INDIANA & MICHIGAN ELECTRIC COMPANY
DONALD C. COOK NUCLEAR PLANT

FACILITY CONCEPTUAL DESIGN DESCRIPTION

FOR THE

TECHNICAL SUPPORT CENTER AND THE EOF.

ATTACHMENT TO AEP:NRC:0531C

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TABLE OF CONTENTS

TABLE OF CONTENTS

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 <u>PLANT SAFETY STATUS DISPLAY (PSSD)</u>:

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 <u>NUCLEAR DATA LINK (NDL)</u>

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. It has a fully automatic fail-over capability . If a central processsing unit (CPU) or a portion of memory fails, the system will automatically reconfigure itself and continue to fully 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 isolators are 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 signal channel has its own Analog/Digital Converter, thus providing a high degree of realiability 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 disturbance. For printing lengthy reports, a hard copy terminal is provided for each Cook Unit Computer System.

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 hard copy output from the CRT screen image.

2.3.3 Emergency Operating Facilities (EOF):

A color CRT terminal, which can access either Cook unit TSC computer, is provided in the Emergency Operating Faciliies. 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.

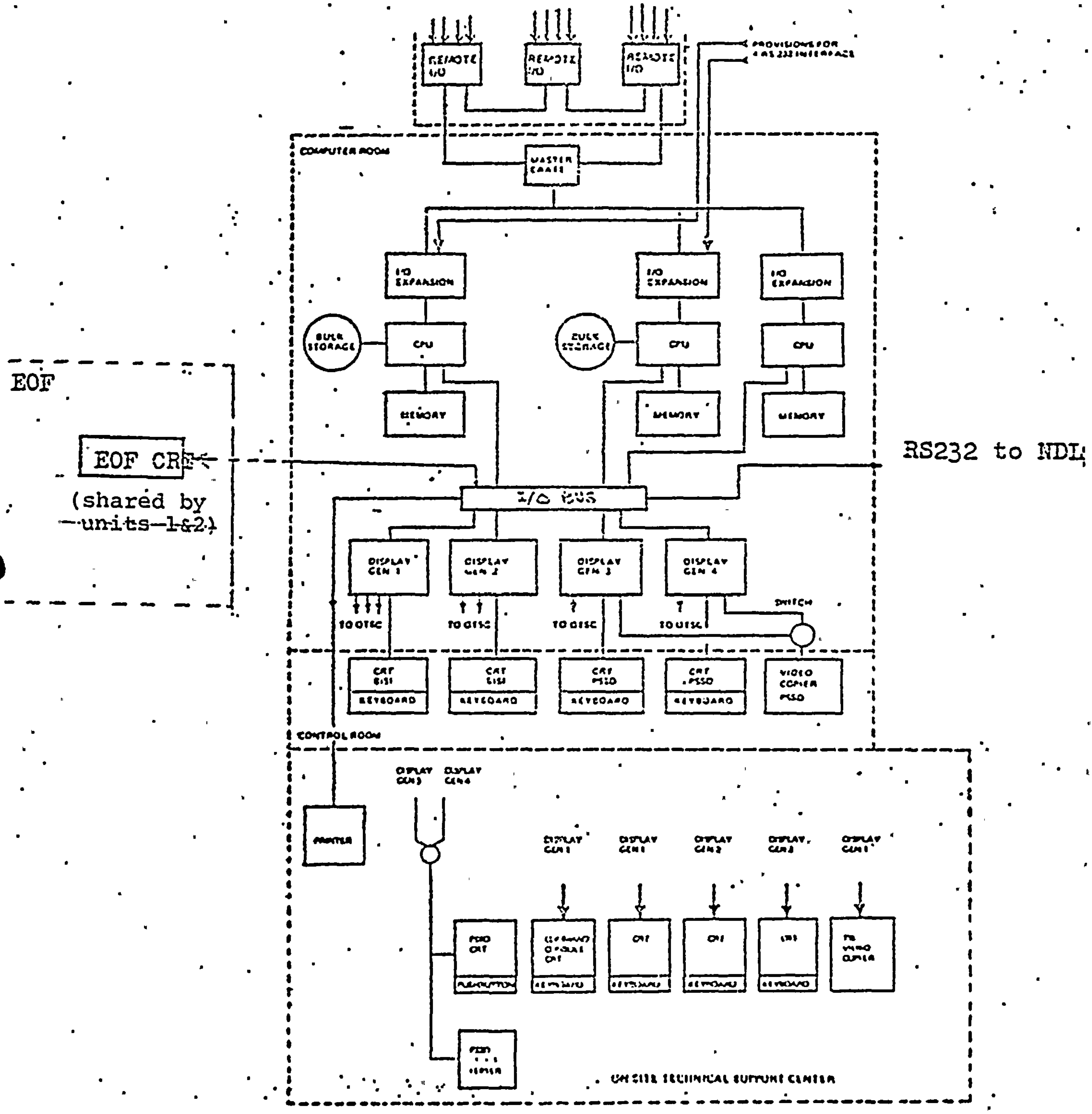

Figure 2.1. Technical Support Complex System Configuration

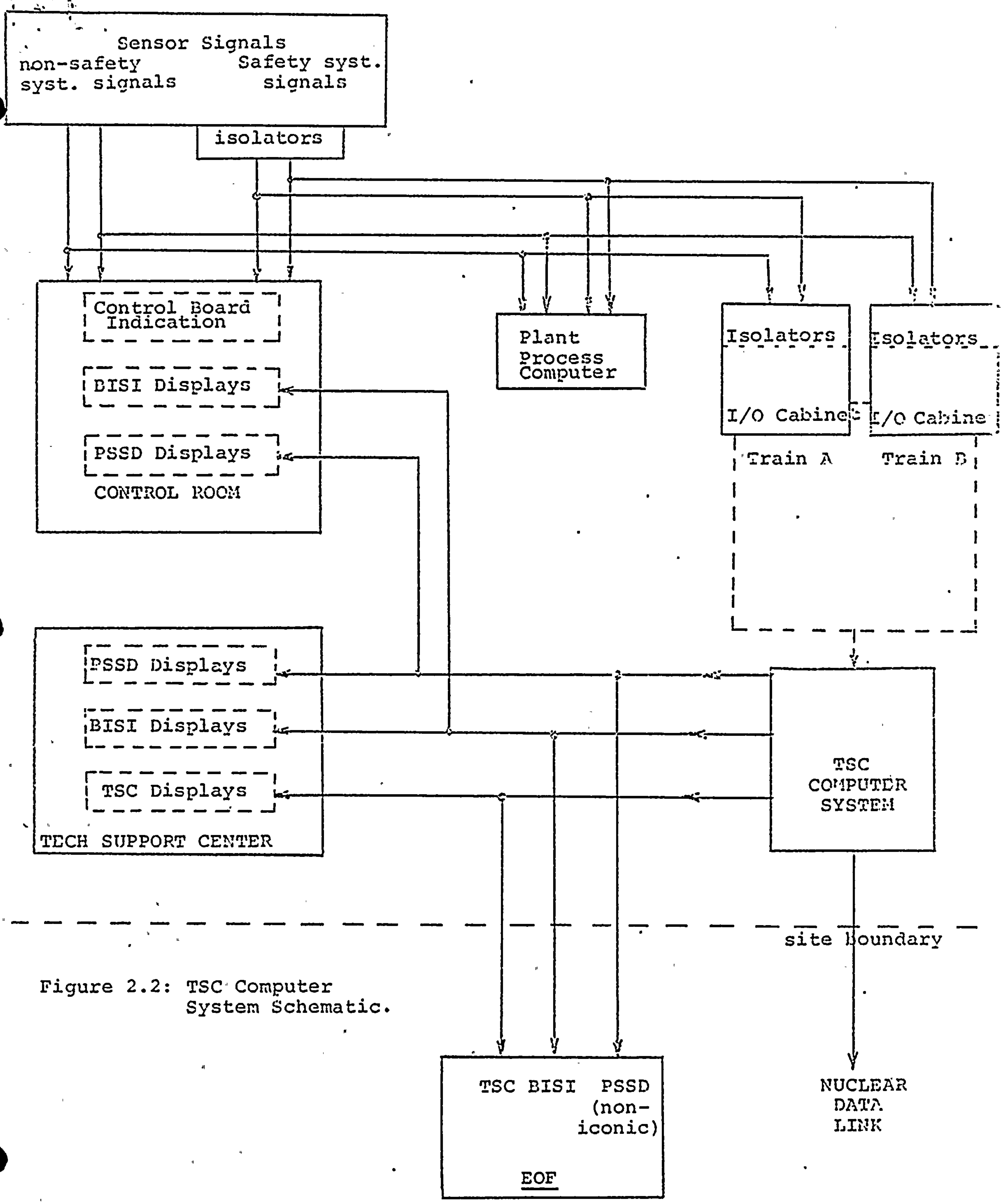

Figure 2.2: TSC Computer System Schematic.

