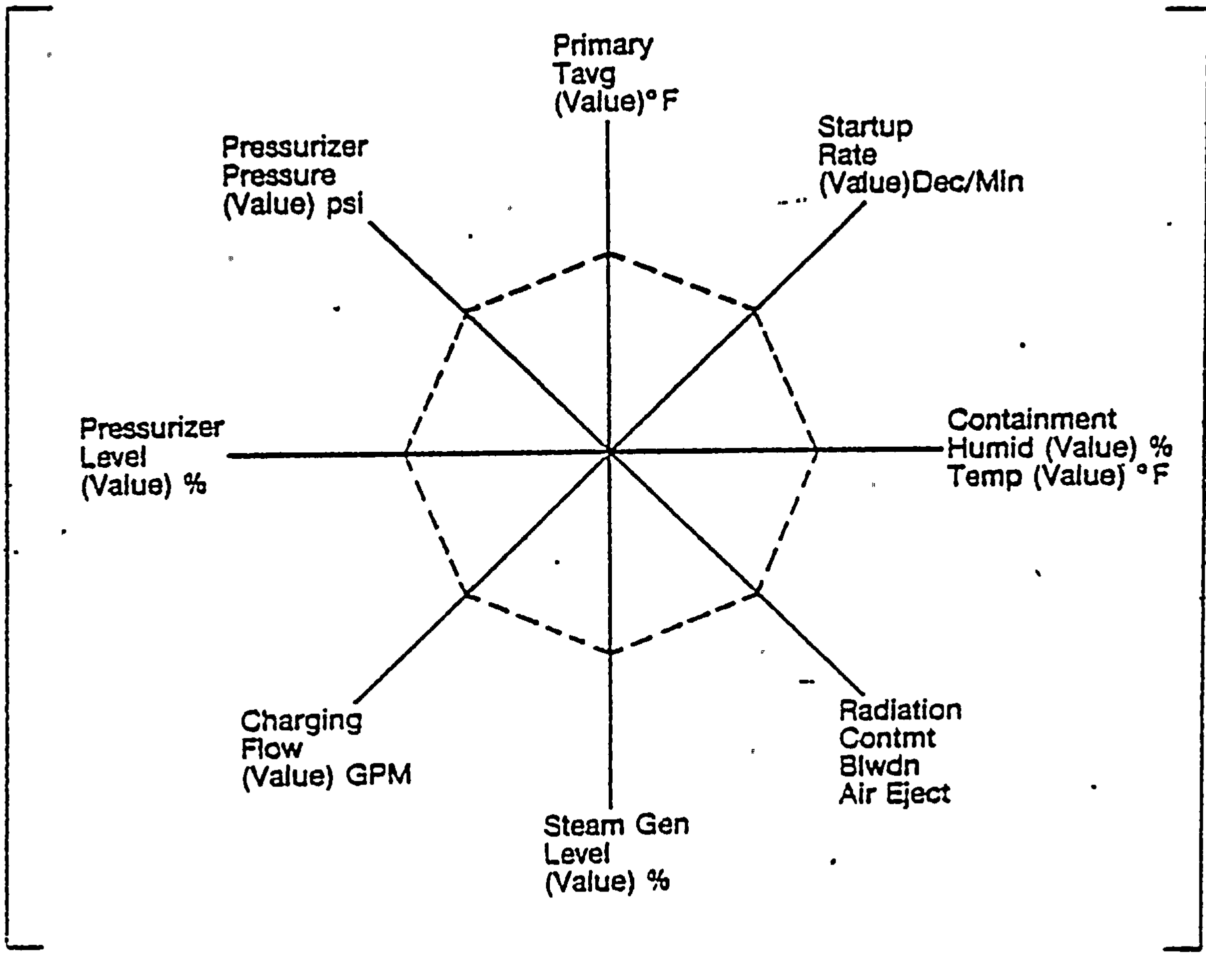


Figure 4-3. Sample Display — Plant Safety Status Display

(a,c,f)

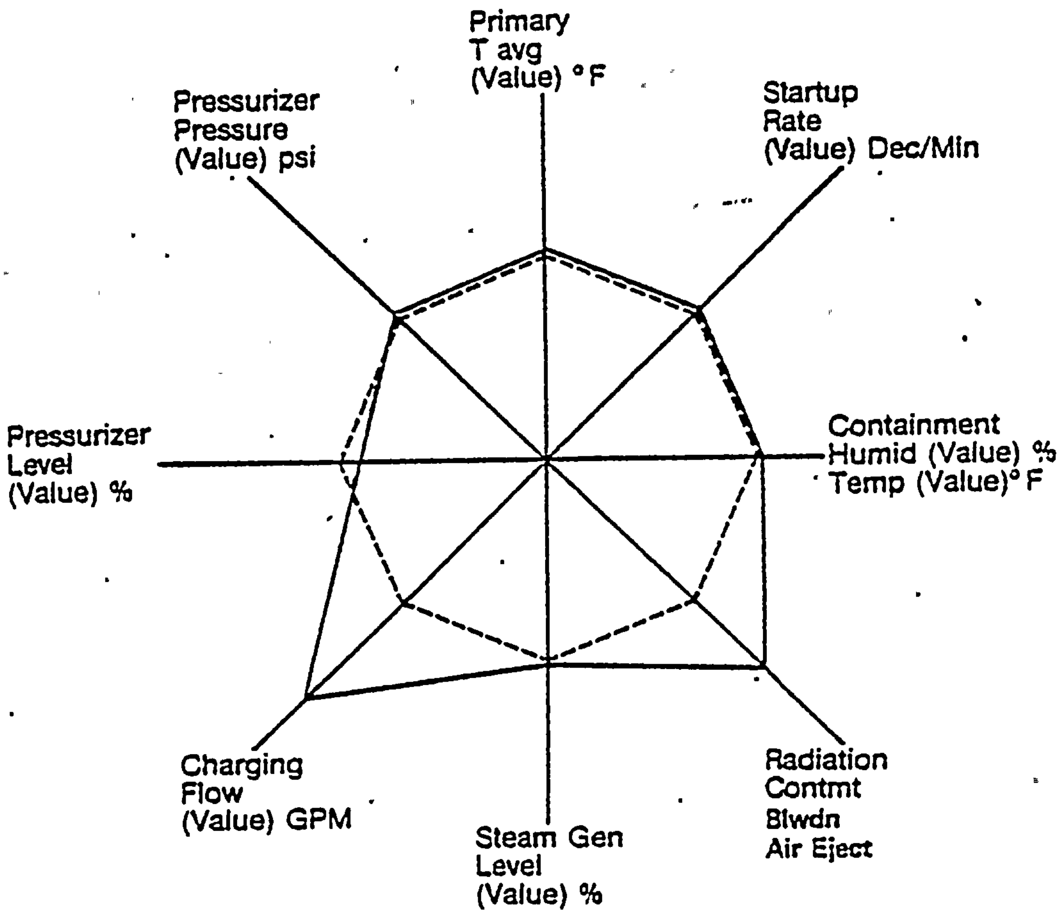
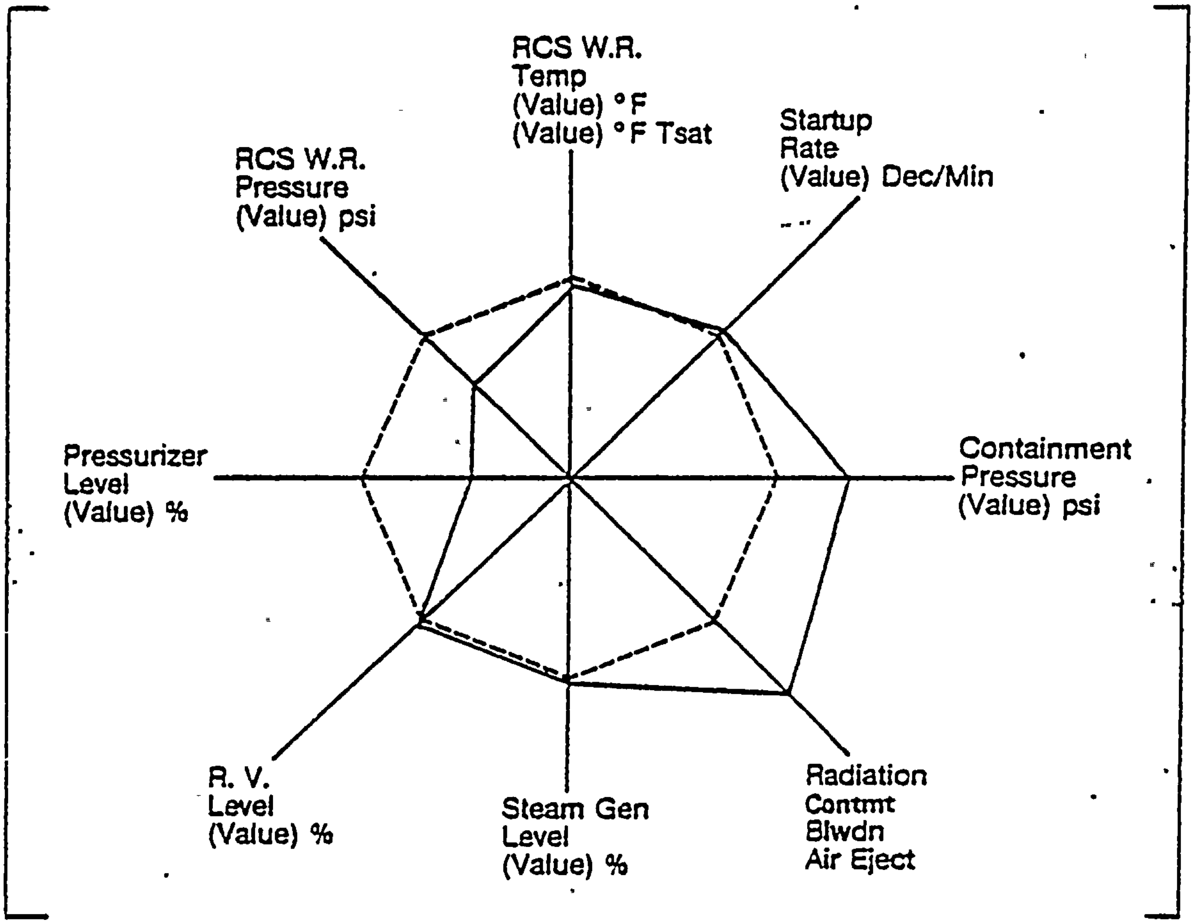


