ENCLOSURE 4

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

OCT 31 1983

MEMORANDUM FOR: [redacted] Chief
Standardization and Special Projects Branch
Division of Licensing

FROM: Brian W. Sheron, Chief
Reactor Systems Branch
Division of Systems Integration

SUBJECT: GRAND GULF UNIT 1 TECHNICAL SPECIFICATIONS

Reactor Systems Branch has reviewed the Grand Gulf Technical Specifications forwarded to us with your October 4, 1983 memorandum. We reviewed the following sections: 2.1, 2.2, 3/4.2.3, 3/4.3.1-5, 3/4.3.8, 3/4.1-2, 3/4.4.3.2, 3/4.4.6.2, 3/4.4.9, 3/4.5.1, 3/4.7.3, 3/4.9.11, Bases 3/4.2.3, 3/4.3.3-5, 3/4.4.1-2, 3/4.4.9, 3/4.5.1-2, 3/4.7.3, 3/4.9.11.

Our comments are enclosed.

Brian W. Sheron

Brian W. Sheron, Chief
Reactor Systems Branch
Division of Systems Integration

cc: R. W. Houston
D. Hoffman
RSB Section B Members

8311100049
XA

Comments on Grand Gulf Unit 1 Technical Specifications

ECCS

074 1. Table 3.3.3-2 - Isolation Actuation Instrumentation Setpoints

Footnotes are indicated for several items but are not provided. The footnote indication should be deleted or the footnote added.

075 2. Table 3.3.3-2 - ECCS Actuation Instrumentation Setpoints

The allowable value given for item A.2.f (LPCI discharge pressure-high) is different than that given in item B.2.e. Either value is acceptable, but the two should be the same.

076 3. Table 3.3.3-3 - ECCS Reponse Time

The LPCI A&B response time is given as ≤ 45 seconds. The LOCA analyses in Chapter 6.3 of the FSAR assumed LPCI injection at 40 seconds. The specification should be changed to ≤ 40 seconds.

4. Section 3/4.2.2 - APRM Setpoints

The time constant for the thermal power monitor needs to be included in the LCO's and surveillance requirements.

TECHNICAL SPECIFICATION PROBLEM SHEET

Item Number: 083 Priority: ~~2A~~ 1D

NRR 1/24/84 /

Identified By Date

Responsible Supervisor

Tech Spec Reference: 3/4.6.3.4

Problem Title: Suppression Pool Makeup Instrumentation

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other):
The actuation instrumentation for the suppression pool makeup system (in Tech Spec 3/4.6.3.4) is not included in the instrumentation section of the Tech Specs.

2. Safety Significance:
Even though the instrumentation is not specifically in the Tech Specs, by definition of operability, the instrumentation is required to be operable per Tech 3/4.6.3.4.

3. Anticipated Resolution:
The Tech Spec should be revised to include the Suppression Pool Makeup System Actuation Instrumentation. Evaluation should show not required, if is required use surveillances and position statements.

4. NRC Response to Item (NRR/IE): ______

NRC Notified: ______ Individual Notified ______ / ______ Date Time

5. Disposition: ______

Items Closed: (How) ______

______ / ______ Date Time

cc: J. E. Cross
R. F. Rogers

Drywell leakage tests are performed with the drywell isolated from the containment. The upper and lower containment pools are filled to normal water level, and the containment air space external to the drywell is vented to the secondary containment atmosphere. The horizontal vents are capped for the preoperational tests to achieve the design drywell internal pressure. The reduced pressure test pressure is less than that required to cause drywell air to bubble through the horizontal vents to the wetwell. The drywell atmosphere is allowed to stabilize for a period of one hour after attaining test pressure. Leakage rate tests commence after the stabilization period.

The maximum allowable in leakage rate into the secondary containment and the means to verify that the inleakage rate has not been exceeded and the bypass leakage rate is discussed in subsection 6.2.3.

The test method is based on drywell atmosphere pressure observations and the known drywell free air volume specified in Table 6.2-47. Leakage rate is calculated from the pressure data, drywell free air volume, and elapsed time.

The periodic drywell leakage test pressures, test duration, and acceptance criteria are specified in Chapter 16. Periodic drywell structural leakage tests are performed at intervals specified in Chapter 16.

The preoperational drywell leakage test shall be limited to the maximum allowable leakage rate of 84,000 scfm at drywell design pressure (30 psig) test and maximum allowable leakage rate of 3,500 scfm at drywell reduced pressure (3 psig) test. Preoperational drywell leakage tests are performed as late as is practical in the construction sequence, but before initial operation. Test duration shall be for a minimum of four (4) hours.

6.2.7 Suppression Pool Makeup System

The suppression pool makeup system provides water from the upper containment pool to the suppression pool by gravity flow following a LOCA. The quantity of water provided is sufficient to account for all conceivable post-accident entrapment volumes (i.e., places where water can be stored while maintaining long-term drywell vent water coverage).

6.2.7.1 Design Basis

The following criteria were used in the design of suppression pool makeup system:

a. The system is redundant with two 100 percent capacity lines. The redundant lines are physically separated and the electrical power and control is separated into two divisions in accordance with IEEE Std 279.

b. The system is Safety Class 2, seismic Category I, and quality group B. |53

c. The minimum long-term post-accident suppression pool water coverage over the top of the top drywell vent is 2 ft.

d. The minimum normal operation LWL suppression pool height above the top drywell vent center line is 7 ft. - 1/3 in. |35 |53

e. The maximum normal operation HWL suppression pool height above the top drywell vent center line is 7 ft. - 6 in.

f. The suppression pool volume, between normal LWL and the minimum post-accident pool level, plus the makeup volume from the upper pool is adequate to supply all possible post-accident entrapment volumes for suppression pool water.

g. The post-accident entrapment volumes causing suppression pool level drawdown include:

1. The free volume inside and below the top of the drywell weir wall.

2. The added water volume needed to fill the vessel from a condition of normal power operation to a post-accident complete fill of the vessel, including the top dome.

3. Volume in the steam lines out to the first MSIV for three lines and out to the second MSIV on one line.

4. An allowance for containment spray holdup on equipment and structural surfaces.

h. No credit for feedwater or HPCS injection from condensate is taken in calculating minimum post-accident suppression pool level.

i. The minimum freeboard distance from suppression pool HWL to the top of the weir wall is adequate to store the upper containment pool makeup volume without flooding into the drywell over the weir wall in case of an inadvertent dump of the upper pool.

j. The minimum normal operation suppression pool volume at LWL is adequate to act as a short-term energy sink without taking credit for upper pool dump. The short-term energy load on the pool consists of hot standby operation for 1-1/2 hours followed by a LOCA.

k. The long-term containment pressure and suppression pool temperature takes credit for the volume added post-accident from the upper containment pool.

l. The upper pool makeup volume dumps within a period so that a minimum vent coverage of 2 feet above the top edge of the top vent is maintained, considering 1) maximum runout flow of all five ECCS pumps, 2) the initial suppression pool water level is at LLWL, and 3) inventory addition to the drywell is through the postulated pipe break.

6.2.7.2 System Design

The piping system consists of two lines which penetrate the separator end of the upper containment pool through the side walls. One line is on either side of the separator pool and then routed down to the suppression pool on opposite sides of the steam tunnel. The elevation of the separator pool penetrations is such as to limit the volume of water which can be dumped to the lower pool. This volume limitation along with adequate weir wall freeboard ensures that no drywell flooding over the weir wall will occur for inadvertent opening of the valves on the suppression pool makeup lines.

The volume of the upper containment pool which is available for suppression pool makeup consists of a maximum 7'-4"-thick slice across the entire upper pool surface area plus the separator pool volume between the top of the separator wall and the makeup system penetration to the upper pool (see Table 6.2-50 for suppression pool geometry). This requires that the refueling gate leading to the dryer storage/fuel transfer pool be removed during power operation. | 55

Each suppression pool makeup line has two normally closed motor-operated butterfly valves in series. The power supply to valves on one line is on the same electrical division. The power supply to the valves on the second line is on a second electrical division. Electrical power is powered from onsite emergency power sources which have divisional separation and redundancy. | 55

The upper pool is dumped by gravity flow after opening the two normally closed valves in series in each line. The valves on both lines receive divisionally separate signals to open. The open signal for each division is derived from either of two suppression pool level sensors. There are a total of four level sensors, two per division.

The dump of the upper pool on low-low suppression pool level insures adequate water volume to keep the suppression pool vents covered for all break sizes. In addition to the low-low suppression pool level dump signal, the upper pool will also be dumped automatically from a timer set for LOCA plus 30 min. This upper pool dump at 30 min. post accident insures that adequate heat sink is available long-term regardless of break size or energy dump sequence. There is also a permissive permitting valve opening only when the LOCA signal exists. This LOCA signal is the same signal which initiates actuation of the ECCS pumps. This combination provides high reliability for the upper containment pool dumping when required but low probability of inadvertent dump by spurious signals. See Figure 6.2-82 for the system P&ID.

The two valves in series on each of the two makeup system dump lines are located near the top of the drywell (approximate elevation 170 ft.) and outside the range of pool swell effects. The makeup system pipes are routed along-side and supported from the drywell wall.

The pipes terminate just below the lowest operating floor support beams to provide an unobstructed free fall to the suppression pool surface. The termination is above the pool high water level to eliminate air clearing loads. The pool swell loading on the makeup system pipe is expected to be relatively small due to the minimum drag cross section of the vertical open-ended cylindrical geometry. Representative tests in the Mark III test facility will help determine the magnitude of side loads from pool swell. The pipe schedule and support design include the effects of internal pressure, seismic loads, and pool swell loads near the suppression pool surface.

Table 6.2-50 gives suppression pool geometry values consistent with the suppression pool makeup system.

6.2.7.3 Design Evaluation

6.2.7.3.1 Initiation

The opening of the makeup system valves is signaled by a series combination of low-low suppression pool level and a LOCA signal permissive (further discussion in subsection 6.2.7.2). The low-low level signal is 18 in. below the normal LWL. Since maximum ECCS pump flow lowers the suppression pool at a rate of 0.86 ft./min., there is a minimum 1-1/2 min. delay between start of ECCS flow and dumping of the upper pool. The delay is actually 1 to 2 min. longer than this because vessel inventory mass is added to the suppression pool during blowdown steam condensation.

This built-in volume integrated delay assures that the drywell pressure transient due to vessel blowdown has ended prior to dumping of the upper pool and corresponding increase of vent submergence.

The makeup system dump valves can also be signaled to open by a LOCA signal in series with a 30-min. timer where the timer itself is started by the LOCA signal. This path of initiation logic is in parallel with the suppression pool low level along with a LOCA permissive and is specifically directed towards insuring that the combined upper pool and suppression pool volumes are available as a heat sink for small breaks which do not lower the suppression pool to the LLWL trip, but continue to dump vessel blowdown energy into the pool. The minimum suppression pool volume, without upper pool dump is adequate to meet all heat sink requirements for any combination sequence of vessel blowdown energy and decay heat energy out to 30 min. A pool dump initiated from the LOCA plus 30 min. timer could result in higher vent submergence than the initial maximum of 7ft. - 6 in. This is no problem in terms of pool swell since all the air would have been purged out of the drywell by the small break flow and only a small steam suppression pool vent flow will persist out to 30 min. Note that action of the drywell vacuum breakers which might re-introduce air into the drywell prior to 30 min. post accident will occur only after complete vessel depressurization and drywell steam condensation on the cold ECCS break overflow of a relatively large break. The hypothesized high vent submergence will also have no effect on peak drywell pressure since the high submergence will occur only during small break flow events and after suppression pool vent clearing had already been established.

6.2.7.3.2 Flow

The suppression pool makeup volume is dumped in approximately 7.5 min. through one of two dump lines. The valves on the suppression pool makeup lines are fully opened within 60 seconds of opening signal application.

6.2.7.3.3 Inadvertent Dump

The design of the opening signal for the suppression pool makeup valves assures high probability that no inadvertent dump will occur. The suppression pool level signal (LLWL) to open the valves is in series with a permissive which allows only the open signal to pass through when a LOCA signal exists on that division. Only a simultaneous signal of suppression pool LLWL and LOCA will automatically open both valves to allow gravity drain of the upper pool to the suppression pool until 30 minutes have passed.

The automatic LOCA signal which provides a permissive for upper pool dump is paralleled with the manual ECCS initiation signal for the respective Divisions 1 and 2. Thus, the upper pool can be dumped manually in accordance with IEEE Std 279; however, there is still single failure protection against inadvertent dump. The LOCA signal plus the timer signal after 30 min. will dump the upper pool. However, the LOCA signal itself is a one-out-of-two-twice combination of high drywell pressure and low vessel water level (see Figure 6.2-82) and a double failure is required to give a spurious LOCA signal.

There are four level switches indicating suppression pool water level with two switches per electrical division. The two level switches in one division are paralleled so that either switch will initiate suppression pool makeup flow (pending LOCA permissive) from the makeup line whose series valves are on the same electrical division as the level switches. Level switches on one electrical division cannot initiate flow from the makeup line whose valves are in a separate electrical division.

There is a remote possibility that a single failure of a suppression pool level switch and a concurrent LOCA event can initiate suppression pool makeup flow from one line so that the makeup flow started at the instant of LOCA. The flow from one makeup line will raise the suppression pool level at a rate of 0.88 ft./min. following full opening of the valves which normally prevent flow. |27

For a large break DBA, the peak drywell pressure occurs at about 1 sec. after the break with the pressure being reduced to the steady flow submergence of the top vent by about 100 sec. Any pool swell induced loading will occur during the first few seconds while drywell air purge is taking place. Thus, the structural loading which will occur following a DBA will occur prior to any significant flow of water from a makeup line which was erroneously signaled to open at the same instant as the DBA.

The peak structural loadings associated with breaks smaller than the DBA are all less than the DBA case and only slightly extended in time. The drywell pressure for all size breaks is reduced to steady flow top vent submergence by 2 min. after the break.

The conclusion is thus that there is no increase in maximum structural loading due to a LOCA when an erroneous signal to initiate suppression pool makeup flow occurs at the instant of LOCA.

An inadvertent dump of the upper pool during any period of plant operation with a pressurized vessel does not represent, in and of itself, any hazard to the public, the plant operating personnel or any plant equipment. The drywell weir wall has sufficient freeboard height between the suppression pool surface and the top of the weir wall to store the entire upper pool makeup volume on top of the normal suppression pool HWL without flooding over the weir wall into the drywell. The only concern is for the extremely low probability that a LOCA might occur during this period of high vent submergence following inadvertent dump. The dumped upper pool makeup volume can be transferred back to the upper pool through the RHR pumps with a 13 min. pumping time. Operating at maximum flow, thus restoring the initial suppression pool water level.

No fuel is stored in the upper pool during plant operation so shielding is no issue for this case. Fuel can be temporarily stored in one end of the upper pool during fuel transfer as part of the refueling operation. This temporary storage pool has sufficient depth that adequate shielding is maintained over the fuel even following inadvertent dump of the upper pool makeup volume to the suppression pool. The 16-foot-high separator wall limits the water height drop over the temporary storage pool to an 8-foot change. This would leave approximately 20 feet of shielding over the top of active fuel temporarily stored even after inadvertant dump.

The only inadvertent dump event which represents a possible hazard to plant operating personnel is a dump event which occurs while fuel is in an elevated position, such as for transit between the reactor cavity and the fuel transfer pit. An 8 ft. upper pool level drop with one bundle in the highest position leaves approximately 1 ft. of water shielding over the top of active fuel. This is adequate for bundle cooling but represents a potential hazard for the plant operating personnel. Radiation alarms at the top of the upper pool will warn personnel to evacuate from the edge of the pool. Several minutes will be available for personnel to step to a safe shielding area out of line of sight of the suspended fuel bundle which is 9 ft below the operating floor. The valve initiation logic is designed with interlocks so that neither automatic nor manual action can open the suppression pool makeup valves while the plant is in the refueling mode.

6.2.7.3.4 Long Term Heat Sink Capability

The capacity of the RHR heat exchangers to safely limit the long-term, post-LOCA suppression pool heatup transient is evaluated on the basis that the drawdown makeup system is activated early in the transient. Specifically, the evaluation assumes that the heat exchangers are activated one-half hour after the LOCA and that at this time the drawdown makeup system water has been added to the main suppression pool inventory. The makeup 30-min. timer will ensure that this condition will exist. The 3.75-min. dump period (8 min. if only one line is operative) is not significant compared to the several hours it takes for the suppression pool peak temperature to be reached.

6.2.7.4 Testing

The suppression pool makeup valves will be periodically manually tested, one at a time, during plant power operation. An interlock prevents this manual testing unless the other valve in series on the same line is closed. The test will verify that the valve will open and close.

Instruments will be periodically tested and inspected.

Preoperational testing will include a complete flow test of the system including a timed dump of the entire makeup volume. Similar flow testing can be performed at any plant shutdown outage; however, the need of such testing is only necessary a few times in the plant lifetime.

6.2.7.5 Instrumentation

There are four suppression pool level sensors, and four suppression pool instrumentation channels, two per division. Level sensor actuation signals for suppression pool makeup in a single electrical division are parallel so that either level sensor provides a signal to open the series valves on only the suppression pool makeup line in the same electrical division as the level sensors.

The level channels are used to monitor continuously the suppression pool level and annunciate at the HWL and LWL setpoints. Each channel is indicated and recorded in the control room. In addition, the LLWL setpoint will both annunciate and provide a signal to actuate the suppression pool makeup flow. The LWL setpoint is 3-1/2 in. below the HWL setpoint, and the LLWL setpoint is 19-1/2 in. below the LWL setpoint, for a total spread of 23 in. between the HWL and LLWL setpoints. 53

ONLY TWO CHANNELS ARE RECORDED

Each level channel consists of 1) a sealed differential pressure cell located beneath the surface of the suppression pool in a stilling well to reduce turbulence and give a better average reading during steam release into the suppression pool; 2) a differential pressure transmitter mounted locally in the containment; 3) power supply located in the control room, and 4) information and alarm outputs located in the control room.

The impulse lines from the sensor to local transmitter are routed so that they will not be damaged by pool swell.

The four level sensors are distributed around the suppression pool with an approximately 90 degree azimuth between them.

An erroneous suppression pool LLWL signal coincident with a LOCA signal which thus results in the initiation of the suppression pool inventory makeup system early in a postulated LOCA has no effect on peak structural loading. The drywell peak pressure for a large break DBA occurs in approximately one second with drywell pressure being reduced to a value equivalent to steady flow against the hydrostatic submergence of the top vents by approximately 100 sec. Peak pool swell loading also occurs in the first few seconds of the DBA while drywell air purge is taking place. No significant increase in vent submergence due to suppression pool makeup flow initiated at the instant of LOCA can occur prior to these peak structural loads. One suppression pool makeup line will raise the suppression pool water level at a rate of approximately 5 ft./min. after the valves are full open.

A level indication for the upper pool is also required to obtain the attention of plant operating personnel if the level drops below that needed for the makeup volume. Level in the upper pool is normally maintained by a continuous overflow of level control weirs. The level is expected to stay nearly constant during plant power operation.

The upper pool and suppression pool temperatures are monitored to insure that the temperature does not exceed technical specification values. This ensures adequate heat sink capability of the suppression pool water, both short and long term.

An annunciator will bring the operator's attention to the situation of having the fuel transfer gate in place while the reactor is in the RUN mode. This gate is left open during plant operation where suppression pool makeup from the upper pool might be required.

6.2.7.6 Materials

The piping which penetrates the separator pool, and welds to the stainless steel pool liner is stainless steel. Piping and valves beyond penetration of the upper containment pool are carbon steel.

6.2.8 References

1. BN-TOP-4, Rev. 1, "Subcompartment Pressure and Temperature Transient Analysis," October, 1977. 3

2. I. E. Idel Chik, "Handbook of Hydraulic Resistance Coefficients of Local Resistance and Friction," USAEC-TR-6630, 1966.

3. Crane Co., "Flow of Fluids," Technical Paper No. 410, 1969, Engineering Division, New York, N.Y.

4. WCAP 7709-L, Supplement 1, Electric Hydrogen Recombiner," April 1972.

5. WCAP 7709-L, Supplement 2, "Electric Hydrogen Recombiner Equipment Qualification Report," September 1973.

6. WCAP 7709-L, Supplement 3, "Electric Hydrogen Recombiner Long Term Tests," January 1974.

7. WCAP 7709-L, Supplement 4, "Electric Hydrogen Recombiner," April 1972.

8. GE Topical Report NEDO-10569A Performance Description for BWR/6 Nuclear System.

9. GE Topical Report NEDO-11013-77 An Analytical Procedure for the Conservative Calculation of Core Metal-Water Reaction Following a Design Basis Load of Coolant Accident.

10. Moody, F.J., "Maximum Flow Rate of a Single Component, Two Phase Mixture," Journal of Heat Transfer, ASME series C, vol. 87, p. 134.

11. James, A. J., "The General Electric Mark III Pressure Suppression Containment Analytical Model," June 1974 (NEDO-20533).

12. James, A. J., "The General Electric Mark III Pressure Suppression Containment Analytical Model," Supplement 1, September 1975 (NEDO-20533-1).

13. Moody, F. J., "Maximum Two Phase Vessel Blowdown from Pipes," Topical Report APED-4827, General Electric Company, 1965.

14. Licensing Topical Report NEDO 10977 "Drywell Integrity Study: Investigation of Potential Cracking for BWR 6/ Mark III Containment," August 1973.

15. Slifer, Bruce, NEDO-10329, "Loss-of-Coolant Accident and Emergency Core Cooling Models," April 1971.

16. ANSI Std. N274 (Draft) dated June 25, 1976, "Containment System Leakage Testing Requirements."

17. "Corrosion Data Survey", compiled by G. A. Nelson, National Association of Corrosion Engineers (Publishers), 1968. Library of Congress Catalog Card No. TA.462.N26. 021.33

18. New Jersey Zinc Co., "Metals Handbook," ASTM Vol. 1, p. 1163, 8th Ed. 44

7.3.1.1.9 Suppression Pool Makeup System

7.3.1.1.9.2 Logic

The suppression pool makeup system consists of two independent and redundant systems. LOCA initiation logic is based on a one-out-of-two-twice arrangement of sensors. Low suppression pool level initiation logic is based on a one-out-of-two arrangement of sensors. Once initiated, the bistable nature of the final control elements of the system seals in the protective action unless terminated by the operator.

Each of the two logic channels is made up of individual solid state bistables and electromechanical relays to provide independence of control components and preserve the independence of the associated mechanical equipment.

In order to minimize the probability of an inadvertent dump of the upper containment pool due to a component failure, separate actuation circuit components are provided for each of the two final control elements (valves) in each of the two redundant channels.

7.3.1.1.9.3 Bypasses

The suppression pool makeup system is bypassed to prevent inadvertent actuation during refueling. The system can also be bypassed manually from a handswitch in the control room. This condition is continuously annunciated.

7.3.1.1.9.4 Interlocks

The suppression pool makeup system is interlocked to prevent actuation without a coincident LOCA. An interlock between the two valves in each line is provided when in the operating mode to prevent inadvertent manual opening of both valves in one line while testing the valves.

7.3.1.1.9.5 Sequencing

A timer is provided in the suppression pool makeup system to initiate the system 30 minutes after a LOCA is detected. Refer to subsection 8.3.1.1 for a discussion of ESF bus load sequencing. There is no other automatic sequencing in the suppression pool makeup system.

7.3.1.1.9.6 Redundancy

Two completely independent and redundant suppression pool makeup lines are provided, including independent and redundant logic systems and mechanical equipment. The two logic systems and their associated motor operated valves are powered from separate ESF buses. Physical and electrical separation is maintained between the two systems.

TABLE 7.3-26

SUPPRESSION POOL MAKEUP

SYSTEM ACTUATED EQUIPMENT LIST

Equipment Number	Figure No. (P&ID)	Description	ESF Division	Function	Signal
F001A	6.2-82	Suppression pool makeup valve	1	OPEN	LOCA, low supp. pool level, manual
F001B	6.2-82	Suppression pool makeup valve	2	OPEN	LOCA, low supp. pool level, manual
F002A	6.2-82	Suppression pool makeup valve	1	OPEN	LOCA, low supp. pool level, manual
F002B	6.2-82	Suppression pool makeup valve	2	OPEN	LOCA, low supp. pool level, manual

TABLE 7.3-27

FAILURE MODES AND EFFECTS ANALYSIS
SUPPRESSION POOL MAKEUP SYSTEM

Failure Mode	Effect on System	Detection	Remarks
Loss of plant instrument air	None	Immediate annunciation on loss of instrument air	
Loss of cooling water systems	None	Annunciation on effected system	
Loss of Power Systems:			
Loss of one ESF ac bus	Motor operated dump and instrument line isolation valves on one system fail as-is	Immediate annunciation on loss of bus	Protective action provided by redundant system if required.
Loss of one ESF dc bus	Affected system fails as-is, cannot be initated	Immediate annunciation on loss of bus	Protective action provided by redundant system if required. If system is in operat: operator may still clo: dump valve by using sy: bypass switch.
Level Sensor failure:			
Upscale	No immediate effect	Periodic testing	
Downscale	Immediate system initiation upon LOCA	Low level alarm on affected channel	No adverse effects on drywell structure or sy operation
AUTO - OFF bypass switch failure			
AUTO	Dump valves cannot be reclosed after system operation.	Periodic testing	
OFF	One system will not operate.	"SPMU System out of service" annunciation on periodic testing.	Protective action prov: by redundant system if quired.
System initiate switch failure			
Open	Prevents manual initiation after LOCA	Periodic testing	Automatic initiation circuits still operati redundant system can b manually initiated if : quired.
Closed	Reduce coincidence required for manual initiation after LOCA	Periodic testing	

"TECH SPEC PRIORITY"

Punchlist Item # 83

Tech Spec 3/4.6.3.4

Priority 1D

TO: Manager of Nuclear Plant Engineering

FROM: Chairman, Prioritization and Disposition Chairman

SUBJECT: Technical Specifications Punchlist Item # 83

PDTS:84/ 0012

DATE: 3/10/84

The subject Tech Spec item has been determined by the Disposition Committee to require Engineering support.

DETAILS: Determine what credit was taken for the Suppression Pool Makeup System for the following cases:

1) LOCA plus (+) 30 minutes actuation

2) LOCA and Suppression Pool Level

Does the level of accident analysis dependence on the above events justify adding more details to the specification (ie. setpoints, minimum operable channels, calibration requirements, etc.)

Your response is required by 3/13/84

Please contact Geo. Zinke at Extension 2695 for further information.

Please refer to the Tech Spec Punchlist item number in your response. Forward your response to Geo. Zinke.

J. C. Roberts
Chairman

LLJ/JCR:sub
cc: Mr. C. L. Tyrone
Mr. J. E. Cross
Mr. D. Stonestreet
Mr. A. S. McCurdy
Mr. S. Hutchins
Mr. J. Hendry
File (Tech Spec Records)

A4.01SWD.

TECHNICAL SPECIFICATION PROBLEM SHEET

Item Number: 083 Priority: 2

NRR 1/24/84 /

Identified By Date Responsible Supervisor

Tech Spec Reference: 3/4.6.3.4

Problem Title: Suppression Pool Makeup Instrumentation

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other):

The actuation instrumentation for the suppression pool makeup system (in Tech Spec 3/4.6.3.4) is not included in the instrumentation section of the Tech Specs.

2. Safety Significance:

Even though the instrumentation is not specifically in the Tech Specs, by definition of operability, the instrumentation is required to be operable per Tech 3/4.6.3.4.

3. Anticipated Resolution:

The Tech Spec should be revised to include the Suppression Pool Makeup System Actuation Instrumentation.

4. NRC Response to Item (NRR/IE): ____

NRC Notified: ____ Individual Notified ____ / ____ Date Time

5. Disposition: ____

Items Closed: (How) ____

/

Date Time

cc: J. E. Cross
R. F. Rogers

Item No. ~~I~~ 11 H — TECHNICAL SPECIFICATION PROBLEM SHEET — Priority ____

MP+L ____ / ____ ____
Identified By — Date — Responsible Supervisor

Tech Spec Reference: 4.6.3.4.c. (P 3/4 6-26)

Problem Title: SUPPRESSION POOL MAKE UP SYSTEM HAS NOT MINIMUM OPERABLE CHANNELS SPECIFIED.

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other):

THE SPMU SYSTEM HAS NO MINIMUM OPERABLE CHANNEL SPEC.

2. Safety Significance:

3. Anticipated Resolution:

4. NRC Response to Item (NRR/IE):

NRC Notified: ____ ____ / ____
Individual Notified — Date — Time

5. Disposition:

Items Closed: (How)

____ / ____
Date — Time

cc: J. E. Cross
R. F. Rogers

III

Item No. P/L 084 TECHNICAL SPECIFICATION PROBLEM SHEET Priority ______

3

1/24/84

NRR A 1/24/84

Identified By Date Responsible Supervisor

IE 2/24/84

Tech Spec Reference: 3.6.3.4 p 3/4 6-26

Problem Title: SUPPRESSION POOL MAKE-UP ~~INSTRUMENTATION~~ NOT CALIBRA... OR MODE APPLICABILITY INADEQUATE

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other):

A. SUPPRESSION POOL MAKE-UP (SPMU) ~~[illegible]~~ TIMERS ARE NOT NOW CALIBRATED BY T.S. REQS. (ARE COVERED UNDER PM PROGRAM).

B. TRANSMITTERS/TRIP UNITS NOW CALIBRATED UNDER ~~P~~ ACCIDENT MONITORING INST. ~~[illegible]~~ T.S. (3.3.7.5) WHICH ONLY REQUIRES IT IN MODES 1+2 VICE 1,2 + 3. (P 3/4 3-69) AS REQUIR...

2. Safety Significance: MINOR - NEEDS TO BE FIXED HOWEVER.

3. Anticipated Resolution: A. ADD TIMERS TO T.S.

B. ADD MODE 3 TO TABLE IF APPROPRIATE FOR INSTRUMENT

4. NRC-Response to Item (NRR/IE):

NRC Notified: ______ ______ /

Individual Notified Date Time

5. Disposition:

Items Closed: (How)

/

Date Time

cc: J. E. Cross
R. F. Rogers

ITEM 084 | NRC IRE 2/24/84
NRR 1/24/84
(TS/FSAR/SER/...)

3.6.3.4 & FSAR 6.2.7.5

NRC Inspector identified during exit (2/24) that our Tech Specs were deficient ~~for not~~ including the suppression pool ~~makeup instrument~~

PRIORITY

SIGNIFICANCE

If a new spec is requested, it may need to be administratively implemented in the interim. The 3.6.3.4 is adequate ~~right now though~~ to cover operability of SPMU. All instruments calibrated by other specs except for the time which are under ~~the~~ our P.M. program. Note: The Tech Specs which calibrate the transmitters/trip units do not have the same mode applicability as SPMU.

ACTION

COMPLETE

DATE (MHL).

PLANNED RESOLUTION

Determine need for change & submit if appropriate. Also correct FSAR.

DATE NRC NOTIFIED & NAME

NRC REACTION

This item was identified by NRR I&C Branch on 10/31/83. It was identified to MP&L by NRR on 1/24/84. It was reidentified by Region I in the 2/24/84 exit as if it were a new item.

ITEM CLOSED/COMMENTS

DATE & NAME

DISPOSITION

B21-LIS-N691A (Low Wl - Wl 1)
B21-PIS-N694A (D/W Press - Hi)
Either

B21-LIS-N691E (Low Wl - Wl 1)
B21-PIS-N694E (D/W Press - Hi)
Either

Both

LPCS Manual Initiation (E21-HS-M613)

Either

E30-HS-M600A (E30 Manual Init)
E30-HS-M600C (E30 Manual Init)
Both

[illegible] (Supp [illegible] Lvl)
[illegible] (Supp [illegible])

Either

Both

C71-PIS-N650A (D/W Press Hi)
[illegible]
E21-HS-M620A (Manual Init)
Either

C71-PIS-N650E (D/W Press - Hi)
B21-LS-[illegible] (Lo Wl - 2)
[illegible] (Manual Init)
Either

Both - 30 Min timer

Either

* Opens E30-F001A and [illegible]-F001A
(Comp. Supp. Containment [illegible])

B21-LIS-N691E (Lo Lvl-1)] Either
B21-PIS-N694E (D/w Press-Hi)
B21-LIS-N691F (Lo Lvl-1)] Either
B21-PIS-N694F (D/w Press-Hi)

Both

RHR B/C Man Init (E12-HS-M617)

Either

Both

E30-HS-M600B] Both
E30-HS-M600D

Either

E30-LIS-N600B
E30-LIS-N600D

Either*

C71-PIS-N650E (D/w Press-Hi)
[illegible]
B21-HS-M630E (Manual Init)

Either

C71-PIS-N650C (D/w Press-Hi)
B21-LIS-[illegible] (Low Lvl-2)
E21-[illegible] (Manual Init)

Either

Both - 30 Minute timer

*Opens E30-F001B and E30-F001E

TECHNICAL SPECIFICATION PROBLEM SHEET

Item Number: 102 Priority: ~~X+A~~ 1C

__________ / __________
Identified By Date

____
Responsible Supervisor

Tech Spec Reference: 3/4.3.7.9

Problem Title: Fire detection instrumentation

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other): Revise Tech Specs to include smoke detectors as specified MPB 83/0328.

2. Safety Significance: Adding smoke detectors will enhance the capability to detect a fire in safety-related equipment.

3. Anticipated Resolution: Add to Tech Specs.
Could use same position statement as item 73

4. NRC Response to Item (NRR/IE): __________

NRC Notified: __________ ______ / ______
Individual Notified Date Time

5. Disposition: __________

Items Closed: (How) __________

__________ / __________
Date Time

cc: J. E. Cross
R. F. Rogers

~~TRANSMITTAL OF PROPOSED CHANGES~~
~~TO GRAND GULF TECHNICAL SPECIFICATIONS~~

~~1. (GGNS - 687) (Resubmittal of Item 1, AECM-83/0411)~~

SUBJECT: Technical Specification Table 3.3.7.9-1 and Technical Specification 3.3.7.9, pages 3/4 3-76 through 3/4 3-80.

NO. -102

DISCUSSION: Technical Specification Table 3.3.7.9-1 provides a listing of fire detection instrumentation. The present table is ordered by building and room numbers within a building whereas the Specification (3.3.7.9) governing the table is ordered by zones. One of the changes to Table 3.3.7.9-1 is to order the table by building, zone numbers within the building, and room numbers within the zone. The present listing of fire detection instruments does not distinguish them as either early warning or actuation of fire suppression systems devices. The proposed change adds Function A (early warning and notification) and Function B (actuation of fire suppression systems) to the table and to the ACTION statements of Technical Specification 3.3.7.9. Other changes to the Fire Detection Instrumentation Table 3.3.7.9-1 include the following:

1. Design Changes
2. Additions to the Table
3. Corrections to the Table

The changes to the table are discussed in detail in the justification section below.

JUSTIFICATION: The following changes to Table 3.3.7.9-1 are the result of design changes made to the fire detection system:

1. In Zone 1-4 for the Control Building, Area OC201 (stairwell) including one smoke detector was added to the table. Safety related cables pass through this area and appropriate fire detection was added to provide coverage.

2. In Zone 1-6 for the Control Building, OC216 (West Corridor) with two smoke detectors was added. Safety related cables pass through this area and appropriate fire detection was added.

3. In Zone 1-23 for the Control Building one smoke detector was added to provide additional coverage.

4. In Zone 2-9 for the Auxiliary Building, one smoke detector was added to provide additional coverage.

The following changes to Table 3.3.7.9-1 constitute additions of areas/zones (and their associated instrumentation) which were inadvertantly left out of the original Technical Specification.

1. Add Zone 1-3 at Elevation 93' in the Control Building. This zone consists of areas OC109 (Decontamination Area), OC115 (Corridor), OC116 (Hot Machine Shop), and OC117

(Corridor). These areas have safety related cable and should be included in the table. A total of 12 smoke detectors is added with these areas.

2. Add area OC306 (Electrical Chase) in Zone 1-10 at Elevation 133' in the Control Building. Safety related cable passes through these areas.

3. Add area OC410 (Battery Room) in Zone 1-14 at Elevation 148' in the Control Building. Specification 3.3.7.9 covers all instrumentation in each Zone listed in Table 3.3.7.9-1 and this area is added to complete the listing for Zone 1-~~10~~.14 The number of Smoke Detectors for Zone 1-~~10~~ 14 increased from 7 to 9 as a result of this change.

4. Add Zone 1-12 at Elevation 133', in the Control Building. This zone includes areas ~~OC304~~ OC304/OC4 2 (Electrical Space~~s~~) and OC305 (Electrical Space). Two additional smoke detectors are added with this change. Safety related cable passes through this zone.

5. Zone 1-13 at Elevation 133' in the Control Building is added since part of Unit 1 control room HVAC equipment is in this zone. Zone 1-13 includes area OC303 (HVAC Room) and adds 16 smoke detectors.

6. Areas OC401 (Corridor), OC408 (Corridor), and OC409 (Electrical Chase) are added to Zone 1-15 along with four additional smoke detectors. Specification 3.3.7.9 covers all instrumentation in each Zone listed in Table 3.3.7.9-1 and these areas (OC401, 408, 409) were added to complete the listing for Zone 1-15.

INSERT A →

8 ~~7~~. Zone 1-19 at Elevation 166' in the Control Building is added because safety related cable passes through this area. This zone includes areas OC514 (Locker Room) and OC506 (Shower AND adds 9 smoke detectors.

9 ~~8~~. Zone 1-21 at Elevation 166' in the Control Building is added because safety related cable passes through this area. This zone includes area OC518 (Electrical Chase) and adds two smoke detectors.

