UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 29, 1998

MEMORANDUM TO: Information and Records Management Branch
Office of Information Resources Management

FROM: Glenn B. Kelly, Senior Project Manager
Project Directorate 3-3, DRPW, NRR

Glenn B Kelly

SUBJECT: INFORMATION TO BE PLACED ON THE DUANE ARNOLD ENERGY CENTER (DAEC) DOCKET (50-331)

The enclosed information documents several faxes between the NRC staff and IES Utilities Inc. regarding DAEC responses to staff requests for information on Section 3.3 of the DAEC improved technical specification submittal. Please place the enclosed information on the DAEC docket.

Enclosures: as stated
Docket Nos. 50-331
cc: See next page

DF01/1

January 29, 1998

MEMORANDUM TO: Information and Records Management Branch
Office of Information Resources Management

FROM: Glenn B. Kelly, Senior Project Manager
Project Directorate 3-3, DRPW, NRR

SUBJECT: INFORMATION TO BE PLACED ON THE DUANE ARNOLD ENERGY CENTER (DAEC) DOCKET (50-331)

The enclosed information documents several faxes between the NRC staff and IES Utilities Inc. regarding DAEC responses to staff requests for information on Section 3.3 of the DAEC improved technical specification submittal. Please place the enclosed information on the DAEC docket.

Enclosures: as stated
Docket Nos. 50-331
cc: See next page

DISTRIBUTION:

Docket File	PUBLIC	G. Grant, DRP, RIII	EAdensam (EGA1)
PD3-3 RF	R. Savio	G. Kelly	ACRS
OGC	P.Ray	R. Laufer	

DOCUMENT NAME: G:\DUANEARN\97197COS.WPD
To receive a copy of this document, indicate in the box C=Copy w/o attachment/enclosure E=Copy with attachment/enclosure N = No copy

OFFICE	PM:PD33	E
NAME	GKelly\ GK	
DATE	01/29 /98	

OFFICIAL RECORD COPY

Duane Arnold Energy Center
3277 DAEC Rd.
Palo, IA 52324

IES UTILITIES

Fax Cover Sheet

To: Rich Laufer

Company: US NRC

Fax Number: (301) 415-3061

From: Tony Browning

Phone Number: (319) 851-7750

IES Fax Number: (319) 851-7364

Number of pages following this page: 4

Date/time: January 14, 1998 *7:30 AM*

Rich,

Attached are the UFSAR table on Diesel Generator loading sequence and simplified logic diagrams for the Emergency Bus Lo.ding Relays. This is a request from Carl Schulten and Ed Tomlinson for question 3.3.5.1 - 15. I don't know if we'll get to discuss this in this afternoon's call or not.

Any Questions, call me. Thanks.

Tony

For Question 3.3.5.1-15

Table 8.3-1 SINGLE DIESEL-GENERATOR LOADING SEQUENCE AND RESPONSE - LOSS-OF-COOLANT ACCIDENT PLUS LOSS OF OFFSITE POWER							
Elapsed Time from Initiation of Event	Load Increment (kW)	Total Load (kW)	Voltage (% of rated)	Recovery Time to 90% Rated Voltage (sec)	Frequency (% of rated)	Recovery Time to 98% Rated Frequency(sec)	Description
0 SEC							LOCA ACCIDENT
3 SEC							LOW REACTOR WATER LEVEL OR HIGH DRYWELL PRESSURE SIGNAL, EDG START SIGNAL
13 SEC			100		100		EDG CLOSES TO BUS, SIGNAL REACTOR RECIRC AND CONTAINMENT ISOLATION, LPCI AND CORE SPRAY INJECT VALVES OPEN
	13						MCC 1B91 LOADS
	135						MCC 1B34A, 1B44A LOADS
	345.1						MCC 1B32 LOADS
	114						MCC 1B34 LOADS
	104	711.1	73	1.2	96.7	3.07	RIVER WATER SUPPLY PUMP START*
18 SEC	612	1323.1	80	1.3	97.2	3.91	CORE SPRAY PUMP START
21 SEC							CORE SPRAY INJECT VALVE OPEN