Figure 4-4. Sample Plant Safety Status Display — Terminate Mode — Primary to Secondary Coolant System Leak (SG Tube Leak)



(a,c,f)

Figure 4-5. Sample Plant Safety Status Display – Mitigate Mode – Primary Coolant System Leak to Containment

5.0 BYPASSED AND INOPERABLE STATUS INDICATION FOR PLANT SAFETY SYSTEMS

5.1 PURPOSE

The purpose of the Bypassed and Inoperable Status Indication (BISI) system is to provide the control room operator with a continuous systems level indication of a bypassed or inoperable condition for the systems comprising the engineered safety features. The system considers the actual status of individual components including systems level bypasses and control room operator entered inputs for components removed from service.

5.2 INPUT DETERMINATION

Bypassed and inoperable status indication is provided for the systems comprising the engineered safety features and their critical support systems. These systems are identified in Table 5.1. This table also identifies the types of components for which monitoring is required, the approximate number of each type of component, and the type of status information needed. This list is generic in nature and will be revised to meet individual plant specific designs.

In the evaluation of system inputs, the components in each system are considered in the light of being in a proper state to perform or support the operation of a safety function. The systems level bypass functions that must also be considered are listed in Table 5.2. In addition to automatically monitored inputs, the system also considers the effect of component or system out of service inputs manually entered by the control room operator.

5.3 MAN-MACHINE INTERFACE

The interface between the operator and this system is provided by redundant CRT displays and keyboard consoles located in the control room. Personnel located in the Onsite Technical Support Center will also be

able to access the same information. The BISI utilizes a structured display hierarchy for the operator interface. The display hierarchy is shown in Figure 3.1.

The primary display, an example of which is shown in Figure 3.2, contains the following information for each of the systems comprising the engineered safety features:

1. Bypassed or inoperable status indication for each affected subsystem on either a systems level and/or train level basis.

2. Identification of whether the condition is due to the inoperable status of a component or auxiliary support such as cooling water, power supply, etc.

(a,c,f) Other levels of displays such as shown in Figure 3.3 provide supporting information on individual components within each subsystem and support system. [An additional display provides a tabulation of all control room operator entered inputs for inoperable components for which automatic monitoring can not be accommodated or for which monitoring does not currently exist].

(a,c,f) Whenever the status of a system becomes inoperable or bypassed, the [control room operator will be alerted by an audible alarm and the primary display will indicate via video highlighting (e.g., flashing, color change, reverse video, etc.) the affected system and subsystem. The operator can then access supporting displays to determine the cause of the bypassed or inoperable condition. The control room operator must acknowledge the abnormal condition in order to silence the audible alarm. Reinstatement of normal system function will also generate a different audible signal.

Two additional capabilities of the BISI are the timing and test functions].

The timing function enables the control room operator to set up a countdown timing function for a system which is bypassed or inoperable. An audible alarm would be generated at the expiration of the operator specified time limit. This feature would aid the control room operator in complying with Technical Specification time limits for systems unavailable for service. (a,c

The test function enables the control room operator to test the effect on systems level status of a change in component status prior to changing the component's status. In response to the control room operator entered input, simulating the affect of changing a component's or system's status, the system determines the resultant effect on system operability and indicates the result to the control room operator.

TABLE 5.1

BYPASSED AND INOPERABLE STATUS INDICATION - COMPONENT INPUTS

(b,c)

System	Components	Status
Emergency core cooling	Valves	Open/Shut
	Pumps	Operable
	Process (level,pressure)	High/Low, etc.
Auxiliary feedwater	Valves	Open/Shut
	Pumps	Operable
	Process	High/Low, etc.
Containment spray	Valves	Open/Shut
	Pumps	Operable
	Process	High/Low, etc.
Containment isolation	Valves	Open/Shut
Auxiliary power system	Breakers	Open/Closed/Out
	Generators	Operable
	Voltages	High/Low
Containment ventilation	Valves	Open/Shut
	Motors	Operable
Containment hydrogen recombiners	Valves	Open/Shut
	Motors	Operable
Component cooling	Valves	Open/Shut
	Pumps	Operable
Service water	Valves	Open/Shut
	Pumps	Operable