10 ~~9~~. Zone 1-22 at Elevation 177' in the Control Building is added because safety related cable passes through this area. This zone includes areas OC601 (Viewing Gallery), OC603 (Emergency Dormitory), ~~and~~ OC608 (Technical Support), ~~and~~ OC611 (Storage Closet) and also adds sixteen smoke detectors. AND NEW AREA HVAC CHASE (EL 177'),

11 ~~10~~. Areas OC706 (West Corridor), OC709 (Electrical Chase), ~~and~~ OC712 (HVAC Room), AND OC610 (ELECTRICAL CHASE) are added to Zone 1-23 at Elevation 189' in the Control Building. Specification 3.3.7.9 covers all instrumentation in each zone listed in Table 3.3.7.9-1 and these areas are added to complete the listing for Zone

INSERT A 7. AREAS OC516 (CABLE SPACE) AND OC517 (CABLE SPACE) ARE ADDED TO ZONE 1-18. SPECIFICATION 3.3.7.9 COVERS ALL INSTRUMENTATION IN EACH ZONE LISTED IN TABLE 3.3.7.9-1 AND THESE AREAS (OC516, ~~AND~~ OC517) WERE ADDED TO COMPLETE THE LISTING FOR ZONE 1-18.

INSERT B 13. ADD ZONE 1-20 AT ELEVATION 189' IN THE CONTROL BUILDING. THIS ZONE CONSISTS OF THE HVAC CHASE AT ELEVATION 189' AND ADDS ONE SMOKE DETECTOR.

1-23. The number of smoke detectors in Zone 1-23 is increased from 15 to 21 with one being added as a design change and five added with areas OC706, OC709, ~~and~~ OC712, and OC610.

12 ~~11~~. Add areas 1A128 (RHR "A" Heat Ex Room), 1A129 (RHR "B" Heat Ex Room), and 1A223 (Passage) to Zone 2-4 in the Auxiliary Building. Areas 1A128 and 1A129 are separated by grating from 1A102 and 1A106, respectively, and as such smoke detectors in area 1A102 serve 1A128 and in 1A106 serve 1A129. Area 1A223 is an area already served by Zone 2-4 instrumentation and is included to complete the Zone 2-4 listing.

INSERT B →

14 ~~13~~ ~~12~~. Add areas 1A524 (Platform) and 1A529 (FPC and CU Room) to Zone 2-9. These areas are added to complete the listing for Zone 2-9.

15 ~~14~~ ~~13~~. Add areas 1A101 (Passage), 1A114 (Fan Coil Area), and 1A117 (Misc. Equip. Area) to Zone 2-17. ~~This~~ These areas ~~is~~ are currently served by Zone 2-17 instrumentation and should be included.

16 ~~15~~ ~~14~~. In Zone 2-10 for the Diesel Generator Building, add three smoke detectors due to the addition of the corridor between the Auxiliary Building and Diesel Generator Building.

17 ~~16~~ ~~15~~. The type of fire protection initiated has been added to the Heat detector column. This administrative change indicates that Halon, CO_2 or Deluge is actuated by the heat detector.

18 ~~17~~ ~~16~~. The Control Room HVAC Intake Plenum Mounted Detectors have been added since they involve control room habitability.

19 ~~18~~ ~~17~~. PGCC Halon systems in the Control Building have been added as Section "g" of the Table. The list of Halon systems is broken down by room, elevation and also by Halon panels within the room. The panels are listed as underfloor modules/Halon panel (Example: 1H13-U713/1H13-P913) and the number of detectors associated with each Halon panel is shown.

20 ~~19~~ ~~18~~. The function designation of A or B (X or Y) is added to the Table to distinguish between those instruments that perform early warning fire detection and notification and those that also actuate fire suppression systems as well as give early warning and notification. The Function A and Function B designation is also added to the ACTION statements of Technical Specification 3.3.7.9 to provide consistency between the Technical Specification and its associated Table. The adding of the Function A and B designations follows the Standard Technical Specification format for Table 3.3.7.9-1. Due to Grand Gulf Fire Detection Instrumentation design the Standard Technical

Specification ACTION statements do not apply. With the addition of function A or B designation, present footnote (2) on page 3/4 3-77 is not needed and is deleted. Present footnote (2) is a duplication of the function A or B requirements.

The following changes to Table 3.3.7.9-1 constitutes correction of errors in the original Technical Specification:

1. Area OC308 (Corridor) at Elevation 133' in the Control Building is moved from Zone 1-10 to Zone 1-11.

2. The number of heat detectors in OC403 - Computer Room of the Control Building is corrected from present 13 to 12. This change does not reflect the deletion of a heat detector from the plant but only a correction to the table.

3. The number of Smoke Detectors in OC503 (Control Room) at Elevation 166' of the Control Building is changed from 17 to 16. This change reflects the temporary split of Unit 1 and Unit 2 Control Rooms. The detector deleted is on the Unit 2 side of the Control Room.

4. Area 1A211 [North Corridor (Partial)] is added to Zone 2-2 of the Auxiliary Building. This area overlaps into Zone 2-2 and also appears in Zone 2-18 as an overlap or interface area.

5. Area 1A314 [South Corridor (Partial)] is added to Zone 2-6 of the Auxiliary Building. This area overlaps into Zone 2-6 and also appears in Zone 2-19 as an overlap or interface area.

6. Area 1A424 [Set Down Area (Partial)] is added to Zone 2-8. This area overlaps into Zone 2-8 and also appears in Zone 2-7 as an overlap or interface area. The number of smoke detectors in Zone 2-7 goes from 12 to 11 and the number in Zone 2-8 goes from 24 to 25 due to Zone assignment of instrumentation.

7. Areas 1A122 [South Corridor (Partial)] and 1A123 [North Corridor (Partial)] are added to Zone 2-14 of the Auxiliary Building. These areas overlap into Zone 2-14 and also appear in Zone 2-17 as an overlap or interface area.

8. Zones 6-9A, 6-9B, and 6-9C for the Diesel Generator Building are corrected to 2-10, 2-11, and 2-12 respectively. Added Corridor between Diesel Generator and Auxiliary Building.

9. Added area numbers for Standby Service Water Pump House. These area numbers were inadvertently omitted.

The format change is proposed so that the Technical Specification will more accurately reflect that the operability of each individual smoke detector affects the entire zone, not just the area in which it is installed.

SIGNIFICANT HAZARDS CONSIDERATION:

The changes to the Fire Detection Instrumentation Table 3.3.7.9-1 constitute additions, corrections, and changes due to design changes to the plant. The design changes add additional equipment and enhance fire detection capability. The additions to the table also enhance fire detection capability. The corrections to the table do not decrease fire detection capability but reflect actual plant systems and instrumentation arrangements. This change does not involve a reduction of safety margins and no significant increase in the probability or consequences of an accident previously evaluated is involved nor is the possibility of a new or different kind of accident from any accident previously evaluated created. Thus the proposed change to the Technical Specifications does not involve any significant hazards considerations.

1.(GGNS-687)

TABLE 3.3.7.9-1
FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION				MINIMUM INSTRUMENTS OPERABLE* HEAT (X/Y)	FLAME(1) (X/Y)	SMOKE(1) (X/Y)
a.	CONTAINMENT BUILDING #					
	1. Return Duct Mounted Detectors					3/0
	ROOM	ELEV	ROOM NAME			
b.	CONTROL BUILDING					
	1. Zone 1-3					12/0
	OC109	93'	Decontamination Area			
	OC115	93'	Corridor			
	OC116	93'	Hot Machine Shop			
	OC117	93'	Corridor			
	2. Zone 1-4					6/0
	OC201	111'	Stairwell			
	OC202	111'	Div I Swgr Rm	0/6(CO_2)		
	OC207	111'	Div I Battery Rm			

* (X/Y): X - is number of Function A (early warning fire detection and notification only) instruments.
Y - is number of Function B (actuation of fire suppression systems and early warning and notification) instruments.

\# The fire detection instruments located within the primary containment are not required to be OPERABLE during the performance of Type A Containment Leakage Rate Tests.

(1) Smoke and flame detectors provide only early warning capability with the exception of:

(a) Zone 1-27 detectors trip closed the door between the OC208/OC208A Remote Shutdown panel rooms.
(b) Containment building return duct mounted detectors' trip the containment cooler fans.
(c) Zone 1-11 and 1-13 detectors initiate the control building purge fan system.
(d) Control Room HVAC Intake Plenum Detectors trip the control room A/C units unless a control room emergency filtration system isolation mode automatic actuation signal is present.

TABLE 3.3.7.9-1
FIRE DETECTION INSTRUMENTATION

ROOM	ELEV	ROOM NAME	MINIMUM INSTRUMENTS OPERABLE* HEAT (X/Y)	FLAME[(1)] (X/Y)	SMOKE[(1)] (X/Y)
3. Zone 1-5					3/0
OC209	111'	Div III Battery Rm			
OC210	111'	Div III Swgr Rm	0/4(CO_2)		
4. Zone 1-6					7/0
OC211	111'	Div II Battery Rm			
OC215	111'	Div II Swgr Rm	0/7(CO_2)		
OC216	111'	West Corridor			
5. Zone 1-10					2/0
OC306	133'	Electrical Chase			
OC307	133'	Electrical Chase			
6. Zone 1-11					13/0
OC302	133'	HVAC Equipment Rm			
OC308	133'	Corridor			
7. Zone 1-12					2/0
OC304/OC412	133'	Electrical Spaces			
OC305	133'	Electrical Space			
8. Zone 1-13					16/0
OC303	133'	HVAC Room			
9. Zone 1-14					9/0
OC403	148'	Computer Room	0/12(Halon)		
OC410	148'	Battery Room			
10. Zone 1-15					15/0
OC401	148'	Corridor			
OC402	148'	Lower Cable Spreading Room	0/7(CO_2)		
OC407	148'	Instr. Motor Gen Rm	0/2(CO_2)		
OC408	148'	Corridor			
OC409	148'	Electrical Chase			
11. Zone 1-18					16/0
OC503/OC516	166'	Control Rm (Unit 1 Side)/Cable Space			
OC504/OC517	166'	U-1 Inst Rack Area/Cable Space			

TABLE 3.3.7.9-1
FIRE DETECTION INSTRUMENTATION

ROOM	ELEV	ROOM NAME	HEAT (X/Y)	FLAME[1] (X/Y)	SMOKE[1] (X/Y)
			MINIMUM INSTRUMENTS OPERABLE*		
12. Zone 1-19					9/0
OC514/OC506	166'	Locker Room/Shower			
13. Zone 1-21					2/0
OC518	166'	Electrical Chase			
14. Zone 1-22					16/0
—	177'	HVAC Chase			
OC601	177'	Viewing Gallery			
OC603/OC616	177'	Emergency Dormitory/Storage Closet			
OC608	177'	Technical Support			
15. Zone 1-23					21/0
OC702	189'	Upper Cable Spreading Room	0/12(CO_2)		
OC706	189'	West Corridor			
OC707	189'	Instr. Motor Gen Rm			
OC709/OC619	189'	Electrical Chases			
OC712	189'	HVAC Room			
16. Zone 1-24					6/0
OC703	189'	Control Cabinet Area	4/0(CO_2)		
17. Zone 1-27					2/0
OC208	111'	Div ~~I~~ Remote Shutdown Panel	0/1(CO_2)		
OC208A	111'	Div ~~II~~ Remote Shutdown Panel	0/1(CO_2)		
18. Control Room HVAC Intake Plenum Mounted Detectors					2/0
19. Zone 1-20	189'	HVAC Chase (No Rm. No.)			1/0

c. AUXILIARY BUILDING

ROOM	ELEV	ROOM NAME	HEAT (X/Y)	FLAME (X/Y)	SMOKE (X/Y)
1. Zone 2-2					23/0
1A211	119'	North Corridor (Partial)			
1A215	119'	South Corridor			
1A222	119'	West Corridor			

TABLE 3.3.7.9-1
FIRE DETECTION INSTRUMENTATION

ROOM	ELEV	ROOM NAME	MINIMUM INSTRUMENTS OPERABLE* HEAT (X/Y)	FLAME(1) (X/Y)	SMOKE(1) (X/Y)
2. Zone 2-3					5/0
1A219	119'	Electrical Swgr Rm	0/2(CO_2)		
1A220	119'	Piping Penetration Rm			
1A221	119'	Electrical Swgr Rm	0/2(CO_2)		
3. Zone 2-4					22/0
1A102	93'	RHR "A" Heat Ex Rm			
1A103	93'	RHR "A" Pump Rm			
1A104	93'	RCIC Pump Rm			
1A105	93'	RHR "B" Pump Rm			
1A106	93'	RHR "B" Heat Ex Rm			
1A128	108'	RHR "A" Heat Ex Rm			
1A129	108'	RHR "B" Heat Ex Rm			
1A202	119'	RHR "A" Heat Ex Rm			
1A203	119'	Piping Penetration Rm			
1A204	119'	Piping Penetration Rm			
1A205	119'	Piping Penetration Rm			
1A206	119'	RHR "B" Heat Ex Rm			
1A207	119'	Electrical Swgr Rm	0/3(CO_2)		
1A208	119'	Electrical Swgr Rm	0/3(CO_2)		
1A209	115'	RWCU Recirc Pump "A" Rm			
1A210	115'	RWCU Recirc Pump "B" Rm			
1A223	128'	Passage			
4. Zone 2-5					5/0
1A318	139'	Electrical Penetration Room	0/2(CO_2)		
1A319	139'	RPV Instr Test Rm			
1A320	139'	Electrical Penetration Room	0/2(CO_2)		
5. Zone 2-6					26/0
1A301	139'	East Corridor			
1A302	139'	Southeast Corridor			
1A303	139'	RHR "A" Heat Ex Rm			
1A304	139'	Piping Penetration Rm			
1A306	139'	Piping Penetration Rm			
1A307	139'	RHR "B" Heat Ex Rm			
1A308	139'	Electrical Penetration Room	0/3(CO_2)		
1A309	139'	Electrical Penetration Room	0/3(CO_2)		
1A314	139'	South Corridor (Partial)			
1A316	139'	North Corridor			

1.(GGNS-687)

TABLE 3.3.7.9-1
FIRE DETECTION INSTRUMENTATION

ROOM	ELEV	ROOM NAME	MINIMUM INSTRUMENTS OPERABLE* HEAT (X/Y)	FLAME(1) (X/Y)	SMOKE(1) (X/Y)
6. Zone 2-7					11/0
1A420	166'	South Corridor (Partial)			
1A424	166'	Set Down Area (Partial)			
1A428	166'	West Corridor			
1A432	166'	FPC & CU Pump Rm			
1A434	166'	South Passage			
7. Zone 2-8					25/0
1A401	166'	Northeast Corridor			
1A402	166'	Steam Tunnel Roof			
1A403	166'	Southeast Corridor			
1A404	166'	Unassigned Area			
1A405	166'	Containment Vent. Equip Room			
1A406	166'	Containment Exhaust Filter Rm			
1A407	166'	MCC Area	0/2(CO_2)		
1A410	166'	MCC Area	0/2(CO_2)		
1A417	166'	North Corridor (Partial)			
1A424	166'	Set Down Area (Partial)			
8. Zone 2-9					10/0
1A519	185'	Storage Area			
1A524	195'	Platform			
1A527	185'	Load Center Area			
1A529	185'	FPC & CU Rm			
9 Zone 2-13					31/0
1A602	208'	Storage Area			
1A603	208'	Passage			
1A604	208'	Fuel Handling Area			
1A606	245'	HVAC Equip Area			
10. Zone 2-14					17/0
1A114	93'	Fan Coil Area			
1A115	93'	Piping Penetration Rm			
1A116	93'	Piping Penetration Rm			
1A117	93'	Misc Equip Area			
1A118	93'	RHR "C" Pump Room			

1.(GGNS-637)

TABLE 3.3.7.9-1
FIRE DETECTION INSTRUMENTATION

ROOM	ELEV	ROOM NAME	MINIMUM INSTRUMENTS OPERABLE* HEAT (X/Y)	FLAME(1) (X/Y)	SMOKE(1) (X/Y)
1A119	93'	LPCS Pump Room			
1A120	93'	CCW Pump & Heat Ex Rm			
1A122	103'	South Corridor (Partial)			
1A123	103'	North Corridor (Partial)			
11. Zone 2-15					1/0
1A539	185'	Cable Chase			
12. Zone 2-17					16/0
1A117	93'	Misc. Equip Area			
1A101	93'	Passage			
1A109	93'	HPCS Pump Rm			
1A111	93'	Piping Penetration Rm			
1A121	103'	East Corridor			
1A122	103'	South Corridor (Partial)			
1A123	103'	North Corridor (Partial)			
1A114	93'	Fan Coil Area			
13. Zone 2-18					20/0
1A201	119'	East Corridor			
1A211	119'	North Corridor (Partial)			
14. Zone 2-19					13/0
1A314	139'	South Corridor (Partial)			
1A321	139'	MCC Area			
1A322	139'	Centrifugal Chiller Area			
1A323	139'	SGTS Area			
1A324	139'	HVAC Equip Area			
1A326	139'	SGTS Area			
15. Zone 2-20					2/0
1A305	139'	Steam Tunnel			
d. DIESEL GENERATOR BUILDING					
1. Zone 2-10				6/0	3/0
1D301	133'	Corridor	0/3 (Deluge)		
1D306	133'	Div III Diesel Gen Room			
1D401	158'	Div III Diesel Gen Room	0/7 (Deluge)		

1.(GGNS-687)

TABLE 3.3.7.9-1
FIRE DETECTION INSTRUMENTATION

ROOM	ELEV	ROOM NAME	MINIMUM INSTRUMENTS OPERABLE* HEAT (X/Y)	FLAME(1) (X/Y)	SMOKE(1) (X/Y)
2. Zone 2-11				6/0	
1D308	133'	Div II Diesel Gen Room			
1D402	158'	Div II Diesel Gen Room	0/7 (Deluge)		
3. Zone 2-12				6/0	
1D310	133'	Div I Diesel Gen Room			
1D403	158'	Div I Diesel Gen Room	0/7 (Deluge)		
e. STANDBY SERVICE WATER PUMP HOUSE					
1. Zone 2-1					4/0
1M110	133'	SSW Pump Rm A			
1M112	133'	SSW Valve Rm A			
2M110	133'	SSW Pump Rm B			
2M112	133'	SSW Valve Rm B			
f. CHARCOAL FILTER TRAINS					
1. Standby Gas Treatment System Filter Train			1/0 (Allison Thermistor Wire)		
Auxiliary Building El. 139'					
2. Control Room Standby Fresh Air System Filter Train			1/0 (Allison Thermistor Wire)		
Control Building El. 133'					
g. CONTROL BUILDING (PGCC HALON SYSTEMS)					
0C503	166'	Control Room (Unit 1 side)			
		Module/Halon Panel			
		1H13-U700/1H13-P900	0/10		10/0
		1H13-U701/1H13-P901	0/10		15/0
		1H13-U702/1H13-P902	0/9		14/0
		1H13-U703/1H13-P903	0/11		17/0
		1H13-U720/1H13-P920	0/7		13/0
		SH13-U730/1H13-P930	0/11		12/0
		1H13-U738/1H13-P938	0/10		12/0
		SH13-U739/SH13-P939	0/5		14/0

1. (GGNS-687)

TABLE 3.3.7.9-1
FIRE DETECTION INSTRUMENTATION

ROOM	ELEV	ROOM NAME	MINIMUM INSTRUMENTS OPERABLE* HEAT (X/Y)	FLAME[1] (X/Y)	SMOKE[1] (X/Y)
OC504	166'	Unit 1 Instrument Rack Area			
		Module/Halon Panel			
		1H13-U710/1H13-P910	0/8		15/0
		1H13-U711/1H13-P911	0/8		14/0
		1H13-U712/1H13-P912	0/8		9/0
		1H13-U714/1H13-P914	0/~~8~~10		13/0
		1H13-U732/1H13-P932	0/8		14/0
		1H13-U733/1H13-P933	0/8		13/0
		1H13-U734/1H13-P934	0/8		13/0
		1H13-U735/1H13-P935	0/8		11/0
OC703	189'	Unit 1 Instrument Rack Area			
		Module/Halon Panel			
		1H13-U713/1H13-P913	0/9		15/0
		1H13-U715/1H13-P915	0/8		10/0
		1H13-U717/1H13-P917	0/8		15/0
		1H13-U736/1H13-P936	0/8		14/0
		1H13-U737/1H13-P937	0/8		10/0

1. (GGNS-687)

INSTRUMENTATION

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.9 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3.7.9-1 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the fire detection instrument is required to be OPERABLE.

ACTION:

With the number of OPERABLE Function A or Function B fire detection instruments less than the Minimum Instruments OPERABLE requirement of Table 3.3.7.9-1:

a. Within 1 hour, establish a fire watch patrol to inspect the or steam Tunnel zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, ~~or~~ drywell, then inspect the primary containment at least once per 8 hours or monitor the containment, ~~and/or~~ drywell air temperature at least once per hour at the locations listed in Specification 4.6.1.8 and 4.6.2.6. or steam Tunnel

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9/15-18

b. Restore the minimum number of instruments to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the instrument(s) to OPERABLE status.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.9.1 Each of the above required fire detection instruments which are accessible during unit operation shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST. Fire detectors which are not accessible during unit operation shall be demonstrated OPERABLE by the performance of a CHANNEL FUNCTIONAL TEST during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

4.3.7.9.2 The NFPA Standard 72D supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

1.(GGNS-687)

TABLE 3.3.7.9-1

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION				MINIMUM INSTRUMENTS OPERABLE* ZONE(1)	HEAT(2)	FLAME	SMOKE(3)
a. Containment Building							
1. Return Duct Mounted Detectors				NA	NA	NA	3
	ROOM NO.	ELEV.	ROOM NAME				
b. Control Building							
1.	OC202	111'	DIV I SWGR RM	1-4	6	NA	4
2.	OC207	111'	DIV I BATTERY RM	1-4	NA	NA	1
3.	OC208	111'	DIV II REMOTE SHUTDOWN PANEL ROOM	1-27	1	NA	1
4.	OC208A	111'	DIV I REMOTE SHUTDOWN PANEL ROOM	1-27	1	NA	1
5.	OC209	111'	DIV III BATTERY RM	1-5	NA	NA	1
6.	OC210	111'	DIV III SWGR RM	1-5	4	NA	2
7.	OC211	111'	DIV II BATTERY RM	1-6	NA	NA	1
8.	OC215	111'	DIV II SWGR RM	1-6	7	NA	4
9.	OC307	133'	ELECTRICAL CHASE	1-10	NA	NA	1
10.	OC308	133'	ELECTRICAL CHASE	1-10	NA	NA	1
11.	OC302	133'	HVAL EQUIP. ROOM	1-11	NA	NA	13
12.	OC402	148'	CABLE SPREADING RM	1-15	7	NA	10
13.	OC403	148'	COMPUTER ROOM	1-14	13	NA	7
14.	OC407	148'	INSTR. MOTOR GEN ROOM	1-15	2	NA	1
15.	OC503 OC504	166'	CONTROL ROOM	1-18	NA	NA	17
16.	OC702	189'	CABLE SPREADING RM	1-23	12	NA	14
17.	OC703	189'	CONTROL CAB. ROOM	1-24	4	NA	6
18.	OC707	189'	INSTR MOTOR GEN. RM	1-23	NA	NA	1

* The fire detection instruments located within the primary containment are not required to be OPERABLE during the performance of Type A Containment Leakage Rate Tests.

(1) Zones apply only to smoke detectors.

(2) Heat detectors provide warning and activation of automatic extinguishing systems.

(3) Smoke detectors provide early warning capability.

(4) Four thermocouples which monitor ambient air temperature will provide early warning capability.

1. (GGNS-687)

TABLE 3.3.7.9-1 (Continued)

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION				MINIMUM INSTRUMENTS OPERABLE*			
	ROOM NO.	ELEV.	ROOM NAME	ZONE(1)	HEAT(2)	FLAME	SMOKE(3)
c. Auxiliary Building							
1.	1A102	93'	RHR 'A' HT EX RM	2-4	NA	NA	1
2.	1A103	93'	RHR 'A' PUMP RM	2-4	NA	NA	2
3.	1A104	93'	RCIC PUMP RM	2-4	NA	NA	2
4.	1A105	93'	RHR 'B' PUMP RM	2-4	NA	NA	2
5.	1A106	93'	RHR 'B' HT EX RM	2-4	NA	NA	1
6.	1A109	93'	HPCS PUMP RM	2-17	NA	NA	2
7.	1A111	93'	PIPING PENETRATION RM	2-17	NA	NA	1
8.	1A114	93'	FAN COIL AREA	2-14	NA	NA	4
9.	1A115	93'	PIPING PENETRATION RM	2-14	NA	NA	1
10.	1A116	93'	PIPING PENETRATION RM	2-14	NA	NA	1
11.	1A117	93'	MISC. EQUIP AREA	2-14	NA	NA	4
12.	1A118	93'	RHR 'C' PUMP ROOM	2-14	NA	NA	2
13.	1A119	93'	LPCS PUMP ROOM	2-14	NA	NA	2
14.	1A120	93'	CCW PUMP AND HX AREA	2-14	NA	NA	3
15.	1A121	103'	EAST CORRIDOR	2-17	NA	NA	5
16.	1A122	103'	SOUTH CORRIDOR	2-17	NA	NA	3
17.	1A123	103'	NORTH CORRIDOR	2-17	NA	NA	5
18.	1A201	119'	EAST CORRIDOR	2-18	NA	NA	6
19.	1A202	119'	RHR 'A' HX RM	2-4	NA	NA	1
20.	1A203	119'	PIPING PENETRATION RM	2-4	NA	NA	2
21.	1A204	119'	PIPING PENETRATION RM	2-4	NA	NA	2
22.	1A205	119'	PIPING PENETRATION RM	2-4	NA	NA	2
23.	1A206	119'	RHR 'B' HX RM	2-4	NA	NA	1
24.	1A207	119'	ELECT. SWGR ROOM	2-4	3	NA	2
25.	1A208	119'	ELECT. SWGR ROOM	2-4	3	NA	2
26.	1A209	115'	RWCU RECIRC PUMP 'A' RM	2-4	NA	NA	1
27.	1A210	115'	RWCU RECIRC PUMP 'B' RM	2-4	NA	NA	1
28.	1A211	119'	NORTH CORRIDOR	2-18	NA	NA	14
29.	1A215	119'	SOUTH CORRIDOR	2-2	NA	NA	5
30.	1A219	119'	ELECT. SWGR RM	2-3	2	NA	2

1.(GGNS-657)

TABLE 3.3.7.9-1 (Continued)

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION				MINIMUM INSTRUMENTS OPERABLE*			
	ROOM NO.	ELEV.	ROOM NAME	ZONE(1)	HEAT(2)	FLAME	SMOKE(3)
c. Auxiliary Building (Continued)							
31.	1A220	119'	PIPING PENETRATION RM	2-3	NA	NA	1
32.	1A221	119'	ELECT. SWGR RM	2-3	2	NA	2
33.	1A222	119'	WEST CORRIDOR	2-2	NA	NA	18
34.	1A301	139'	NORTHEAST CORRIDOR	2-6	NA	NA	2
35.	1A302	139'	SOUTHEAST CORRIDOR	2-6	NA	NA	1
36.	1A303	139'	RHR 'A' HX RM	2-6	NA	NA	1
37.	1A304	139'	PIPING PENETRATION RM	2-6	NA	NA	1
38.	1A305	139'	STEAM TUNNEL	2-20	NA(4)	NA	~~2~~ NA
39.	1A306	139'	PIPING PENETRATION RM	2-6	NA	NA	1
40.	1A307	139'	RHR 'B' HX RM	2-6	NA	NA	1
41.	1A308	139'	ELECT. PENETRATION RM	2-6	3	NA	2
42.	1A309	139'	ELECT. PENETRATION RM	2-6 2-6	3	NA	2 3
43.	1A314	139'	SOUTH CORRIDOR	2-19	NA	NA	3
44.	1A316	139'	NORTH CORRIDOR	2-6	NA	NA	13
45.	1A318	139'	ELECT. PENETRATION RM	2-5	2	NA	2
46.	1A319	139'	RPV INSTR. TEST RM	2-5	NA	NA	1
47.	1A320	139'	ELECT. PENETRATION RM	2-5	2	NA	2
48.	1A321	139'	MCC AREA	2-19	NA	NA	3
49.	1A322	139'	CENTRIFUGAL CHILLER AREA	2-19	NA	NA	4
50.	1A323	139'	SGTS AREA	2-19	NA	NA	1
51.	1A324	139'	HVAC EQUIP AREA	2-19	NA	NA	1
52.	1A326	139'	SGTS AREA	2-19	NA	NA	1
53.	1A401	166'	NORTHEAST CORRIDOR	2-8	NA	NA	2
54.	1A402	166'	STEAM TUNNEL ROOF	2-8	NA	NA	1
55.	1A403	166'	SOUTHEAST CORRIDOR	2-8	NA	NA	2
56.	1A404	166'	UNASSIGNED AREA	2-8	NA	NA	1
57.	1A405	166'	CNTMT VENT. EQUIP RM	2-8	NA	NA	1
58.	1A406	166'	CNTMT EXHAUST FILTER AND VENT ROOM	2-8	NA	NA	1

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1.(GGNS-687)

TABLE 3.3.7.9-1 (Continued)

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION				MINIMUM INSTRUMENTS OPERABLE*			
	ROOM NO.	ELEV.	ROOM NAME	ZONE(1)	HEAT(2)	FLAME	SMOKE(3)
c. Auxiliary Building (Continued)							
59.	1A407	166'	MCC AREA	2-8	2	NA	1
60.	1A410	166'	MCC AREA	2-8	2	NA	1
61.	1A417	166'	NORTH CORRIDOR	2-8	NA	NA	14
62.	1A420	166'	SOUTH CORRIDOR	2-7	NA	NA	4
63.	1A424	166'	SET DOWN AREA	2-7	NA	NA	2
64.	1A428	166'	WEST CORRIDOR	2-7	NA	NA	4
65.	1A432	166'	FPC AND CU PUMP RM	2-7	NA	NA	1
66.	1A434	166'	PASSAGE	2-7	NA	NA	1
-67.	1A519	185'	STORAGE AREA	2-9	NA	NA	4
-68.	1A527	185'	LOAD CENTER AREA	2-9	NA	NA	5
69.	1A539	185'	CABLE CHASE	2-15	NA	NA	1
70.	1A602	208'10"	STORAGE AREA	2-13	NA	NA	6
71.	1A603	208'10"	PASSAGE	2-13	NA	NA	3
72.	1A604	208'10"	FUEL HANDLING AREA	2-13	NA	NA	13
73.	1A606	245'	HVAC EQUIP AREA	2-13	NA	NA	9
d. Diesel Generator Building							
1.	Unit 1 El. 158'-0" HPCS Generator			6-9A	7	6	NA
2.	Unit 1 El. 158'-0" Bus B Generator			6-9B	7	6	NA
3.	Unit 1 El. 158'-0" Bus A Generator			6-9C	7	6	NA
e. Standby Service Water Pump House							
1.	Pump House A			2-1	NA	NA	1
2.	Valve Room A			2-1	NA	NA	1
3.	Pump House B			2-1	NA	NA	1
4.	Valve Room B			2-1	NA	NA	1
f. Charcoal Filter Trains							
1.	Standby Gas Treatment System Filter Train Auxiliary Building El. 139'-0"			NA	1 (Allison Thermistor Wire)	NA	NA
2.	Control Room Standby Fresh Air System Filter Train, Control Building El. 133'-0"			NA	1 (Allison Thermistor Wire)	NA	NA

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JUN 22 1983
NUCLEAR SERVICE
M.P.&L.CO.

Bechtel Power Corporation
Engineers — Constructors
15740 Shady Grove Road
Gaithersburg, Maryland 20877-1454
301-258-3000

June 15, 1983

102

Mr. J. F. Pinto, Manager
Nuclear Plant Engineering
Grand Gulf Nuclear Station
Mississippi Power & Light Company
Post Office Box 756
Port Gibson, Mississippi 39150

Dear Mr. Pinto:

Nuclear QA Is Applicable
Middle South Energy, Inc.
Grand Gulf Nuclear Station
Bechtel Job No. 15026
File: 0262/L-860.0/L-952.0
Safety Evaluations for Amendment 56
FSAR Change Notices
MPB-83/0328

As requested by your Mr. R. G. Bearden, we have performed safety evaluations (enclosed) on the Bechtel initiated FSAR Change Notices scheduled for Amendment 56, using the guidelines of 10 CFR 50.59. These evaluations supercede the evaluations attached to MPB-83/0312, dated May 27, 1983.

If you have any questions, please contact us.

Very truly yours,

R. S. Trickovic

R. S. Trickovic
Project Engineer

SWK/mm
Applicable Systems: None
Enclosures: Safety Evaluations for Amendment 56
FSAR Change Notices
1727, 1728, 1737, 1740, 1744, 1748, 1756,
1758, 1767, 1784, 1788, 1796, 1806, 1808,
1815, 1845, and 1883

cc: J. P. McGaughy, Jr., w/1
T. H. Cloninger, w/1
T. E. Reaves, w/1
C. K. McCoy, w/1
L. F. Dale, 2w/2
J. D. Richardson, w/1
W. D. Archdeacon, w/o
J. F. Hudson, w/1

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Safety Evaluations for Amendment 56 FSAR
Change Notices 1727, 1728, 1737, 1740, 1744, 1748, 1756,
1758, 1767, 1784, 1783, 1796, 1806, 1808, 1815, and 1883.

Grand Gulf Nuclear Station - Unit 1
Safety Evaluation for FSAR Amendment 56
(Change Notice 1727)

FSAR Change Notice 1727 revises Table 9A-4, Item II.E, "Hydrostatic Hose Tests," to be consistent with the testing requirements of 10 CFR 50, Appendix R, and Technical Specifications 4.7.6.5 and 4.7.6.6. The purpose of the change notice is to update the FSAR to be consistent with the Technical Specifications and plant surveillance procedures.

The change described above does not require a change in the Technical Specifications or constitute an unreviewed safety question as defined in 10 CFR 50.59. Therefore this change will not adversely affect the health and safety of the public. On this basis it is considered acceptable.

Originator [illegible signature]

Date June 15, 1983

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Grand Gulf Nuclear Station - Unit 1
Safety Evaluation for FSAR Amendment 56
(Change Notice 1728)

FSAR Change Notice 1728 changes the phrase "Fire Protection Plan" in FSAR Table 9A-1 to "Fire Protection Program." This is an editorial change to make Table 9A-1 consistent with revised FSAR Appendix 9B and MP&L letter AECM-82/143, which transmitted Appendix 9B to the NRC.

The change described above does not require a change in the Technical Specifications or constitute an unreviewed safety question as defined in 10 CFR 50.59. Therefore, this change will not adversely affect the health and safety of the public. On this basis it is considered acceptable.

Originator H S Montgomery

Date June 15, 1983

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Grand Gulf Nuclear Station - Unit 1
Safety Evaluation for FSAR Amendment 56
(Change Notice 1737)

FSAR Change Notice 1737 revises Table 5.2-5 to provide an updated listing of Reactor Coolant Pressure Boundary (RCPB) valves. Additions/deletions of valves generally resulted from design changes from the initial FSAR issuance and are based on input provided by General Electric. These valve description changes are consistent with information provided in other sections of the FSAR. Valves were procured in accordance with the applicable quality group requirements. Valve closure time data is consistent with the latest General Electric analyses and design requirements.

The change described above does not require a change in the Technical Specifications or constitute an unreviewed safety question as defined in 10 CFR 50.59. Therefore, this change will not adversely affect the health and safety of the public. On this basis it is considered acceptable.

Originator RCG DC Gloie

Date 6-15-83

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Grand Gulf Nuclear Station - Unit 1
Safety Evaluation for FSAR Amendment 56
(Change Notice 1740)

FSAR Change Notice 1740 updates Figure 9.5-1, "Fire Protection System, Unit 1," to reflect the deletion of a flow restricting orifice on the fill line to the fire water storage tanks. The purpose of the orifice was to limit the flow of plant service water (PSW) to ensure that sufficient PSW would be available to the turbine building cooling water system (TBCW). The orifice was deleted when TBCW operation demonstrated that the orifice was unnecessary. This modification does not adversely affect operation of the TBCW.

The change described above does not require a change in the Technical Specifications or constitute an unreviewed safety question as defined in 10 CFR 50.59. Therefore, this change will not adversely affect the health and safety of the public. On this basis it is considered acceptable.

Originator: [signature]

Date 6-15-83

0321015111

Grand Gulf Nuclear Station - Unit 1
Safety Evaluation for FSAR Amendment 56
(Change Notice 1744)

FSAR Change Notice 1744 updates various equipment location drawings in Appendix 9A for the auxiliary, containment, diesel generator, standby service water, control, turbine, and water treatment buildings. The drawings presently in the FSAR are out of date. These changes are necessary to clarify the actual equipment locations as part of the updated fire hazards analysis.

The change described above does not require a change in the Technical Specifications or constitute an unreviewed safety question as defined in 10 CFR 50.59. Therefore, this change will not adversely affect the health and safety of the public. On this basis it is considered acceptable.

Originator f RC Sloric

Date 5-15-83

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Grand Gulf Nuclear Station - Unit 1
Safety Evaluation for FSAR Amendment 56
(Change Notice 1748)

FSAR Change Notice 1748 updates Figure 9A-47a to show the as-built location of equipment in the auxiliary and containment buildings at Els. 161'-10" and 166'-0". The drawing presently in the FSAR is out of date. This change is necessary to clarify the actual equipment locations as part of the updated fire hazards analysis.

The change described above does not require a change in the Technical Specifications or constitute an unreviewed safety question as defined in 10 CFR 50.59. Therefore, this change will not adversely affect the health and safety of the public. On this basis it is considered acceptable.

Originator: RC Eloric

Date 6-15-83

Grand Gulf Nuclear Station - Unit 1
Safety Evaluation for FSAR Amendment 56
(Change Notice 1756)

FSAR Change Notice 1756 corrects the description of smoke detection and isolation of the main control room. The FSAR incorrectly states that isolation of the control room and initiation of the emergency filtration unit are accomplished by a high smoke concentration in the air intake duct, and that a fire in the control room air conditioning equipment room is an initiating event for isolation of the control room. This change notice corrects the FSAR to state that upon sensing smoke in the mixing plenum of outside air and return air, the control room is isolated and the air conditioning unit is shut down. The operator then either places the control room in a recirculation mode if the source of smoke is external to the control room, or he starts the purge fans if the source of smoke is internal to the control room. This is consistent with Technical Specification 4.7.2.d, in that a smoke signal is not required to start the control room emergency filtration system. This is also consistent with statements made by the NRC in SER Section 6.4. The revised description reflects the operational configuration upon which the system was licensed, and no post-operating license modification has been made. The results of accidents evaluated in the FSAR are not affected by this change.