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 push-buttons 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
Charging 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 transmit the required radiation signals to the TSC computer from this new Radiation Monitor System

TABLE 2-3

TSC INSTRUMENT BASIS

PARAMETER	INITIAL EVENT DIAGNOSIS*	BASIS

(b,c)

TABLE 2-3 (Continued)

TSC INSTRUMENT BASIS

PARAMETER	INITIAL EVENT DIAGNOSIS*	BASIS

(b,c)

TABLE 2-3 (Continued)

TSC INSTRUMENT BASIS

PARAMETER	INITIAL EVENT DIAGNOSIS*	BASIS	(b,c)

TABLE 2-3 (Continued)

TSC INSTRUMENT BASIS

PARAMETER	INITIAL EVENT DIAGNOSIS*	BASIS

(b,c)

TABLE 2-3 (Continued)

TSC INSTRUMENT BASIS

PARAMETER	INITIAL EVENT DIAGNOSIS*	BASIS	(b,c)

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TABLE 3.3 (Continued)

TSC INSTRUMENT BASIS

PARAMETER	INITIAL EVENT DIAGNOSIS*	BASIS

(b,c)

TABLE 3.3 (Continued)

TSC INSTRUMENT BASIS

PARAMETER	INITIAL EVENT DIAGNOSIS*	BASIS	(b,c)

TABLE 2-3 (Continued)

TSC INSTRUMENT BASIS

PARAMETER	INITIAL EVENT DIAGNOSIS*	BASIS	(b,c)

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)

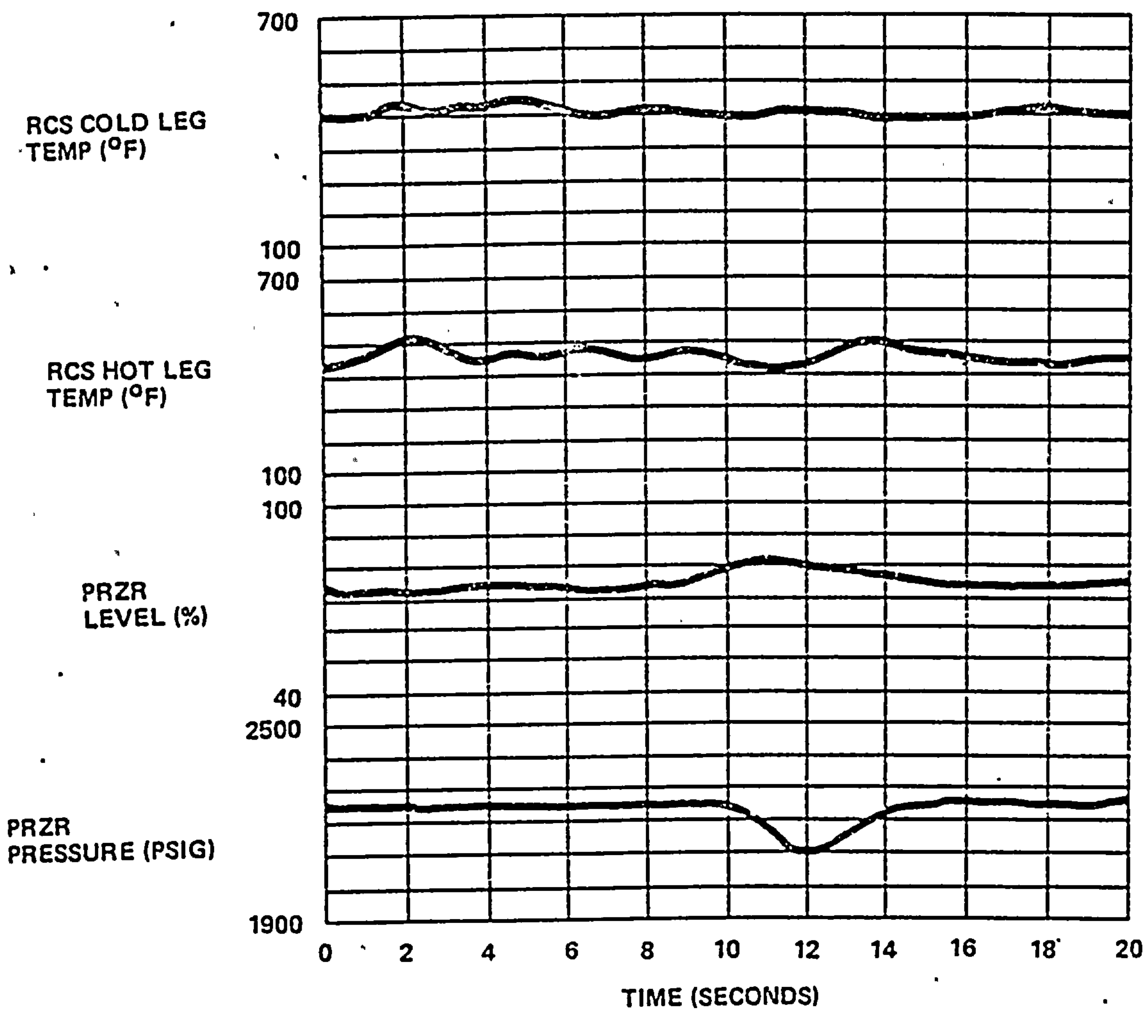


Figure 3.3 Graphic Display at Onsite Technical Support Center (Example)

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 [(b,c,e)

]

Because of the fact that [(b,c,e)

] the PSSD incorporates [(b,c,e)

]

(b,c,e) [

]. The parameters available for

(b,c,e) [

(b,c,e)]. The role for which the PSSD provides [

] is as follows:

(b,c,e) [

[(b,c,e)

]

By addressing [(b,c,e)

]

[(b,c,e)

]

In defining the inputs for the PSSD, [ ] as follows: (b,c,e)

[(b,c,e)

]

In response to the [(b,c,e)

]

(b,c,e) In order to satisfy the [

].

4.3 MAN-MACHINE INTERFACE

The PSSD system will process the defined input data set of plant param-
(a,b,c) eters at [ ] and generate displays for redundant PSSD
(a,c) dedicated CRTs located in the control room. [
]

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 [
]
depicted in Figure 4-1. The display system structure should be defined
(b,c) such that it [
] are defined as follows:

(b,c) [

].

[(b,c)

]

The display structure shown in Figure 4-2 [(a,c,f)

]

(a,c,f)

A major problem associated with the man-machine interface is the

(a,c,f)

. Figure

(a,c,f) 4-3 is an illustration of the display.

(a,c,f) Figures 4-4 and 4-5 are preliminary versions of 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)

(a,c,f)

The information at [(a,c,f)

]

TABLE 4-1

PLANT SAFETY STATUS DISPLAY - SAFETY GOALS - TERMINATE MODE TRANSIENTS

(b,c,e)

[ ]

5435A

TABLE 4-2

PLANT SAFETY STATUS DISPLAY - SAFETY
GOALS - MITIGATE MODE TRANSIENTS

(b,c,e)

TABLE 4-2 (Continued)

PLANT SAFETY STATUS DISPLAY - SAFETY
GOALS - MITIGATE MODE TRANSIENTS

(b,c,e)

5435A

TABLE 4-3

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

5435A

TABLE 4-4

PLANT SAFETY STATUS DISPLAY - MITIGATE MODE PARAMETERS

b,c,e) [ ]

(a,c)

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 5.1.

The primary display, an example of which is shown in Figure 5.2, 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.

