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

JUN 8 1982

MEMORANDUM FOR: W. Houston, Assistant Director for Radiation Protection, DSI
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T. Speis, Assistant Director for Reactor Safety, DSI

FROM: Robert L. Tedesco, Assistant Director for Licensing, DL
Gus C. Lainas, Assistant Director for Safety Assessment, DL

SUBJECT: IP REQUEST FOR EXPERT REVIEW OF KRSKO PLANT CHANGES

In our memorandum of May 28, 1982 we provided each of you with the inital information received from IP related to the Krsko plant changes necessitated by the recent steam generator problems. We have recently received the enclosed additional information from IP:

(1) Memorandum, J. Dular, NEK to S. Smith, U. S. Embassy, Belgrade.
(2) Operating procedures for counter-flow preheat steam generator main feedwater bypass system with concurrent feedwater flow, April 1982.
(3) Minimization of counter-flow preheat steam generator preheater pressure transients system description, April 1982.
(4) Draft FSAR changes.
(5) Krsko feedwater system modification for split flow operation.

Please use the enclosed information in completing the June 25, 1982 milestone discussed in our May 28, 1982 memorandum. W. Kane at x27050 is the project manager for this activity.

Frank J. Miraglia
for Robert L. Tedesco, Assistant Director
for Licensing
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T. Ippolito /for
Gus C. Lainas, Assistant Director
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DRAFT FSAR CHANGES

TABLE 5.1-1

REACTOR COOLANT SYSTEM DESIGN AND OPERATING PARAMETERS FOR NORMAL STEADY-STATE FULL POWER OPERATING CONDITIONS

Nominal Operating Pressure, psig	2235
Total System Volume (including pressurizer and surge line), cu. ft.	6423
System Liquid Volume (including pressurizer water at maximum guaranteed power), cu. ft.	6011
Pressurizer Heater Capacity, kW	1,000
Pressurizer Relief Tank Volume, cu. ft.	1,100

System Thermal and Hydraulic Data
(Based on Thermal Design Flow)

Total Primary Heat Output, MWt	1882
Thermal Design Flows, gpm	
Loop	94,500
Reactor	189,000
Total Reactor Flow, 10^6 lb/hr	~~71.1~~ 70.9
Temperatures, °F	
Reactor Vessel Outlet	~~615.9~~ 617.5
Reactor Vessel Inlet	~~549.5~~ 551.5
Steam Generator Steam	~~535.1~~ 534.6
Feedwater	430.0
Steam Pressure, psia	920
Total Steam Flow, 10^6 lb/hr	8.17
Best Estimate Flows, gpm	
Loop	~~101,400~~ 100,700
Reactor	~~[illegible]~~ 201,600
Mechanical Design Flows, gpm	
Loop	~~106,500~~ 104,700
Reactor	~~213,030~~ 209,400

5.6 INSTRUMENTATION APPLICATION

Process control instrumentation is provided for the purpose of acquiring data for the key process parameters of the reactor coolant system (including the reactor coolant pump motors) as well as for the residual heat removal system. The pick-off points for the reactor coolant system are shown in the three sheets of the flow diagrams (Figure 5.1-1); and for the residual heat removal system, in flow diagram Figure 5.5-4. In addition to providing input signals for the protection system and the plant control systems, the instrumentation sensors furnish input signals for monitoring and/or alarming purposes for the following parameters:

1. Temperatures
2. Flows
3. Pressures
4. Water levels

In general the~~se~~ir input signals are used for the following purposes:

1. Provide input to the reactor trip system for reactor trips as follows:

   a. Overtemperature ΔT
   b. Overpower ΔT
   c. Low pressurizer pressure
   d. High pressurizer pressure
   e. High pressurizer water level
   f. Low primary coolant flow

   It is noted that the following parameter~~s~~, which ~~are~~ is also sensed to generate an input to the reactor trip system, while not part of the reactor coolant system, ~~are~~ is included here for purposes of completeness:

~~8. Low feedwater flow~~

9 ~~h~~. Low low steam generator water level

2. Provide input to the engineered safety features actuation system as follows:

   a. Pressurizer low pressure

   It is noted that the following parameters, which are also sensed to generate an input to the engineered safety features actuation system, while not part of the reactor coolant system, are included here for purposes of completeness:

   b. Low steam line pressure

   c. Hi-Hi steam flow or High steam flow coincident with low-low (T_{avg})

   d. Hi-1 containment pressure

   e. Hi-2 containment pressure

   f. Hi-3 containment pressure

3. Furnished input signals to the nonsafety-related system, such as the plant control systems and surveillance circuits so that:

   a. Reactor coolant average temperature (T_{avg}) will be maintained within prescribed limits. The resistance temperature detector instrumentation is identified on Figure 5.1-1, Sheet 3.

Thus an analysis of smaller pump suction breaks is representative of the spectrum of break sizes.

The LOCA analysis calculational model is typically divided into three phases which are: 1) blowdown, which includes the period from accident occurrence (when the reactor is at steady state full power operation) to the time when zero break flow is first calculated, 2) refill, which is from the end of blowdown to the time the ECCS fills the vessel lower plenum, and 3) reflood, which begins when water starts moving into the core and continues until the end of the transient. For the pump suction break, consideration is given to a possible fourth phase; that is, froth boiling in the steam generator tubes after the core has been quenched. For a description of the calculational model used for the mass and energy release analysis, see Reference 20.

Basis of the Analysis

1. Assumptions

The following items ensure that the core energy release is conservatively analyzed for maximum containment pressure.

a. Maximum expected operating temperature (617.5 ~~616.1~~°F)

b. Allowance in temperature for instrument error and dead band (+4°F)

c. Margin in volume (1.4%)

d. Allowance in volume for thermal expansion (1.6%)

e. Margin in core power associated with use of engineered safeguards design rating (ESDR)

f. Allowance for calorimetric error (2% of ESDR) prevent spurious trips caused by short term voltage perturbations. The coincidence logic and interlocks are given in Table 7.2-1.

d. Reactor coolant pump bus underfrequency trip

This trip is required to protect against low flow resulting from bus underfrequency, for example a major power grid frequency disturbance. The function of this trip is to trip the reactor for an underfrequency condition. The setpoint of the underfrequency relays is adjustable between 44 and 49 Hz.

There are two underfrequency sensing relays connected to each reactor coolant pump bus. Signals from relays connected to the buses (time delayed up to approximately 0.1 seconds to prevent spurious trips caused by short term frequency perturbations) will trip the reactor if the power is above P-7.

Figure 7.2-1, Sheet 5, shows the logic for the Reactor Coolant System low flow trips.

~~5. Steam Generator Trips~~

~~The specific trip functions generated are as follows:~~

~~a. Low feedwater flow trip~~

~~This trip protects the reactor from a sudden loss of the heat sink. The trip is actuated by steam/feedwater flow mismatch (one out of two) in coincidence with low water level (one out of two) in any steam generator.~~

~~Figure 7.2-1, Sheet 7, shows the logic for this trip function.~~

~~There are no interlocks associated with this trip.~~

5. ~~b.~~ Low-low steam generator water level trip

This trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch of insufficient magnitude to cause a low feedwater flow reactor trip. This trip is actuated on two out of three low-low water level signals occurring in any steam generator.

The logic is shown on Figure 7.2-1, Sheet 7.

6. Turbine Trip - Reactor Trip (anticipatory)

The turbine trip-reactor trip is actuated by two-out-of-three logic from emergency trip fluid system signals or by all closed signals from the turbine steam stop valves. A turbine trip causes a direct reactor trip above P-7. The reactor trip on turbine trip provides additional protection and conservatism beyond that required for the health and safety of the public. This trip is included as part of good engineering practice and prudent design. No credit is taken in any of the safety analyses (Chapter 15) for this trip.

The turbine provides anticipatory trips to the reactor protection system from contacts which change position when the turbine stop valves close or when the turbine emergency trip fluid system pressure goes below its setpoint.

3. Protection System ranges

Protection system ranges are tabulated in Table 7.2-3. Range selection for the instrumentation covers the expected range of the process variable being monitored during power operation. Limiting setpoints are at least 5 percent from the end of the instrument span.

7.2.1.3 Final System Drawings

Functional block diagrams are furnished in Figure 7.2-1 (Sheets 1-15) and additional drawings for the I&C systems are included at the end of sections 7.2, 7.6, 7.6 and in the referenced topical reports. See Table 7.3-6 for additional references.

7.2.2 ANALYSES

7.2.2.1 Failure Mode and Effects Analyses

A failure mode and effects analysis of the Reactor Trip System has been performed. Results of this study and a fault tree analysis are presented in Reference [4].

7.2.2.2 Evaluation of Design Limits

While most setpoints used in the Reactor Protection System are fixed, there are variable setpoints, most notably the overtemperature ΔT and overpower ΔT setpoints. All setpoints in the Reactor Trip System have been selected on the basis of engineering design and safety studies. The capability of the Reactor Trip System to prevent loss of integrity of the fuel cladding and/or Reactor Coolant System pressure boundary during Condition II and III transients is demonstrated in the Safety Analysis, Chapter 15. These safety analyses are carried out using those setpoints determined from results of the engineering design studies. Setpoint limits are presented in the Technical Specifications. A discussion on the intent for each of the various reactor trips and the accident analysis (where appropriate) which utilizes this trip is presented be[illegible]. It should be noted that the selected trip setpoints all provide for margin before protective action is actually required to allow for uncertainties and instrument errors. The design meets the requirements of Criteria 10 and 20 of the 1971 GDC.

7.2.2.2.1 Trip Setpoint Discussion

It has been pointed out previously that below a DNB ratio of 1.3 there is likely to be significant local fuel cladding failure. The DNB ratio existing at any point in the core for a given core design can be determined as a function of the core inlet temperature, power output, operating pressure and flow. Consequently, core safety limits in terms of a DNBR equal to 1.30 for the hot channel can be developed as a function of core ΔT, T_{avg} and pressure for a specified flow as illustrated by the solid lines in Figure 7.2-~~1a~~3. Also shown as solid lines in Figure 7.2-~~1a~~3 are the loci of conditions equivalent to 118 percent of power as a function of ΔT and T_{avg} representing the overpower (KW/ft) limit on the fuel. The dashed lines indicate the maximum permissable set point (ΔT) as a function of T_{avg} and pressure for the overtemperature and overpower reactor trip. Actual values of setpoint constants in the equation representing the dashed lines are as given in the Technical Specification, Section 16.2.3. These values are conservative to allow for instrument errors. The design meets the requirements of Criteria 10, 15, 20 and 29 of the 1971 GDC.

DNBR is not a directly measurable quantity; however, the process variables that determine DNBR are sensed and evaluated. Small isolated changes in various process variables may not individually result in violation of a core safety limit; whereas the combined variations, over sufficient time, pressurizer water level control. A failure in the level control system could fill or empty the pressurizer at a slow rate (on the order of half an hour or more), which allows ample time for corrective action by the operator.

The high water level trip setpoint provides sufficient margin such that the undesirable condition of discharging liquid coolant through the safety valves is avoided. Even at full power conditions, which would produce the worst thermal expansion rates, a failure of the water level control would not lead to any liquid discharge through the safety valves. This is due to the automatic high pressurizer pressure reactor trip actuating at a pressure sufficiently below the safety valve setpoint, or to the high pressurizer water level reactor trip.

7.2.2.3.5 Steam Generator Water Level and Feedwater Flow

The basic function of the reactor protection circuits associated with low steam generator water level ~~and low feedwater flow~~ is to preserve the steam generator heat sink for removal of long term residual heat. Should a complete loss of feedwater occur, the reactor would be tripped on ~~coincidence of steam/feedwater flow mismatch and low steam generator level or on~~ low-low steam generator water level. In addition, redundant auxiliary feedwater pumps are provided to supply feedwater in order to remove residual heat from the reactor.

Th~~ese~~ is reactor trip~~s~~ acts before the steam generators are dry to reduce the required capacity and increase the time available for starting these auxiliary feedwater pumps and to minimize the thermal transient on the Reactor Coolant System and steam generators. Therefore, the following reactor trip circuit~~s~~ ~~are~~ is provided for each steam generator to ensure that sufficient initial thermal capacity is available in the steam generator at the start of the transient:

~~1.~~ ~~The low feedwater flow trip detects a mismatch in steam and feedwater flow (one-out-of-two) coincident with low steam generator water levels for a steam generator in any loop.~~

~~2.~~ A low-low steam generator water level regardless of steam - feedwater flow mismatch;

It is desirable to minimize thermal transients on a steam generator for credible loss of feedwater accidents. Hence, it should be noted that controller malfunctions caused by a protection system failure effect only one steam generator; ~~the steam generator level signal used in the feedwater control originates separately from that used in the low feedwater reactor trip.~~

~~A spurious high signal from the feedwater flow channel being used for control would cause a reduction in feedwater flow preventing that channel from ultimately tripping. However, the mismatch between steam demand and feedwater flow produced by this spurious signal will actuate alarms to alert the operator of this situation in time for manual correction or, if the condition is allowed to continue, the reactor will eventually trip on a low-low water level signal independent of indicated feedwater flow.~~

A spurious low signal from the feedwater flow channel being used for control would cause an increase in feedwater flow. The mismatch between steam flow and feedwater flow produced by the spurious signal would actuate alarms to alert the operator of the situation in time for manual correction. If the condition continues, a two out of three high-high steam generator water level signal in any loop, independent of the indicated feedwater flow, will cause main feedwater pump trip and isolation and trip the turbine. The turbine trip will result in a subsequent reactor trip. The High-High Steam Generator Water Level trip is an equipment protective trip preventing excessive moisture carryover which could damage the turbine blading.

In addition, the three element feedwater controller incorporates reset action on the level error signal, such that with expected controller settings a rapid increase or decrease in the flow signal would cause only a small change in level before the controller would compensate for the level error. A slow change in the feedwater signal would have no effect at all. A spurious low or high steam flow signal would have the same effect as high or low feedwater signal, discussed above.

A spurious high steam generator water level signal from the protection channel used for control will tend to close the feedwater valve. However, before a reactor trip would occur, two out of three channels for a steam generator would have to indicate a high water level. A spurious low steam generator water level signal will tend to open the feedwater valve. Again, before a reactor trip would occur, two out of three channels in a loop would have to indicate a low water level. Any slow drift in the water level signal will permit the operator to respond to the level alarms and take corrective action. Automatic protection is provided in case the spurious high level reduces feedwater flow sufficiently to cause low level in the steam generator. The reactor will trip ~~either on low feedwater flow coincident with low water level or, ultimately,~~ on low-low steam generator water level. Automatic protection is also provided in case the spurious low level signal increases feedwater flow sufficiently to cause high level in the steam generator. A turbine trip and feedwater isolation would occur on two out of three high-high steam generator water level in any loop.

7.2.2.4 Additional Postulated Accidents

Loss of plant instrument air or loss of component cooling water is discussed in Section 7.3. Load rejection and turbine trip are discussed in further detail in Section 7.7.

TABLE 7.2-1 (CONTINUED) (Sheet 2 of 2)

LIST OF REACTOR TRIPS

Reactor Trip	Coincidence Logic	Interlocks	Comments
11. Low reactor coolant flow	2/3 per loop	Interlocked with P-7	Blocked below P-7
12. Reactor coolant pump breakers open	1/2 breakers, 1 breaker per bus	Interlocked with P-7	Blocked below P-7
13. Reactor coolant pump bus undervoltage	1/2 per bus on both buses	Interlocked with P-7	Low voltage on all buses permitted below P-7
14. Reactor coolant pump bus underfrequency	1/2 per bus on both buses	Interlocked with P-7	Under frequency on 2 buses will trip all reactor coolant pump breakers and cause reactor trip; reactor trip blocked below P-7
~~15. Low feedwater flow~~	~~1/2 per loop*~~	~~No interlocks~~	
~~16~~ 15. Low-low steam generator water level	2/3 per loop	No interlocks	
~~17~~ 16. Safety injection signal	Coincident with actuation of safety injection	No interlocks	(See Section 7.3 for Engineered Safety Features actuation conditions)
~~18~~ 17. Turbine-generator trip a) Low trip fluid pressure b) Turbine stop valve close	 2/3 2/2	Interlocked with P-7	Blocked below P-7
~~19~~ 18. Manual	1/2	No interlocks	

~~* 1/2 steam/feedwater flow mismatch in coincidence with 1/2 low steam generator water level.~~

TABLE 7.2-3 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

Reactor Trip Signal	Typical Range	Typical Trip Accuracy	Typical Time Response (sec)
11. Low reactor coolant flow	0 to 120% of rated flow	± 2.75 percent of full flow within range of 70 percent to 100 percent of full flow (1)	1.0
12. Reactor coolant pump bus undervoltage	0 to 100% rated voltage	±1 percent of rated voltage	1.2
13. Reactor coolant pump bus underfrequency	40 to 55 Hz	±0.1 Hz	0.6
~~14. Low feedwater flow~~	~~0 to 120% Max. Calc. feedwater flow~~	~~±6.5% (2)~~	~~2.0~~
~~15~~ 14. Low-low steam generator water level	± ~ 6 ft. from nominal full load water level	±2.3 percent of Δp signal over pressure range of 700 to 1200 psig	2.0
~~16~~ 15. Turbine Trip			1.0

NOTES FOR TABLE 7.2-3

(1) Reproducibility (see definitions in Sec 7.1)

~~(2) 1/2 steam/feedwater flow mismatch in coincidence with 1/2 low steam generator water level.~~

~~Channel accuracy of feedwater flow analog signal is ±2.5 percent of maximum calculated feedwater flow.~~

~~Accuracy of steam flow singal is ±3 percent of maximum calculated flow over the pressure range of 700 to 1200 psig.~~

(Sheet 4 of 5)

TABLE 7.2-4 (Continued)

REACTOR TRIP CORRELATION

	ACCIDENT[a]	TECH. SPEC.[b]
zer ssure	1) Uncontrolled Rod Cluster Control Assembly Bank Withdrawal At Power (15.2.2) 2) Loss of External Electrical Load and/or Turbine Trip (15.2.7)	16.2.3.3.2
rizer ater Trip	1) Uncontrolled Rod Cluster Control Assembly Bank at Power (15.2.2) 2) Loss of External Electrical Load and/or Turbine Trip (15.2.7)	16.2.3.3.3
Reactor ant Flow	1) Partial Loss of Forced Reactor coolant Flow (15.2.5) 2) Loss of Off-Site Power to the Station Auxiliaries (Station Blackout) (15.2.9) 3) Complete Loss of Forced Reactor Coolant Flow (15.3.4)	16.2.3.3.2
actor olant Pump eaker Trip	Not used nor credit taken for in any Accident Analysis; provided as additional feature to enhance safety	
eactor Coolant ump Bus Undervoltage Trip	1) Complete Loss of Forced Reactor Coolant Flow (15.3.4)	16.2.3.3.2
Reactor Coolant Pump Bus Underfrequency Trip	1) Complete Loss of Forced Reactor Coolant Flow (15.3.4)	16.2.3.3.2
~~Low Feedwater Flow Trip~~	~~1) Loss of Normal Feedwater (15.2.8)~~	~~See note c~~

TABLE 7.2-4 (Continued)

REACTOR TRIP CORRELATION

TRIP		ACCIDENT[a]	TECH. SPEC.[b]
~~17~~ 6)	Low-low Steam Generator Water Level Trip	1) Loss of Normal Feedwater (15.2.8)	16.2.3.3.3
~~18~~ 7)	Turbine Trip-Reactor Trip	1) Loss of External Electrical Load and/or Turbine Trip (15.2.7)	See note c.
		2) Loss of Off-Site Power to the Station Auxiliaries (Station Blackout) (15.2.9)	16.2.3.3.2
~~19~~ 8)	Safety Injection Signal Actuation Trip	1) Accidental Depressurization of the Main Steam System (15.2.13)	See note d.
~~20~~ 19)	Manual Trip	Available for all Accidents (Chapter 15)	See note c.

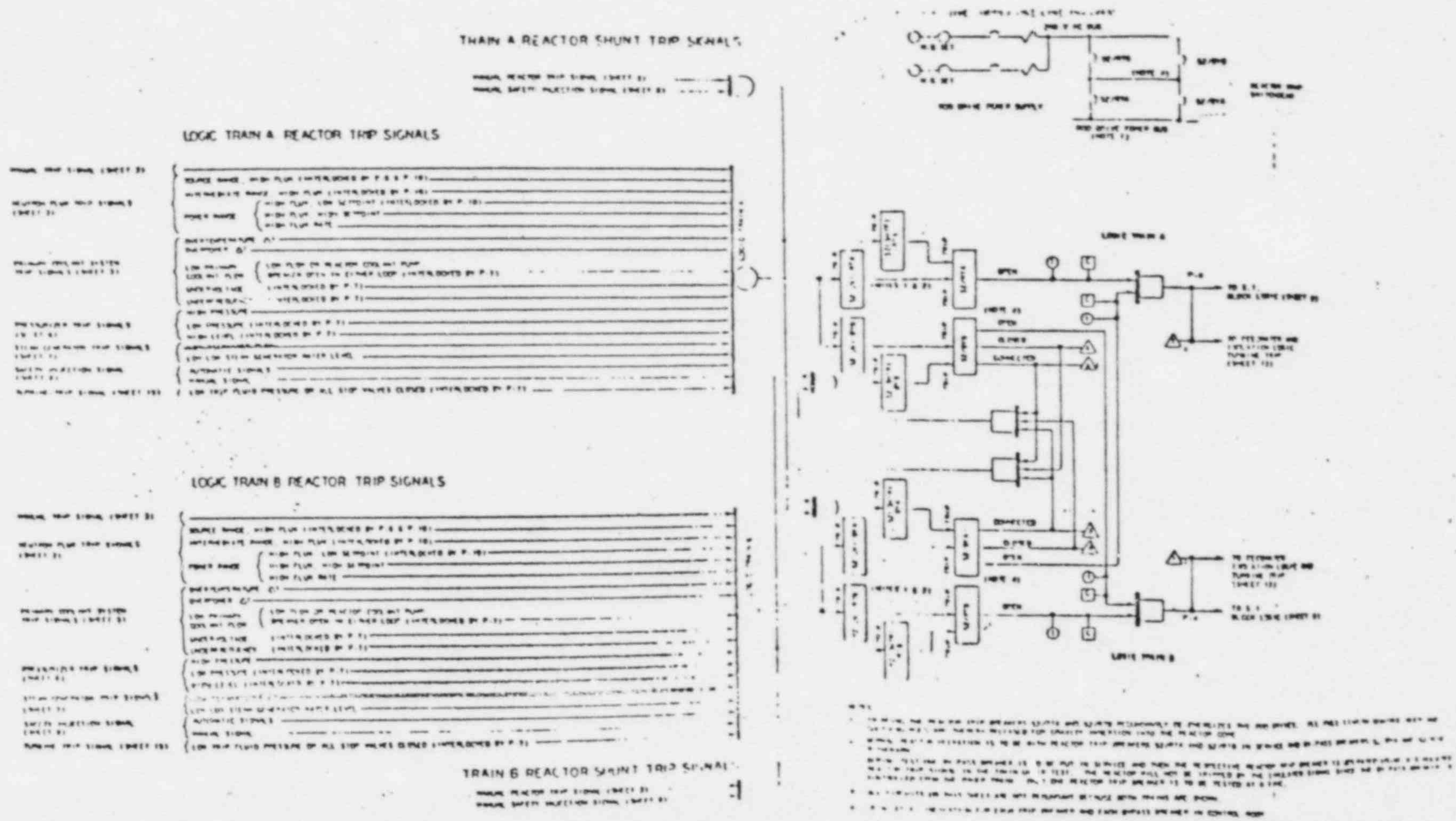
NOTES:

a. References refer to accident analyses presented in Chapter 15.

b. References refer to technical specifications presented in Chapter 16.

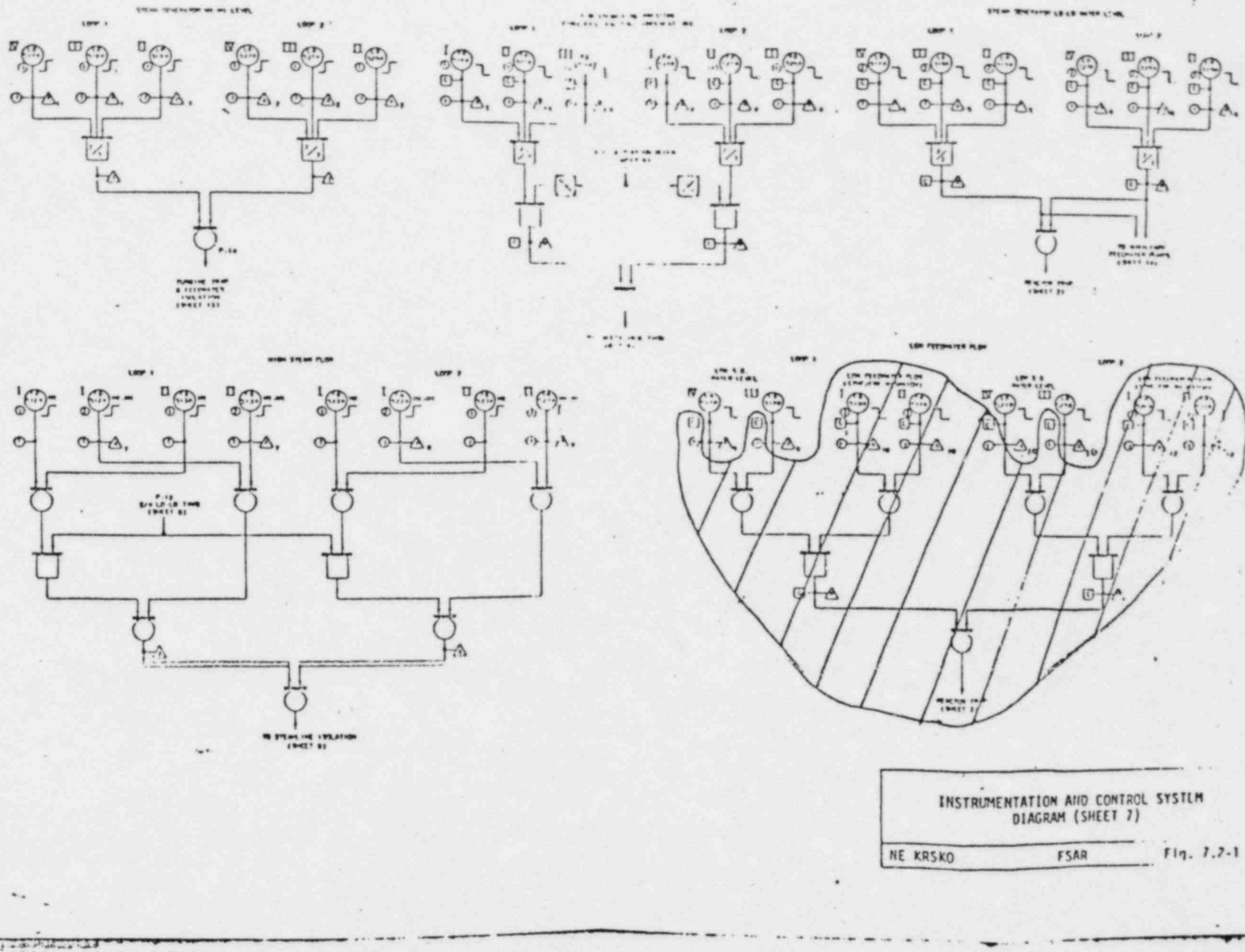
c. A technical specification is not required because this trip is not assumed to function in the accident analyses.

d. Accident assumes that the reactor is tripped at end of life (EOL) which is the worst initial condition for this case. Pressurizer low pressure-low level is the first out trip of Safety Injection.



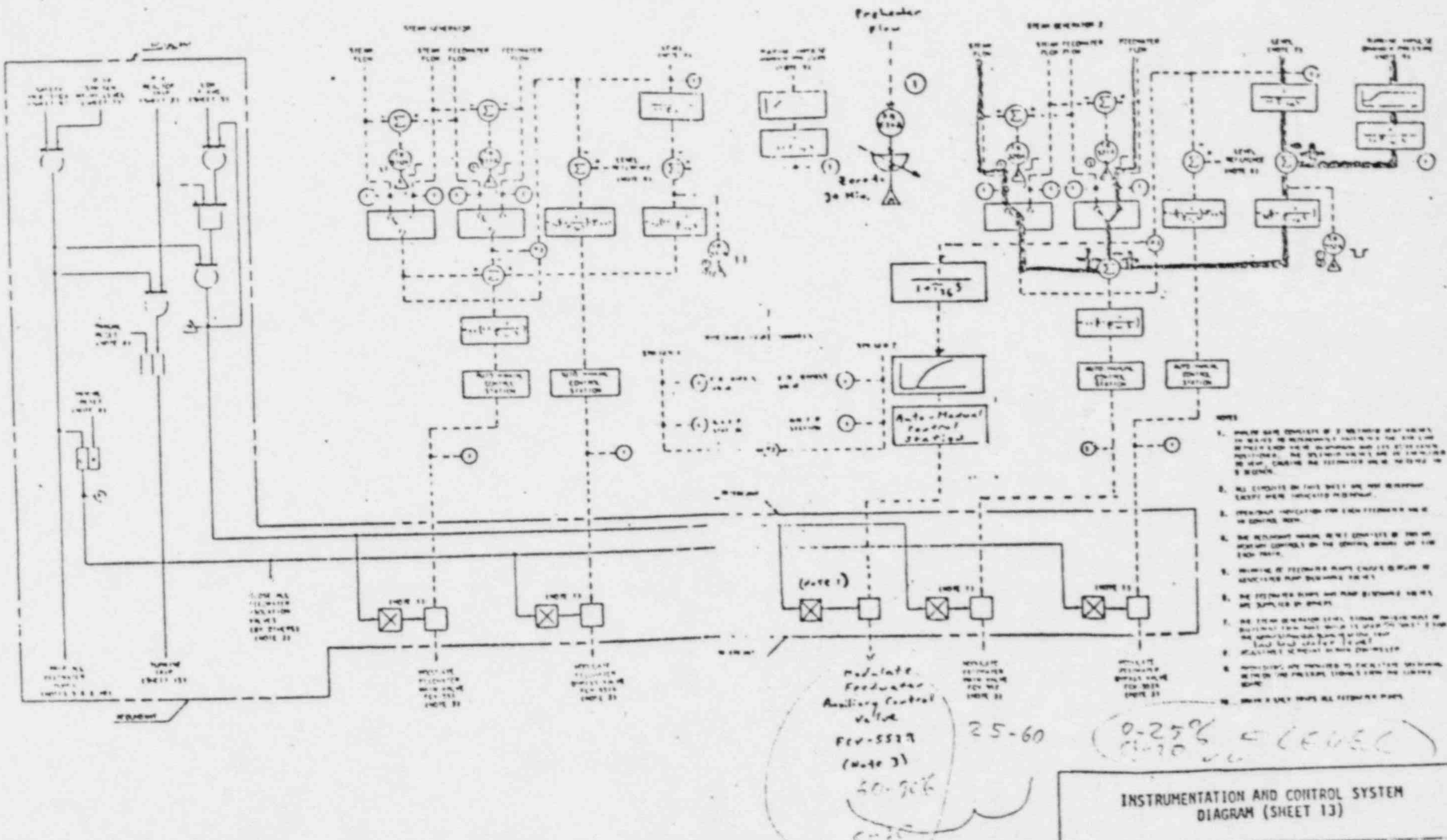
INSTRUMENTATION AND CONTROL SYSTEM
DIAGRAM (SHEET 2)

NE KRSKO FSAR Fig. 7.2-1



INSTRUMENTATION AND CONTROL SYSTEM
DIAGRAM (SHEET 7)

NE KRSKO	FSAR	Fig. 7.2-1



INSTRUMENTATION AND CONTROL SYSTEM DIAGRAM (SHEET 13)

NE KRSKO | FSAR | Fig. 7.2-1

7COW

The feedwater pumps are designed in accordance with the requirements of the Hydraulic Institute Standards. Design points for these pumps are selected to satisfy the requirements of the turbine thermal cycle at the maximum guaranteed condition plus margins for wear and surges. The feedwater pumps are also capable of maintaining steam generator water level during a load rejection and steam dump at 96 percent flow at a steam generator pressure of 972 psia.

High pressure feedwater heaters are designed, fabricated, inspected, tested and stamped in accordance with the ASME Code, Section VIII, Division 1. Thermal performance of these feedwater heaters is governed by Heat Exchange Institute Standards.

The feedwater system equipment parameters are listed in Table 10.4-2.

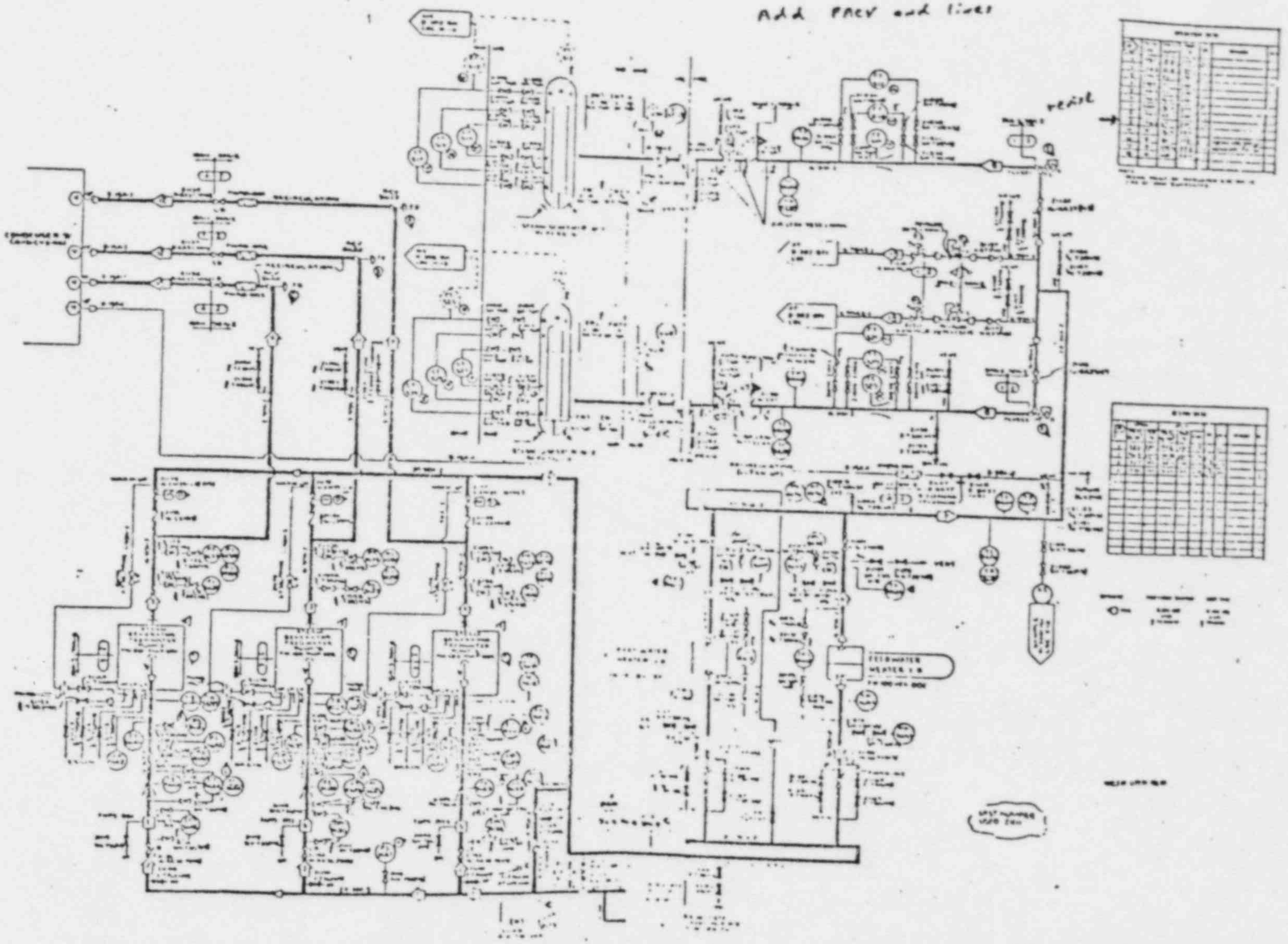
10.4.7.2.2 System Description

The feedwater system includes three 50 percent capacity motor driven main feed pumps, two parallel high pressure feedwater heaters and associated piping, valves and controls.

At loads above 15 percent, normal operating control is achieved by using a three element system consisting of inputs proportional to steam flow, feedwater flow and steam generator water level to control the position of the feedwater regulating valves. At loads of 15 percent and below, the steam generator level is maintained with the feedwater bypass control valve. The bypass control valve is automatically positioned with inputs proportional to the steam generator water level or the bypass control valve is manually positioned from the control room. These controls are always in operation except during a safety injection signal or a reactor trip coincident with low T_{avg}. The modulating signal is blocked in these cases.

add discussion on "60 percent and higher"

The main feed pumps are driven by a constant speed motor through speed increasing gearing. A low flow bypass to the main condenser is provided for use during startup.



FEEDWATER
NE KRSKO · FSAR · FIG. 10.1-3

plant and of the Reactor Coolant System. The overpower - overtemperature protection (neutron overpower, overtemperature and overpower ΔT trips) prevents any power increase which could lead to a DNBR less than 1.30.

One example of excess heat removal from the primary system is the transient associated with the accidental opening of the feedwater bypass valve which diverts flow around the low pressure feedwater heaters. In the event of an accidental opening of the bypass valve, there is a sudden reduction in feedwater inlet temperature to the steam generators. This increased subcooling will create a greater load demand on the Reactor Coolant System.

Another example of excessive feedwater flow would be a full opening of a feedwater control valve due to a feedwater control system malfunction or an operator error. At power this excess flow causes a greater load demand on the Reactor Coolant System due to increased subcooling in the steam generator. With the plant at no-load conditions the addition of cold feedwater may cause a decrease in Reactor Coolant System temperature and thus a reactivity insertion due to the effects of the negative moderator coefficient of reactivity. Continuous addition of excessive feedwater is prevented by the steam generator high-high level trip, which closes the feedwater valves.

15.2.10.2 Analysis of Effects and Consequences

15.2.10.2.1 Method of Analysis

The excessive heat removal due to a feedwater control valve malfunction transient is analyzed by using the detailed digital computer code ~~MARVEL~~ LOFTRAN[1]. This code simulates a multi-loop system, neutron kinetics, the pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

The system is analyzed to demonstrate plant behavior in the event of a feedwater bypass valve malfunction. Feedwater temperature reduction due to low pressure heater bypass valve actuation in conjunction with an inadvertent trip of the heater drain pump is considered.