The change described above does not require a change in the Technical Specifications or constitute an unreviewed safety question as defined in 10 CFR 50.59. Therefore, this change will not adversely affect the health and safety of the public. On this basis it is considered acceptable.

Originator [signature]

Date 6 15 83

0321015514

Grand Gulf Nuclear Station - Unit 1
Safety Evaluation for FSAR Amendment 56
(Change Notice 1758)

FSAR Change Notice 1758 clarifies the methods used to enclose cables in the control room suspended ceiling area to include instrumentation cables run in all-metal solid-bottom trays which do not have metal covers for the top of the tray. These cable trays are entirely wrapped by a fire-retardant material.

Because an exposure fire is not postulated in the suspended ceiling area, the damage potential is limited to faults in the cable. Therefore, the wrapping of the cable trays with a fire-retardant material is acceptable under Regulatory Guide 1.75 and IEEE Standard 384-1974.

The change described above does not require a change in the Technical Specifications or constitute an unreviewed safety question as defined in 10 CFR 50.59. Therefore, this change will not adversely affect the health and safety of the public. On this basis it is considered acceptable.

Originator [signature]

Date 6/15/83

Grand Gulf Nuclear Station - Unit 1
Safety Evaluation for FSAR Amendment 56
(Change Notice 1767)

FSAR Change Notice 1767 updates the fire and smoke detection system figures. The changes reflect design changes which increased area coverage by ionization smoke detectors and added flame detectors in the diesel generator building. The capability to detect a fire in safety-related equipment has been enhanced. Hence no unreviewed safety question, as defined in 10 CFR 50.59, is involved.

However, a Technical Specification change is needed. Technical Specification Table 3.3.7.9-1 should be revised to include the smoke detectors in Areas 1D301 and 1A529.

Originator RC Bloric

Date 6-15-83

0 3 2 1 0 1 5 1 6

Grand Gulf Nuclear Station - Unit 1
Safety Evaluation for FSAR Amendment 56
(Change Notice 1784)

FSAR Change Notice 1784 changes the air flow rate of the turbine building smoke exhaust system from its design minimum of 27,000 cfm to its tested minimum of 19,000 cfm. As such, this change to the FSAR reflects the as-built condition of the smoke exhaust system.

Although the design minimum flow rate could not be achieved, the effect of the lower flow rate on the mitigation of the effects of smoke on equipment in the turbine building is negligible. Based on the negligible effects of the reduction in flow rate, this system still meets the requirements of Appendix A to Branch Technical Position APCSB 9.5-1.

The change described above does not require a change in the Technical Specifications or constitute an unreviewed safety question as defined in 10 CFR 50.59. Therefore, this change will not adversely affect the health and safety of the public. On this basis it is considered acceptable.

Originator S.K. Bagai

Date 6-15-83

Grand Gulf Nuclear Station - Unit 1
Safety Evaluation for FSAR Amendment 56
(Change Notice 1783)

FSAR Change Notice 1783 provides a clarification which specifies that the vacuum breakers on the return lines to the pools are stop check valves. This is an editorial change, involving no hardware modifications.

The change described above does not require a change in the Technical Specifications or constitute an unreviewed safety question as defined in 10 CFR 50.59. Therefore, this change will not adversely affect the health and safety of the public. On this basis it is considered acceptable.

Originator ____________

Date 6-15-83

Grand Gulf Nuclear Station - Unit 1
Safety Evaluation for FSAR Amendment 56
(Change Notice 1796)

FSAR Change Notice 1796 adds new Figure 9A-55 to show the exact boundaries of the exposure fire areas in the auxiliary building corridors at Els. 93'-0" and 103'-0". This is a clarification of the fire hazards analysis. The analysis itself has not been changed.

The change described above does not require a change in the Technical Specifications or constitute an unreviewed safety question as defined in 10 CFR 50.59. Therefore, this change will not adversely affect the health and safety of the public. On this basis it is considered acceptable.

Originator 7/J.B. Montgomery

Date June 15, 1983

0 2 1 0 1 5 1 9

Grand Gulf Nuclear Station - Unit 1
Safety Evaluation for FSAR Amendment 56
(Change Notice 1806)

FSAR Change Notice 1806 is an editorial correction to Figure 1.2-1 (drawing 01-C001). The FSAR currently has an incorrect drawing as Figure 1.2-1. This change notice replaces it with the correct drawing. This editorial change does not involve any physical modifications to the plant.

The change described above does not require a change in the Technical Specifications or constitute an unreviewed safety question as defined in 10 CFR 50.59. Therefore, this change will not adversely affect the health and safety of the public. On this basis it is considered acceptable.

Originator f RC Sloric

Date 5-15-83

0 3 2 1 0 . 1 5 2 0

Grand Gulf Nuclear Station - Unit 1
Safety Evaluation for FSAR Amendment 56
(Change Notice 1008)

FSAR Change Notice 1008 updates Figure 9.4-10 (Drawing M-1103A) to add a smoke detector in the auxiliary building ventilation system. This modification is an enhancement to the smoke detection capability in this area.

The change described above does not require a change in the Technical Specifications or constitute an unreviewed safety question as defined in 10 CFR 50.59. Therefore, this change will not adversely affect the health and safety of the public. On this basis it is considered acceptable.

Originator RC Illoric

Date 6-15-83

0321011521

Grand Gulf Nuclear Station - Unit 1
Safety Evaluation for FSAR Amendment 56
(Change Notice 1815)

FSAR Change Notice 1815 incorporates editorial changes in the fire protection plan figures of Appendix 9A. These editorial changes clarify notes describing the fire rating of floor separations. Since no plant modifications are involved, the analysis of Appendix 9A is not affected.

The change described above does not require a change in the Technical Specifications or constitute an unreviewed safety question as defined in 10 CFR 50.59. Therefore, this change will not adversely affect the health and safety of the public. On this basis it is considered acceptable.

Originator [signature]

Date 6-15-83

Grand Gulf Nuclear Station - Unit 1
Safety Evaluation for FSAR Amendment 56
(Change Notice 1845)

FSAR Change Notice 1845 adds elastomeric (antisweat) insulation to the list of noncombustible materials defined in Appendix 9A. This material has a flame spread-fuel contribution rating of 25-30, as measured by ASTM-E-84, which is not significantly greater than the present FSAR criterion of 25-25. Due to the conservative nature of the fire hazards analysis, inclusion of this insulation as a combustible material would not result in significant increases in the calculated heat loads in the plant.

The change described above does not require a change in the Technical Specifications or constitute an unreviewed safety question as defined in 10 CFR 50.59. Therefore, this change will not adversely affect the health and safety of the public. On this basis it is considered acceptable.

Originator *. ec Bosic
Date 6-13-83

0 3 2 1 0 1 5 2 3

Grand Gulf Nuclear Station - Unit 1
Safety Evaluation for FSAR Amendment 56
(Change Notice 1883)

FSAR Change Notice 1883 is an update and clarification of FSAR subsection 9.5.1 and Appendix 9A. In general, the FSAR changes can be grouped into the following categories:

1. Subsection 9.5.1

   a. Subsection 9.5.1 was revised to describe the fire protection systems and equipment in their as-built condition.

   b. Subsection 9.5.1.2.2.7 was revised to clarify that smoke detection is not provided for the minimal amount of safety-related equipment in non-Category I buildings since this equipment is designed to fail safe in the event of a fire.

2. Appendix 9A:

   a. Section 7.2 has been revised to delete the listings of the redundant safe shutdown related cables. The subsections from which these listings were deleted now include only an indication whether redundant safe shutdown related cables are routed through that area, or an adjacent exposure fire area, and the separation that exists between redundant safe shutdown related cables. The listing, by room, of redundant safe shutdown related cables will be maintained at the site on drawing 9645-E-0731. This was done in order to decrease the number of future FSAR revisions that would be necessary as a result of design changes in the facility.

   b. Section 7.2 and Table 9A-2 have been revised to clarify that manual hose streams and portable fire extinguishers are not necessarily in a given room, but are accessible to that room. FSAR Figure 9.5-4 provides the actual locations for hose stations and portable fire extinguishers. All references to "portable water and dry chemical fire extinguishers" have been changed to "portable fire extinguishers" since there are other types, such as halon, at the plant.

   c. Section 7.2 and Table 9A-2 have been revised to clearly indicate the exact room location of the ionization smoke detectors. For the few instances where credit is taken for area coverage by the ionization smoke detector(s) located in an adjacent area, such credit has been identified.

   d. Section 7.2 and Table 9A-2 have been revised to indicate that area smoke detector coverage of a room could be provided by one or more detectors, by referring to them as "detector(s)." FSAR figures 9A-10 through 9A-33 provide the actual locations of the smoke detectors. Additionally, as a result of a MP&L review, certain areas were identified which did not meet the design requirements for ionization smoke detector coverage. These areas are rooms OC201, OC216, HVAC Chase (El. 177', area bounded by column lines 15.[illegible] and J8-K), and HVAC chase (El. 189', area bounded by column lines 18.9-20 and J.8-K), all of which are in the control building.

IL0032

Smoke detection is currently being provided for these areas. The FSAR should be revised to show these detectors in a later amendment.

e. As discussed in the June 10, 1982, meeting between MP&L and NRC Staff (Chemical Engineering - J. Stang) on fire protection, Appendix 9A Sections 2.0 and 7.2.1 have been revised to include a detailed discussion of the corridors at Els. 93' and 103' of the auxiliary building which are separated by metal grate floors. As detailed in these sections, extensive consideration has been given to the effects of, and protection from, a postulated exposure fire in any one of these areas.

FSAR Figure 9A-55 is added via Change Notice 1796 to clarify the exposure fire areas and their relation to each other.

f. Table 9A-2 has been revised to incorporate the latest values for electrical cable heat loads in the auxiliary, control, diesel generator buildings, and the standby service water pump house. These revised values represent a more realistic, up-to-date tabulation of the maximum possible cable heat load for a given area. Original values listed in the table were based on preliminary design estimates of expected cable loadings. The conservative basis for these revised values is discussed in Appendix 9A, Section 4.0, the notes and comments to Table 9A-2, and Table 9A-3.

g. Table 9A-2 has been revised, to clearly indicate the fire rating of the walls that enclose a given area. Fire-rated barriers are shown on Figures 9A-3 through 9A-9.

h. Appendix 9A, Section 7.2.2 and Table 9A-2 have been revised, to indicate which fire protection systems dedicated solely to Unit 2 are not required for Unit 1 operation and will not be provided prior to Unit 1 operation.

i. Appendix 9A, Sections 3.0, 6.0, 7.0, Table 9A-4, and the response to NRC Question 013.24 have been revised to clarify that smoke detection is not provided for the minimal amount of safety-related equipment in non-Category I buildings since this equipment is designed to fail safe in the event of a fire.

All of the above changes to Appendix 9A are either a clarification or an update to the existing document. Therefore, this change does not constitute an unreviewed safety question as defined in 10 CFR 50.59 and will not adversely affect the health and safety of the public.

Implementation of this change notice requires a revision to the plant Technical Specifications. Table 3.3.7.9-1 should be revised to include the smoke detector(s) in the areas listed below.

Areas For Which A
Technical Specification Change
Is Required

0C109	0C410	0C609
0C115	0C412	0C610
0C116	0C506	0C612
0C303	0C514	0C616
0C304	0C516	0C706
0C305	0C517	0C709
0C306	0C518	0C712
0C401	0C601	1D301
0C408	0C603	1A101
0C409	0C608	1A529

In addition, rooms 1A102 and 1A106 should be written as 1A102/1A128 and 1A106/1A129 respectively.

Originator PC Blair

Date 4-15-83

8305260120

PDR

TECHNICAL SPECIFICATION PROBLEM SHEET

Item Number: 112 Priority: Z IC

______ / ______
Identified By Date

______
Responsible Supervisor

Tech Spec Reference: 3/4.3.2 (TS 3.3.1.c, TS 3.3.2.b)

Problem Title: Isolation Instrumentation Channel/Trip

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other):

The proposed change brings present action "b" of T.S. 3.3.2; its associated "*" notation, and the "*" notation of T.S. 3.3.1 in accordance with STS. The STS wording does not require an inoperable channel to be placed in a tripped condition where this would cause the Trip Function to occur.

2. Safety Significance: This change helps to clarify the GGNS Tech Specs.

3. Anticipated Resolution: Incorporate change. Tech Spec change has been submitted
could live with spec.

4. NRC Response to Item (NRR/IE): ______

NRC Notified: ______ ______ / ______
Individual Notified Date Time

5. Disposition: ______

Items Closed: (How) ______

______ / ______
Date Time

cc: J. E. Cross
R. F. Rogers

18. (GGNS - X40)

SUBJECT: Technical Specifications 3.3.1 and 3.3.2, pages 3/4 3-1 and 3/4 3-9.

DISCUSSION: Action "b" of Technical Specification 3.3.2 requires that with the number of OPERABLE channels less than required by the MINIMUM OPERABLE channels per Trip System requirement for one trip system, place that trip system in the tripped condition* within one hour. The proposed change is to adopt the Standard Technical Specification wording that allows placing the inoperable channel(s) and/or that trip system in the tripped condition* within one hour.

Present "*" notation at the bottom of pages 3/4 3-1 and 3/4 3-9 requires that with a design providing only one channel per trip system, an inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. The proposed change is to change the present "*" notation to agree with the applicable Standard Technical Specification "*" notation.

JUSTIFICATION: The Standard Technical Specification wording for action "b" of Technical Specification 3.3.2, its associated "*" notation, and the "*" notation of Technical Specification 3.3.1 does not require an inoperable channel to be placed in the tripped condition where this would cause the Trip Function to occur. However, restoration of the channel is required within 2 hours or ACTIONS required by Table 3.3.2-1 for that Trip Function shall be taken. The Standard Technical Specification clarifies present action "b" of Technical Specification 3.3.2 and "*" notation which addresses a design providing only one channel per trip system.

SIGNIFICANT HAZARDS CONSIDERATION:

The proposed change brings present action "b" of Technical Specification 3.3.2, its associated "*" notation, and the "*" notation of Technical Specification 3.3.1 in conformance with the Standard Technical Specifications. This change does not involve the reduction of safety margins and no significant increase in the probability or consequences of an accident previously evaluated is involved nor is the possibility of a new or different kind of accident from any accident previously evaluated created. Thus the proposed change to the Technical Specification does not involve any significant hazards consideration.

18. (GGNS - X40)

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel and/or that trip system in the tripped condition* within one hour. The provisions of Specification 3.0.4 are not applicable.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within one hour and take the ACTION required by Table 3.3.1-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3 1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip functional unit shown in Table 3.3.1-2 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip system.

---

*~~With a design providing only one channel per trip system,~~ A an inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.1-1 for that Trip Function shall be taken.

** [illegible] the [illegible] inoperable [illegible] in the tripped condition, except when this would [illegible] the Trip Function to occur.

IS, (GE/S-XYZ)

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The isolation actuation instrumentation channels shown in Table 3.3.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.2-2 and with ISOLATION SYSTEM RESPONSE TIME as shown in Table 3.3.2-3.

APPLICABILITY: As shown in Table 3.3.2-1.

ACTION:

a. With an isolation actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel(s) and/or that trip system in the tripped condition* within one hour. The provisions of Specification 3.0.4 are not applicable.

c. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within one hour and take the ACTION required by Table 3.3.2-1.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each isolation actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.2.1-1.

4.3.2.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.2.3 The ISOLATION SYSTEM RESPONSE TIME of each isolation trip function shown in Table 3.3.2-3 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months, where N is the total number of redundant channels in a specific isolation trip system.

---

*~~With a design providing only one channel per trip system,~~ An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.2-1 for that Trip Function shall be taken.

**... the trip system with more inoperable channels in the tripped condition, except ... this would cause the Trip F...

TECHNICAL SPECIFICATION PROBLEM SHEET

Item Number: 116 Priority: 2/A

______ / ______ ______
Identified By Date Responsible Supervisor

Tech Spec Reference: 3/4.3.3.1 (Tbl. 4.3.3.1-1.A.2.e)

Problem Title: LPCS Pump Discharge High

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other):
A reference to note (a) is to be addedd to the CHANNEL CALIBRATION entry for item A.2.e.

2. Safety Significance: Proposed change adds a requirement to calibrate the above trip unit monthly. Thisrevision makes this trip unit consistent with those of other Div 1, 2, and 3 trip systems.

3. Anticipated Resolution: Implement change.
Programmaticly control surveillance Frequency.

4. NRC Response to Item (NRR/IE): ______

NRC Notified: ______ ______ / ______
Individual Notified Date Time

5. Disposition: ______

Items Closed: (How) ______

______ / ______
Date Time

cc: J. E. Cross
R. F. Rogers

28. (GGNS - 788)

SUBJECT: Technical Specification Table 4.3.3.1-1, page 3/4 3-31.

DISCUSSION: A reference to note (a) is to be added to the CHANNEL CALIBRATION entry for item A.2.e, LPCS Pump Discharge Pressure - High. This reference was inadvertently omitted from the Technical Specification.

JUSTIFICATION: The proposed change adds a requirement to calibrate the above trip unit monthly. This revision will make the calibration requirements from this trip unit consistent with those of other Division 1, 2, and 3 trip systems.

SIGNIFICANT HAZARDS CONSIDERATION:

The proposed change is an additional surveillance requirement. It does not significantly increase the probability or consequences of an accident previously evaluated nor does it create the possibility of a new or different accident from any previously evaluated. It does not constitute a significant hazards consideration.

TABLE 4.3.3.1-1

E[illegible]ERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

[illegible]	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
DIV[illegible] [illegible] SYSTEM				
1. [illegible] (LPCI MODE) AND LPCS SYSTEM				
[illegible]ctor Vessel Water Level - [illegible] Low [illegible], Level 1	S	M	R(a)	1, 2, 3, 4*, 5*
[illegible]ll [illegible]ssure - High	S	M	R(a)	1, 2, 3
[illegible]CI Pump A Start Time Delay Relay	NA	M	Q	1, 2, 3, 4*, 5*
[illegible]nual Initiation	NA	R(b)(c)	Q(d)	1, 2, 3, 4*, 5*
2. [illegible] DEPRESSURIZATION SYSTEM [illegible] "A"				
[illegible]ctor Vessel Water Level - Low Low Low, Level 1	S	M	R(a)	1, 2, 3
[illegible]ll Pressure-High	S	M	R(a)	1, 2, 3
[illegible] Timer	NA	M	Q	1, 2, 3
[illegible]ctor Vessel Water Level - Low, Level 3	S	M	R(a)	1, 2, 3
[illegible] Pump Discharge Pressure-High	S	M	R(a)	1, 2, 3
[illegible]CI Pump A Discharge Pressure-High	S	M	R(a)	1, 2, 3
[illegible]ual Initiation	NA	R(b)	NA	1, 2, 3
DIVI[illegible] SYSTEM				
1. [illegible] C (LPCI MODE)				
[illegible]ctor Vessel Water Level - Low Low Low, Level 1	S	M	R(a)	1, 2, 3, 4*, 5*
[illegible]ll Pressure - High	S	M	R(a)	1, 2, 3
[illegible]CI Pump B Start Time Delay Relay	NA	M	Q	1, 2, 3, 4*, 5*
[illegible]nual Initiation	NA	R(b)(c)	Q(d)	1, 2, 3, 4*, 5*

(GGNS-108)

NO Changes On This Page - Information Only

TABLE 4.3.3.1-1 (Continued)

28. (GGNS-758)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

NOTATION

- \# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 135 psig.
- * Applicable when the system is required to be OPERABLE per Specification 3.5.2 or 3.5.3.
- ** Required when ESF equipment is required to be OPERABLE.
- (a) Calibrate trip unit at least once per 31 days.
- (b) Manual initiation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual initiation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days as a part of circuitry required to be tested for automatic system actuation.
- (c) Manual initiation test shall include verification of the OPERABILITY of the LPCS and LPCI injection valve interlocks. (See Note 1)
- (d) This calibration shall consist of the CHANNEL CALIBRATION of the LPCS and LPCI injection valve interlocks with the interlock setpoint verified to be $\leq$ 150 psig. (See Note 1)
- (e) Functional Testing of Time Delay Not Required

Note 1: Until restart after the first refueling outage, the requirements of (c) and (d) above do not apply.

"TECH SPEC PRIORITY"

Punchlist Item # See ATTAched

Tech Spec See ATTAched

Priority See ATTAched

TO: Manager of Nuclear Plant Engineering

FROM: Chairman, Prioritization and Disposition Chairman

SUBJECT: Technical Specifications Punchlist Item # See ATTached

PDTS:84/ 0014

DATE: 3/10/84

The subject Tech Spec item has been determined by the Disposition Committee to require Engineering support.

DETAILS: This letter identifies requested response dates for the following Tech Spec problems:

# 199	Letter No. PDTS 84/0001	# 015	Letter No. PDTS 84/0007
# 180	Letter NO. PDTS 84/0002	# 198	Letter No. PDTS 84/0008
# 033	Letter NO. PDTS 84/0003	# 202	Letter NO. PDTS 84/0009
# 054	Letter NO. PDTS 84/0004	# 213	Letter NO. PDTS 84/0010
# 001	Letter NO. PDTS 84/0005	# 219	Letter NO. PDTS 84/0011
# 016	Letter NO. PDTS 84/0006	# 118	Letter NO. PDTS 84/0013

It is requested that the responses to the above items be Completed by 3/13/84

Please contact Jerry RoberTs at Extension 2695 for further information.

Please refer to the Tech Spec Punchlist item number in your response. Forward your response to George ZINKE.

J. C. Roberts
Chairman

LLJ/JCR:swb

cc: Mr. C. L. Tyrone
Mr. J. E. Cross
Mr. D. Stonestreet
Mr. A. S. McCurdy
Mr. S. Hutchins
Mr. J. Hendry
File (Tech Spec Records)

A4/61swb1

Tech Spec Problem No.	Tech Spec	Priority
199	Table 3.3.6-1.5	1B
180	4.8.4.3	1D
033	Table 3.3.8-2	1B
054	3/4.3.8	1B
001	3/4.5.1	1B
016	3/4.3.8	1B
015	3/4.3.2	1D
198	3/4.3.7	1B
202	3/4.3.7	1B
213	3/4.3.3	1B
219	Figure 3.4.6.1-1	1B
168	3.6.3.1	1B

TECHNICAL SPECIFICATION PROBLEM SHEET

Item Number: 139 Priority: 1~~X~~B

_______________ / _______
Identified by Date

____
Responsible Supervisor

Tech Spec Reference: Table 3.7.4-2

Problem Title: Corrections to Snubber Table 3.7.4-2

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other):
Change to Tech Spec Table 3.7.4-2 Mechanical Snubber Table requries the following corrections: 1) Addition of snubbers that were inadvertently left off the table. 2) Deletion of snubbers that have been voided, superceded or incorrect. 3) Correct typos 4) Add Non-Q Mech. snubbers included in stress analysis of Q-piping.

2. Safety Significance:
The Tech Spec Table is non-conservative, some snubbers included in design analysis are missing.

3. Anticipated Resolution:
Reference Item No. 006-Existing snubber Tech Spec to be revised, will not have a list of snubbers.
Live with spec note: STS no longer requires listing of snubbers to be in Tech Specs.

4. NRC Response to Item (NRR/IE): _______________

NRC Notified: _______________ _____ / _____
Individual Notified Date Time

5. Disposition: _______________

Items Closed: (How) _______________

_____ / _____
Date Time

cc: J. E. Cross
R. F. Rogers

31. (GGNS - 521)

SUBJECT: Technical Specification Table 3.7.4-2, pages 3/4 7-16 through 3/4 7-25.

DISCUSSION: Technical Specification Table 3.7.4-2 contains a listing of Safety Related Mechanical Snubbers. The purpose of this change is to make necessary corrections to the table that include the following:

1. Additions of snubbers that were inadvertently left off the table.

2. Deletion of snubbers that have been voided, superceded, or incorrectly placed in the table.

3. Correct typos

4. Add Non-Q Mechanical Snubbers which were included in the stress analysis of Q-piping.

As part of this change, snubbers with the same number and location are grouped together to prevent duplicate listings. The snubbers that are grouped together are indicated by number in parentheses.

The changes to Table 3.7.4-2 are discussed in detail in the justification section.

JUSTIFICATION: The changes to Table 3.7.4-2 are administrative in nature and do not involve changes to the plant. These changes are Table corrections due to typo's, inadvertent omissions from the table, and deletions due to superceded, voided or incorrect listings. The inclusion of Non-Q Snubbers is provided since they are on piping which is included in stress analysis calculations that affect Q piping. The changes to Table 3.7.4-2 are justified below:

1. Additions to the table due to inadvertent omissions include:

   a. Q1B33G122R01, Area 11, Elevation 108

   b. Q1B33G355R01, Area 11, Elevation 102 - One of these snubbers was submitted in AECM-83/0314. There should be two at elevation 102.

   c. Q1B21G023R020, Area 11, Elevation 120

   d. Q1B21G163R01, Area 11, Elevation 113

   e. Q1B21G163R02, Area 11, Elevation 113

   [illegible], Elevation [illegible]

g. Q1B21G369R01 (2), Area 11, Elevation 148

h. Q1B21G382R02 (2), Area 11, Elevation 155 - One additional snubber is at this location.

i. Q1B21G384R01, Area 11, Elevation 152

j. Q1E12G025C01 (2), Area 8, Elevation 95 - One additional snubber is at this location.

k. Q1E31G122R01 (2), Area 11, Elevation 149 - One additional snubber is at this location.

l. Q1E51G003R12 (2), Area 8, Elevation 106.

m. Elevations are added as follows:

Q1G33G002R24, Elevation 102
Q1P41G007R19, Elevation 144
Q1P41G007R20, Elevation 144
Q1P41G007R23 (2), Elevation 138
Q1P41G007R24 (2), Elevation 137

2. Deletions to the table are the following:

a. Q1B21G023R08, Area 11, Elevation 126 - Only one Q1B21G023R08 snubber exists. One remains in the table.

b. Q1B21G032R04, Area 11, Elevation 127 - Only one Q1B21G032R04 snubber exists. One remains in the table.

c. Q1B21G218R02 and Q1B21G218R03, Area 11, Elevation 161. These two were not installed in the plant.

d. Q1B21G382R01, Area 11, Elevation 155. This snubber was voided and not installed.

e. Q1P41G002C03, Area 8, Elevation 95. This snubber was superceded by Q1E12G025C01 (2), Area 8, Elevation 95.

3. Typo corrections include the following.

a. Present Q1B21G022R10 should be Q1B21G023R10.

b. Presently there are two Q1E21G002R01 snubbers. One of these should be corrected to [illegible].

[illegible] of these should be corrected to [illegible].

[illegible] should be Elevation [illegible].

e. Elevation for Q1B21G226R01 should be 173' instead of present 172'.

f. Both of Q1B21G195R02 snubbers are at elevation 160' instead of present 160' for one and 161' for the other.

4. Non-Q Mechanical Snubbers are added to the Table as Section 2. Section 1 of the Table is Safety Related Mechanical Snubbers and Table 3.7.4-2 title is changed to "Mechanical Snubbers" to allow the two separate sections. The Non-Q Mechanical Snubbers are added because even though they are on Non-Q piping this was included in Stress analysis calculations for adjacent Q piping.

5. Duplicate listing of snubbers is avoided by listing only one and using parenthesis to indicate the number of snubbers with identical listings.

SIGNIFICANT HAZARDS CONSIDERATION:

The changes proposed to Table 3.7.4-2 are administrative in nature and do not involve changes to the plant. This change does not involve the reduction of safety margins and no significant increase in the probability or consequences of an accident previously evaluated is involved nor is the possibility of a new or different kind of accident from any accident previously evaluated created. Thus the proposed change to the Technical Specification does not involve any significant hazards consideration.

TABLE 3.7.4-2

~~SAFETY RELATED~~ MECHANICAL SNUBBERS*,#‡

[illegible] [illegible] [illegible]RTED MECHANICAL SNUBBERS

SNUBBER NO.	AREA	ELEVATION
[illegible]TION SYSTEM		
[illegible]R01 (2)	11	117
~~[illegible]R01~~	~~11~~	~~117~~
[illegible]R01	11	102
[illegible]R02 (2)	11	102
~~[illegible]R03~~	~~11~~	~~102~~
[illegible]R05	11	101
[illegible]G10[illegible]C01	11	101
[illegible]G10[illegible]R01	11	101
[illegible]G10[illegible]R02 (2)	11	101
~~[illegible]R02~~	~~11~~	~~101~~
[illegible]C01	11	101
[illegible]R01 (3)	11	101
~~[illegible]R02~~	~~11~~	~~101~~
~~[illegible]R01~~	~~11~~	~~101~~
[illegible]R02 (2)	11	101
~~[illegible]R02~~	~~11~~	~~101~~
RECIRCULATION SYSTEM (Continued)		
Q1B33G112R02	11	101
Q1B33G124R01	11	122
Q1B33G128C01 (2)	11	121
~~Q1B33G128C01~~	~~11~~	~~121~~
Q1B33G129C01	11	121
Q1B33G262R02	11	103
Q1B33G265C01	11	102
Q1B33G265R04	11	107
Q1B33G265R05	11	112
Q1B33G322R01 (2)	11	112
~~Q1B33G322R01~~	~~11~~	~~112~~
Q1B33G331R02	11	111
Q1B33G337R02	11	109
Q1B33G339R01	11	111
Q1B33G346R01	11	105
Q1B33G355R01 (2)	11	~~103~~ 102
Q1B33G318R01	11	~~108~~ 102
Q1B33G122R01	11	108

*[illegible] may be added to safety related systems without prior License Amendment to Table 3.7.4-2 provided that a revision to Table 3.7.4-2 is included with the next License Amendment request.

‡ The number in parentheses is the number of snubbers associated with the component support. If no number in parentheses appears, there is only one snubber associated with the support.

TABLE 3.7.4-2 (Continued)

~~SAFETY RELATED~~ MECHANICAL SNUBBERS*, **

SAFETY RELATED MECHANICAL SNUBBERS

SNUBBER NO.	AREA	ELEVATION
5. MAIN STEAM SYSTEM		
Q1B21G022C04	11	141
Q1B21G022R01 (2)	11	135
~~Q1B21G022R01~~	~~11~~	~~135~~
Q1B21G022R03 (2)	11	133
~~Q1B21G022R03~~	~~11~~	~~133~~
Q1B21G022R06 (2)	11	124
~~Q1B21G022R06~~	~~11~~	~~124~~
Q1B21G022R12 (2)	11	132
~~Q1B21G022R12~~	~~11~~	~~132~~
Q1B21G022R13 (2)	11	131
~~Q1B21G022R13~~	~~11~~	~~131~~
Q1B21G022R14	11	126
Q1B21G022R15	11	125
Q1B21G022R16	11	121
Q1B21G023R03	11	137
Q1B21G023R05	11	133
Q1B21G023R06 (2)	11	133
~~Q1B21G023R06~~	~~11~~	~~133~~
Q1B21G023R0[illegible]	11	126
~~Q1B21G023R0[illegible]~~	~~11~~	~~126~~
Q1B21G023R09	11	122
Q1B21G023R10	11	122
Q1B21G023R11 (2)	11	120
~~Q1B21G023R11~~	~~11~~	~~120~~
Q1B21G023R14	11	141
Q1B21G023R15 (2)	11	141
~~Q1B21G023R15~~	~~11~~	~~141~~
Q1B21G023R16	11	133
Q1B21G023R17	11	121

SNUBBER NO.	AREA	ELEVATION
MAIN STEAM SYSTEM (Continued)		
Q1B21G023R18 (2)	11	119
~~Q1B21G023R18~~	~~11~~	~~119~~
Q1B21G024C01	11	131
Q1B21G024R04	11	137
Q1B21G024R05 (2)	11	132
~~Q1B21G024R05~~	~~11~~	~~132~~
Q1B21G024R06	11	125
Q1B21G024R07 (2)	11	119
~~Q1B21G024R07~~	~~11~~	~~119~~
Q1B21G024R11	11	138
Q1B21G024R12 (2)	11	127
~~Q1B21G024R12~~	~~11~~	~~127~~
Q1B21G024R13	11	123
Q1B21G024R17	11	120
Q1B21G025R02	11	128
Q1B21G025R03	11	125
Q1B21G025R04 (2)	11	124
~~Q1B21G025R04~~	~~11~~	~~124~~
Q1B21G025R05	11	120
Q1B21G026C01 (2)	11	143
~~Q1B21G026C01~~	~~11~~	~~143~~
Q1B21G026C02 (2)	11	143
~~Q1B21G026C02~~	~~11~~	~~143~~
Q1B21G026R01	11	143
Q1B21G026R02 (2)	11	153
~~Q1B21G026R02~~	~~11~~	~~153~~
Q1B21G026R03	11	149
Q1B21G026R04 (2)	11	153
~~Q1B21G026R04~~	~~11~~	~~153~~
Q1B21G023R20	11	120

TABLE 3.7.4-2 (Continued)

~~SAFETY RELATED~~ MECHANICAL SNUBBERS*, **

1. [illegible] RELATED MECHANICAL SNUBBERS

SNUBBER NO.	AREA	ELEVATION
MAIN STEAM SYSTEM (Continued)		
Q1B21G026R05	11	143
Q1B21G026R06 (2)	11	143
~~Q1B21G026R06~~	~~11~~	~~143~~
Q1B21G026R07	11	143
Q1B21G026R08	11	149
Q1B21G030R03 (2)	8	143
~~Q1B21G030R03~~	~~8~~	~~143~~
Q1B21G032R03	11	129
Q1B21G032R04	11	127
~~Q1B21G032R04~~	~~11~~	~~127~~
Q1B21G032R05 (2)	11	120
~~Q1B21G032R05~~	~~11~~	~~120~~
[illegible]G123R[illegible]	11	165
[illegible]G126[illegible]	11	159
~~[illegible]~~	~~11~~	~~193~~
Q1B21G127R01 (2)	11	193
Q1B21G127R04	11	186
Q1B21G139R01	11	150
Q1B21G139R02	11	150
Q1B21G141R01	11	173
Q1B21G142R01 (2)	11	173
~~Q1B21G142R01~~	~~11~~	~~173~~
Q1B21G144R01	11	173
Q1B21G146C03 (2)	11	169
~~Q1B21G146C03~~	~~11~~	~~169~~
Q1B21G146C04	11	169
Q1B21G146R03	11	173
Q1B21G147C02	11	167
Q1B21G148C01 (2)	11	173
MAIN STEAM SYSTEM (Continued)		
~~Q1B21G148C01~~	~~11~~	~~173~~
Q1B21G149R01 (2)	11	172
~~Q1B21G149R01~~	~~11~~	~~172~~
Q1B21G153C01	11	174
Q1B21G153C02	11	182
Q1B21G153C03 (2)	11	171
~~Q1B21G153C03~~	~~11~~	~~171~~
Q1B21G153R01	11	181
Q1B21G153R02 (2)	11	175
~~Q1B21G153R02~~	~~11~~	~~175~~
Q1B21G153R03 (2)	11	172
~~Q1B21G153R03~~	~~11~~	~~172~~
Q1B21G153R05 (2)	11	170
~~Q1B21G153R05~~	~~11~~	~~173~~
Q1B21G162R01	11	113
Q1B21G171R01	11	165
Q1B21G174C01 (2)	11	196
~~Q1B21G174C01~~	~~11~~	~~196~~
Q1B21G174R01	11	197
Q1B21G174R02	11	196
Q1B21G175R01 (2)	11	153
~~Q1B21G175R01~~	~~11~~	~~153~~
Q1B21G175R02 (2)	11	158
~~Q1B21G175R02~~	~~11~~	~~158~~
Q1B21G180R01	11	152
Q1B21G180R02 (2)	11	158
~~Q1B21G180R02~~	~~11~~	~~158~~
Q1B21G180R03	11	161
Q1B21G181C01	11	150
Q1B21G163R01	11	113
Q1B21G163R02	11	113

TABLE 3.7.4-2 (Continued)

~~SAFETY RELATED~~ MECHANICAL SNUBBERS*, **

SAFETY RELATED MECHANICAL SNUBBERS

SNUBBER NO.	AREA	ELEVATION
MAIN STEAM SYSTEM (Continued)		
Q1B21G183R01 (2)	11	152
~~Q1B21G183R01~~	~~11~~	~~152~~
Q1B21G183R02	11	151
Q1B21G191R01	11	161
Q1B21G19[illegible] (2)	11	159
~~[illegible]~~	~~11~~	~~159~~
Q1B21G195R01	11	161
Q1B21G195R02 (2)	11	160
~~Q1B21G195R02~~	~~11~~	~~161~~
Q1B21G196R01 (2)	11	151
~~Q1B21G196R01~~	~~11~~	~~151~~
Q1B21G197R01 (2)	11	157
~~Q1B21G197R01~~	~~11~~	~~157~~
Q1B21G201R01	11	158
Q1B21G201R02 (2)	11	157
~~Q1B21G201R02~~	~~11~~	~~157~~
Q1B21G204R01	11	152
Q1B21G204R02 (2)	11	160
~~Q1B21G204R02~~	~~11~~	~~160~~
Q1B21G205R01	11	159
Q1B21G205R02 (2)	11	160
~~Q1B21G205R02~~	~~11~~	~~160~~
Q1B21G208R01	11	157
Q1B21G208R03	11	160
Q1B21G210R01 (2)	11	157
~~Q1B21G210R01~~	~~11~~	~~157~~
Q1B21G213R01	11	151
~~Q1B21G213R02~~	~~11~~	
Q1B21G181R01	11	153