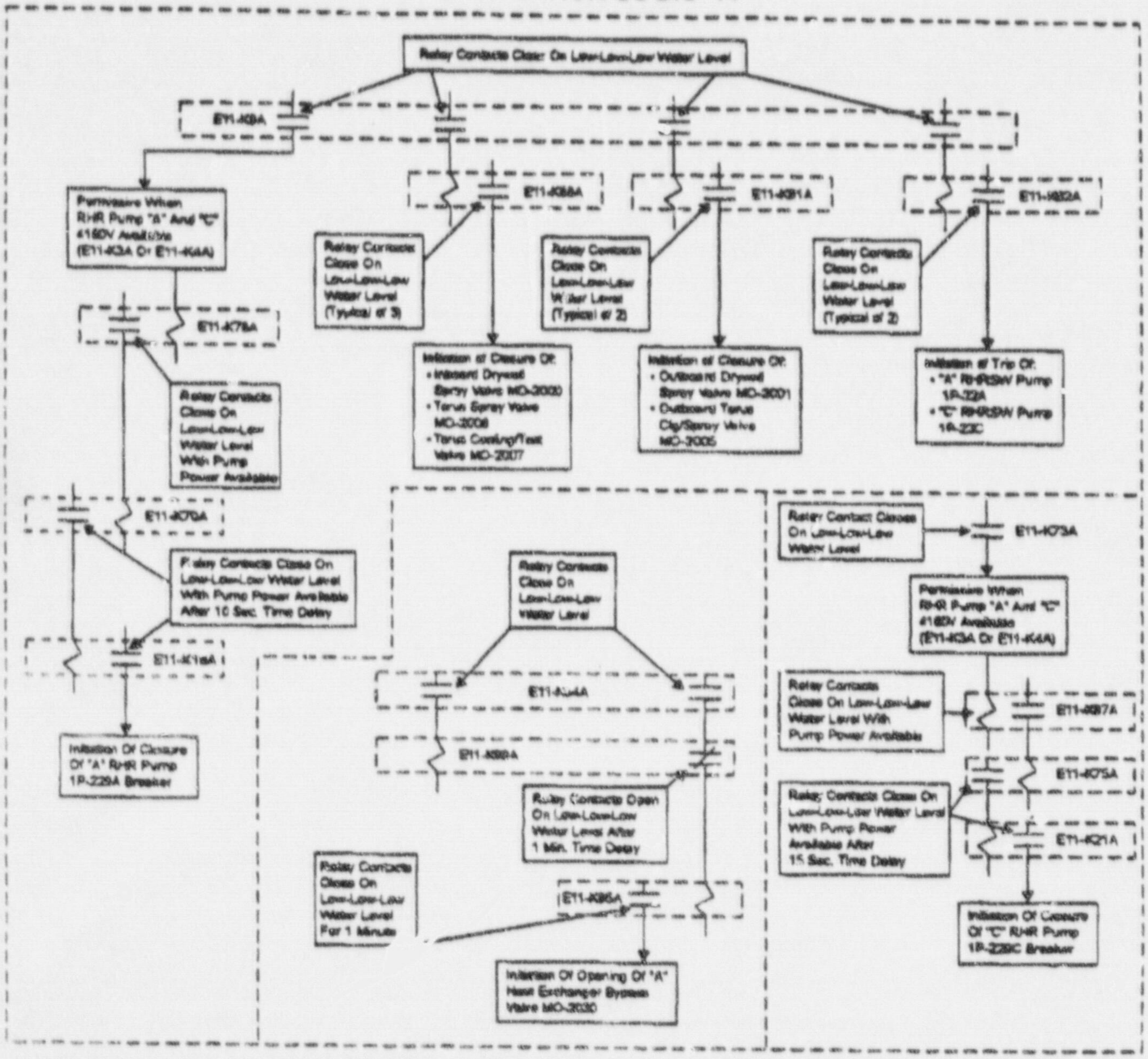
Table 8.3-1
SINGLE DIESEL-GENERATOR LOADING SEQUENCE AND RESPONSE -
LOSS-OF-COOLANT ACCIDENT PLUS LOSS OF OFFSITE POWER