Excessive feedwater addition due to a control system malfunction or operator error which allows a feedwater control valve to open fully is considered. Two cases are analyzed as follows:

1. Accidental opening of one feedwater control valve with the reactor just critical at zero load conditions assuming a conservatively large negative moderator temperature coefficient characteristic of end of core life conditions.

2. Accidental opening of one feedwater control valve with the reactor in automatic control at full power.

The reactivity insertion rate following a feedwater system malfunction is calculated with the following assumptions:

a. For the feedwater control valve incident at full power, one feedwater control valve is assumed to malfunction resulting in a step increase to 155 ~~120%~~ of nominal feedwater flow to one steam generator.

b. For the feedwater control valve accident at zero load condition, a feedwater control valve malfunction occurs which results in a step increase in flow to one steam generator from zero to 100% of the nominal full load value for one steam generator.

c. For the zero load condition, feedwater temperature is at a conservatively low value of 70°F.

In the case of an accidental full opening of one feedwater control valve with the reactor at zero power and the above mentioned assumptions, the maximum reactivity insertion rate is less than the maximum reactivity insertion rate analyzed in Section 15.2.1, Uncontrolled Control Rod Assembly Withdrawal from a Subcritical Condition, and therefore, the results of the analyses are not presented. It should be noted that if the incident occurs with the unit just critical at no load, the reactor may be tripped by the power range high neutron flux trip (low setting) set at approximately 25 percent.

The full power case (EOL, with control) gives the largest reactivity feedback and results in the greatest power increase. A turbine trip and reactor trip is actuated when the nuclear flux level exceeds the power range high nuclear flux trip setpoint of 118% of nominal.

For all excessive feedwater cases continuous addition of cold feedwater is prevented by closure of all feedwater control valves, a trip of the feedwater pumps, and closure of the feedwater pump discharge valves on steam generator high-high level signal.

Transient results, see Figures 15.2-24 and 15.2-25, show the increase in nuclear power and T_{avg} associated with the increased thermal load on the reactor. Steam generator level rises until the feedwater flow is terminated as a result of the high-high steam generator level turbine trip. The DNB ratio does not drop below 1.3 as shown in Figure 15.2-26.

15.2.10.3 Conclusions

Results show that the consequences of excess load increases due to opening the low pressure heater bypass valve are more moderate than those considered for the Excessive Load Increase Accident. Additionally, it has been shown that the reactivity insertion rate which occurs at no load following excessive feedwater addition is less than the maximum value considered in the analysis of the rod withdrawal from a subcritical condition. Also, the DNB ratios encountered for excessive feedwater addition at power are ~~well~~ above the limiting value of 1.30.

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS

Accident	Event	Time (sec.)
Loss of Normal Feedwater (Continued)	Peak water level in pressurizer occurs	280
Excessive feedwater at full load	One main feedwater control valve fails fully open	0
	Minimum DNBR occurs	55.5 ~~66.0~~
	Feedwater flow isolated due to high-high steam generator level	62 ~~192.0~~
Excessive Load Increase 1. Manual Reactor Control (BOL)	10% step load increase	0
	Equilibrium conditions reached (approximate times only)	150
2. Manual Reactor Control (EOL)	10% step load increase	0
	Overtemperature ΔT reactor trip point reached	24.1

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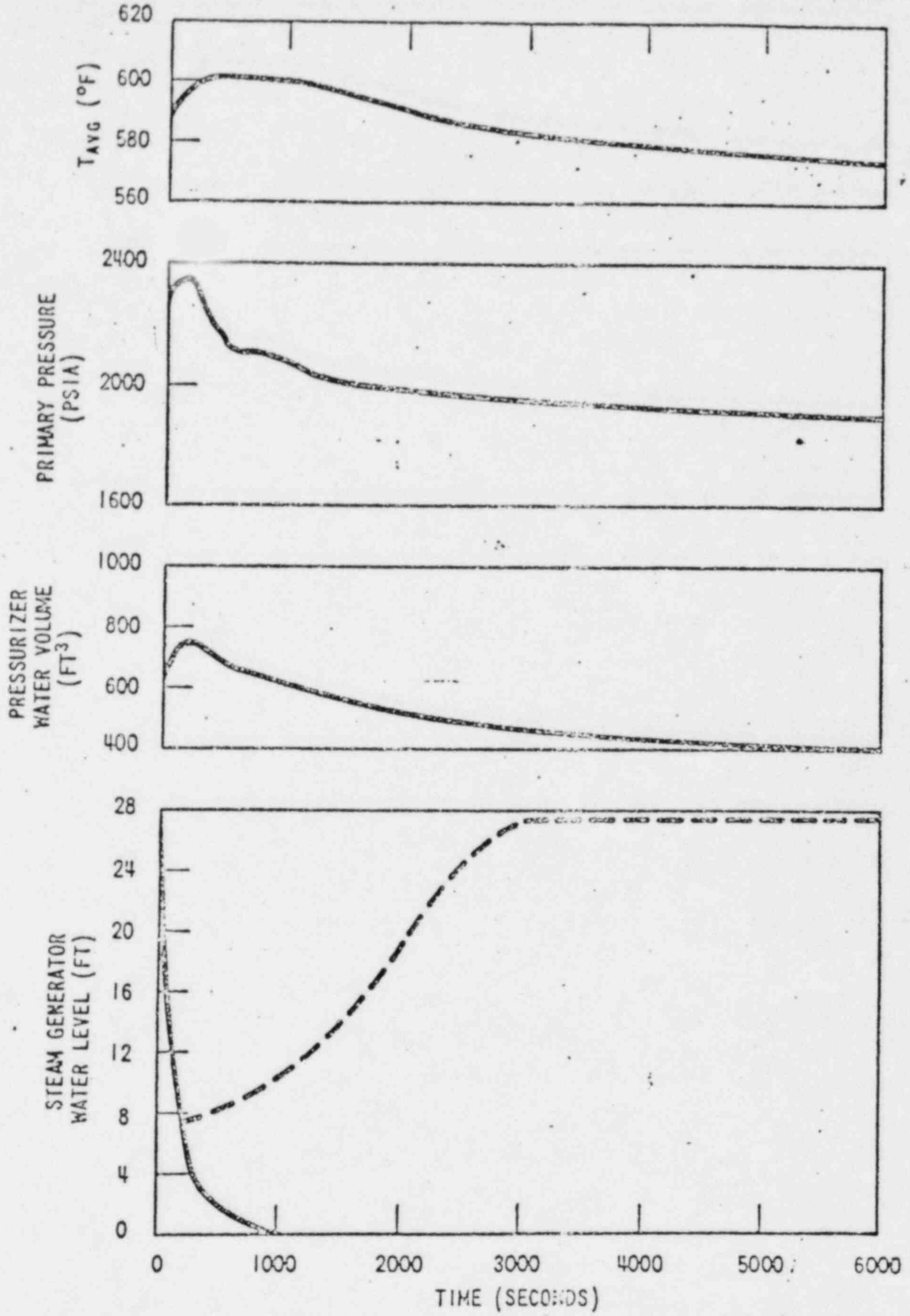

LOSS OF NORMAL FEEDWATER

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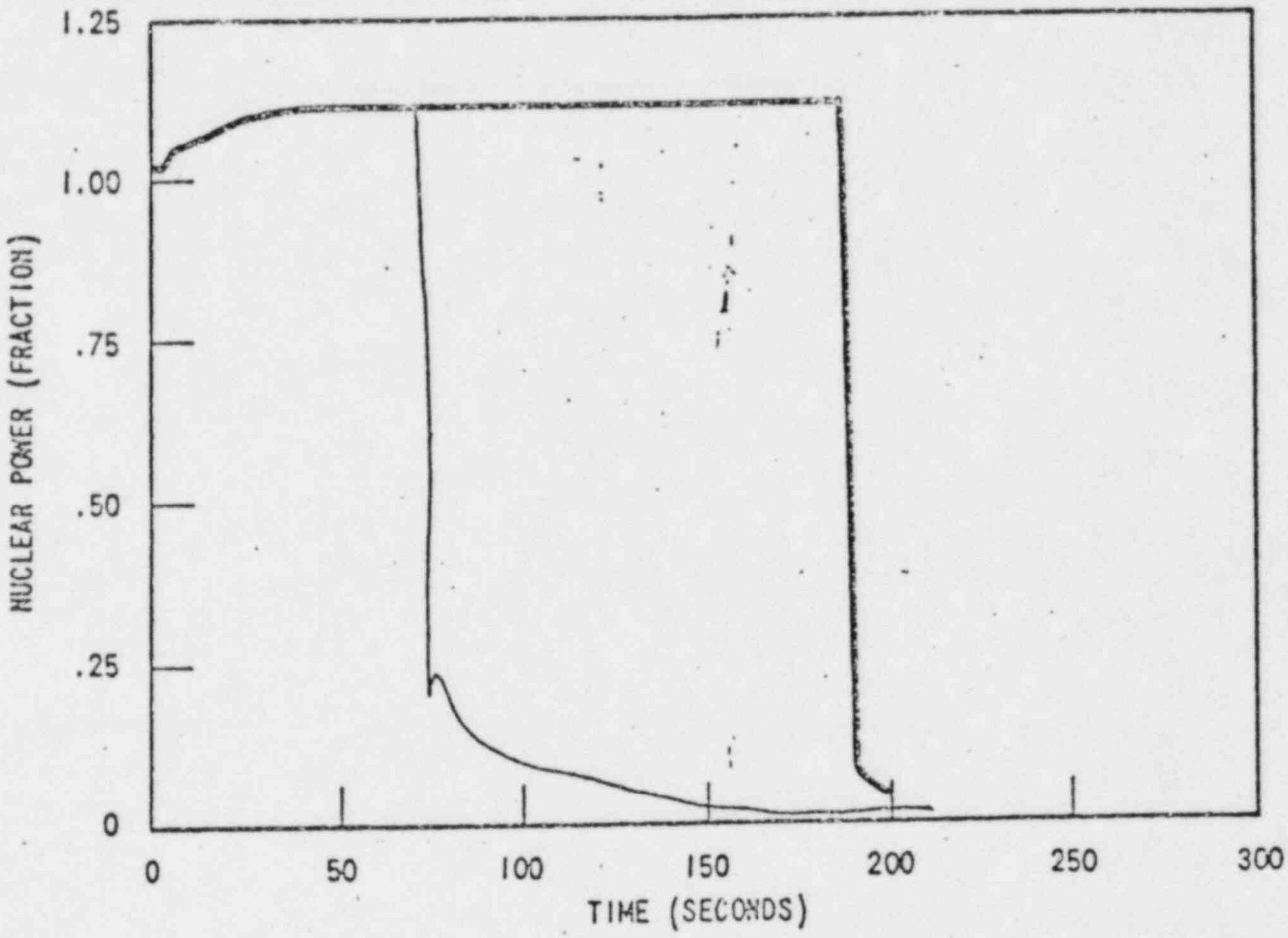
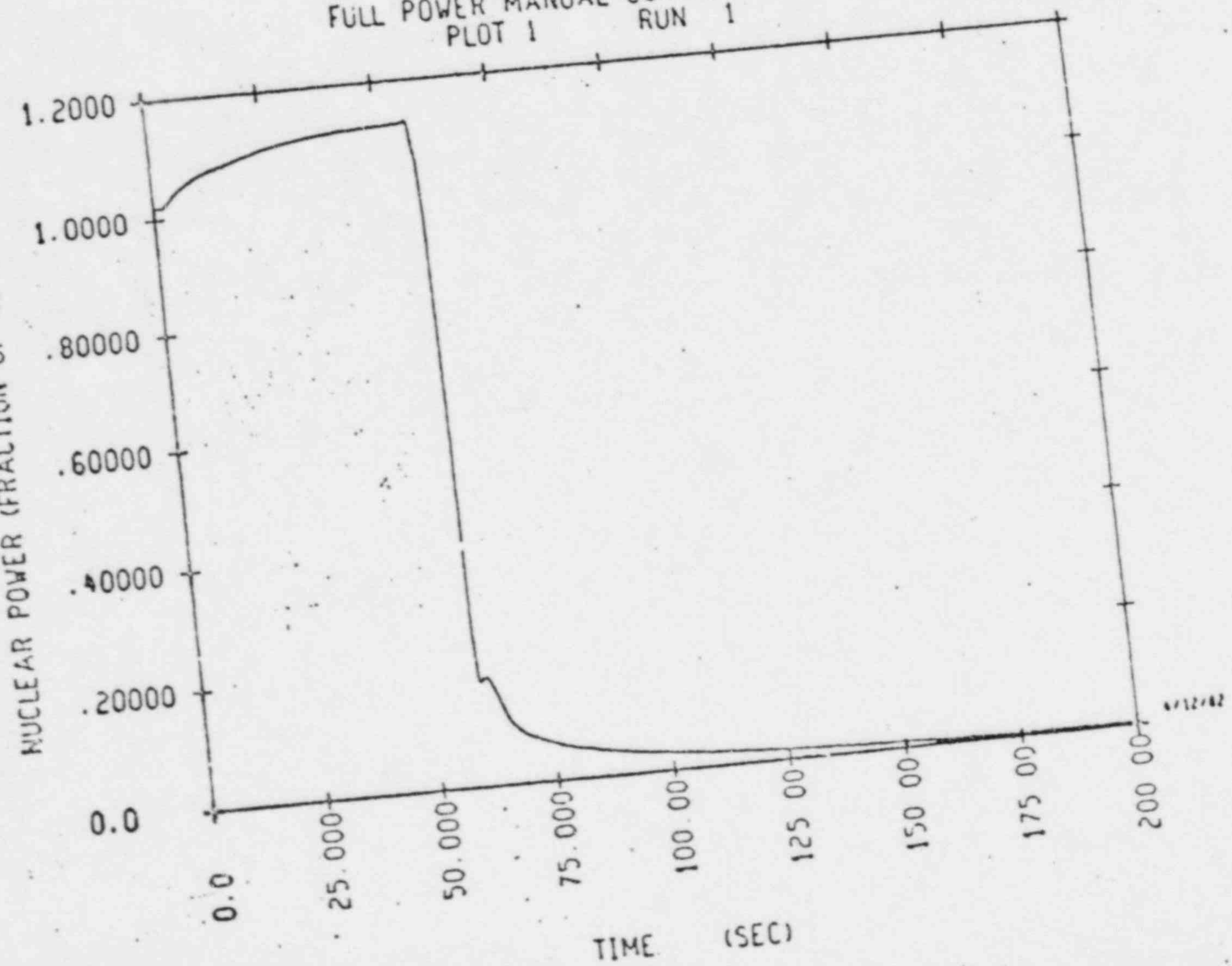


EXCESS FEED ACCIDENT FEED CONTROL
VALVE FAILURE AT FULL LOAD

NE KRSKO FSAR Fig. 15.2-24

STAR10 DATA PROGRA M RWS
FULL POWER MANUAL CONTROL 1 SSWF
PLOT 1 RUN 1



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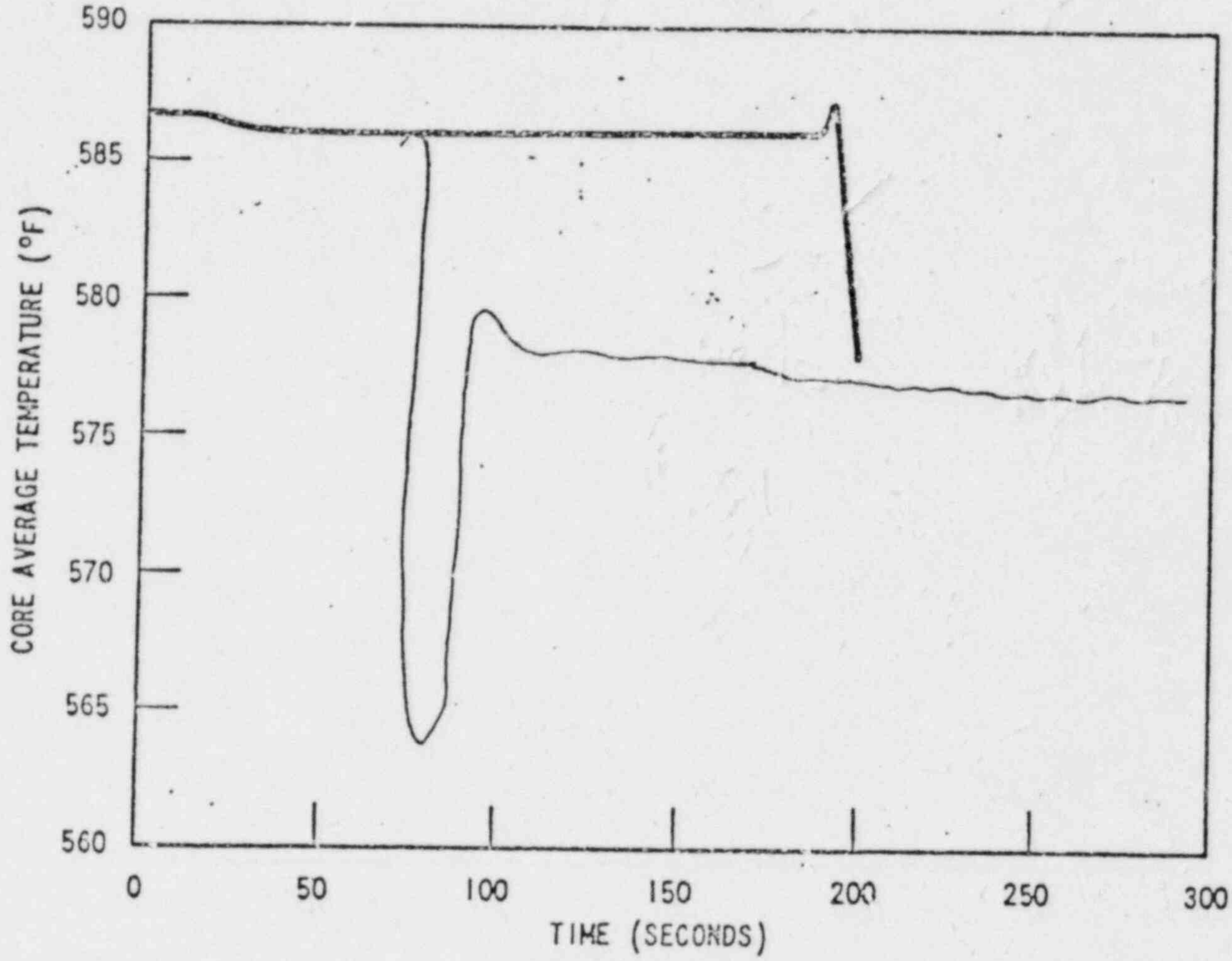
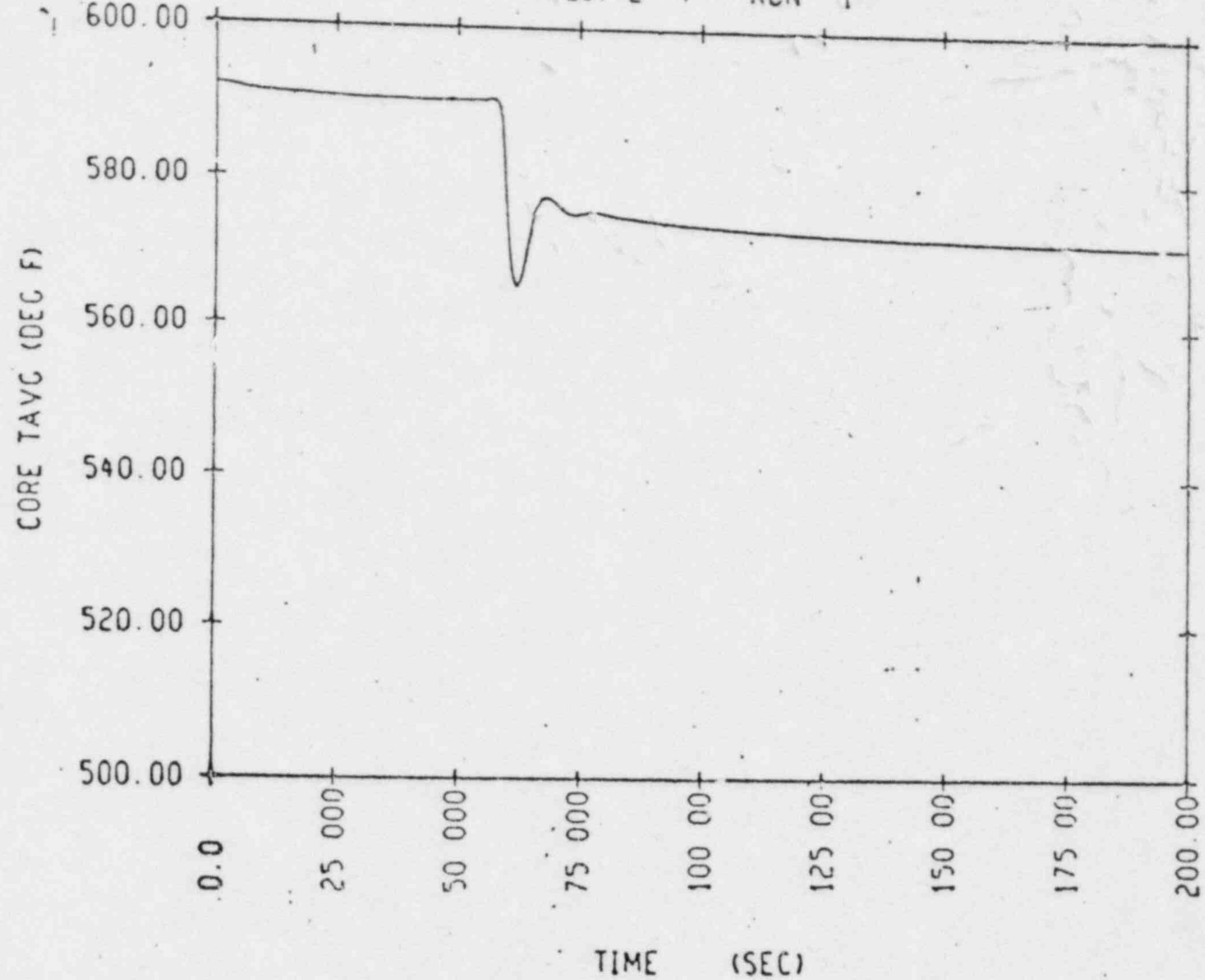


EXCESS FEED ACCIDENT FEED CONTROL VALVE FAILURE AT FULL LOAD		
NE KRSKO	FSAR	Fig. 15.2-25

FULL POWER MANUAL CONTROL [illegible]-55WF
PLOT 2 RUN 1



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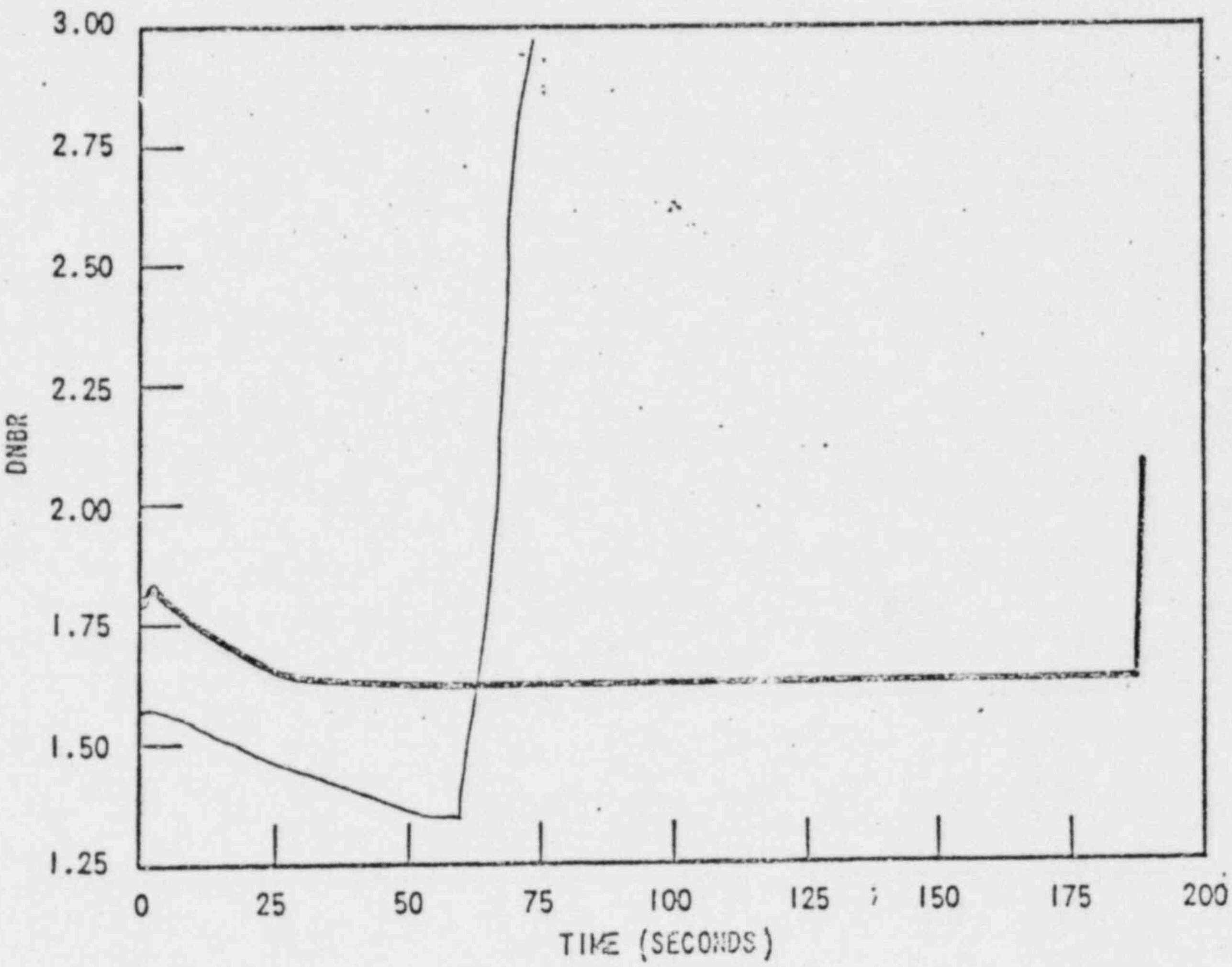
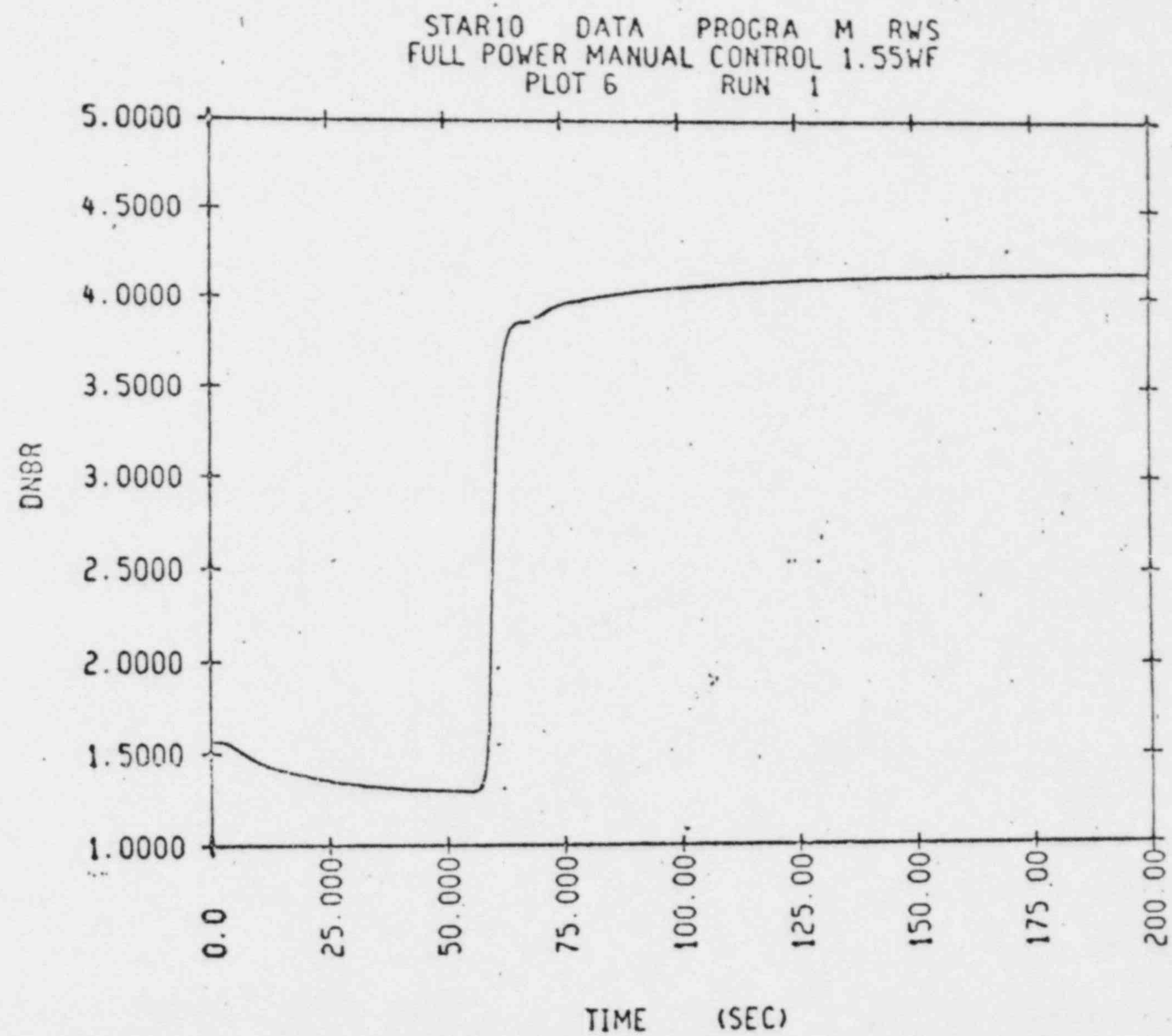
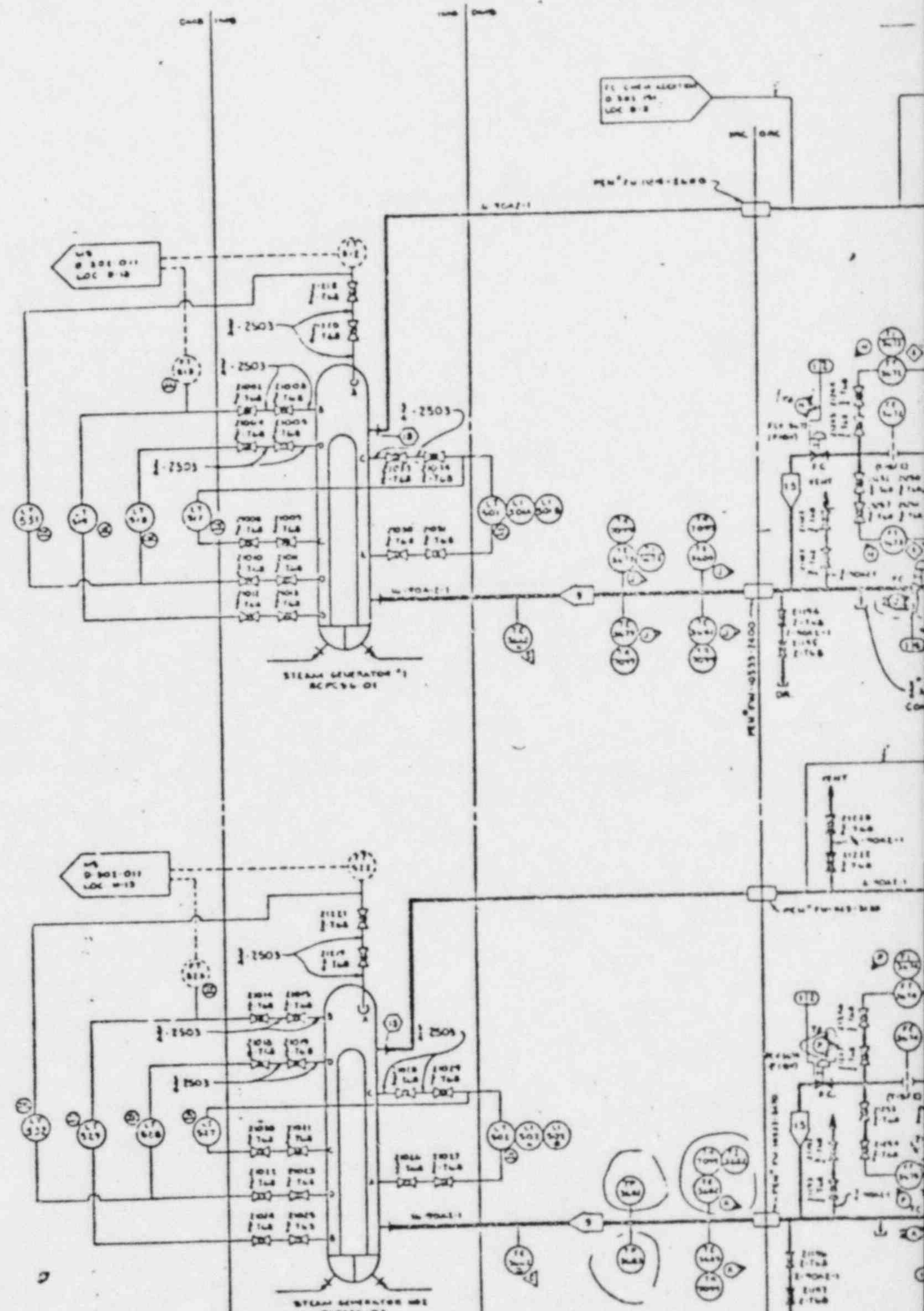


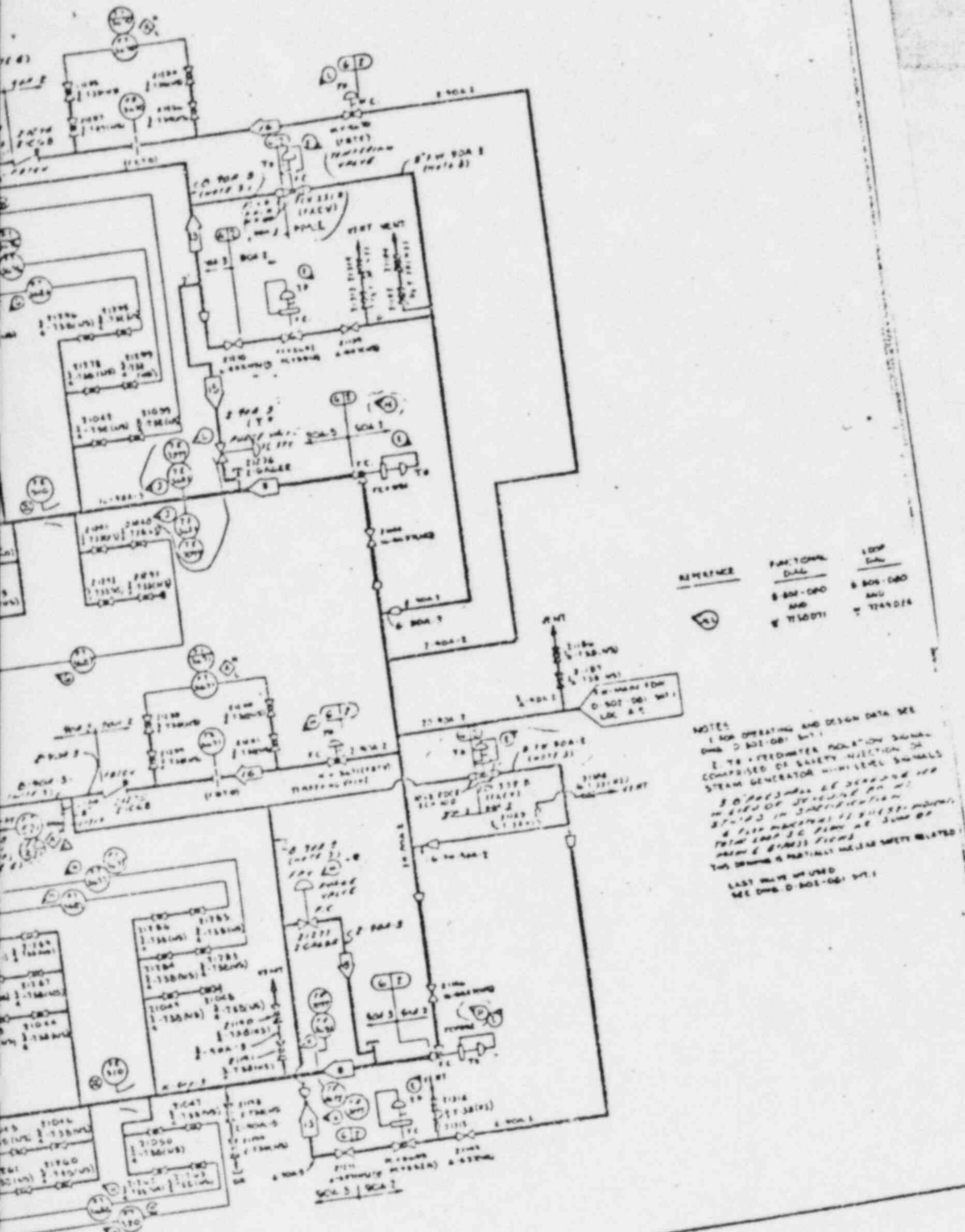
EXCESS FEED ACCIDENT

NE KRSKO	FSAR	Fig. 15.2-26


STAR10 DATA PROGRA M RWS
FULL POWER MANUAL CONTROL 1.55WF
PLOT 6 RUN 1
DNBR
5.0000
4.5000
4.0000
3.5000
3.0000
2.5000
2.0000
1.5000
1.0000
0.0
25.000
50.000
75.000
100.00
125.00
150.00
175.00
200.00
TIME (SEC)



STEAM GENERATOR #1
STEAM GENERATOR NO2



FEEDWATER

Sheet 2 of 2

FIG. 10.1-3

FSAR

NE KRSKO

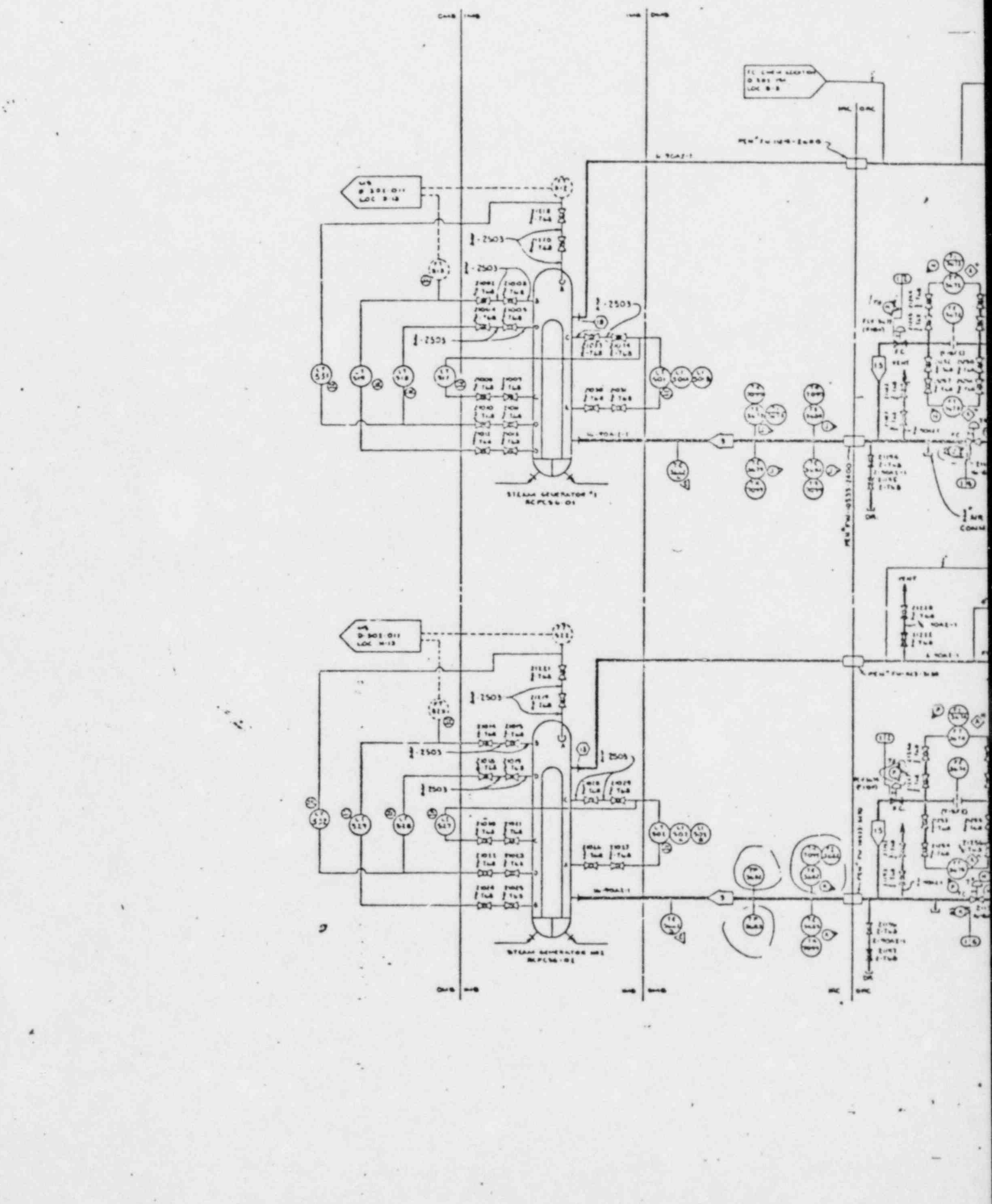


TABLE 10.4-3

FEEDWATER SYSTEM FAILURE ANALYSIS

COMPONENT	MALFUNCTION	COMMENT
Feedwater Pump	While two pumps are running one operating pump trips.	Remaining feedwater pump runs out; no effect on NSSS.
	While one pump is running one operating pump trips.	Standby feed pump automatically starts; no effect on NSSS.
Main Feedwater Flow Control Valve	Valve fails closed above 20% power. Valve stuck full open.	Reactor trip. Turbine trip/Reactor trip.
Feedwater Isolation Valve	Valve fails closed above 20% power. Valve stuck full open.	Reactor trip. Turbine trip/Reactor trip.
Feedwater Piping	Postulated pipe rupture.	See Section 10.4.7.2.3.
Feedwater Preheater Bypass Valve	Valve fails closed below 20% power. Valve fails closed above 70% power.	Reactor trip. Feedwater control valve opens more. Hi preheater flow alarm sounds.
Feedwater Bypass Control Valve	Valve fails closed below 20% power. Valve fails open below 20% power.	Reactor trip. Turbine trip/Reactor trip.
Feedwater Auxiliary Control Valve	Valve fails closed above 70% power.	Feedwater control valve opens more. Hi preheater flow alarm so
	Valve fails open below 30% power.	Turbine trip/Reactor trip.