SNUBBER NO.	AREA	ELEVATION
MAIN STEAM SYSTEM (Continued)		
Q1B21G213R02 (2)	11	152
Q1B21G217R02	11	159
~~Q1B21G218R02~~	~~11~~	~~161~~
~~Q1B21G218R03~~	~~11~~	~~161~~
Q1B21G219R01 (2)	11	157
~~Q1B21G219R01~~	~~11~~	~~157~~
Q1B21G222R01	11	160
Q1B21G224R01	11	152
Q1B21G225R01	11	147
Q1B21G226C03	11	168
Q1B21G226R01 (2)	11	173
~~Q1B21G226R01~~	~~11~~	~~172~~
Q1B21G304R01	11	156
Q1B21G306R01	11	151
Q1B21G311R01 (2)	11	152
~~Q1B21G311R01~~	~~11~~	~~152~~
Q1B21G355R01	11	147
Q1B21G357C03	11	148
Q1B21G359C03	11	148
Q1B21G361C03	11	147
Q1B21G372R01 (2)	11	148
~~Q1B21G372R01~~	~~11~~	~~148~~
Q1B21G382R02 (2)	11	155
~~Q1B21G382R01~~	~~11~~	~~155~~
Q1B21G423R01	11	147
Q1B21G424R01	11	147
Q1B21G490R03	11	152
Q1B21G384R01	11	152
Q1B21G369R01 (2)	11	148

TABLE 3.7.4-2 (Continued)

~~SAFETY-RELATED~~ MECHANICAL SNUBBERS*, **

I. [illegible] RELATED MECHANICAL SNUBBERS

[illegible]	AREA	ELEVATION
[illegible] SYSTEM		
[illegible]G113C[illegible]	11	185
[illegible]G113C[illegible]	11	181
[illegible]G113[illegible]02	11	181
[illegible]G113R03	11	181
[illegible]G117C02	11	145
[illegible]G117R01	11	151
[illegible]G119R01 (2)	11	129
~~[illegible]19R01~~	~~11~~	~~129~~
[illegible]G119R03	11	114
[illegible]G119R04	11	112
[illegible]G119R05	11	112
[illegible]G120C05	11	155
[illegible]G124R01	11	159
[illegible]G124R03	11	162
[illegible]L HEAT REMOVAL SYSTEM		
Q1E12G009R03	7	134
Q1E12G009R04	7	134
Q1E12G009R05	8	134
Q1E12G009R06	8	134
Q1E12G010R02	8	105
Q1E12G010R04	8	103
Q1E12G010R05	8	125
Q1E12G010R07	8	133
Q1E12G010R10	8	142
Q1E12G010R11	8	142
Q1E12G010R13 (2)	8	113
~~Q1E12G010R13~~	~~8~~	~~113~~

SNUBBER NO.	AREA	ELEVATION
RESIDUAL HEAT REMOVAL SYSTEM (Continued)		
Q1E12G010R15	8	103
Q1E12G010R16	8	104
Q1E12G010R17 (2)	8	104
~~Q1E12G010R17~~	~~8~~	~~104~~
Q1E12G010R18 (2)	8	96
~~Q1E12G010R18~~	~~8~~	~~96~~
Q1E12G011R02 (3)	8	99
~~Q1E12G011R02~~	~~8~~	~~99~~
~~Q1E12G011R02~~	~~8~~	~~99~~
Q1E12G012R02 (2)	7	114
~~Q1E12G012R02~~	~~7~~	~~114~~
Q1E12G012R04	7	142
Q1E12G012R05	7	142
Q1E12G012R08	8	104
Q1E12G012R09	8	102
Q1E12G012R13	7	119
Q1E12G012R15	7	133
Q1E12G012R16	7	99
Q1E12G012R18	11	133
Q1E12G012R19	11	133
Q1E12G013C01	7	110
Q1E12G013C02	7	130
Q1E12G013R02 (2)	7	115
~~Q1E12G013R02~~	~~7~~	~~115~~
Q1E12G013R03	7	110
Q1E12G013R04	7	119
Q1E12G013R05 (2)	7	100
~~Q1E12G013R05~~	~~7~~	~~100~~
Q1E12G013R06 (3)	7	120

TABLE 3.7.4-2 (Continued)

~~SAFETY-RELATED~~ MECHANICAL SNUBBERS*, # ##

SA[illegible] [illegible]ATED MECHANICAL SNUBBERS

RESIDUAL HEAT REMOVAL SYSTEM (Continued)

SNUBBER NO.	AREA	ELEVATION
~~[illegible]R05~~	~~7~~	~~120~~
~~[illegible]G01[illegible]R06~~	~~7~~	~~120~~
[illegible]G01[illegible]R07	7	121
[illegible]G01[illegible]R08	7	105
[illegible]G013R11	7	97
[illegible]G01[illegible]C01	8	110
[illegible]G01[illegible]C03	8	106
[illegible]G01[illegible]C04	8	130
[illegible]G011R01 (2)	8	129
~~[illegible]R01~~	~~8~~	~~129~~
[illegible]G01[illegible]R03 (1)	8	98
~~[illegible]R03~~	~~8~~	~~98~~
[illegible]G01[illegible]R04 (3)	8	122
~~[illegible]R04~~	~~8~~	~~122~~
~~[illegible]R04~~	~~8~~	~~122~~
[illegible]2G01[illegible]R05	8	105
[illegible]2G014R07	8	106
[illegible]G01[illegible]R10 (2)	8	109
~~[illegible]R10~~	~~8~~	~~109~~
[illegible]2G01[illegible]R11 (2)	8	110
~~[illegible]R11~~	~~8~~	~~110~~
[illegible]G015R02	11	156
[illegible]G015R04 (2)	11	143
~~[illegible]5R04~~	~~11~~	~~143~~
[illegible]2G015R06	11	143
[illegible]2G015R07	11	214
[illegible]2G015R08	11	210
[illegible]G015R11	11	143
[illegible]2G015R17	11	210

RESIDUAL HEAT REMOVAL SYSTEM (Continued)

SNUBBER NO.	AREA	ELEVATION
Q1E12G015R19	11	214
Q1E12G015R20	11	144
Q1E12G015R21 (2)	11	140
~~Q1E12G015R21~~	~~11~~	~~140~~
Q1E12G015R28 (3)	11	192
~~Q1E12G015R28~~	~~11~~	~~192~~
~~Q1E12G015R28~~	~~11~~	~~192~~
Q1E12G015R33 (2)	11	205
~~Q1E12G015R33~~	~~11~~	~~205~~
Q1E12G015R38	11	157
Q1E12G016C01	11	143
Q1E12G016R01	11	146
Q1E12G016R02	11	143
Q1E12G016R03	11	143
Q1E12G016R05 (2)	11	143
~~Q1E12G016R05~~	~~11~~	~~143~~
Q1E12G019R05 (2)	8	139
~~Q1E12G019R05~~	~~8~~	~~139~~
Q1E12G019R07	8	149
Q1E12G019R08	7	149
Q1E12G019R09 (2)	7	143
~~Q1E12G019R09~~	~~7~~	~~143~~
Q1E12G020R01 (2)	8	140
~~Q1E12G020R01~~	~~8~~	~~148~~
Q1E12G020R02 (2)	7	148
~~Q1E12G020R02~~	~~7~~	~~148~~
Q1E12G020R03	8	140
Q1E12G020R04 (2)	8	148
~~Q1E12G020R04~~	~~8~~	~~148~~

TABLE 3.7.4-2 (Continued)

~~SAFETY RELATED~~ MECHANICAL SNUBBERS*, # #

I. S[illegible] RELATED MECHANICAL SNUBBERS

SNUBBER NO.	AREA	ELEVATION
d. RESIDUAL HEAT REMOVAL SYSTEM (Continued)		
Q1E12G020R05	7	147
Q1E12G020R07 (2)	7	147
~~Q1E12G020R07~~	~~7~~	~~147~~
Q1E12G020R09	7	147
Q1E12G021R01	8	147
Q1E12G021R03 (2)	8	146
~~[illegible]R03~~	~~8~~	~~146~~
Q1E12G025R01	8	110
Q1E12G119R02	7	152
[illegible]59R01	7	126
[illegible]59R03	7	126
[illegible]59R04	7	131
[illegible]25C01 (2)	8	95
e. LPCS SYSTEM		
Q1E21G001R05	9	96
Q1E21G001R07 (2)	9	96
~~Q1E21G001R07~~	~~9~~	~~96~~
Q1E21G002R01	11	150
Q1E21G002R02	11	150
Q1E21G002R03	11	151
Q1E21G002R04	11	153
Q1E21G002R05	11	153
Q1E21G002R06	11	153
Q1E21G002R07	11	150
f. HPCS SYSTEM		
Q1E22G001R10 (2)	8	96
~~Q1E22G001R10~~	~~8~~	~~96~~
Q1E22G002R02 (2)	8	96
~~Q1E22G002R02~~	~~8~~	~~96~~
Q1E22G002R03	8	96
Q1E22G003R01	11	153
Q1E22G003R02	11	153
Q1E22G003R03	11	149
Q1E22G003R04	11	150
Q1E22G003R05	11	151
g. RCS LEAK DETECTION SYSTEM		
Q1E31G116R01	11	169
Q1E31G122R01 (2)	11	149
Q1E31G124R01 (2)	11	151
~~Q1E31G124R01~~	~~11~~	~~151~~
Q1E31G126C01	11	149
Q1E31G140R01	11	159
Q1E31G140R02 (2)	11	159
~~Q1E31G140R02~~	~~11~~	~~159~~
Q1E31G148R01 (2)	11	151
~~Q1E31G148R01~~	~~11~~	~~151~~
Q1E31G149R01 (2)	11	151
~~Q1E31G149R01~~	~~11~~	~~151~~

TABLE 3.7.4-2 (Continued)

~~SAFETY-RELATED~~ MECHANICAL SNUBBERS*, **

1. [illegible] RELATED MECHANICAL SNUBBERS

SNUBBER NO.	AREA	ELEVATION
[illegible] DETECTION SYSTEM (Continued)		
[illegible]1G168R01	11	158
[illegible]1G174R01 (2)	11	151
~~[illegible]74R01~~	~~11~~	~~151~~
[illegible]1G176C01	11	147
[illegible]31G178R08	11	179
[illegible]31G178R05	11	179
[illegible]31G181R01	11	156
[illegible]1G243R01	11	144
[illegible]31G243R02	11	140
[illegible]31G246R01 (2)	11	144
~~[illegible]6R01~~	~~11~~	~~144~~
[illegible] LEAKAGE CONTROL SYSTEM		
[illegible]2G103C01 (2)	8	122
~~[illegible]103C01~~	~~8~~	~~122~~
[illegible]2G106C01	8	121
[illegible]2G199C01	8	122
[illegible]2G119C01	8	148
[illegible]TER LEAKAGE CONTROL SYSTEM		
[illegible]192R01	8	145
[illegible] SYSTEM		
[illegible]51G001R05	8	104
[illegible]51G001R06	8	109
[illegible]1G001R09	11	133

SHUBBER NO.	AREA	ELEVATION
RCIC SYSTEM (Continued)		
Q1E51G001R10 (2)	11	134
~~Q1E51G001R10~~	~~11~~	~~134~~
Q1E51G001R15	11	178
Q1E51G001R17 (2)	11	190
~~Q1E51G001R17~~	~~11~~	~~190~~
Q1E51G001R18	11	194
Q1E51G001R19 (2)	11	194
~~Q1E51G001R19~~	~~11~~	~~194~~
Q1E51G003R03	7	126
Q1E51G003R04	7	117
Q1E51G003R05 (2)	7	127
~~Q1E51G003R05~~	~~7~~	~~127~~
Q1E51G003R07	8	112
Q1E51G003R08 (2)	8	112
~~Q1E51G003R08~~	~~8~~	~~110~~
Q1E51G003R09 (2)	8	109
~~Q1E51G003R09~~	~~8~~	~~109~~
Q1E51G003R10	8	105
Q1E51G003R11 (2)	8	100
~~Q1E51G003R11~~	~~8~~	~~100~~
Q1E51G003R12 (2)	8	106
Q1E51G004C02 (2)	8	97
~~Q1E51G004C02~~	~~8~~	~~97~~
Q1E51G004R01 (2)	8	98
~~Q1E51G004R01~~	~~8~~	~~98~~
Q1E51G004R05 (2)	8	106
~~Q1E51G004R05~~	~~8~~	~~106~~
Q1E51G004R06 (2)	8	96
~~Q1E51G004R06~~	~~8~~	~~96~~

TABLE 3.7.4-2 (Continued)

~~SAFETY-RELATED~~ MECHANICAL SNUBBERS*, **

1. SAFETY RELATED MECHANICAL SNUBBERS

SNUBBER NO.	AREA	ELEVATION
RCIC SYSTEM (Continued)		
Q1E51G004R07 (2)	8	97
~~Q1E51G004R07~~	~~8~~	~~97~~
Q1E51G004R08 (2)	11	164
~~Q1E51G004R08~~	~~11~~	~~164~~
Q1E51G004R11	8	97
Q1E51G004R13 (2)	11	167
~~Q1E51G004R13~~	~~11~~	~~167~~
Q1E51G004R14 (2)	11	152
~~Q1E51G004R14~~	~~11~~	~~152~~
Q1E51G150R03 (2)	11	143
~~Q1E51G150R03~~	~~11~~	~~143~~
COMBUSTIBLE GAS CONTROL SYSTEM		
Q1E61G001R07	11	189
RWCU SYSTEM		
Q1G33G00[illegible]03 (2)	11	113
~~Q1G33G00[illegible]03~~	~~11~~	~~113~~
Q1G33G00[illegible]R03 (2)	8	136
~~Q1G33G00[illegible]R03~~	~~8~~	~~136~~
Q1G33G002R05 (2)	11	140
~~Q1G33G002R05~~	~~11~~	~~140~~
Q1G33G002R08 (2)	11	102
~~Q1G33G002R08~~	~~11~~	~~102~~
Q1G33G002R09 (3)	11	102
~~Q1G33G002R09~~	~~11~~	~~102~~
~~Q1G33G002R09~~	~~11~~	~~102~~

SNUBBER NO.	AREA	ELEVATION
RWCU SYSTEM (Continued)		
Q1G33G002R10 (2)	11	102
~~Q1G33G002R10~~	~~11~~	~~102~~
Q1G33G002R11	11	102
Q1G33G002R12	11	102
Q1G33G002R13 (2)	11	102
~~Q1G33G002R13~~	~~11~~	~~102~~
Q1G33G002R14 (2)	11	102
~~Q1G33G002R14~~	~~11~~	~~102~~
Q1G33G002R16	11	112
Q1G33G002R17 (2)	8	125
~~Q1G33G002R17~~	~~8~~	~~125~~
Q1G33G002R18	8	116
Q1G33G002R19	8	116
Q1G33G002R21 (2)	11	102
~~Q1G33G002R21~~	~~11~~	~~102~~
Q1G33G002R22	11	102
Q1G33G002R24	11	102
Q1G33G011R01	11	140
Q1G33G011R03 (2)	11	145
~~Q1G33G011R03~~	~~11~~	~~145~~
Q1G33G012R01 (2)	11	142
~~Q1G33G012R01~~	~~11~~	~~142~~
Q1G33G012R02	11	152
Q1G33G105R01 (3)	11	103
~~Q1G33G105R01~~	~~11~~	~~103~~
~~Q1G33G105R01~~	~~11~~	~~103~~

TABLE 3.7.4-2 (Continued)

~~SAFETY RELATED~~ MECHANICAL SNUBBERS*, **

1. SAFETY RELATED MECHANICAL SNUBBERS

	AREA	ELEVATION
ESSC SYSTEM		
Q1G41G006R01	9	114
Q1G41G006R07 (3)	7	99
~~Q1G41G006R07~~	~~7~~	~~99~~
~~Q1G41G006R07~~	~~7~~	~~99~~
Q1G41G015R09	11	204
Q1G41G016C08	11	163
Q1G41G016R04	11	166
Q1G41G016R24	11	163
Q1G41G016R27 (2)	11	203
~~Q1G41G016R27~~	~~11~~	~~203~~
Q1G41G016R28 (2)	11	206
~~Q1G41G016R28~~	~~11~~	~~206~~
Q1G41G016R32	11	197
Q1G41G018R06	9	197
SSW SYSTEM		
Q1P41G001R14 (2)	7	98
~~Q1P41G001R14~~	~~7~~	~~98~~
~~Q1P41G002R09~~	~~8~~	~~95~~
Q1P41G002R10 (2)	8	106

SNUBBER NO.	AREA	ELEVATION
SSW SYSTEM (Continued)		
~~Q1P41G002R10~~	~~8~~	~~106~~
Q1P41G002R12 (2)	8	106
~~Q1P41G002R12~~	~~8~~	~~106~~
Q1P41G006C01	8	99
Q1P41G006C17	8	99
Q1P41G007R19	025A	144
Q1P41G007R20	025A	144
Q1P41G007R23 (2)	025A	138
~~Q1P41G007R23~~	~~025A~~	
Q1P41G007R24 (2)	025A	137
~~Q1P41G007R24~~	~~025A~~	

o. CCW SYSTEM

Q1P42G002R06 (2)	9	193
~~Q1P42G002R06~~	~~9~~	~~193~~
Q1P42G002R07 (2)	9	186
~~Q1P42G002R07~~	~~9~~	~~186~~
Q1P42G002R11 (2)	9	186
~~Q1P42G002R11~~	~~9~~	~~186~~
Q1P42G002R13 (2)	9	186
~~Q1P42G002R13~~	~~9~~	~~186~~

TABLE 3.7.4-2 (Continued)

MECHANICAL SNUBBERS*,**

NO[illegible]CAL SNUBBERS

a. [illegible] STEAM SYSTEM

SNUBBER NUMBER	AREA	ELEVATION
[illegible]1G1[illegible]BR[illegible]	11	148
[illegible]1G1[illegible]BR[illegible]2	11	147
[illegible]1G1[illegible]1C02	11	137
[illegible]1G1[illegible]2C03	11	136
[illegible]1G1[illegible]3R01 (2)	11	138
[illegible]1G1[illegible]3R04	11	136
[illegible]1G2[illegible]1R01 (2)	11	163
N1B33G108C02	11	101
N1B33G108R03 (2)	11	101
N1B33G108R05	11	101
N1B33G108R06 (2)	11	101
N1B33G108R07	11	101
N1B33G119R04	11	112
N1B33G120R03	11	101
N1B33G123C01	11	102
N1B33G362R03	11	102

b. [illegible]RCULATION SYSTEM

SNUBBER NUMBER	AREA	ELEVATION
[illegible]G104R02	11	102
[illegible]G105C01	11	101
[illegible]G105C03	11	101
[illegible]G105C04	11	101
[illegible]G105C05	11	101
[illegible]G105R01	11	101
[illegible]G106R01	11	102
[illegible]G107R01	11	102
[illegible]G107R02	11	102

c. RESIDUAL HEAT REMOVAL SYSTEM

SNUBBER NUMBER	AREA	ELEVATION
N1E12G172R02	11	129
N1E12G212R01	11	136
N1E12G212R03	11	133

TABLE 3.7.4-2 (Continued)

MECHANICAL SNUBBERS*,**

[illegible] CAL SNUBBERS

[illegible] NUMBER	AREA	ELEVATION
d. [illegible] DIESEL GEN. SYSTEM		
[illegible]5G004R01	12	165
[illegible]5G004R02 (2)	12	152
[illegible]4R03 (2)	12	152
[illegible]5R01	12	165
[illegible]5R02 (2)	12	152
[illegible]5R03 (2)	12	152
e. [illegible] CORE ISOLATING COOLING SYSTEM		
[illegible]20R01	11	127
f. [illegible] CLEANUP [illegible]YSTEM		
[illegible]3G002[illegible]01	7	120
[illegible]33G002R02	8	118
[illegible]3G002R03	8	123
[illegible]3G002R04	8	123

SNUBBER NUMBER	AREA	ELEVATION
REACTOR WATER CLEANUP SYSTEM		
N1G33G002R05 (2)	11	147
N1G33G002R08 (2)	11	164
N1G33G002R10 (2)	11	147
N1G33G002R11 (3)	11	180
N1G33G002R12 (3)	11	180
N1G33G002R13	11	178
N1G33G002R14	8	120
N1G33G002R21	8	120

TECHNICAL SPECIFICATION PROBLEM SHEET

Item Number: 146 Priority: ~~1~~ 1A

_______________ / _______
Identified By Date

____
Responsible Supervisor

Tech Spec Reference: SER 9.5.4.1/Tech Spec 6.8

Problem Title: Control Room Ceiling Work

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other):
Per SER, work in control room ceiling may only be performed in cold shutdown.

2. Safety Significance:
Unimplemented SER requirement.

3. Anticipated Resolution:
Revise admin section of Tech Specs and implement controls.
Use Administrative controls.

4. NRC Response to Item (NRR/IE): _______________

NRC Notified: _______________ _______ / _______
Individual Notified Date Time

5. Disposition: _______________

Items Closed: (How) _______________

_______ / _______
Date Time

cc: J. E. Cross
R. F. Rogers

SUBJECT: Technical Specification 6.8.1, page 6-14

No. -146

DISCUSSION: Section 9.5.4.1 of the Grand Gulf Safety Evaluation Report (SER) states that in Amendment 49 to the FSAR, MP&L agreed to maintain the access door to the concealed ceiling space above the control room locked at all times with strict key control, and to provide technical specification requirements that prohibit work of any kind in the concealed area unless the plant is in a cold shutdown situation.

At present, procedures do exist which meet the intent of this commitment; however, they are not consolidated nor are they all included in the technical specifications. MP&L has therefore implemented an Access Control Program and will add this program to Section 6.8.l of the Grand Gulf Technical Specifications.

JUSTIFICATION: Although Grand Gulf has various procedures, programs, and administrative controls that together meet the intent of our SER commitment concerning the control room concealed ceiling space, the new Access Control Program will better address controlling access to those areas that requires additional restrictions. The Access Control Program will consolidate into one document a list of areas requiring additional restrictions and address the specific concerns involving each area. All areas included in the Access Control Program will be under strict control of the Shift Supervisor.

SIGNIFICANT HAZARDS CONSIDERATION:

The proposed addition of an Access Control Program to the Administrative Controls Section of the Grand Gulf Technical Specification constitutes an additional limitation not presently included in the Technical Specifications. The proposed change does not involve: a) the reduction of safety margins; b) an increase in the probability or consequences of a previously evaluated accident; or c) the possible creation of a new or different kind of accident. Thus the proposed change does not involve a significant hazards consideration.

L14sd!

j. Access Control Program for those areas required to be under strict control of the Shift Supervisor, (e.g., see Section 9.5.4.1 of NUREG-0831).

SAFETY LIMIT VIOLATION (Continued)

c. The Safety Limit Violation Report shall be submitted to the Commission, the SRC and the Senior Vice President - Nuclear within 14 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. PROCESS CONTROL PROGRAM implementation.

h. OFFSITE DOSE CALCULATION MANUAL implementation.

i. Quality Assurance Program for effluent and environmental monitoring, using the guidance in Regulatory Guide 4.15, February 1979.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed as required by 6.5, above, prior to implementation and shall be reviewed periodically as set forth in administrative procedures.

6.8.3 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the:

1. RCIC system outside containment containing steam or water, except the drain line to the main condenser.

2. RHR system outside containment containing steam or water, except the line to the LRW system and headers that are isolated by manual valves.

3. HPCS system.

4. LPCS system.

5. Hydrogen analyzers of the combustible gas control system.

TECHNICAL SPECIFICATION PROBLEM SHEET

Item Number: 180 Priority: ~~3a~~ ~~1C~~ 1D

Identified By / Date

Responsible Supervisor

Tech Spec Reference: 4.8.4.3

Problem Title: ~~B~~ RPS Undervoltage Setpoint

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other):

The under-voltage setpoint should be changed to reflect a more accurate operating range.

2. Safety Significance: None: Enhancement only.

3. Anticipated Resolution: Revise setpoint.
   live with.

4. NRC Response to Item (NRR/IE):

   NRC Notified: Individual Notified Date / Time

5. Disposition:

   Items Closed: (How)

Date / Time

cc: J. E. Cross
R. F. Rogers

Item No. 180

TECHNICAL SPECIFICATION PROBLEM SHEET

Priority 3a

Identified By ______ Date ____/____ Responsible Supervisor K

Tech Spec Reference: 4.8.4.3

Problem Title: RPS UNDERVOLTAGE SETPOINT

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other): THE UNDER-VOLTAGE SETPOINT SHOULD BE CHANGED TO REFLECT A MORE ACCURATE OPERATING RANGE.

2. Safety Significance: NONE, ENHANCEMENT ONLY.

3. Anticipated Resolution: REVISE SETPOINT.

4. NRC Response to Item (NRR/IE): ______

NRC Notified: ______ Individual Notified ______ Date ____/____ Time

5. Disposition: ______

Items Closed: (How) ______

______/______ Date Time

cc: J. E. Cross
R. F. Rogers

"TECH SPEC PRIORITY"

Punchlist Item # 180

Tech Spec 4.8.4.3

Priority 1C

TO: Manager of Nuclear Plant Engineering

FROM: Chairman, Prioritization and Disposition Chairman

SUBJECT: Technical Specifications Punchlist Item # 180

PDTS:84/ 0002

DATE: 3/10/84

The subject Tech Spec item has been determined by the Disposition Committee to require Engineering support.

DETAILS: Determine the setpoint range and provide the proper justification for the change

Please contact Joe Hendry at Extension 2675 for further information.

Please refer to the Tech Spec Punchlist item number in your response. Forward your response to G. Zinke.

J. C. Roberts
for Chairman

LLJ/JCR:swb
cc: Mr. C. L. Tyrone
Mr. J. E. Cross
Mr. D. Stonestreet
Mr. A. S. McCurdy
Mr. S. Hutchins
Mr. J. Hendry
File (Tech Spec Records)

A4/61swb1

"TECH SPEC PRIORITY"

Punchlist Item # See ATTached

Tech Spec See ATTached

Priority See ATTached

TO: Manager of Nuclear Plant Engineering

FROM: Chairman, Prioritization and Disposition Chairman

SUBJECT: Technical Specifications Punchlist Item # See ATTached

PDTS:84/ 0014

DATE: 3/10/84

The subject Tech Spec item has been determined by the Disposition Committee to require Engineering support.

DETAILS: This letter identifies requested response dates for the following Tech Spec problems:

#199 Letter No. PDTS 84/0001	#015 Letter No. PDTS 84/0007
#180 Letter NO. PDTS 84/0002	#198 Letter No. PDTS 84/0008
#033 Letter NO. PDTS 84/0003	#202 Letter NO. PDTS 84/0009
#054 Letter NO. PDTS 84/0004	#313 Letter NO. PDTS 84/0010
#001 Letter NO. PDTS 84/0005	#219 Letter NO. PDTS 84/0011
#016 Letter NO. PDTS 84/0006	#118 Letter NO. PDTS 84/0013

It is requested that the responses to the above items be completed by 3/13/84

Please contact Jerry RoberTs at Extension 2695 for further information.

Please refer to the Tech Spec Punchlist item number in your response. Forward your response to George ZINKE.

J. C. Roberts
Chairman

LLJ/JCR:swb

cc: Mr. C. L. Tyrone
Mr. J. E. Cross
Mr. D. Stonestreet
Mr. A. S. McCurdy
Mr. S. Hutchins
Mr. J. Hendry
File (Tech Spec Records)

A4/61swb1

Tech Spec Problem No.	Tech Spec	Priority
199	Table 3.3.6-1.5	1B
180	4.8.4.3	1D
033	Table 3.3.8-2	1B
054	3/4.3.8	1B
001	3/4.5.1	1B
016	3/4.3.8	1B
015	3/4.3.2.	1D
198	3/4.3.7	1B
202	3/4.3.7	1B
213	3/4.3.3	1B
219	Figure 3.4.6.1-1	1B
168	3.6.3.1	1B

"TECH SPEC PRIORITY"

MEMO TO: J. F. Pinto, Manager of Nuclear Plant Engineering

FROM: C. L. Tyrone, Project Manager

SUBJECT: Handling of Tech Spec Review Items

TSRO: 84/0001

DATE: March 11, 1984

This memorandum confirms our conversation of March 10, 1984. At that time, your assistance was requested in resolving discrepancies on eleven priority 1 items. Since then two items have been added. These items are all previously identified items which require early resolution with the NRC. A response is needed on these items by 12:30 PM on March 11, 1984. A list is attached.

Furthermore, all items of any priority identified (or previously known) which are being handled on this program require expeditious handling. This includes areas where requests are originated from other interfacing organizations such as the Plant or Nuclear Services. In any case where conflicts regarding highest priority is not clear, I am available to provide clarification.

It is suggested that you arrange 7 day a week support in this area as it is needed and arrange for all NPE personnel who will be involved in this effort to be available (or on call) in a manner that will support the Tech Spec Review program.

CLT

SHH:sad
Attachment

cc: J. B. Richard (w/a
J. P. McGaughy (w/a)
J. F. Pinto (w/a)
J. E. Cross (w/a)
T. H. Cloninger (w/a)
H. J. Green (w/a)
R. C. Fron (w/a)
D. W. Stonestreet (w/a)
[redacted]
T. E. Reaves, Jr. (w/a)
S. M. Feith (w/a)
J. G. Cesare (w/a)
G. W. Smith (w/a)
L. R. McKay (w/a)
L. C. Burgess (w/a)
File (Tech Spec Records) (w/a)

LIST OF CURRENT PRIORITY 1
ITEMS REQUIRING NPE SUPPORT

PDTS:84/	P/L #	Date Sent
001	199	3/10/84
002	180	3/10/84
003	033	3/10/84
004	054	3/10/84
005	001	3/10/84
006	016	3/10/84
007	015	3/10/84
008	198	3/10/84
009	202	3/10/84
010	213	3/10/84
011	219	3/10/84
012	083	3/10/84
013	168	3/10/84

TECHNICAL SPECIFICATION PROBLEM SHEET

Item Number: 180 Priority: ~~3C~~ ~~1C~~ 1D

_______________ / _______________
Identified By Date

____
Responsible Supervisor

Tech Spec Reference: 4.8.4.3

Problem Title: ~~B~~RPS Undervoltage Setpoint

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other):

The under-voltage setpoint should be changed to reflect a more accurate operating range.

2. Safety Significance: None: Enhancement only.

3. Anticipated Resolution: Revise setpoint.
live with.

4. NRC Response to Item (NRR/IE): _______________

NRC Notified: _______________ _______ / _______
Individual Notified Date Time

5. Disposition: _______________

Items Closed: (How) _______________

_______ / _______
Date Time

cc: J. E. Cross
R. F. Rogers

Item No. 180

TECHNICAL SPECIFICATION PROBLEM SHEET

Priority 3a

Identified By	/ Date	K Responsible Supervisor

Tech Spec Reference: 4.8.4.3

Problem Title: RPS UNDERVOLTAGE SETPOINT

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other): THE UNDER-VOLTA SETPOINT SHOULD BE CHANGED TO REFLECT a MORE ACCURA OPERATING RANGE.

2. Safety Significance: NONE, ENHANCEMENT ONLY.

3. Anticipated Resolution: REVISE SETPOINT.

4. NRC Response to Item (NRR/IE):

NRC Notified: ______ Individual Notified ______ Date / Time

5. Disposition:

Items Closed: (How)

Date / Time

cc: J. E. Cross
R. F. Rogers

"TECH SPEC PRIORITY"

Punchlist Item # 180

Tech Spec 4.8.4.3

Priority 1C

TO: Manager of Nuclear Plant Engineering

FROM: Chairman, Prioritization and Disposition Chairman

SUBJECT: Technical Specifications Punchlist Item # 180

PDTS:84/ 0002

DATE: 3/10/84

The subject Tech Spec item has been determined by the Disposition Committee to require Engineering support.

DETAILS: Determine the setpoint range and provide the proper justification for the change

Please contact Joe Hendry at Extension 2675 for further information.

Please refer to the Tech Spec Punchlist item number in your response. Forward your response to G. Zinke.

George [signature]
J. C. Roberts
for Chairman

LLJ/JCR:swb
cc: Mr. C. L. Tyrone
Mr. J. E. Cross
Mr. D. Stonestreet
Mr. A. S. McCurdy
Mr. S. Hutchins
Mr. J. Hendry
File (Tech Spec Records)

A4/61swb1

"TECH SPEC PRIORITY"

Punchlist Item # See ATTAched

Tech Spec See ATTAched

Priority See ATTAched

TO: Manager of Nuclear Plant Engineering

FROM: Chairman, Prioritization and Disposition Chairman

SUBJECT: Technical Specifications Punchlist Item # See ATTached

PDTS:84/ 0014

DATE: 3/10/84

The subject Tech Spec item has been determined by the Disposition Committee to require Engineering support.

DETAILS: This letter identifies requested response dates for the following Tech Spec problems:

#199 Letter No. PDTS 84/0001	#015 Letter No. PDTS 84/0007
#180 Letter NO. PDTS 84/0002	#198 Letter No. PDTS 84/0009
#033 Letter No. PDTS 84/0003	#202 Letter No. PDTS 84/0009
#054 Letter No. PDTS 84/0004	#213 Letter No. PDTS 84/0010
#001 Letter No. PDTS 84/0005	#219 Letter No. PDTS 84/0011
#016 Letter No. PDTS 84/0006	#118 Letter No. PDTS 84/0013

It is requested that the responses to the above items be completed by 3/13/84

Please contact Jerry Roberts at Extension 2695 for further information.

Please refer to the Tech Spec Punchlist item number in your response. Forward your response to George Zinke.

J. C. Roberts
Chairman

LLJ/JCR:swb
cc: Mr. C. L. Tyrone
Mr. J. E. Cross
Mr. D. Stonestreet
Mr. A. S. McCurdy
Mr. S. Hutchins
Mr. J. Hendry
File (Tech Spec Records)

A4/61swb1

Tech Spec Problem No.	Tech Spec	Priority
199	Table 3.3.6-1.5	1B
180	4.8.4.3	1D
033	Table 3.3.1-2	1B
054	3/4.3.8	1B
001	3/4.5.1	1B
016	3/4.3.8	1B
015	3/4.3.2	1D
198	3/4.3.7	1B
202	3/4.3.7	1B
213	3/4.3.3	1B
219	Figure 3.4.6.1-1	1B
168	3.6.3.1	1B

"TECH SPEC PRIORITY"

MEMO TO: J. F. Pinto, Manager of Nuclear Plant Engineering

FROM: C. L. Tyrone, Project Manager

SUBJECT: Handling of Tech Spec Review Items

TSRO: 84/0001

DATE: March 11, 1984

This memorandum confirms our conversation of March 10, 1984. At that time, your assistance was requested in resolving discrepancies on eleven priority 1 items. Since then two items have been added. These items are all previously identified items which require early resolution with the NRC. A response is needed on these items by 12:30 PM on March 11, 1984. A list is attached.

Furthermore, all items of any priority identified (or previously known) which are being handled on this program require expeditious handling. This includes areas where requests are originated from other interfacing organizations such as the Plant or Nuclear Services. In any case where conflicts regarding highest priority is not clear, I am available to provide clarification.

It is suggested that you arrange 7 day a week support in this area as it is needed and arrange for all NPE personnel who will be involved in this effort to be available (or on call) in a manner that will support the Tech Spec Review program.

CLT

SHH:sad
Attachment

cc: J. B. Richard (w/a
J. P. McGaughy (w/a)
J. F. Pinto (w/a)
J. E. Cross (w/a)
T. H. Cloninger (w/a)
H. J. Green (w/a)
R. C. Fron (w/a)
D. W. Stonestreet (w/a)

[redacted]
T. E. Reaves, Jr. (w/a)
S. M. Feith (w/a)
J. G. Cesare (w/a)
G. W. Smith (w/a)
L. R. McKay (w/a)
L. C. Burgess (w/a)
File (Tech Spec Records) (w/a)

LIST OF CURRENT PRIORITY 1
ITEMS REQUIRING NPE SUPPORT

PDTS:84/	P/L #	Date Sent
001	199	3/10/84
002	180	3/10/84
003	033	3/10/84
004	054	3/10/84
005	001	3/10/84
006	016	3/10/84
007	015	3/10/84
008	198	3/10/84
009	202	3/10/84
010	213	3/10/84
011	219	3/10/84
012	083	3/10/84
013	168	3/10/84

TECHNICAL SPECIFICATION PROBLEM SHEET

Item Number: 199 Priority: 1A

Steve Logan /
Identified By Date

Responsible Supervisor

Tech Spec Reference: Table 3.3.6-1.5

Problem Title: Scram Discharge Volume Bypass

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other):
Difference between STS and our Tech Specs. Add 5:b Scram Trip Bypass 2 1,2,5* 62.

2. Safety Significance: Interlock being verified in Surveillances every 18 months during LSFT, a discrepancy does exist in which non-conservative operation of the plant might occur.

3. Anticipated Resolution: Submit Tech Spec change to incorporate Scram Discharge Volume bypass so as to conform with Standard.
Cover in surveillance and use Tech Spec position statement.

4. NRC Response to Item (NRR/IE): ______

NRC Notified: ______ Individual Notified ______ / ______ Date Time

5. Disposition: ______

Items Closed: (How) ______

/
Date Time

cc: J. E. Cross
R. F. Rogers

Item No. 141 TECHNICAL SPECIFICATION PROBLEM SHEET Priority 1A

Steve Logan /
Identified By Date

Responsible Supervisor

Tech Spec Reference: Table 3.3.6-1.5

Problem Title: Scram Discharge Volume Bypass

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other): Difference between STS and our tech specs

add

5.b Scram Trip Bypass 2 1,2,5* 62

2. Safety Significance: ~~RBM~~ RRD 3/1/87 interlock being verified in Surveillances every 18 months during LSFT. A discrepancy does exist in which non conservative operation of the plant might occur.

3. Anticipated Resolution: Submit Tech Spec change to incorporate Scram Discharge Volume bypass so as to conform with Standard.

4. NRC Response to Item (NRR/IE):

NRC Notified:
Individual Notified Date / Time

5. Disposition:

Items Closed: (How)

/
Date Time

cc: J. E. Cross
R. F. Rogers

"TECH SPEC PRIORITY"

Punchlist Item # 199

Tech Spec Table 3.3.6-1.5

Priority 1A

TO: Manager of Nuclear Plant Engineering

FROM: Chairman, Prioritization and Disposition Chairman

SUBJECT: Technical Specifications Punchlist Item # 199

PDTS:84/ 0001

DATE: 3/10/84

The subject Tech Spec item has been determined by the Disposition Committee to require Engineering support.

DETAILS: Verify design vs STS Table 3.3-1 item 5b and Table 4.3.1-1 item 5b. Particular attention should be made to loop channels, action statements and testability.