23 SEC	496	1819.1	80	1.3	97.2	3.91	FIRST RHR (LPCI) PUMP START
28 SEC	496	2315.1	83	1.1	97.1	2.45	SECOND RHR (LPCI) PUMP START
31 SEC							LPCI VALVES OPEN
43 SEC							REACTOR RECIRC VALVES CLOSE, COMPLETION OF LPCI SEQUENCE
73 SEC	66	2381.1	--	--	--	--	CONTROL BUILDING CHILLER START
After 10 min*	-162	2219.1	--	--	--	--	MCC LOAD REDUCTION
	-496	1723.1	--	--	--	--	TRIP RHR PUMP
	470	2193.1	--	--	--	--	RHR SERVICE WATER PUMP START
	470	2663.1	--	--	--	--	RHR SERVICE WATER PUMP START

NOTE: * Additional loads may be applied at operator's discretion.
† If the RWS pump selected for auto start was running prior to the LOOP, the selected RWS pump will trip and will restart 120 seconds after the diesel generator output breaker closes.

LOW PRESSURE COOLANT INJECTION (LPCI) SYSTEM REACTOR VESSEL WATER LEVEL - LOW-LOW-LOW (2.a)

LPCI INITIATION LOGIC "A"



Minimum Channel Requirements for LPCI Mode of RHR Initiation Capability

In order to maintain the capability to initiate the LPCI Mode of RHR Loop "A" on the Reactor Water Level - Low-Low-Low Trip Function, the following combination of channels is required to be Operable. A channel in the tripped condition is considered to be Operable.