TABLE 16.3-3 (Continued) (Sheet 3 of

REACTOR TRIP INSTRUMENTATION LIMITING OPERATING CONDITIONS

No.	Functional Unit	No. of Channels	No. of Channels To Trip	Min. Operable Channels	Min. Degree of Redundancy	Operator Action If Conditions of Column 3 or 4 Cannot be Met
18.	Hi-Hi Steam Generator Level or S.I. (Turbine trip and feedwater isolation)	3/loop	2/loop (any loop)	2^+/loop	1/loop	Maintain hot shutdown
~~19.~~	~~Steam Flow/Feedwater Flow Mismatch and Low Steam Generator Water~~	~~2/loop - level~~	~~1/loop - level coincident with~~	~~1/loop - level~~	~~1/loop - level~~	~~Maintain hot shutdown~~
		~~2/loop - flow mismatch~~	~~1/loop - flow mismatch in same loop~~	~~1/loop - flow mismatch~~	~~1/loop - flow mismatch~~	~~Maintain hot shutdown~~

* If the plant is operating above 75 percent of rated power with one excore nuclear channel out of service, then the core quadrant power tilt shall be determined once a day by the movable incore detectors (at least 2 thimbl per quandrant).

** When 2 out of 4 power channels are greater than 10 percent full power, hot shutdown is not required.

*** If one of two intermediate range channels greater than 10^{-10} amps., hot shutdown is not required.

\+ Inoperable channels are placed in the trip mode. Once placed in the trip mode, the channels can be considere operable for purposes of meeting this specification.

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

Channel Description	Check	Calibrate	Test	Remarks
10. Rod Position Bank Counters	S (1,2)	N.A.	R	1) With analog rod position 2) Following rod motion in excess of six in. when computer is out of service
~~12. Steam Generator Flow Mismatch~~	~~S~~	~~R**~~	~~M~~	
11. Steam Generator Level	S	R**	M	
13. Charging Flow	S	A**	N.A.	
14. Residual Heat Removal Pump Flow	S (when in operation)	A**	N.A.	
15A. Boric Acid Tank Level	W	A**	N.A.	
15B. Boric Acid Tank Temperature	W	R	R	
16. Refueling Water Storage Tank Level	W	A	N.A.	
17. Volume Control Tank Level	S	A**	N.A.	
18A. Containment Pressure (SIS signal)	S	R**	$M^{(1)}$	1) Isolation valve signal
18B. Containment Pressure (Streamline Isol)	S	R**	M	Narrow range containment pressure (-3.0, +3.0 psig excluded)

10.4.7 CONDENSATE AND FEEDWATER SYSTEMS

10.4.7.1 Condensate System

No change this page!

10.4.7.1.1 Design Basis

The condensate pumps take suction from the condenser hotwell and develop sufficient head to overcome the system pressure drop and transport the condensate to the suction of the main feedwater pumps at the required net positive suction head. The condensate system is shown on Figure 10.1-4.

Condensate can be recirculated to the condenser during low load conditions, maintaining cooling flow requirements to the steam jet air ejector condenser and gland steam condenser. A bypass, directly from the discharge of the condensate pumps to the main feedwater pump suction, is provided to maintain condensate flow at a reduced system pressure drop should the heater drain pumps not be available, a condensate pump become inoperable, or a load rejection occur. The bypass system is controlled by sensing heater drain pump minimum discharge flow, feed pump suction pressure, and load rejection. In addition, the lower temperature condensate that is bypassed prevents flashing from occurring at the suction of the feedwater pumps during severe transient conditions.

The condensate pumps are designed in accordance with the requirements of the Hydraulic Institute Standards. The low pressure feedwater heaters are designed, fabricated, inspected and tested in accordance with the ASME Code, Section VIII, Division 1. System piping and valves are designed in accordance with ANSI B31.1 and ANSI B16.5, respectively. The condensate system, except for the condensate storage tank, is non-nuclear safety class. The condensate storage tank is designed, fabricated, inspected and tested in accordance with ASME Code Section III, Seismic Category I, since it is the primary inventory source for the auxiliary feedwater system.

No Change

10.4.7.1.2 System Description

The condensate system consists of three 50 percent capacity vertical condensate pumps, five stages of closed feedwater heaters and associated piping, valves, controls and equipment condensers. Condensate is pumped from the hotwell storage area by two normally operating condensate pumps through two 50 percent capacity, parallel streams of low pressure heaters to the feedwater pumps.

The feedwater heaters are shell and U tube heat exchangers with two heating zones except for heater #2, a condensing zone and an integral drain cooling zone. There are two 50 percent capacity, parallel streams of low pressure heaters and each stream can be bypassed if it is out of service. Turbine load must be adjusted according to the manufacturer's instructions when a feedwater stream is isolated. The three lowest pressure heaters are located inside the condenser neck. Heaters #5 and #6 are in a single shell, duplex arrangement. Drains from the #3, #4, #5, and #6 heaters are cascaded back to the condenser. Drains from the remaining heaters and the moisture separator reheaters are collected in the heater drain tank and then pumped into the condensate system piping just upstream of the main feedwater pump.

10.4.7.1.3 Safety Evaluation

A standby condensate pump is provided which starts on low feedwater pump suction pressure.

System makeup is provided directly from the condensate storage tanks to the condenser hot well.

The condensate system is not required for safe shutdown of the reactor and is not a safety related system.

No Change!

10.4.7.1.4 Tests and Inspections

Condensate system equipment is shop tested hydrostatically and operationally. Feedwater heaters are tested in accordance with the requirements of Section VIII, Division I, of the ASME Boiler and Pressure Vessel Code.

10.4.7.1.5 Instrumentation

Pressure and temperature test points, recorders and indicators are provided to test and monitor system performance. Equipment operating conditions and system motor operated valve positions are indicated in the control room.

In the nuclear safety class portion of the system, instrumentation is provided for monitoring the condensate storage tank level.

10.4.7.2 Feedwater System

10.4.7.2.1 Design Basis

The feedwater system is designed to pump feedwater through one stage of high pressure feedwater heaters for the maintenance of steam generator water level during startup, 0-100 percent power operation, shutdown and steam dump conditions. The feedwater system is shown on Figure 10.1-3.

Feedwater system components from and including the feedwater isolation valves to the steam generators are Safety Class 2, Seismic Category I. System components upstream of the containment isolation valves are non-nuclear safety class.

Feedwater piping upstream of the feedwater isolation valves in the turbine building is designed in accordance with ANSI B31.1, 1973 Addenda B, and valves conform to ANSI B16.5-1968. Safety Class 2 feedwater system piping, valves and other pressure containing parts are designed in accordance with ASME Code, Section III, Class 2, 1971 Winter, 1972 Addenda.

The feedwater pumps are designed in accordance with the requirements of the Hydraulic Institute Standards. Design points for these pumps are selected to satisfy the requirements of the turbine thermal cycle at the maximum guaranteed condition plus margins for wear and surges. The feedwater pumps are also capable of maintaining steam generator water level during a load rejection and steam dump at 96 percent flow at a steam generator pressure of 972 psia.

High pressure feedwater heaters are designed, fabricated, inspected, tested and stamped in accordance with the ASME Code, Section VIII, Division 1. Thermal performance of these feedwater heaters is governed by Heat Exchange Institute Standards.

The feedwater system equipment parameters are listed in Table 10.4-2.

10.4.7.2.2 System Description

The feedwater system includes three 50 percent capacity motor driven main feed pumps, two parallel high pressure feedwater heaters and associated piping, valves and controls.

Out.

At loads above 15 percent, normal operating control is achieved by using a three element system consisting of inputs proportional to steam flow, feedwater flow and steam generator water level to control the position of the feedwater regulating valves. At loads of 15 percent and below, the steam generator level is maintained with the feedwater bypass control valve. The bypass control valve is automatically positioned with inputs proportional to the steam generator water level or the bypass control valve is manually positioned from the control room. These controls are always in operation except during a safety injection signal or a reactor trip coincident with low T_{avg}. The modulating signal is blocked in these cases.

The main feed pumps are driven by a constant speed motor through speed increasing gearing. A low flow bypass to the main condenser is provided for use during startup.

10.4.7.2.2 System Description

The Main Feedwater System consists of three (50 percent) capacity motor driven centrifugal feedwater pumps, two (50 percent) parallel high pressure feedwater heaters with bypass, separate feedwater control stations and preheater bypass control stations for each steam generator, piping, valves, and associated instrumentation.

The Main Feedwater System provides feedwater to the steam generator under all operating conditions from 0 to 100 percent power. For loads below approximately 20 percent power; feedwater is delivered to the 6-inch auxiliary feedwater nozzle on the steam generator. Between 20 and 70 percent power; flow is delivered through the 16-inch main feedwater nozzle on the steam generator. Above 70 percent power; approximately 70 percent of the required flow is maintained to the 16-inch nozzle and the difference is supplied through the 6-inch auxiliary nozzle. The main feedwater control valves control flow in the 16-inch main feedwater line. Flow in the bypass line is controlled through the feedwater bypass control valves for plant loads from 0 to 20 percent, or through the feedwater auxiliary control valve for plant loads above 70 percent.

Each of the three feedwater pumps take suction from a common manifold through locked open suction isolation valves and flow measuring elements and strainers, and discharge into a common header through tilting disc check valves and motor operated stop check valves. Separate minimum flow recirculation lines to the condenser are located between the pump discharge nozzles and the tilting disc check valve. These recirculation lines are fitted with an air operated diaphragm control valve, pressure breakdown orifices, and a locked open isolation valve at the condenser.

A feedwater pump bypass line equipped with a check valve and a manual stop check valve connects the feedwater pump suction header (condensate side) with the feedwater pump discharge header. This line is used to fill the steam generator and to provide a supply of feedwater from the condensate system during start-up. A single pipe connects the feedwater pump discharge header with feedwater heaters 1A and 1B. At the feedwater heaters, the flow path is divided and passes through the two half-capacity heaters. A common bypass permits either heater to be isolated while full flow to the steam geneators is maintained.

After the feedwater heaters and/or bypass, the feedwater is recombined into a single pipe to ensure an even temperature distribution of water being supplied to the steam generators. Above 20 percent load, two separate main feedwater control stations regulate the flow of feedwater to their respective steam generators. Each main control station consists of a flow measuring element, an air operated diaphragm control valve, and associated manual isolation valves. As plant load is increased above 70 percent, a similar control is maintained on the feedwater control valves with the feedwater auxiliary control valves regulating bypass flow to maintain approximately 70 percent flow through the main 16-inch nozzle.

A three element feedwater regulating system controls the position of the main feedwater control valve. The three elements are steam flow, steam generator water level, and feedwater flow.

Each feedwater control station also contains a feedwater bypass control valve and a feedwater auxiliary control valve which are used during periods of operation below 20 percent load and above 70 percent load, respectively. The feedwater bypass control valve is controlled remote manually or automatically, in accordance with steam generator level demand when plant operation is below 20 percent. When plant operation is above 70 percent, the feedwater auxiliary control valve is automatically controlled to maintain steam generator flow through the 16-inch nozzle to approximately 70 percent. The feedwater auxiliary control valves can also be manually controlled.

Downstream from the main feedwater control valve, the feedwater passes through a check valve. At that point, the piping changes from Non-Safety Class to Class 2 and remains Safety Class 2 to the 16-inch connection on the steam generator Similarly, downstream from the feedwater bypass control valve and the feedwater auxiliary control valve, flow passes through the feedwater preheater bypass valve, at which point the piping changes from Non-Safty Class to Safety Class 2. It then remains Safety Class 2 to the 6-inch auxiliary nozzle on the steam generator.

During plant start-up and low power operation, feedwater flow is directed through the 6-inch preheater bypass line. A very small warm-up flow passes through the main feedwater line via the feedwater isolation bypass valve and the feedwater purge valve. This mode prevents cold water from being injected into the preheater section of the steam generator during low power operation and potential water hammer damage. When power reaches approximately 20 percent and the main feedwater line is sufficiently warmed, all flow is transferred to the 16-inch main feedwater line. Once flow exceeds 70 percent of full load flow, the preheater bypass line will flow that amount required to maintain approximately 70 percent flow through the main feedwater nozzle.

10.4.7.2.2 System Description (cont'd)

During reduction from full power, feedwater flow is automatically transferred from both steam generator nozzles at 70 percent load and then is manually transferred from the main nozzle to the auxiliary nozzle at approximately 20 percent power. At any time the main feedwater isolation valve is open or the feedwater control valve is opened, a low flow is maintained through the auxiliary feedwater nozzle via the feedwater tempering valve. This flow, taken upstream of the feedwater control valves, prevents thermal shock to the auxiliary nozzle if flow is suddenly transferred from the main nozzle or if auxiliary feedwater is required.

Auxiliary feedwater flow and chemical feed to the main feedwater system is accomplished through separate injection lines on each 6-inch preheater bypass line.

Two of the three feedwater pumps have an auto start feature if the other trips. If all three feedwater pumps are tripped, then auxiliary feedwater is automatically initiated with both Train A and Train B electrical signals.

10.4.7.2.3 Safty Evaluation

A pipe break in one loop of the main feedwater system will not damage the other intact feedwater loop. Also, a feedwater pipe break will not propagate to cause a main steam or reactor coolant loop pipe break. The above protection is provided by separation, restraints, equipment orientation, and jet impingement barriers. Following a feedwater break in the intermediate building, all control valves and isolation valves on the intact feedwater loop remain operable.

The main feedwater, feedwater preheater bypass, feedwater isolation, feedwater control, feedwater auxiliary control, feedwater bypass control, feedwater tempering, and feedwater purge valves are designed to close within five seconds after receipt of a closure signal, to provide isolation if a feedwater line breaks either inside or outside the reactor building. These valves are designed to fail closed and to close upon loss of control air or Train A or Train B electric power, except for the main feedwater isolation and feedwater preheater bypass valves.

The main feedwater isolation valve operator utilizes a nitrogen precharge to close the valve. Valve opening is accomplished by an air pump and hydraulic system which, through a series of solenoid valves in the hydraulic line, are required to be energized. The valve closes on loss of electric signal and slowly bleeds closed on loss of air supply to the valve air pump.

The feedwater preheater bypass valve similarly utilizes a nitrogen precharge to close the valve. However, valve opening is hydraulic with an electric pump rather than an air pump. Solenoids on the valve are required to be energized to pump the hydraulic fluid and close the valve. No air is required for this valve. Any of the following signals initiate valve closure:

1. Safety injection signal.
   a. Low steamline pressure.
   b. Low pressurizer pressure.
   c. High containment pressure.
   d. Manual safety injection signal.
2. Either steam generator hi-hi level.

10.4.7.2.3 Safety Evaluation (cont'd)

The above signals also result in a Train A feedwater pump trip signal which trips all running feedwater pumps. When the safety injection signal is bypassed, either Pump 1 or Pump 2 starts automatically. This does not affect feedwater system isolation, since the isolation valves remain closed and feedwater is bypassed to the condenser.

No Change

Table 10.4-3 presents a feedwater system failure analysis.

10.4.7.2.4 Tests and Inspections

Major equipment is periodically inspected to ensure proper conditions and operation. Additional testing includes:

1. Hydrostatic testing of safety class system piping and equipment following construction and prior to placing the system in service.

2. Hot functional testing subsequent to cold hydrostatic testing.

3. Normal operational checking and routine maintenance of the system.

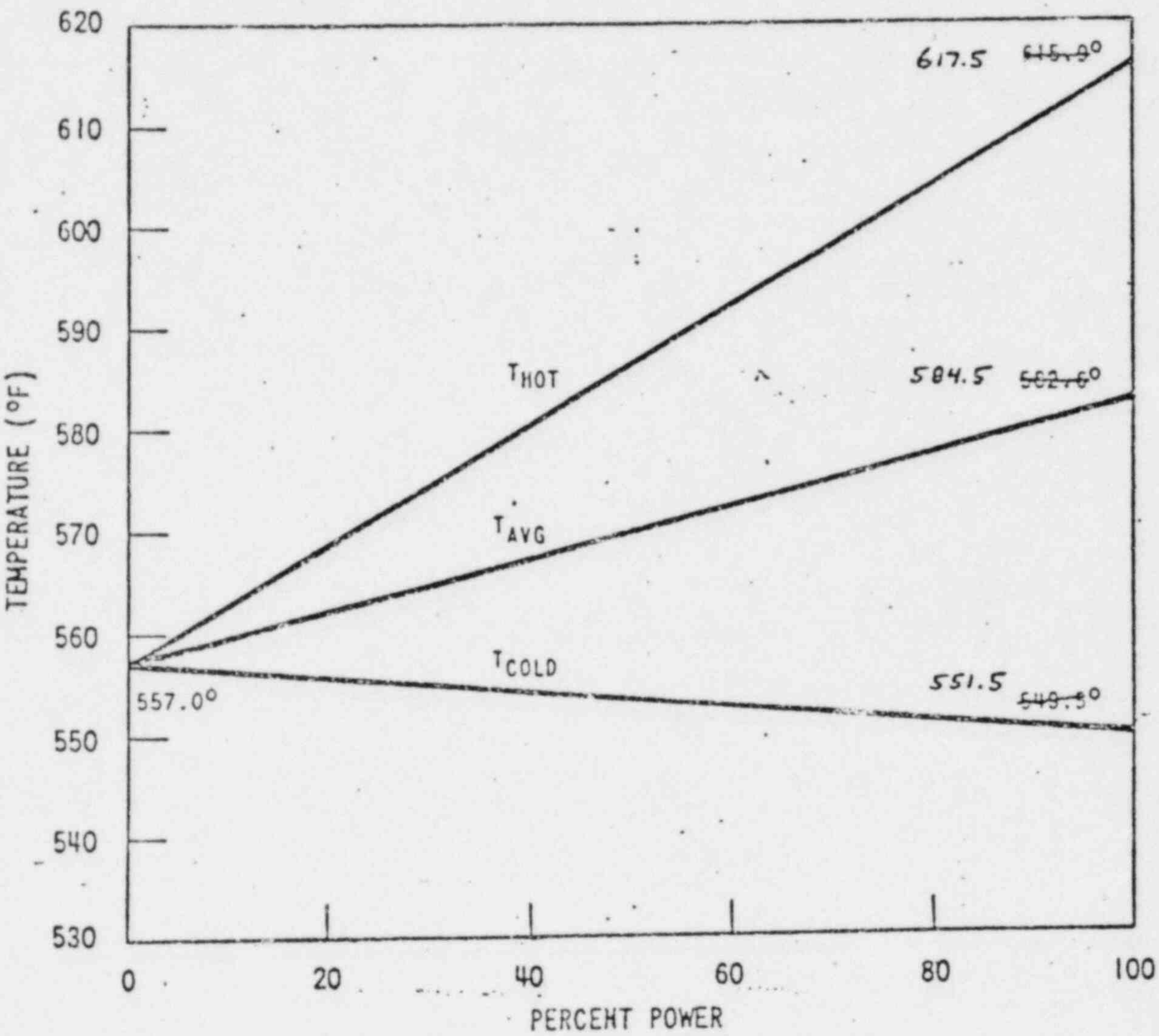
Inservice inspection of safety class portions of the system is performed in accordance with applicable ASME Code, Section XI, requirements.

10.4.7.2.5 Instrumentation

The following instrumentation is supplied for the non-nuclear safety class portion of the system to permit operator evaluation of major equipment performance and to provide a performance record:

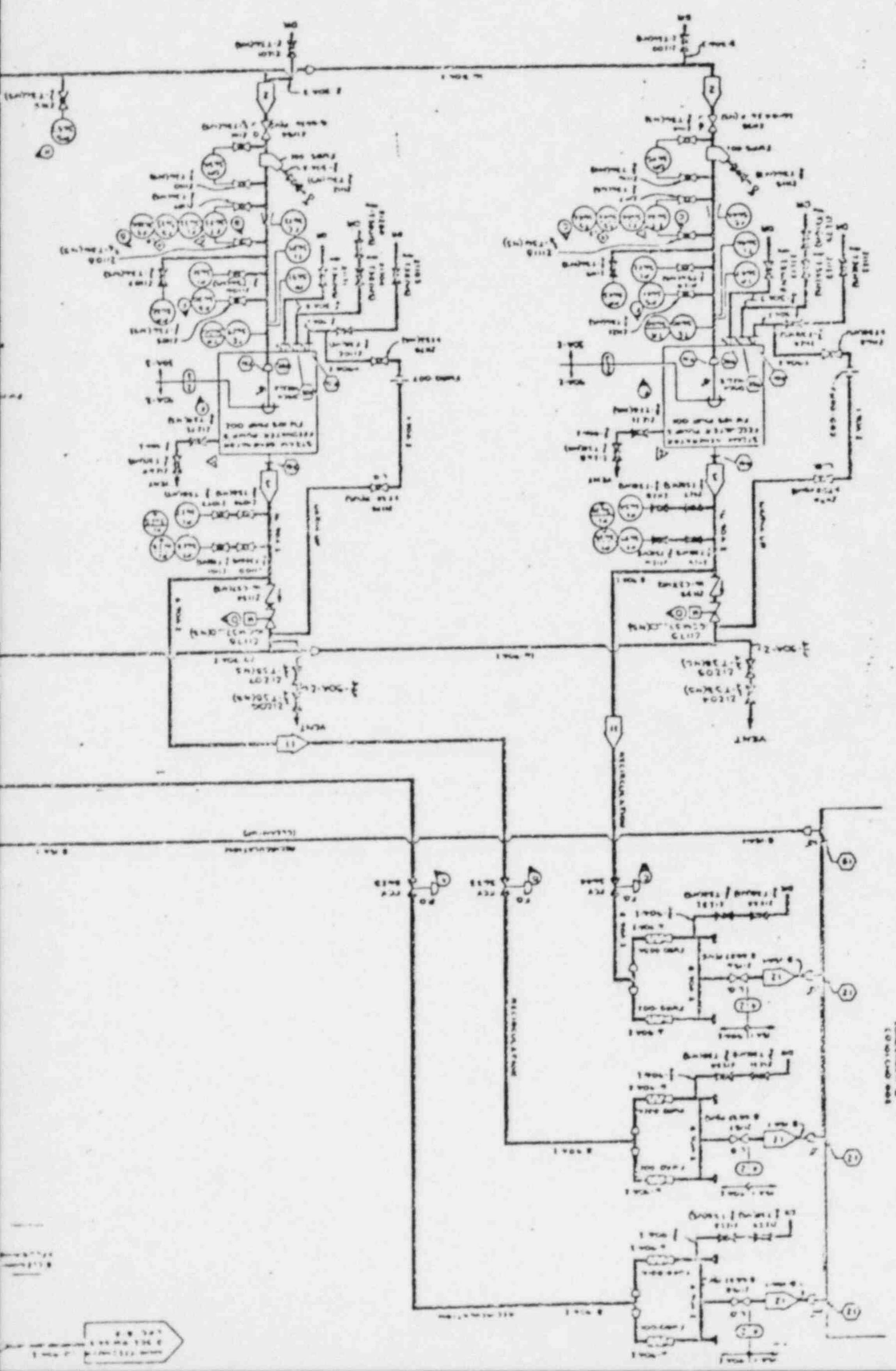
1. Pressure indicators, switches and test connections.
2. Flow indicators, controllers and test connections.
3. Temperature indicators, recorders and test connections.

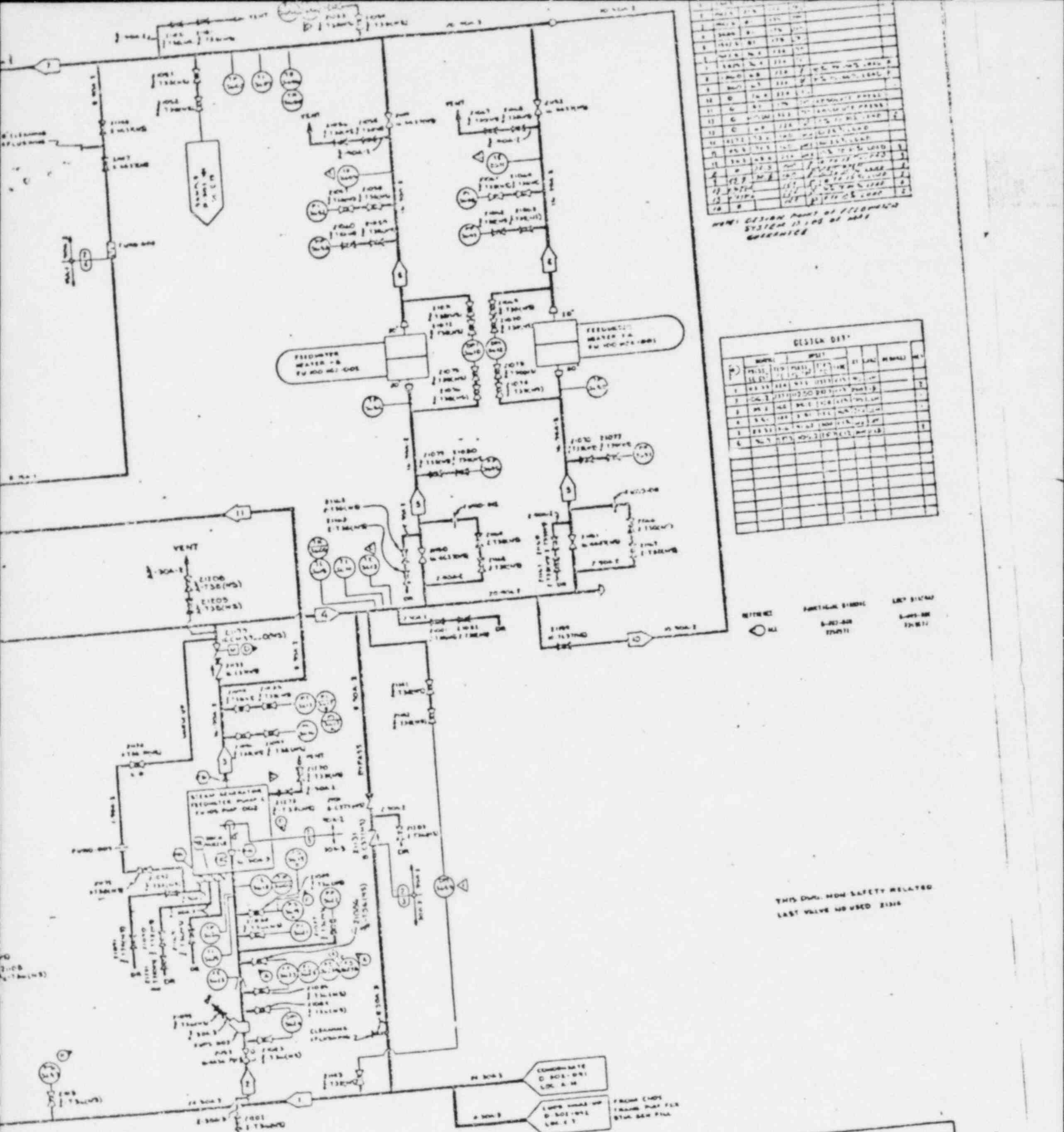
In the nuclear safety class portion of the system, instrumentation is provided to permit monitoring of the temperature of feedwater entering the steam generators as well as water level in the steam generators. Instrumentation associated with the development of control signals for the feedwater isolation valves is discussed in Section 7.3.



REACTOR COOLANT SYSTEM TEMPERATURE - PERCENT POWER MAP

NE KRSKO FSAR FIGURE 5.3-1





FEEDWATER

Sheet 1 of 2

NE KRSKO FSAR FIG. 10.1-3

KRSKO FEEDWATER SYSTEM MODIFICATION FOR SPLIT FLOW OPERATION

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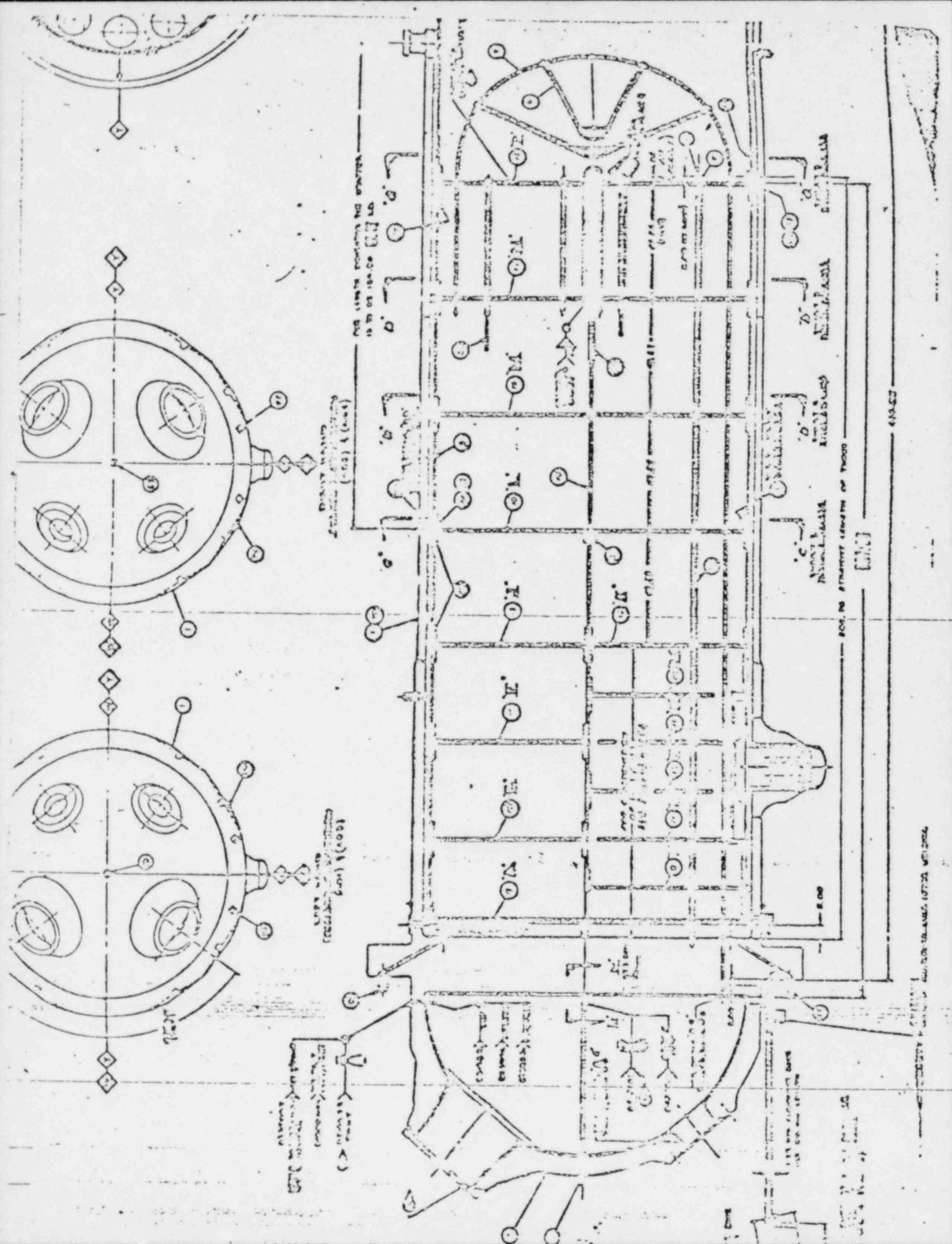
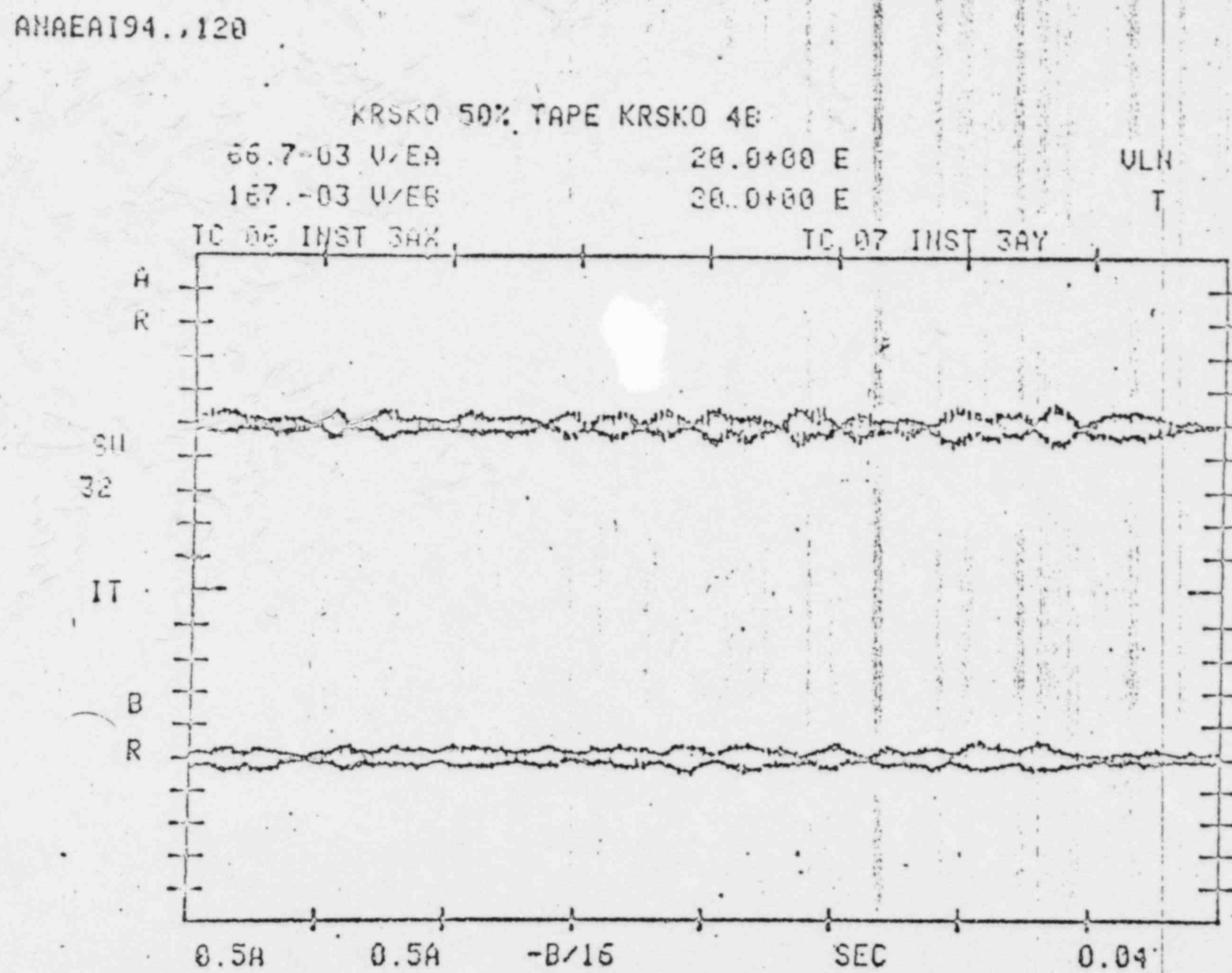