Please contact Joe Hendry at Extension 2675 for further information.

Please refer to the Tech Spec Punchlist item number in your response. Forward your response to G. Zinke.

George Alan Zinke
J. C. Roberts
for Chairman

LLJ/JCR:swb

cc: Mr. C. L. Tyrone
Mr. J. E. Cross
Mr. D. Stonestreet
Mr. A. S. McCurdy
Mr. S. Hutchins
Mr. J. Hendry
File (Tech Spec Records)

A4/61swb1

"TECH SPEC PRIORITY"

Punchlist Item # See ATTAched

Tech Spec See ATTAched

Priority See ATTAched

TO: Manager of Nuclear Plant Engineering

FROM: Chairman, Prioritization and Disposition Chairman

SUBJECT: Technical Specifications Punchlist Item # See ATTached

PDTS:84/ 0014

DATE: 3/10/84

The subject Tech Spec item has been determined by the Disposition Committee to require Engineering support.

DETAILS: This letter identifies requested response dates for the following Tech Spec problems:

# 199 Letter No. PDTS 84/0001	#015 Letter No. PDTS 84/0007
# 180 Letter NO. PDTS 84/0002	#198 Letter No. PDTS 84/0008
# 033 Letter NO. PDTS 84/0003	#202 Letter NO. PDTS 84/0009
# 054 Letter NO. PDTS 84/0004	#213 Letter No. PDTS 84/0010
# 001 Letter NO. PDTS 84/0005	#219 Letter NO. PDTS 84/0011
# 016 Letter NO. PDTS 84/0006	#118 Letter NO. PDTS 84/0013

It is requested that the responses to the above items be completed by 3/13/84

Please contact Jerry RoberTs at Extension 2695 for further information.

Please refer to the Tech Spec Punchlist item number in your response. Forward your response to George ZINKE.

J. C. Roberts
Chairman

LLJ/JCR:swb

cc: Mr. C. L. Tyrone
Mr. J. E. Cross
Mr. D. Stonestreet
Mr. A. S. McCurdy
Mr. S. Hutchins
Mr. J. Hendry
File (Tech Spec Records)

A4/61swb1

Tech Spec Problem No.	Tech Spec	Priority
199	Table 3.3.6-1.5	1B
180	4.8.4.3	1D
033	Table 3.3.8-2	1B
054	3/4.3.8	1B
001	3/4.5.1	1B
016	3/4.3.8	1B
015	3/4.3.2	1D
198	3/4.3.7	1B
202	3/4.3.7	1B
213	3/4.3.3	1B
219	Figure 3.4.6.1-1	1B
168	3.6.3.1	1B

"TECH SPEC PRIORITY"

MEMO TO: J. F. Pinto, Manager of Nuclear Plant Engineering

FROM: C. L. Tyrone, Project Manager

SUBJECT: Handling of Tech Spec Review Items

TSRO: 84/0001

DATE: March 11, 1984

This memorandum confirms our conversation of March 10, 1984. At that time, your assistance was requested in resolving discrepancies on eleven priority 1 items. Since then two items have been added. These items are all previously identified items which require early resolution with the NRC. A response is needed on these items by 12:30 PM on March 12, 1984. A list is attached.

Furthermore, all items of any priority identified (or previously known) which are being handled on this program require expeditious handling. This includes areas where requests are originated from other interfacing organizations such as the Plant or Nuclear Services. In any case where conflicts regarding highest priority is not clear, I am available to provide clarification.

It is suggested that you arrange 7 day a week support in this area as it is needed and arrange for all NPE personnel who will be involved in this effort to be available (or on call) in a manner that will support the Tech Spec Review program.

CLT

SHH:sad
Attachment

cc: J. B. Richard (w/a
J. P. McGaughy (w/a)
J. F. Pinto (w/a)
J. E. Cross (w/a)
T. H. Cloninger (w/a)
H. J. Green (w/a)
R. C. Fron (w/a)
D. W. Stonestreet (w/a)
[redacted]
T. E. Reaves, Jr. (w/a)
S. M. Feith (w/a)
J. G. Cesare (w/a)
G. W. Smith (w/a)
L. R. McKay (w/a)
L. C. Burgess (w/a)
File (Tech Spec Records) (w/a)

LIST OF CURRENT PRIORITY 1
ITEMS REQUIRING NPE SUPPORT

PDTS:84/	P/L #	Date Sent
001	199	3/10/84
002	180	3/10/84
003	033	3/10/84
004	054	3/10/84
005	001	3/10/84
006	016	3/10/84
007	015	3/10/84
008	198	3/10/84
009	202	3/10/84
010	213	3/10/84
011	219	3/10/84
012	083	3/10/84
013	168	3/10/84

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

199

July 7, 1980

TO ALL OPERATING BOILING WATER REACTORS (BWR'S)

MAEC-801260

Gentlemen:

As you know, the staff has proposed improvements in the scram discharge volume (SDV) designs for BWR control rod drive systems to reduce susceptibility to common cause failures (NUREG-0460). We now request that you amend the Technical Specifications for your facility with respect to control rod drive scram discharge volume capability. The basis for our request is founded in events which have occurred in operating BWR's involving common cause failures of SDV limit switches and SDV drain valve operability. In IE Bulletin 80-14 dated June 13, 1980, you were requested to implement procedures and administrative controls to ensure that the SDV is operable during reactor operation. While the function of the bulletin was to effect immediate action with regard to this problem, the proposed Technical Specifications will strengthen the provisions for assuring continued operability of the control rod drive system during reactor operation.

You are requested to propose Technical Specification changes for your facility to provide surveillance requirements for SDV vent and drain valves and LCO/surveillance requirements for RPS and control rod block SDV limit switches. To assist you in preparing your submittal, we have enclosed a copy of Model Technical Specifications which would be sufficient to provide the assurance we seek. These Technical Specifications have been proposed by our staff to be incorporated into the next revision of the General Electric Standard Technical Specifications. Applicable changes are marked by vertical lines in the margins. Unchanged pages are included for completeness. Your proposal should use the enclosure as a guide and should include an appropriate Safety Analysis as a basis.

It is requested that you submit your proposed Technical Specifications with the basis within 90 days of receipt of this letter. If you have any questions about this request, please contact your Project Manager.

Sincerely,

Darrell G. Eisenhut

Darrel G. Eisenhut, Director
Division of Licensing

Enclosure:
Model Technical Specifications

~~8008050064~~

3/4.1.3 CONTROL RODS

CONTROL ROD OPERABILITY

LIMITING CONDITION FOR OPERATION

3.1.3.1 All control rods shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With one control rod inoperable due to being immovable, as a result of excessive friction or mechanical interference, or known to be untrippable:

   1. Within one hour:

      a) Verify that the inoperable control rod, if withdrawn, is separated from all other inoperable control rods by at least two control cells in all directions, and

      b) Disarm the associated directional control valves hydraulically by closing the insert and withdraw isolation valves,

      or be in at least HOT SHUTDOWN within the next 12 hours.

   2. Restore the inoperable control rod to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. With one or more control rods inoperable for causes other than addressed in ACTION a, above:

   1. If the inoperable control rod(s) is withdrawn, within one hour:

      a) Verify that the inoperable withdrawn control rod(s) is separated from all other inoperable control rods by at least two control cells in all directions, and

      b) Demonstrate the insertion capability of the inoperable withdrawn control rod(s) by inserting the control rod(s) at least one notch by drive water pressure within the normal operating range*, or

      c) Fully insert the inoperable withdrawn control rod(s) and disarm the associated directional control valves either:

         1) Electrically, or

         2) Hydraulically by closing the drive water and exhaust water isolation valves.

*The inoperable control rod may then be withdrawn to a position no further withdrawn than its position when found to be inoperable.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

2. If the inoperable control rod(s) is inserted, within one hour disarm the associated directional control valves either:

   a) Electrically, or

   b) Hydraulically by closing the drive water and exhaust water isolation valves.

3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

c. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The scram discharge volume drain and vent valves shall be verified open at least once per 31 days.*

4.1.3.1.2 All withdrawn control rods not required to have their directional control valves disarmed electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch:

a. At least once per 7 days when above the preset power level of the RWM and RSCS, and

b. At least once per 24 hours when above the preset power level of the RWM and RSCS and any control rod is immovable as a result of excessive friction or mechanical interference.

4.1.3.1.3 All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.1.3.2, 4.1.3.4, 4.1.3.5, 4.1.3.6 and 4.1.3.7.

4.1.3.1.4 All withdrawn control rods shall be determined OPERABLE by demonstrating the scram discharge volume drain and vent valves OPERABLE, when control rods are scram tested per Specification 4.1.3.2, by verifying that the drain and vent valves:

a. Close within __ seconds after receipt of a signal for control rods to scram, and

b. Open when the scram signal is reset or the scram discharge volume trip is bypassed.

*These valves may be closed intermittently for testing under administrative control.

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position (6), based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed (7.0) seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding (7.0) seconds:

a. Declare the control rod(s) with the slow insertion time inoperable, and

b. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with three or more control rods with maximum scram insertion times in excess of (7.0) seconds, or

c. Be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig and, during single control rod scram time tests, the control rod drive pumps isolated from the accumulators:

a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS or after a reactor shutdown that is greater than 120 days,

b. For specifically affected individual control rods following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and

c. For 10% of the control rods, on a rotating basis, at least once per 120 days of operation.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place at least one inoperable channel in the tripped condition within one hour.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one inoperable channel in at least one trip system* in the tripped condition within one hour and take the ACTION required by Table 3.3.1-1.

c. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip function shown in Table 3.3.1-2 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function.

*If both channels are inoperable in one trip system, select at least one inoperable channel in that trip system to place in the tripped condition, except when this would cause the Trip Function to occur.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
8. Scram Discharge Volume Water Level - High	1, 2, 5$^{(h)}$	2	4
9. Turbine Stop Valve - Closure	1$^{(i)}$	4$^{(j)}$	7
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	1$^{(i)}$	2$^{(j)}$	7
11. Reactor Mode Switch in Shutdown Position	1, 2, 3, 4, 5	1	8
12. Manual Scram	1, 2, 3, 4, 5	1	9

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION

ACTION 1 - In OPERATIONAL CONDITION 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

ACTION 2 - Lock the reactor mode switch in the Shutdown position within one hour.

ACTION 3 - Be in at least STARTUP within 2 hours.

ACTION 4 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

ACTION 5 - Be in at least HOT SHUTDOWN within 6 hours.

ACTION 6 - Be in STARTUP with the main steam line isolation valves closed within 2 hours or in at least HOT SHUTDOWN within 6 hours.

ACTION 7 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure to $\leq$ (250) psig, equivalent to THERMAL POWER less than (30)% of RATED THERMAL POWER, within 2 hours.

ACTION 8 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 3 or 4, verify all insertable control rods to be fully inserted within one hour.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

ACTION 9 In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 3 or 4, lock the reactor mode switch in the Shutdown position within one hour.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

*Except movement of IRM, SRM or special movable detectors, or replacement of LPRM strings provided SRM instrumentation is OPERABLE per Specification 3.9.2.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

b) The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn* and shutdown margin demonstrations performed per Specification 3.10.3.

(c) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than (11) LPRM inputs to an APRM channel.

(d) These functions are not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.10.1.

(e) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(f) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(g) Also actuates the standby gas treatment system.

(h) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(i) These functions are automatically bypassed when turbine first stage pressure is ≤ (250) psig, equivalent to THERMAL POWER less than (30)% of RATED THERMAL POWER.

(j) Also actuates the EOC-RPT system.

*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - Upscale	NA
b. Inoperative	NA
2. Average Power Range Monitor*:	
a. Neutron Flux - Upscale, (15)%	NA
b. Flow Biased Simulated Thermal Power - Upscale	$\leq$ (0.09)**
c. Fixed Neutron Flux - Upscale, (118)%	$\leq$ (0.09)
d. Inoperative	NA
e. LPRM	NA
3. Reactor Vessel Steam Dome Pressure - High	$\leq$ (0.55)
4. Reactor Vessel Water Level - Low, Level 3	$\leq$ (1.05)
5. Main Steam Line Isolation Valve - Closure	$\leq$ (0.06)
6. Main Steam Line Radiation - High	NA
7. Primary Containment Pressure - High	NA
8. Scram Discharge Volume Water Level - High	NA
9. Turbine Stop Valve - Closure	$\leq$ (0.06)
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	$\leq$ (0.08)#
11. Reactor Mode Switch in Shutdown Position	NA
12. Manual Scram	NA

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel. (This provision is not applicable to Construction Permits docketed after January 1, 1978. See Regulatory Guide 1.18, November 1977.)

**Not including simulated thermal power time constant.

#Measured from start of turbine control valve fast closure.

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High	NA	M	R	1, 2, 5
9. Turbine Stop Valve - Closure	NA	M	R	1
10. Turbine Control Valve Fast Closure Trip Oil Pressure - Low	NA	M	Q	1
11. Reactor Mode Switch in Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	M	NA	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) Within 24 hours prior to startup, if not performed within the previous 7 days.
(c) The IRM and SRM channels shall be determined to overlap for at least (   ) decades during each startup and the IRM and APRM channels shall be determined to overlap for at least (   ) decades during each controlled shutdown, if not performed within the previous 7 days.
(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference greater than 2%. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.
(e) This calibration shall consist of the adjustment of the APRM readout to conform to a calibrated flow signal.
(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.

3/4.3.6 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6. The control rod withdrawal block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE with their trip setpoints set consistant with the values shown in the Trip Setpoint column of Table 3.3.6-2.

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

a. With a control rod withdrawal block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, take the ACTION required by Table 3.3.6-1.

c. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

SURVEILLANCE REQUIREMENTS

4.3.6 Each of the above required control rod withdrawal block trip systems and instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1.

TABLE 3.3.6-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. ROD BLOCK MONITOR[(a)]			
a. Upscale	2	1*	60
b. Inoperative	2	1*	60
c. Downscale	2	1*	60
2. APRM			
a. Flow Biased Simulated Thermal Power - Upscale	4	1	61
b. Inoperative	4	1, 2, 5	61
c. Downscale	4	1	61
d. Neutron Flux - Upscale, Startup	4	2, 5	61
3. SOURCE RANGE MONITORS			
a. Detector not full in(b)	3	2	61
	2	5	61
b. Upscale[(c)]	3	2	61
	2	5	61
c. Inoperative[(c)]	3	2	61
	2	5	61
d. Downscale[(d)]	3	2	61
	2	5	61
4. INTERMEDIATE RANGE MONITORS			
a. Detector not full in (e)	6	2, 5	61
b. Upscale	6	2, 5	61
c. Inoperative	6	2, 5	61
d. Downscale[(e)]	6	2, 5	61
5. SCRAM DISCHARGE VOLUME			
a. Water Level-High	2	1, 2, 5**	62
b. Scram Trip Bypassed	1	1, 2, 5**	62
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW			
a. Upscale	2	1	62
b. Inoperative	2	1	62
c. (Comparator) (Downscale)	2	1	62

TABLE 3.3.6-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

ACTION

ACTION 60 - Take the ACTION required by Specification 3.1.4.3.

ACTION 61 - With the number of OPERABLE Channels:

a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.

b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.

ACTION 62 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour.

NOTES

* With THERMAL POWER $\geq$ (20)% of RATED THERMAL POWER.

** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

a. The RBM shall be automatically bypassed when a peripheral control rod is selected.

b. This function shall be automatically bypassed if detector count rate is > 100 cps or the IRM channels are on range (2) or higher.

c. This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

d. This function shall be automatically bypassed when the IRM channels are on range 3 or higher.

e. This function shall be automatically bypassed when the IRM channels are on range 1.

TABLE 3.3.6-2

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale	$\leq$ 0.66 W + (40)%	$\leq$ 0.66 W + (43)%
b. Inoperative	NA	NA
c. Downscale	$\geq$ (5)% of RATED THERMAL POWER	$\geq$ (3)% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Simulated Thermal Power - Upscale	$\leq$ 0.66 W + (42)%*	$\leq$ 0.66 W + (45)%*
b. Inoperative	NA	NA
c. Downscale	$\geq$ (5)% of RATED THERMAL POWER	$\geq$ (3)% of RATED THERMAL POWER
d. Neutron Flux - Upscale Startup	$\leq$ (12)% of RATED THERMAL POWER	$\leq$ (14)% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$\leq (2 \times 10^5)$ cps	$\leq (5 \times 10^5)$ cps
c. Inoperative	NA	NA
d. Downscale	$\geq$ (3) cps	$\geq$ (2) cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$\leq$ (100/125) of full scale	$\leq$ (110/125) of full scale
c. Inoperative	NA	NA
d. Downscale	$\geq$ (5/125) of full scale	$\geq$ (3/125) of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level High	$\leq$ (18) gallons	$\leq$ (18) gallons
b. Scram Trip Bypassed	NA	NA
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	$\leq$ (__/__) of full scale	$\leq$ (__/__) of full scale
b. Inoperative	NA	NA
c. (Comparator) (Downscale)	$\leq$ (10)% flow deviation	$\leq$ (__)% flow deviation

*The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

TABLE 4.3.6-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	NA	S/U(b), M	Q	1*
b. Inoperative	NA	S/U(b), M	NA	1*
c. Downscale	NA	S/U(b), M	Q	1*
2. APRM				
a. Flow Biased Simulated Thermal Power - Upscale	NA	S/U(b), M	Q	1
b. Inoperative	NA	S/U(b), M	NA	1, 2, 5
c. Downscale	NA	S/U(b), M	Q	1
d. Neutron Flux - Upscale, Startup	NA	S/U(b), M	Q	2, 5
3. SOURCE RANGE MONITORS				
a. Detector not full in	NA	S/U(b), W(c)	NA	2, 5
b. Upscale	NA	S/U(b), W(c)	Q	2, 5
c. Inoperative	NA	S/U(b), W(c)	NA	2, 5
d. Downscale	NA	S/U(b), W(c)	Q	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	S/U(b), W(c)	NA	2, 5
b. Upscale	NA	S/U(b), W(c)	Q	2, 5
c. Inoperative	NA	S/U(b), W(c)	NA	2, 5
d. Downscale	NA	S/U(b), W(c)	Q	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	NA	Q	R	1, 2, 5**
b. Scram Trip Bypassed	NA	M	NA	1, 2, 5**
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	NA	S/U(b), M	Q	1
b. Inoperative	NA	S/U(b), M	NA	1
c. (Comparator) (Downscale)	NA	S/U(b), M	Q	1

TABLE 4.3.6-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES:

a. Neutron detectors may be excluded from CHANNEL CALIBRATION.

b. Within 24 hours prior to startup, if not performed within the previous 7 days.

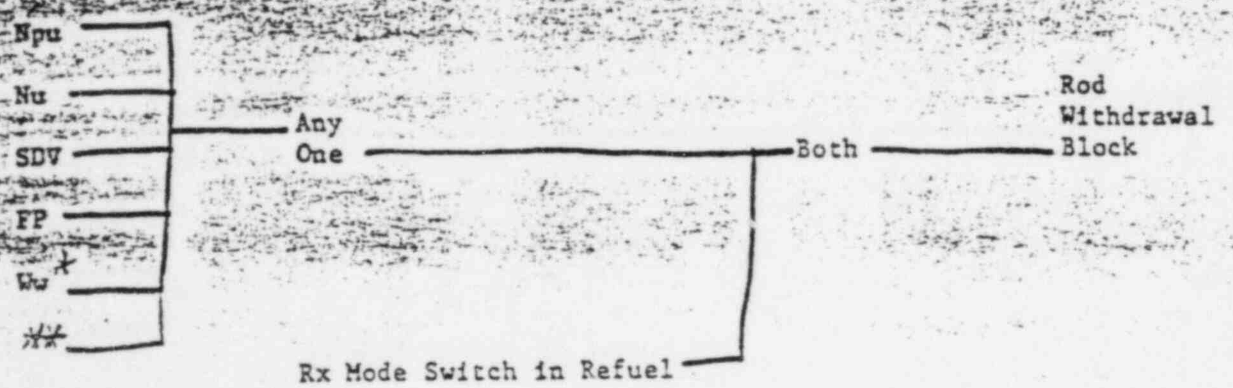
c. When making an unscheduled change from OPERATIONAL CONDITION 1 to OPERATIONAL CONDITION 2, perform the required surveillance within 12 hours after entering OPERATIONAL CONDITION 2.

* With THERMAL POWER $\geq$ (20)% of RATED THERMAL POWER.

** With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

DEFINITIONS FOR
"CHANNELS", "TRIP SYSTEMS", AND "TRIP FUNCTIONS"
FOR CONTROL ROD BLOCK INSTRUMENTATION TABLE 3.3.6-1 (Continued)

Rx Mode Swith in Refuel



Where: Mr = Rx Mode Switch in Run
Ms = Rx Mode Switch in Startup
Mrf = Rx Mode Switch in Refuel
FP = Refueling Equipment Rod Block Inputs.**

** CRD Scram Discharge Volume Bypass Switch in Bypass

* Wu - If RFCS not in operation, a rod block is initiated. This is the only rod block input from [illegible]

**The refueling equipment rod blocks are specified in Technical Specification 3/4.9.1

TECHNICAL SPECIFICATION PROBLEM SHEET

Item Number: 213 Priority: IA

_______________ / _______ ____________________
Identified By Date Responsible Supervisor

Tech Spec Reference: Incorp. Represent. of ECCS Man. Init. Logic

Problem Title: 3.3.3-1

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other):

Tech Spec Table 3.3.3-1 Items 3.3.3-1.A.2.9 and 3.3.3-1.B.2.f list minimum operable channels per trip system as 1/valve per plant design this should be 2/system.

2. Safety Significance:

Since ECCS Manual Initiation Logic (ADS) requires two manual initiation inputs, the plant could be operated in a non-conservative condition.

3. Anticipated Resolution:

Change Tech Spec Table 3.3.3-1 to reflect as built design of plant.
under evaluation (could include in surveillances)

4. NRC Response to Item (NRR/IE): ____________________

NRC Notified: ____________________ _______ / _______
Individual Notified Date Time

5. Disposition: ____________________

Items Closed: (How) ____________________

_______ / _______
Date Time

cc: J. E. Cross
R. F. Rogers

SUBJECT: Table 3.3.3-1 of Technical Specificatio
3.3.3, pages 3/4 3-25 and 3/4 3-27.

No. - 213

DISCUSSION: Two changes to the minimum number of operable channels are required on Table 3.3.3-1 to make the table reflect the Automatic Depressurization System (ADS) actuation logic.

Grand Gulf has two ADS trip systems (A and B). Each of these trip systems has two hand switches that must be manually actuated to initiate a system trip. Therefore the minimum number of operable channels (hand switches) should be two (2) per trip system.

Table 3.3.3-1, items A.2.g and B.2.f presently lists "1/valve" for the minimum number of operable channels per trip function; this should be changed to "2/system". Two per system is consistent with system actuation logic requirements.

The one per valve requirement presently listed refers to those hand switches that actuate the individual ADS valves and not to those handswitches that actuate the entire population of ADS valves, (i.e., an ADS trip system).

Implementation of the proposed changes to the minimum number of operable channels will invalidate the associated action statements. Action 32 is no longer applicable, as it refers to individual ADS valves. The applicable action statement would now be ACTION 31, which refers to the ADS trip system.

In addition to deleting ACTION 32 from items A.2.g and B.2.f, the action statement itself requires revision to delete the now misleading reference to individual ADS valves. To accomplish this the words "ADS valve or" should be deleted from action statement number 32.

JUSTIFICATION: The proposed change is necessary to make the information presented in Table 3.3.3-1 consistent with the Grand Gulf ADS trip system actuation logic. Two manual hand switches are installed in each of the two ADS trip systems and both hand switches in a system must be operable in order to manually initiate that system. The proposed change in the minimum number of operable channels (hand switches) per trip function from "1/valve" to "2/system" will accurately reflect this requirement.

The proposed change of action statements is necessary so that ~~[illegible]~~ manual initiation of the ADS will be associated with the actuation of the ADS trip system rather than the actuation of individual ADS valves.

SIGNIFICANT HAZARDS CONSIDERATION: The proposed changes ~~to the Technical Specifications will~~ make the information presented in Table 3.3.3-1 consistent with the as built ADS actuation logic requirements and are ~~It is also~~ conservative in that the new action statement (ACTION 31) requires the ADS trip system be declared inoperable immediately rather than within eight (8) hours as was allowed in ACTION 32.

The primary purpose of the proposed changes is to clarify Table 3.3.3-1 so that the reader will understant that the actuation instrument channels of concern are those associated with the AD. system and not with the individual ADS valves.

The proposed changes do not involve a) the reduction of safety margins, b) an increase in the probability or consequences of a previously evaluated accident, or c) the possible creation of a new or different kind of accident. Thus the proposed changes do not involve a significant hazards consideration.

TABLE 3.3.3-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION(a)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
A. DIVISION I TRIP SYSTEM			
1. RHR-A (LPCI MODE) & LPCS SYSTEM			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2(b)	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	2(b)	1, 2, 3	30
c. LPCI Pump A Start Time Delay Relay	1	1, 2, 3, 4*, 5*	31
d. Manual Initiation	1/system	1, 2, 3, 4*, 5*	32
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"#			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2(b)	1, 2, 3	30
b. Drywell Pressure - High	2(b)	1, 2, 3	30
c. ADS Timer	1	1, 2, 3	31
d. Reactor Vessel Water Level - Low, Level 3 (Permissive)	1	1, 2, 3	31
e. LPCS Pump Discharge Pressure-High (Permissive)	2	1, 2, 3	31
f. LPCI Pump A Discharge Pressure-High (Permissive)	2	1, 2, 3	31
g. Manual Initiation	~~1/valve~~ 2/system	1, 2, 3	~~32~~ 31
B. DIVISION 2 TRIP SYSTEM			
1. RHR B & C (LPCI MODE)			
a. Reactor Vessel Water Level - Low, Low Low, Level 1	2(b)	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	2(b)	1, 2, 3	30
c. LPCI Pump B Start Time Delay Relay	1	1, 2, 3, 4*, 5*	31
d. Manual Initiation	1/system	1, 2, 3, 4*, 5*	32
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"#			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2(b)	1, 2, 3	30
b. Drywell Pressure - High	2(b)	1, 2, 3	30
c. ADS Timer	1	1, 2, 3	31
d. Reactor Vessel Water Level - Low, Level 3 (Permissive)	1	1, 2, 3	31
e. LPCI Pump B and C Discharge Pressure - High (Permissive)	2/pump	1, 2, 3	31
f. Manual Initiation	~~1/valve~~ 2/system	1, 2, 3	~~32~~ 31

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION

ACTION 30 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

a. With one channel inoperable, place the inoperable channel in the tripped condition within one hour* or declare the associated system(s) inoperable.

b. With more than one channel inoperable, declare the associated system(s) inoperable.

ACTION 31 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, declare the associated ADS trip system or ECCS inoperable.

ACTION 32 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 8 hours or declare the associated ~~ADS valve or~~ ECCS inoperable.

ACTION 33 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

a. For one trip system, place that trip system in the tripped condition within one hour* or declare the HPCS system inoperable.

b. For both trip systems, declare the HPCS system inoperable.

ACTION 34 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour* or declare the HPCS system inoperable.

*The provisions of Specification 3.0.4 are not applicable.

Item No. 213

TECHNICAL SPECIFICATION PROBLEM SHEET

Priority 1A

CK

Identified By / Date

Responsible Supervisor

Tech Spec Reference: 3.3.3-1

Problem Title: Inaccurate representation of ECCS manual initiation logic input

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other):

Tech Spec Table 3.3.3-1 Items 3.3.3-1.A.2.g and 3.3.3-1.B.2.f list minimum operable channels per trip system as 1/valve. Per plant design this should be 2/system.

2. Safety Significance: Since ECCS manual initiation logic (ADS) requires two manual initiation inputs, the plant could be operated in a non-conservative condition.

3. Anticipated Resolution: Change Tech Spec Table 3.3.3-1 to reflect as built design of plant.

4. NRC Response to Item (NRR/IE):

NRC Notified: Individual Notified   Date / Time

5. Disposition:

Items Closed: (How)

Date / Time

cc: J. E. Cross
R. F. Rogers

"TECH SPEC PRIORITY"

Punchlist Item # 213

Tech Spec 3/4.3.3

Priority 1A

TO: Manager of Nuclear Plant Engineering

FROM: Chairman, Prioritization and Disposition Chairman

SUBJECT: Technical Specifications Punchlist Item # 213

PDTS:84/ 0010

DATE: 3/10/84

The subject Tech Spec item has been determined by the Disposition Committee to require Engineering support.

DETAILS: Verify that the min. operable channels listed in T.S. Table 3.3.3-1, items A.2.g & B.2.f should be "2/system" rather than "1/valve".

Please contact Joe Hendry at Extension 2675 for further information.

Please refer to the Tech Spec Punchlist item number in your response. Forward your response to G. Zinke.

J. C. Roberts
Chairman

LLJ/JCR:swb

cc: Mr. C. L. Tyrone
Mr. J. E. Cross
Mr. D. Stonestreet
Mr. A. S. McCurdy
Mr. S. Hutchins
Mr. J. Hendry
File (Tech Spec Records)

A4/61swb1

"TECH SPEC PRIORITY"

Punchlist Item # See ATTached

Tech Spec See ATTached

Priority See ATTached

TO: Manager of Nuclear Plant Engineering

FROM: Chairman, Prioritization and Disposition Chairman

SUBJECT: Technical Specifications Punchlist Item # See ATTached

PDTS:84/ 0014

DATE: 3/10/84

The subject Tech Spec item has been determined by the Disposition Committee to require Engineering support.

DETAILS: This letter identifies requested response dates for the following Tech Spec problems:

# 199	Letter No. PDTS 84/0001	# 015	Letter No. PDTS 84/0007
# 180	Letter NO. PDTS 84/0002	# 198	Letter No. PDTS 84/0008
# 033	Letter NO. PDTS 84/0003	# 202	Letter NO. PDTS 84/0009
# 054	Letter NO. PDTS 84/0004	# 213	Letter No. PDTS 84/0010
# 001	Letter NO. PDTS 84/0005	# 219	Letter No. PDTS 84/0011
# 016	Letter NO. PDTS 84/0006	# 1[illegible]8	Letter NO. PDTS 84/0013

It is requested that the responses to the above items be completed by 3/13/84

Please contact Jerry RoberTs at Extension 2695 for further information.

Please refer to the Tech Spec Punchlist item number in your response. Forward your response to George ZINKE.

J. C. Roberts
Chairman

LLJ/JCR:swb
cc: Mr. C. L. Tyrone
Mr. J. E. Cross
Mr. D. Stonestreet
Mr. A. S. McCurdy
Mr. S. Hutchins
Mr. J. Hendry
File (Tech Spec Records)

A4/61swb1

Tech Spec Problem No.	Tech Spec	Priority
199	Table 3.3.6-1.5	1B
180	4.8.4.3	1D
033	Table 3.3.8-2	1B
054	3/4.3.8	1B
001	3/4.5.1	1B
016	3/4.3.8	1B
015	3/4.3.2	1D
198	3/4.3.7	1B
202	3/4.3.7	1B
213	3/4.3.3	1B
219	Figure 3.4.6.1-1	1B
168	3.6.3.1	1B

"TECH SPEC PRIORITY"

MEMO TO: J. F. Pinto, Manager of Nuclear Plant Engineering

FROM: C. L. Tyrone, Project Manager

SUBJECT: Handling of Tech Spec Review Items

TSRO: 84/0001

DATE: March 11, 1984

This memorandum confirms our conversation of March 10, 1984. At that time, your assistance was requested in resolving discrepancies on eleven priority 1 items. Since then two items have been added. These items are all previously identified items which require early resolution with the NRC. A response is needed on these items by 12:30 PM on March 12, 1984. A list is attached.

Furthermore, all items of any priority identified (or previously known) which are being handled on this program require expeditious handling. This includes areas where requests are originated from other interfacing organizations such as the Plant or Nuclear Services. In any case where conflicts regarding highest priority is not clear, I am available to provide clarification.

It is suggested that you arrange 7 day a week support in this area as it is needed and arrange for all NPE personnel who will be involved in this effort to be available (or on call) in a manner that will support the Tech Spec Review program.

CLT

SHH:sad
Attachment

cc: J. B. Richard (w/a
J. P. McGaughy (w/a)
J. F. Pinto (w/a)
J. E. Cross (w/a)
T. H. Cloninger (w/a)
H. J. Green (w/a)
R. C. Fron (w/a)
D. W. Stonestreet (w/a)
[redacted]
T. E. Reaves, Jr. (w/a)
S. M. Feith (w/a)
J. G. Cesare (w/a)
G. W. Smith (w/a)
L. R. McKay (w/a)
L. C. Burgess (w/a)
File (Tech Spec Records) (w/a)

LIST OF CURRENT PRIORITY 1
ITEMS REQUIRING NPE SUPPORT

PDTS:84/	P/L #	Date Sent
001	199	3/10/84
002	180	3/10/84
003	033	3/10/84
004	054	3/10/84
005	001	3/10/84
006	016	3/10/84
007	015	3/10/84
008	198	3/10/84
009	202	3/10/84
010	213	3/10/84
011	219	3/10/84
012	083	3/10/84
013	168	3/10/84

DEFINITIONS FOR
"CHANNELS", "TRIP SYSTEMS", AND "TRIP FUNCTIONS"
FOR EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION TABLE 3.3.3-1

RHR and LPCS Trip Systems

Trip Unit	Parameter	Logic	
B21-LIS-N691A(1)	RPV Level - Lvl 1		
B21-LIS-N691E(1)	RPV Level - Lvl 1	One out of Two Twice	Initiate LPCS
B21-PIS-N694A *(1)	Drywell Pressure - HI		
B21-PIS-N694E (1)	Drywell Pressure - HI	Either	Initiate RHR A (LPCI MODE) TRIP FUNCTION
E12-HS-M617	Manual Initiation		
E12-K70A	LPCI Pump A Start Time Delay Relay	Both	
		TRIP SYSTEM	
B21-LIS-N691B(1)	RPV Level - Lvl 1		Initiate RHR C (LPCI Mode)
B21-LIS-N691F(1)	RPV Level - Lvl 1	One out of Two Twice	
B21-PIS-N694B(1)	Drywell Pressure - HI		
B21-PIS-N694F(1)	Drywell Pressure - HI	Either	
E12-HS-M617	Manual Initiation		
E12-K70B	LPCI Pump B Start Time Delay Relay	Both	Initiate RHR B (LPCI Mode) TRIP FUNCTION
		TRIP SYSTEM	

* One Channel (typical of 12 shown for RHR and LPCS)
(1) Also actuates the associated division diesel generator.

ADS Trip Systems
ADS "A"

Trip Unit	Parameter	Logic	
B21-LIS-N691A	RPV Level - Lvl 1	Three Out of three	
B21-LIS-N695A *	RPV Level - Lvl 3		
B21-PIS-N694A	Drywell Pressure - HI	Both	
B21-K5A	ADS Timer (105 Seconds)	Either	
B21-HS-M629A	Manual Initiation		
E21-PIS-N652	LPCS Pump Disch Pr. - HI	Both	
E12-PIS-N655A	LPCI Pump A Disch. Pr. - HI	Either	Energize Pilot Sol. A TRIP FUNCTION
B21-LIS-N691E	RPV Level - Lvl 1	Both	
B21-PIS-N694E	Drywell Pressure - HI	Both	
B21-HS-M629E	Manual Initiation	Either	
E21-PIS-N653	LPCS Pump Disch. Pr. - HI	Both	
E12-PIS-N656A	LPCI Pump A Disch. Pr. - HI	Either	
		TRIP SYSTEM	

* One Channel (Typical of 12 shown for ADS "A")

DEFINITIONS FOR
"CHANNELS", "TRIP SYSTEMS", AND "TRIP FUNCTIONS"
FOR EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION TABLE 3.3.3-1 (Continued)

ADS "B"

Trip Unit	Parameter	Logic				
B21-LIS-N691B	RPV Level - Lvl 1	Three Out of three	Both	Either	Both	Both — Energize Pilot Sol. B TRIP FUNCTION
B21-LIS-N695B *	RPV Level - Lvl 3					
B21-PIS-N694B	Drywell Pressure - HI					
B21-K5B	ADS Timer (105 Seconds)					
B21-HS-M629B	Manual Initiation					
E12-PIS-N655B	LPCI Pump B Disch Pr. - HI	Either				
E12-PIS-N655C	LPCI Pump C Disch. Pr. - HI					
B21-LIS-N691F	RPV Level - Lvl 1	Both	Either	Both		
B21-PIS-N694F	Drywell Pressure - Hi					
B21-HS-M629F	Manual Initiation					
E12-PIS-N656B	LPCI Pump B Disch. Pr. - HI	Either				
E12-PIS-N656C	LPCI Pump C Disch. Pr. - HI					

TRIP SYSTEM

* One Channel (Typical of 12 shown for ADS "B")

HPCS System

Trip Unit	Parameter	Logic		
E22-HS-M616	Manual Initiation		Any One	Initiates HPCS
B21-LIS-N673C	RPV Level - Lvl 2	One out of two Twice		
B21-LIS-N673G	RPV Level - Lvl 2 *			
B21-LIS-N673L	RPV Level - Lvl 2			
B21-LIS-N673R	RPV Level - Lvl 2			
B21-PIS-N667C	Drywell Pressure HI	One out of Two Twice		
B21-PIS-N667G	Drywell Pressure HI			
B21-PIS-N667L	Drywell Pressure HI			
B21-PIS-N667R	Drywell Pressure HI			
B21-LS-N674C	RPV Level - Lvl 8	Two Out of Two		Close HPCS Pump Disch Valve
B21-LS-N674L	RPV Level - Lvl 8			
E22-LIS-N654C	Cond Stg Tk Lvl - Low		Any One	Switch HPCS Pump Suction from CST to Supp. Pool
E22-LIS-N654G	Cond Stg Tk Lvl - Low			
E22-LIS-N655C	Supp. Pool Wtr Lvl - High			
E22-LIS-N655G	Supp. Pool Wtr Lvl - High			

TRIP FUNCTION

* One Channel (Typical of 15 shown for HPCS)

TECHNICAL SPECIFICATION PROBLEM SHEET

Item Number: 235 Priority: 1A

Ron Davis /
Identified By Date

Responsible Supervisor

Tech Spec Reference: 4.6.1.3.a

Problem Title: Containment Air Locks

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other):
The 72 hour time frame given is a 10CFR 50 Appendix J requirement A # sign should be added to this time frame since 4.0.2 can not apply

2. Safety Significance:

3. Anticipated Resolution:
Add # sign to 72 hours, submit Tech Spec change.
Cou'd use Position Statement and programaticly control surveillance frequency.