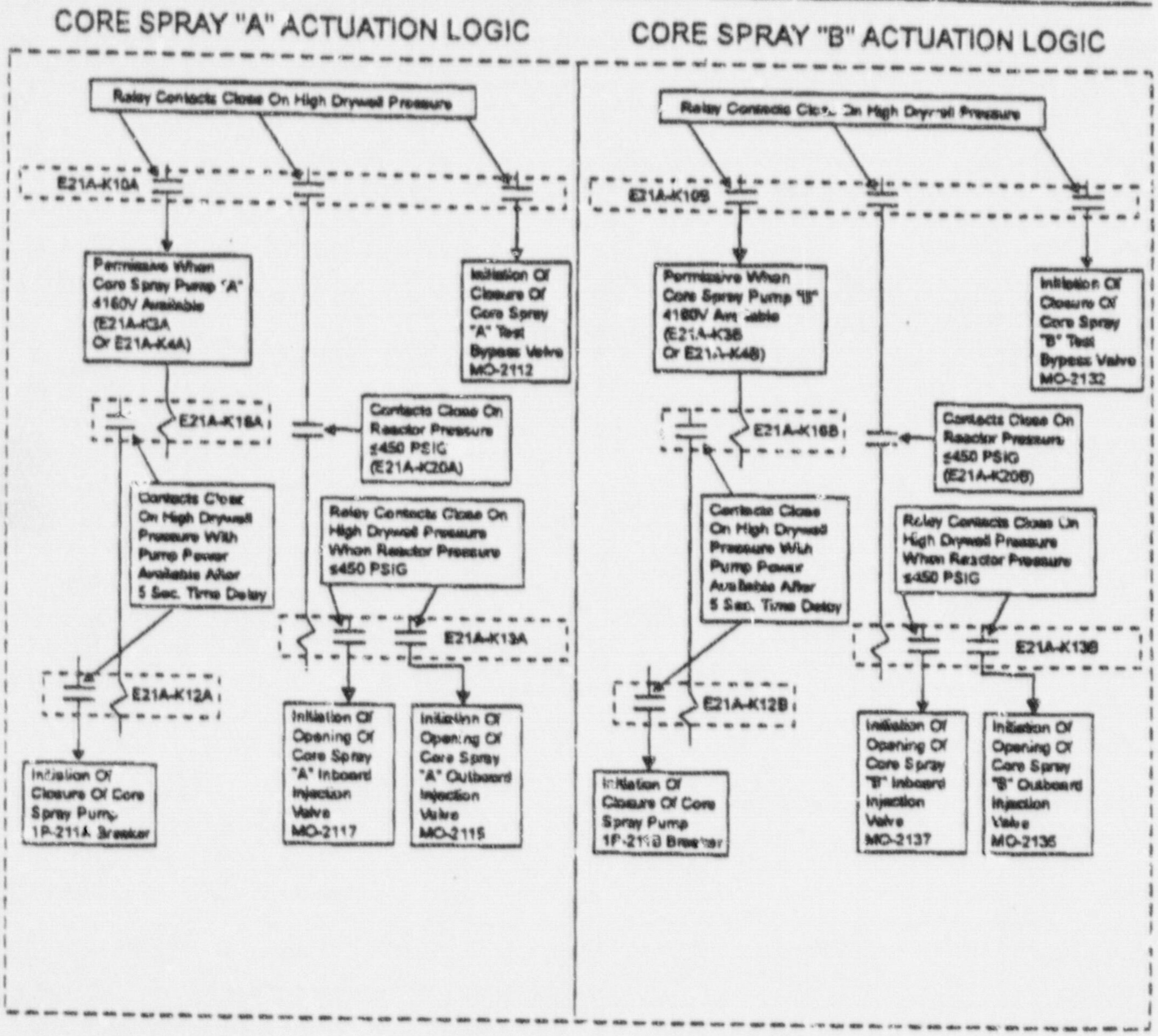
LIS-4531 OR LIS-4532

AND

LIS-4533 OR LIS-4534

ELEMENTARY REFFRENCES APED-E21-006 SHEETS 1, 2 APED-E11-007 SHEETS 3, 4, 5, 7, 8, 13, 15, 17	TABLE 3.3.5.1-1 EMERGENCY CORE COOLING SYSTEM INSTRUMENTATION LOW PRESSURE COOLANT INJECTION (LPCI) SYSTEM REACTOR VESSEL WATER LEVEL - LOW-LOW-LOW	LFD-ECCS-05 REV. 1, 06/17/96 SHEET 2 OF 3

CORE SPRAY SYSTEM DRYWELL PRESSURE - HIGH (1.b)



Minimum Channel Requirements for Core Spray Initiation Capability

In order to maintain the capability to initiate Core Spray System "A" or Core Spray System "B" on the Drywell Pressure - High Trip Function, the following combination of channels is required to be Operable. A channel in the tripped condition is considered to be Operable.

PS-4310B OR PS-4311B

AND

PS-4312B OR PS-4313B

ELEMENTARY REFERENCES APED-E21-006 SHEETS 1, 2, 4	TABLE 3.3.5.1-1 EMERGENCY CORE COOLING SYSTEM INSTRUMENTATION	LFD-ECCS-02 REV. 1, 06/17/96
	CORE SPRAY SYSTEM DRYWELL PRESSURE - HIGH	SHEET 2 OF 2

Duane Arnold Energy Center
32. DAEC Rd.
Palo, IA 52324

UTILITIES

Fax Cover Sheet

To: Rich Laufer

Company: US NRC

Fax Number: (301) 415-3061

From: Tony Browning

Phone Number: (319) 851-7750

IES Fax Number: (319) 851-7364

Number of pages following this page: 11

Date/time: January 14, 1998 7:30 AM

Rich,

Attached is a package of information regarding the Feedwater/Turbine Trip instruments. The first part is a short write-up that I did to try and explain our position and to clear up what appears to be a misunderstanding by the Staff over what our UFSAR says about this trip. In addition to UFSAR pages, I included a write-up from GE's fuel topical report (which is part of our licensing basis by reference in our UFSAR, TS and Core Operating Limits Report (COLR)) and the results of the Feedwater Controller Failure transient out of our reload analysis for this cycle's COLR.

As we discussed yesterday, we will try one more time to resolve this at our level, using this additional information for the Staff. Otherwise, we'll need to appeal this issue to a higher management level, as we will have great difficulty incorporating this LCO into our ITS on our timetable. First, Engineering tells us, if they drop everything else that they are doing, it will take them 4 manweeks to generate and verify the setpoint calculation for the Allowable Value for the calibration SR, as one does not currently exist. Second, what safety basis do I use to write my BASES from, as our analysis shows it doesn't meet Criteria 3 at the DAEC. "Because the NRC said so" doesn't work either. Lastly, as this is a more-restrictive change than the CTS, I think that we'd have trouble writing a convincing M-DOC that says that plant operation wouldn't be adversely affected, as we'd be adding LCO requirements and surveillances on instruments that we have not demonstrated improve plant safety or lower plant risk.