Other levels of displays such as shown in Figure 5.3 provide supporting information on individual components within each subsystem and support

(a,c,f) system. [

]

Whenever the status of a system becomes inoperable or bypassed, the

(a,c,f) [

]

[(a,c,f

]

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,f

Figure 5.1 Display Structure — Bypassed and Inoperable Status Indication

(a,c,f)

Figure 5.2 Primary Display -- Bypassed and Inoperable Status Indication

(a,c,f)

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. Transformer isolators are provided for all analog input signals and optical isolators are provided for all digital input signals. In addition, all signals from the Reactor Protection Channels are taken after the existing safety grade isolators. Therefore, the interfacing of the TSC computer system to the existing plant instrumentation will not result in any degradation of the control room, protection system, controls or other plant functions.

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)

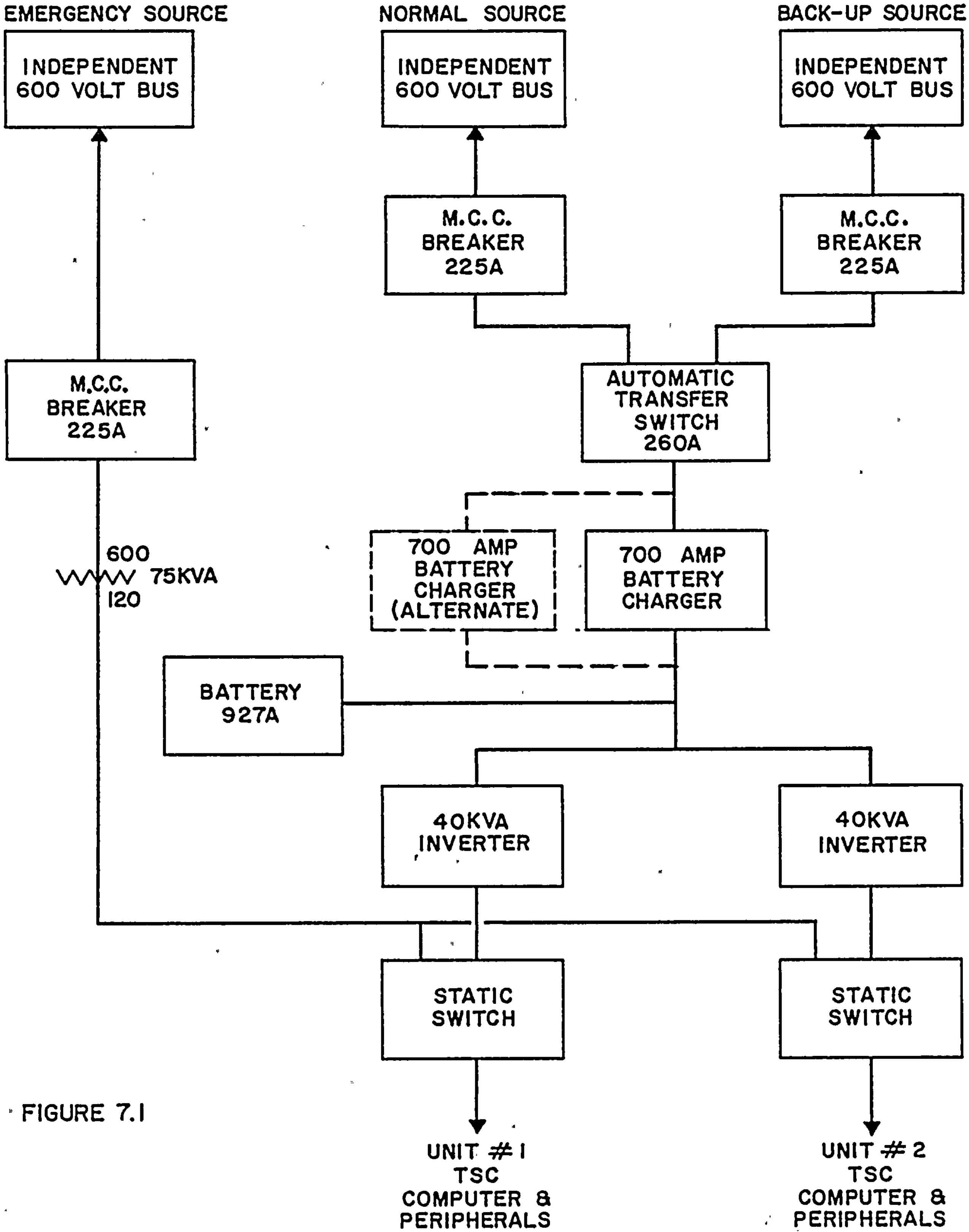


FIGURE 7.1

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).

8. TSC AND EOF FUNCTIONS

8.1 TASK FUNCTIONS PERFORMED BY INDIVIDUALS IN THE TSC:

The emergency functions/tasks performed by individuals required to report to the TSC are described by the following:

8.1.1 RADIATION MONITORING

- Coordinate activities of field assessment teams.
- Receive Data from personnel in the field.
- Provide RADIATION MONITORING DATA to appropriate personnel.
- Dosimetry Control.

8.1.2 DOSE ASSESSMENT:

- Receive data from communications personnel on Radiation Monitoring and Meteorological conditions.
- Receive Radiation Monitoring Data from Radiation Monitoring Director.
- Perform Dose Assessment calculations.
- Provide recommended protective actions, as neceessary.
- Assist in classification of event.
- Place Control Room Data in format useful to off-site agencies.

8.1.3 COMMUNICATIONS

- Receive event data from Control Room.
- Place event data in format useful for Dose Assessment personnel, Plant Status Evaluation personnel, and Off-site Agencies.

- Communicate event conditions to the Berrien County Sheriff's Department; Michigan State Police; the Joint Public Information Center; AEP Service Corporation; and Industry Support Groups.

8.1.4 TECHNICAL SUPPORT

- Provide technical support to Operations personnel in areas such such as Coré Analysis, Chemical Control, Cycle Evaluation, and instrumentation.
- Provide Independent Evaluation of the Safety status of the Unit.

8.1.5 MANAGEMENT SUPPORT

- Provide for availability of site personnel as needed.
- Provide direction and priorities for site personnel activities.
- Provide evaluation of Emergency measures to be taken on-site and off-site.

8.2 EMERGENCY FUNCTIONS PERFORMED IN THE TSC/EOF FOR EACH EMERGENCY CLASS:

8.2.1 UNUSUAL EVENT:

Plant conditions requiring declaration of an "Unusual Event" are not expected to require activation of the Technical Support Center or Emergency Operations Facility. The Emergency functions previously described will be delegated/coordinated from the Control Room.

8.2.2 ALERT:

Plant conditions requiring declaration of an "Alert" vary in severity level from the upper bounds of "Unusual Event" to the lower bounds of a "Site Emergency". An "Alert" classification therefore, may require performance of portions of the functions, described above in Section 8.1, in the Technical Support Center and the Control Room, or it may require performance of all of the emergency functions in the Technical Support Center and the Emergency Operations facility.

The degree of activation of the TSC/EOF is a function of time as well as of event severity. At the time the event occurs all emergency functions will be performed in the Control Room, with first priority for operator actions given to event mitigation.

Approximately 1 hour after event occurrence the required emergency functions will be divided between the Control Room and the TSC. If the event is of low severity within the "Alert" category, a majority of these functions will be performed in the Control Room, as applicable to the event. If the event is of high severity within the "Alert" category, it is expected that the majority of these functions will be performed in the TSC, as applicable to the event.

If the event continues for a long period of time, such as 24 hours or more, response group arrivals at the site will require full activation of the TSC and a partial activation of the EOF, independent of the relative severity of the "Alert" event. All applicable emergency functions will be performed in the TSC with the exception of dose assessment, which may shift to the EOF.

8.2.3 SITE AND GENERAL EMERGENCY

A "Site Emergency" will require full activation of the TSC. Except for the dose assessment function, which may shift to the EOF, all applicable emergency functions of Section 8.1 will be performed in the TSC.

A "Site Emergency" is not expected to occur instantaneously; however, should this occur, the TSC will be activated, staffed and assume emergency functions from the Control Room, within 1 hour of event occurrence. The EOF is expected to be activated within 4 to 6 hours and it will assume the dose assessment function from the TSC.