4. NRC Response to item (NRR/IE): ___

NRC Notified: ___ Individual Notified / Date / Time

5. Disposition: ___

Items Closed: (How) ___

/
Date Time

cc: J. E. Cross
R. F. Rogers

D. L. CLARIDGE

SUBJECT: TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT 4.6.1.3.a, PAGE 3/4 6-6

NO. 235

DISCUSSION: THIS CHANGE PROPOSES TO ADD A "#" AFTER "WITHIN 72 HOURS" AT THE BEGINNING OF ~~STATEMENT~~ OF TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT 4.6.1.3.a, IN ORDER TO MAKE THE FOOTNOTE APPLICABLE.

JUSTIFICATION: THE REQUIREMENT, AS PRESENTLY WORDED, IMPLIES THAT SPECIFICATION 4.0.2 IS APPLICABLE, WHICH WOULD ALLOW UP TO A 25% EXTENSION OF THE 72 HOUR TIME ALLOTMENT TO DEMONSTRATE OPERABILITY OF THE CONTAINMENT AIR LOCK(S). HOWEVER, 10CFR 50 APPENDIX J ALLOWS ONLY 72 HOURS FOR THIS ACTION, MAKING REWORDING OF THE SPECIFICATION NECESSARY TO COMPLY WITH 10 CFR 50. THE ADDITION OF THE FOOTNOTE AT THE INDICATED LOCATION WOULD PROVIDE THE NECESSARY RESTRICTION WITHOUT REWRITING THE SPECIFICATION.

SIGNIFICANT HAZARDS CONSIDERATION:

THIS CHANGE LIMITS THE TIME ALLOWED TO DEMONSTRATE OPERABILITY OF THE CONTAINMENT AIR LOCKS TO WITHIN 72 HOURS AFTER EACH CLOSING. THIS CHANGE CONSTITUTES AN ADDITIONAL LIMITATION NOT PRESENTLY INCLUDED IN THE TECHNICAL SPECIFICATIONS, AND PROVIDES COMPLIANCE WITH THE REQUIREMENTS OF 10 CFR 50 APPENDIX J. THE ADDITION OF THIS FOOTNOTE DOES NOT REDUCE THE TECHNICAL CONTENT OF THE TECHNICAL SPECIFICATION NOR DOES IT a) REDUCE THE MARGIN OF SAFETY, b) INCREASE THE PROBABILITY OR CONSEQUENCES OF A PREVIOUSLY EVALUATED ACCIDENT OR c) CREATE THE POSSIBILITY OF A NEW OR DIFFERENT KIND OF ACCIDENT. THUS THE PROPOSED CHANGE TO THE TECHNICAL SPECIFICATION DOES NOT INVOLVE A SIGNIFICANT HAZARDS CONSIDERATION.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours# after each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage rate less than or equal to 2 scf per hour when the gap between the door seals is pressurized to Pa, 11.5 psig.

b. By conducting an overall air lock leakage test at P_a, 11.5 psig, and verifying that the overall air lock leakage rate is within its limit:

1. At least once per 6 months#, and

2. Prior to establishing PRIMARY CONTAINMENT INTEGRITY when maintenance has been performed on the air lock that could affect the air lock sealing capability.*

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

d. By verifying each airlock door inflatable seal system OPERABLE by:

1. Demonstrating each of the two inflatable seal pressure instrumentation channels per airlock door OPERABLE by performance of a:

a) CHANNEL FUNCTIONAL TEST at least once per 31 days, and

b) CHANNEL CALIBRATION at least once per 18 months,

with a low pressure setpoint of $\geq$ 60 psig.

2. At least once per 7 days, verifying seal air flask pressure to be greater than or equal to 60 psig.

3. At least once per 18 months, conducting a seal pneumatic system leak test and verifying that system pressure does not decay more than 2 psig from 90 psig within 48 hours.

#The provisions of Specification 4.0.2 are not applicable.

*Exemption to Appendix J of 10 CFR 50.

TECHNICAL SPECIFICATION PROBLEM SHEET

Item Number: 243 Priority: 1B

/

Identified By Date Responsible Supervisor

Tech Spec Reference: 4.4.7, 4.6.4.3

Problem Title: MSIU Stroke Time Definition

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other):

ASME code IWV-3413 defines full stroke "from initiation of the actuating signal to the end of the actuating cycle." G.E. design spec states "valve closing time (100% valve travel)" FSAR (large pipe break analysis) gives 5.5 sec. (.5 sec. for actuation delay) for MSIV closure. Should value timing include actuation delays (i e., time from handswitch in CLOSE until valve starts CLOSED.)

2. Safety Significance:

3. Anticipated Resolution:

Resolve definition of valve stroke (ASME IWV-3413 vs. G.E. design spec vs. FSAR).

4. NRC Response to Item (NRR/IE):

NRC Notified: /

Individual Notified Date Time

5. Disposition:

Items Closed: (How)

/

Date Time

cc: J. E. Cross

R. F. Rogers

Rev. 2, 3/10/84

I/121swb95

Item No. 243

TECHNICAL SPECIFICATION PROBLEM SHEET

Priority 1A

Identified By / Date

Responsible Supervisor

Tech Spec Reference: 4.4.7/4.6.4.3

Problem Title: MSIV stroke time definition

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other): ASME code IWV-3413 defines full stroke "from initiation of the actuating signal to the end of the actuating cycle. GE design spec states "valve closing time (100 percent valve travel)" FSAR (large pipe break analysis) gives 5.5 sec (.5 sec. for actuation delay) for MSIV closure. Should valve timing include actuation delays (ie time from handswitch in CLOSE until valve starts closed.)

2. Safety Significance:

3. Anticipated Resolution: Resolve definition of valve stroke (ASME IWV-3413 vs GE design spec vs FSAR)

4. NRC Response to Item (NRR/IE):

NRC Notified: Individual Notified / Date / Time

5. Disposition:

Items Closed: (How)

Date / Time

cc: J. E. Cross
R. F. Rogers

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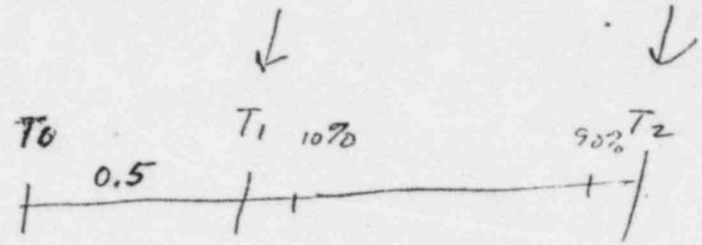


3 sec.

4 sec. → 3 sec.

ISI $T_0 \rightarrow T_2$

Tech Spec Bases $T_1 \rightarrow T_2$ 3 - 5 sec

$T_0 \rightarrow T_1 \approx 0.5$ sec.

ARTICLE IWV-3000
TEST REQUIREMENTS

IWV-3100 PRESERVICE TESTS

Each valve, after installation and prior to service, shall be tested as required by this Subsection. These tests shall be conducted under conditions similar to those to be experienced during subsequent inservice tests. Safety and relief valves which will be removed and bench tested during subsequent inservice tests need not be installed prior to the preservice test.

IWV-3200 VALVE REPLACEMENT, REPAIR, AND MAINTENANCE

When a valve or its control system has been replaced or repaired or has undergone maintenance[1] that could affect its performance, and prior to the time it is returned to service, it shall be tested to demonstrate that the performance parameters which could be affected by the replacement, repair, or maintenance are within acceptable limits.

IWV-3300 VALVE POSITION INDICATOR VERIFICATION

Valves with remote position indicators shall be observed at least once every 2 years to verify that valve operation is accurately indicated.

[1]Adjustment of stem packing, removal of the bonnet, stem assembly, or actuator, and disconnection of hydraulic or electrical lines are examples of maintenance that could affect valve performance parameters.

IWV-3400 INSERVICE TESTS, CATEGORY A AND B VALVES

IWV-3410 VALVE EXERCISING TEST

IWV-3411 Test Frequency

Category A and B valves shall be exercised at least once every 3 months, except as provided by IWV-3412(a), IWV-3415, and IWV-3416.

IWV-3412 Exercising Procedure

(a) Valves shall be exercised to the position required to fulfill their function unless such operation is not practical during plant operation. If only limited operation is practical during plant operation, the valve shall be part-stroke exercised during plant operation and full-stroke exercised during cold shutdowns. Valves that cannot be exercised during plant operation shall be specifically identified by the Owner and shall be full-stroke exercised during cold shutdowns. Full-stroke exercising during cold shutdowns for all valves not full-stroke exercised during plant operation shall be on a frequency determined by the intervals between shutdowns as follows: for intervals of 3 months or longer, exercise during each shutdown; for intervals of less than 3 months, full-stroke exercise is not required unless 3 months have passed since last shutdown exercise.

(b) The necessary valve disk movement shall be determined by exercising the valve while observing an appropriate indicator which signals the required change of disk position, or observing indirect evidence, such as changes in system pressure, flow rate, level, or temperature, which reflect stem or disk position.

IWV-3413 Power Operated Valves

(a) The limiting value of full-stroke time of each power operated valve shall be specified by the Owner. Full-stroke time is that time interval from initiation of

the actuating signal to the end of the actuating cycle.

(b) The stroke time of all power operated valves shall be measured to the nearest second, for stroke times 10 sec or less, or 10% of the specified limiting stroke time for full-stroke times longer than 10 sec whenever such a valve is full-stroke tested.

IWV-3414 Valves in Regular Use

Valves which operate in the course of plant operation at a frequency which would satisfy the exercising requirements of this Subsection need not be additionally exercised, provided that the observations otherwise required for testing are made and analyzed during such operation and are recorded in the plant record at intervals no greater than specified in IWV-3411.

IWV-3415 Fail-Safe Valves

When practical, valves with fail-safe actuators shall be tested by observing the operation of the valves upon loss of actuator power. If these valves cannot be tested once every 3 months, they shall be tested during each cold shutdown; in case of frequent cold shutdowns, these valves need not be tested more often than once every 3 months.

IWV-3416 Valves in Systems Out of Service

For a valve in a system declared inoperable or not required to be operable, the exercising test schedule need not be followed. Within 30 days prior to return of the system to operable status, the valves shall be exercised and the schedule resumed in accordance with requirements of this Article.

IWV-3417 Corrective Action

(a) If, for power operated valves, an increase in stroke time of 25% or more from the previous test for valves with full-stroke times greater than 10 sec or 50% or more for valves with full-stroke times less than or equal to 10 sec is observed, test frequency shall be increased to once each month until corrective action is taken, at which time the original test frequency shall be resumed. In any case, any abnormality or erratic action shall be reported.

(b) If a valve fails to exhibit the required change of valve stem or disk position or exceeds its specified limiting value of full-stroke time by this testing, then corrective action shall be initiated immediately. If the condition is not, or cannot be, corrected within 24 hr, the valve shall be declared inoperative. When corrective action is required as a result of tests made during cold shutdown, the condition shall be corrected before startup. A retest showing acceptable operation shall be run following any required corrective action before the valve is returned to service.

IWV-3420 VALVE LEAK RATE TEST

IWV-3421 Scope

Category A valves shall be leak tested except that valves which function in the course of plant operation in a manner that demonstrates functionally adequate seat tightness need not be leak tested. In such cases, the valve record shall provide the basis for the conclusion that operational observations constitute satisfactory demonstration.

IWV-3422 Frequency

Tests shall be conducted at least once every 2 years.

IWV-3423 Differential Test Pressure

Valve seat leakage tests shall be made with the pressure differential in the same direction as when the valve is performing its function, with the following exceptions.

(a) Globe-type valves may be tested with pressure under the seat.

(b) Butterfly valves may be tested in either direction, provided their seat construction is designed for sealing against pressure on either side.

(c) Gate valves with two-piece disks may be tested by pressurizing them between the seats.

(d) Valves (except check valves) may be tested in either direction if the function differential pressure is 15 psi (100 kPa) or less.

(e) Leakage tests involving pressure differentials lower than function pressure differentials are permitted in those types of valves in which service pressure will tend to diminish the overall leakage channel opening, as by pressing the disk into or onto the seat with greater force. Gate valves, check valves, and globe-type valves, having function pressure differential applied over the seat, are examples of valve applications satisfying this requirement. When leakage tests are made in such cases using pressures lower than function maximum pressure differential, the observed

TECHNICAL SPECIFICATION PROBLEM SHEET

Item Number: 001 Priority: ~~1A~~ 1B

_____ / _____ _____
Identified By Date Responsible Supervisor

Tech Spec Reference: 3/4.5.1

Problem Title: ADS Valves Operability Requirement

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other): Tech Spec requires at least 7 operable ADS valves in Div. 1 & 2. However, the bases indicate that ADS controls only 7 selected SRVs, while the FSAR only takes credit for 6. Thus allowing one valve out of service for 14 days. It appears the tech specs should require 8 operable ADS valves, since ADS controls 8 valves not 7 and the FSAR assumes loss of one valve.

2. Safety Significance: Tech Specs is non conservative since it requires only 7 operable valves versus 8.

3. Anticipated Resolution: Change tech specs for required operable ADS valves from 7 to 8. After review of FSAR Chap. 15 Accident Analysis.
Could use Tech Spec position statement

4. NRC Response to Item (NRR/IE): _____

NRC Notified: _____ _____ / _____
Individual Notified Date Time

5. Disposition: _____

Items Closed: (How) _____

_____ / _____
Date Time

cc: J. E. Cross
R. F. Rogers

P/L #001

TECHNICAL SPECIFICATION REVIEW SHEET

LEAD REVIEW BY:

☒ GE ☐ BPC ☐ R&ES ☐ ADMIN

TECHNICAL SPECIFICATION: 3.5.1 REF PROB SHEET NO. 001
SYSTEM/SUBJECT: ECCS, ADS VALVE OUT OF SERVICE ANALYSIS
BASES SUBSECTION: 3/4.5
PARAMETER: - NUMBER OF ADS VALVES
ASSOCIATED DOCUMENTS REVIEWED:
FSAR(AMEND 57) 6.3.3
DESIGN DOCUMENTS/REV: N/A
RELATED TECH SPEC SECTIONS: N/A
SER(SUPP.4): N/A
OTHER: REF. T.S. PROBLEM SHEET ITEM NO. 001 (attached

CONCLUSIONS: It is acceptable for an ADS valve to be out of service for an extended period of time and still meet acceptance criteria of 10 CFR 50.46

☒ Attached Sheets (2) A. TECH SPEC & BASES REVISION.
(21) B. PROPOSED FSAR REVISIONS.

RECOMMENDED ACTION: Revise FSAR and Tech Spec & Bases per attached sheets.

☐ Attached Sheets ( )

Reviewed by B. F. Shoopak, LCE

BSE [signature] 3/9/84
Signature/Date

ONSITE REVIEW
[] Concur
[] Do Not Concur
☐ Recommended Action: __________

☐ Attached Sheets ( )

Bechtel/GE Systems Engineer/Date

MP&L NPE Engineer/Date

MP&L SRO/Date

NSSS/BOP Manager/Date

ATTACHMENT: TECHNICAL SPECIFICATION REVIEW CHECKLIST.

TECHNICAL SPECIFICATION PROBLEM SHEET

Item Number: 001 Priority: 1A

_______________ / _______________
Identified By Date Responsible Supervisor

Tech Spec Reference: 3/4.5.1

Problem Title: ADS Valves Operability Requirement

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other): Tech Spec requires at least 7 operable ADS valves in Div. 1 & 2. However, the bases indi- that ADS controls only 7 selected SRVs, while the FSAR only takes credit for 6. Thus allowing one valve out of service for 14 days. It appears the tech specs should require 8 operable ADS valves, since ADS controls 8 valves not 7 and the FSAR assumes loss of one valve.

2. Safety Significance: Tech Specs is non conservative since it requires only 7 operable valves versus 8.

3. Anticipated Resolution: Change tech specs for required operable ADS valves from 7 to 8. After review of FSAR Chap. 15 Accident Analysis.

4. NRC Response to Item (NRR/IE):

NRC Notified: _______________ _______________ / _______________
Individual Notified Date Time

5. Disposition:

Items Closed: (How)

_______________ / _______________
Date Time

cc: J. E. Cross
R. F. Rogers

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ECCS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.1 ECCS divisions 1, 2 and 3 shall be OPERABLE with:

a. ECCS division 1 consisting of:

1. The OPERABLE low pressure core spray (LPCS) system with a flow path capable of taking suction from the suppression pool and transferring the water through the spray sparger to the reactor vessel.

2. The OPERABLE low pressure coolant injection (LPCI) subsystem "A" of the RHR system with a flow path capable of taking suction from the suppression pool and transferring the water to the reactor vessel.

3. At least ~~7~~ 8 OPERABLE ADS valves.

b. ECCS division 2 consisting of:

1. The OPERABLE low pressure coolant injection (LPCI) subsystems "B" and "C" of the RHR system, each with a flow path capable of taking suction from the suppression pool and transferring the water to the reactor vessel.

2. At least ~~7~~ 8 OPERABLE ADS valves.

c. ECCS division 3 consisting of the OPERABLE high pressure core spray (HPCS) system with a flow path capable of taking suction from the suppression pool and transferring the water through the spray sparger to the reactor vessel.

APPLICABILITY: OPERATIONAL CONDITION 1, 2*# and 3*.

ACTION:

a. For ECCS division 1, provided that ECCS divisions 2 and 3 are OPERABLE:

1. With the LPCS system inoperable, restore the inoperable LPCS system to OPERABLE status within 7 days.

2. With LPCI subsystem "A" inoperable, restore the inoperable LPCI subsystem "A" to OPERABLE status within 7 days.

3. With the LPCS system inoperable and LPCI subsystem "A" inoperable, restore at least the inoperable LPCI subsystem "A" or the inoperable LPCS system to OPERABLE status within 72 hours.

4. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

*The ADS is not required to be OPERABLE when reactor steam dome pressure is less than or equal to 135 psig.

#See Special Test Exception 3.10.5.

ECCS-OPERATING and SHUTDOWN (Continued)

into the reactor, but no credit is taken in the safety analyses for the condensate storage tank water.

With the HPCS system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified automatic depressurization system and both the LPCS and LPCI systems. In addition, the reactor core isolation cooling (RCIC) system, a system for which no credit is taken in the safety analysis, will automatically provide makeup at reactor operating pressures on a reactor low water level condition. The HPCS out-of-service period of 14 days is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems.

The surveillance requirements provide adequate assurance that the HPCS system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test with reactor vessel injection requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

Upon failure of the HPCS system to function properly after a small break loss-of-coolant accident, the automatic depressurization system (ADS) automatically causes selected safety-relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds 135 psig even though low pressure core cooling systems provide adequate core cooling up to 350 psig.

ADS automatically controls ~~seven~~ eight selected safety-relief valves although the safety analysis only takes credit for ~~six~~ seven valves. It is therefore appropriate to permit one valve to be out-of-service for up to 14 days without materially reducing system reliability.

3/4.5.3 SUPPRESSION POOL

The supression pool is required to be OPERABLE as part of the ECCS to ensure that a sufficient supply of water is available to the HPCS, LPCS and LPCI systems in the event of a LOCA. This limit on suppression pool minimum water volume ensures that sufficient water is available to permit recirculation cooling flow to the core. The OPERABILITY of the suppression pool in OPERATIONAL CONDITIONS 1, 2 or 3 is required by Specification 3.6.3.1.

Repair work might require making the suppression pool inoperable. This specification will permit those repairs to be made and at the same time give assurance that the irradiated fuel has an adequate cooling water supply when the suppression pool must be made inoperable, including draining, in OPERATIONAL CONDITION 4 or 5.

In OPERATIONAL CONDITION 4 and 5 the suppression chamber minimum required water volume is reduced because the reactor coolant is maintained at or below 200°F. Since pressure suppression is not required below 212°F, the minimum required water volume is based on NPSH, recirculation volume, and vortex prevention plus a 1'2" safety margin for conservatism.

TABLE 6.3-2 (Cont.)

o	Initiating signals low-low-low water level or high drywell pressure	ft. above top of active fuel psig	≥1.0 ≤2.0
o	Maximum allowed (runout) flow	GPM	9100
o	Maximum allowed delay time from initiating signal to pump at rated speed	sec	27.0
o	Injection valve fully open	sec after DBA	≤40.0
	High-Pressure Core Spray		
o	Vessel pressure at which flow may commence	psid	1177
o	Minimum flow available at vessel to pump suction head		See Figure 6.3-3
o	Initiating signals low-low water level or high drywell pressure	ft. above top of active fuel psig	≥10.5 ≤2.0
o	Maximum allowed (runout) flow	GPM	9100
o	Maximum allowed delay time from initiating signal to rated flow available and injection valve wide open	sec	27.0
	Automatic Depressurization System		
o	Total number of valves installed		8
o	Number of valves used in analysis		8 (1)
o	Minimum Flow Capacity of 8 valves at vessel pressure	lb/hr psid (vessel suppression pool)	6.4×10^6 1125

NOTE (1) Additional LOCA analyses with 7 ADS valves in section 6.3.3.8 justify 1 ADS valve out of service for an extended period of time.

Amend. 48 6/81

insert new section 6.3.3.8

d. Peak cladding temperature as a function of time from REFLOOD

The same variables resulting from the analysis of a less limiting small break are shown in Figures 6.3-51 through 6.3-54.

6.3.3.7.7 Calculations for Other Break Locations

Reactor water level and vessel pressure from SAFE/REFLOOD and peak cladding temperature and convective heat transfer coefficients from REFLOOD are shown in Figures 6.3-55 through 6.3-58 for the HPCS line break, Figure 6.3-59 through 6.3-62 for the feedwater line break, and in Figures 6.3-63 through 6.3-66 for the main steam line break inside the containment. 46

An analysis was done for the main steam line break outside the containment. Reactor water level and vessel pressure from SAFE/REFLOOD and peak cladding temperature and convective heat transfer coefficients from REFLOOD are shown in Figures 6.3-73 through 6.3-76. 46

6.3.3.~~8~~9 LOCA Analysis Conclusions

Having shown compliance with the applicable acceptance criteria of subsection 6.3.3.2, it is concluded that the ECCS will perform its function in an acceptable manner and meet all of the 10 CFR 50.46 Acceptance Criteria, given operation at or below the maximum average planar linear heat generation rates in Table 6.3-6.

6.3.4 Tests and Inspections

6.3.4.1 ECCS Performance Tests

All systems of the ECCS are tested for their operational ECCS function during the pre-operational and/or startup test program. Each component is tested for power source, range, direction of rotation, set point, limit switch setting, torque switch setting, etc. Each pump is tested for flow capacity for comparison with vendor data. (This test is also used to verify flow measuring capability). The flow tests involve the same suction and discharge source; i.e., suppression pool or condensate storage tank.

All logic elements are tested individually and then as a system to verify complete system response to emergency signals including the ability of valves to revert to the ECCS alignment from other positions.

Finally the entire system is tested for response time and flow capacity taking suction from its normal source and delivering flow into the reactor vessel. This last series of tests is performed with power supplied from both offsite power and onsite emergency power.

6.3.3.8 ADS Valve Out of Service Calculations

The purpose of this section is to quantify the effect on the LOCA calculations of an ADS valve out of service.

As demonstrated in the previous sections, for small breaks the unavailability of the High Pressure Core Spray (HPCS) system (as a result of the break or an assumed single failure) will result in the highest calculated peak cladding temperature (PCT). For these cases the emergency core cooling systems remaining include the Automatic Depressurization System (ADS) and some low pressure ECCS. Here the ADS is required to rapidly depressurize the vessel below the shutoff head of the low pressure ECCS.

If an ADS valve is out of service in addition to the assumed worst single failure, the ADS will depressurize the vessel slower. This will result in a delay of low pressure ECCS and, in general, a corresponding delay in reflooding time and increase in the PCT. However, the significance of the ADS decreases as larger break sizes are considered because of the increasing depressurization due to mass loss through the break. Therefore, the maximum impact on the PCT due to an ADS valve out of service will be determined by the recalculation of the small break spectrum.

Figure 6.3-77 is a comparison between the two break spectrum calculations for recirculation line breaks. The increase in PCT for an ADS valve out of service is shown to be about 115°F with a maximum PCT of 1477°F occurring at a break size of 0.09 ft². Reactor water level, vessel pressure, heat transfer coefficients and peak cladding temperature versus time for this limiting case are shown in Figures 6.3-78 through 6.3-81.

The maximum core spray line break with an LPCS diesel generator failure was also reevaluated with an ADS valve out of service. For this case reactor water level, vessel pressure, heat transfer coefficients and peak cladding temperature versus time are shown in Figures 6.3-82 through 6.3-85. The increase in PCT for an ADS valve out of service is shown to be 95°F yielding a PCT for this case of 1784°F.