TABLE 5.2

BYPASSED AND INOPERABLE STATUS INDICATION -
SYSTEM LEVEL BYPASS FUNCTIONS

Safety injection

- Low pressurizer pressure

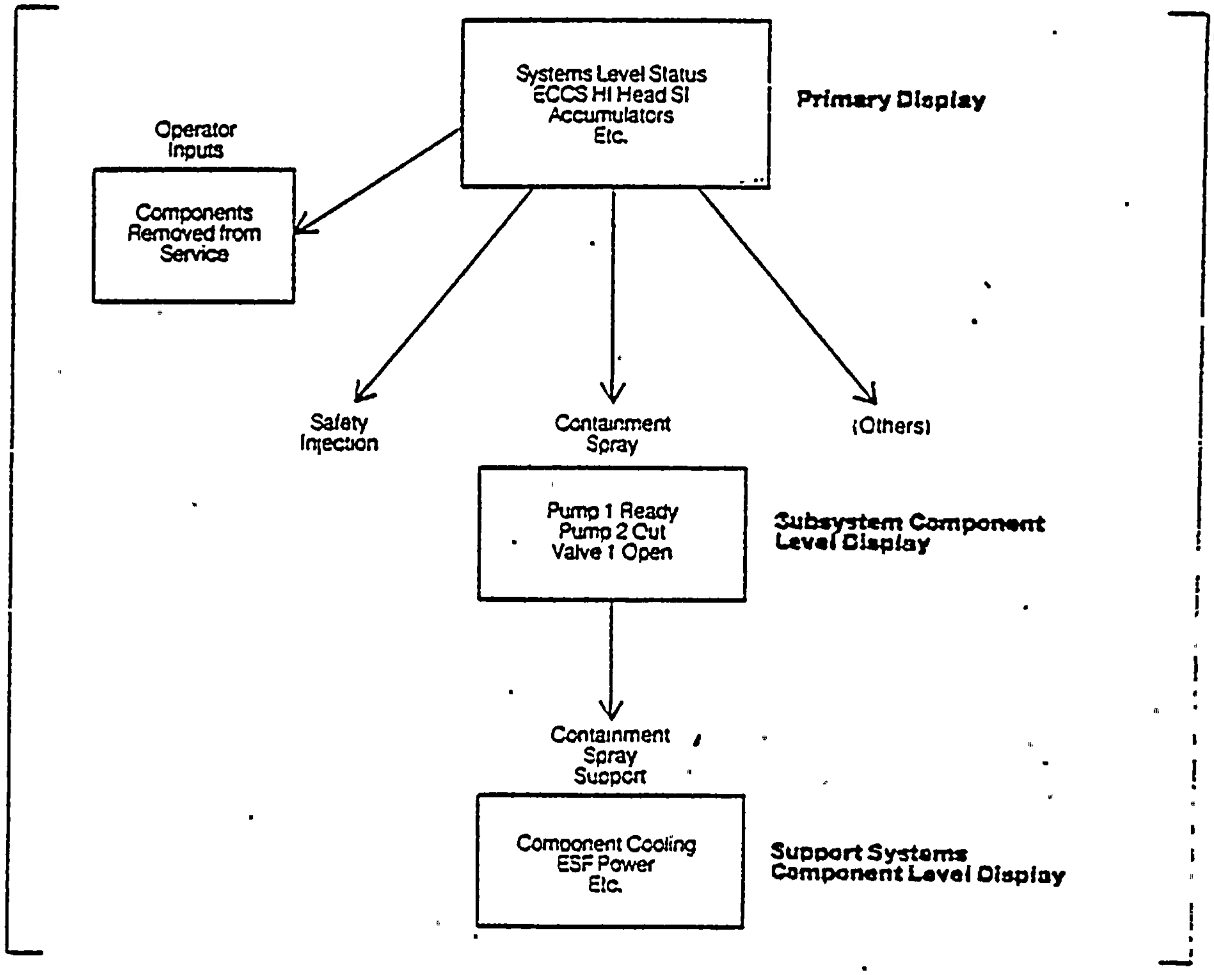
- Low steamline pressure

- Manual reset

Steamline isolation

Steam dump interlock

Steam generator blowdown isolation



(a,c,

Figure 5.1 Display Structure — Bypassed and Inoperable Status Indication

(a,c,f)

BYPASSED AND INOPERABLE STATUS DISPLAY	
SYSTEMS	STATUS
Emergency Core Cooling -	
High Head SI	Operable
Intermediate Head SI	Operable
Low Head SI	Operable
Accumulators	Operable
Auxiliary Feedwater	Operable
Containment Isolation	Operable
Containment Spray	Inoperable - Train A Component
Containment Ventilation	Operable
Safeguards Power Source	Operable

Figure 5.2 Primary Display – Bypassed and Inoperable Status Indication

(a,c,f)

CONTAINMENT SPRAY					
Train A		Train B		Train C	
VLV101 Pump A Suct	Open	VLV201 Pump B Suct	Open	VLV301 Pump C Suct	Open
VLV111 NAOH Supply	Open	VLV211 NAOH Supply	Open	VLV311 NAOH Supply	Open
Pump A	Operable	Pump B	Operable	Pump C	Operable
VLV102 Pump A Outlet	Closed	VLV202 Pump B Outlet	Open	VLV302 Pump C Outlet	Open
VLV103 Headr A Outlet	Closed	VLV203 Headr B Outlet	Closed	VLV303 Headr C Outlet	Closed
VLV121 Recirc A	Closed	VLV221 Recirc B	Closed	VLV321 Recirc C	Closed
Refueling Water Storage Tank					
LS100 Level	Normal				
LS101 Level	Normal				
LS102 Level	Normal				
LS103 Level	Normal				
NAOH Spray Additive					
LS200 Level	Normal	TS200 Temp	Normal		
LS201 Level	Normal	TS201 Temp	Normal		
LS202 Level	Normal	TS202 Temp	Normal		

Figure 5.3 Secondary Display — Bypassed and Inoperable Status Information

6. TSC INSTRUMENTATION

As described in Section 2, most of the input signals to the TSC computer are taken from the existing instruments which also provide signals for the Control Room indicators. This approach will provide consistent data in both the control room, Onsite Technical Support Center and the EOF. The input signals to the TSC computer therefore have the same high quality, accuracy and reliability as the control room signal. Inputs to the TSC computer provide transformer isolation for all analog input signals and all digital input signals are optically isolated. In addition, all signals from the Reactor Protection Channels are taken after the existing safety grade isolators. The interfacing of the TSC Computer to the existing plant instrumentation was designed so as not to result in any degradation of the control room, protection system, controls or other plant functions. Any degredation that is noted during checkout and integrated systems testing will be corrected.

7. TSC POWER SUPPLY SYSTEMS

7.1 POWER TO THE TSC COMPUTER SYSTEM:

The power requirements of the TSC Computer System will be satisfied though the use of an uninterruptible power supply system (UPS). This UPS system will provide the TSC computers and peripheral equipment with a high quality, transient free power source.

7.1.1 THE UPS SYSTEM:

Figure 7.1 shows a one-line diagram (schematic) for the UPS system. The system consists of redundant battery chargers, battery, static inverters, and static transfer switches. Under normal conditions, the battery charger converts AC to DC and supplies it to the inverter. The battery charger also keeps the battery at full charge. The inverter converts the DC to AC in order to supply the load requirements of the TSC computers and their peripheral equipment.

7.1.2 CONSEQUENCES OF POWER SUPPLY INTERRUPTION:

If there is a power reduction (dip or degradation) or loss (failure) of the AC power source, the UPS battery becomes the primary source of DC to the inverter, rather than the battery charger which has lost its normal source of AC power supply. The battery will be sized to supply the inverter load requirement for a period of 30 minutes. This allows a sufficient time interval in which a diesel generator (backup AC source) can be made available to provide power to the inverter. In the unlikely event of loss or

TSC POWER SUPPLY SYSTEM
(CONCEPTUAL DESIGN)