Any Questions, call me. Thanks.

Tony

Duane Arnold - FW/TT on RPV High Level

- Per UFSAR Section 7.7.1, this trip is not required for safety. It is an "Auxiliary Function" of the Feedwater Control System.

- There is no write-up in the UFSAR regarding the trip of the Main Turbine outside of the Chapter 15.1.1 write-up.

- With regard to the UFSAR 15.1.1 write-up, there are three main points to be made:

  1) The write-up does not say that the High Level trip mitigates the event. On the contrary, without the turbine trip (and ensuing pressurization transient), the consequences of the event are insignificant. (See attached figure 2 from current cycle's transient analysis) Power level (and ΔMCPR) have levelled out at a new steady state of only a couple of % above normal long before the turbine trip at the high level trip setpoint. (Note: see also write-up from GESTAR-II which describes this event in better detail

than the DAEC UFSAR. The impact on MCPR from the increased feedwater flow (and increase in core inlet subcooling that increases power level and lowers MCPR margin) is separate and distinct from the pressurization transient from the "eventual" turbine trip. The reactor trip is necessary to mitigate the turbine trip event, not the high water level trip of the turbine.

2) The UFSAR says that the high level trip prevents moisture carryover to the turbine. This is merely a design feature for equipment protection ~~feature~~ only, not a safety function, as corroborated by no discussion of this feature in the Turbine Control section of the UFSAR (section 7.7.2).

3) We need to point out a minor discrepancy in the existing UFSAR write-up. This high level trip setpoint for the feedwater/turbine trip is not in the current TS; only the high level trip for HPCI and RCIC. The HPCI and RCIC trip is safety-related and is provided by a different set of instruments and logic (ref. UFSAR 7.3.1.1.2.1 & 7.4.1.2).

7.7 CONTROL SYSTEMS NOT REQUIRED FOR SAFETY

This section discusses control systems whose functions are not essential for the safety of the plant. These systems are the feedwater control system, the turbine-generator controls, the reactor manual control system, and the process computer system.

7.7.1 FEEDWATER SYSTEM CONTROL AND INSTRUMENTATION

7.7.1.1 Power Generation Objective

The power generation objective of the feedwater control system is to maintain a preestablished water level in the reactor vessel during normal plant operation.

7.7.1.2 Power Generation Design Basis

The feedwater control system regulates the feedwater flow (1) to maintain adequate water level in the reactor vessel according to the requirements of the system operators and (2) to prevent the exposure of the reactor core over the power range of the reactor

7.7.1.3 System Description

During normal plant operation, the feedwater control system automatically regulates feedwater flow into the reactor vessel. The system can be manually operated. The feedwater control system includes the two main feedwater control valves and one feedwater startup control valve.

The feedwater flow control instrumentation measures the water level in the reactor vessel, the feedwater flow rate into the reactor vessel, and the steam flow rate from the reactor vessel. During normal operation, these three measurements are used in controlling feedwater flow.

The optimum reactor vessel water level is determined by the requirements of the steam separators. The separators limit water carry-over in the steam going to the turbines and limit steam carry-under in water returning to the core. The water level in the reactor vessel is normally maintained within ± 2 in. of the optimum level during normal operation. This control capability is achieved by comparing feedwater flow to the reactor vessel with the steam flow from the reactor vessel to provide an anticipatory level error signal. The feedwater flow is regulated by adjusting the feedwater control valves to deliver the required flow to the reactor vessel.