FIGURE 6.3-77

PEAK CLADDING TEMPERATURE VERSUS BREAK AREA

GRAND GULF

RECIRCULATION SUCTION SMALL BREAK, HPCS FAILURE

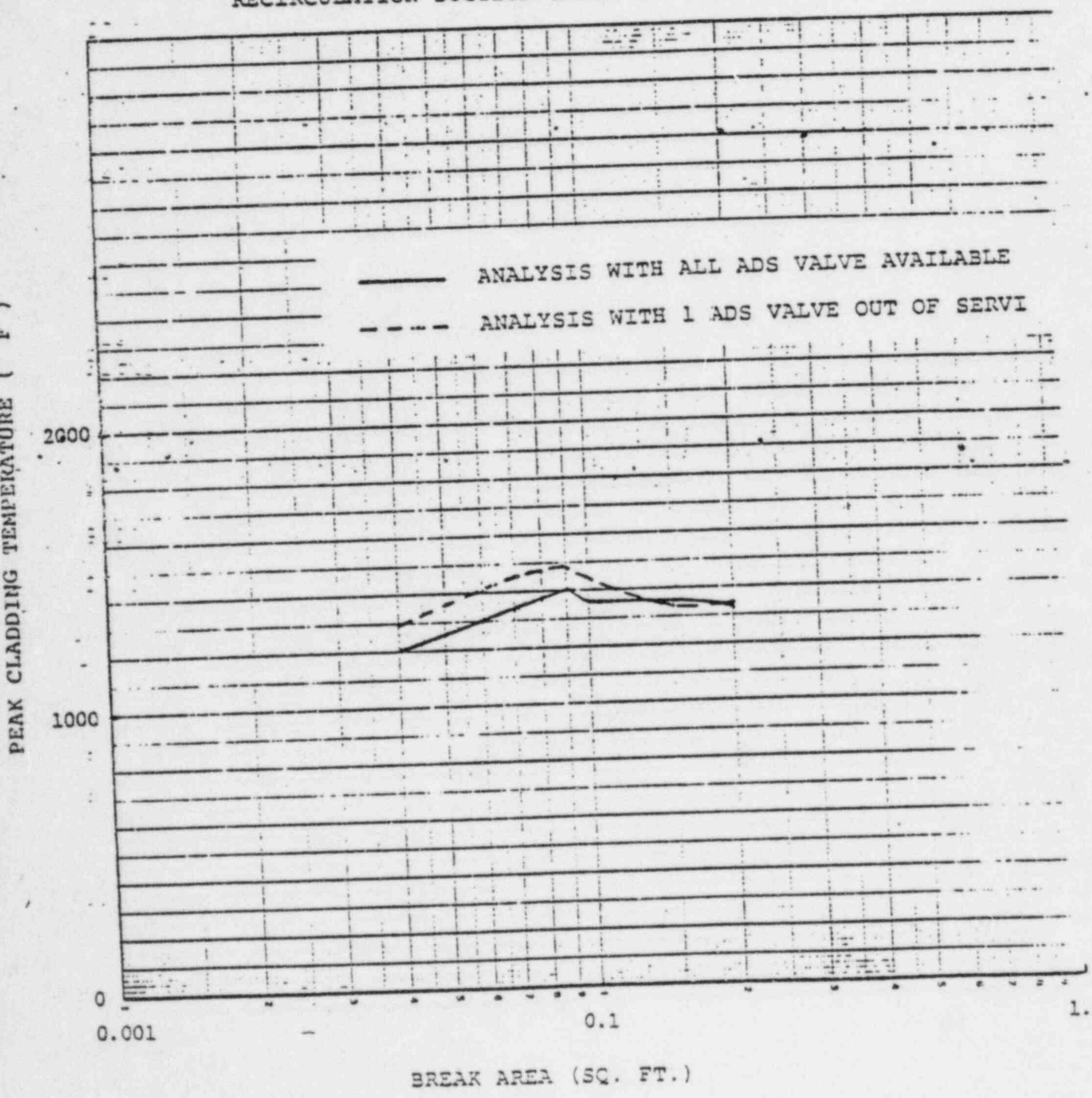


FIGURE 6.3-78

CONVECTIVE HEAT TRANSFER COEFFICIENT VERSUS TIME AFTER BREAK

GRAND GULF

.09 SQ. FT. BREAK (HIGHEST TEMPERATURE SMALL BREAK), RECIRCULATION SUCTION BREAK, HPCS FAILURE

WITH 1 ADS VALVE OUT OF SERVICE

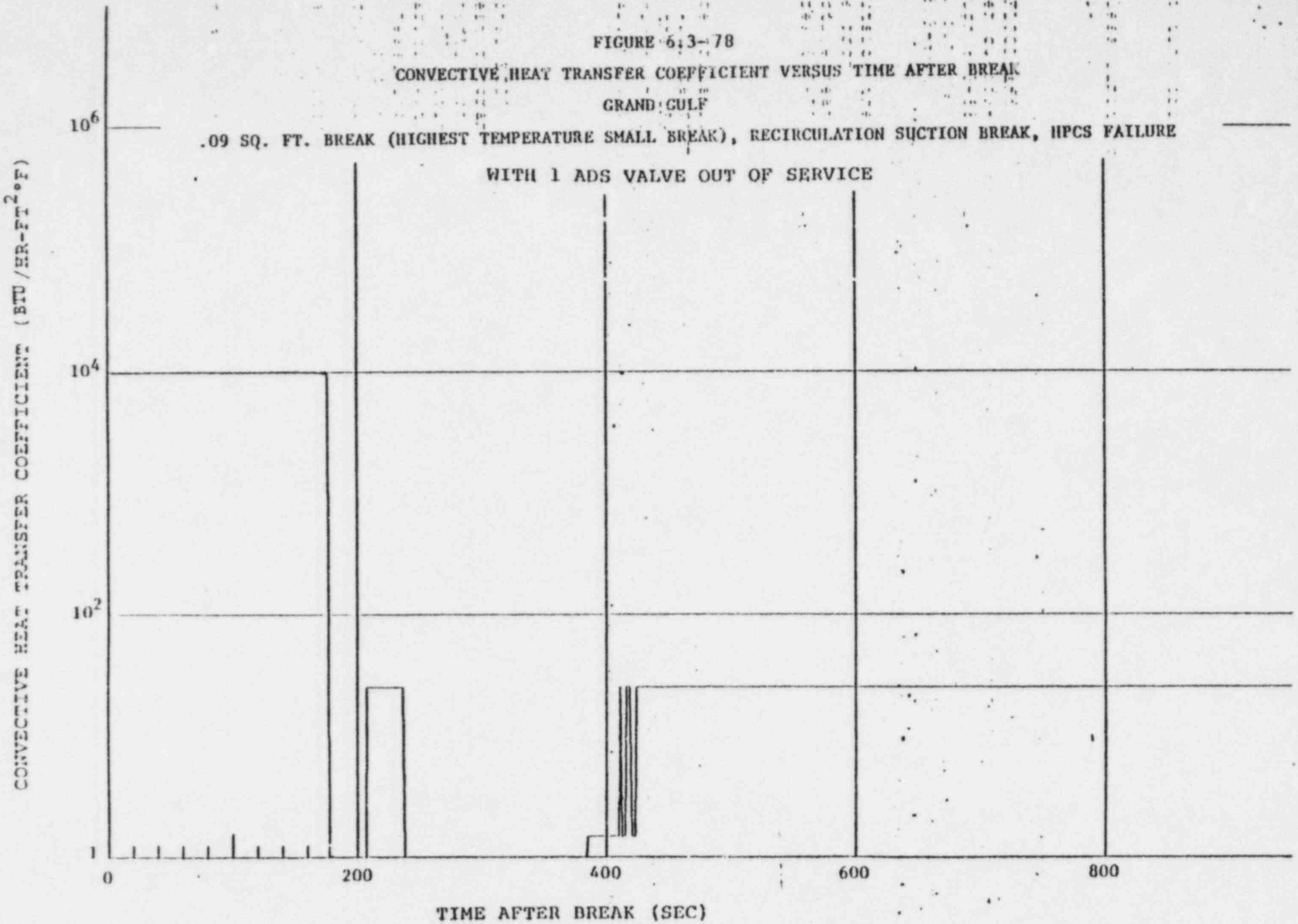


FIGURE 6.3-79

WATER LEVEL INSIDE THE SHROUD VERSUS TIME AFTER BREAK

GRAND GULF

.09 SQ. FT. BREAK (HIGHEST TEMPERATURE SMALL BREAK), RECIRCULATION SUCTION BREAK, HPCS FAILURE

WITH 1 ADS VALVE OUT OF SERVICE

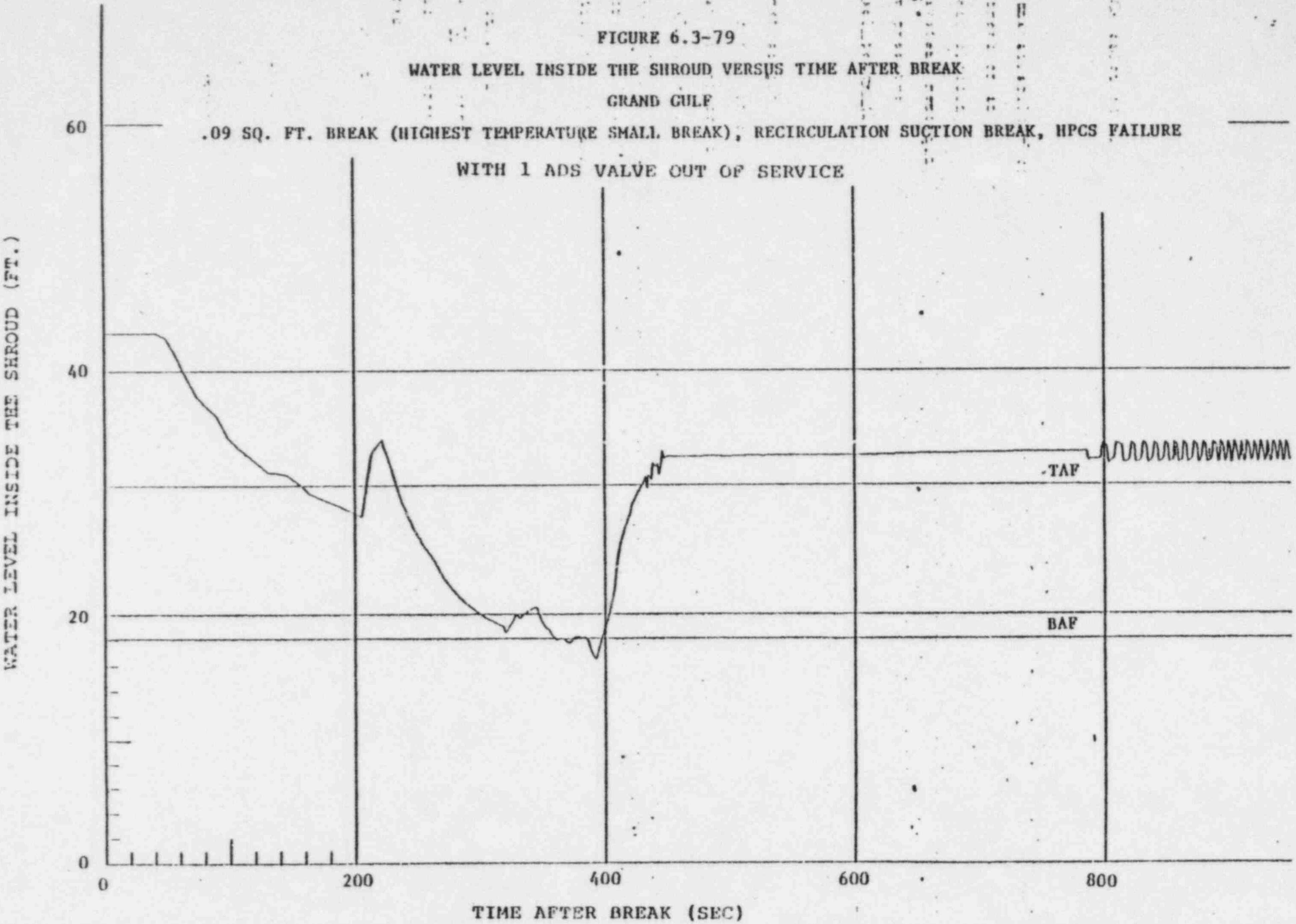


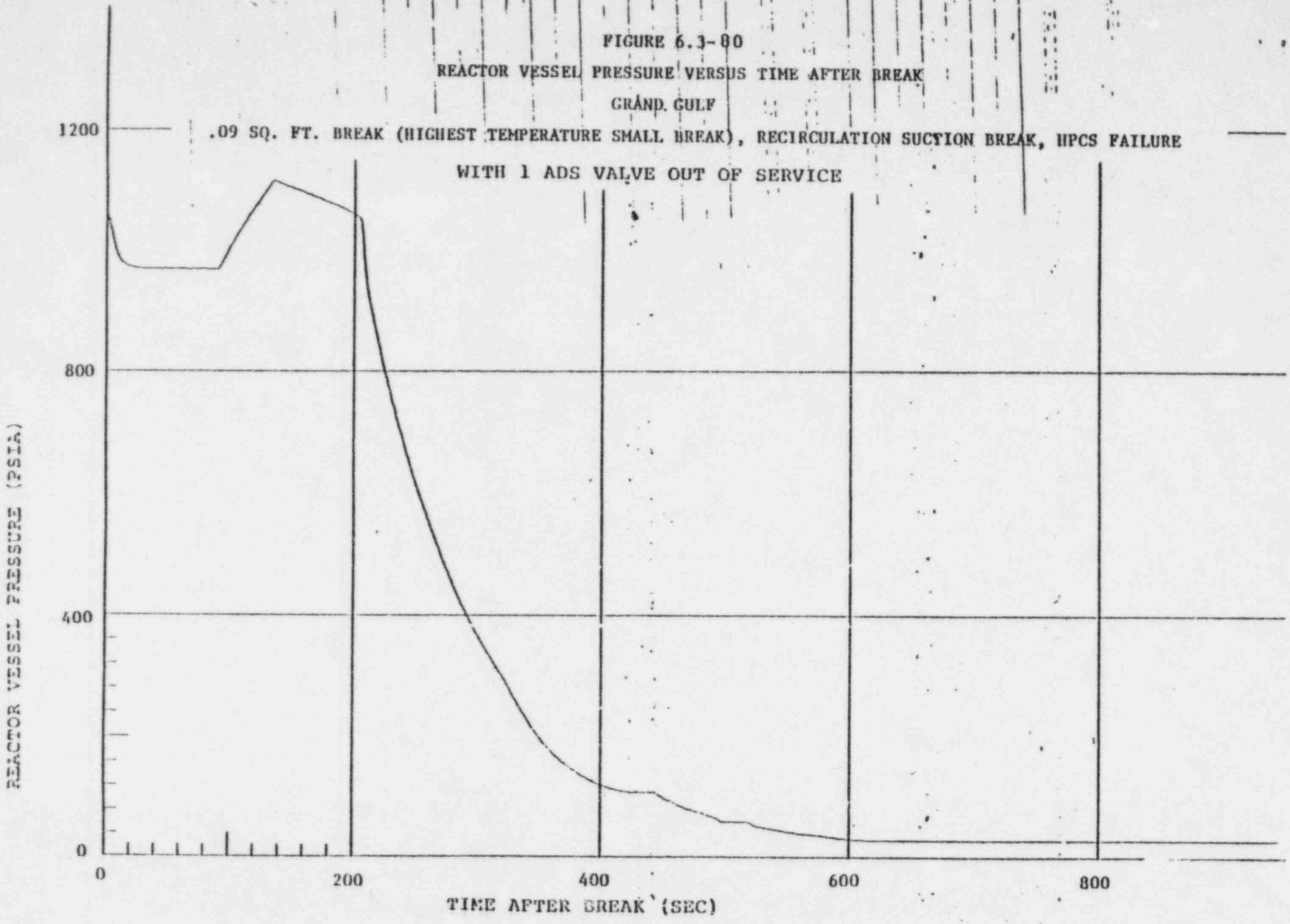
FIGURE 6.3-80

REACTOR VESSEL PRESSURE VERSUS TIME AFTER BREAK

GRAND GULF

.09 SQ. FT. BREAK (HIGHEST TEMPERATURE SMALL BREAK), RECIRCULATION SUCTION BREAK, HPCS FAILURE

WITH 1 ADS VALVE OUT OF SERVICE



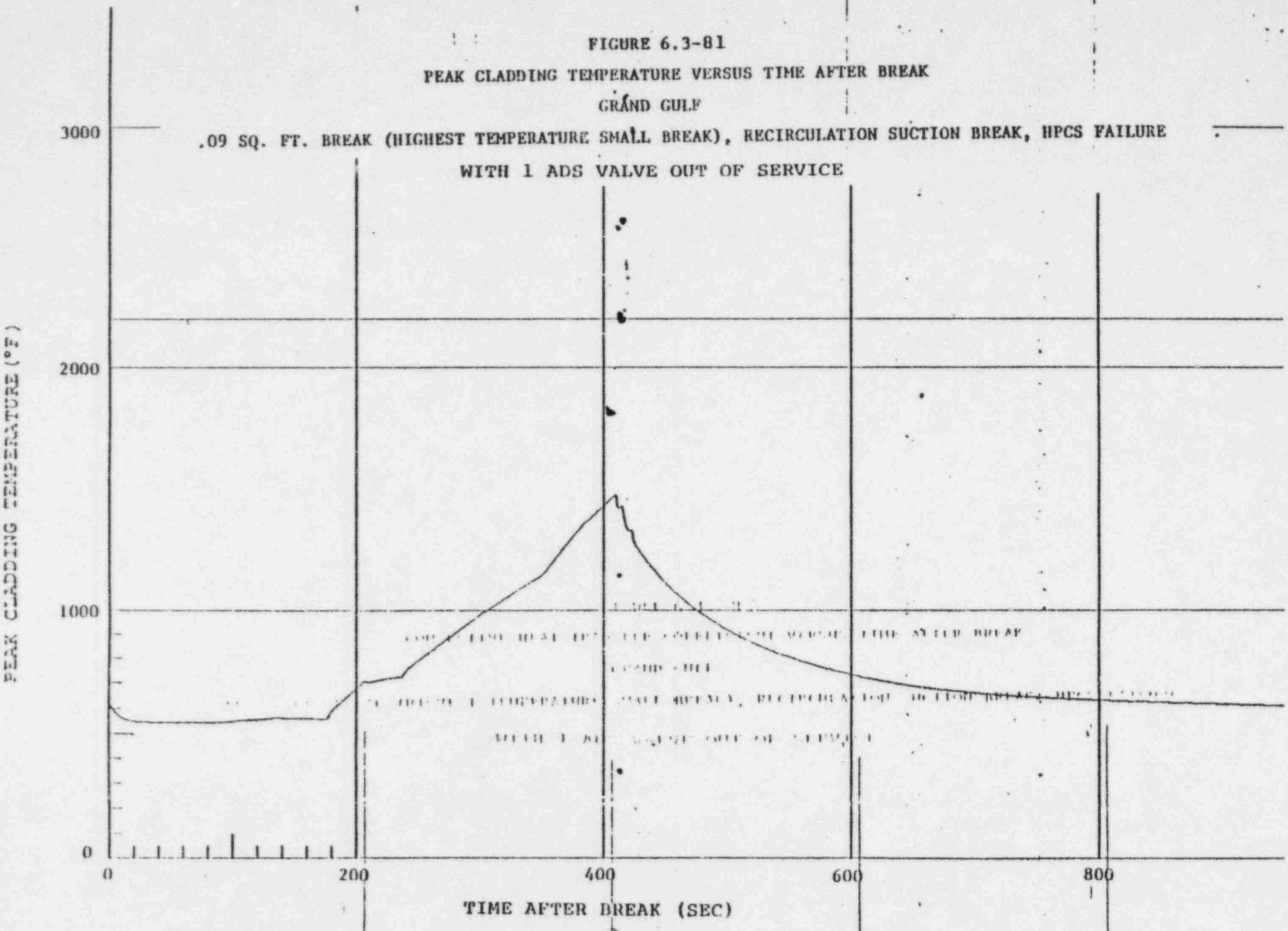


FIGURE 6.3-81

PEAK CLADDING TEMPERATURE VERSUS TIME AFTER BREAK

GRAND GULF

.09 SQ. FT. BREAK (HIGHEST TEMPERATURE SMALL BREAK), RECIRCULATION SUCTION BREAK, HPCS FAILURE

WITH 1 ADS VALVE OUT OF SERVICE

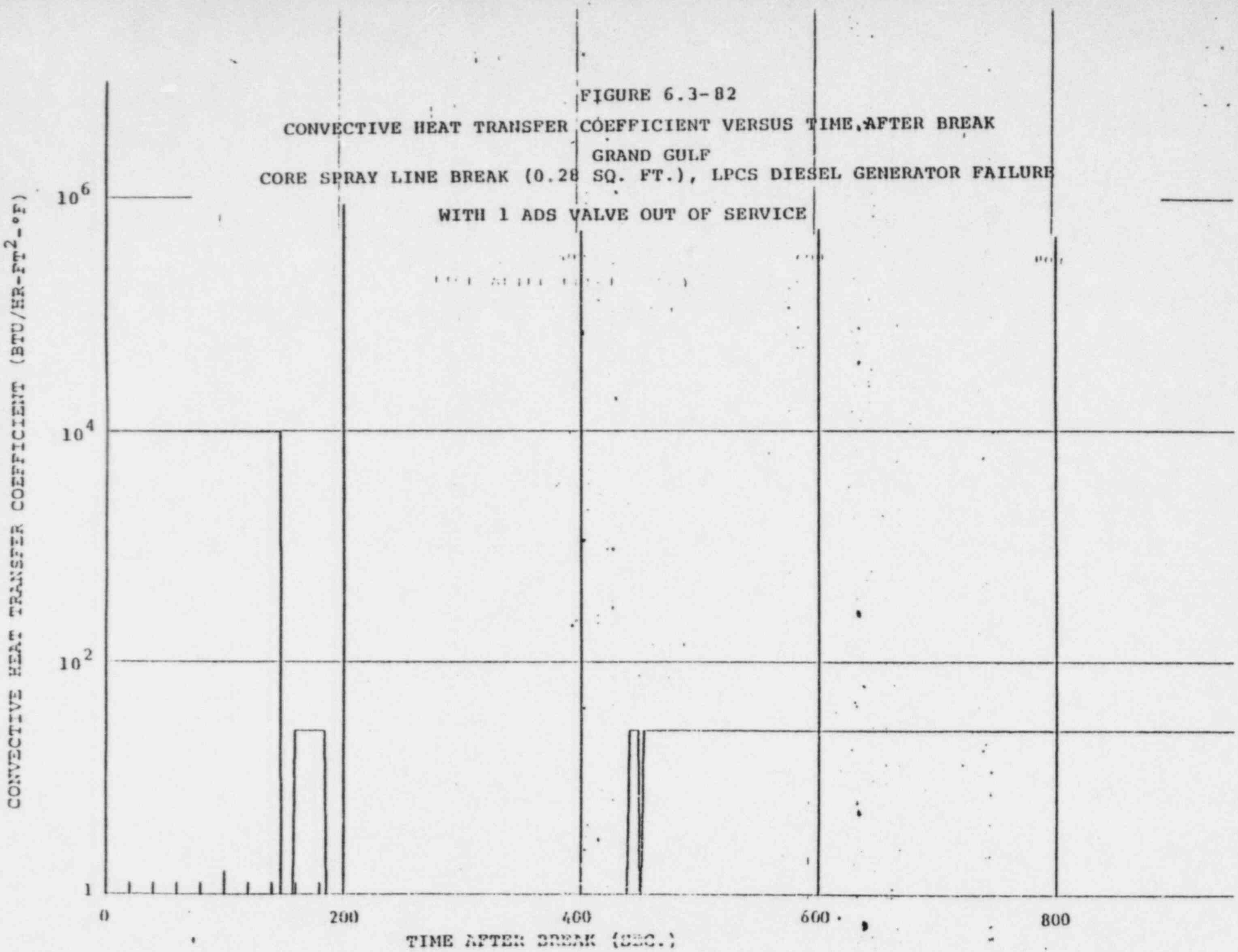


FIGURE 6.3-82
CONVECTIVE HEAT TRANSFER COEFFICIENT VERSUS TIME AFTER BREAK
GRAND GULF
CORE SPRAY LINE BREAK (0.28 SQ. FT.), LPCS DIESEL GENERATOR FAILURE
WITH 1 ADS VALVE OUT OF SERVICE
CONVECTIVE HEAT TRANSFER COEFFICIENT (BTU/HR-FT2-°F)
10^6
10^4
10^2
1
0
200
400
600
800
TIME AFTER BREAK (SEC.)

FIGURE 6.3-83

WATER LEVEL INSIDE THE SHROUD VERSUS TIME AFTER BREAK

GRAND GULF

CORE SPRAY LINE BREAK (0.28 SQ. FT.), LPCS DIESEL GENERATOR FAILURE

WITH 1 ADS VALVE OUT OF SERVICE

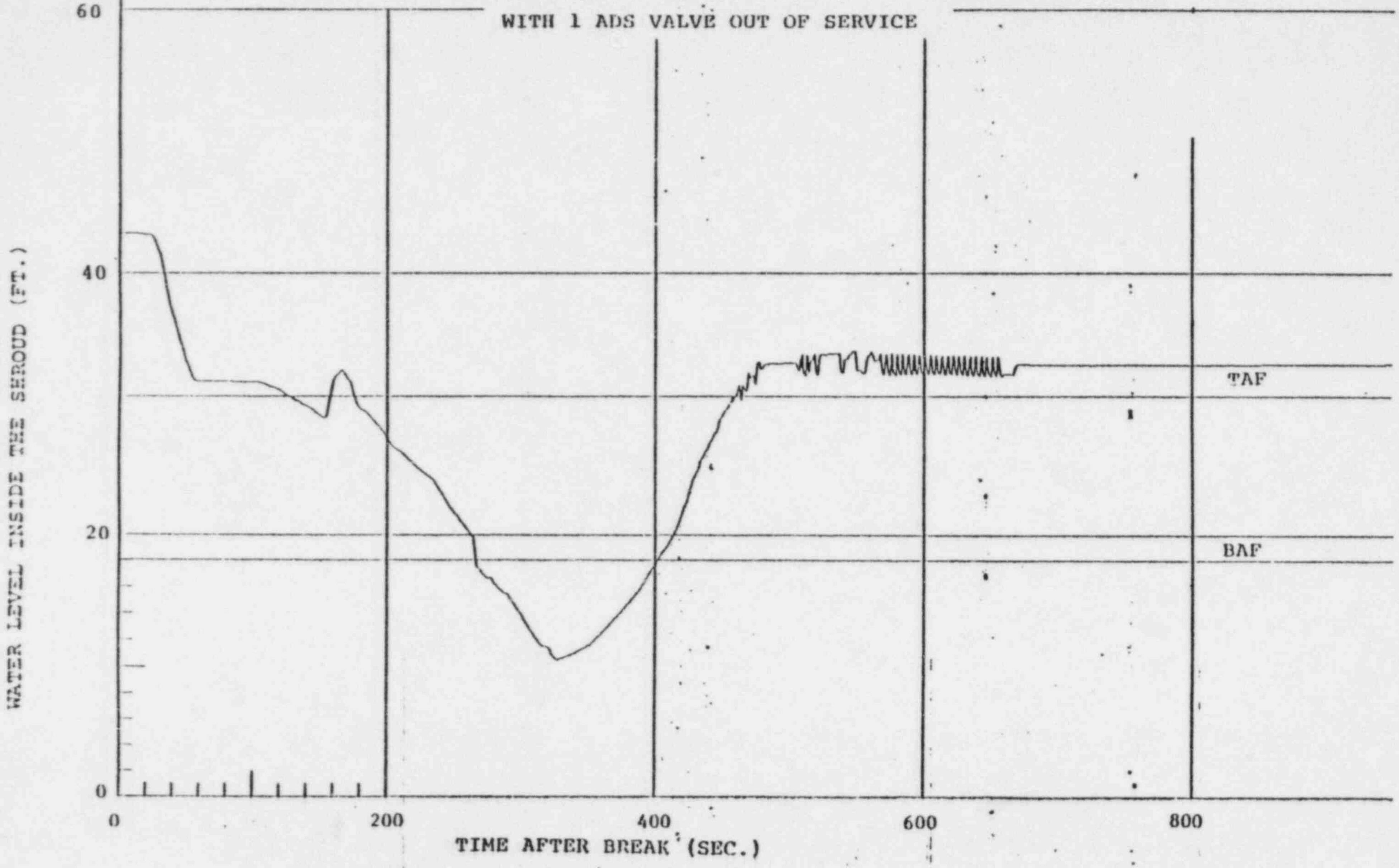


FIGURE 6.3-84

REACTOR VESSEL PRESSURE VERSUS TIME AFTER BREAK

GRAND GULF

CORE SPRAY LINE BREAK (0.28 SQ. FT.), LPCS DIESEL GENERATOR FAILURE

WITH 1 ADS VALVE OUT OF SERVICE

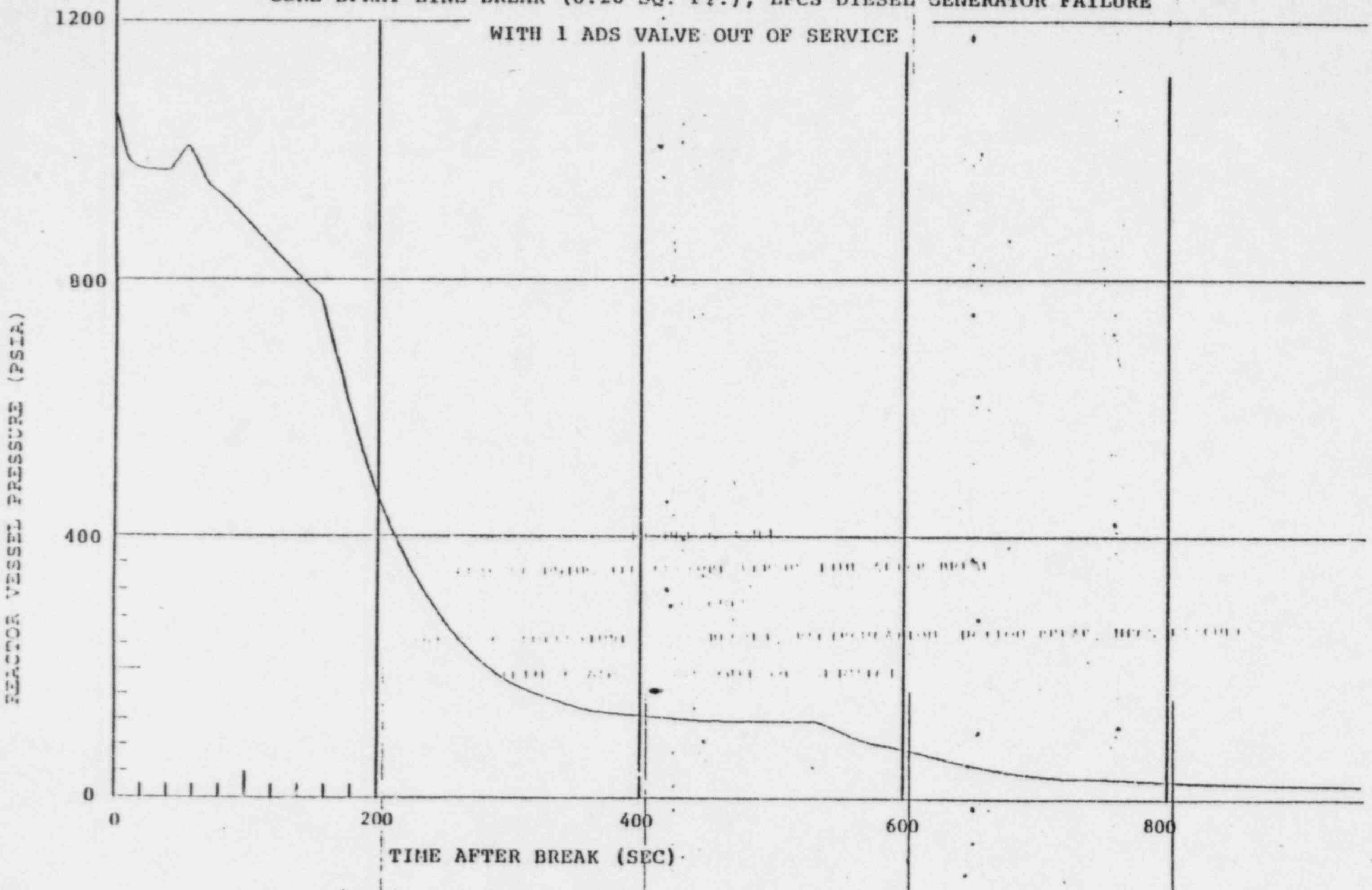


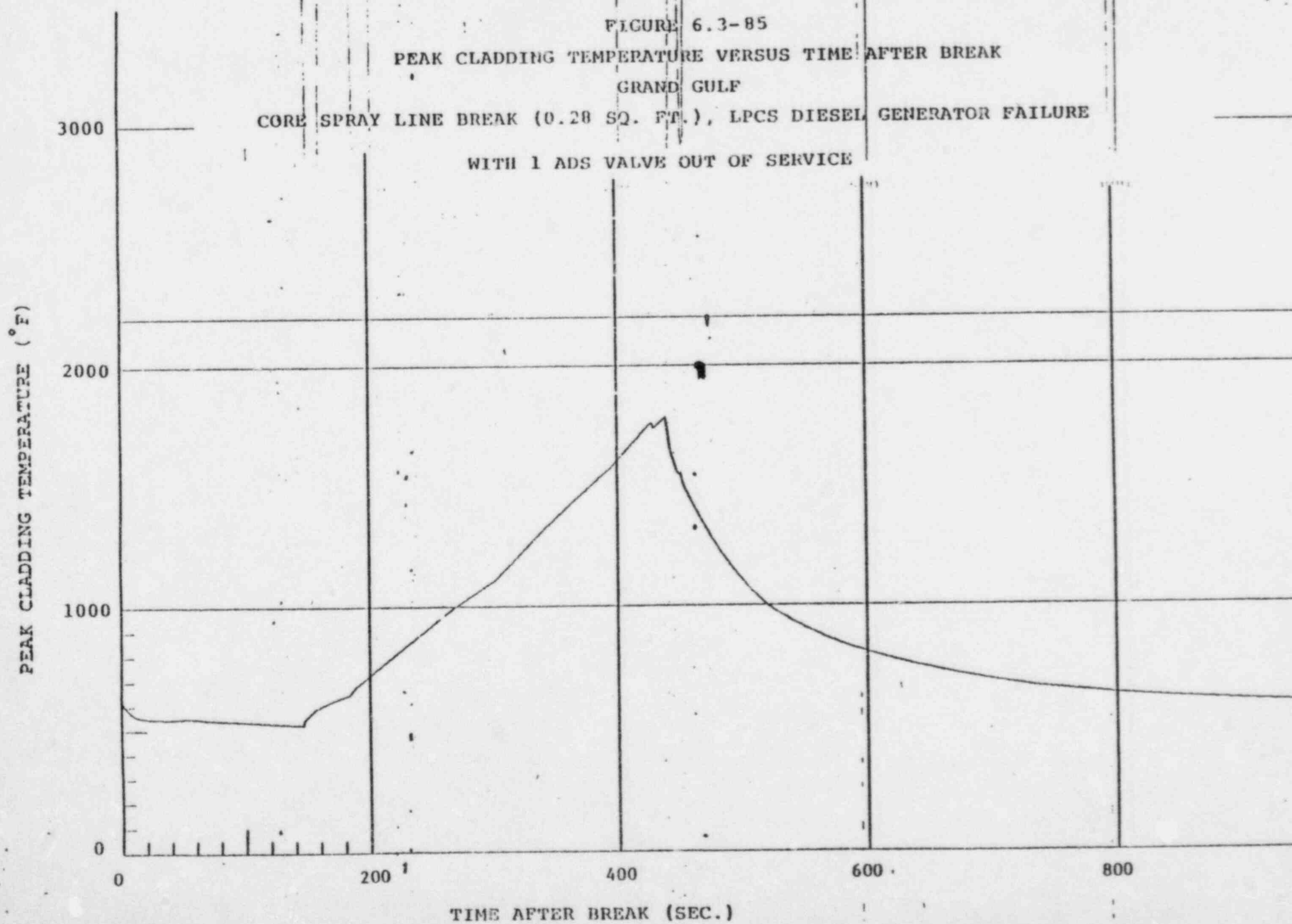
FIGURE 6.3-85

PEAK CLADDING TEMPERATURE VERSUS TIME AFTER BREAK

GRAND GULF

CORE SPRAY LINE BREAK (0.28 SQ. FT.), LPCS DIESEL GENERATOR FAILURE

WITH 1 ADS VALVE OUT OF SERVICE



greater than 10 minutes will have less severe consequences than diversion of 10 minutes.

It should also be noted that the dotted line curve of Figure 212.24-1 is bounding for small breaks (<approximately 0.01 ft^2) since only one LPCI pump (the minimum possible since LPCI pump 51
"C" does not divert) is assumed to operate for these breaks. 29
Consequently, if one assumed a new scenario whereby diversion occurred at times later than ten minutes, the Figure 212.24-1 curve would peak at a smaller break area, and the PCT at that peak would be the same as that on the dotted line curve for that break area.

To resolve the concern of the NRC staff that premature diversion of low pressure coolant injection (LPCI) flow to containment sprays could adversely affect core cooling, the Grand Gulf symptom-based emergency procedures have been constructed to caution the operator against such diversion unless adequate core cooling is assured. These procedures clearly identify 51
LPCI diversion as secondary to the core cooling requirements except in those instances, outside the plant design envelope, which involve multiple failures and for which maintenance of containment integrity is required to minimize risk to the environment.

Insert 1

1

Consideration of an ADS Valve Out of Service

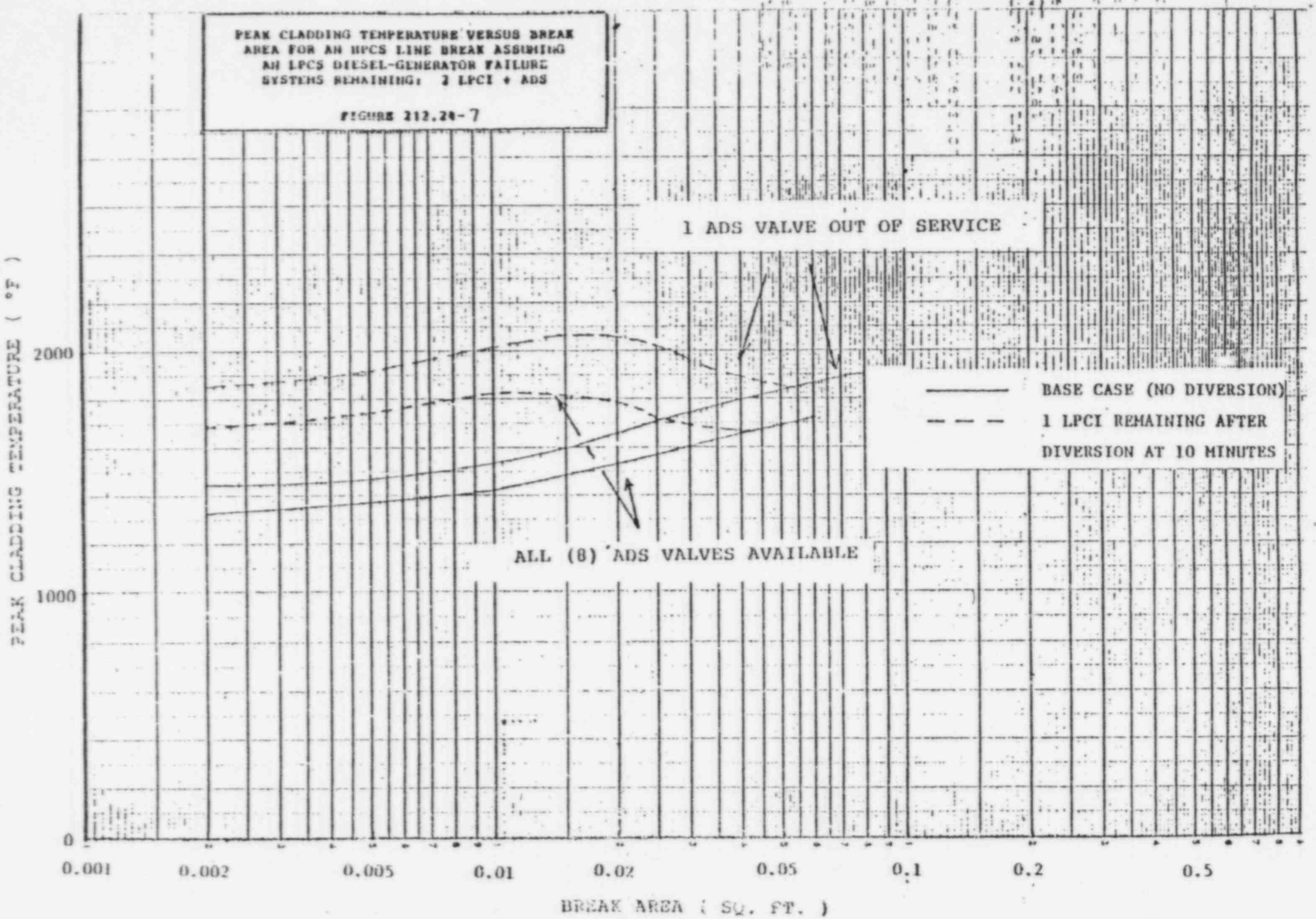
The purpose of this section is to quantify the effect of an ADS valve out of ~~service on~~ the above LOCA analysis.

The above analysis demonstrated that the HPCS line break with failure of the LPCS diesel generator results in the highest calculated peak cladding temperatures (PCT). For these cases the emergency core cooling systems remaining are 2 LPCI plus the ADS. Here the ADS is required to rapidly depressurize the vessel below the shutoff head of the low pressure ECCS.

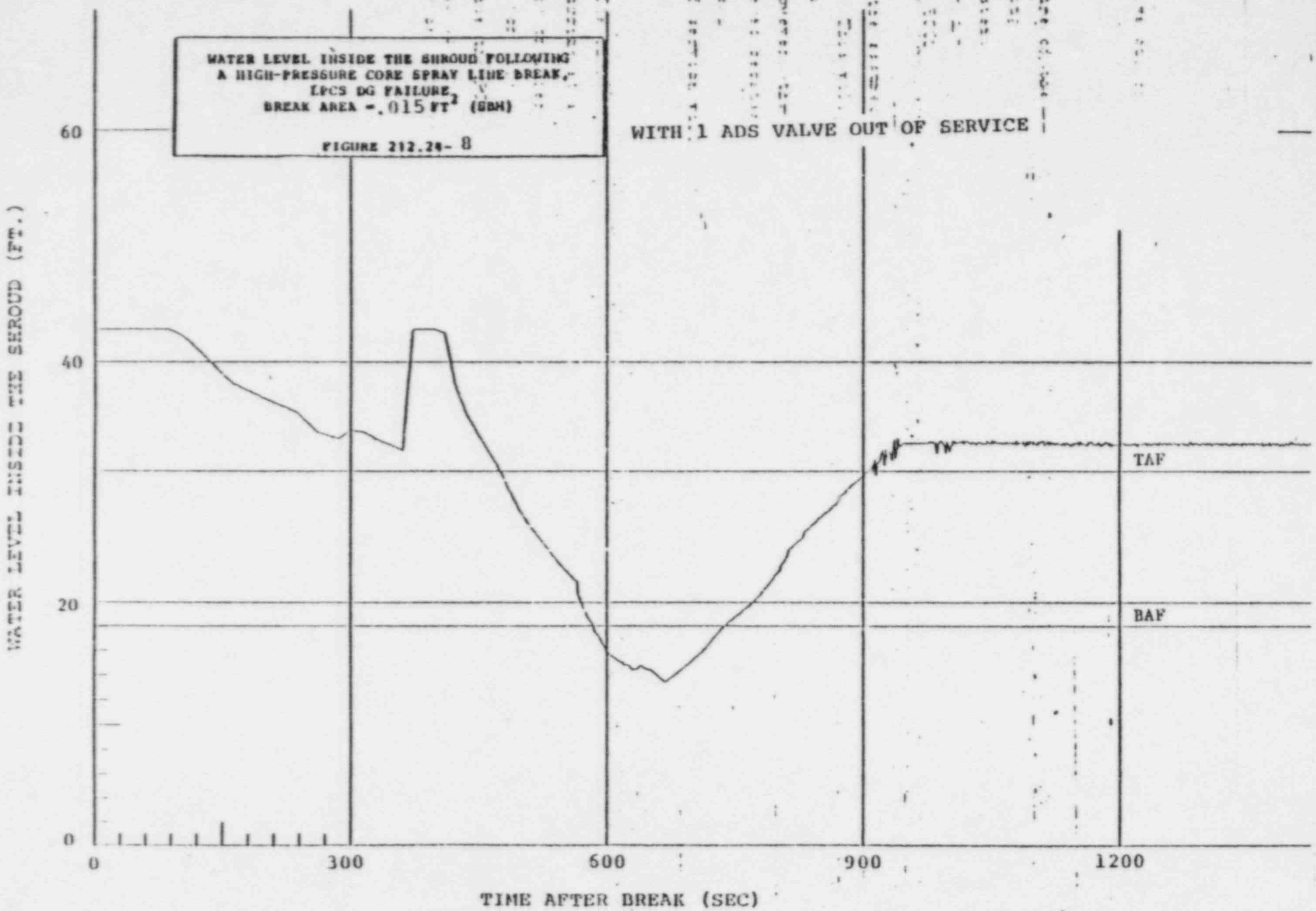
If an ADS valve is out of service and unavailable, in addition to the assumed worst single failure, the ADS will depressurize the vessel slower. This will result in a delay of low pressure ECCS injection and, in general, a corresponding delay in reflooding time and increase in the PCT. However, the significance of the ADS decreases as larger break sizes are considered because of the increasing depressurization due to mass loss through the break. Therefore, the maximum impact on the PCT due to an ADS valve out of service will be determined by the recalculation of the small break spectrum.

Figure 212.24-7 shows the impact of an ADS valve out of service. The increase in PCT is shown to be about 270°F with a maximum PCT of 2064°F occurring at a break size of 0.015 ft². Reactor water level, vessel pressure, heat transfer coefficients, peak cladding temperature, and ECCS flow versus time for this limiting case are shown in Figures 212.24-8 through 212.24-12. Since the PCT for this limiting case is still below the 2200°F limit, no change in the operating MAPLEGR limit is required due to an ADS valve out of service, to meet the 10CFR50.46 licensing limits.

All peak cladding temperatures for the cases affected by an ADS valve out of service are still well below the PCT for the maximum recirculation line break which is unaffected by the number of ADS valves available as discussed earlier. Therefore based on the results of the LOCA analyses presented in this section, it is concluded that it is acceptable for an ADS valve to be out of service for an extended period of time without changing the operating MAPLHGR limits to meet the 10CFR50.46 licensing limits.

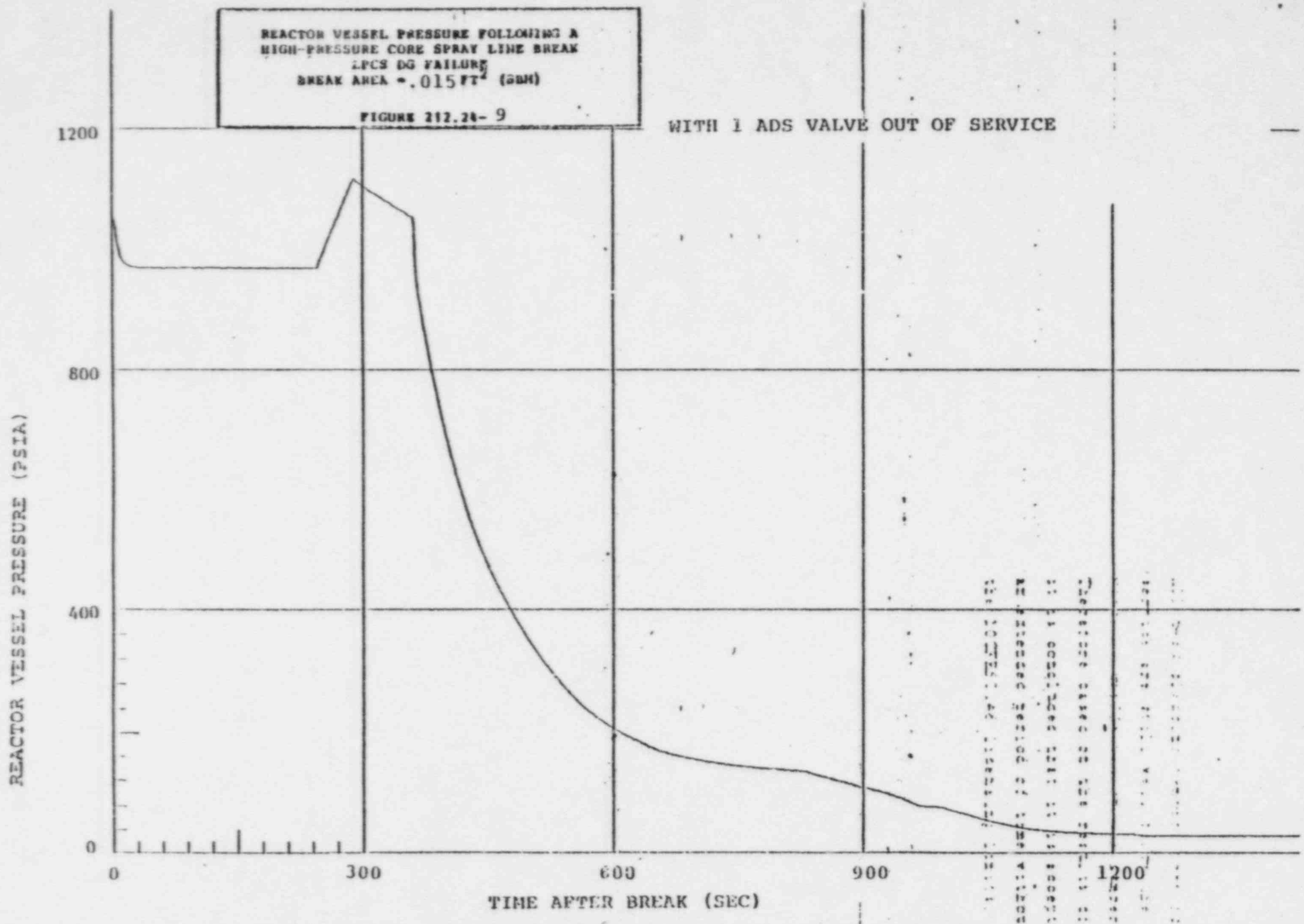


PEAK CLADDING TEMPERATURE VERSUS BREAK AREA FOR AN HPCS LINE BREAK ASSUMING AN LPCS DIESEL-GENERATOR FAILURE SYSTEMS REMAINING: 2 LPCI + ADS
FIGURE 212.24-7
1 ADS VALVE OUT OF SERVICE
BASE CASE (NO DIVERSION)
1 LPCI REMAINING AFTER DIVERSION AT 10 MINUTES
ALL (8) ADS VALVES AVAILABLE
PEAK CLADDING TEMPERATURE (°F)
2000
1000
0
0.001
0.002
0.005
0.01
0.02
0.05
0.1
0.2
0.5
BREAK AREA (SQ. FT.)