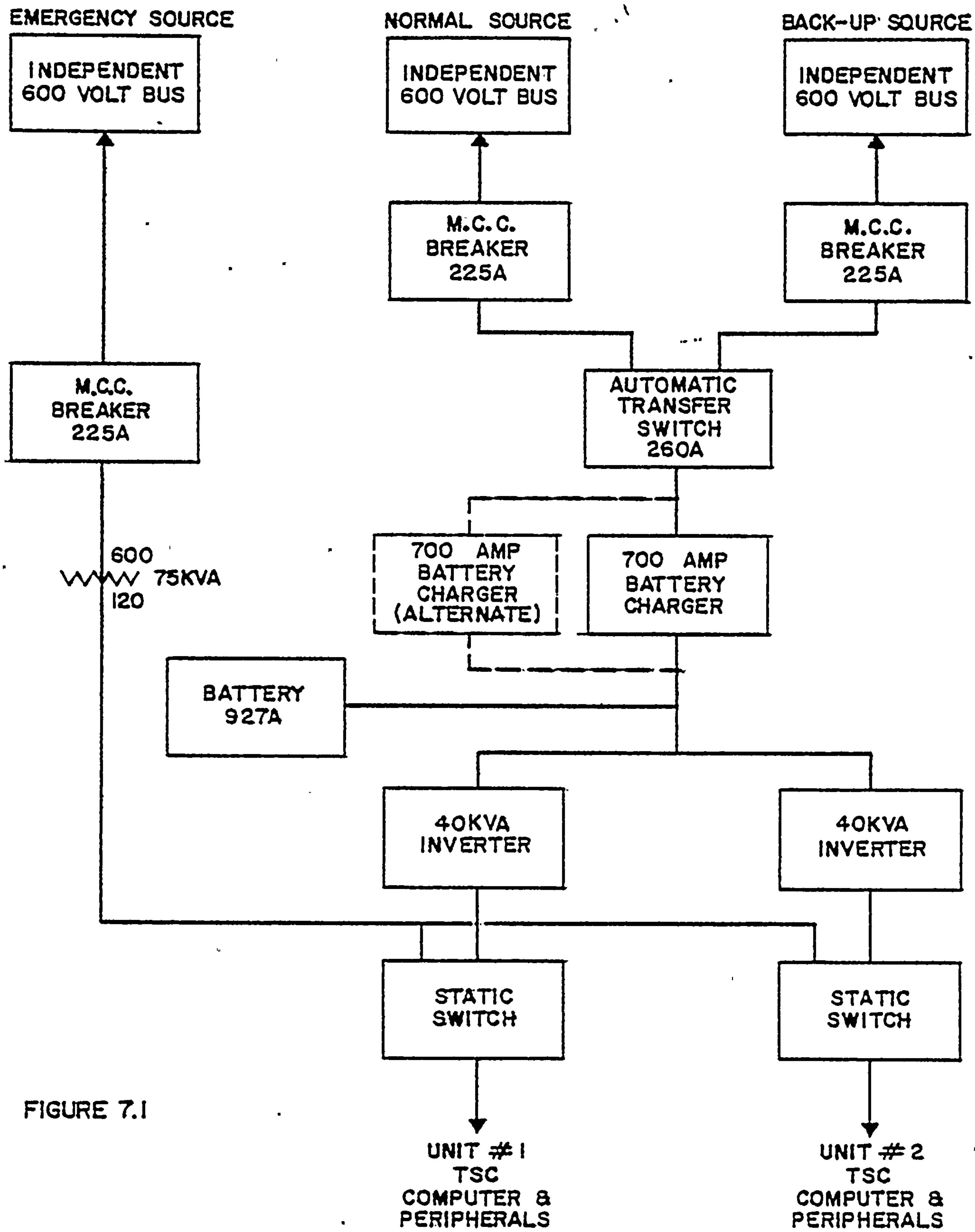


FIGURE 7.1

6/1/81

unavailability of both the normal and backup AC sources, the static switch will be used for transfer, if necessary, to the emergency AC source.

7.2 POWER TO THE TSC COMPLEX:

Standard balance-of-plant (BOP) sources will provide the TSC with power for lighting and convenience receptacles. For additional protection, the lighting fixtures are provided with battery packs for continued operation in the event of loss of the BOP power supply. The HVAC equipment will be supplied from an Essential Services System bus (AC source).

Section 8.0

Original pages AEP-58 through AEP-62 have been deleted from this submittal. The descriptive information that was contained therein can be found in the DCCNP Emergency Plan.

Section 9.0

Original pages AEP-63 through AEP-65 have been deleted from this submittal. Listings of plant records, plant specific reference material, general technical reference material, plant procedures and reports that are available to personnel working in the TSC are provided in general company internal documents which pertain to the subject matter.

Attachment 1 to AEP:NRC:0916I

REASONS AND 10 CFR 50.92 ANALYSES

FOR CHANGES TO THE

DONALD C. COOK NUCLEAR PLANT

UNIT 2 TECHNICAL SPECIFICATIONS

The Technical Specification (T/S) changes included in this letter are, in general, those necessary to support the safety analyses performed by Exxon Nuclear Company (ENC) for the Unit 2 Cycle 6 reload. In addition to these changes, however, we have included additional changes which are intended to make the T/Ss clearer, easier to use, or more consistent with the Standard Technical Specifications (STSs) for Westinghouse Pressurized Water Reactors, NUREG-0452, Rev. 4 (or Draft Rev. 5, where applicable).

A summary of the changes has been included as Attachment 10 to this letter. It includes a brief description of each change, as well as the reason for the change, and, where applicable, references to the safety analyses the change is based on. This attachment includes an overview of the changes, as well as our 10 CFR 50.92 justifications for no significant hazards consideration. Please note that the changes will be referred to by their numbers, which are given in the "Description of Change" column in Attachment 10.

We have grouped the changes into 12 separate types for ease of discussion. These changes are discussed below.

1. Editorial Changes

The first group of changes to be discussed consists of those that are purely editorial in nature. These changes are numbered 1, 2, 5, 6, 12, 20, 21, 24, 35, 36, 45, 50, 60, 62, 69, 74, 81, 83, 84, 88, 90, 93, 94, 97, 98 and 105 in Attachment 10. These changes are proposed to enhance the readability of the T/Ss, to achieve consistency between the Unit 1 and 2 T/Ss, or to achieve consistency with the STSs, as described in Attachment 10.

Per 10 CFR 50.92, a proposed amendment will involve a no significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated,
(2) create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or
(3) involve a significant reduction in a margin of safety.

Criterion 1

These changes, being editorial in nature and intended to improve the readability of the T/Ss, will not reduce in any way requirements or commitments in the existing T/Ss. Thus, no increase in the probability or consequences of a previously evaluated accident would be expected.

Criterion 2

These purely editorial changes will not create the possibility of a new or different kind of accident from any previously evaluated, because all accident analyses and nuclear design bases remain unchanged.

Criterion 3

The proposed amendment will not involve a significant reduction in margin of safety, because, as discussed above, all accident analyses and nuclear design bases remain unchanged.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The first of these examples refers to changes that are purely administrative in nature: for example, changes to achieve consistency throughout the T/Ss, correction of an error, or a change in nomenclature. This group of proposed changes is intended to achieve consistency between the Unit 1 and 2 T/Ss, to achieve greater consistency with the STS format, or to improve the overall readability of the T/S document. As these changes are purely editorial and do not impact safety in any way, we believe the Federal Register example cited is applicable and that the changes involve no significant hazards consideration.

2. Removal of 3-Loop Technical Specifications

A second category of changes involves removal of Technical Specification provisions for 3 reactor coolant loop operation in Operational Modes 1 and 2. These are changes numbered 3, 7, 16, 29, 30, 31, 46, 56, 59, 61, 67, 91, 99, and 100 in Attachment 10. This category includes all changes involving removal of 3-loop provisions except for those associated with Functional Unit 1.e. (Differential Pressure Between Steam Lines-High) on Engineered Safety Features (ESF) Actuation Instrumentation Table 3.3-3. Three-loop changes associated with this ESF signal are discussed in Category 5 of this Attachment.

License Condition 2.C.3(j) for Unit 2 prohibits operation with less than 4 pumps at power levels above the P-7 permissive (approximately 11% of rated thermal power). As a matter of practice, we have extended this restriction to cover all of Modes 1 and 2. As T/Ss covering 3-loop operation in Modes 1 and 2 are therefore not necessary, we propose to remove them to streamline the document.

Included in this group of changes is the deletion of T/S 3/4.4.1.4. Although this specification contains provisions for less than 4-loop operation in modes other than 1 and 2, the requirements for other modes which remain applicable are addressed identically in other T/Ss, as specified below:

Action Statement (Below P-7)	Where Addressed
a	T/S 3.4.1.1
b	T/Ss 3.4.1.2 and 3.4.1.3
c	Not needed, since 3-loop operation in Modes 1 and 2 will be prohibited.

Per 10 CFR 50.92, a proposed amendment will involve a no significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated,
(2) create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or
(3) involve a significant reduction in a margin of safety.

Criterion 1

This group of changes will extend the license condition prohibiting 3-loop operation above the P-7 permissive to include all of Modes 1 and 2. Thus, the changes would be expected, as a minimum, to reduce the probability, or consequences of a previously evaluated accident.

Criterion 2

Since these changes place additional restrictions on plant operation, they would not be expected to create the possibility of a new or different kind of accident from any previously analyzed or evaluated.