A "General Emergency" will require full activation of the TSC and EOF and the functions performed by site personnel assigned in these facilities is expected to be identical to those functions performed for a "Site Emergency."

8.3 FUNCTIONS OF INDIVIDUALS REPORTING TO THE EOF:

The emergency function/task performed by indviduals required to report to the EOF are described in detail in the DCCNP Emergency plan Chapter 12.3.3.3 and are generalized below by the following four categories:

- Coordination of Off-site Radiological Monitoring and Dose Assessment.
- Technical Support of Plant Recovery Operations.
- Management Support of Recovery Operations.
- Communication with Offsite Agencies.

9. TSC RECORDS AND DATA AVAILABILITY

It is necessary to make available in the Technical Support Center the reference material and data source material needed to make a technical evaluation of an accident or emergency situation. Therefore, up-to-date plant specific documents and general technical references needed to implement this function will be maintained in the Technical Support Center.

9.1 CONTROLLED PLANT SPECIFIC REFERENCE MATERIAL:

For plant specific reference, the following controlled material will be kept in the TSC:

- Technical Sepcifications.
- Abnormal and Emergency Operating Procedures.
- Detailed elementary electrical diagrams, and detailed flow diagrams. Included with this information are plant arrangement diagrams showing component locations.
- Contour area map with population distribution and overlays for plume evaluation.
- Donald C. Cook Nuclear Plant Emergency Plan with procedures.
- System Descriptions.
- Precautions, Limitations and Setpoints
- Plant Technical Data Book containing curves for reactivity control, rod worth, RSC temperature and pressure limits and secondary plant performance.

The above material will be controlled by the Donald C. Cook Plant document control system which is governed by Plant Manager Instruction 2030, entitled Document Control.

9.2 UNCONTROLLED INFORMATION AND TECHNICAL REFERENCE MATERIALS:

In addition, other plant information which is useful will be present in the TSC:

- Pump and fan performance curves.
- Final Safety Analysis Report.
- Annunciator Layouts.
- Tank Volume/level curves.
- Appropriate Plant Manager Procedures.
- Reference copies of miscellaneous emergency procedures; and Unit Vent Emergency Release Level Determination and Secondary System Emergency Release Determination guides.

General technical reference materials will also be available, such as:

- Steam tables.
- Chart of Nuclides.
- Standard Handbook for Electrical Engineers.
- Handbook of Chemistry and Physics.
- Standard Handbook for Mechnaical Engineers.
- Thermodynamics Handbook.
- Nuclear Reactor Engineering Handbook.
- Radiological Health Handbook.
- Instrument Engineers Handbook.
- Pump Handbook.

The above-mentioned items will be maintained in the TSC readily available and other materials deemed necessary may be added in the future at any time.

9.3 <u>OTHER DATA, RECORDS, AND INFORMATION:</u>

The following references are immediately available to the personnel in the Control Room:

- Normal Operating Procedures
- Abnormal Operating Procedures
- Emergency Operating Procedures.

The following reports and information are available in the plant library:

- Plant Operating Records
- Plant Nuclear Safety Review Committee Reports
- Vendor manuals, and component level drawing and sketches
- State of Michigan and local emergency preparedness plans.

The plant library is located in the office building, which is in close proximity to the TSC and Control Rooms.

Attachment C to AEP:NRC:0745H

Responses to Questions on Unit 1 Cycle 8

Thermal Hydraulics (Non-Proprietary)

QUESTIONS 1 through 3:

What is the basis for using the WRB-1 correlation for 15x15 OFA?
What is the basis for the DNBR limit of 1.17 for WRB-1 applied to 15x15 OFA?
Are there any critical heat flux data?

RESPONSE:

The WRB-1 CHF correlation is based entirely on rod bundle data and has been shown to provide a significant improvement in DNB predictive capability for Westinghouse fuel designs with type "R" mixing vane grids. The NRC has recognized this increased accuracy and concurred that a 95/95 limit DNBR of 1.17 is appropriate for 12 ft and 14 ft 17x17 standard and optimized fuel assemblies, and 12 ft 15x15 standard fuel assemblies with the type "R" mixing vane grid (Ref. 1 and 2). Based on the semi-empirical nature of the correlation, the NRC has imposed restrictions on its applicability to other PWR designs. Specifically, the Safety Evaluation Report stated that, "The correlation should not be applied to any PWR geometry which has not been specifically tested or which has not been bracketed by the test data. The important parameters to which this applies are: rod size, rod pitch, heated length, mixing vane design and grid spacing."

The 15x15 optimized design is virtually identical to the 15x15 R-grid design in that the [

]+ (a,c)

As will be discussed below, similar scaling techniques have been used for designing the 17x17 OFA and 14x14 OFA grids, and DNB testing has shown that the WRB-1 correlation correctly predicts the performance of those designs without modifications.

Based on the previous success of this grid scaling technique (as demonstrated by the 17x17 OFA and 14x14 OFA DNB test results) and the similarity of the 15x15 OFA and R-grid geometries, use of the WRB-1 CHF correlation with a design limit of 1.17 is justified for the 15x15 OFA design.

17x17 OFA DNB Test Results

Geometrically the 17x17 OFA design differs from the standard 17x17 R-grid design in that:

1) The fuel rod diameter was reduced from 0.374 inch to [ ]+ inch. (a,c)

2) The Zircaloy type "R" grid is [ ]+ than the Inconel type "R" grid which has previously been DNB tested. (a,c)

In order to minimize the effect of the grid dimensional changes on DNB performance, special care was taken to preserve the important type "R" mixing vane characteristics. [(a,c)

]+ DNB testing of the 17x17 OFA geometry demonstrated the success of this scaling approach--the WRB-1 correlation predicted the data well without any modifications, using the same performance factor as was used for the 17x17 standard fuel. Repeatability studies (Ref. 3) have shown that the accuracy of the WRB-1 correlation is essentially identical for the 17x17 OFA and standard geometries, indicating that no additional component of variance is introduced by the grid dimensional changes. In other words, the correlation correctly accounted for the equivalent diameter effects and the scaling approach correctly accounted for the grid dimensional changes.

14x14 OFA DNB Test Results

The 14x14 optimized geometry differs from the standard geometry in that:

1) The fuel rod diameter has been reduced from 0.422 inch to [ ]+inch. (a,c)

2) The Zircaloy type "R" grid is [ ]+ (a,c) than the Inconel type "R" grid which had previously been DNB tested.

A CHF test series of the 14x14 OFA typical cell geometry has been performed to verify that the WRB-1 correlation correctly predicts the effect on CHF of the equivalent diameter change, and that the grid scaling approach introduces no additional component of variance. As will be discussed below, the results indicate that the WRB-1 correlation predicts the 14x14 OFA data with essentially the same accuracy as for the geometry from which it was scaled.

Test Facilities

The test facilities and testing procedures used for the 14x14 OFA CHF tests were the same as those described in References 4 and 5. The test section was similar to the 0.422 inch rod bundle described in Reference 4, except that the mixing vane grid dimensions were modified slightly in order to accommodate the new rod diameter and the change from Inconel to Zircaloy. The modified grid design has retained the type "R" grid features. Figure 1 shows a sketch of the 14x14 OFA typical cell test bundle cross section. The axial locations of the grids and thermocouples are shown in Figure 2, and Figure 3 shows the cosine axial power distribution used for the tests.

CHF Data Evaluations

The data were reduced using the THINC subchannel code, in the same manner as described previously in References 4 and 5. The WRB-1 correlation of Reference 6 was used to predict the critical heat flux. The performance factor used was the same as that employed for the 0.422 inch data evaluations [ ]+, since the mixing vane grid size was [(a,b,c)

]+ As discussed above, this (a,c)

approach had previously worked quite well with the 17x17 OFA CHF data.

The results of the data reduction are shown in Table 1. The average measured-to predicted critical heat flux ratio for the data set is [ ]+ with a (b,c) sample standard deviation of [ ]+. These values were compared to those (b,c) from the 0.422 inch rod bundle tests with 26 inch grid spacing, the geometry from the original WRB-1 R-grid database which is closest to the 14x14 OFA geometry. As shown in Table 2 the agreement is excellent, indicating that the WRB-1 correlation correctly accounts for the geometry changes and that the choice of performance factor is appropriate. Also given in Table 2 is a comparison of the 17x17 standard and OFA DNB statistics. It is apparent that the WRB-1 correlation's ability to predict CHF is essentially identical for standard and OFA fuel designs.