STEAM GENERATOR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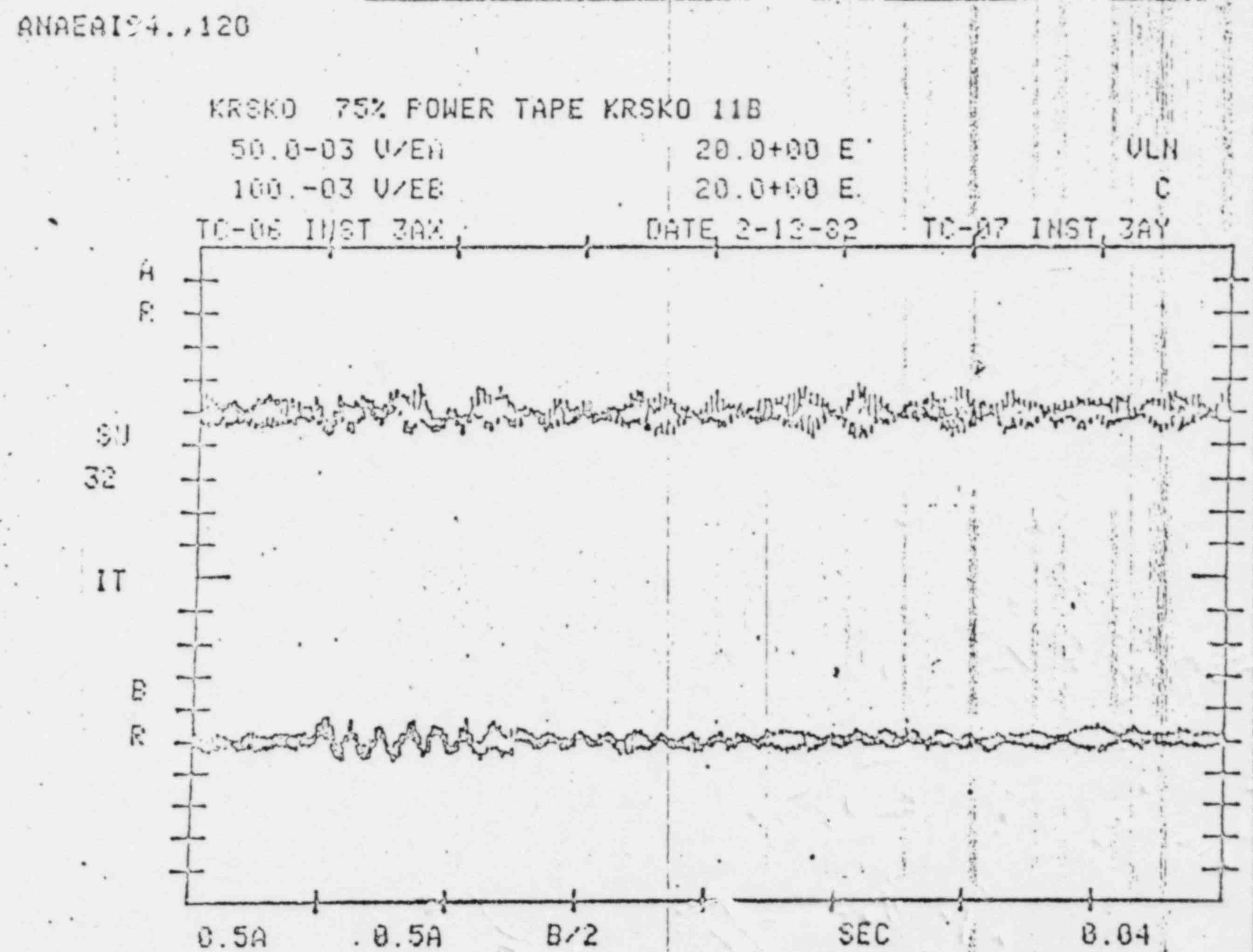
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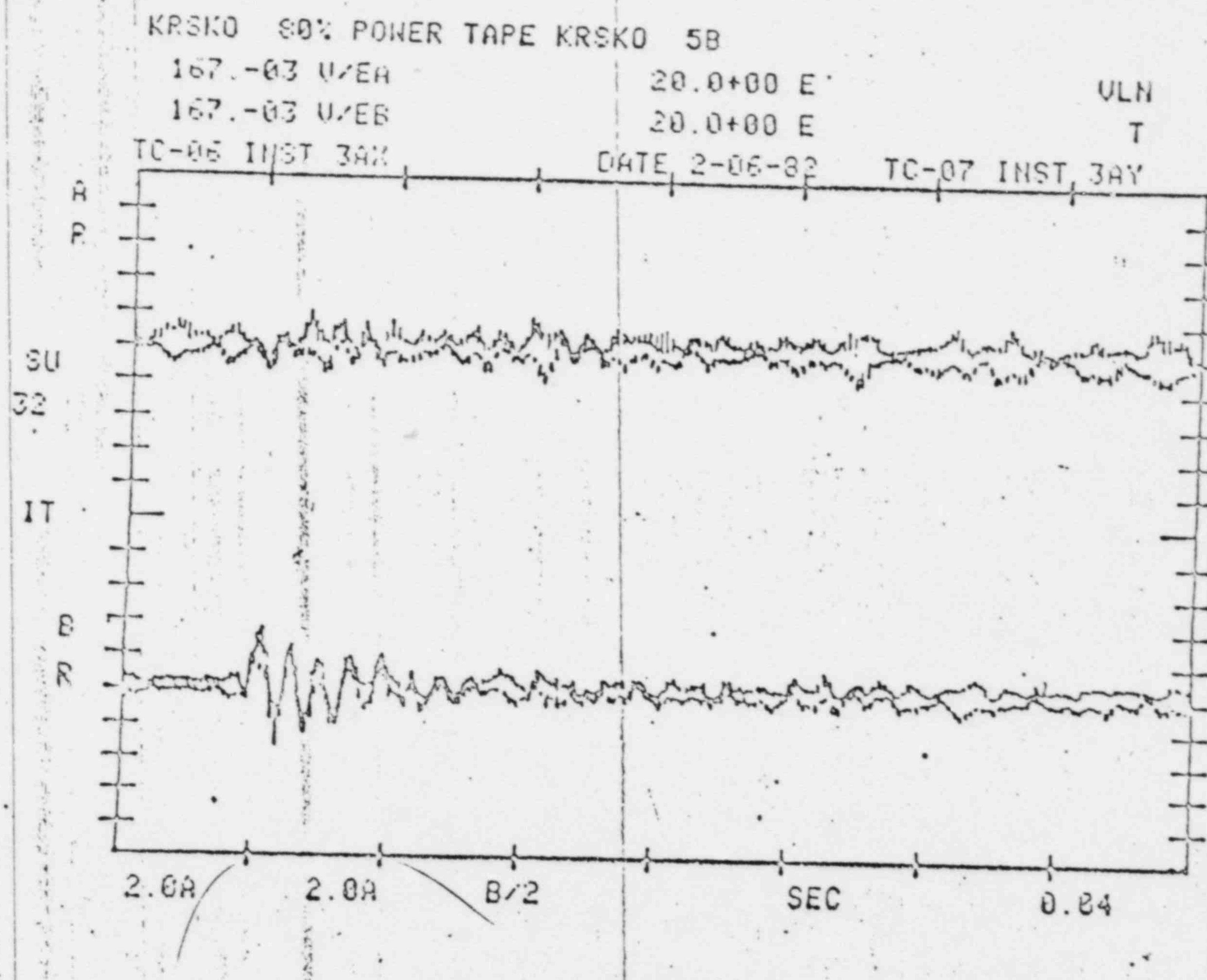

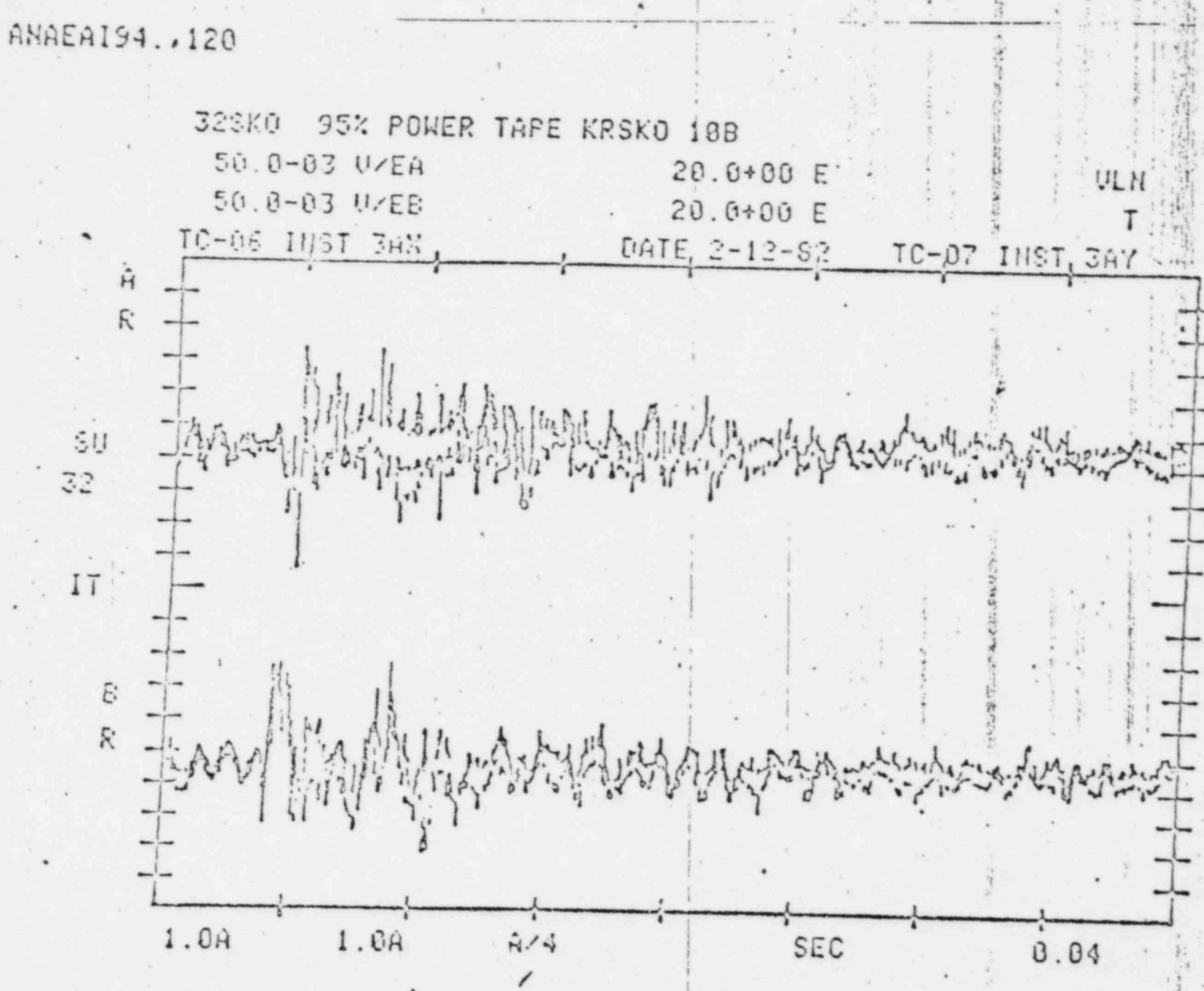
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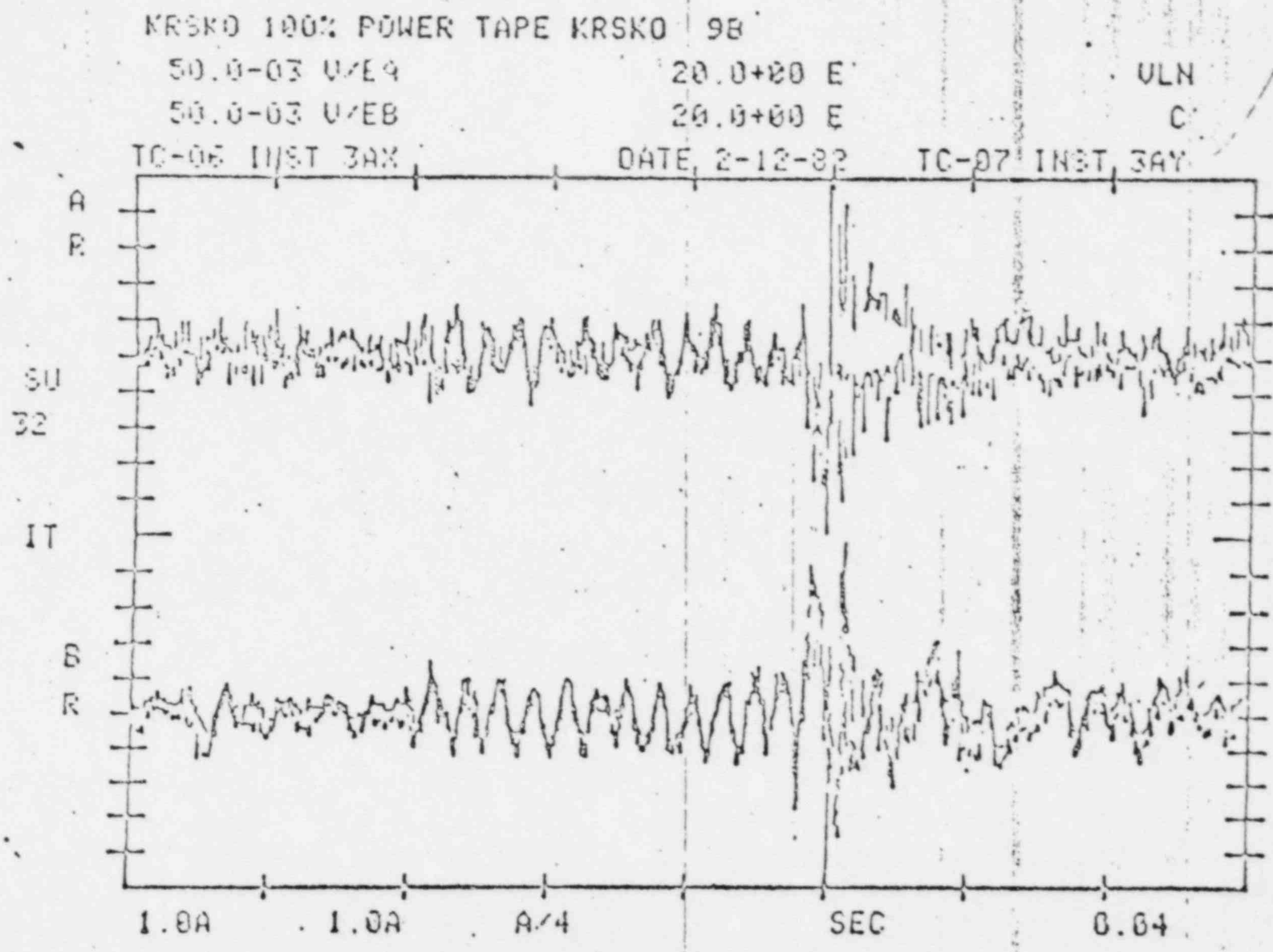
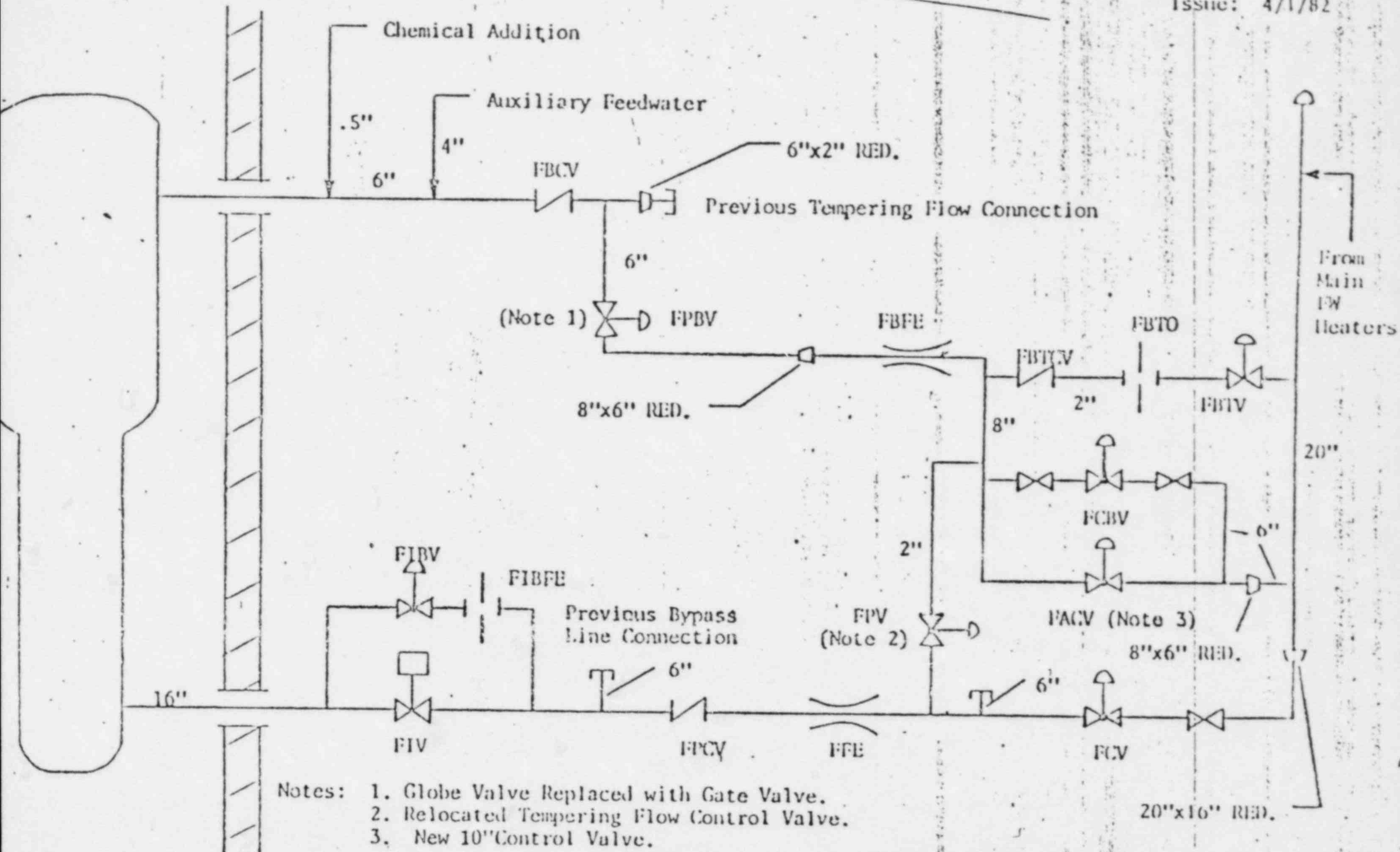


FLUID SYSTEMS MODIFICATION

INFORMATION

KRSKO FEEDWATER BYPASS SYSTEM: PROPOSED ARRANGEMENT

Issue: 4/1/82



Notes:
1. Globe Valve Replaced with Gate Valve.
2. Relocated Tempering Flow Control Valve.
3. New 10" Control Valve.

- FW Preheater Check Valve
- FW Isolation Valve
- FW Isolation Bypass Valve
- FW Control Valve
- FW Control Bypass Valve

FBFE - FW Bypass Flow Element
FFE - FW Flow Element
FBTV - FW Bypass Tempering Valve
FBCV - FW Bypass Check Valve
FPBV - FW Preheater Bypass Valve

FIBFE - FW Isolation Bypass Flow Element
FBTO - FW Bypass Tempering Orifice
FBTCV - FW Bypass Tempering Check Valve
FPV - FW Purge Valve
FACV - FW Aux. Control Valve

KRSKO FEEDWATER BYPASS SYSTEM CHANGES

- THE PREHEATER BYPASS LINE CONNECTION HAS BEEN RELOCATED FROM DOWNSTREAM OF THE FEEDWATER CONTROL VALVE (FCV) TO UPSTREAM OF THIS VALVE.

- A NEW 10" FEEDWATER AUXILIARY CONTROL VALVE (FACV) WITH ASSOCIATED 8" PIPING HAS BEEN INSTALLED IN PARALLEL WITH THE EXISTING FEEDWATER CONTROL BYPASS VALVE (FCBV).

- THE FEEDWATER PREHEATER BYPASS VALVE (FPBV) HAS BEEN CHANGED FROM A GLOBE TO A GATE VALVE.

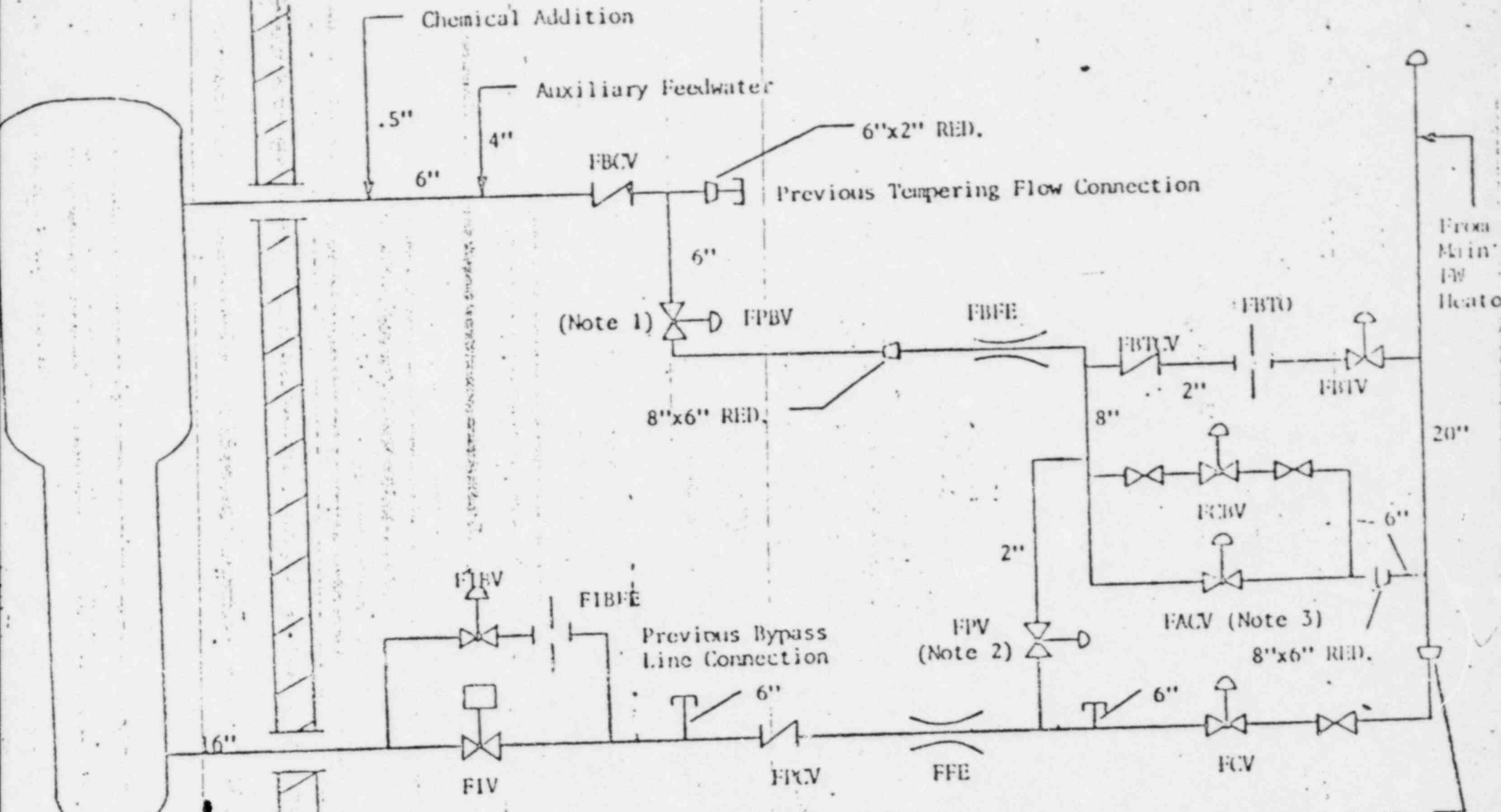
- A NEW LINE FOR PURGING HAS BEEN ADDED AROUND THE FCV.

- A NEW VENTURI FLOW METER HAS BEEN INSTALLED IN THE BYPASS LINE.

- THE PRESENT, PROTECTION GRADE SG WATER LEVEL TRANSDUCER/TRANSMITTER ARE NO LONGER USED FOR CONTROL FUNCTIONS. FOR THE CONTROL FUNCTIONS, A SEPARATE CONTROL GRADE SG WATER LEVEL TRANSDUCER/TRANSMITTER HAS BEEN ADDED.

- THE PRESENT FEED/STEAM MISMATCH SIGNAL USED IN COINCIDENT WITH THE SG LO WATER LEVEL SIGNAL HAS BEEN ELIMINATED. A 2/3 SG LO-LO WATER LEVEL SIGNAL (FROM EQUIPMENT NOT USED FOR CONTROL FUNCTIONS) IS NOW USED FOR REACTOR TRIP.

- FVC IS NOW CONTROLLED TO LIMIT THE FEEDWATER FLOW INTO THE SG MAIN NOZZLE TO 70%. ABOVE 70% THE FEEDWATER AUXILIARY CONTROL VALVE IS AUTOMATICALLY OPENED TO CONTROL FLOW BETWEEN 70 & 100% POWER.



Notes:

1. Globe Valve Replaced with Gate Valve.
2. Relocated Tempering Flow Control Valve.
3. New Control Valve.

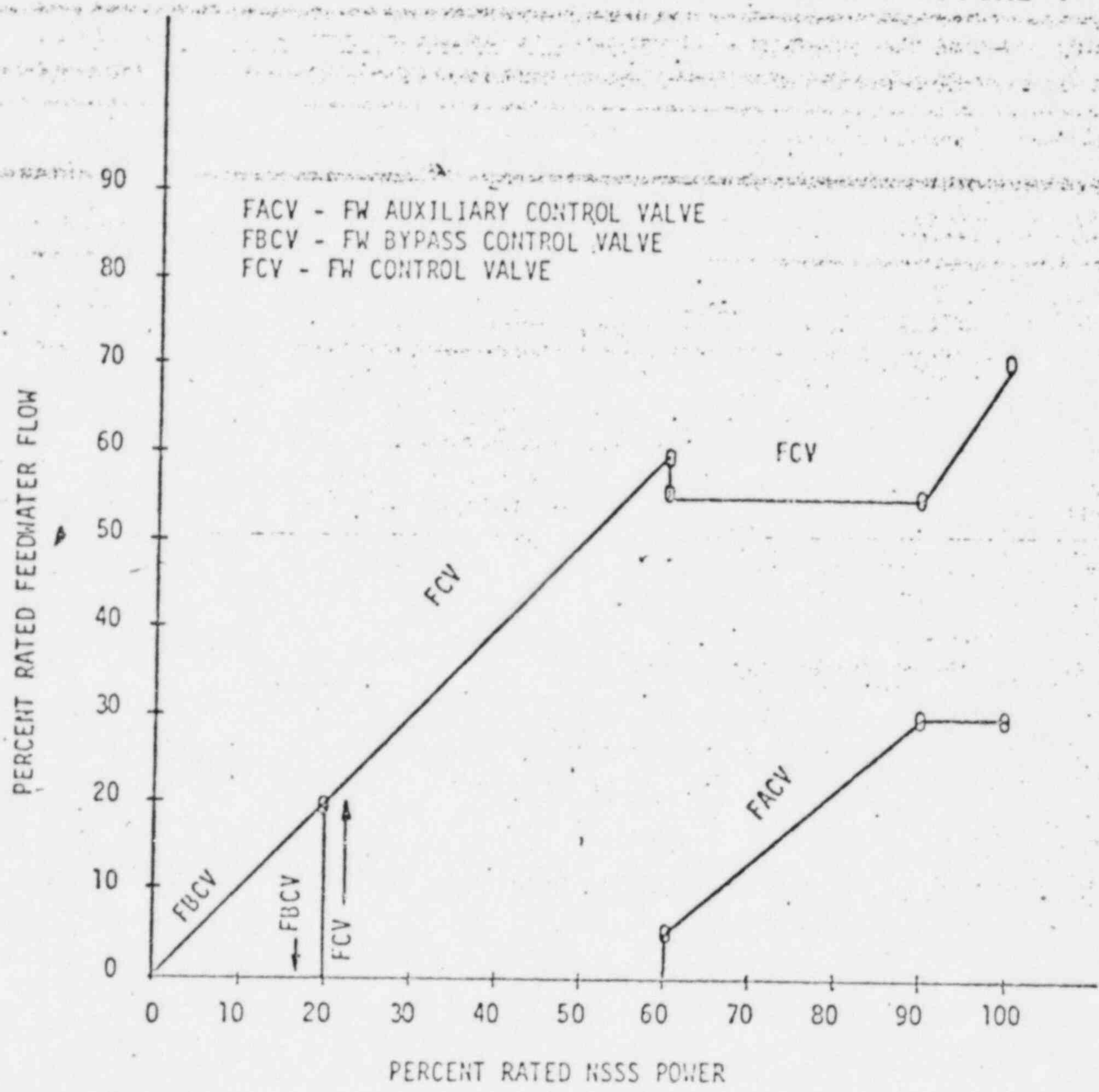
CV - FW Preheater Check Valve
IV - FW Isolation Valve
IBV - FW Isolation Bypass Valve
CV - FW Control Valve
BV - FW Control Bypass Valve

FBFE - FW Bypass Flow Element
FFE - FW Flow Element
FBTV - FW Bypass Tempering Valve
FBCV - FW Bypass Check Valve
FPBV - FW Preheater Bypass Valve

FIBFE - FW Isolation Bypass Flow Element
FBTO - FW Bypass Tempering Orifice
FBTCV - FW Bypass Tempering Check Valve
FPV - FW Purge Valve
FACV - FW Aux. Control Valve

KRSKO NUCLEAR POWER PLANT

FEEDWATER SYSTEM CONTROL VALVES OPERATION



KRSKO FEEDWATER BYPASS

SYSTEM MODIFICATION

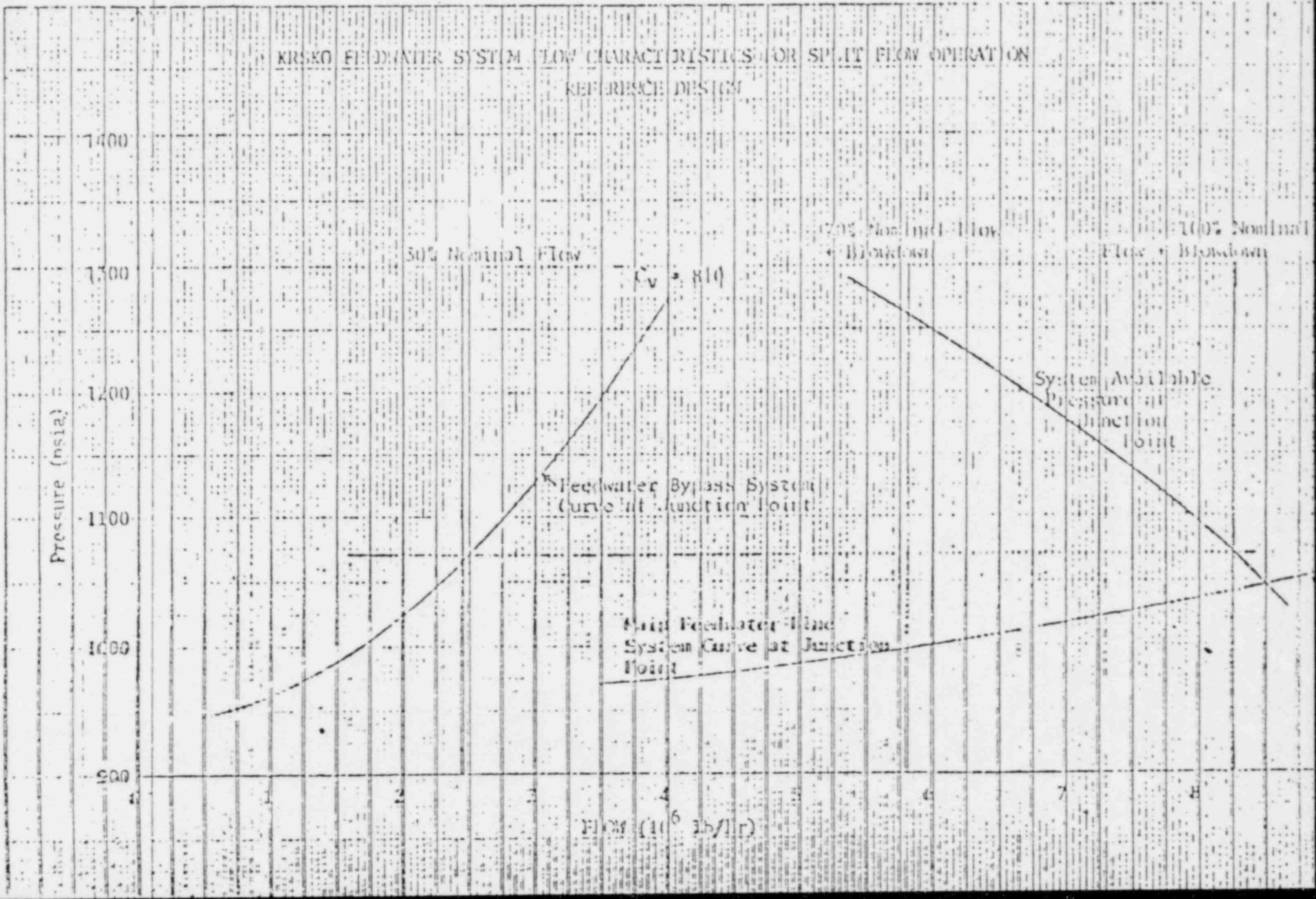
BASIC QUESTIONS

- o CAN A 70/30 FEEDWATER FLOW SPLIT BE OBTAINED.
- o WILL THE SYSTEM CONTROL PERFORM IN A STABLE MANNER.
- o WILL THE CHANGE IN THE REACTOR COOLANT SYSTEM PARAMETERS RESULTING FROM A 70/30 SPLIT FEEDWATER FLOW BE ACCEPTABLE.
- o ARE PREVIOUS CRITERIA ESTABLISHED FOR MINIMIZATION OF COUNTER-FLOW PREHEAT SG PREHEATER PRESSURE TRANSIENTS MET.
- o ARE PRESENT SAFETY/REGULATORY REQUIREMENTS MET.

K-E 46 1320

KRSKO FEEDWATER SYSTEM FLOW CHARACTERISTICS FOR SPLIT FLOW OPERATION

REFERENCE DESIGN



Paramclers for 70-30 Feedflow Split

WESTINGHOUSE PROPRIETARY CLASS 2

POWER CAPABILITY PARAMETERS

OWNER UTILITY: Sauske Elektrarne and Elektroprivreda

~~PLANT NAME:~~ Krsko (KRK)

UNIT NUMBER: 1

BASIC COMPONENTS

Reactor Vessel, ID, in.	132	Core bypass, %	4.5
Core		Isolation Valves	No
Number of Assemblies	121	Number of Loops	2
Rod Array	16x16	Steam Generator	
Rod OD, in.	0.374	Model	D4
Number of grids	8R	Shell design pressure, psia	1200
Active Fuel Length, in.	144	Reactor Coolant Pump	
Number of control rods, FL/PL	33/4	Model/Weir	100D/NA
Internals Type	KRK	Pump motor, hp	7000
		Frequency, Hz	50

THERMAL DESIGN PARAMETERS

	N LOOP	N-1 LOOP	OPERATING
NSSS Power, %	100		
MWt	1882		
10^6 BTU/hr	6421.6		
Reactor Power, MWt	1876		
10^6 BTU/hr	6401.2		
Thermal Design Flow, Loop gpm	94500		
Reactor 10^6 lb/hr	70.9		
Reactor Coolant Pressure, psia	2250		
Reactor Coolant Temperature, °F			
Core outlet	620.3 (1)		
Vessel outlet	617.5 (1)		
Core average	587.8 (1)		
Vessel average	584.5 (1)		
Vessel/core inlet	551.5 (1)(2)		
Steam Generator outlet	551.2 (1)		
Steam Generator			
Steam Temperature, °F	534.6		
Steam Pressure, psia	920		
Steam Flow, 10^6 lb/hr total	8.17		
Feed Temperature, °F	430		
Moisture, % max	0.25		
App. F'F', hr sq ft °F/BTU	0.00005		
Zero Load Temperature, °F	557		

HYDRAULIC DESIGN PARAMETERS

Best Estimate Flow (gpm)/Head (ft)	
Pump Design Point, Flow (gpm)/Head (ft)	— 100,700/272
Mechanical Design Flow, gpm	104700

REASON FOR CHANGE

NOTES

(1) Temperatures reflect increase to offset 70-30 feed flow split

(2) FSAR Basis TCOLD: 552°F

Basis:

Reference: Issue No:

KRK EI&C SYSTEM MODIFICATION

- CONTROL AND PROTECTION SYSTEM
  - STEAM GENERATOR TRIP SIGNALS
  - FEEDWATER CONTROL AND ISOLATION
- PREHEATER BYPASS SYSTEM

PROTECTION SYSTEM IMPACT

STEAM GENERATOR PROTECTION

- DELETE LOW FEEDWATER FLOW REACTOR TRIP.