Criterion 3

Since 3-loop operation in all of Modes 1 and 2 will be prohibited, additional margin to DNB under accident conditions should result. Thus, margin of safety should be increased rather than decreased.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The second of these examples refers to changes that impose additional limitations, restrictions, or controls not presently included in the T/Ss. Since prohibition of 3-loop operation in Modes 1 and 2 constitutes a restriction which the current T/Ss do not have, we believe this example is applicable and that the changes involve no significant hazards consideration.

3. Additional Restrictions Because of Safety Analyses

A third group of changes involves inclusion of proposed new requirements in the T/Ss. The new requirements are proposed to make the T/Ss consistent with the safety analyses performed by ENC in support of the Cycle 6 reload, or to achieve consistency with the STS. These changes are numbered 9, 22, 51, 52, 55, 63, 64, 70, 72, 73, 80, 82, 86, 92, and 102 in Attachment 10. The applicable references to the safety analyses are included there also.

Per 10 CFR 50.92, a proposed amendment will involve a no significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated,

(2) create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or
(3) involve a significant reduction in a margin of safety.

Criterion 1

These changes constitute additional restrictions on the plant in terms of T/S mode applicability, surveillance requirements, or Action Statement requirements. Since none of these changes reduce in any way previous safety requirements, they would not be expected to result in an increase in the probability or consequences of an accident previously evaluated.

Criterion 2

These changes will place additional restrictions on plant operation and will increase, rather than reduce, requirements for safety. Therefore, they should not create the possibility of a new or different kind of accident from any previously analyzed or evaluated.

Criterion 3

These changes add additional safety requirements, and in no way reduce any existing requirements. Thus, no reduction in margin of safety will occur because of these changes.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The second of these examples refers to changes that impose additional limitations, restrictions, or controls not presently included in the T/Ss. These changes impose additional restrictions on the plant for consistency with the Cycle 6 safety analyses or the STSs. Thus, we believe that this example is applicable and that the changes involve no significant hazards consideration.

4. Refueling Water Storage Tank Changes

A fourth group of changes involves T/Ss 3.1.1.3, 3.1.2.3, 3.1.2.5, 3.4.1.2, 3.4.1.3, and 3.9.8.1 specifically as they apply to borated water addition or positive reactivity addition from the Refueling Water Storage Tank (RWST). These are changes numbered 25, 26, 27, 87, 89, and 104 in Attachment 10.

T/S 3.1.1.3 requires reactor coolant flow of at least 3000 gpm during dilution of the Reactor Coolant System (RCS) boron concentration in any mode. T/Ss 3.4.1.2 and 3.4.1.3 require at least one coolant loop to be in operation during boron dilution in Modes 3, 4, and 5. T/S 3.9.8.1 requires 3000 gpm of coolant flow via the Residual Heat Removal System during boron dilution in Mode 6. T/Ss 3.1.2.3 and 3.1.2.5 prohibit positive reactivity addition in Modes 5 and 6 with charging pumps or boric acid transfer pumps inoperable, respectively. Because of concerns with literal T/S compliance, questions have arisen as to the applicability of these specifications during the times when we add water to the RCS from an operable RWST, specifically when the boron concentration of the RWST is lower than the RCS.

The RWST minimum boron concentrations stated in the T/Ss were established to ensure that adequate shutdown margin is maintained, and are consistent with numbers assumed by ENC in their Cycle 6 reload analyses. Because of this, it is our belief that the boron dilution restrictions of the T/Ss listed above were not meant to be applicable during water addition from the RWST, provided the boron concentration in the RWST exceeds the minimum requirements stated in the T/Ss. We have documented this interpretation in the past (see our letter AEP:NRC:0975A, dated February 28, 1986); this change is submitted only to formalize this interpretation.

Per 10 CFR 50.92, a proposed amendment will involve a no significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated,
(2) create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or
(3) involve a significant reduction in a margin of safety.

Criterion 1

Our review has determined that the T/S RWST minimum boron concentrations are sufficient to ensure that adequate shutdown margin is maintained throughout the entire core life. Additionally, the RWST boron concentrations are consistent with those assumed in the LOCA analyses performed by ENC. Thus, we conclude that these changes will not significantly increase the probability or consequences of an accident previously evaluated.

Criterion 2

The proposed amendment will not create the possibility of a new or different kind of accident from any previously evaluated. It has been determined that the RWST boron concentration is sufficient to ensure adequate shutdown margin from all expected operating conditions. The consequences of adding water from an operable RWST which is at a lower boron concentration than the RCS is therefore bounded, and no new or different kind of accident from those previously evaluated would be expected.

Criterion 3

Because these changes lessen operating restrictions, it can be expected that a reduction in safety margin may occur. However, because the RWST minimum boron concentrations are sufficient to provide adequate shutdown margin from all expected operating conditions, this reduction in safety margin would be insignificant.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The sixth of these examples refers to changes which may result in some increase to the probability of occurrence or consequences of a previously analyzed accident, but where the results are

clearly within limits established as acceptable. As discussed above, these changes relax requirements related to boron dilution or positive reactivity addition, but are clearly bounded by our shutdown margin analyses. Thus, we conclude that the example cited is applicable and that the changes involve no significant hazards considerations.

5. Changes to the Differential Pressure Between Steam Lines-High ESF Actuation Signal

The fifth group of proposed changes involve Functional Unit 1.e (Differential Pressure Between Steam Lines-High) under the Engineering Safety Feature (ESF) Actuation System Instrumentation Table 3.3-3. These changes are numbered 67, 68, and 71 in Attachment 10. Specifically, we are proposing to change the footnote designator for the Channels to Trip column of the 3-loop section to a quadruple pound sign, and to add a corresponding new footnote to the Table 3.3-3 notations on T/S page 3/4 3-21. Additionally, we propose to revise the functional unit to prohibit 3-loop operation in Modes 1 and 2, consistent with Category 2 of this attachment.

The Differential Pressure Between Steam Lines-High actuation differs from other ESF actuation signals in that a signal from one loop is compared to signals in the other loops. The current footnote associated with this signal for the 3-loop case states: "The channels associated with the protective functions derived from the out of service Reactor Coolant Loop shall be placed in the tripped mode." This could be construed to mean that all channels in the out of service loop should be tripped. This in turn would result in an ESF actuation. It is our belief that the footnote as applied to this functional unit means to trip the bistables which indicate low active loop steam pressure relative to the idle loop. This action reduces the ESF actuation logic for the active loop differential pressures from 2 out of 3 to 1 out of 2, and thus permits 3-loop operation in Mode 3 since 2 channels per steam line are necessary for a trip.

Per 10 CFR 50.92, a proposed amendment will involve a no significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated,
(2) create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or
(3) involve a significant reduction in a margin of safety.

The prohibition of 3-loop operation in Modes 1 and 2 is consistent with the changes included in Category 2 of this attachment. The 10 CFR 50.92 analysis is thus identical and will not be repeated here. The 10 CFR 50.92 analyses included in this category are therefore only those involved in rewriting the Differential Pressure Between Steam Lines-High footnote in T/S Table 3.3-3.

Criterion 1

The changes included in this group are editorial in nature, intended only to clarify the ESF Actuation System Instrumentation Table (3.3-3) as it

applies to the Differential Pressure Between Steam Lines-High actuation signal. Thus, no significant increase in the probability or consequences of a previously evaluated accident should occur.

Criterion 2

The proposed amendment will not create the possibility of a new or different kind of accident from any previously evaluated because these changes, being editorial in nature, will not impact existing safety analyses or the nuclear design bases.

Criterion 3

The proposed amendment will not involve a significant reduction in margin of safety because, as discussed above, all accident analyses and nuclear design bases remain unchanged as a result of these proposed T/S changes.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The first of these examples refers to changes that are purely administrative in nature: for example, changes to achieve consistency throughout the T/Ss, correction of an error, or a change in nomenclature. This group of proposed changes is intended only to clarify the T/Ss, to avoid the possibility that they may be misread. As these changes are editorial and do not impact safety in any way, we believe that the Federal Register example cited is applicable and that the changes involve no significant hazards consideration.