T-tests and F-tests have been performed for each of these standard/OFA data set pairs in order to evaluate the effect of the geometry changes on the accuracy of the WRB-1 correlation. Table 3 shows the results of these tests. It can be seen that the hypothesis that the WRB-1 correlation predicts the DNB behavior of the OFA geometries with the same accuracy as the standard R-grid geometries cannot be rejected at a 5% significance level, with the exception of the [ ]+ comparison. (b,c) For that comparison the OFA data had an appreciably lower variance. A smaller variance is indicative of better correlation accuracy, so failure of the F-test is no reason for concern. Therefore, the results of these tests indicate that no additional component of variance is introduced by the grid dimensional changes.

FIGURE 1

+b,c

FIGURE 2

AXIAL GRID AND CHF DETECTOR LOCATIONS,
14X14 OPTIMIZED CHF TEST SECTION

+b,

+b,c

Q''_{LOC}/Q''_{AVG}, LOCAL-TO-AVERAGE HEAT FLUX RATIO

Z, AXIAL DISTANCE FROM BEGINNING OF HEATED LENGTH (INCHES)

Figure 3. Axial Heat Flux Distribution

TABLE 1 - CHF TEST RESULTS FOR 14x14 OFA TYPICAL CELL USING WRB-1 CORRELATION

(b,c)

RUN NO.	INLET PRESSURE (PSIA)	INLET TEMP (F)	INLET MASS VELOCITY (X10E6 LBM/HR-SQFT)	LOCAL QUALITY (P)	LOCAL HEAT FLUX (X10E6 BTU/HR-SQFT) MEAS.	LOCAL HEAT FLUX (X10E6 BTU/HR-SQFT) PRED.	M/P CHF (WRB-1)	ELEVATION FROM INLET (INCHES) PRED.	ELEVATION FROM INLET (INCHES) MEAS. (b,c)
W2242							1.0599		
W2243							1.1582		
W2244							1.0593		
W2245							1.1064		
W2246							1.1149		
W2247							1.1483		
W2248							1.0943		
W2249							1.1320		
W2250							1.1482		
W2251							1.1118		
W2252							.9741		
W2253							1.1036		
W2254							1.0983		
W2255							1.0939		
W2256							1.0844		
W2257							1.0783		
W2258							1.1018		
W2259							.9514		
W2260							1.1338		
W2261							1.0550		
W2262							.8592		
W2263							.9104		
W2264							.9466		
W2265							1.0739		
W2266							1.0461		
W2267							.9942		
W2268							1.0154		
W2269							1.0146		
W2270							1.0065		
W2271							.9615		

TABLE 1 (CONTINUED)

CHF TEST RESULTS FOR 14x14 QFA TYPICAL CELL

RUN NO.	INLET PRESSURE (PSIA)	INLET TEMP (F)	INLET MASS VELOCITY (X10E6 LBM/HR-SQFT)	LOCAL QUALITY (P)	LOCAL HEAT FLUX (X10E6 BTU/HR-SQFT) MEAS.	LOCAL HEAT FLUX (X10E6 BTU/HR-SQFT) PRED.	M/P CHF (WRB-1)	ELEVATION FROM INLET (INCHES) PRED.	ELEVATION FROM INLET (INCHES) MEAS.
							+(b,c)		+b,c
W2272							1.1481		
W2273							1.0612		
W2274							.9741		
W2314							.9314		
W2315							1.0735		
W2316							1.1411		
W2317							1.1465		

L = 14 FT
DE = .5040 IN
4 RODS 100%
12 RODS 85%

ROD O.D. = [.400 IN] +a,c
ZIRC SPRING MV GRIDS 26 IN SPACING
INNER ROD/OUTER ROD POWER = 1.1765

TABLE 2

STATISTICAL COMPARISON OF STANDARD AND OPTIMIZED FUEL CHF RESULTS USING THE WRB-1 CORRELATION

+b,

TABLE 3

F-test and t-test Results for Standard/OFA

Data Set Pairs in Table 2

*For these tests the 0.422 inch rod DNB data sets have been grouped.

REFERENCES

1) Letter, D. F. Ross, Jr. (NRC) to D. B. Vassallo (NRC), Subject: Topical Report Evaluation for WCAP-8762, April 10, 1978.

2) Letter, R. L. Tedesco (NRC) to T. M. Anderson (Westinghouse), Subject: Acceptance for Referencing Topical Report WCAP-9401 (P)/WCAP-9402 (NP), May 7, 1981.

3) Beaumont, M. D., Skaritka, J., "Verification Testing and Analyses of the 17x17 Optimized Fuel Assembly," WCAP-9401, March 1979.

4) K. W. Hill, F. E. Motley, F. F. Cadek, A. H. Wenzel, "Effect of 17x17 Fuel Assembly Geometry on DNB," WCAP-8926-P-A (Westinghouse Proprietary) and WCAP-8297-A (Non-proprietary), February 1975.

5) F. E. Motley, A. H. Wenzel, F. F. Cadek, "Critical Heat Flux Testing of 17x17 Fuel Assembly Geometry with 22-inch Grid Spacing," WCAP-8536, (Westinghouse Proprietary) and WCAP-8537 (Non-proprietary), May 1975.

6) Motley, F. E., Hill, K. W., Cadek, F. F., Shefcheck, J. J., "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," WCAP-8762, (Westinghouse Proprietary), July 1976.

QUESTION 4:

What is the reason for using two critical heat flux correlations in the same core?

RESPONSE:

The Exxon fuel currently in the D.C. Cook Unit 1 core was originally licensed with the W-3 critical heat flux correlation (Reference: D.C. Cook Unit 1 Final Safety Analysis Report). Therefore, the Westinghouse analyses of the Exxon fuel during the transition cycles have also utilized the W-3 correlation.

The WRB-1 critical heat flux correlation was developed from a large body of Westinghouse mixing vane grid rod bundle CHF data, and has been shown to predict CHF for fuel designs with the type "R" grid with better accuracy than previous correlations. The WRB-1 correlation was, therefore, selected for analyses of the Westinghouse 15x15 optimized fuel design, which uses mixing vane grids of the type "R" design. Further justification for the use of the WRB-1 correlation for the 15x15 OFA design is provided in the responses to Questions 1 through 3.

NRC QUESTION NO. 5

A 5% DNBR penalty for the transition mixed core is used in this reload (D.C. Cook Unit 1 Cycle 8) as a result of analysis using the same methods as applied for the 17x17 OFA and 17x17 LOPAR cores. Provide your analyses and results.

RESPONSE

Attached are results of the analyses which were performed in order to calculate the D.C. Cook Unit 1 Cycle 8 transition core DNBR penalty of 5%.

TABLE 1

COMPUTER ANALYSES MADE TO JUSTIFY TRANSITION CORE METHODS

+(a,c)

Run	Configuration (Figure No.)	Pressure (psia)	Inlet Temperature (°F)	Power (% of 16.84 MWt/Assy)	Flow (% of 1900 gpm/assy)	Axial Power Distribution (Figure No.)
1						
2						
3						
4						
5						
6						
7						
8						
9						
10						
11						
12						
13						
14						
15						
16						
17						
18						
19						
20						
21						
22						
23						
24						
25						

TABLE 2 - RESULTS OF TRANSITION CORE
DNB PENALTY SENSITIVITY STUDIES

Type	Runs	ΔDNBR (%)

+(a,c)

FIGURE 1

TRANSITION PATTERN 1

+(a,c)

Key:

ENC - ENC 15x15 Fuel Assembly

OFA - W 15x15 OFA

FIGURE 2

TRANSITION PATTERN 2

+(a,c)

Key:

ENC - ENC 15x15 Fuel Assembly

OFA - W 15x15 OFA

FIGURE 3

REPRESENTATIVE AXIAL POWER DISTRIBUTION

+(a,c)

FIGURE 4

REPRESENTATIVE AXIAL POWER DISTRIBUTION

+(a,c)

FIGURE 5

REPRESENTATIVE AXIAL POWER DISTRIBUTION

+(a,c)