7.7.1.3.1 Reactor Vessel Water Level Measurement

Reactor vessel water level is measured by three identical, independent sensing instrument loops (Figure 7.7-1). A differential-pressure transmitter senses the difference between the pressure caused by a constant reference column of water and the pressure caused by the variable height of water in the reactor vessel. The differential-pressure transmitter is installed on lines that serve other systems (see Section 7.6.4). A total of three level differential pressure transmitters to each transmits a level signal to a level indicator and a level switch. Two of the level signals are selectable and the selected level signal is used to provide the level control function to the level controller. The selected signal also feeds a computer point, a level switch and a recorder. The signal of the non-selectable third level differential pressure transmitter only feeds an indicator and a level switch. three pressure transmitters feed three reactor vessel pressure indicators, respectively, in the control room. Signals from two of the three pressure transmitters are selectable and the selected signal is fed to a recorder and a computer point in the control room. The level signal from two of the three sensing systems can be selected by the operator as the signal to be used for feedwater flow control. The selected water level and the reactor vessel pressure signals are continually recorded in the control room.

7.7.1.3.2 Steam Flow Measurement

Steam flow is sensed at each main steam line flow restrictor by a differential-pressure transmitter equipped with square root functions. Signals from these differential-pressure transmitters are added to provide a linear signal proportional to the total steam flow rate. Individual steam line flow signals are indicated in the control room. The total steam flow signal is used for feedwater flow control, and is also recorded in the control room.

7.7.1.3.3 Feedwater Flow Measurement

Feedwater flow is sensed at a flow element in each feedwater line by differential-pressure transmitters. Each feedwater signal is linearized by square root converters. Then the individual mass flow signals are summed to provide a total mass flow signal for the feedwater flow control system. The total feedwater mass flow signal is also recorded in the control room.

In order to increase the reliability of feedwater flow indication, redundant flow measuring devices are installed on a local instrument rack in the turbine building.

The feedwater flow control system is a three-element control system. The three inputs are vessel level, feedwater flow and steam flow. The latter two constitute a flow mismatch that provides level error anticipation.

The setpoint of the master controller can be changed to a predetermined level by a manual push button at the operators discretion, and the controller will control the reactor level at that predetermined level automatically if the controller is in the Automatic Mode. The setpoint transfer function will not be in effect when the controller is in Manual Mode. This function is primarily designed to provide a convenient setpoint change for the operator during the execution of IPOI 5, immediate actions in response to a SCRAM.

Optional Automatic Operation

A single-element control signal (reactor vessel water level) can be used to replace the above three-element signal. In such cases, the operator switches the controller input to the "1 element" signal. The reactor level signal is fed to the level controller through a dynamic compensator and a converter and signal isolator. Reactor water level is then controlled by the reactor level signal in accordance with the controller setpoint.

Auxiliary Functions

Alarms are provided for high and low reactor water level and for high pressure. A loss of power signal to the feedwater control valve, Manual/Auto (M/A) stations, and the feedwater startup control valve, or a loss of service air supply to the feedwater startup control valve, or a loss of service air supply to the valves will cause the valves to lock up as is. Both power failure and low air pressure are annunciated. The feedwater startup control valve has annunciation in the control room for 90% or greater open. This annunciation indicates that the startup valve is approaching maximum flow and that action should be taken to transfer to one of the feedwater control valves. The level control system provides interlocks and control functions to other systems. When one out of two reactor feed pumps is lost and coincident or subsequent low water level exists, recirculation flow is limited to within the power capabilities of one reactor feed pump (see Section 7.7.5). This action aids in avoiding a low-level scram by limiting the steaming rate. Reactor recirculation flow is also limited on sustained low feedwater flow to ensure that adequate net positive suction head will be provided for the recirculation system.

Two-out-of-three narrow range vessel water level signals at the Hi trip setpoint will cause the feed pumps to trip. Controls to reset the trip are located on panel 1C05. The Reactor Feed Pumps (RFPs) High RPV Level Trip Defeat override may be used in support of the Emergency Operating Procedures (EOPs) in lieu of jumpers and lifted leads. This defeat allows restoration of the feed pumps for flooding above the normal level either in support of RPV Flooding Contingency or the Primary Containment Flooding Contingency. The single key-lock switch has an amber light and individually annunciates on front panel 1C-14 when taken to override.