6. Changes to the Power-Operated Relief Valve (PORV) Specification, 3/4.11.4

The sixth group of proposed changes involve a redraft of T/S 3/4.11.4, concerning the Pressurizer Power-Operated Relief Valves (PORVs). These changes are number 95 in Attachment 10. Specifically, we are proposing to change T/S 3/4.11.4 to require that at least 2 PORVs be available in Modes 1, 2, and 3. For purposes of this specification, "available" means that the PORV is operable with its solenoid deenergized and that the block valve is operable and energized. This differs from the present T/S, which allows all 3 PORVs to be inoperable, provided their associated block valves are closed. The proposed changes are intended to ensure that PORV relief capability is available to assist in RCS depressurization following a steam generator tube rupture without offsite power, and to respond to comments made by members of your staff at a meeting held with us in Bethesda, MD on December 13, 1984.

Per 10 CFR 50.92, a proposed amendment will involve a no significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated,
(2) create the possibility of new or different kind of accident from any accident previously analyzed or evaluated, or
(3) involve a significant reduction in a margin of safety.

Criterion 1

This group of changes constitutes additional restrictions placed on PORV (and associated block valve) operability requirements. Since no restrictions associated with the PORVs are reduced in any way by this group of changes, we conclude that these changes will not increase the probability or consequences of a previously analyzed accident.

Criterion 2

Since these changes place additional restrictions on plant operation and in no way reduce present safety restrictions, they would not be expected to create the possibility of a new or different kind of accident from any previously analyzed or evaluated.

Criterion 3

These changes add additional restrictions on the PORVs, designed primarily to ensure that PORV relief valve capability is available to assist in RCS depressurization following a steam generator tube rupture. Thus, these changes would be expected to increase, rather than decrease, safety margins.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The second of these examples refers to changes that impose additional limitations, restrictions, or controls not presently included in the T/Ss. Since this group of changes will require PORVs to be operable in Modes 1 through 3 (where previously no operability requirement existed), they clearly constitute additional restrictions. Thus, we conclude that the example cited is applicable and that no significant hazards are involved.

7. Addition of T/S 4.0.4 Exemptions

The seventh group of proposed changes are those which add T/S 4.0.4 exemptions to existing T/Ss. These changes are numbered 44, 65, 66, and 103 in Attachment 10. For the first of these changes, a T/S 4.0.4 exemption has been proposed for the flow measurement performed after each refueling and for all flow surveillances for the DNB T/S, 4.2.5.1 (see numbers 44 in Attachment 10). (The flow specification has been moved from the $F_{\Delta H}$ specification (3/4.2.3) to the DNB specification (3/4.2.5.1) for consistency with Unit 1 specifications.) This exemption is required because flow is measured using secondary calorimetric and primary temperature measurements, which can only be performed at or near full power. The flow instrumentation is calibrated based on this measurement.

Exemptions have also been provided for several Nuclear Instrumentation System (NIS) calibrations (see numbers 65 and 66 in Attachment 10) in T/S Table 4.3-1. Of these, those proposed for source range and intermediate range detector calibrations appear in STS, Rev. 4. STS, Rev. 4 also provides this exemption for the incore detector, excore power range

detector cross-calibration performed after refueling. Our proposal extends this exemption to the quarterly incore detector, excore power range detector cross-calibration in order to address the situation where an unscheduled outage of significant duration causes the surveillance interval for this calibration to lapse. This exemption is proposed for the daily power range, neutron flux heat balance because it is required to be performed above 15% rated thermal power by T/S. It is also proposed for the monthly incore-excore axial offset comparison for the same reason. These exemptions are needed to address unscheduled outages for which the surveillance interval has lapsed. An exemption from T/S 4.0.4 for the source range channel functional test is proposed. This exemption addresses the situation that results from a reactor trip after continuous power operation of more than 1.25 times 31 days. This surveillance cannot be performed at power without damaging the source range detectors.

Exemptions from T/S 4.0.4 are proposed for the single-loop and two-loop loss-of-flow trip calibrations of T/S Table 4.3-1. These are required because these calibrations are based on the primary flow measurement taken at or near full power which was discussed above in relation to flow instrumentation. These changes are numbered 65 and 66 in Attachment 10.

Exemptions from T/S 4.0.4 are proposed for the f(Δ I) penalties associated with the Overpower Δ T and Overtemperature Δ T trips. These exemptions are required because the f(Δ I) module is calibrated to data obtained from the incore detector, excore power range detector cross-calibration. As is implied by the exemption of this calibration from T/S 4.0.4 on a refueling frequency, which is already available in STS, Rev. 4, this calibration must be performed at power, in the applicable mode. The calibration is performed at power so that an appreciable signal can be obtained on the incore detectors and the excore detectors. These changes are numbered 65 and 66 in Attachment 10.

Lastly, an exemption from T/S 4.0.4 is proposed for Surveillance 4.7.1.5 (see number 103 in Attachment 10.) This exemption is required because T/S 3.7.1.5, Steam Generator Stop Valves, is applicable to Mode 3, and Surveillance 4.7.1.5, which measures stop valve closure time, must be performed in Mode 3. In order to demonstrate the required closure time for the steam generator stop valves, steam pressure must be in the normal operating range corresponding to primary temperature above the P-12 setpoint. Therefore, secondary pressure for this test must be above approximately 800 psig for which saturation temperature is well above the 350^{o}F Mode 3 boundary. An exemption is also proposed for Beginning of Cycle to enter Mode 2 for physics testing provided the steam generator stop valves are closed. This provision allows continuation of the startup program with steam generators isolated in the event that secondary side work is not complete.

Per 10 CFR 50.92, a proposed amendment will involve a no significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated, .

(2) create the possibility of new or different kind of accident from any accident previously analyzed or evaluated, or
(3) involve a significant reduction in a margin of safety.

Criterion 1

The changes in this section are necessary to make the T/Ss accurately reflect limitations associated with surveillances which must be performed in the applicable mode. Additionally, the changes are needed to address the fact that unscheduled outages can and do occur, and when they do surveillances can expire with no way to correct the situation until the unit returns to power. Where possible we have followed the guidance given by the STSs, expanding it as necessary to address the situations just described. As these changes are consistent with the guidance provided by the STSs, we believe that any increase in the probability of occurrence or consequences of an accident previously analyzed, or any reduction in margins of safety, would be insignificant.

Criterion 2

Since these changes require neither physical changes to the plant nor changes to the safety analyses, it is concluded that they will not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3

Please see our discussion on Criterion 1, above.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. Example 6 refers to changes which may result in some increase to the probability or consequences of a previously analyzed accident, but where the results of the change are clearly within acceptable limits. It is our belief that these changes are necessary to reflect limitations inherent in surveillance testing methods employed by the Cook Plant, and the changes reflect further clarification of the intent of the original T/S as is indicated by the type of T/S in these areas that is permitted by later revisions of the STS. In light of this, we believe the reasons for this group of changes to be consistent with Example 6.

8. Changes to Existing T/S Values

The eighth group of proposed changes involve values of parameters presently included in the T/Ss that are being revised to reflect the assumptions used in the various safety analyses performed in support of the Unit 2 Cycle 6 reload. These changes are numbered 4, 8, 10, 11, 13, 14, 15, 17, 18, 19, 23, 28, 34, 40, 42, 47, 48, 49, 54, 76, 78, 79, and 101 in Attachment 10. That attachment also includes references to the specific sections of the accident analyses on which the changes are based.

Two types of changes included in this group need further explanation. The first are changes to allowances to permit operation with RdF RTDs. These are included in the changes numbered 8, 10, 14, 19, 42, 47, 48, 76, and 78 in Attachment 10. During the Unit 2 Cycle 6 refueling outage, we will be replacing all of our existing Rosemount RTDs with RTDs manufactured by the RdF Corporation. Because the uncertainties associated with these new RTDs are different from those associated with the older Rosemount RTDs, it is necessary to revise some T/S values accordingly. We used the revised uncertainties to obtain Technical Specification setpoints from the analysis values calculated by Exxon Nuclear Company. Certain setpoints were affected by both a change in analysis value and the revised allowances. For your convenience, we have included the Westinghouse Electric Corporation safety evaluation for the RdF RTD installation (WCAP-11080) as Attachment 3 to this letter.