FIGURE 6

REPRESENTATIVE AXIAL POWER DISTRIBUTION

+(a,c)

FIGURE 7 MASS VELOCITY VS. ELEVATION

(a,c)

FIGURE 8 LOCAL QUALITY VS. ELEVATION

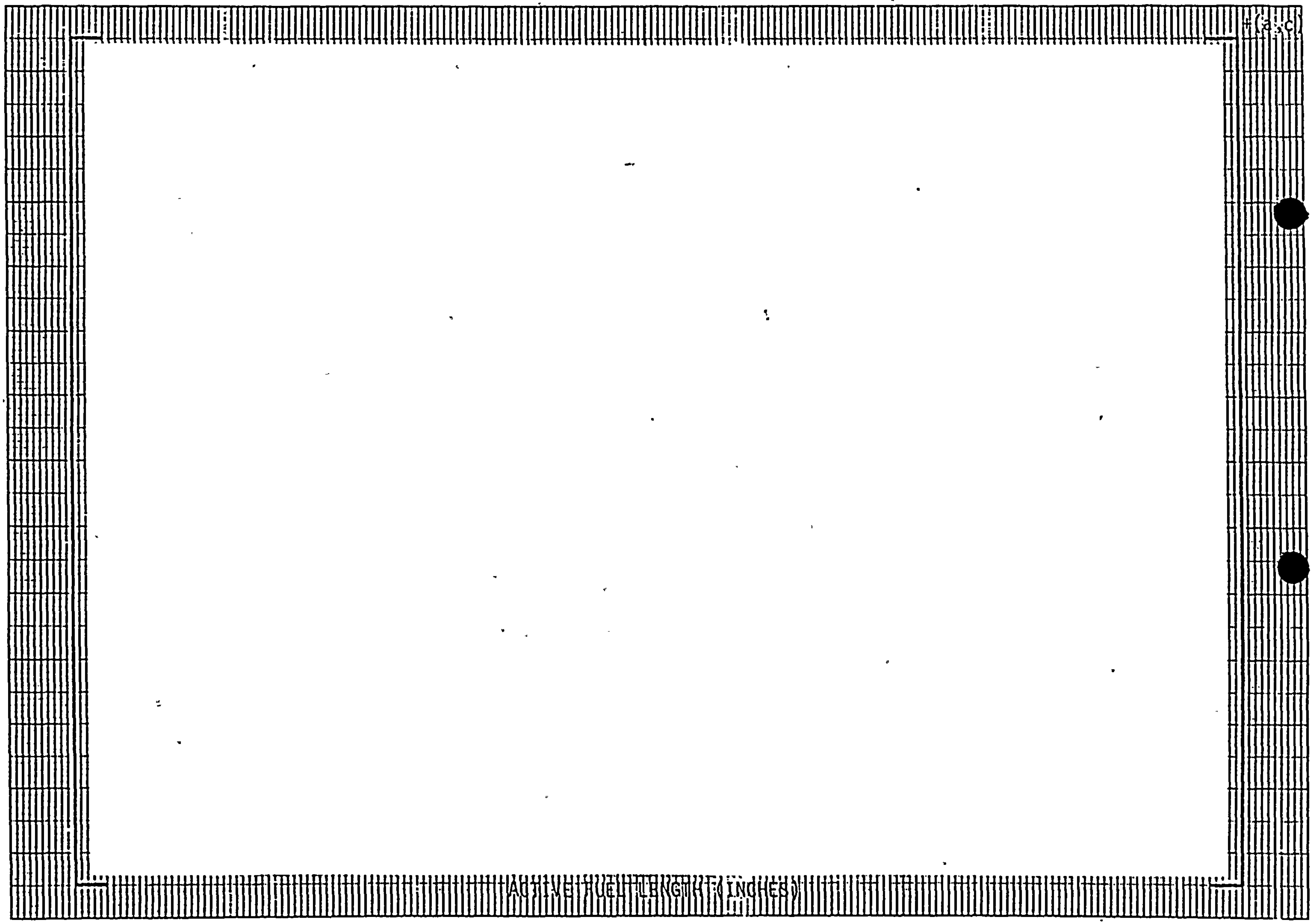
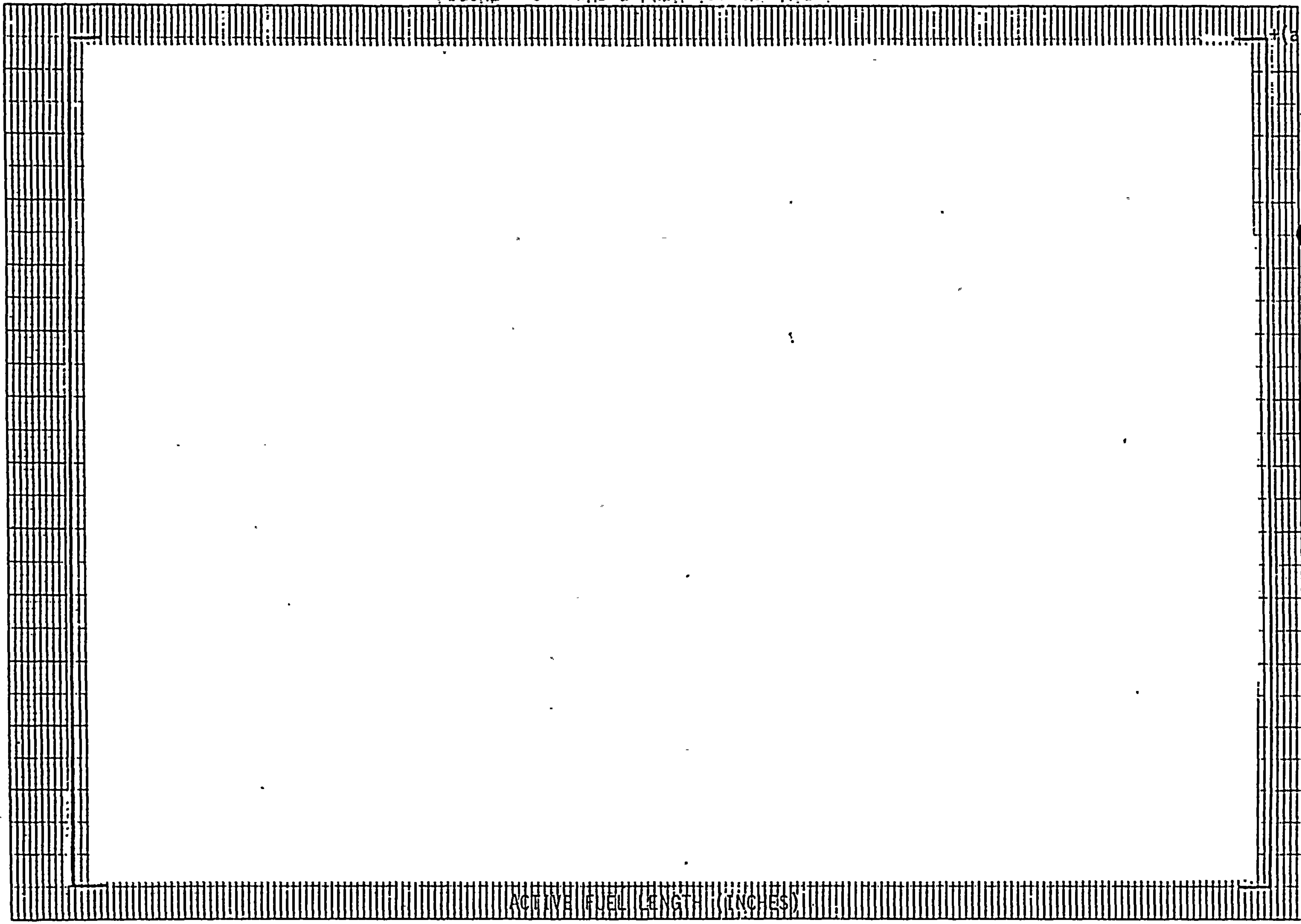


FIGURE 9 WRB-1 DNBR VS. ELEVATION



Attachment D to AEP:NRC:0745H

Response to Question on Fuel Seismic Analysis

STRUCTURAL ANALYSIS OF MIXED OFA/EXXON REACTOR CORE DURING LOCA EVENTS AND DURING SEISMIC EVENTS

An accident analysis was performed to establish the structural adequacy of the optimized fuel assembly design for use in the D.C. Cook Unit 1 (AEP) Plant. The specific objective was to determine the maximum fuel assembly response during a seismic or LOCA accident and to verify that the Westinghouse 15x15 optimized fuel assemblies remain coolable. An analysis described in Section 3.5 of the Donald C. Cook Unit 1 July, 1982 updated FSAR, dealt with the adequacy of the Exxon fuel assemblies when mixed with Westinghouse 15x15 standard (Inconel grid) fuel assemblies. This analysis bounds a mixed core consisting of Exxon and Westinghouse 15x15 optimized fuel assemblies. Since this plant currently contains Exxon fuel assemblies, the reactor core was modeled using various core loading patterns that contained both Westinghouse and Exxon fuel assemblies.