STEAM BREAK PROTECTION

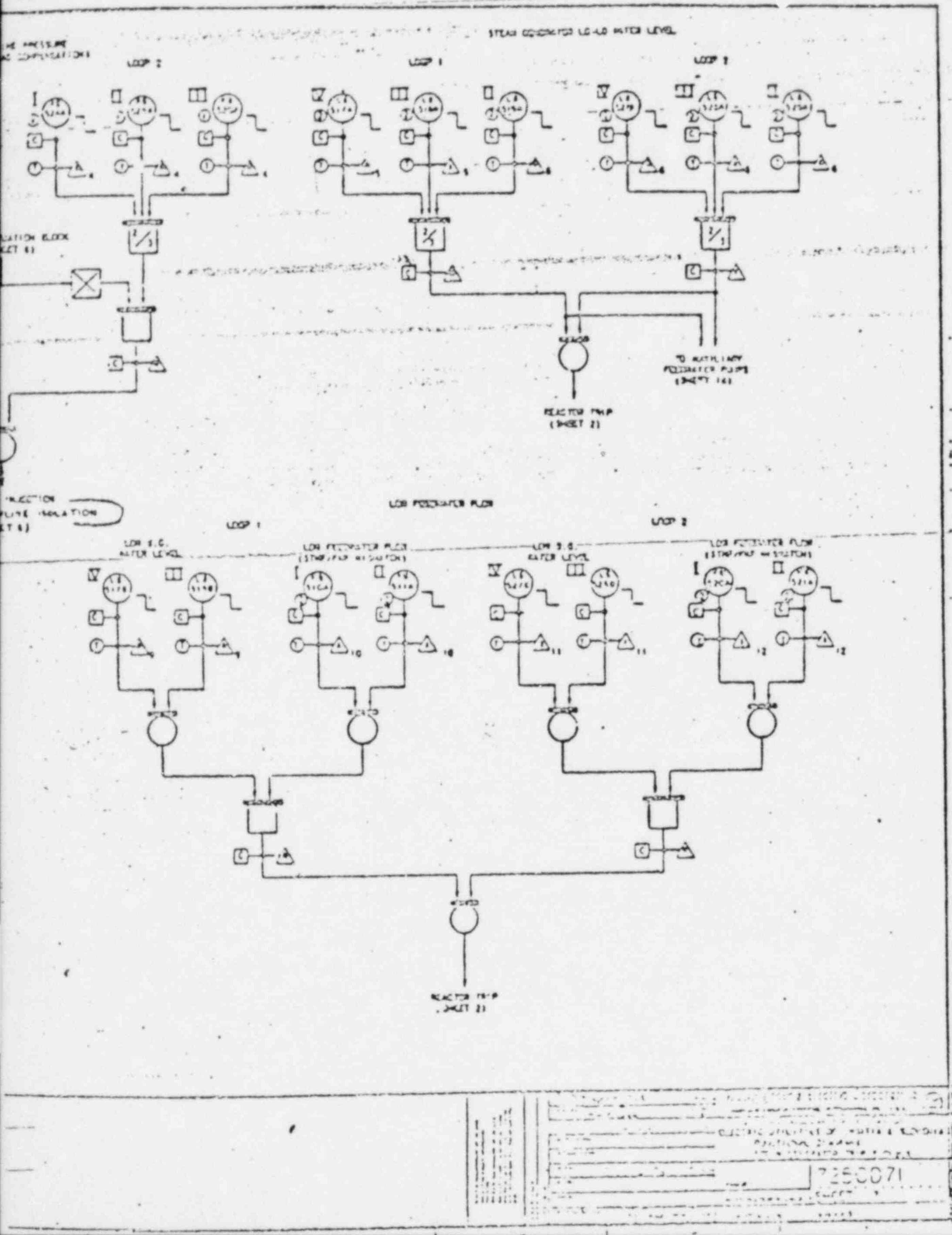
- THE NEW FEEDWATER AUXILIARY CONTROL VALVE MUST BE CLOSED BY FEEDWATER ISOLATION SIGNAL.

LOW FEEDWATER FLOW TRIP

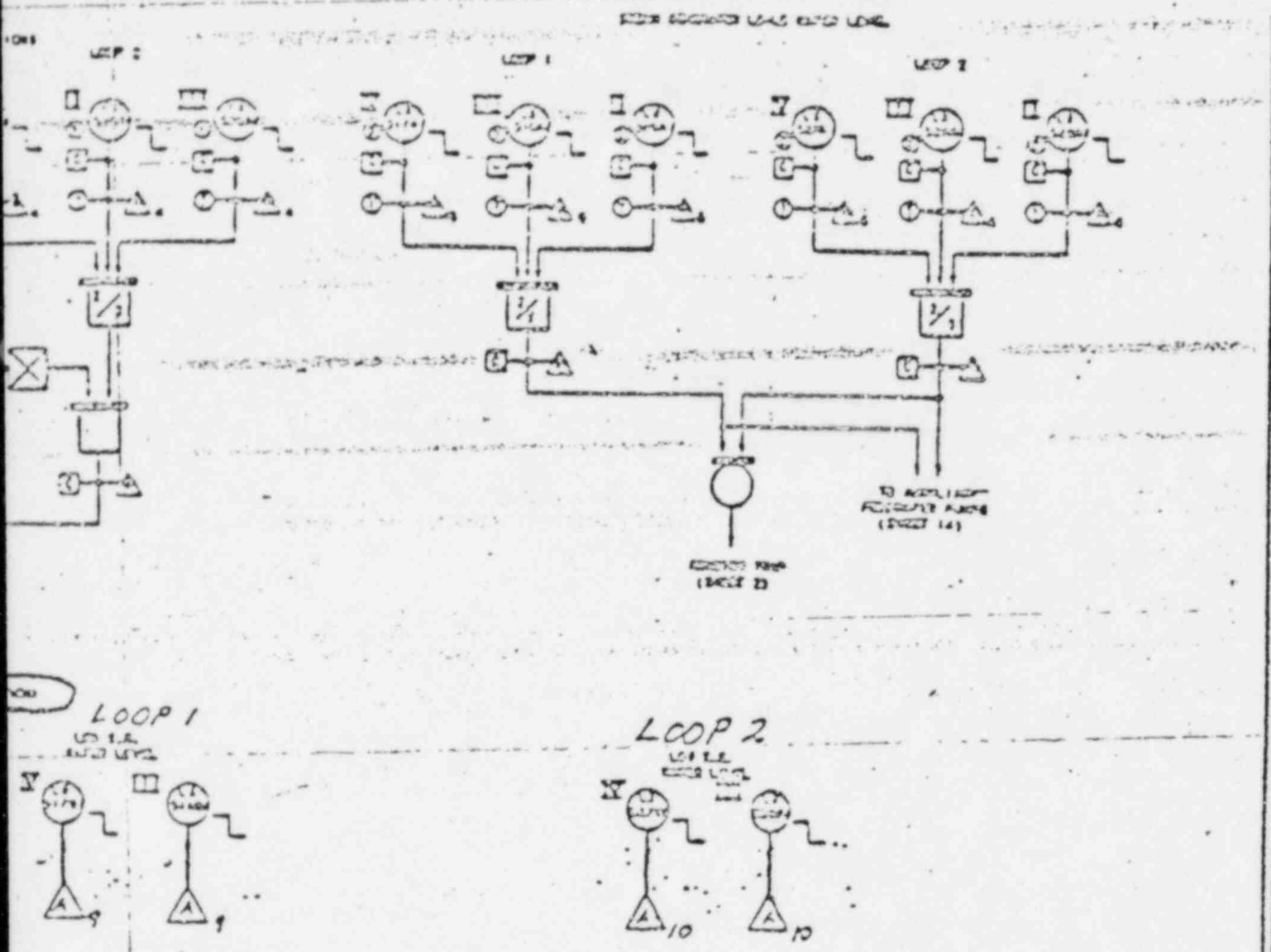
THE LOW FEEDWATER FLOW TRIP (ACTUALLY TRIP ON LOW S/G LEVEL COINCIDENT WITH STEAM / FEED MISMATCH) IS DELETED TO ALLOW REPLACING TWO SAFETY - GRADE FEEDWATER FLOW CHANNELS WITH A SINGLE CONTROL - GRADE CHANNEL.

WITH THE LOW FEEDWATER FLOW TRIP DELETED, NONE OF THE THREE SAFETY - GRADE S/G LEVEL CHANNELS CAN BE USED FOR FEEDWATER CONTROL. A FOURTH S/G LEVEL CHANNEL (CONTROL - GRADE) IS ADDED TO EACH STEAM GENERATOR TO PROVIDE LEVEL SIGNAL FOR CONTROL SYSTEM. THE NEW STEAM GENERATOR CHANNEL IS "TEED - OFF" FROM EXISTING S/G TAPS. (IEEE - 279 CONSIDERATIONS)

ORIGINAL TRIP SIGNALS



KPK FEEDWATER BYPASS SYSTEM MODIFICATION
- DELETE LOW FEEDWATER FLOW REACTOR TRIP



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CONTROL SYSTEM IMPACT

- NEW CONTROL - GRADE S/G LEVEL CHANNEL USED FOR FEEDWATER CONTROL

- ONE CONTROL GRADE FEEDWATER FLOW CHANNEL REPLACES TWO SAFETY - GRADE CHANNELS

- TOTAL FEEDWATER FLOW SIGNAL MUST BE DERIVED BY ELECTRICALLY SUMMING BYPASS LINE AND MAIN LINE FLOW SIGNALS

- AUTOMATIC STEAM GENERATOR LEVEL CONTROL CONTINUES TO BE PERFORMED BY THE THREE - ELEMENT CONTROLLER ACTING ON THE MAIN CONTROL VALVE (FCV). THE TOTAL FEEDWATER FLOW SIGNAL IS AN INPUT TO THIS CONTROLLER.

- SEPARATE NEW CONTROLLER IS USED TO AUTOMATICALLY OPEN THE AUXILIARY CONTROL VALVE (FACV) SO THAT THE PREHEATER FLOW DOES NOT EXCEED 70 PERCENT AT HIGHER POWER LEVELS. THE FACV POSITION IS PROGRAMMED AS A FUNCTION OF THE TOTAL FEEDWATER FLOW

CONTROL SYSTEM IMPACT (CONT'D.)

- NEW INDICATORS

  PREHEATER FLOW, NEW S/G LEVEL SIGNAL, FACV CONTROL SIGNAL

- NEW STATUS LIGHTS

  FACV OPEN / CLOSED LIGHTS

- NEW CONTROLS

  ADD FACV AUTO / MANUAL STATION

  DELETE FEEDWATER FLOW CHANNEL SELECTOR SWITCH

- NEW ALARM

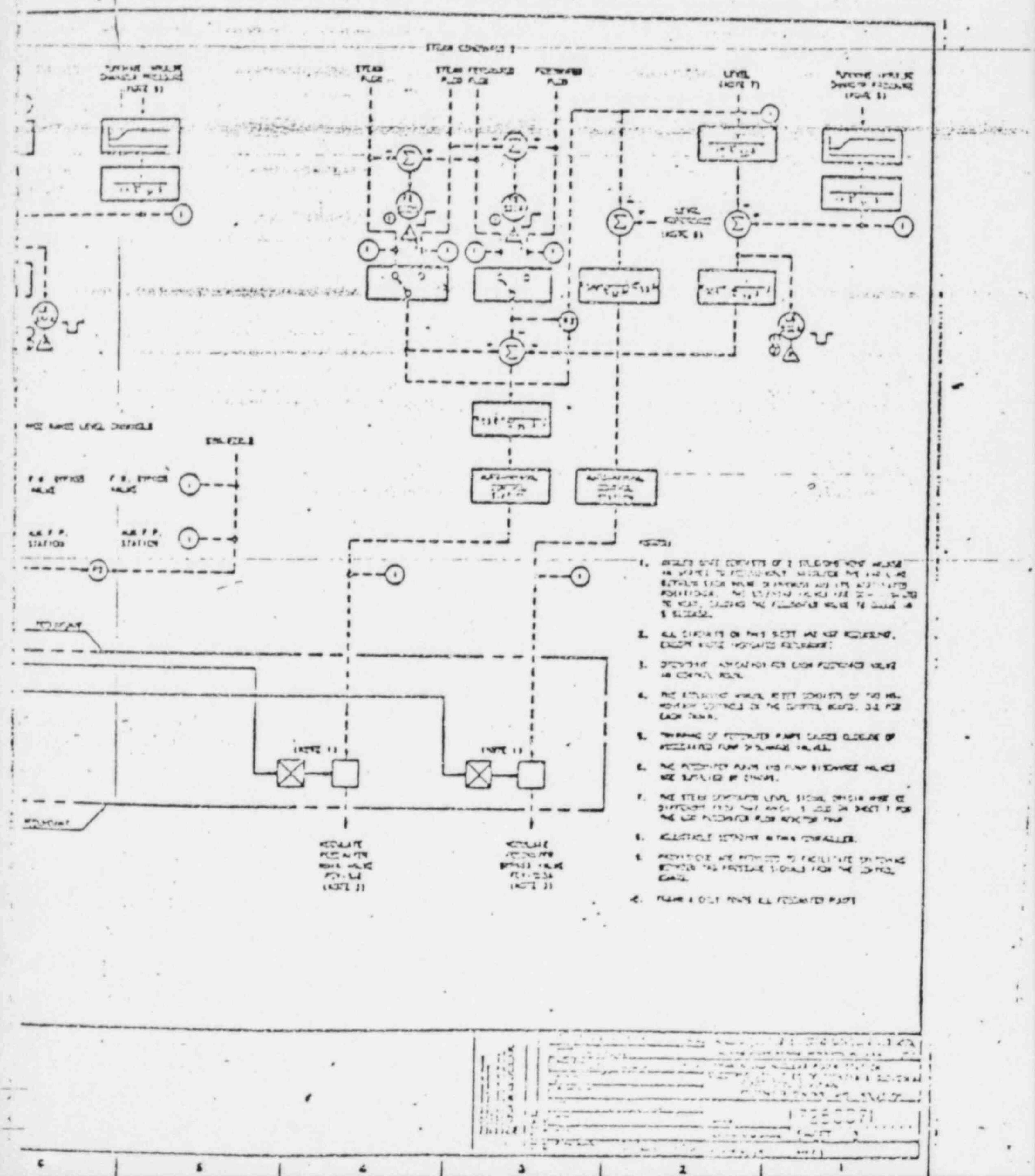
  HIGH PREHEATER FLOW ALARM - TIMER PRO[illegible] TO PREVENT ALARM ACTUATION DURING TRANSIENTS

- NEW TRANSFER FUNCTIONS

  FILTER ON FLOW SIGNAL USED FOR FACV CONTROL

  FUNCTION GENERATOR TO DERIVE FACV CONTROL SIGNAL FROM FILTERED FLOW SIGNAL

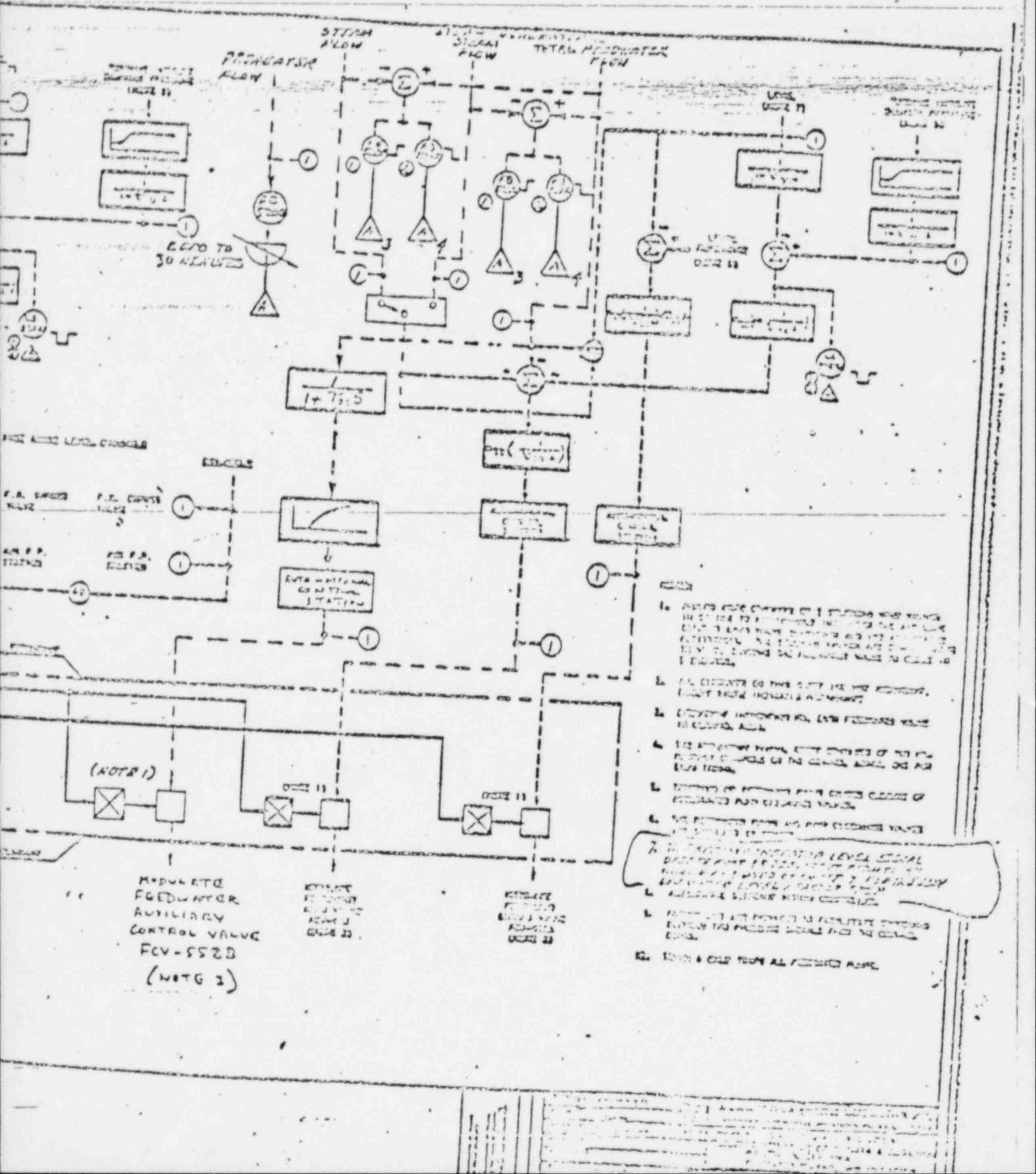
ORIGINAL CONTROL SYSTEM



NNK FEEDWATER BYPASS SYSTEM MODIFICATION

- PROGRAM FCBV OPENING AS A FUNCTION

FEEDWATER FLOW



CONTROL SYSTEM STABILITY

ANALYTICAL RESULTS

- SYSTEM AS PROPOSED USING FEED FLOW SIGNAL FOR FACV CONTROL IS STABLE FOR ALL REASONABLE LOOP GAINS. INCREASING GAIN BY FACTOR OF 5 LED TO INSTABILITY, BUT THIS COULD BE CORRECTED BY INCREASING FILTER TIME CONSTANT.

- SYSTEM BASED ON FACV CONTROL WITH STEAM FLOW SIGNAL HAS MORE MARGIN TO INSTABILITY.

NOTE: STEAM FLOW SIGNAL WAS EVALUATED AS AN ALTERNATIVE TO TOTAL FEED FLOW SIGNAL. CHOICE FEED FLOW SIGNAL FOR CONTROL WAS BASED ON CONSIDERATIONS OTHER THAN STABILITY:

- FEED FLOW SIGNAL HAS MORE STEADY - STATE ACCURACY (2 - 3% IMPROVEMENT)

- FEED FLOW SIGNAL IS LESS NOISY THAN STEAM FLOW SIGNAL (EXPERIENCE)

- USE OF FEED FLOW SIGNAL MINIMIZES FLOW OVERSHOOT INTO PREHEATER FOLLOWING LOAD REJECTION AND OTHER TRANSIENTS

PREHEATER BYPASS

SYSTEM IMPACT

FEEDWATER BYPASS SYSTEM LOGIC CHANGES

- FLOW SWITCHOVER FROM THE AUXILIARY TO THE MAIN NOZZLE IS NOW PERFORMED BY MANUALLY OPENING THE FCV AND CLOSING THE FCBV. VALVES ARE MANUALLY OPERATED IN REVERSE ORDER FOR SWITCHOVER FROM MAIN TO AUXILIARY NOZZLE.

- HIGH FEEDWATER TEMPERATURE SIGNAL IS NOW USED TO OPEN FIV IN PLACE OF HIGH FLOW SIGNAL. EITHER HIGH FLOW OR HIGH TEMPERATURE WILL KEEP THE FIV OPEN (LOAD REJECTION CONSIDERATION).

- FCV - CLOSED SIGNAL IS NOW USED AS PERMISSIVE FOR FEEDWATER PURGING USING FPV AND FIBV.

- FCV - CLOSED SIGNAL IS NOW USED TO SHUT OFF AUXILIARY NOZZLE TEMPERING FROM USING FBTV.

ORIGINAL LOGIC

WESTINGHOUSE PROPRIETARY CLASS 2

12,397-4

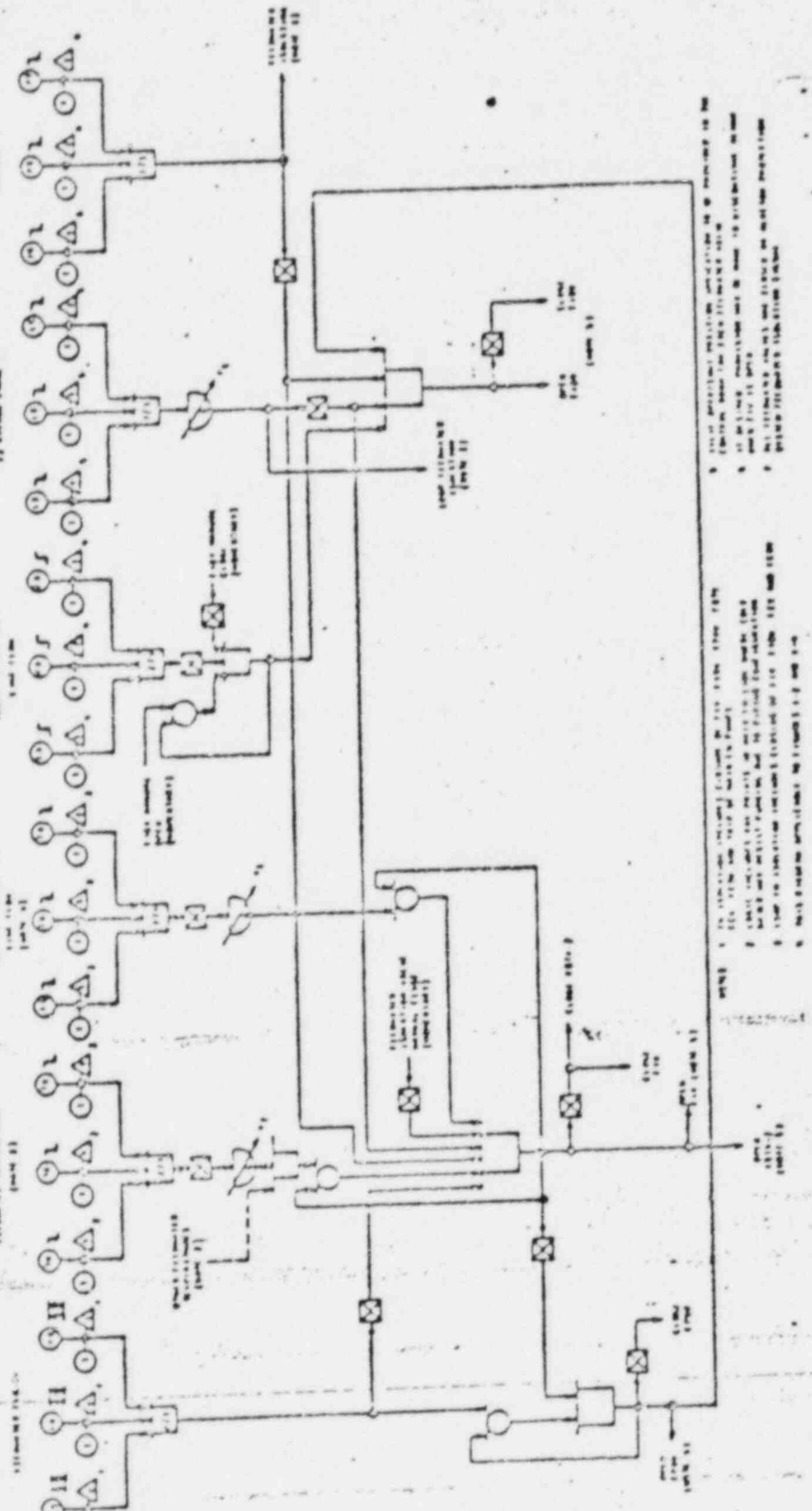


Figure I-5. Logic Diagram — Feedwater Bypass System (Forward Flushing)

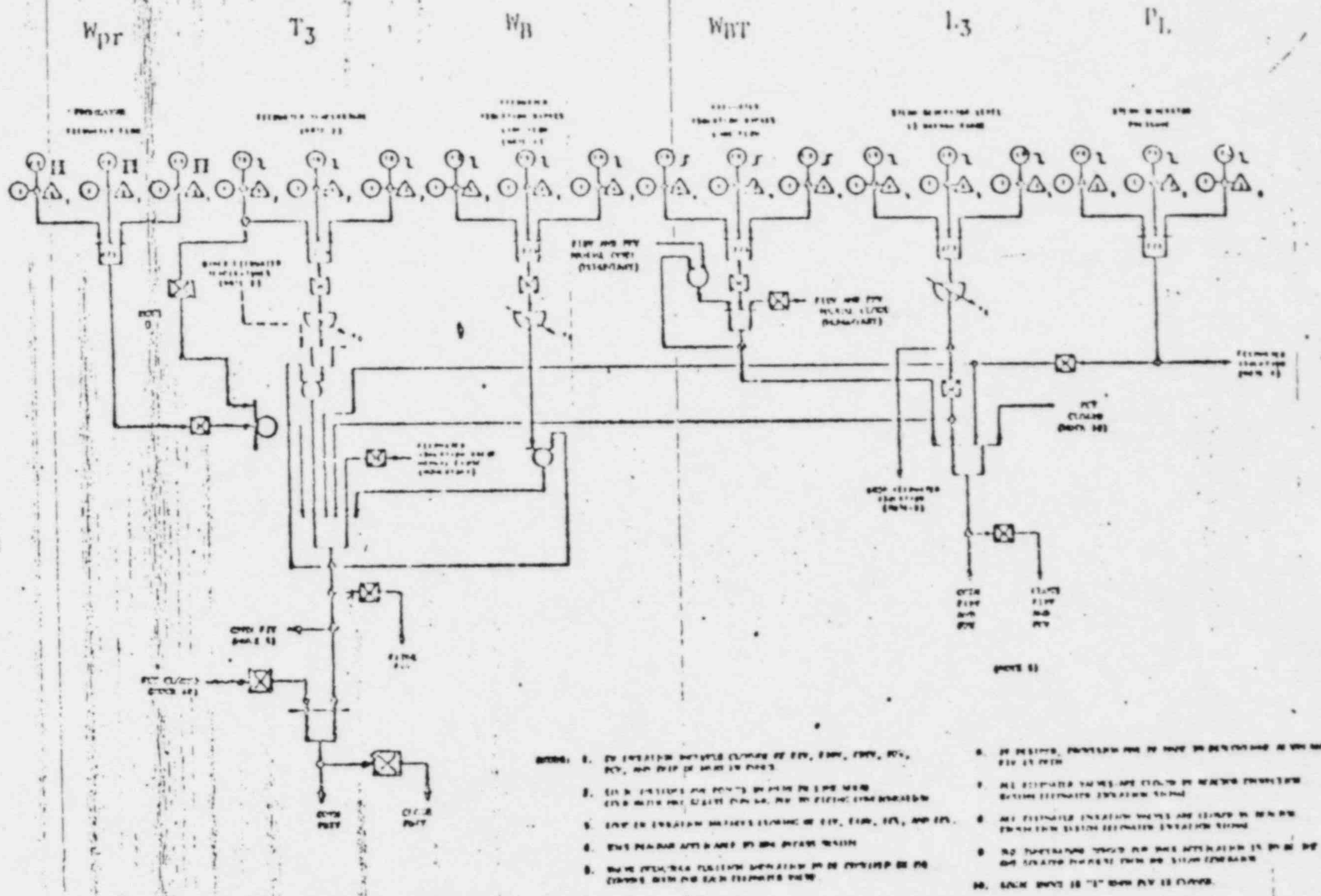


Figure 3-1. Logic Diagram - Steam Generator Preheater Waterhammer Control

KRK FEEDWATER MIX

MAIN CONTROL BOARD CHANGE

1. CHANGES SCALE ON INDICATORS FI-510 AND FI-520

2. CUT CONTROL BOARD, INSTALL AND WIRE TWO INDICATORS LI-531 AND LI-532

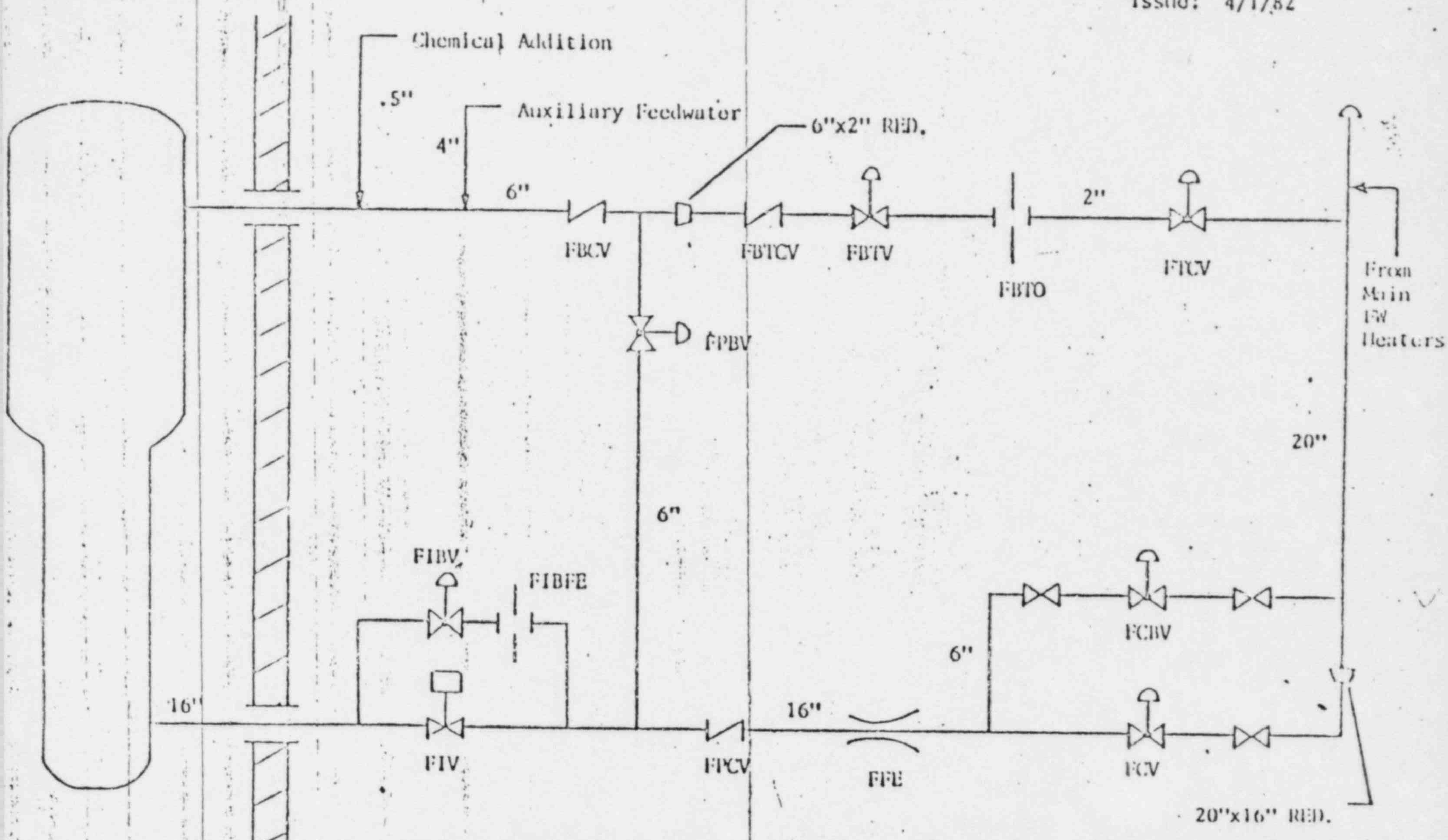
3. REMOVE SWITCH MODULES FS-510Z (ITEM 149) AND FS-520Z (ITEM 150), ENLARGE TWO CUTOUT, INSTALL ADAPTER PLATES AND INSTALL AUTO/MANUAL STATIONS FK-551B AND FK-552B

4. PROVIDE BLANK LIGHT BOX LENS FOR ANNUNCIATOR LIGHT BOX AL805 AND FOR FEEDWATER VALVES STATUS LIGHT BOX (CONTROL BOARD SECTION B)

NUCLEAR SAFETY EVALUATION

KRSKO FEEDWATER BYPASS SYSTEM: PRESENT ARRANGEMENT

Issue: 4/1/82



FIV - Feedwater Isolation Valve
FIBV - Feedwater Isolation Bypass Valve
FIBFE - Feedwater Isolation Bypass Flow Element
FPCV - Feedwater Preheater Check Valve
FCV - Feedwater Control Valve
FCBV - Feedwater Control Bypass Valve

FBCV - Feedwater Bypass Check Valve
FBTCV - Feedwater Bypass Tempering Check Valve
FBTV - Feedwater Bypass Tempering Valve
FBTO - Feedwater Bypass Tempering Orifice
FTCV - Feedwater Tempering Control Valve

SAFETY EVALUATION

- FEEDWATER SYSTEM MODIFICATIONS

- ASSOCIATED SYSTEMS AND COMPONENTS
  - DESIGN PARAMETER CHANGES
  - MECHANICAL SYSTEMS AND COMPONENTS
  - CONTROL AND PROTECTION SYSTEMS
  - ACCIDENT ANALYSES
  - OPERATOR ACTION

- FSAR REVIEW

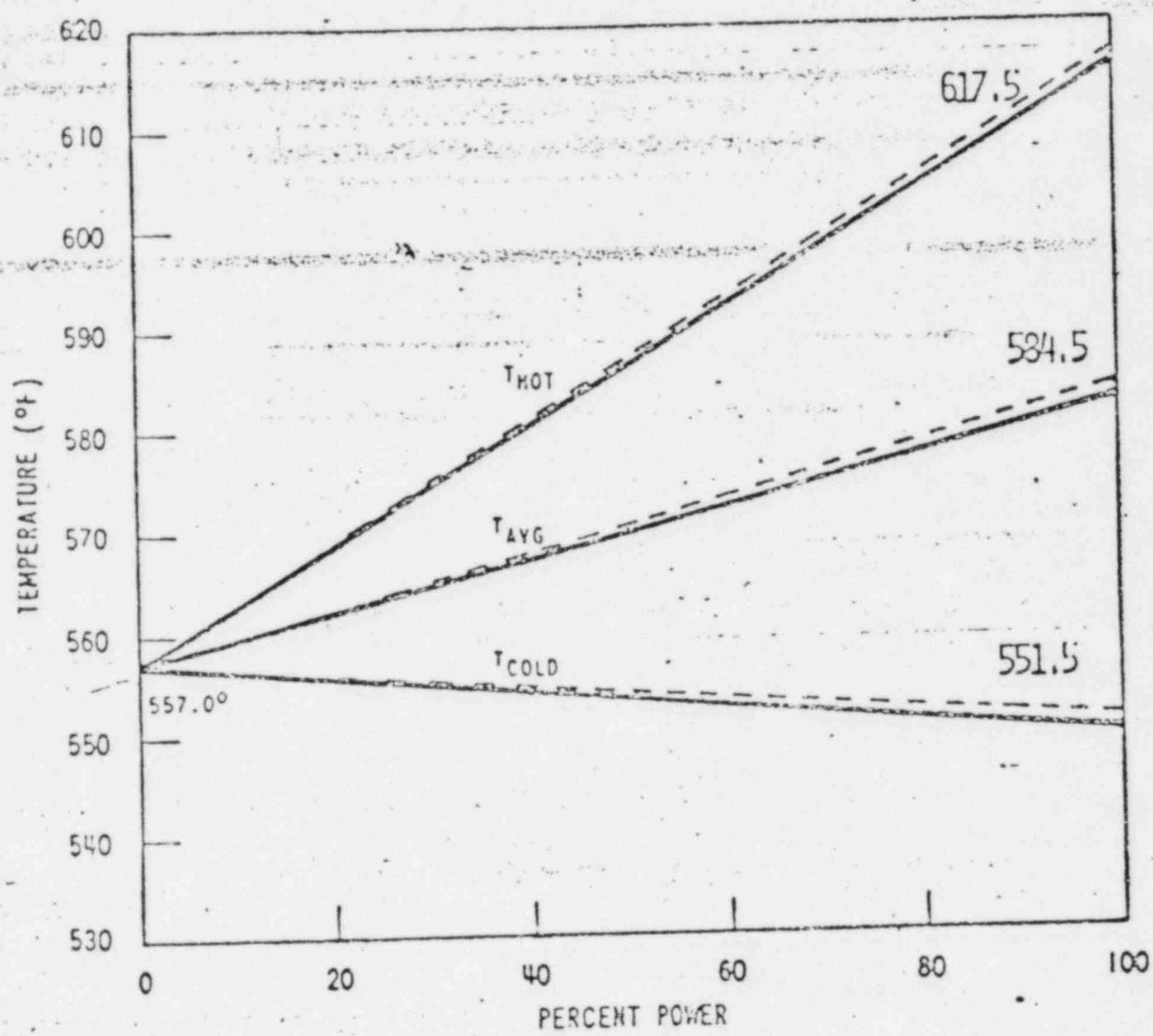
COMPARISON OF STEAM GENERATOR PARAMETERS

Design Parameter	Present System	Proposed System
Steam Temperature, °F	534.6	534.6
Steam Pressure, psia	920	920
Steam Flow, 10^6 lb/hr total	8.17	8.17
Feedwater Temperature*, °F	430	430
Moisture, % maximum	0.25	0.25

*Six stages of feedwater heaters in operation

COMPARISON OF REACTOR COOLANT SYSTEM PARAMETERS

Thermal and Hydraulic Design Parameters	Design Conditions Current	Design Conditions Proposed
NSSS Power, MWt	1882	1882
Reactor Core Heat Output, MWt	1876	1876
System Pressure, Normal psia	2250	2250
Total Inlet Thermal Flow Rate, gpm	189,000	189,000
Total Inlet Thermal Flow Rate, lbm/hr	71.05×10^6	70.9×10^6
Core Effective Flow Rate for Heat Transfer, lbm/hr	67.8×10^6	67.7×10^6
Reactor Coolant System Temperatures, °F		
Nominal Reactor Vessel/Core Inlet	549.9	551.5
Average Rise in Vessel	66.2	66.0
Average in Vessel	583.0	584.5
Average Rise in Core	69.0	68.8
Average in Core	586.3	587.8
No Load	557.0	557.0