15.1 EVENTS RESULTING IN A REACTOR VESSEL WATER TEMPERATURE DECREASE

Events that result directly in a reactor vessel water temperature decrease are those that either increase the flow of cold water to the vessel or reduce the temperature of water being delivered to the vessel. The three events that result in the most severe transients in this category are the following:

1. Feedwater controller failure - maximum demand.

2. Loss of a feedwater heater.

3. Inadvertent HPCI pump start.

15.1.1 FEEDWATER CONTROLLER FAILURE - MAXIMUM DEMAND

The failure of the feedwater controller in the direction of increased feedwater flow results in a moderator temperature decrease causing a reactor power increase through the effect of the negative reactivity void coefficient.

The transient response of the plant to a failure of the feedwater controller resulting in a demand for maximum feedwater flow is presented in Reference 3 of Section 15.0. The transient is initiated from the low end of the automatic recirculation flow control range, which produces a more severe steam flow/feed flow mismatch and level transient than that resulting at other initial conditions. The feedwater pumps are assumed to attain a maximum capability of 115% of rated flow.

Water level increases during the initial part of the transient. The high-water-level turbine trip is not initiated until the sensed [illegible] has increased to the preset value specified in the Technical Specification ([illegible] high-water-level trip also trips the high-pressure coolant injection and reactor core isolation cooling turbines and the feedwater pumps). The turbine trip prevents excessive moisture carryover to the turbine. Scram occurs consequential to the turbine trip, limiting the neutron flux peak and fuel thermal transient so that no fuel damage occurs.

The DAEC is analyzed for this event with the End-of-Cycle Recirculation Pump Trip (EOC-RPT) and both Turbine Bypass Valves (TBV) out-of-service. The thermal limit impact of these out-of-service conditions, or any combination thereof, are contained in the References 3 and 39 of Section 15.0. The Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) will assure power reduction with the EOC-RPT out of service. However, operation of ATWS-RPT does not impact the fuel thermal limit calculations for this event.

The turbine bypass valves open to limit the peak pressure in the steam line. The turbine bypass valves subsequently close, bringing the pressure in the reactor vessel under control during reactor shutdown.

15.1.2 LOSS OF A FEEDWATER HEATER

A feedwater heater can be lost in at least two ways: (1) if the steam extraction line to the heater is shut, the heat supply to the heater is removed, producing a gradual cooling of the feedwater; (2) a bypass line is usually provided so that the feedwater flow can be passed around rather than through the heater. In either case, the reactor vessel receives cooler feedwater, which produces an increase in core inlet subcooling. Because of the negative void reactivity coefficient, an increase in core power results.

Reference 3 of Section 15.0 describes the response of the plant to the loss of 100°F of the feedwater hating capability of the plant. This represents the maximum expected single heater (or group of heaters) which can be tripped or bypassed by a single event. The reactor is assumed to be at turbine-generator design conditions on manual recirculation flow control when the heater is lost. The feedwater flow delay time of approximately 25 sec between the heaters and the feedwater sparger is neglected. The plant would continue at steady-state conditions during this delay period. Neutron flux increases above the initial value to produce turbine design steam flow with the higher inlet subcooling. However, a significant amount of the energy from the subsequent neutron power increase would be utilized in heating the subcooled inlet water. Therefore, thermal power would increase at a very moderate rate. If power exceeded the normal full-power flow control line, the operator would be expected to insert control rods to return the power and flow to their normal range. If this were not done, the neutron flux could exceed the scram setpoint where a scram would occur. Because nuclear system pressure remains essentially constant during this transient, the nuclear system process barrier is not threatened by high internal pressure.

15.1.3 INADVERTENT HPCI PUMP START

Several systems are available for providing high pressure cold water to the vessel. An inadvertent start of the largest of these supplies, the 3,000 gpm HPCI system, produces the most severe transient of this type.

The HPCI system is inadvertently started and cold water is introduced at the feedwater spargers. Injection of this auxiliary supply of water with the feedwater flow will cause reactor water level to increase. The feedwater controller will compensate by decreasing feedwater flow. Introduction of this cold water into the reactor will increase core inlet subcooling which reduces the void fraction. The negative void reactivity coefficient will induce an increase in reactor power. Reactor power will increase and may reach the high flux scram setpoint. The power increase will generate a corresponding increase in steam flow, feedwater flow, and system pressure.