The second group of changes needing clarification are changes involved with the $f(\Delta I)$ penalty which is applied to the Overtemperature ΔT and Overpower ΔT reactor trip setpoints. (These are changes numbered 15 and 18 in Attachment 10.) There is only one $f(\Delta I)$ module, which serves both of these trips. This module places a penalty on these trip functions in the event of an axial imbalance in neutron flux between the top and bottom halves of the core. The $f(\Delta I)$ penalty was not required as an input to the Overpower ΔT trip for previous Unit 2 cycles, and thus $f_2(\Delta I)$ is presently set equal to zero in T/S Table 2.2-1. The new analyses performed by ENC apply the $f(\Delta I)$ penalty to both Overpower and Overtemperature ΔT. The ENC analyses resulted in different $f(\Delta I)$ functions for these two trips. However, because they share the same $f(\Delta I)$ module, a single $f(\Delta I)$ function that conservatively bounds these two functions was chosen for the proposed T/Ss.

Per 10 CFR 50.92, a proposed amendment will involve a no significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated,
(2) create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or
(3) involve a significant reduction in a margin of safety.

The changes included in this group are necessary to support safety analyses performed by ENC and Westinghouse Electric Corporation (as referenced by Attachment 10) in support of the Cycle 6 reload. These analyses have not yet been accepted by the Commission. Our conclusion of no significant hazards considerations, which is supported below, is therefore contingent upon Commission acceptance.

Criterion 1

The safety analyses performed for Cycle 6 addressed all previously analyzed accidents. The analyses, which are referenced in Attachment 10, demonstrated that no significant increase in the probability or consequences of a previously evaluated accident is expected to occur.

Criterion 2

The safety analyses performed for Cycle 6 addressed all applicable accidents found in the Standard Review Plan for relevancy to Cook. Many of those addressed had not previously been evaluated for D. C. Cook Unit 2. Therefore, we conclude that, to the best of our knowledge, this group of changes will not create the possibility of a new or different kind of accident from any accident previously analyzed.

Criterion 3

The safety analyses performed for Cycle 6 (as referenced by Attachment 10) have demonstrated that acceptable margins of safety are maintained for all accidents which were addressed.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The second of these examples refers to changes resulting from a nuclear reactor core reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. These changes are similar to this example in that the Cycle 6 reload is very similar to previous reloads in terms of enrichment, power distribution, and fuel type. Although minor changes have occurred (e.g., F_Q was increased from 2.04 to 2.10), the changes were analyzed and found not to significantly impact applicable margins to safety. Thus, we conclude that the example cited is relevant and that no significant hazards consideration is involved.

9. Separation of Flow Rate and $F^N_{\Delta H}$

The ninth group of changes involve revisions to T/S 3/4.2.3, Nuclear Enthalpy Hot Channel Factor ($F^N_{\Delta H}$). These changes are numbered 41, 42, 43, 48 in Attachment 10. In the present T/Ss, RCS flow rate and $F^N_{\Delta H}$ may be "traded off" against one another (i.e., a lower measured RCS flow rate is acceptable provided $F^N_{\Delta H}$ is also acceptably lower). In the proposed T/S 3/4.2.3, we have eliminated the ability to trade off flow for $F^N_{\Delta H}$. $F^N_{\Delta H}$ is now defined in T/S 3.2.3 only as a function of rated thermal power. RCS flow rate in Mode 1 has been moved to proposed T/S 3/4.2.5.1, which contains the Mode 1 DNB parameters. Although the Action Statements and surveillance requirements have been revised to reflect this separation, no requirement appropriate for either of the two has been deleted or made less severe. No flux mapping is required in the DNB Action statement, because flux mapping is used to measure $F^N_{\Delta H}$, not flow.

The proposed changes included in this group are only those changes involved in separating flow rate and $F^N_{\Delta H}$ in the T/S. Changes to existing T/S values for flow are included in Category 8 of this attachment.

Per 10 CFR 50.92, a proposed amendment will involve a no significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated,

(2) create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or
(3) involve a significant reduction in a margin of safety.

Criterion 1

This group of proposed changes in no way removes or reduces any safety requirements, nor does it require physical changes to the plant. Thus, it is not expected to involve a significant increase in the probability or consequences of a previously evaluated accident.

Criterion 2

These proposed changes will not create the possibility of a new or different kind of accident from any previously analyzed, because, being primarily editorial in nature, they impact neither the accident analyses nor the nuclear design bases.

Criterion 3

The proposed changes will not involve a significant reduction in margin of safety, because, as discussed above, all accident analyses and nuclear design bases remain unchanged. Since these changes actually represent additional restrictions (in that we will no longer be able to trade off RCS flow rate for $F^N_{\Delta H}$) it could be anticipated that an increase, rather than decrease, in the margin to DNB under accident conditions might actually result.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The first example refers to purely administrative changes to the T/S: for example, changes to achieve consistency throughout the T/Ss, correction of an error, or a change in nomenclature. These changes are similar to this example in that RCS flow rate and $F^N_{\Delta H}$ are being separated with no reduction in requirements, primarily to make the Unit 2 T/Ss more similar to those for Unit 1.

The second example published in the Federal Register refers to changes that constitute additional limitations, restrictions, or controls not presently included in the T/Ss: for example, more stringent surveillance requirements. These changes are similar to this example in that we will be prohibiting ourselves from trading off RCS flow rate for $F^N_{\Delta H}$.

For the reasons provided above, we conclude that the examples cited are relevant and that this group of proposed changes involves no significant hazards consideration.

10. Changes to the P-12 Interlock Description

The tenth group of proposed changes involves the P-12 Interlock description included in T/S Table 3.3-3. These changes are numbered 75 and 77 in Attachment 10. The P-12 Interlock receives input from the T_{ave} low-low bistables. These bistables are calibrated to trip when the temperature decreases to 541°F as specified in T/S Table 3.3-4.

With 2 out of 4 bistables tripped, P-12 permits the manual block of the Low Steam Line Pressure Safety Injection, causes steam line isolation under conditions of high steam flow, and removes the arming signal to condenser steam dump. With 3 of 4 T_{ave} channels above the reset point, which is greater than 541°F, the manual block of Low Steamline Pressure Safety Injection is defeated or prevented and the condenser steam dump is enabled.

The present T/S description of the P-12 Interlock is confusing in that it neglects the trip and reset points, and instead describes P-12 in terms of conditions above 544°F and below 540°F. If this description is read literally, it could be inferred that P-12 is established when T_{ave} is greater than or equal to 544°F and when T_{ave} is less than 540°F. Additionally, the manual block of safety injection actuation would not be permitted until below 540°F, when in fact the setpoint is 541°F. We propose to rewrite P-12 in terms of the 541°F setpoint, which is similar to the methodology utilized in Rev. 4 of the STS, in order to better reflect the functioning of this interlock.

In addition to the changes described above, we have revised the P-12 function description. The current description states that the Safety Injection associated with P-12 occurs on high steam line flow and low steam line pressure. The D. C. Cook Unit 2 ESF design provides a Safety Injection on Low Steam Line pressure which does not require a coincident signal from P-12 Low Low T_{ave}. This particular Safety Injection may be blocked if the P-12 Low Low T_{ave} signal is present. High steam line flow coincident with P-12 Low Low T_{ave} does not provide a Safety Injection; it does however cause a steamline isolation.

Per 10 CFR 50.92, a proposed amendment will involve a no significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated,
(2) create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or
(3) involve a significant reduction in a margin of safety.

Criterion 1

These changes, being editorial in nature and intended only to more accurately describe the functioning of the P-12 interlock, will not reduce in any way requirements or commitments which are presently included in the T/Ss. Thus, no increase in the probability or consequences of a previously evaluated accident would be expected.