REACTOR COOLANT SYSTEM TEMPERATURE - PERCENT POWER MAP

NE KRSKO FSAR FIGURE 5.3-1

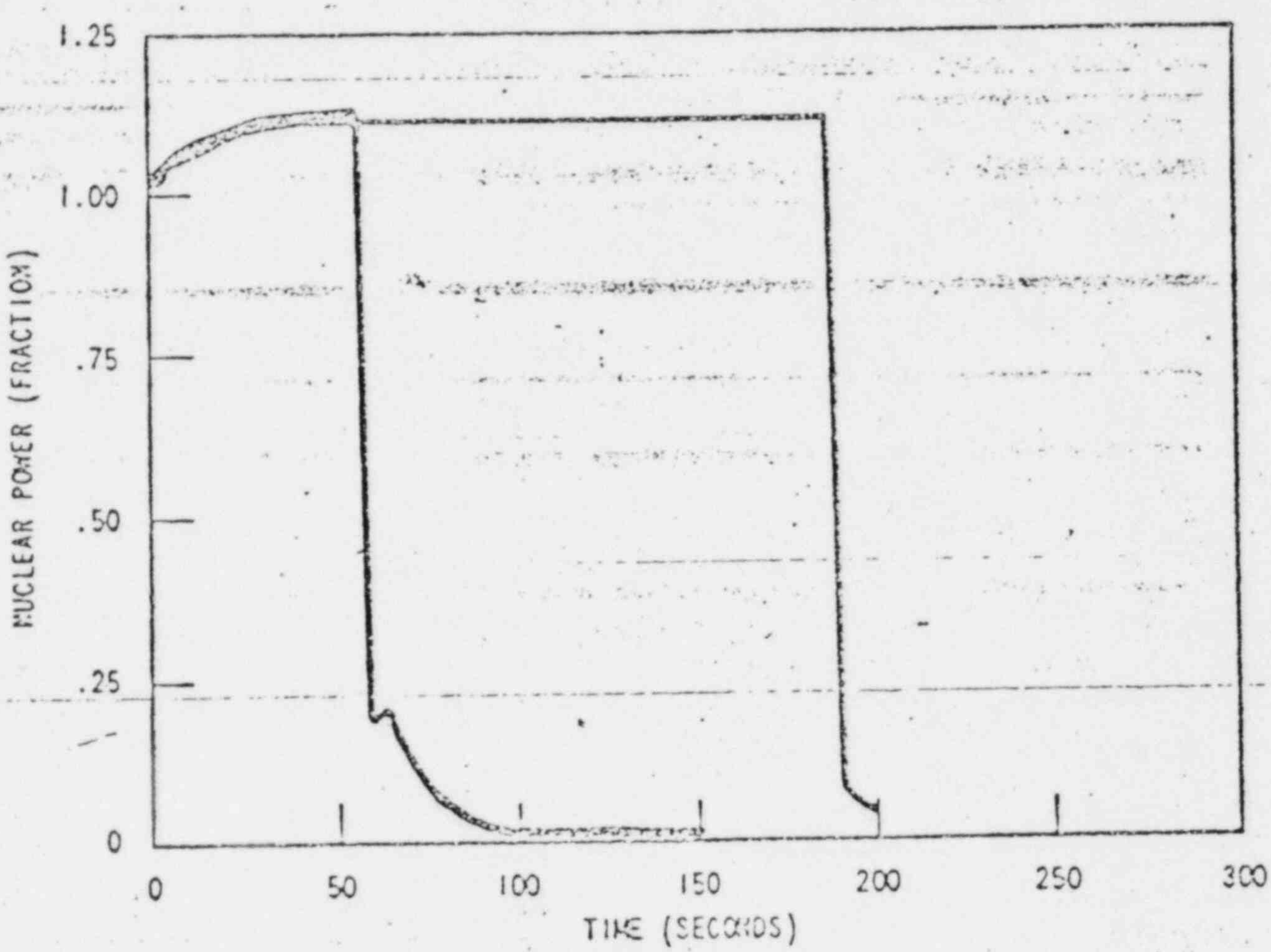
TABLE 3.3

REVIEW OF MECHANICAL SYSTEMS AND COMPONENTS AFFECTED BY FEEDWATER SYSTEM MODIFICATIONS*

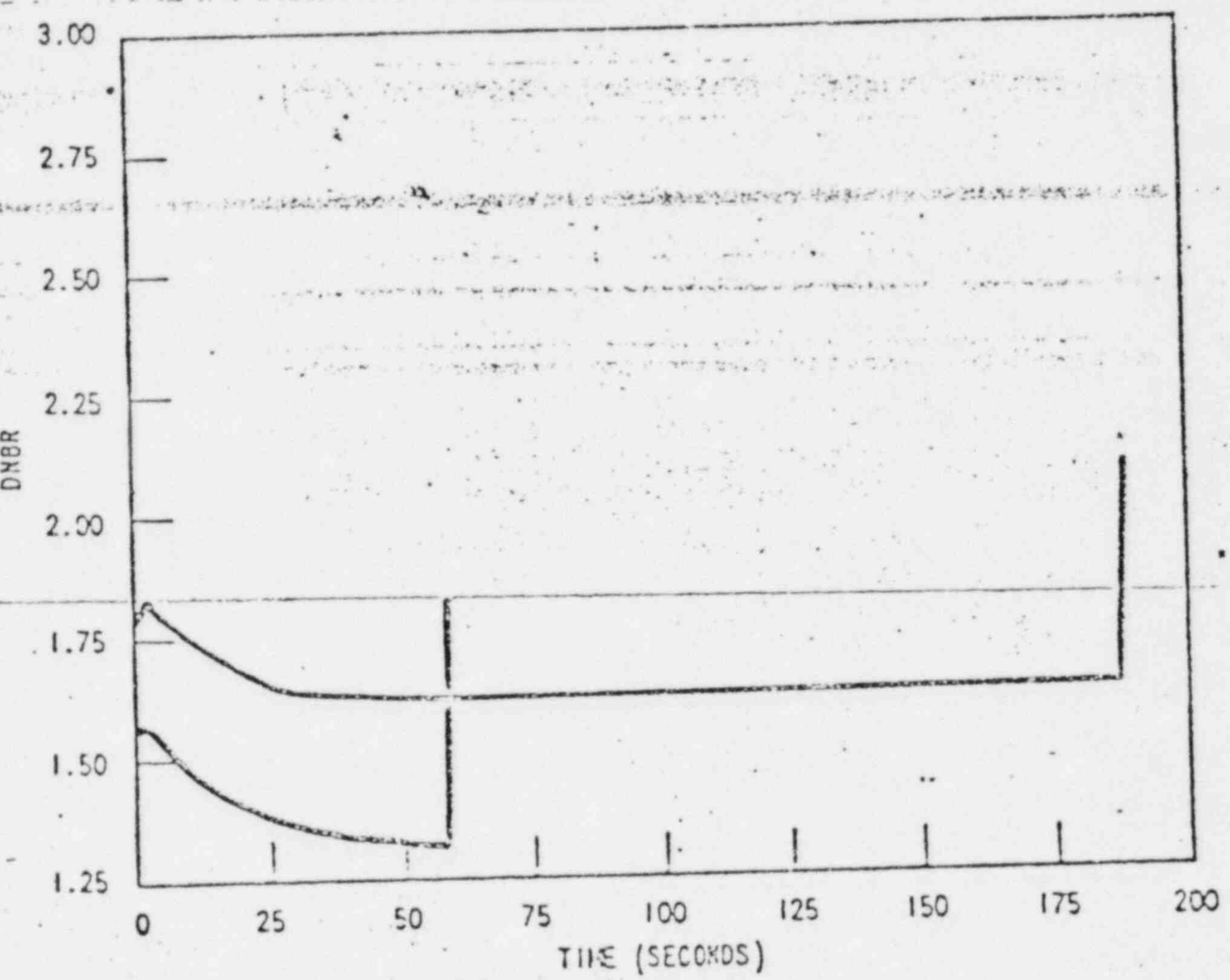
System/Component	Affected? Normal	Transient	Design Document(s) Revisions Required	Remarks
Reactor Coolant System	Yes	Yes		
Reactor Vessel	Yes	Yes		
CRDM Housings	No	No	-	
Steam Generators	Yes	Yes		
Pressurizer	No	Yes	No	
Loop Piping/Fittings	Yes	Yes	No	
RTD Bypass Manifold	Yes	Yes	No	
RC Thermowell/Boss	Yes	Yes	No	
Safety Valves	No	Yes	No	
PORVs	No	No	-	
RCPB Valves	Yes	Yes	No	
Reactor Coolant Pumps	Yes	Yes		
Chemical and Volume Control System				
Regenerative Hx				
Letdown Hx				
Demineralizers				
Reactor Coolant Filter				
Volume Control Tank				
CC/PD Pumps				

TABLE 3.3 (Continued)

System/Component	Affected? Normal	Affected? Transient	Design Decoment(s) Revisions Required	Remarks
Seal Water Filters				
Letdown Orifices				
Excess Letdown HX				
Seal Water Hx				
Boric Acid Tanks				
BA Transfer Pump				
BA Blender				
BA Filter				
BA Tank Orifice				
RCP Seal Bypass Orifice				
Boron Thermal Regeneration System				
Moderating HX				
Letdown Chiller HX				
Letdown Reheat HX				
Thermal Regen, Demineralizer				
Emergency Core Cooling System	No	No	-	
Residual Heat Removal System	No	No		
Boron Recycle System	No	No	-	

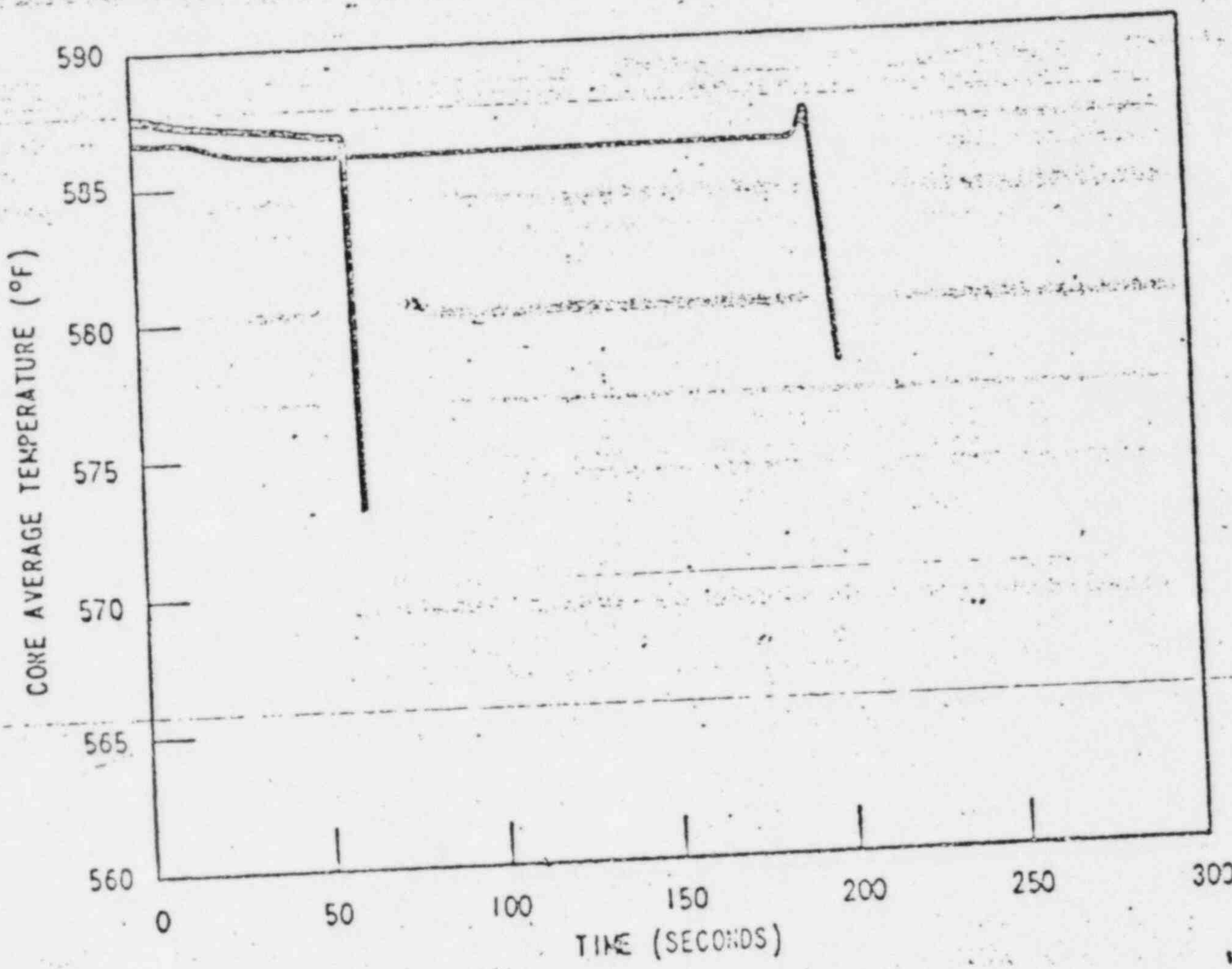


EXCESS FEED ACCIDENT FEED CONTROL
VALVE FAILURE AT FULL LOAD



EXCESS FEED ACCIDENT

NE KRSKO | FSAR | Fig. 15.2-26



EXCESS FEED ACCIDENT FEED CONTROL
VALVE FAILURE AT FULL LOAD

NE KRSKO FSAR Fig. 15.2-25

FSAR REVIEW

o MECHANICAL SYSTEMS AND COMPONENTS

3.9	MECHANICAL SYSTEMS AND COMPONENTS
3.11	ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT
5.3	THERMAL HYDRAULIC SYSTEM DESIGN
5.5	COMPONENT AND SUBSYSTEM DESIGN
9.2	WATER SYSTEMS (CCWS)
9.3	PROCESS AUXILIARIES (CVCS)
10.4.7	CONDENSATE AND FEEDWATER SYSTEMS

o CONTROL AND PROTECTION SYSTEMS

3.10	SEISMIC DESIGN OF CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT
5.6	INSTRUMENTATION APPLICATION
7.2	REACTOR TRIP SYSTEM
7.7	CONTROL SYSTEMS NOT REQUIRED FOR SAFETY

FSAR REVIEW

- ACCIDENT ANALYSIS

6.2.1	MASS/ENERGY RELEASE TO CONTAINMENT
15.2.8	LOSS OF NORMAL FEEDWATER
15.2.10	EXCESS HEAT REMOVAL DUE TO FEEDWATER SYSTEM MALFUNCTIONS
15.4.1	MAJOR REACTOR COOLANT SYSTEM PIPE RUPTURE (LOCA)

- TECHNICAL SPECIFICATIONS

16.3	LIMITING CONDITIONS FOR OPERATION (STEAM AND POWER CONVERSION SYSTEMS)
16.4	SURVEILLANCE REQUIREMENTS

FSAR REVIEW SCHEDULE

WORKING DRAFT -- APRIL 23

TYPED PROOF's — MAY 1

FSAR AMENDMENT — MAY 20

EQUIPMENT DELIVERY

STATUS

KRSKO STEAM G

ITEM DESCRIPTION	RESPONSIBLE PARTY	P.O. NO.	ESTIMATED DELIVERY
1. 4 Thermowells with RTD's.	WNI-Grogan/Pavick	KSG-022	5/05/82
2. 2 MCB Level Indicators for S/G Level.	WNI-Grogan	KSG-023	D 4/09/82
3. 2 FW Flow Venturi's for 8 Inch Bypass Line.	WNI-Moore/Brady	KSG-035	5/08/82
4. 2 Pressure Transmitters for FW Flow Venturi's.	WNI-Grogan/Pavick	KSG-019	5/05/82
5. 6 Inch Pipe Caps Safety Class Quantity 4.	WNI-Moore/Brady	KSG-016	S 4/15/82
6. Ten 8 Inch to 6 Inch Reducers, Seven 8 Inch Schedule 100 Long Radius ELL, Eight 2 Inch 3000 LB. SW Boss, Eight 2 Inch 3000 LB. SW Half Coupling, Four 6 Inch Weldolets, Four 6 Inch Schedule 80 ELLS, Four 8X10 Reducers.	WNI-Moore/Brady	KSG-017	S 4/08/82 S 4/16/82 4/26/82
7. Hangers	WNI-Moore/Fullen	KSG-021	Partial S - 4/16/82. Balance R/A - 4/23/82, 4/30/82
8. #2 Narrow Range S/S Level Inst.	SOD-Beacomb	KSG-037	S 4/1/82
9. Four Instrument Isolation. Valves for Blowdown Connections on NR S/G Level - Safety Valves.	SITE-Comoletti	-	Available
10. Eight Instrument Isolation Valves for Feedwater Flow Transmitters Non-Safety 3/4T-38.	SITE-Comoletti	-	Available

ITEM DESCRIPTION	RESPONSIBLE PARTY	P.O. NO.	ESTIMATED DELIVERY
11. Safety Class Instrument Tubing - 50 Meters.	SITE-Comoletti	-	Available
12. Non-Safety Instrument Tubing	SITE-Comoletti	-	Available
13. Safety Class Piping 5.5 Meters 6 Inch Schedule 80 1-90 Degree Bend - 5D Radius 6 Inch Schedule 80 1-180 Degree Bend - 5D Radius 6 Inch Schedule 80.	SITE-Reuler	KSG-020	D 4/08/82
14. Non-Safety Piping 54.4 Meters 8 Inch Schedule 100 9 Each 90 Degree Bends - 5D Radius 8 Inch Schedule 100.	SITE-Reuler	KSG-020	D 4/08/82
15. 7300 Cards - NSSS	SOD-Chambers	N/A	4/30/82
16. 7300 Cards - 8 New for BOP Racks	WNI-Grogan/Pavick	KSG-032	30 Weeks to Replace, Cards Available.
17. Deleted			
18. Piping Insulation 12 Meters of 6 Inch Pipe 60 Meters of 8 Inch Pipe Small Quantity of 2 Inch Pipe Small Quantity of Tubing.	SITE-Melville	-	Available
19. Twisted Shielded Pair Cable 2 S/G Level to Process Rack 2 Venturi to Process Rack Miscellaneous Other Cable.	SITE-Janes	-	Available

ITEM DESCRIPTION	RESPONSIBLE PARTY	P.O. NO.	ESTIMATED DELIVERY
20. 12 W Relays	WNI-Moore	KSG-036	5/01/82
21. 2 - 6 Inch - 900 LB.	WNI-Moore/Powell	KRA-159	4/30/82
22. 2 - 10 Inch Flow Control Valves.	SITE-McKeown Spain	N/A	6/01/82
23. Amp Connectors.	WNI-Bohn	KSG-038	4/21/82
24. 2 - Control Board Manual Auto Stations.	SOD-Greisheimer	KSG-039	4/30/82
25. Deleted			
26. 3/4" Conduit + Connections	WNI-Bohn	KSG-033	S 4/16/82
27. 4 ASCO Solenoids	WNI-Sitler	KRA-631	5/15/82
28. MCB Wire	WNI-Bohn	KSG-041	5/15/82
29. Burndy Lugs	WNI-Bohn	KSG-042	4/23/82

Revised FSAR pages

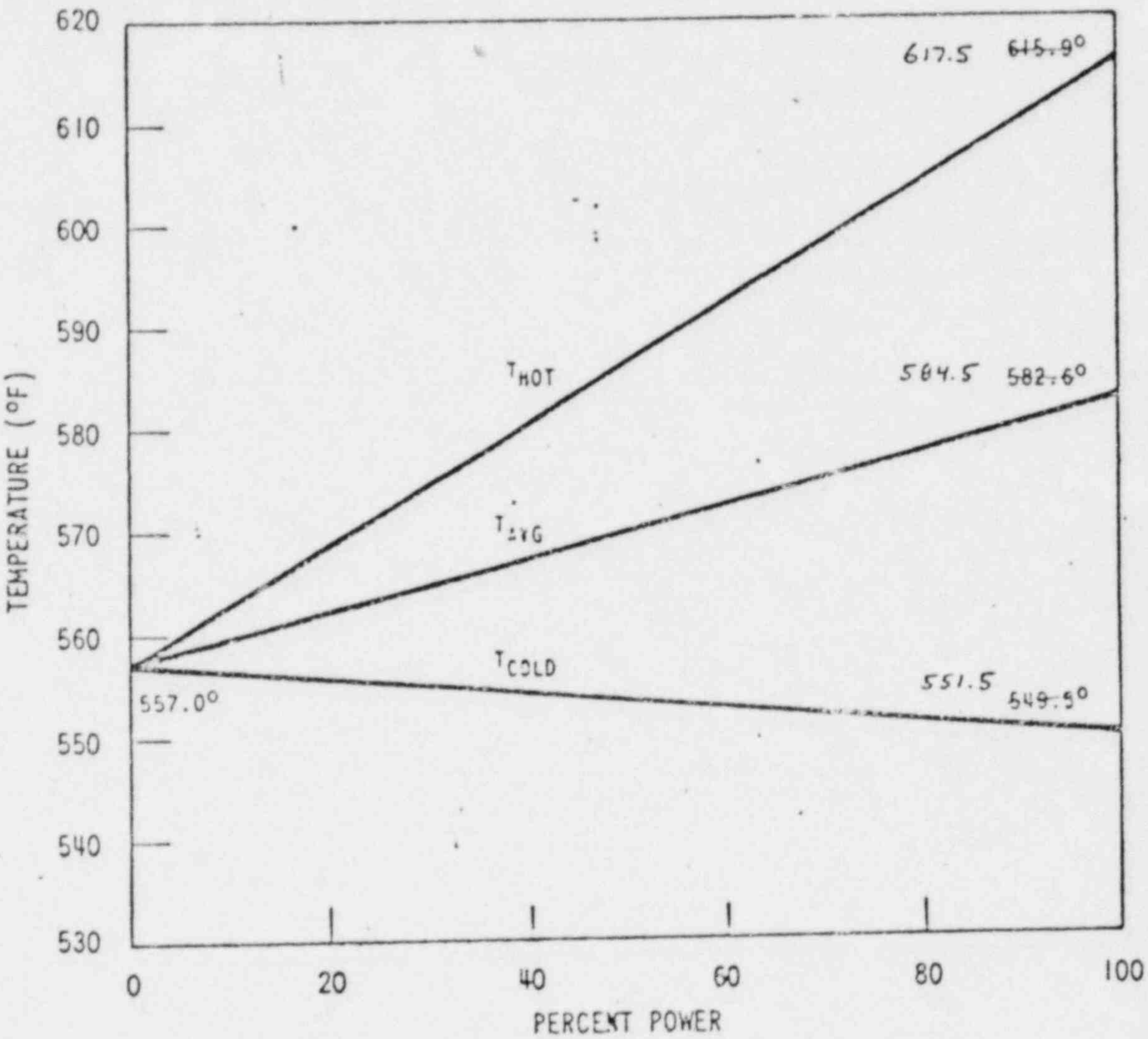
TABLE 5.1-1

REACTOR COOLANT SYSTEM DESIGN AND OPERATING PARAMETERS
FOR NORMAL STEADY-STATE FULL POWER OPERATING CONDITIONS

Nominal Operating Pressure, psig	2235
Total System Volume (including pressurizer and surge line), cu. ft.	6423
System Liquid Volume (including pressurizer water at maximum guaranteed power), cu. ft.	6011
Pressurizer Heater Capacity, kW	1,000
Pressurizer Relief Tank Volume, cu. ft.	1,100

System Thermal and Hydraulic Data
(Based on Thermal Design Flow)

Total Primary Heat Output, MWt	1882
Thermal Design Flows, gpm	
Loop	94,500
Reactor	189,000
Total Reactor Flow, 10^6 lb/hr	~~71.1~~ 70.9
Temperatures, °F	
Reactor Vessel Outlet	~~615.9~~ 617.5
Reactor Vessel Inlet	~~549.5~~ 551.5
Steam Generator Steam	~~535.1~~ 534.6
Feedwater	430.0
Steam Pressure, psia	920
Total Steam Flow, 10^6 lb/hr	8.17
Best Estimate Flows, gpm	
Loop	~~101,400~~ 100,700
Reactor	~~202,800~~ 201,600
Mechanical Design Flows, gpm	
Loop	~~106,500~~ 104,700
Reactor	~~213,000~~ 209,400



REACTOR COOLANT SYSTEM TEMPERATURE - PERCENT POWER MAP

NE KRSKO	FSAR	FIGURE 5.3-1

5.6 INSTRUMENTATION APPLICATION

Process control instrumentation is provided for the purpose of acquiring data for the key process parameters of the reactor coolant system (including the reactor coolant pump motors) as well as for the residual heat removal system. The pick-off points for the reactor coolant system are shown in the three sheets of the flow diagrams (Figure 5.1-1); and for the residual heat removal system, in flow diagram Figure 5.5-4. In addition to providing input signals for the protection system and the plant control systems, the instrumentation sensors furnish input signals for monitoring and/or alarming purposes for the following parameters:

1. Temperatures
2. Flows
3. Pressures
4. Water levels

In general there input signals are used for the following purposes:

1. Provide input to the reactor trip system for reactor trips as follows:

    a. Overtemperature ΔT
    b. Overpower ΔT
    c. Low pressurizer pressure
    d. High pressurizer pressure
    e. High pressurizer water level
    f. Low primary coolant flow

    It is noted that the following parameters, which are also sensed to generate an input to the reactor trip system, while not part of the reactor coolant system, are included here for purposes of completeness:

~~g. Low feedwater flow~~

g ~~h.~~ Low low steam generator water level

2. Provide input to the engineered safety features actuation system as follows:

   a. Pressurizer low pressure

   It is noted that the following parameters, which are also sensed to generate an input to the engineered safety features actuation system, while not part of the reactor coolant system, are included here for purposes of completeness:

   b. Low steam line pressure

   c. Hi-Hi steam flow or High steam flow coincident with low-low (T_{avg})

   d. Hi-1 containment pressure

   e. Hi-2 containment pressure

   f. Hi-3 containment pressure

3. Furnished input signals to the nonsafety-related system, such as the plant control systems and surveillance circuits so that:

   a. Reactor coolant average temperature (T_{avg}) will be maintained within prescribed limits. The resistance temperature detector instrumentation is identified on Figure 5.1-1, Sheet 3.

Thus an analysis of smaller pump suction breaks is representative of the spectrum of break sizes.

The LOCA analysis calculational model is typically divided into three phases which are: 1) blowdown, which includes the period from accident occurrence (when the reactor is at steady state full power operation) to the time when zero break flow is first calculated, 2) refill, which is from the end of blowdown to the time the ECCS fills the vessel lower plenum, and 3) reflood, which begins when water starts moving into the core and continues until the end of the transient. For the pump suction break, consideration is given to a possible fourth phase; that is, froth boiling in the steam generator tubes after the core has been quenched. For a description of the calculational model used for the mass and energy release analysis, see Reference 20.

Basis of the Analysis

1. Assumptions

   The following items ensure that the core energy release is conservatively analyzed for maximum containment pressure.

   a. Maximum expected operating temperature (~~616.1~~ 617.5°F)

   b. Allowance in temperature for instrument error and dead band (+4°F)

   c. Margin in volume (1.4%)

   d. Allowance in volume for thermal expansion (1.6%)

   e. Margin in core power associated with use of engineered safeguards design rating (ESDR)

   f. Allowance for calorimetric error (2% of ESDR) prevent spurious trips caused by short term voltage perturbations. The coincidence logic and interlocks are given in Table 7.2-1.

d. Reactor coolant pump bus underfrequency trip

This trip is required to protect against low flow resulting from bus underfrequency, for example a major power grid frequency disturbance. The function of this trip is to trip the reactor for an underfrequency condition. The setpoint of the underfrequency relays is adjustable between 44 and 49 Hz.

There are two underfrequency sensing relays connected to each reactor coolant pump bus. Signals from relays connected to the buses (time delayed up to approximately 0.1 seconds to prevent spurious trips caused by short term frequency perturbations) will trip the reactor if the power is above P-7.

Figure 7.2-1, Sheet 5, shows the logic for the Reactor Coolant System low flow trips.

~~5. Steam Generator Trips~~

~~The specific trip functions generated are as follows:~~

~~a. Low feedwater flow trip~~

~~This trip protects the reactor from a sudden loss of the heat sink. The trip is actuated by steam/feedwater flow mismatch (one out of two) in coincidence with low water level (one out of two) in any steam generator.~~

~~Figure 7.2-1, Sheet 7, shows the logic for this trip function.~~

~~There are no interlocks associated with this trip.~~

5. ~~b.~~ Low-low steam generator water level trip

This trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch of insufficient magnitude to cause a low feedwater flow reactor trip. This trip is actuated on two out of three low-low water level signals occurring in any steam generator.

The logic is shown on Figure 7.2-1, Sheet 7.

6. Turbine Trip - Reactor Trip (anticipatory)

The turbine trip-reactor trip is actuated by two-out-of-three logic from emergency trip fluid system signals or by all closed signals from the turbine steam stop valves. A turbine trip causes a direct reactor trip above P-7. The reactor trip on turbine trip provides additional protection and conservatism beyond that required for the health and safety of the public. This trip is included as part of good engineering practice and prudent design. No credit is taken in any of the safety analyses (Chapter 15) for this trip.

The turbine provides anticipatory trips to the reactor protection system from contacts which change position when the turbine stop valves close or when the turbine emergency trip fluid system pressure goes below its setpoint.

3. Protection System ranges

Protection system ranges are tabulated in Table 7.2-3. Range selection for the instrumentation covers the expected range of the process variable being monitored during power operation. Limiting setpoints are at least 5 percent from the end of the instrument span.

7.2.1.3 Final System Drawings

Functional block diagrams are furnished in Figure 7.2-1 (Sheets 1-15) and additional drawings for the I&C systems are included at the end of sections 7.2, 7.6, 7.6 and in the referenced topical reports. See Table 7.3-6 for additional references.

7.2.2 ANALYSES

7.2.2.1 Failure Mode and Effects Analyses

A failure mode and effects analysis of the Reactor Trip System has been performed. Results of this study and a fault tree analysis are presented in Reference [4].

7.2.2.2 Evaluation of Design Limits

While most setpoints used in the Reactor Protection System are fixed, there are variable setpoints, most notably the overtemperature ΔT and overpower ΔT setpoints. All setpoints in the Reactor Trip System have been selected on the basis of engineering design and safety studies. The capability of the Reactor Trip System to prevent loss of integrity of the fuel cladding and/or Reactor Coolant System pressure boundary during Condition II and III transients is demonstrated in the Safety Analysis, Chapter 15. These safety analyses are carried out using those setpoints determined from results of the engineering design studies. Setpoint limits are presented in the Technical Specifications. A discussion on the intent for each of the various reactor trips and the accident analysis (where appropriate) which utilizes this trip is presented below. It should be noted that the selected trip setpoints all provide for margin before protective action is actually required to allow for uncertainties and instrument errors. The design meets the requirements of Criteria 10 and 20 of the 1971 GDC.

7.2.2.2.1 Trip Setpoint Discussion

It has been pointed out previously that below a DNB ratio of 1.3 there is likely to be significant local fuel cladding failure. The DNB ratio existing at any point in the core for a given core design can be determined as a function of the core inlet temperature, power output, operating pressure and flow. Consequently, core safety limits in terms of a DNBR equal to 1.30 for the hot channel can be developed as a function of core ΔT, T_{avg} and pressure for a specified flow as illustrated by the solid lines in Figure 7.2-~~1a~~3. Also shown as solid lines in Figure 7.2-~~1a~~3 are the loci of conditions equivalent to 118 percent of power as a function of ΔT and T_{avg} representing the overpower (KW/ft) limit on the fuel. The dashed lines indicate the maximum permissable set point (ΔT) as a function of T_{avg} and pressure for the overtemperature and overpower reactor trip. Actual values of setpoint constants in the equation representing the dashed lines are as given in the Technical Specification, Section 16.2.3. These values are conservative to allow for instrument errors. The design meets the requirements of Criteria 10, 15, 20 and 29 of the 1971 GDC.

DNBR is not a directly measurable quantity; however, the process variables that determine DNBR are sensed and evaluated. Small isolated changes in various process variables may not individually result in violation of a core safety limit; whereas the combined variations, over sufficient time, pressurizer water level control. A failure in the level control system could fill or empty the pressurizer at a slow rate (on the order of half an hour or more), which allows ample time for corrective action by the operator.

The high water level trip setpoint provides sufficient margin such that the undesirable condition of discharging liquid coolant through the safety valves is avoided. Even at full power conditions, which would produce the worst thermal expansion rates, a failure of the water level control would not lead to any liquid discharge through the safety valves. This is due to the automatic high pressurizer pressure reactor trip actuating at a pressure sufficiently below the safety valve setpoint, or to the high pressurizer water level reactor trip.

7.2.2.3.5 Steam Generator Water Level and Feedwater Flow

The basic function of the reactor protection circuits associated with low steam generator water level ~~and low feedwater flow~~ is to preserve the steam generator heat sink for removal of long term residual heat. Should a complete loss of feedwater occur, the reactor would be tripped on ~~coincidence of steam/feedwater flow mismatch and low steam generator level or on~~ low-low steam generator water level. In addition, redundant auxiliary feedwater pumps are provided to supply feedwater in order to remove residual heat from the reactor.

Th~~ese~~is reactor trip~~s~~ acts before the steam generators are dry to reduce the required capacity and increase the time available for starting these auxiliary feedwater pumps and to minimize the thermal transient on the Reactor Coolant System and steam generators. Therefore, the following reactor trip circuit~~s~~ ~~are~~ is provided for each steam generator to ensure that sufficient initial thermal capacity is available in the steam generator at the start of the transient:

1. ~~The low feedwater flow trip detects a mismatch in steam and feedwater flow (one-out-of-two) coincident with low steam generator water levels for a steam generator in any loop.~~

~~2.~~ A low-low steam generator water level regardless of steam - feedwater flow mismatch;

It is desirable to minimize thermal transients on a steam generator for credible loss of feedwater accidents. Hence, it should be noted that controller malfunctions caused by a protection system failure effect only one steam generator; ~~the steam generator level signal used in the feedwater control originates separately from that used in the low feedwater reactor trip.~~

~~A spurious high signal from the feedwater flow channel being used for control would cause a reduction in feedwater flow preventing that channel from ultimately tripping. However, the mismatch between steam demand and feedwater flow produced by this spurious signal will actuate alarms to alert the operator of this situation in time for manual correction or, if the condition is allowed to continue, the reactor will eventually trip on a low-low water level signal independent of indicated feedwater flow.~~

A spurious low signal from the feedwater flow channel being used for control would cause an increase in feedwater flow. The mismatch between steam flow and feedwater flow produced by the spurious signal would actuate alarms to alert the operator of the situation in time for manual correction. If the condition continues, a two out of three high-high steam generator water level signal in any loop, independent of the indicated feedwater flow, will cause main feedwater pump trip and isolation and trip the turbine. The turbine trip will result in a subsequent reactor trip. The High-High Steam Generator Water Level trip is an equipment protective trip preventing excessive moisture carryover which could damage the turbine blading.

In addition, the three element feedwater controller incorporates reset action on the level error signal, such that with expected controller settings a rapid increase or decrease in the flow signal would cause only a small change in level before the controller would compensate for the level error. A slow change in the feedwater signal would have no effect at all. A spurious low or high steam flow signal would have the same effect as high or low feedwater signal, discussed above.

A spurious high steam generator water level signal from the protection channel used for control will tend to close the feedwater valve. However, before a reactor trip would occur, two out of three channels for a steam generator would have to indicate a high water level. A spurious low steam generat r water level signal will tend to open the feedwater valve. Again, before reactor trip would occur, two out of three channels in a loop would have to indicate a low water level. Any slow drift in the water level signal will permit the operator to respond to the level alarms and take corrective action. Automatic protection is provided in case the spurious high level reduces feedwater flow sufficiently to cause low level in the steam generator. The reactor will trip either on low feedwater flow coincident with low water level or, ultimately, on low-low steam generator water level. Automatic protection is also provided in case the spurious low level signal increases feedwater flow sufficiently to cause high level in the steam generator. A turbine trip and feedwater isolation would occur on two out of three high-high steam generator water level in any loop.

7.2.2.4 Additional Postulated Accidents

Loss of plant instrument air or loss of component cooling water is discussed in Section 7.3. Load rejection and turbine trip are discussed in further detail in Section 7.7.

TABLE 7.2-1 (CONTINUED) (Sheet 2 of 2)

LIST OF REACTOR TRIPS

Reactor Trip	Coincidence Logic	Interlocks	Comments
11. Low reactor coolant flow	2/3 per loop	Interlocked with P-7	Blocked below P-7
12. Reactor coolant pump breakers open	1/2 breakers, 1 breaker per bus	Interlocked with P-7	Blocked below P-7
13. Reactor coolant pump bus undervoltage	1/2 per bus on both buses	Interlocked with P-7	Low voltage on all buses permitted below P-7
14. Reactor coolant pump bus underfrequency	1/2 per bus on both buses	Interlocked with P-7	Under frequency on 2 buses will trip all reactor coolant pump breakers and cause reactor trip; reactor trip blocked below P-7
15. ~~Low feedwater flow~~	~~1/2 per loop*~~	~~No interlocks~~	
5 16. Low-low steam generator water level	2/3 per loop	No interlocks	
6 17. Safety injection signal	Coincident with actuation of safety injection	No interlocks	(See Section 7.3 for Engineered Safety Features actuation conditions)
7 18. Turbine-generator trip a) Low trip fluid pressure b) Turbine stop valve close	 2/3 2/2	Interlocked with P-7	Blocked below P-7
8 19. Manual	1/2	No interlocks	

~~* 1/2 steam/feedwater flow mismatch in coincidence with 1/2 low steam generator water level.~~

TABLE 7.2-3 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

Reactor Trip Signal	Typical Range	Typical Trip Accuracy	Typical Time Response (sec)
11. Low reactor coolant flow	0 to 120% of rated flow	± 2.75 percent of full flow within range of 70 percent to 100 percent of full flow (1)	1.0
12. Reactor coolant pump bus undervoltage	0 to 100% rated voltage	±1 percent of rated voltage	1.2
13. Reactor coolant pump bus underfrequency	40 to 55 Hz	±0.1 Hz	0.6
14. ~~Low feedwater flow~~	~~0 to 120% Max. Calc. feedwater flow~~	~~±6.5% (2)~~	~~2.0~~
15. Low-low steam generator water level	± ~6 ft. from nominal full load water level	±2.3 percent of Δp signal over pressure range of 700 to 1200 psig	2.0
16. Turbine Trip			1.0

NOTES FOR TABLE 7.2-3

(1) Reproducibility (see definitions in Sec 7.1)

~~(2) 1/2 steam/feedwater flow mismatch in coincidence with 1/2 low steam generator water level.~~

~~Channel accuracy of feedwater flow analog signal is ±2.5 percent of maximum calculated feedwater flow.~~

~~Accuracy of steam flow singal is ±3 percent of maximum calculated flow over the pressure range of 700 to 1200 psig.~~

TABLE 7.2-4 (Continued) (Sheet 4 of 5)

REACTOR TRIP CORRELATION

TRIP	ACCIDENT[a]	TECH. SPEC.[b]
10) Pressurizer High Pressure Trip	1) Uncontrolled Rod Cluster Control Assembly Bank Withdrawal At Power (15.2.2)	16.2.3.3.2
	2) Loss of External Electrical Load and/or Turbine Trip (15.2.7)	
11) Pressurizer High Water Level Trip	1) Uncontrolled Rod Cluster Control Assembly Bank at Power (15.2.2)	16.2.3.3.3
	2) Loss of External Electrical Load and/or Turbine Trip (15.2.7)	
12) Low Reactor Coolant Flow	1) Partial Loss of Forced Reactor coolant Flow (15.2.5)	16.2.3.3.2
	2) Loss of Off-Site Power to the Station Auxiliaries (Station Blackout) (15.2.9)	
	3) Complete Loss of Forced Reactor Coolant Flow (15.3.4)	
13) Reactor Coolant Pump Breaker Trip	Not used nor credit taken for in any Accident Analysis; provided as additional feature to enhance safety	
14) Reactor Coolant Pump Bus Undervoltage Trip	1) Complete Loss of Forced Reactor Coolant Flow (15.3.4)	16.2.3.3.2
15) Reactor Coolant Pump Bus Underfrequency Trip	1) Complete Loss of Forced Reactor Coolant Flow (15.3.4)	16.2.3.3.2
~~16) Low Feedwater Flow Trip~~	~~1) Loss of Normal Feedwater (15.2.8)~~	~~See note c.~~

TABLE 7.2-4 (Continued) (Sheet 5 of 5)

REACTOR TRIP CORRELATION

TRIP		ACCIDENT[a]	TECH. SPEC.[b]
6 17)	Low-low Steam Generator Water Level Trip	1) Loss of Normal Feedwater (15.2.8)	16.2.3.3.3
7 18)	Turbine Trip-Reactor Trip	1) Loss of External Electrical Load and/or Turbine Trip (15.2.7)	See note c.
		2) Loss of Off-Site Power to the Station Auxiliaries (Station Blackout) (15.2.9)	16.2.3.3.2
8 19)	Safety Injection Signal Actuation Trip	1) Accidental Depressurization of the Main Steam System (15.2.13)	See note d.
19 20)	Manual Trip	Available for all Accidents (Chapter 15)	See note c.