GE Nuclear Energy

24A5369
Revision 0
Class I
September 1996

Supplemental Reload Licensing Report for Duane Arnold Energy Center Reload 14 Cycle 15

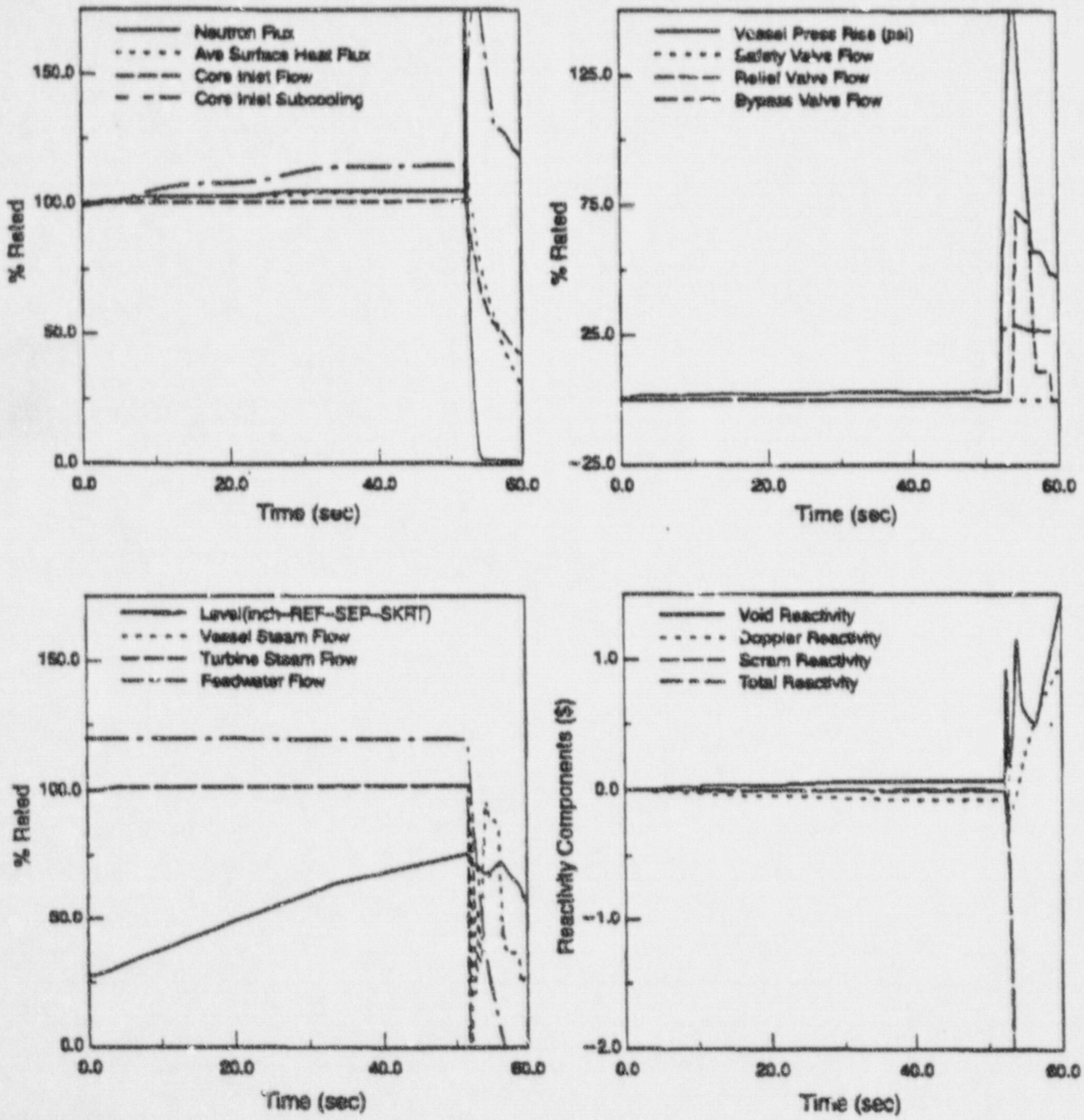


Figure 2 Plant Response to FW Controller Failure (BOC15 to EOC15 STANDARD)