Criterion 2

These changes, being purely editorial, will not create the possibility of a new or different kind of accident from any previously evaluated because all accident analyses and nuclear design bases remain unchanged.

Criterion 3

The proposed amendment will not involve a significant reduction in margin of safety, because, as discussed above, all accident analyses and nuclear design bases remain unchanged.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The first of these examples refers to changes which are purely administrative in nature: for example, a change to achieve consistency throughout the T/Ss, correction of an error, or a change in nomenclature. This group of proposed changes is similar to this example in that the changes are purely editorial, intended to make the T/Ss more accurately reflect the functioning of the P-12 interlock. No physical changes to the plant or its procedures will be necessary because of these changes. Thus, we conclude that the example cited is applicable and that this group of changes involve no significant hazards consideration.

11. Simplifications to Power Distribution and APDMS T/S

The purpose of the eleventh group of proposed changes is to delete reference to the Axial Power Distribution Monitoring System (APDMS) from the T/Ss and to simplify the Power Distribution Limits T/Ss. These changes are numbered 32, 33, 37, 38, 39, 53, and 85 in Attachment 10.

The APDMS is an option currently provided in the T/Ss. It is required to be operable by T/S 3.3.3.7 when it is being used for monitoring axial power distribution. Power operation is permitted above the Allowable Power Level (APL) and below Rated Thermal Power provided additional surveillance is performed using the APDMS in accordance with T/S 4.2.6.1. In practice, however, the APDMS can be somewhat more limiting than APL. More importantly, experience has shown that APDMS causes extensive wear and tear on the Movable Incore Detector System, which the APDMS uses for data acquisition. This effect results in serious maintenance problems on a system which contains parts which are highly radioactive. For these reasons, it was decided not to operate with APDMS. Therefore, we are proposing to delete T/S 3/4.3.3.7, and to revise T/Ss 3/4.2.2 ($F_Q(Z)$) and 3/4.2.6 (Axial Power Distribution) to remove material related to APDMS.

In conjunction with the above, we have rewritten T/S 3/4.2.6. The proposed T/S contains the limits and surveillances required to establish and maintain APL, and has also been renamed accordingly. Most of the surveillance requirements of T/S 4.2.2 have been moved to T/S 4.2.6 in order to further simplify these T/Ss. It should be noted that the 2% penalty applied to $F^M_Q(Z)$ for increasing $F_{\Delta H}$ by T/S 4.2.2.2.e has been incorporated into the definition of APL in the proposed T/S 3.2.6. No requirements or limits currently in T/Ss 3/4.2.2 or 3/4.2.6, other than those related to APDMS and those discussed in the next paragraph, have been removed or reduced in our proposed revisions.

In addition to the changes described above, T/S 3.2.2 has also been revised to eliminate the need to place the reactor in Hot Standby to perform the Overpower ΔT trip setpoint reduction when this setpoint is

required to be reduced by Action Statement a. Our review of this requirement has determined that the reduction can be performed while the reactor is at power. The change in setpoint can be accomplished one channel at a time with bistables on the affected channel in the tripped configuration; therefore, there is no need to impose a transient on the reactor systems, which is inherent in changing from Modes 1 to 3. This change is consistent with guidance provided in Draft Rev. 5 of the STS.

Per 10 CFR 50.92, a proposed amendment will involve a no significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated,
(2) create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or
(3) involve a significant reduction in a margin of safety.

<u>Criterion 1</u>

The changes included in this group (with the exception of the Overpower Δ T trip setpoint reduction) should not involve a significant increase in the probability or consequences of an accident previously evaluated. These changes are administrative in nature and do not delete any requirements other than those associated with APDMS. As described earlier, APDMS is an option and is not required by T/Ss. For the Overpower Δ T trip setpoint reduction, the change is consistent with guidance provided by the Commission through the issuance of Draft Rev. 5 to the STSs. Although the changes may increase the probability or consequences of an accident, the results should be no worse than those previously accepted by the Commission through their issuance of Draft Rev. 5 to the STSs.

<u>Criterion 2</u>

The changes other than the Overpower Δ T trip setpoint reduction are administrative in nature. They do not introduce any new modes of plant operation, nor do they require physical changes to the plant. The changes associated with the Overpower Δ T trip setpoint are consistent with guidance provided by the Commission through the issuance of Draft Rev. 5 of the STSs and are presumed to be acceptable on that basis. Thus, we conclude that the changes will not create the possibility of a new or different kind of accident from any previously analyzed or evaluated.

<u>Criterion 3</u>

The changes included in this group (other than the Overpower Δ T trip setpoint reduction) should not involve a significant reduction in safety margins, since they are purely administrative and in no way reduce previous requirements for safety. Changes associated with the Overpower Δ T trip setpoint reduction may involve reductions in safety margins, but the results of the change are clearly within limits found acceptable to the Commission through their issuance of Draft Rev. 5 of the STSs.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The first of these examples refers to changes which are purely administrative in nature: for example, to achieve consistency throughout the T/Ss, to correct an error, or to make a change in nomenclature. The changes in this group (other than the Overpower Δ T trip setpoint reduction) are purely administrative in nature. They are intended to improve T/S readability by eliminating the APDMS option not currently exercised, and by rearranging the T/Ss to make them easier to use. No reductions in safety requirements will occur as a result of these changes.

As for the Overpower Δ T trip setpoint reduction, this change is similar to Example 6 published in the Federal Register. This example refers to changes which may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria. The elimination of the requirement to place the reactor in Hot Standby to perform the reduction does constitute a relaxation of a previous requirement, but the results of the change have been found acceptable by the Commission through their issuance of Draft Rev. 5 to the STSs.

Based on the above, we conclude that the examples cited are applicable and that the changes involve no significant hazards consideration.

12. Changes for Consistency With STS

The twelfth group of proposed changes consist of those that are requested to make our T/Ss more consistent with Rev. 4 of the STS. These are the changes numbered 57, 58, and 96 in Attachment 10, which also includes a description of the changes.

Per 10 CFR 50.92, a proposed amendment will involve a no significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated,
(2) create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or
(3) involve a significant reduction in a margin of safety.

Criterion 1

As these changes in general represent relaxation of current T/S requirements, they may involve an increase in the probability or consequences of an accident previously analyzed. The results of the changes, however, have been reviewed and found acceptable by the Commission through their issuance of Rev. 4 to the STSs. Thus, we conclude that any increase in probability or consequences would not be significant.

Criterion 2

As these changes will involve no physical plant changes and no T/S changes which are not consistent with Rev. 4 of the STSs, we conclude that they should not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3

Because these changes represent relaxation of present T/S requirements, they could potentially involve a reduction in safety margin. However, these changes are all consistent with those found acceptable by the Commission in Rev. 4 of the STSs. Thus, we conclude that any reduction in margins of safety are insignificant.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The sixth example refers to changes which may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria. The changes included in this group are consistent with Rev. 4 of the STSs. Although they may reduce safety requirements, the results of this change have been evaluated and found acceptable by the Commission.

Based on the above, we conclude that the example cited is applicable and that the change involves no significant hazards consideration.

Changes to the Bases

In addition to the changes to the T/Ss described above, we have also proposed changes to the Bases section to reflect both changes in the safety analyses and changes in the T/Ss. Descriptions of these changes have been included in Attachment 10.

Conclusion

In conclusion, we believe that the proposed changes do not involve significant hazards consideration because operation of D.C. Cook Unit 2 in accordance with these changes would not:

(1) involve a significant increase in the probability of occurrence or consequences of an accident previously analyzed,
(2) create the possibility of a new or different kind of accident from any accident previously evaluated, or
(3) involve a significant reduction in a margin of safety.

This conclusion is based on our evaluation of the changes, which has determined that all proposed changes which are not administrative in nature, consistent with the STS, or consistent with the design basis of the plant are clearly traceable to the Cycle 6 safety analyses, as referenced by Attachment 10. Assuming Commission acceptance of these analyses, it is our belief that they successfully demonstrate that applicable safety limits and margins to safety will be maintained.