NOTES:

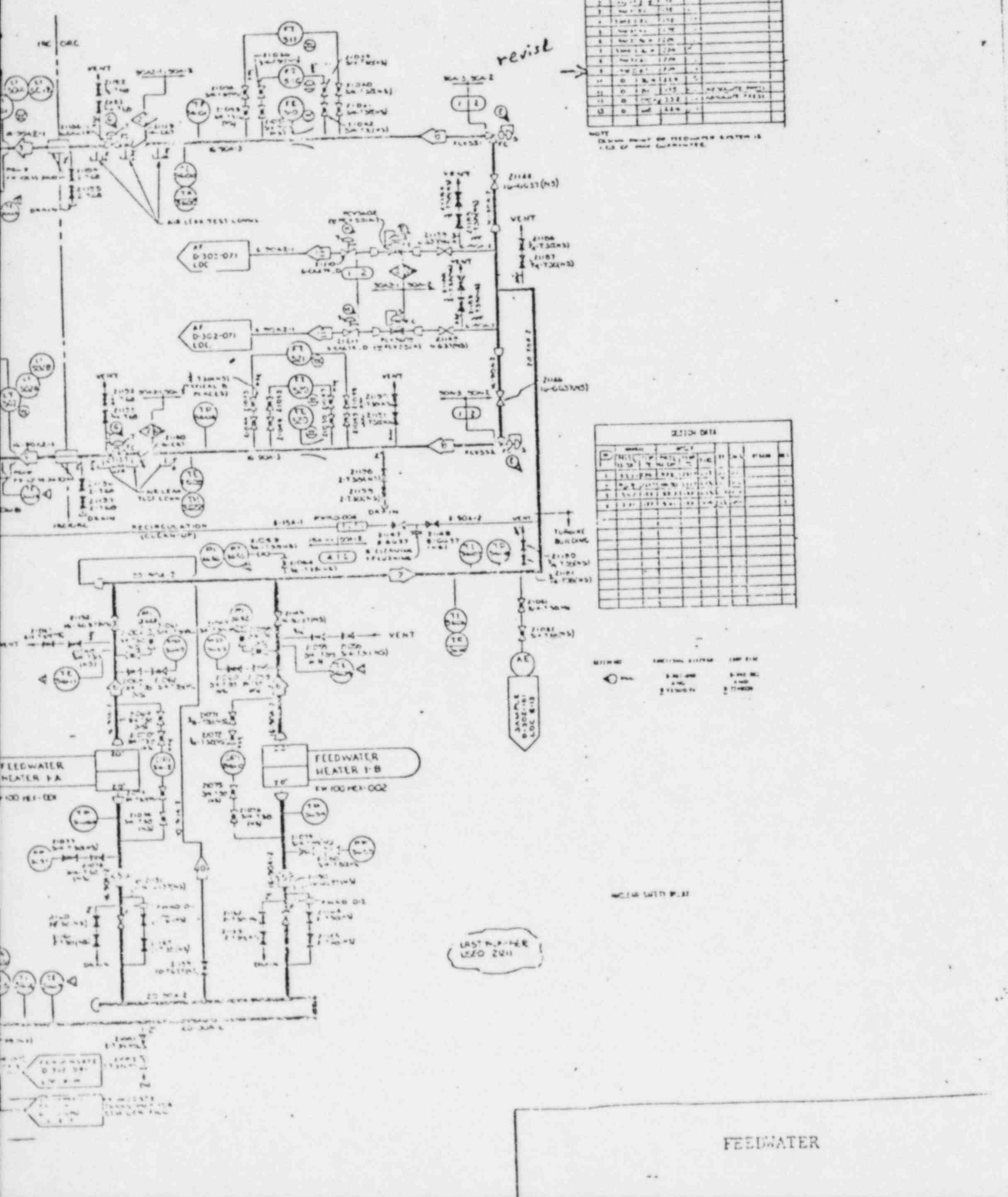
a. References refer to accident analyses presented in Chapter 15.

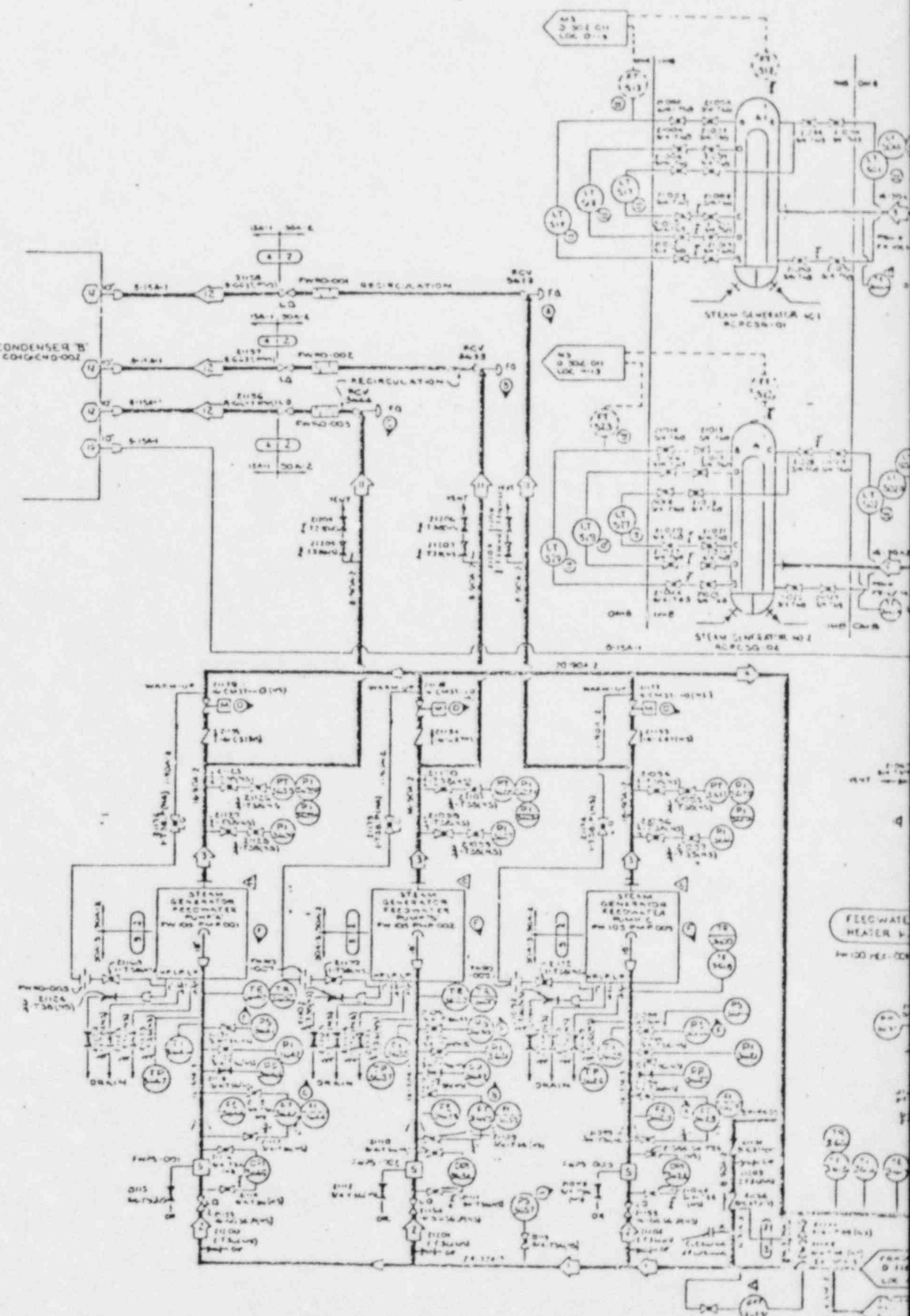
b. References refer to technical specifications presented in Chapter 16.

c. A technical specification is not required because this trip is not assumed to function in the accident analyses.

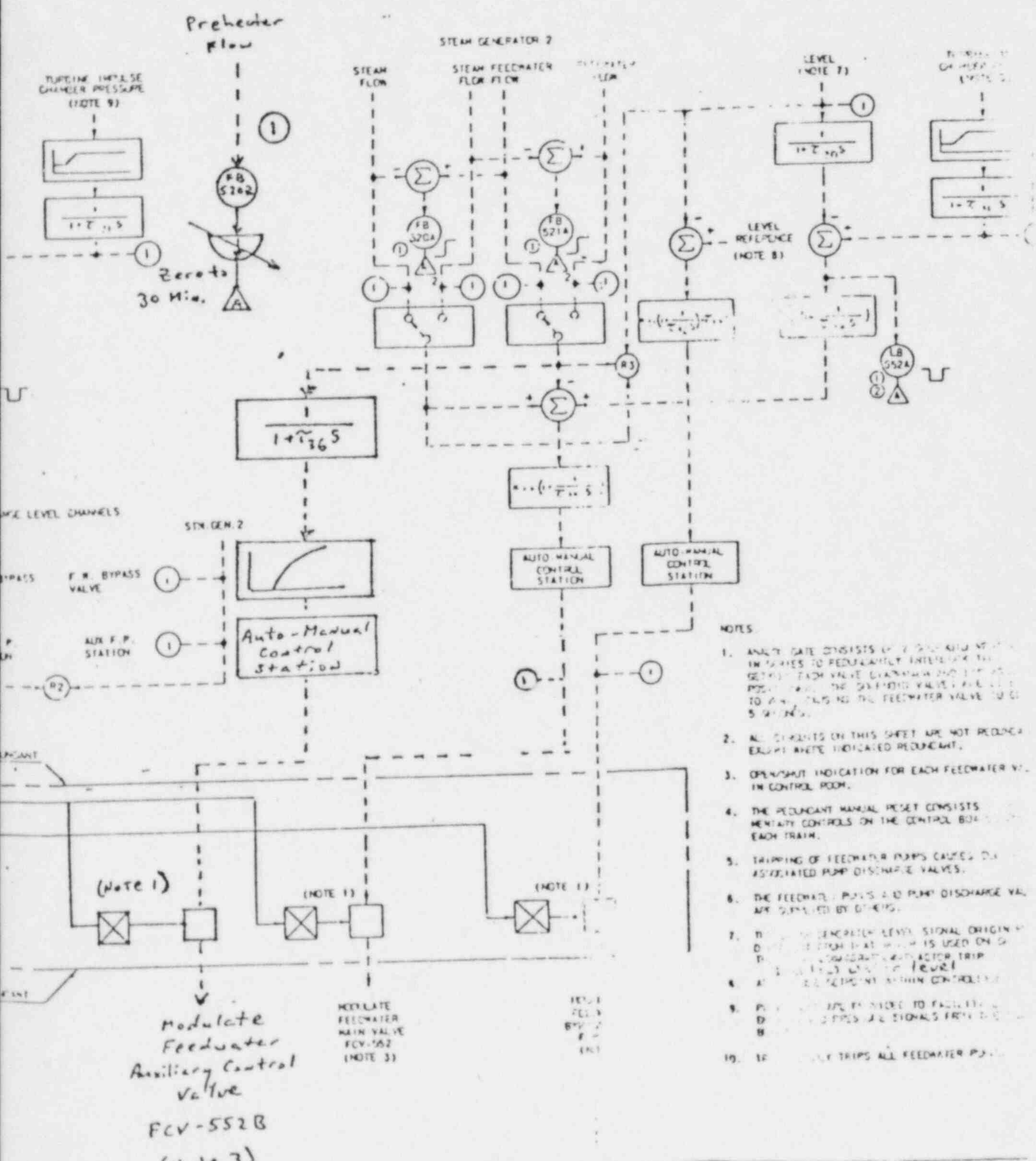
d. Accident assumes that the reactor is tripped at end of life (EOL) which is the worst initial condition for this case. Pressurizer low pressure-low level is the first out trip of Safety Injection.

Add FACV and lines




CONDENSER 'B'
RECIRCULATION
RECIRCULATION
STEAM GENERATOR NO 1
STEAM GENERATOR NO 2
STEAM GENERATOR FEEDWATER PUMP
STEAM GENERATOR FEEDWATER PUMP
STEAM GENERATOR FEEDWATER PUMP
WARM-UP
DRAIN
VENT
FEEDWATER HEATER

Note: SG 2 depicted

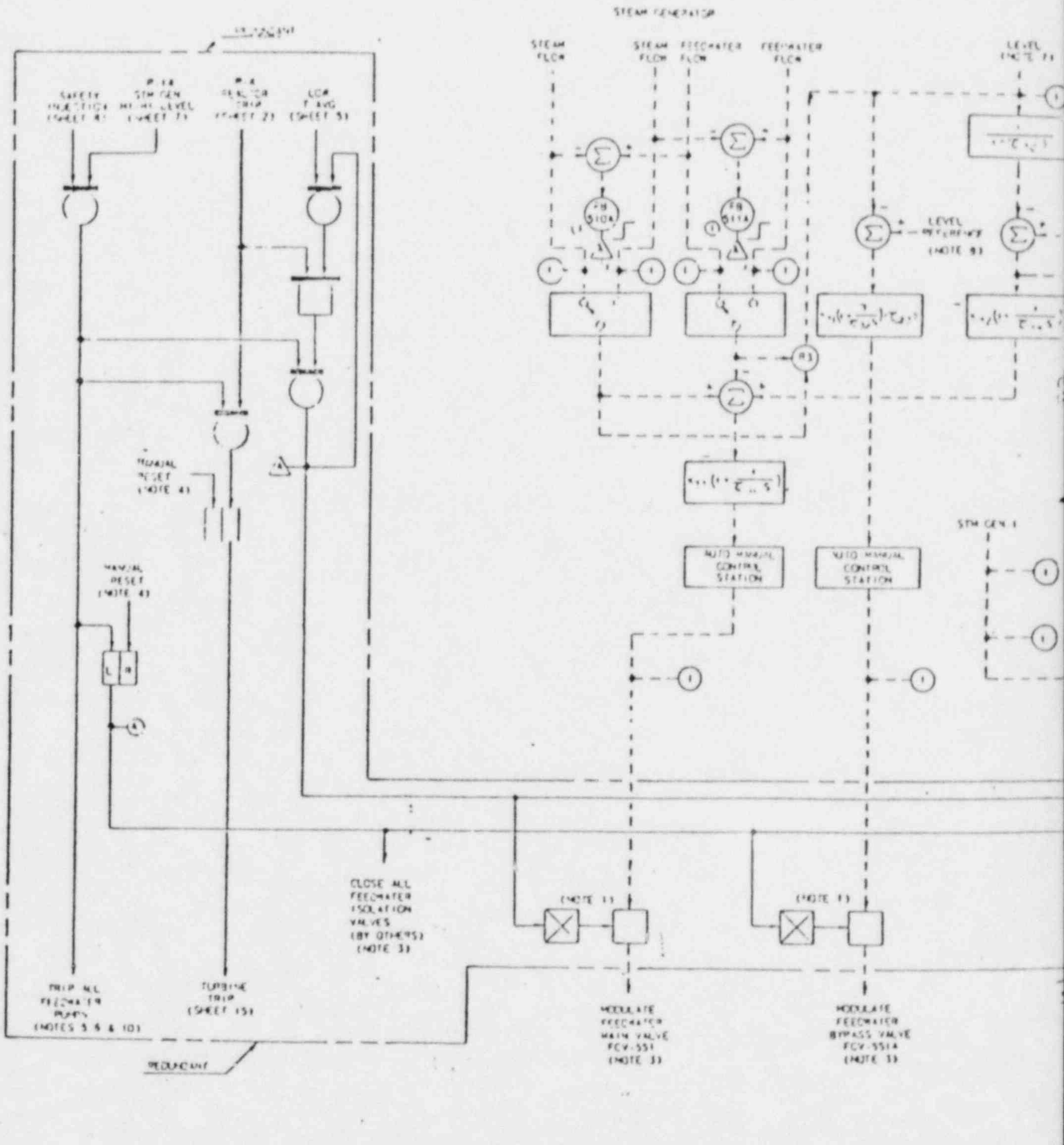


NOTES

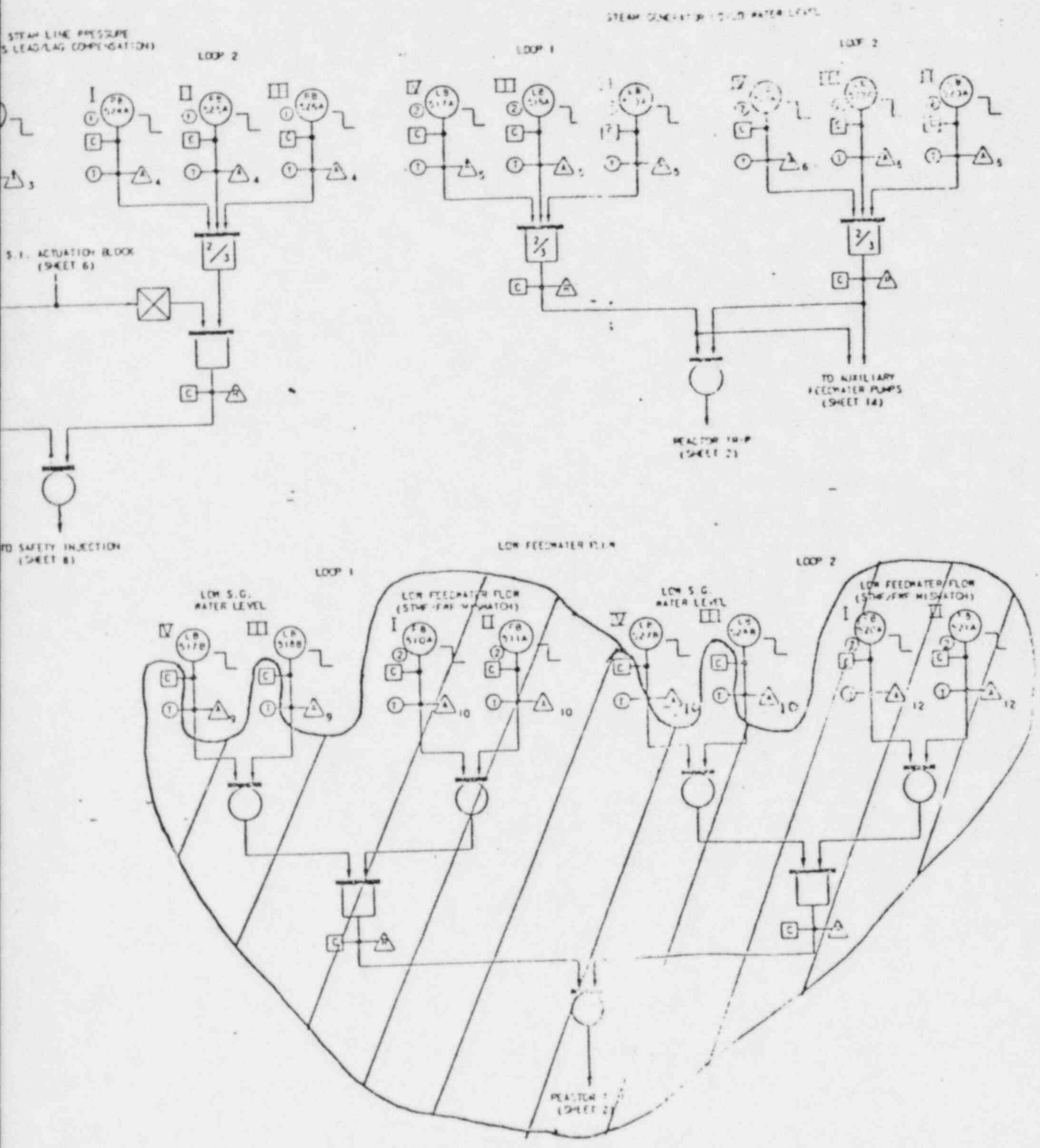
1. ANALOG GATE CONSISTS OF ... IN SERIES TO REDUNDANTLY INTERRUPT THE ... EACH VALVE DIAPHRAGM ... THE SOLENOID VALVES ... TO ... THE FEEDWATER VALVE ...
2. ALL CIRCUITS ON THIS SHEET ARE NOT REDUNDANT EXCEPT WHERE INDICATED REDUNDANT.
3. OPEN/SHUT INDICATION FOR EACH FEEDWATER VALVE IN CONTROL ROOM.
4. THE REDUNDANT MANUAL RESET CONSISTS ... MENTARY CONTROLS ON THE CONTROL BOARD ... EACH TRAIN.
5. TRIPPING OF FEEDWATER PUMPS CAUSES ... ASSOCIATED PUMP DISCHARGE VALVES.
6. THE FEEDWATER PUMPS ... PUMP DISCHARGE VALVES ARE SUPPLIED BY OTHERS.
7. ... GENERATOR LEVEL SIGNAL ORIGIN ... FROM THAT WHICH IS USED ON ... REDUNDANT ... REACTOR TRIP ... level
8. ... SETPOINT WITHIN CONTROL ...
9. ... ARE PROVIDED TO FACILITATE ... PRESSURE SIGNALS FROM ...
10. ... TRIPS ALL FEEDWATER PUMPS.

INSTRUMENTATION AND CONTROL SYSTEM DIAGRAM (SHEET 13)

NE KRSKO | FSAR | Fig. 7.

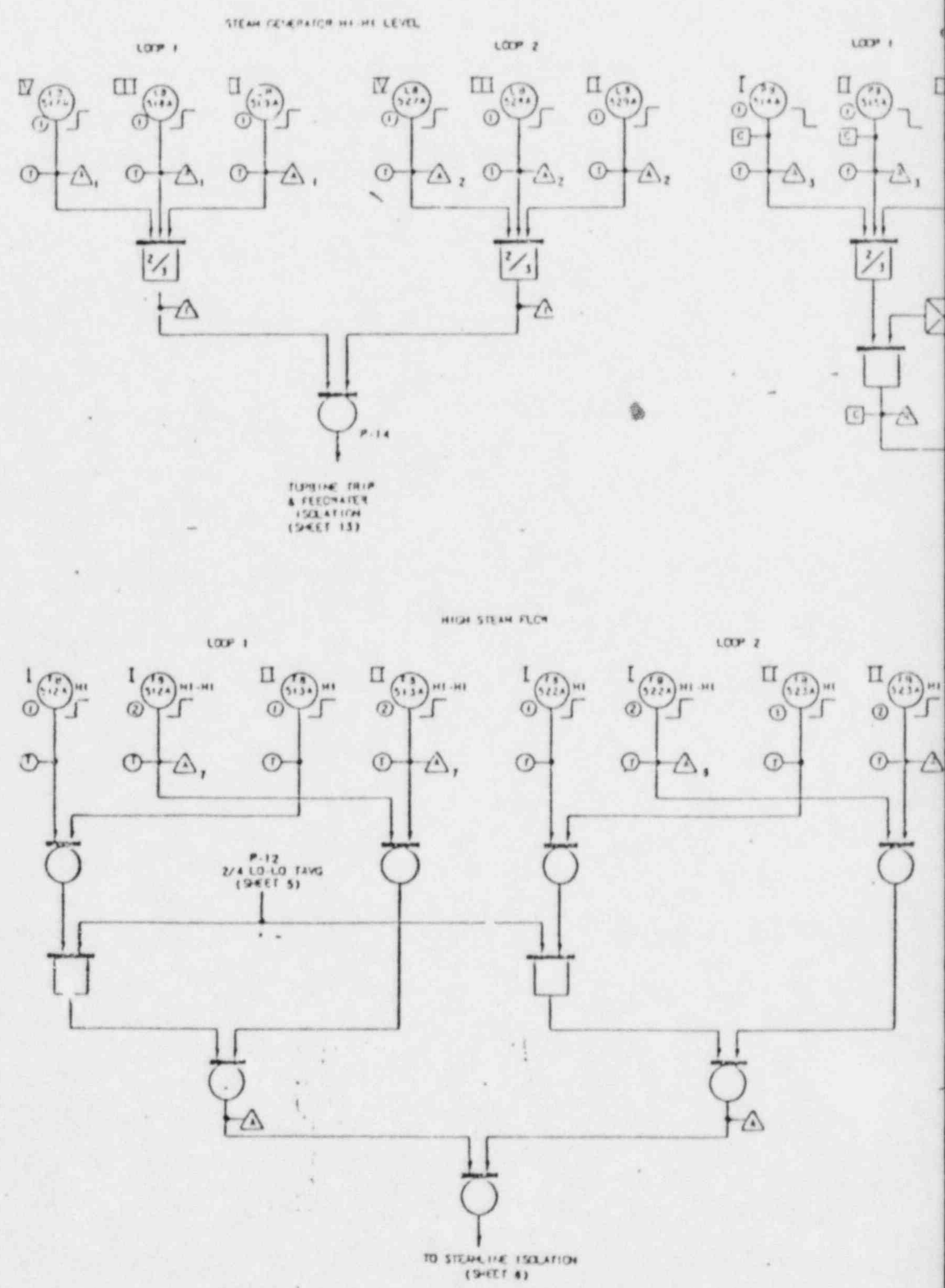


STEAM GENERATOR
STEAM FLOW
STEAM FLOW
FEEDWATER FLOW
FEEDWATER FLOW
LEVEL (NOTE 7)
LEVEL REFERENCE (NOTE 8)
SAFETY INJECTION (SHEET 8)
STM GEN HI-HI LEVEL (SHEET 7)
REACTOR TRIP (SHEET 2)
LOW T AVG (SHEET 5)
MANUAL RESET (NOTE 4)
MANUAL RESET (NOTE 4)
AUTO MANUAL CONTROL STATION
AUTO MANUAL CONTROL STATION
STM GEN I
CLOSE ALL FEEDWATER ISOLATION VALVES (BY OTHERS) (NOTE 3)
TRIP ALL FEEDWATER PUMPS (NOTES 5, 6 & 10)
TURBINE TRIP (SHEET 15)
MODULATE FEEDWATER MAIN VALVE FCV-551 (NOTE 3)
MODULATE FEEDWATER BYPASS VALVE FCV-551A (NOTE 3)
REDUNDANT

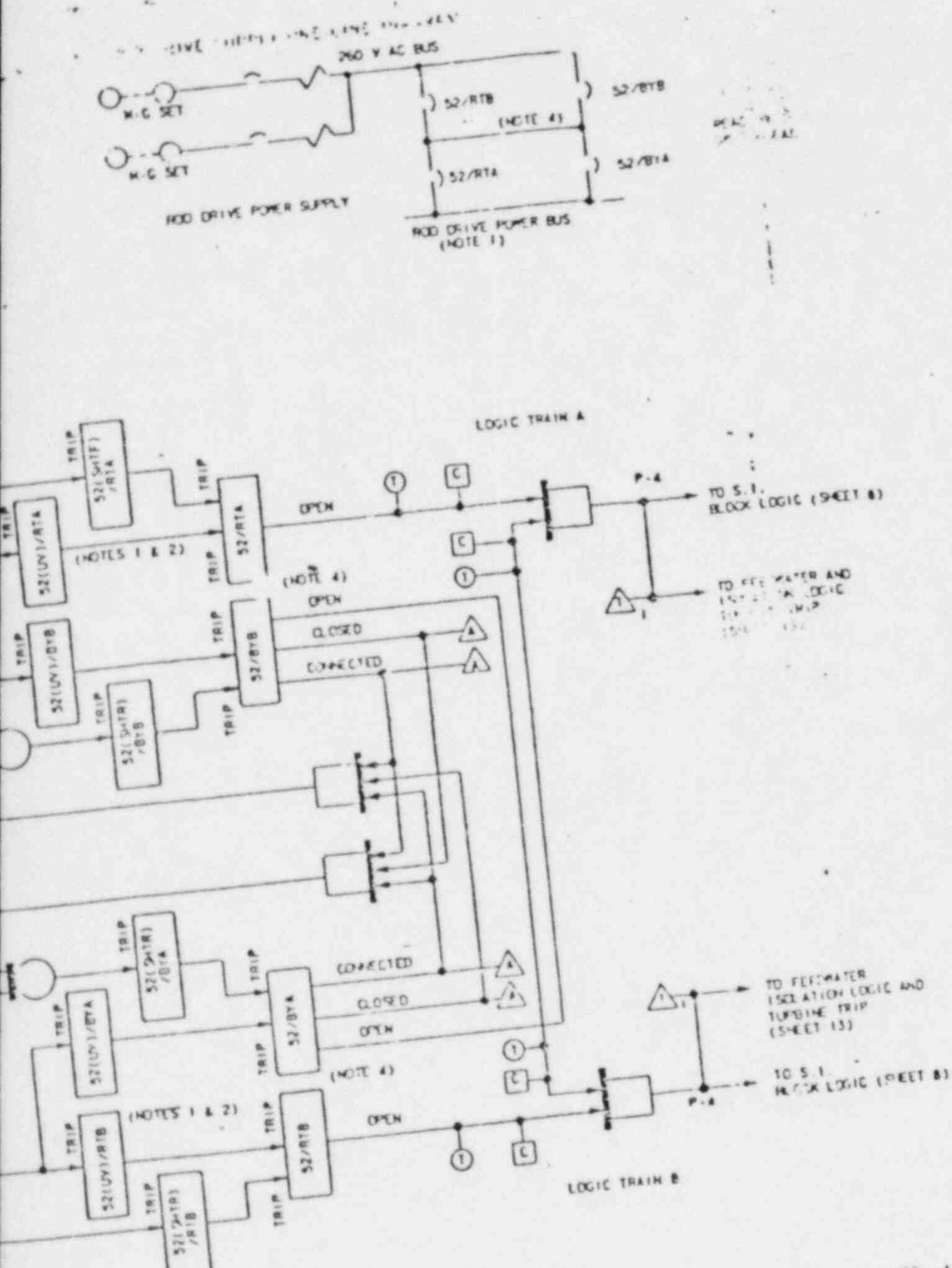


INSTRUMENTATION AND CONTROL SYSTEM
DIAGRAM (SH-- 7)

NE KRSKO -FS-- Fig. 7.2-



STEAM GENERATOR HI-HI LEVEL
LOOP 1
LOOP 2
LOOP 1
P-14
TURBINE TRIP
& FEEDWATER
ISOLATION
(SHEET 13)
HIGH STEAM FLOW
LOOP 1
LOOP 2
P-12
2/4 LO-LO TAVG
(SHEET 5)
TO STEAMLINE ISOLATION
(SHEET 8)



NOTES

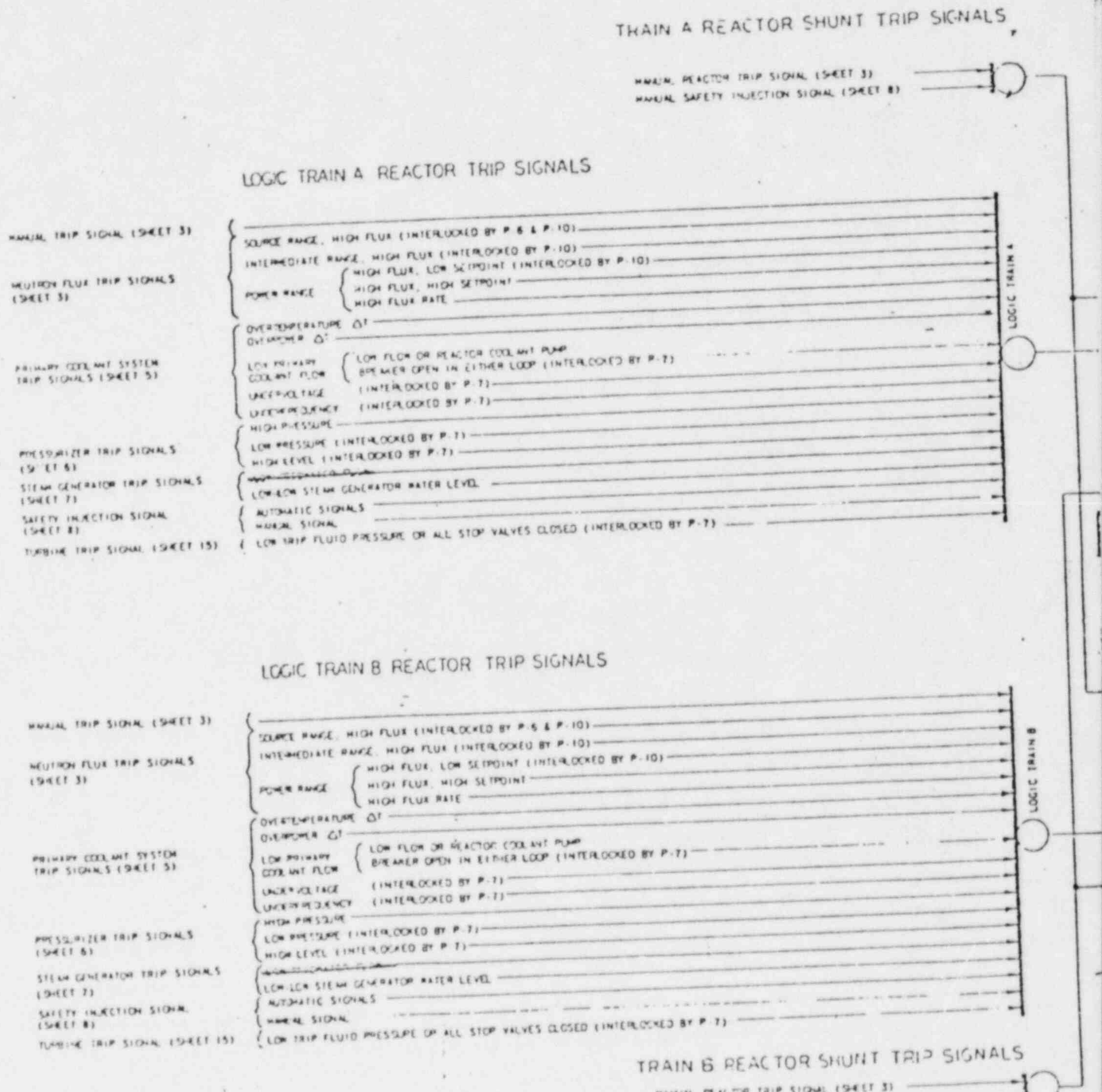
1. TRIPPING THE REACTOR TRIP BREAKERS 52/RTA AND 52/RTB REDUNDANTLY DE-ENE... ROD DRIVES. ALL FULL LENGTH CONTROL RODS AND SHUTDOWN RODS ARE THEREBY RELEASED FOR GRAVITY INSERTION INTO THE REAC...
2. NORMAL REACTOR OPERATION IS TO BE WITH REACTOR TRIP BREAKERS 52/RTA AND ... SERVICE AND BY-PASS BREAKERS 52/BYA AND 52/BYB WITHDRAWN.
   DURING TEST ONE BY-PASS BREAKER IS TO BE PUT IN SERVICE AND THEN THE ... REACTOR TRIP BREAKER IS OPERATED USING A SIMULATED REACTOR TRIP SIGNAL IN THE TRAIN UNDER TEST. THE REACTOR WILL NOT BE ... THE SIMULATED SIGNAL SINCE THE BY-PASS BREAKER IS CONTROLLED FROM THE OTHER TRAIN. ONLY ONE REACTOR TRIP BREAKER IS TO ... A TIME.
3. ALL CIRCUITS ON THIS SHEET ARE NOT REDUNDANT BECAUSE BOTH TRAINS ARE SU... EACH BYPASS BREAKER IN ...
4. OPEN/CLOSED INDICATION FOR EACH TRIP BREAKER AND EACH BYPASS BREAKER IN ...