In order to perform the reactor core structural analysis, the mechanical properties of the Westinghouse 15x15 (Inconel grid type) fuel assembly design were used to simulate the Exxon fuel properties in the modeling of the Exxon fuel. It has been confirmed that this modeling of the Exxon fuel is conservative in predicting the response of the Westinghouse 15x15 optimized fuel.

LOCA ANALYSIS

The fuel assembly response resulting from the most limiting main coolant pipe break was analyzed using time history numerical techniques. The vessel motion for the LOCA accident produces substantial lateral loads on the reactor core, and therefore, a finite element model similar to the seismic model described in Ref. (1) was used to determine the fuel assembly deflections and grid impact forces.

The reactor core finite element model which simulates the fuel assembly interaction during lateral excitation consists of fuel assemblies arranged in a planar array with inter-assembly gaps. For the D.C. Cook Unit 1 Plant,

fifteen (15) fuel assemblies which correspond to the maximum number of assemblies across the core diameter were used in the model. The fuel assemblies are schematically represented by individual beam elements as shown in Figure 1. A spring and lumped mass system model consistent with the model described in Ref. (2) was used to represent the simplified fuel assembly elements in Figure 1. The discrete masses and spring rates were calculated directly from the fuel assembly frequencies and corresponding mode shapes. This type of model was adopted because it inherently provides an accurate representation of the fuel assembly higher natural frequencies and mode shapes.

Because of the mixed core consideration, four (4) fuel assembly reactor core patterns were selected for analysis. The Westinghouse/Exxon fuel assembly relative locations for the various patterns are shown in Figure 2. These reactor core reload patterns are consistent with typical reload configurations.

The time history motion for the upper and lower core plates and the barrel at the upper core plate elevation are simultaneously applied to the simulated reactor core model as illustrated in Figure 1. The three time history motions were obtained from a time history analysis involving a finite element model of the reactor vessel and internals.

The fuel assembly response, namely the displacements and grid impact forces, was obtained with the reactor core model by using the core plate and barrel motions that result from a reactor vessel inlet nozzle break. The reactor vessel inlet break has been shown to produce the limiting structural loads for the fuel assembly. The maximum grid impact forces for both the LOCA and seismic accidents occur at the peripheral fuel assembly locations adjacent to the baffle wall. The grid impact forces are appreciably lower for fuel assembly locations inward from the peripheral fuel. For the lateral blowdown case, only a small (outer) portion of the core experiences large grid impact forces.

The grid maximum impact forces and fuel assembly maximum deflection obtained from the nozzle inlet break for the four reload patterns are given in Table 1. An examination of Table 1 shows only minor differences in the fuel assembly maximum deflection and grid impact forces for the various reload patterns.

Table 1

LOCA INDUCED FUEL ASSEMBLY FORCES AND DEFLECTION

	Case 1*	Case 2	Case 3	Case 4
Grid Max. Impact Force (% of Allowable Limit)	58	59	56	59
Fuel Assembly Max. Deflection (in)	.73	.75	.75	.75

----

*Refer to Figure 2 for core pattern

Seismic Analysis

A seismic analysis of the reactor internals was performed using a synthesized time history wave which produced a response spectra that enveloped the D.C. Cook Unit 1 Plant design requirement. The time history results obtained from that analysis were used as input to the model shown in Figure 1 to obtain the reactor core seismic response. Since the reactor core responses obtained from the LOCA analysis were essentially the same, only three of the four core reload patterns were analyzed for the seismic accident.

The grid maximum impact forces and fuel assembly maximum deflections obtained from the seismic analysis of the three reload reference patterns are given in Table 2. The results of the analysis show that Case 1 is the most limiting pattern based on the grid impact forces. The homogeneous core consisting of all Westinghouse optimized fuel assemblies, which is Case 4, exhibited the most margin.

Table 2

SEISMIC INDUCED FUEL ASSEMBLY FORCES AND DEFLECTIONS

	Case 1	Case 3	Case 4
Grid Max. Impact Force (% of Allowable Limit)	80	75	54
Fuel Assembly Max. Deflection (in)	.71	.86	.89

FUEL ASSEMBLY COMPONENT STRESSES

The stresses induced in the various fuel assembly components were assessed based on the most limiting seismic and LOCA accident conditions. The fuel assembly axial forces resulting from the LOCA accident were the primary source of the stresses in the thimble guide tube and fuel assembly nozzles. As a result of faulted condition transient loading, the induced stresses in a fuel rod are generally very low. They were caused by bending due to the fuel assembly deflections during the seismic accident. A summary of the LOCA induced stresses, expressed in terms of a percentage of the allowable stress limits, for the fuel assembly major components is given in Table 3.

TABLE 3

FUEL ASSEMBLY COMPONENT STRESSES FOR LOCA ACCIDENT

(Percent of Allowable)

Component	Uniform Stresses (Membrane/Direct)	Combined Stresses (Membrane + Bending)
Thimble	45.5	53.2
Fuel Rod*	13.8	13.9
Top Nozzle Plate	1.0	8.3
Bottom Nozzle Plate	1.0	31.6

----

*Includes primary operating stress

The fuel assembly component stresses which result from the vertical effects of the LOCA accident were directly combined with the seismic induced stresses and a summary of the combined stresses is given in Table 4.

Table 4

FUEL ASSEMBLY COMPONENT STRESS FOR COMBINED SEISMIC/LOCA ACCIDENT

(Percent of Allowable)

Component	Uniform Stresses (Membrane/Direct)	Combined Stresses (Membrane + Bending)
Thimble	46.3	57.9
Fuel Rod*	13.9	14.6
Top Nozzle Plate	1.0	8.3
Bottom Nozzle Plate	1.0	31.6

----

*Includes primary operating stress

CONCLUSIONS

Based on the grid impact forces and fuel assembly component stress margins, it is concluded that the Westinghouse 15x15 optimized fuel assemblies will remain coolable for seismic or LOCA accidents.

REFERENCES

1. WCAP 8236, "Safety Analysis of 17x17 Fuel Assembly for Combined Seismic and Loss of Coolant Accident", L.T. Gesinski.

2. WCAP 9401, "Verification Testing and Analysis of the 17x17 Optimized Fuel Assembly", M.D. Beaumont, et.al.

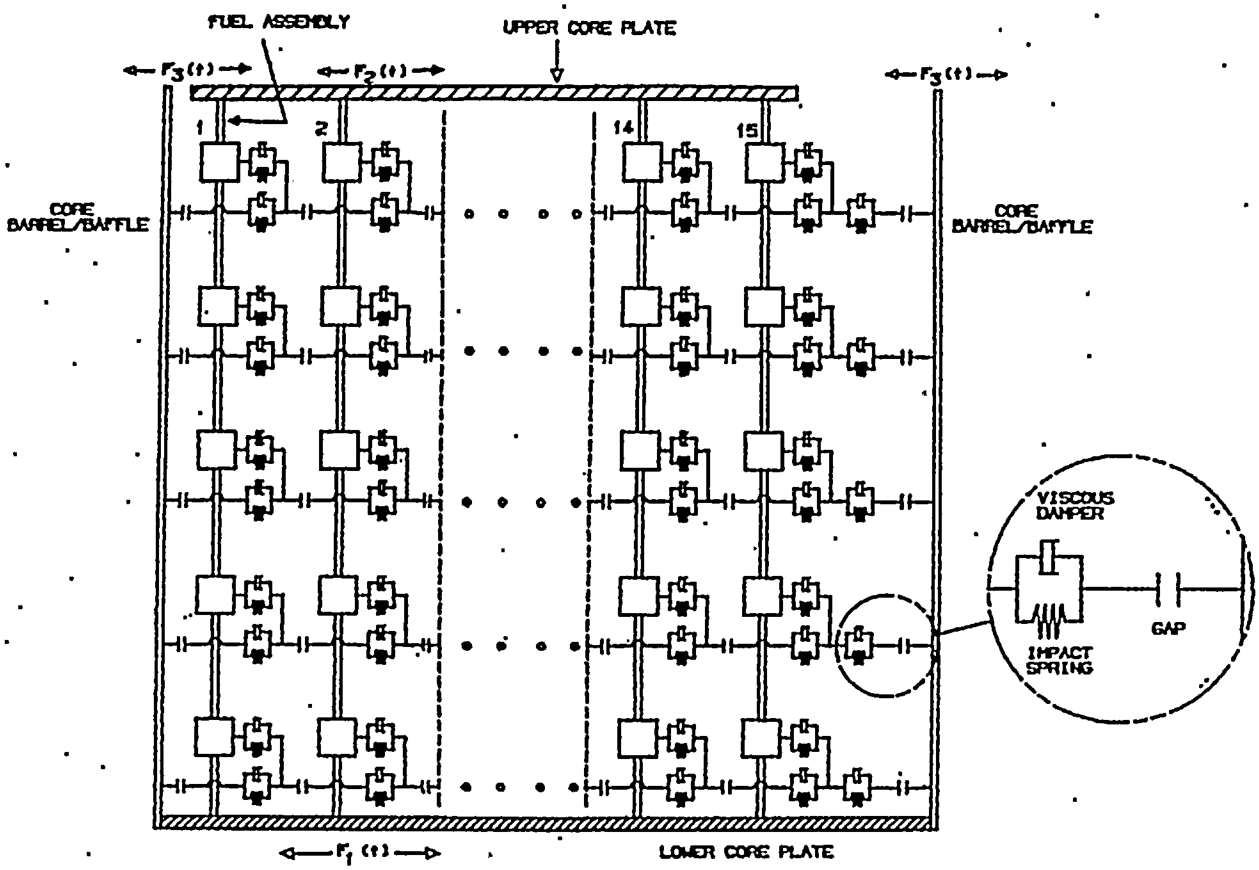


FIGURE 1 SCHEMATIC REPRESENTATION OF REACTOR CORE FINITE ELEMENT MODEL

CORE BARREL/BAFFLE

CORE BARREL/BAFFLE

FA. NO.

CASE	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	CORE
1	W	E	E	E	E	E	E	E	E	E	E	E	E	E	W	TRANSITION CORE
2	W	E	E	E	W	W	W	E	W	W	W	E	E	E	W	TRANSITION CORE
3	W	E	W	E	W	E	W	E	W	E	W	E	W	E	W	TRANSITION CORE
4	W	W	W	W	W	W	W	W	W	W	W	W	W	W	W	HOMOGENEOUS CORE

LEGEND:

E - EXXON FUEL ASSEMBLY

W - WESTINGHOUSE FUEL ASSEMBLY

FIGURE 2 REACTOR CORE FUEL ASSEMBLY RELOAD PATTERN

Attachment E to AEP:NRC:0745H

Response to Questions on WCAP-9500 Methods and Design Criteria

Q.1 Referring to Sections 4.2.1.1 to 4.2.1.3 of the NRC SER on WCAP-9500, confirm that the design acceptance criteria stated in these SER sections are satisfied for the Cook Unit 1 15x15 optimized fuel.

Response

The design acceptance criteria are satisfied except for Section 4.2.1.3(d) of the SER. The Cook Unit 1 licensing bases does not require the combining of seismic and LOCA forces during a safe shutdown earthquake event. Additional information on seismic/LOCA is provided in the response to Question 3, Part (c).

Q.2 Referring to Sections 4.2.3.1 to 4.2.3.3 of the NRC SER on WCAP-9500, confirm that approved methods were used for the Cook Unit 1 optimized fuel. Note and justify changes to approved methods given in WCAP-9500 which were used for the Cook Unit 1 fuel.

Response

Approved methods noted in the SER sections are used for the Cook Unit 1 fuel with the following clarifications or exceptions:

1. With respect to Section 4.2.3.2(d), the revised PAD fuel thermal safety model (WCAP-8720, Addendum 2) has been used in the safety analyses of all non-LOCA transients. In a July 29, 1983 letter to the NRC the following is stated, "In all cases, it was determined that the use of the fuel temperatures predicted by the revised PAD model has a slight impact on the non-LOCA safety analyses and the appropriate design bases are still met."

2. With respect to Sections 4.2.3.2(f) and 4.2.3.3(c), cladding rupture, cladding ballooning and flow blockage during LOCA incidents are accommodated in the Cook Unit 1 analyses by use of the approved 1981 large break ECCS evaluation model (WCAP-9220-P-A). Approval of the 1981 ECCS model represents a generic resolution of the clad swelling and ballooning open items stated in the WCAP-9500 SER.

3. With respect to Section 4.2.3.3(d), the seismic and LOCA forces were not combined since this was not required by the Cook Unit 1 licensing bases. For additional information see the response to Q3, part (c).

Q.3 Per the NRC SER cover letter on WCAP-9500, provide the following plant specific information:

a) How was the rod bow penalty accounted for?

b) Confirm that the predicted clad collapse time exceeds the expected lifetime of the fuel.

c) Confirm that the appropriate seismic and LOCA forces on the fuel assemblies are within acceptable bounds.

d) What are the fuel surveillance plans?

<u>Response 3(a)</u>

The Westinghouse 15x15 OFA is assumed to have identical gap closure as the Westinghouse 15x15 LOPAR, since the parameters (fuel rod diameter, clad thickness, and grid spacing) used in analytically determining gap closure are identical. Thus, the NRC-approved full flow rod bow penalty(1) applied to Westinghouse 15x15 LOPAR is applicable to Westinghouse 15x15 OFA. This rod bow penalty is 12.5% DNBR. Sufficient margin between the safety analysis limit DNBR and the design limit DNBR is maintained to accommodate this penalty as

well as the transition core DNBR penalty. The additional penalty of 2.4% DNBR at loss-of-flow conditions is covered explicitly in the loss-of-flow analysis for Westinghouse 15x15 OFA.

Reference to 3(a) Response:

1. Stolz, J. F., NRC Letter to T. M. Anderson, Westinghouse, "Staff Review of WCAP-8691," April 5, 1979.

Response 3(b)

Clad flattening (collapse) calculations, performed using the NRC approved clad flattening model (WCAP-8377), confirm that clad flattening will not occur during the expected lifetime of the fuel. Predicted clad flattening time for D.C. Cook Unit 1 Region 10 fuel is in excess of 40000 EFPH.

Response 3(c)

Consistent with Section 4.2.3.3(d) of the SER for WCAP-9500 a plant specific analysis of several cases covering the D. C. Cook Unit 1 mixed-core configurations was done. The analysis demonstrates that grid impact forces on the OFA for the most limiting case are:

<60% of allowable load for LOCA and
<80% of allowable load for seismic

However, since the D. C. Cook Unit 1 original design basis does not combine grid impact forces due to response to the LOCA and seismic events, we have not combined grid impact forces in this analysis.

Analyses also show that the major fuel assembly component stresses are less than the allowable. Hence, it is concluded the Westinghouse 15x15 OFA will remain in a coolable configuration under the postulated LOCA or seismic events.

Response 3(d)

A routine fuel inspection program will be implemented on the irradiated and discharged optimized fuel from the initial reload region. The program will involve visual examinations on a representative sample of assemblies from the initial fuel region at each refueling until this fuel is discharged. Visual observations will include, but not be limited to, crud buildup, rod bowing, grid strap conditions and missing components. Additional fuel inspections would be performed depending on the results of operational monitoring, including coolant activity, and the visual fuel inspections.

Attachment F to AEP:NRC:0745H

Revised Peaking Factor Limit Report