INSTRUMEN... AND CONTROL ...
... M (SHEET 2)

FSAR

Fig 7.2-1

... KRSKO



TRAIN A REACTOR SHUNT TRIP SIGNALS
MANUAL REACTOR TRIP SIGNAL (SHEET 3)
MANUAL SAFETY INJECTION SIGNAL (SHEET 8)
LOGIC TRAIN A REACTOR TRIP SIGNALS
MANUAL TRIP SIGNAL (SHEET 3)
NEUTRON FLUX TRIP SIGNALS (SHEET 3)
SOURCE RANGE, HIGH FLUX (INTERLOCKED BY P-6 & P-10)
INTERMEDIATE RANGE, HIGH FLUX (INTERLOCKED BY P-10)
POWER RANGE
HIGH FLUX, LOW SETPOINT (INTERLOCKED BY P-10)
HIGH FLUX, HIGH SETPOINT
HIGH FLUX RATE
PRIMARY COOLANT SYSTEM TRIP SIGNALS (SHEET 5)
OVERTEMPERATURE ΔT
OVERPOWER ΔT
LOW PRIMARY COOLANT FLOW
LOW FLOW OR REACTOR COOLANT PUMP BREAKER OPEN IN EITHER LOOP (INTERLOCKED BY P-7)
UNDERVOLTAGE (INTERLOCKED BY P-7)
UNDERFREQUENCY (INTERLOCKED BY P-7)
PRESSURIZER TRIP SIGNALS (SHEET 6)
HIGH PRESSURE
LOW PRESSURE (INTERLOCKED BY P-7)
HIGH LEVEL (INTERLOCKED BY P-7)
STEAM GENERATOR TRIP SIGNALS (SHEET 7)
LOW-LOW STEAM GENERATOR WATER LEVEL
SAFETY INJECTION SIGNAL (SHEET 8)
AUTOMATIC SIGNALS
MANUAL SIGNAL
TURBINE TRIP SIGNAL (SHEET 15)
LOW TRIP FLUID PRESSURE OR ALL STOP VALVES CLOSED (INTERLOCKED BY P-7)
LOGIC TRAIN A
LOGIC TRAIN B REACTOR TRIP SIGNALS
MANUAL TRIP SIGNAL (SHEET 3)
NEUTRON FLUX TRIP SIGNALS (SHEET 3)
SOURCE RANGE, HIGH FLUX (INTERLOCKED BY P-6 & P-10)
INTERMEDIATE RANGE, HIGH FLUX (INTERLOCKED BY P-10)
POWER RANGE
HIGH FLUX, LOW SETPOINT (INTERLOCKED BY P-10)
HIGH FLUX, HIGH SETPOINT
HIGH FLUX RATE
PRIMARY COOLANT SYSTEM TRIP SIGNALS (SHEET 5)
OVERTEMPERATURE ΔT
OVERPOWER ΔT
LOW PRIMARY COOLANT FLOW
LOW FLOW OR REACTOR COOLANT PUMP BREAKER OPEN IN EITHER LOOP (INTERLOCKED BY P-7)
UNDERVOLTAGE (INTERLOCKED BY P-7)
UNDERFREQUENCY (INTERLOCKED BY P-7)
PRESSURIZER TRIP SIGNALS (SHEET 6)
HIGH PRESSURE
LOW PRESSURE (INTERLOCKED BY P-7)
HIGH LEVEL (INTERLOCKED BY P-7)
STEAM GENERATOR TRIP SIGNALS (SHEET 7)
LOW-LOW STEAM GENERATOR WATER LEVEL
SAFETY INJECTION SIGNAL (SHEET 8)
AUTOMATIC SIGNALS
MANUAL SIGNAL
TURBINE TRIP SIGNAL (SHEET 15)
LOW TRIP FLUID PRESSURE OR ALL STOP VALVES CLOSED (INTERLOCKED BY P-7)
LOGIC TRAIN B
TRAIN B REACTOR SHUNT TRIP SIGNALS
MANUAL REACTOR TRIP SIGNAL (SHEET 3)
MANUAL SAFETY INJECTION SIGNAL (SHEET 8)

The feedwater pumps are designed in accordance with the requirements of the Hydraulic Institute Standards. Design points for these pumps are selected to satisfy the requirements of the turbine thermal cycle at the maximum guaranteed condition plus margins for wear and surges. The feedwater pumps are also capable of maintaining steam generator water level during a load rejection and steam dump at 96 percent flow at a steam generator pressure of 972 psia.

High pressure feedwater heaters are designed, fabricated, inspected, tested and stamped in accordance with the ASME Code, Section VIII, Division 1. Thermal performance of these feedwater heaters is governed by Heat Exchange Institute Standards.

The feedwater system equipment parameters are listed in Table 10.4-2.

10.4.7.2.2 System Description

The feedwater system includes three 50 percent capacity motor driven main feed pumps, two parallel high pressure feedwater heaters and associated piping, valves and controls.

At loads above 15 percent, normal operating control is achieved by using a three element system consisting of inputs proportional to steam flow, feedwater flow and steam generator water level to control the position of the feedwater regulating valves. At loads of 15 percent and below, the steam generator level is maintained with the feedwater bypass control valve. The bypass control valve is automatically positioned with inputs proportional to the steam generator water level or the bypass control valve is manually positioned from the control room. These controls are always in operation except during a safety injection signal or a reactor trip coincident with low T_{avg}. The modulating signal is blocked in these cases.

add discussion on "60 percent and higher"

The main feed pumps are driven by a constant speed motor through speed increasing gearing. A low flow bypass to the main condenser is provided for use during startup.

plant and of the Reactor Coolant System. The overpower - overtemperature protection (neutron overpower, overtemperature and overpower ΔT trips) prevents any power increase which could lead to a DNBR less than 1.30.

One example of excess heat removal from the primary system is the transient associated with the accidental opening of the feedwater bypass valve which diverts flow around the low pressure feedwater heaters. In the event of an accidental opening of the bypass valve, there is a sudden reduction in feedwater inlet temperature to the steam generators. This increased subcooling will create a greater load demand on the Reactor Coolant System.

Another example of excessive feedwater flow would be a full opening of a feedwater control valve due to a feedwater control system malfunction or an operator error. At power this excess flow causes a greater load demand on the Reactor Coolant System due to increased subcooling in the steam generator. With the plant at no-load conditions the addition of cold feedwater may cause a decrease in Reactor Coolant System temperature and thus a reactivity insertion due to the effects of the negative moderator coefficient of reactivity. Continuous addition of excessive feedwater is prevented by the steam generator high-high level trip, which closes the feedwater valves.

15.2.10.2 Analysis of Effects and Consequences

15.2.10.2.1 Method of Analysis

The excessive heat removal due to a feedwater control valve malfunction transient is analyzed by using the detailed digital computer code MARVEL[1]. This code simulates a multi-loop system, neutron kinetics, the pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

The system is analyzed to demonstrate plant behavior in the event of a feedwater bypass valve malfunction. Feedwater temperature reduction due to low pressure heater bypass valve actuation in conjunction with an inadvertent trip of the heater drain pump is considered.

Excessive feedwater addition due to a control system malfunction or operator error which allows a feedwater control valve to open fully is considered. Two cases are analyzed as follows:

1. Accidental opening of one feedwater control valve with the reactor just critical at zero load conditions assuming a conservatively large negative moderator temperature coefficient characteristic of end of core life conditions.

2. Accidental opening of one feedwater control valve with the reactor in automatic control at full power.

The reactivity insertion rate following a feedwater system malfunction is calculated with the following assumptions:

a. For the feedwater control valve incident at full power, one feedwater control valve is assumed to malfunction resulting in a step increase to ~~120~~ 155% of nominal feedwater flow to one steam generator.

b. For the feedwater control valve accident at zero load condition, a feedwater control valve malfunction occurs which results in a step increase in flow to one steam generator from zero to 100% of the nominal full load value for one steam generator.

c. For the zero load condition, feedwater temperature is at a conservatively low value of 70°F.

In the case of an accidental full opening of one feedwater control valve with the reactor at zero power and the above mentioned assumptions, the maximum reactivity insertion rate is less than the maximum reactivity insertion rate analyzed in Section 15.2.1, Uncontrolled Control Rod Assembly Withdrawal from a Subcritical Condition, and therefore, the results of the analyses are not presented. It should be noted that if the incident occurs with the unit just critical at no load, the reactor may be tripped by the power range high neutron flux trip (low setting) set at approximately 25 percent.

The full power case (EOL, with control) gives the largest reactivity feedback and results in the greatest power increase. A turbine trip and reactor trip is actuated when the nuclear flux level exceeds the power range high nuclear flux trip setpoint of 118% of nominal.

For all excessive feedwater cases continuous addition of cold feedwater is prevented by closure of all feedwater control valves, a trip of the feedwater pumps, and closure of the feedwater pump discharge valves on steam generator high-high level signal.

Transient results, see Figures 15.2-24 and 15.2-25, show the increase in nuclear power and T_{avg} associated with the increased thermal load on the reactor. Steam generator level rises until the feedwater flow is terminated as a result of the high-high steam generator level turbine trip. The DNB ratio does not drop below 1.3 as shown in Figure 15.2-26.

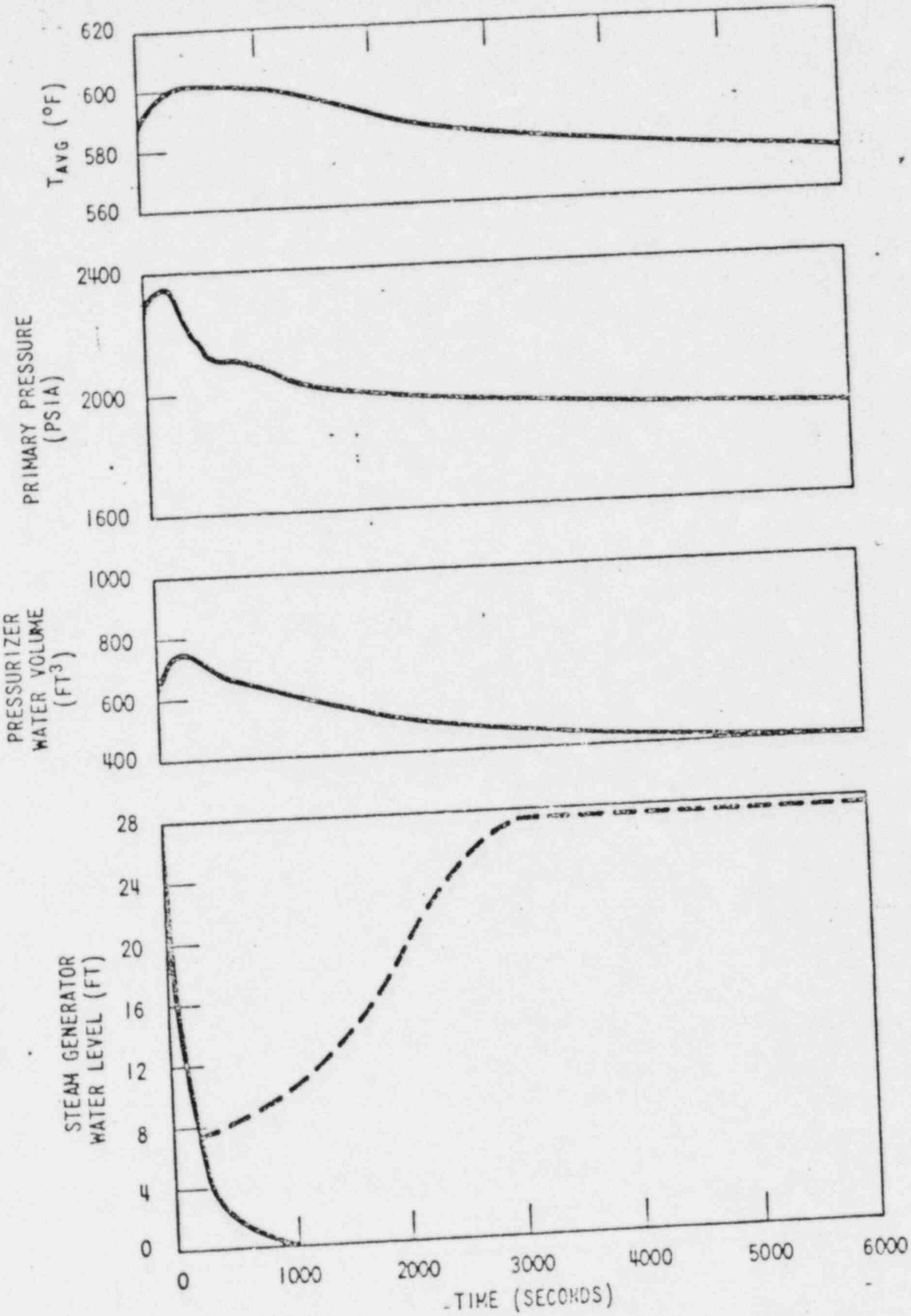
15.2.10.3 Conclusions

Results show that the consequences of excess load increases due to opening the low pressure heater bypass valve are more moderate than those considered for the Excessive Load Increase Accident. Additionally, it has been shown that the reactivity insertion rate which occurs at no load following excessive feedwater addition is less than the maximum value considered in the analysis of the rod withdrawal from a subcritical condition. Also, the DNB ratios encountered for excessive feedwater addition at power are well above the limiting value of 1.30.

TIME SEQUENCE OF EVENTS FOR
CONDITION II EVENTS

Accident	Event	Time (sec.)
Loss of Normal Feedwater (Continued)	Peak water level in pressurizer occurs	280
Excessive feedwater at full load	One main feedwater control valve fails fully open	0
	Minimum DNBR occurs	~~66.0~~ 55.5
	Feedwater flow isolated due to high-high steam generator level	~~192.0~~ 62
Excessive Load Increase		
1. Manual Reactor Control (BOL)	10% step load increase	0
	Equilibrium conditions reached (approximate times only)	150
2. Manual Reactor Control (EOL)	10% step load increase	0
	Overtemperature ΔT reactor trip point reached	24.1

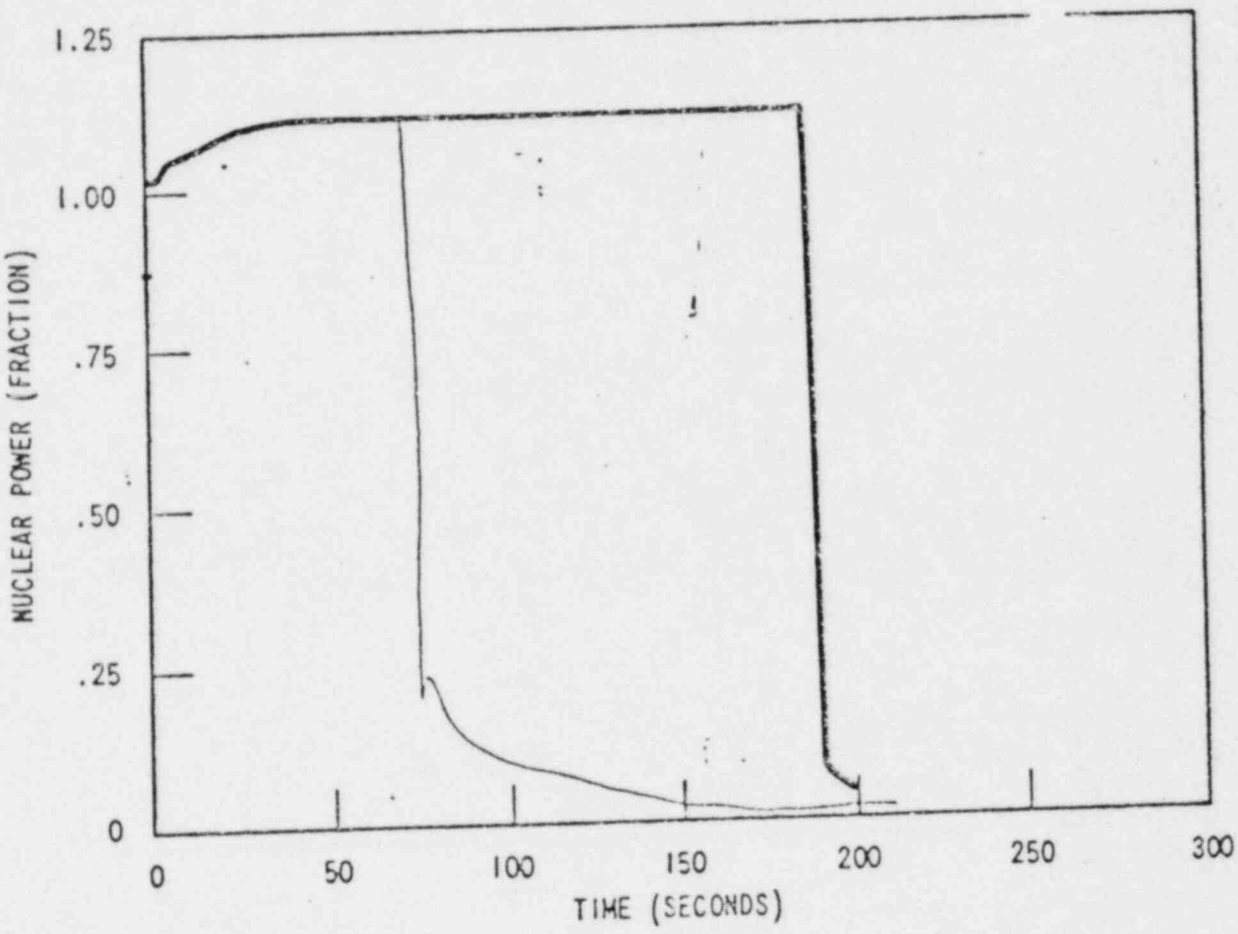
Replace with new figures



LOSS OF NORMAL FEEDWATER

NE KRSKO -- FSAR Fig. 15.2-23

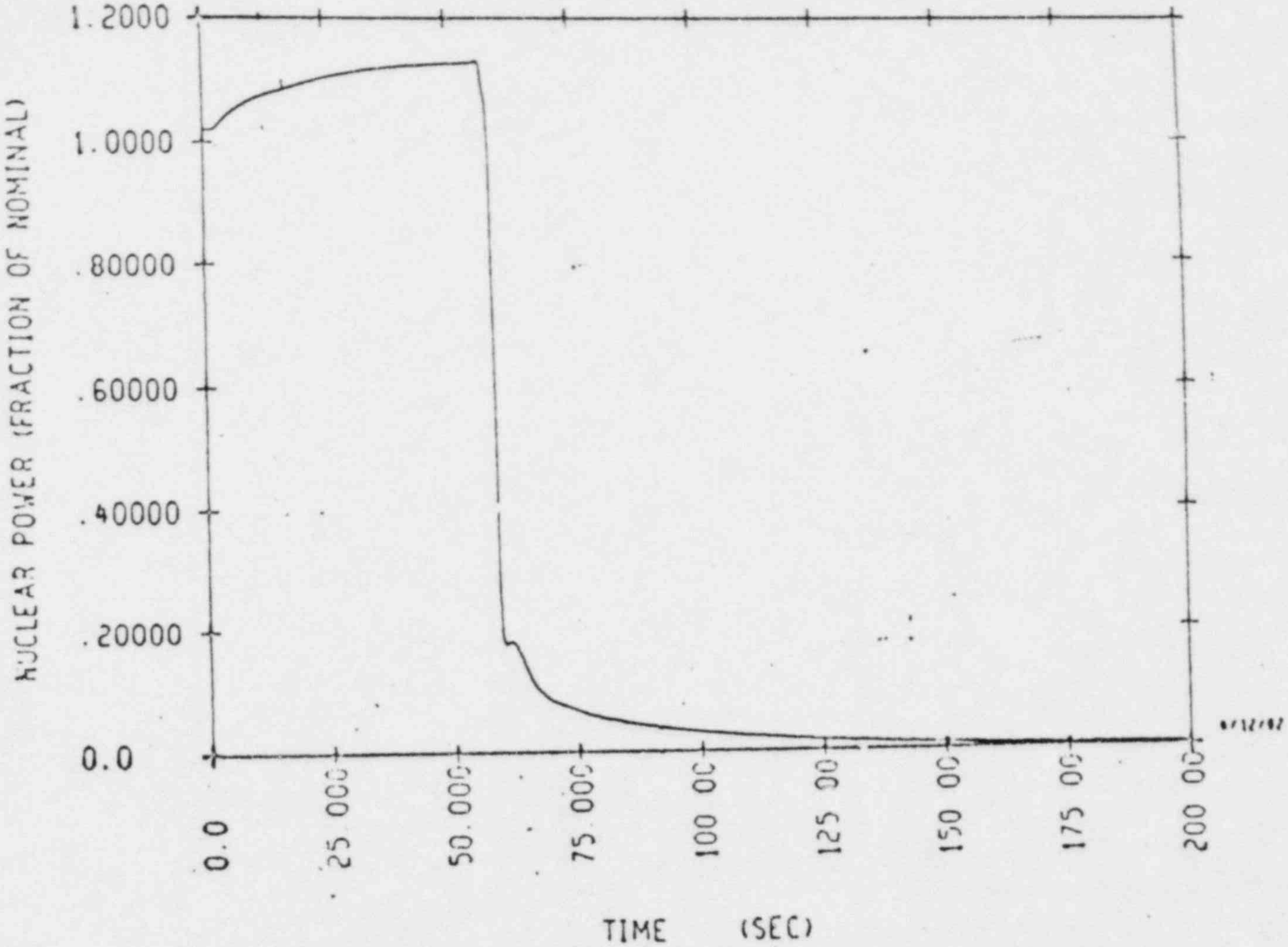
replace with new figure



EXCESS FEED ACCIDENT FEED CONTROL
VALVE FAILURE AT FULL LOAD

NE KRSKO	FSAR	Fig. 15.2-24

STARTO DATA PROGRA M RWS
FULL POWER MANUAL CONTROL 1 55WF
PLOT 1 RUN 1



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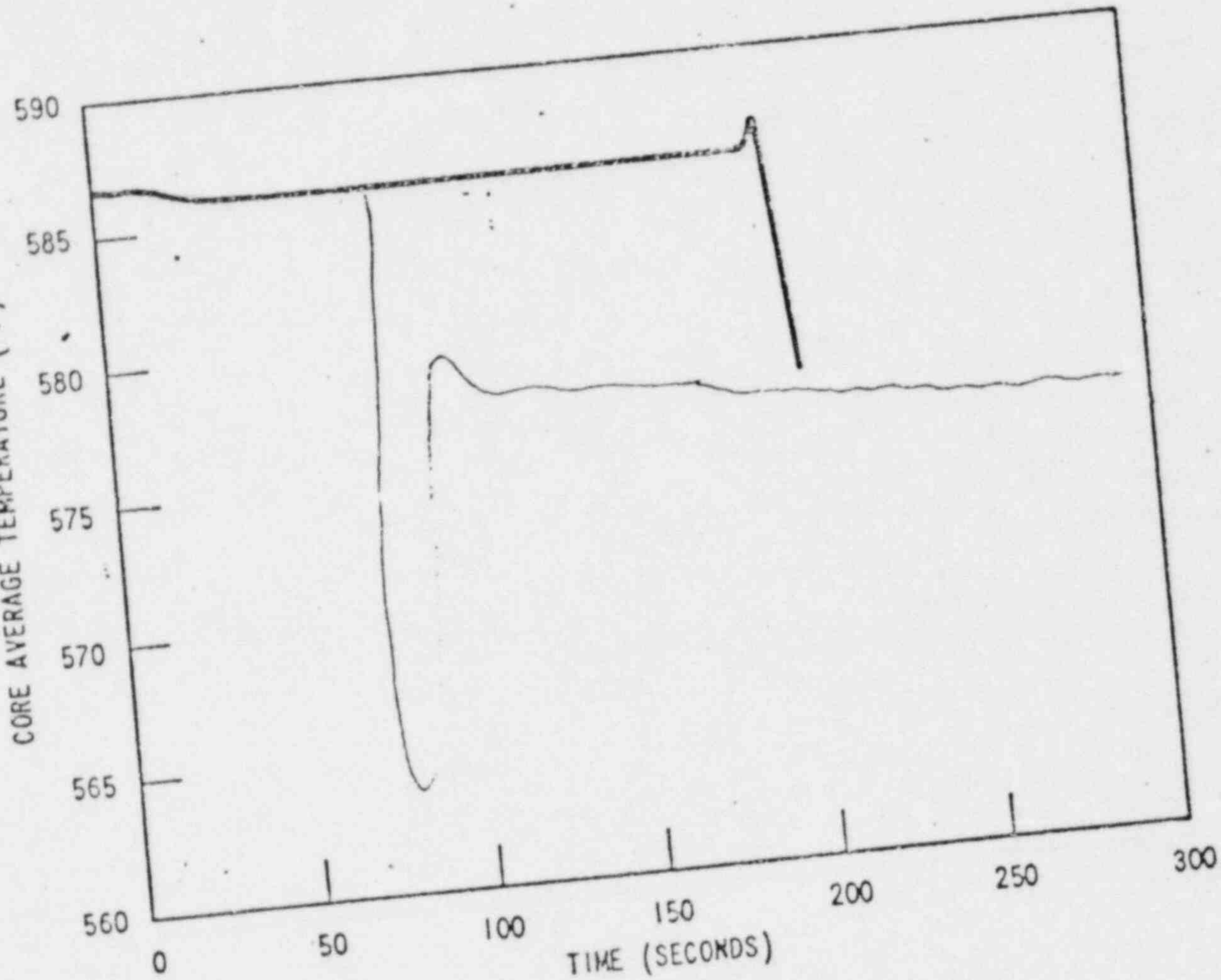
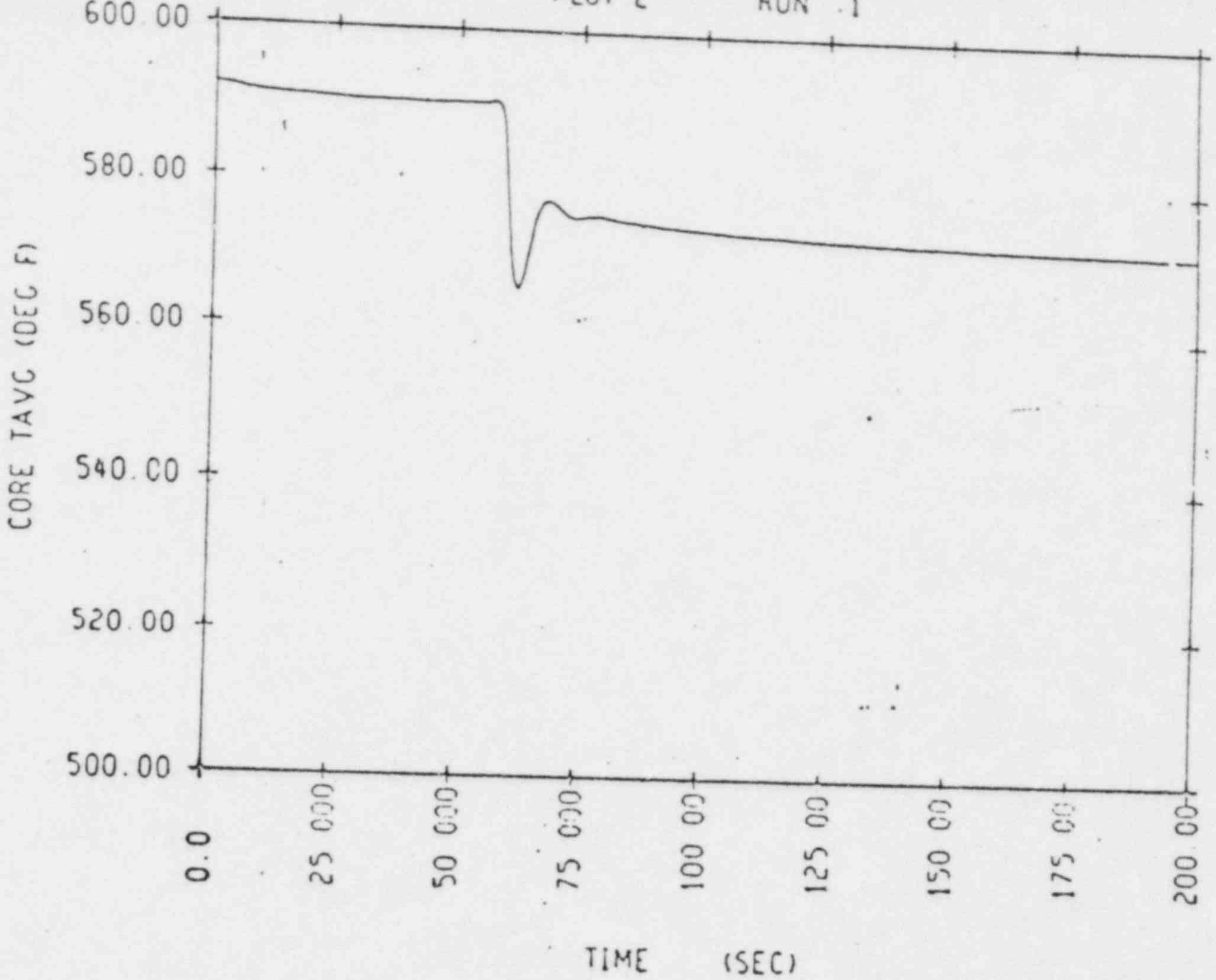


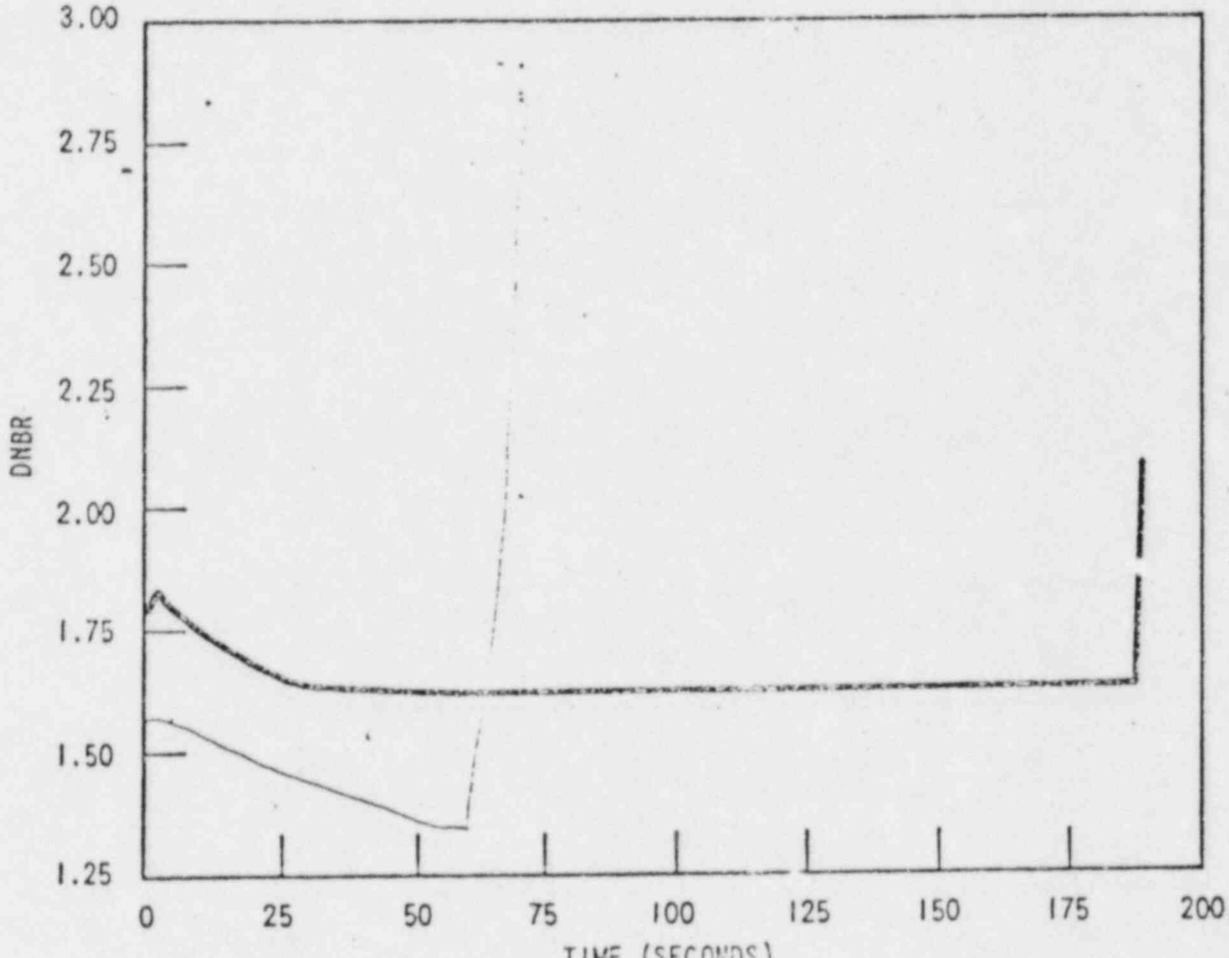
EXCESS FEED ACCIDENT FEED CONTROL
VALVE FAILURE AT FULL LOAD

NE KRSKO	FSAR	Fig. 15.2-25

FULL POWER MANUAL CONTROL 1 55WF
PLOT 2 RUN 1



replace with new figure



EXCESS FEED ACCIDENT

NE KESKO | FSAR | Fig. 15.2-26

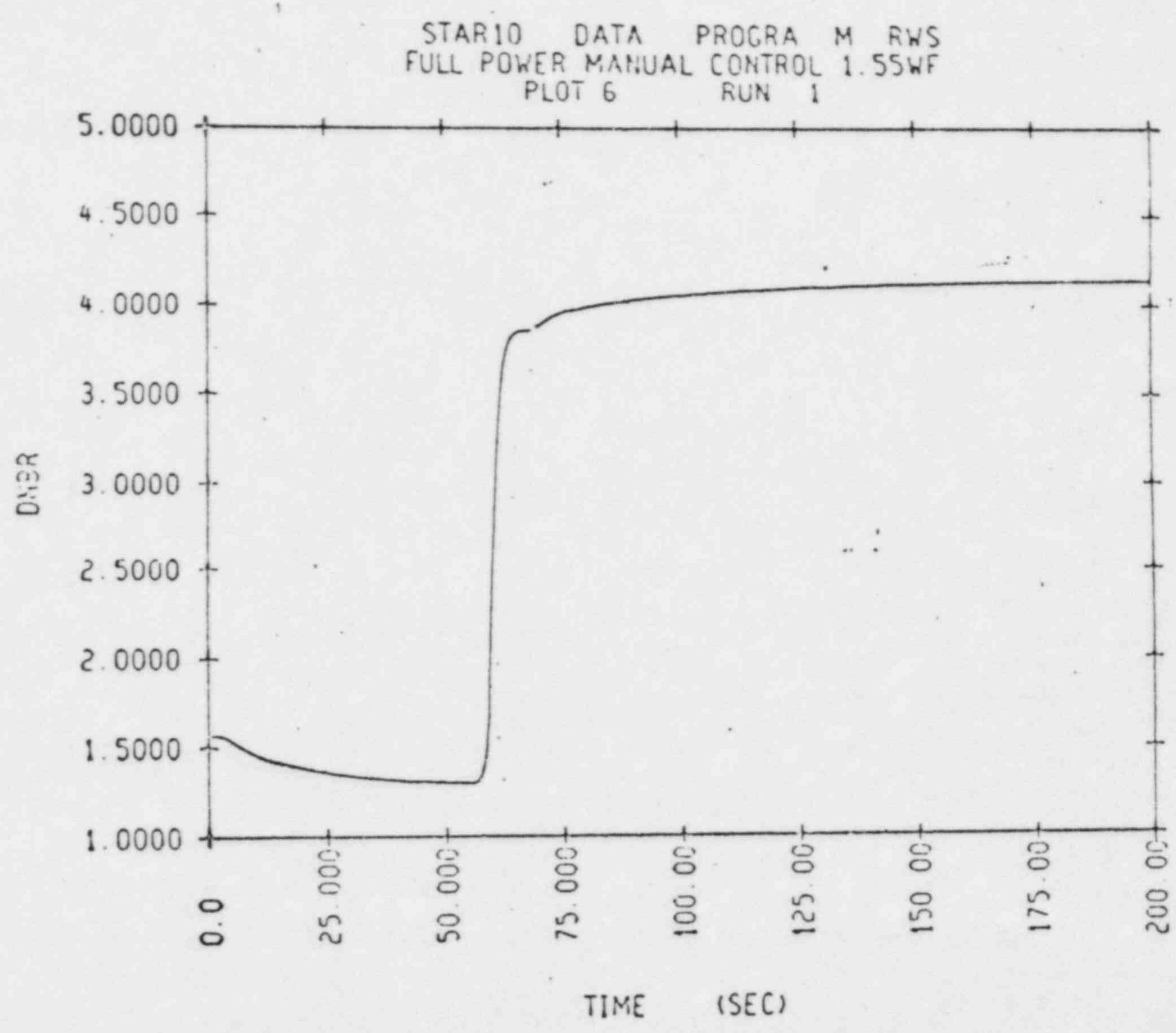

STAR10 DATA PROGRA M RWS
FULL POWER MANUAL CONTROL 1.55WF
PLOT 6 RUN 1
DNBR
5.0000
4.5000
4.0000
3.5000
3.0000
2.5000
2.0000
1.5000
1.0000
0.0
25.000
50.000
75.000
100.00
125.00
150.00
175.00
200.00
TIME (SEC)

TABLE 16.3-3 (Continued)

REACTOR TRIP INSTRUMENTATION LIMITING OPERATING CONDITIONS

No.	Functional Unit	No. of Channels	No. of Channels To Trip	Min. Operable Channels	Min. Degree of Redundancy	Operator Action If Conditions of Column 3 or 4 Cannot be Met
18.	Hi-Hi Steam Generator Level or S.I. (Turbine trip and feedwater isolation)	3/loop	2/loop (any loop)	2^{+}/loop	1/loop	Maintain hot shutdown
~~19.~~	~~Steam Flow/Feedwater Flow Mismatch and Low Steam Generator Water~~	~~2/loop level~~	~~1/loop level coincident with~~	~~1/loop level~~	~~1/loop level~~	~~Maintain hot shutdown~~
		~~2/loop flow mismatch~~	~~1/loop flow mismatch in same loop~~	~~1/loop flow mismatch~~	~~1/loop flow mismatch~~	~~Maintain hot shutdown~~

* If the plant is operating above 75 percent of rated power with one excore nuclear channel out of service, then the core quadrant power tilt shall be determined once a day by the movable incore detectors (at least 2 thimbles per quandrant).

** When 2 out of 4 power channels are greater than 10 percent full power, hot shutdown is not required.

*** If one of two intermediate range channels greater than 10^{-10} amps., hot shutdown is not required.

\+ Inoperable channels are placed in the trip mode. Once placed in the trip mode, the channels can be considered operable for purposes of meeting this specification.

TABLE 16.4-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Test	Remarks
10.	Rod Position Bank Counters	S (1,2)	N.A.	R	1) With analog rod position 2) Following rod motion in excess of six in. when computer is out of service
11.	Steam Generator Level	S	R**	M	
12.	~~Steam Generator Flow Mismatch~~	~~S~~	~~R**~~	~~M~~	
13.	Charging Flow	S	A**	N.A.	
14.	Residual Heat Removal Pump Flow	S (when in operation)	A**	N.A.	
15A.	Boric Acid Tank Level	W	A**	N.A.	
15B.	Boric Acid Tank Temperature	W	R	R	
16.	Refueling Water Storage Tank Level	W	A	N.A.	
17.	Volume Control Tank Level	S	A**	N.A.	
18A.	Containment Pressure (SIS signal)	S	R**	M(1)	1) Isolation valve signal
18B.	Containment Pressure (Streamline Isol)	S	R**	M	Narrow range containment pressure (-3.0, +3.0 psig excluded)