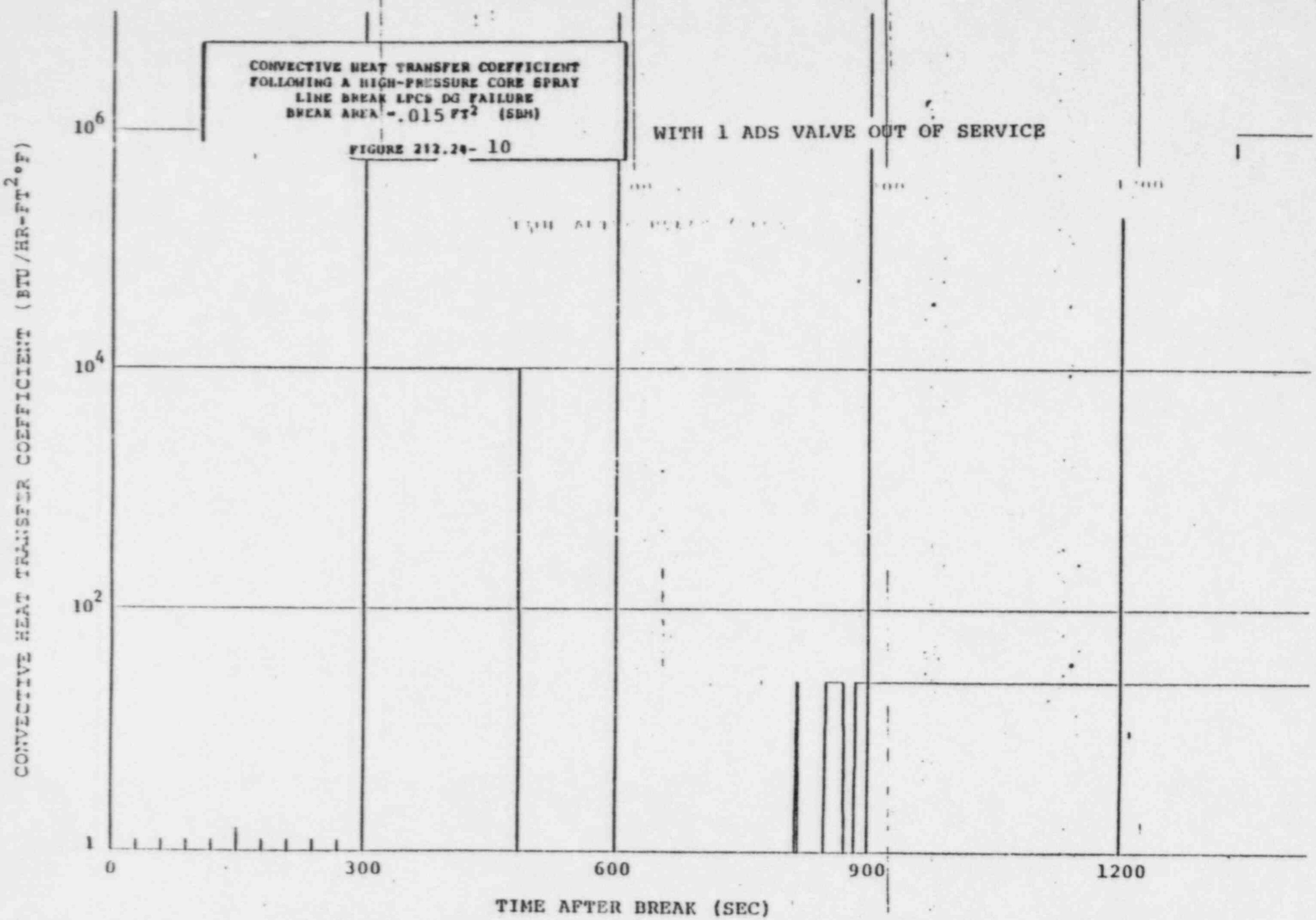
WATER LEVEL INSIDE THE SHROUD FOLLOWING A HIGH-PRESSURE CORE SPRAY LINE BREAK, LPCS DG FAILURE, BREAK AREA = .015 FT^2 (SBM)

FIGURE 212.24- 8

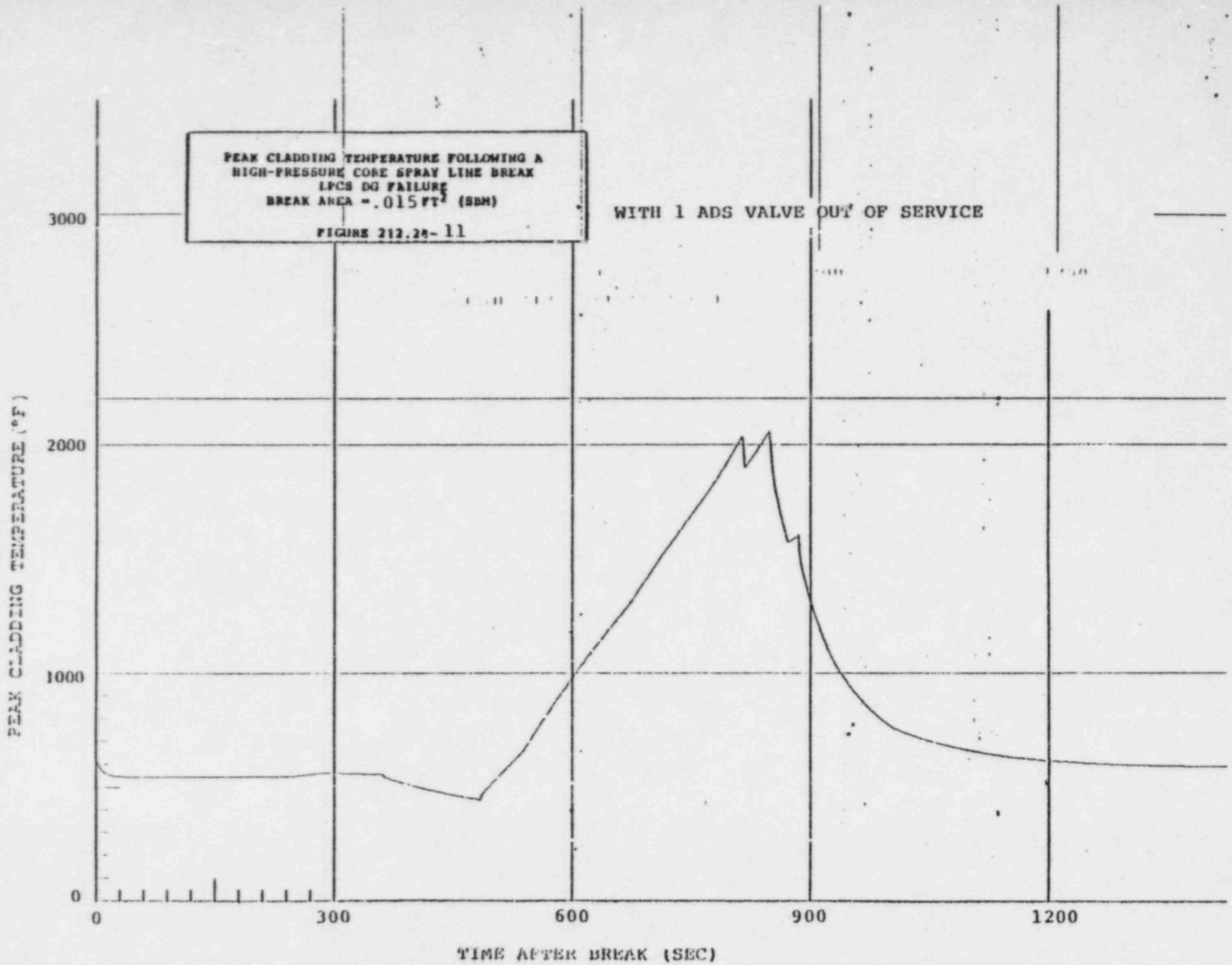


REACTOR VESSEL PRESSURE FOLLOWING A
HIGH-PRESSURE CORE SPRAY LINE BREAK
LPCS DG FAILURE
BREAK AREA = .015 FT² (SBM)

FIGURE 212.24- 9

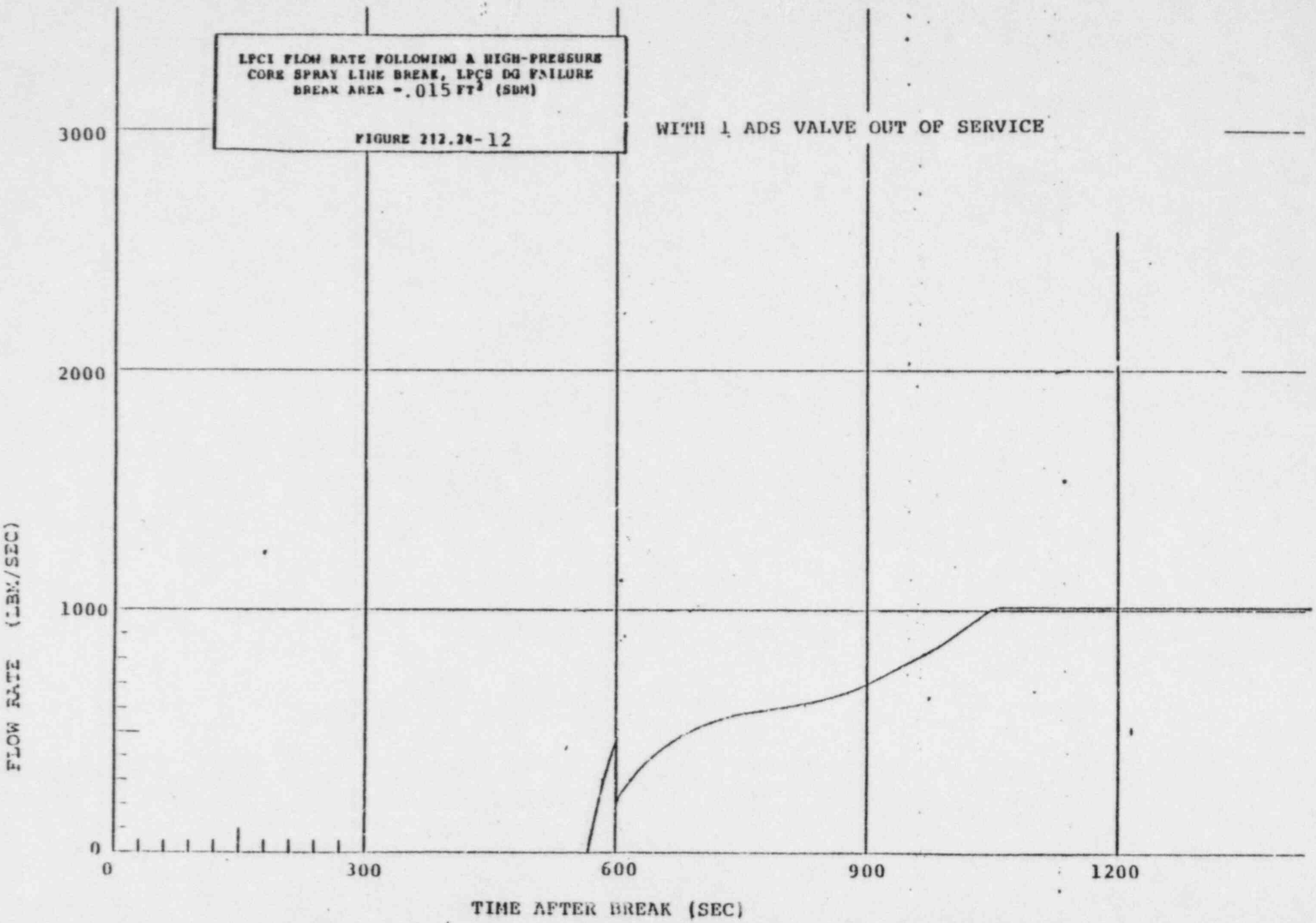


CONVECTIVE HEAT TRANSFER COEFFICIENT
FOLLOWING A HIGH-PRESSURE CORE SPRAY
LINE BREAK LPCS DG FAILURE
BREAK AREA -.015 FT^2 (SBM)
FIGURE 212.24- 10
WITH 1 ADS VALVE OUT OF SERVICE
CONVECTIVE HEAT TRANSFER COEFFICIENT (BTU/HR-FT^2°F)
10^6
10^4
10^2
1
0
300
600
900
1200
TIME AFTER BREAK (SEC)



PEAK CLADDING TEMPERATURE FOLLOWING A
HIGH-PRESSURE CORE SPRAY LINE BREAK
LPCS DG FAILURE
BREAK AREA = .015 FT^2 (SBM)

FIGURE 212.28-11


LPCI FLOW RATE FOLLOWING A HIGH-PRESSURE
CORE SPRAY LINE BREAK, LPCS DG FAILURE
BREAK AREA =.015 FT² (SBM)
FIGURE 212.24-12
WITH 1 ADS VALVE OUT OF SERVICE
3000
2000
1000
0
FLOW RATE (LBM/SEC)
0
300
600
900
1200
TIME AFTER BREAK (SEC)

TECHNICAL SPECIFICATION PROBLEM SHEET

Item Number: 005 Priority: ~~1A~~ 1B

_____ / _____
Identified By Date

_____
Responsible Supervisor

Tech Spec Reference: 3/4.3.2 - (3.3.2-1.4.h)

Problem Title: Min. Operable Channels for RWCU isolation from SLCS initiation

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other): Table 3.3.2-1.4.h lists the minimum operable channels per trip system as "NA" for RWCU isolation from a SLCS initiation. Therefore, the ACTION statements for the Tech Spec do not apply and the channel may remain out of service indefinitely with no action required.

2. Safety Significance: Tech Specs are confusing and could lead to non-conservative plant operation.

3. Anticipated Resolution: Change tech specs to require logic channel to be operable and an indicated ACTION taken if channel is inop.
Could use Tech Spec Position Statement

4. NRC Response to Item (NRR/IE): _____

NRC Notified: _____ _____ / _____
Individual Notified Date Time

5. Disposition: _____

Items Closed: (How) _____

_____ / _____
Date Time

cc: J. E. Cross
R. F. Rogers

SUBJECT: Technical Specification Tables 3.3.2-1 and 4.3.2.1-1 item 4.h, pages 3/4 3-12, 3/4 3-14, 3/4 3-22, and 3/4 3-23a.

No. -005

DISCUSSION: Technical Specification Table 3.3.2-1, item 4.h (Standby Liquid Control System (SLCS) initiation) provides a signal to isolate the Reactor Water Cleanup (RWCU) system upon SLCS actuation. This interlock is provided to ensure that the boron concentration injected into the reactor vessel by SLCS is not depleted by normal RWCU system operation.

This proposed change to the Technical Specifications consists of three parts. The first part of the change involves changing the minimum operable channels per trip system from "NA" to one (1) for SLCS initiation. The second part of the change replaces Operational Condition 3 with Operational Condition 5 and adds "##" footnote to require the SLCS Initiation function to be operable in Operational Condition 5 only when control rods are withdrawn. The third part of this change replaces present action statement 27 for the SLCS Initiation function with new action 30 on Table 3.3.2-1 which requires the affected SLCS pump to be declared inoperable.

JUSTIFICATION: In order to enter ACTION a or b of Technical Specification 3.3.2, the number of OPERABLE channels must be less than that required by the Minimum OPERABLE channels per Trip System in Table 3.3.2-1. Presently there are no channels required to be OPERABLE for the SLCS Initiation function under RWCU Isolation. Therefore, the ACTION statements for Technical Specification 3.3.2 can never be entered for the SLCS Initiation function. The SLCS Initiation function design consists of one (1) channel per trip system. Adding the one (1) channel per trip system will reflect system design and allow entering appropriate action statements for the SLCS Initiation function.

The present applicable operational conditions for the SLCS in Technical Specification 3/4.1.5 are not the same as the ones for the SLCS Initiation function in Tables 3.3.2-1 and 4.3.2.1-1. Technical Specification 3/4.1.5 requires the SLCS to be operable in Operational Conditions 1, 2, and 5* where the "*" footnote applies with any control rod withdrawn but not to control rods removed per Technical Specification 3.9.10.1 or 3.9.10.2. The applicable operational conditions for the SLCS Initiation function in Tables 3.3.2-1 and 4.3.2.1-1 are 1, 2, and 3. The applicable operational conditions for these specifications should be identical since the specifications involve the same SLCS system. This part of the proposed technical specification change deletes Operational Condition 3 and adds 5## to item 4.h of Tables 3.3.2-1 and 4.3.2.1-1 (the "*" footnote from Technical Specification 3.1.5 and the "##" footnote are identical). Control rods cannot be pulled in Operational Condition 3, therefore, the SLCS system and its associated isolation function are not required for this case.

Present Action Statement 27 for the SLCS Initiation function in Table 3.3.2-1 requires the RWCU isolation valves to be closed and the RWCU system to be declared inoperable. This action can have an adverse impact on reactor water quality at power. The new proposed Action 30 for the SLCS Initiation function will require that the affected SLCS pump be declared inoperable. This new Action 30 will then require entry into Action a.1 or b.1 of Technical Specification 3.1.5 to determine action requirements for an inoperable SLCS pump. There are two SLCS systems in the Grand Gulf design. SLCS "A" initiation will close RWCU outboard isolation valve G33-F004. SLCS "B" initiation will close RWCU inboard isolation valves G33-F001 and F251. Initiation of either SLCS "A" or "B" will cause isolation of the RWCU system, therefore, one SLCS pump can be out of service without adversely affecting the isolation capability of the SLCS initiation function.

SIGNIFICANT HAZARDS CONSIDERATION:

The addition of one (1) minimum operable channel per trip system for the SLCS initiation function constitutes an additional limitation not presently in the Technical Specifications. The change to the applicable operational conditions was made to promote consistency among Technical Specification 3.1.5, Table 3.3.2-1 and Table 4.3.2.1-1. Changing the action statement for the SLCS Initiation function from the present Action 27 to the new Action 30 on Table 3.3.2-1 is made to ensure reactor water quality by not isolating RWCU but still retaining the isolation function from the redundant SLCS system. Also, since the affected SLCS pump is declared inoperable by the new Action 30, the SLCS Initiation function must be restored within the time constraints of the Action Statements of Technical Specification 3.1.5. The proposed changes do not involve a) the reduction of safety margins, b) an increase in the probability or consequences of a previously evaluated accident, or c) the possible creation of a new or different kind of accident. Therefore, the proposed changes do not involve a significant hazards consideration.

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL (a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
4. REACTOR WATER CLEANUP SYSTEM ISOLATION (Continued)				
f. Main Steam Line Tunnel Ambient Temperature - High	8	1	1, 2, 3	27
g. Main Steam Line Tunnel Δ Temp. - High	8	1	1, 2, 3	27
h. SLCS Initiation	8(i)	NA	1, 2, ~~3~~ 5##	~~27~~ 30
i. Manual Initiation	8	2	1, 2, 3	26
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High	4	1	1, 2, 3	27
b. RCIC Steam Supply Pressure - Low	4, 9(m)	1	1, 2, 3	27
c. RCIC Turbine Exhaust Diaphragm Pressure - High	4	2	1, 2, 3	27
d. RCIC Equipment Room Ambient Temperature - High	4	1	1, 2, 3	27
e. RCIC Equipment Room Δ Temp. - High	4	1	1, 2, 3	27
f. Main Steam Line Tunnel Ambient Temperature - High	4	1	1, 2, 3	27
g. Main Steam Line Tunnel Δ Temp. - High	4	1	1, 2, 3	27
h. Main Steam Line Tunnel Temperature Timer	4	1	1, 2, 3	27

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION

ACTION

ACTION 20 - Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 21 - Close the affected system isolation valve(s) within one hour or:
  a. In OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
  b. In Operational Condition *, suspend CORE ALTERATIONS, handling of irradiated fuel in the primary containment and operations with a potential for draining the reactor vessel.

ACTION 22 - Restore the manual initiation function to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION 23 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 24 - Be in at least STARTUP within 6 hours.

ACTION 25 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within one hour.

ACTION 26 - Restore the manual initiation function to OPERABLE status within 8 hours or close the affected system isolation valves within the next hour and declare the affected system inoperable.

ACTION 27 - Close the affected system isolation valves within one hour and declare the affected system inoperable.

ACTION 28 - Lock the affected system isolation valves closed within one hour and declare the affected system inoperable.

ACTION 29 - Close the affected system isolation valves within one hour and declare the affected system or component inoperable or:
  a. In OPERATIONAL CONDITION 1, 2 or 3 be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
  b. In OPERATIONAL CONDITION # suspend CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

ACTION 30 - Declare the affected SLCS Pump inoperable.

NOTES

* When handling irradiated fuel in the primary or secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

\# During CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

(a) See Specification 3.6.4, Table 3.6.4-1 for valves in each valve group.

(b) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

(c) Also actuates the standby gas treatment system.

(d) Also actuates the control room emergency filtration system in the isolation mode of operation.

(e) Two upscale-Hi Hi, one upscale-Hi Hi and one downscale, or two downscale signals from the same trip system actuate the trip system and initiate isolation of the associated containment and drywell isolation valves.

with any control rod withdrawn, Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
4. REACTOR WATER CLEANUP SYSTEM ISOLATION (Continued)				
f. Main Steam Line Tunnel Ambient Temperature - High	S	M	R	1, 2, 3
g. Main Steam Line Tunnel Δ Temp. - High	S	M	R	1, 2, 3
h. SLCS Initiation	NA	M(b)	NA	1, 2, ~~3~~ 5 ##
i. Manual Initiation	NA	M(a)	NA	1, 2, 3
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High	S	M	R	1, 2, 3
b. RCIC Steam Supply Pressure - Low	S	M	R	1, 2, 3
c. RCIC Turbine Exhaust Diaphragm Pressure - High	S	M	R	1, 2, 3
d. RCIC Equipment Room Ambient Temperature - High	S	M	R	1, 2, 3
e. RCIC Equipment Room Δ Temp. - High	S	M	R	[illegible], 2, 3
f. Main Steam Line Tunnel Ambient Temperature - High	S	M	R	1, 2, 3
g. Main Steam Line Tunnel Δ Temp. - High	S	M	R	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
6. RHR SYSTEM ISOLATION (Continued)				
e. Drywell Pressure - High	S	M	R	1, 2, 3
f. Manual Initiation	NA	M(a)	NA	1, 2, 3

*When handling irradiated fuel in the primary or secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

**When reactor steam pressure ≥ 1045 psig and/or any turbine stop valve is open.

#During CORE ALTERATION and operations with a potential for draining the reactor vessel.

(a) Manual initiation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual initiation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days as part of circuitry required to be tested for automatic system isolation.

(b) Each train or logic channel shall be tested at least every other 31 days.

(c) Calibrate trip unit at least once per 31 days.

##With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2

DEFINITIONS FOR
"CHANNELS", "TRIP SYSTEMS", AND "TRIP FUNCTIONS"
FOR GROUP 8 ISOLATION (CONTINUED)

Trip Unit	Parameter	Logic	
E31-FS-N609B	RWCU Δflow-high	Two out of two	
E31-KIS-R615B	RWCU Δflow-timer		
E31-TS-N620B	RWCU area temp-high		
E31-TS-N621B	RWCU area temp-high		
E31-TS-N622B	RWCU area temp-high *		
E31-TS-N623B	RWCU area temp-high		
E31-TS-N624B	RWCU area temp-high		
E31-TS-N625B	RWCU area temp-high		
E31-TS-N626B	RWCU area temp-high		
E31-TS-N627B	RWCU area temp-high		
E31-TDS-N612B	RWCU area Δtemp-high		
E31-TDS-N613B	RWCU area Δtemp-high		
E31-TDS-N614B	RWCU area Δtemp-high	Any One	Close inboard Group 8 valves TRIP FUNCTION
E31-TDS-N615B	RWCU area Δtemp-high		
E31-TDS-N616B	RWCU area Δtemp-high		
E31-TDS-N617B	RWCU area Δtemp-high		
E31-TDS-N618B	RWCU area Δtemp-high		
E31-TDS-N619B	RWCU area Δtemp-high		
B21-LS-N682B	RPV Level 2	Two out of two	
B21-LS-N682C	RPV Level 2		
E31-TS-N604B	MSL tunnel temp-high		
E31-TDS-N605B	MSL tunnel Δtemp-high		
C41-HS-M601B	SLCS initiation (Note 1)		
B21-HS-M630B	Manual initiation	Two out of two	TRIP SYSTEM
B21-HS-M630C	Manual initiation		

* One Channel (typical of 25 shown on this page)
Note 1 - Closes only valve G33-F001 & F251

I3sp12

TECHNICAL SPECIFICATION PROBLEM SHEET

Item Number: 015 Priority: S/A-1B

_____ / _____
Identified By Date

_____
Responsible Supervisor

Tech Spec Reference: Table 2.2.1-1 / 3.3.2-2 /3.3.3-2 / 3.3.8-2

Problem Title: Drywell/Containment Pressure Setpoints

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other): Drywell & containment instrumentation uses absolute pressure transmitters with a psig set point. Barametric pressure changes will affect instrumentation. Pressure set point change may be required to account for barometric pressure changes.

2. Safety Significance: This is not a Tech Spec problem but a hardware problem of significance.

3. Anticipated Resolution: NPE has this item for action (PMI 83/12028) to determine resolution. The setpoints have been lowered by .5 psi as an interim measure. Is incorporated in Surveillance procedures.

4. NRC Response to Item (NRR/IE): _____

NRC Notified: _____ _____ / _____
Individual Notified Date Time

5. Disposition: _____

Items Closed: (How) _____

_____ / _____
Date Time

cc: J. E. Cross
R. F. Rogers

pc 15

15

PM JA MM AMS

OCT 27 1983

GGNS

MEMO TO: C. K. McCoy

FROM: J. F. Pinto

SUBJECT: Drywell Pressure Set Points

FILE NO.: 0290/15635

PMI: 83/12,028

DATE: October 26, 1983

PSRC review 10/27/83

BACKGROUND: Plant Staff has written PMI 83/9213 which NPE replied to on PMI 83/11,417. Subsequent discussions with the Preliminary Safety Review Committee has clarified Plant Staff's concerns about the lowering of the drywell pressure set points to the 1.23 PSIG value.

DISCUSSION: NPE has been convinced that further evaluation of plant conditions is necessary prior to a final evaluation as to the safety concerns of this set point change.

ACTION: Plant Staff is to shut off the containment mini purge (M41 system). This will isolate the containment with the exception of allowed leakage and should insure that the drywell pressure will not be affected by atmospheric pressure changes. This isolation will allow leaving the drywell pressure transmitter set points at their present setting of 1.73 PSIG. NPE will pursue the following action items:

1. Evaluate how closely drywell pressure will follow atmospheric pressure (i.e., how much time lag).

2. Design basis for drywell pressure transmitters as applied to the various safety analysis in the FSAR and the Humphrey concerns.

3. Reevaluate the drywell pressure transmitter set point based on the above findings and normal and adverse atmospheric pressure conditions.

4. Evaluate alternatives to the reduction of operating margins for any set point change.

5. Evaluate loss of drywell cooling's affect on drywell pressure transmitters.

Any questions concerning these items should be directed to J. D. Voss at extension 2710.

J. F. Pinto
Manager
Nuclear Plant Engineering

JDV:gaf
cc: See Page 2

MEMO TO: C. K. McCoy
FROM: J. F. Pinto
SUBJECT: Drywell Pressure Set Points
PMI: 83/12,028
DATE: October 26, 1983

cc: J. B. Richard
J. P. McGaughy
T. E. Reaves
L. F. Dale
A. S. McCurdy
T. H. Cloninger
C. R. Hutchinson
J. W. Yelverton
S. M. Feith
W. F. Adcock
J. Cross
File (Project)
File (Plant)
File (NPE)

A12.03

MEMO TO: C. K. McCoy

FROM: J. F. Pinto

SUBJECT: Drywell Pressure Setpoint - Unit 1

RE.: PMI 83/9585 and 83/9213

PMI: 83/11,417

FILE NO.: 0290/0164/15635

DATE: October 7, 1983

OK

~~NSM~~

GGNS DATE OCT 12 83

INFO ACTION REPLY

Pls have PSRC review and provide recommended action, prepare reply to ASM JFP and memo to JPM for my sig.
CKM
Due 10/31/83

Do this on time

BACKGROUND:

NPE issued PMI 83/9213 which required changing the Setpoint for the Drywell Pressure Instruments from 1.73 to 1.23 psig due to the variations of atmospheric pressure at the Grand Gulf Nuclear Station from a low of 14.2 psia to a high of 15.2 psia. Plant Staff has issued PMI 83/9585 which disagrees with the setpoint change and justification.

DISCUSSION:

The following are the Plant Staff reasons for not changing the setpoints and NPE's reply to Plant Staff reasons:

1. Reduced setpoint and operating margins will cause unnecessary LOCA signal initiations and subject the plant to unnecessary transients.

   1A. NPE agrees with this statement and furthermore states that the long term fix to this problem should not be to "Live" with the reduced setpoints but to work with GE and the Accident Analysis to raise the allowable setpoint value.

2. The barometric pressure required to produce the low condition is a temporary condition and then only lasts for a short period of time as it will take a certain length of time for the drywell pressure to equalize with the outside atmospheric pressure. Furthermore storms of this nature usually move through a given location fairly quickly.

   2A. The barometric pressure change required to make the instrument setting nonconservative in regard to instrument drift and tech spec requirements may not be a temporary condition.

   NOTE: GGNS is located in a tornado zone.

MEMO TO: C. K. McCoy
FROM: J. F. Pinto
SUBJECT: Drywell Pressure Setpoint - Unit 1
DATE: October 7, 1983

Any time the local atmospheric pressure is below 14.7 psia the instrument setpoint is nonconservative, this can be caused by a stationary low front. As for the time for drywell pressure to equalize with atmospheric pressure the low pressure condition in drywell would last approximately as long as the condition lasts outside (since the equalization time necessary for the pressure to increase could be approximately equal to the equalization time for pressure to decrease).

In conclusion NPE contends that not lowering the drywell pressure setpoints as requested gives a much higher probability of placing the plant in a non conservative condition. This condition is real and does not justify leaving the setpoints at 1.73 psig.

NPE does agree that a design change should be made either to change the instruments or change the allowable limits form 2.0 psig to 2.5 psig for the drywell pressure trip points.

ACTION:

NPE has evaluated PMI 83/9213 and does not concur with Plant Staff recommendation for leaving the present the drywell pressure trip setpoints at 1.73 psig as prescribed in their PMI 83/9213.

J. F. Pinto
Manager
Nuclear Plant Engineering

JV:vwb

cc: J. B. Richard
J. P. McGaughy
T. E. Reaves
L. F. Dale
A. S. McCurdy
T. H. Cloninger
C. R. Hutchinson
J. W. Yelverton
File (Project)(2)
File (Plant)
File (NPE)

MEMO TO: J. F. Pinto

FROM: C. K. McCoy

SUBJECT: Drywell Pressure Setpoint - Unit 1

RE: PMI-83/9213

PMI: 83/9585

We have reviewed the subject memo and disagree that the Drywell Pressure setpoints should be lowered to 1.23 psig.

The basis for this requirement is that a low barometric of 28.95" Hg was recorded in 1969 as a result of severe weather (Hurricane Camille). This low pressure would have caused a trip setpoint greater than that allowed by Tech Specs. This is a result of having absolute pressure transmitters vs. Differential Pressure Transmitters to monitor Drywell Pressure.

Plant Staff's position of not changing the setpoints is based upon:

1) Although the letter states that adequate operating margin remains with the lower setpoints, we contend that the reduced operating margins will cause unnecessary LOCA signal initiations and thus subject the plant to unnecessary transients.

2) The low barometric pressure required to produce such a condition is a temporary condition and then only last for a short period of time as it will take a certain length of time for Drywell Pressure to equalize with atmospheric pressure and storms of this nature move through a given location fairly quickly.

In conclusion, Plant Staff contends that the lowering of the setpoints is non-conservative and the probability of this occurring is remote enough such that the setpoints should remain as is. A design change should be initiated to change the pressure transmitters to Differential Pressure Units during the first refueling outage and/or changes to the Analytical Limits be made.

C.K. McCoy

C. K. McCoy
8/31/83

ASM/CRH/CKM:pjc

cc: J. P. McGaughy
J. B. Richard
File (Project)
File (Plant)

MEMO TO: C. K. McCoy

FROM: J. F. Pinto

SUBJECT: Surveillance Test Program

System: C71-B21

FILE NO: 0290/0164/15635

PMI: 83/9213

DATE: August 18, 1983

BACKGROUND: Mechanical (I&C) was requested to review and verify set point values. RE: ITEM B15

DISCUSSION: Mechanical (I&C) has prepared a Set Point Verification Package (SPVP) B21-B, Rev. 0.
SUPERCEDES THE SETPOINT IN SPVP E12-X PMI 83/6930

ACTION: The SPVP is being issued for your use. Please review the attached SPVP and provide comments to Mechanical (I&C) for further resolution.

Any questions concerning these items should be directed to J D VOSS at extension 2710.

J. F. Pinto

J. F. Pinto
Manager
Nuclear Plant Engineering

BL/GK

JDV:BL:

Attachment

cc: J. B. Richard, w/o
J. P. McGaughy, w/o
L. F. Dale, w/o
A. S. McCurdy, w/o
T. H. Cloninger, w/o
C. R. Hutchinson, w/o
J. W. Yelverton, w/o
W. F. Adcock, w/o
T. E. Reaves, w/o
J. D. Kegerreis, w/a
G. E. Baker, w/a
D. E. Stewart, w/a
File (System), w/a
File (Project), w/a
File (Plant), w/o
File (PMI), w/o
File (NPE), w/o

NPE 02/7-83

As an interim measure to insure concurrence with Technical Specification limits and also to prevent violation of Analytical or Technical Specification limits, the Drywell Press Trip Unit Set Points shall be changed to 1.23 psig in accordance with FDDR-JB1-1936, Rev. 0 (attached). The would insure instrument trip at the proper set point.

In addition, the following changes will be made at a later date:

1. Change trip unit scales from 14.7 - 19.7 psia to 14.0 psia - 19.0 psia.

2. Revise calibration of pressure transmitters to 14.0 to 19.0 psia.

3. Revise set point to a psia value.

4. Revise the analytical limit to 2.5 psig greater than the average ambient atmospheric pressure in G.E. Design Specification Data Sheets.

GENERAL ELECTRIC

FIELD DEVIATION DISPOSITION REQUEST

DATE OF ISSUE

FDDR NO. JB1-1936

REVISION 0

SHEET 1 OF 2

PROJECT Grand Gulf UNIT 1

EQUIPMENT (MPL OR DESCRIPTION OR BOTH) Drywell Pressure Trip

FDDR ORIGINATOR J. D. Schoepf DATE 8/17/83

DOCUMENT NO.	SH NO.	REV.	TITLE
See Below			

DEVIATION DESCRIPTION

Existing setpoints for B21-694 A, B, E, F and C71-N650 A, B, C, D, may allow drywell pressure to exceed the 2 psig analytical limit.

SITE QC CONCURRENCE DATE FIELD CONCURRANCE DATE

☐ SUGGESTED DISPOSITION ☐ EXPEDITED DISPOSITION

Add note to B21-694 on drawing 22A3856AA and paragraph 4.3.11.9 on drawing 22A3771AE and any other applicable drawing stating that nominal setpoint of this absolute pressure instrument can be reduced _ .5 PSI above maximum drywell operating pressure if necessary to accomodate barometric pressure changes. This margin provides adequate spurious trip avoidance. A larger margin should be used if feasible.

QA requirements: None, Document Change Only

DISPOSITION NEED DATE

EXPEDITED DISPOSITION APPROVAL BY T.A. by O. J. Foster DATE 8/17/83

FINAL DISPOSITION

ECA/ECN FDI NO.

JUSTIFICATION OF DISPOSITION DECISION (SAFETY, RELIABILITY)

DESIGN VERIFICATION STATEMENT

VERIFIED BY DATE

APPROVALS	DATE
QUALITY	
MATL APPL ENGR	
LEAD SYSTEM ENGR	
ENGR MANAGER	
RESPONSIBLE ENGR	

DRF NO. IF APPLICABLE

APPROVALS	DATE

DISTRIBUTION CODE

INTERNAL	EXTERNAL

THIS EQUIPMENT IS SAFETY RELATED ☐ YES ☐ NO
SAFETY FUNCTION IS AFFECTED ☐ YES ☐ NO

COMPLETION RECORD REQUIRED BY R. E. ☐ YES ☐ NO

SUPPLIER ACTION REQUIRED ☐ YES ☐ NO

FIELD WORK ORDER NO.

DISPOSITION COMPLETE DATE

SITE QUALITY CONTROL

FIELD MANAGER

GENERAL ELECTRIC

SUPPLEMENTAL SHEET

FIELD DEVIATION DISPOSITION REQUEST

FDDR NO. JB1-1936

REVISION 0

SHEET 2 OF 2

DOCUMENTS REQUIRED TO BE CHANGED

DOCUMENT IDENT NO.	DOCUMENT TITLE/TYPE	SYSTEM OR PANEL MPL NO	UPDATED WITH FDDR ISSUE	
			YES	NO
22A3856AA	Design Spec Data Sh.	B21-4020		
22A3771AE	Design Spec Data Sh.	C71-4010		

A.R. SMITH; 8-16-83

THE ATTACHED 2 SHEETS ARE PROVIDED BY OJ FOSTER & MILT LANE. PLEASE REVIEW AND IF OK WITH YOU, PASS ON TO JOHN PINTO.

WITH THIS APPROACH, ENGINEERING DOES NOT BELIEVE THAT THE 2.5 PSIG ANALYTICAL LIMIT NEEDS TO BE IMPLEMENTED.

MILT LANES DATA ON SK MRL 8 15 83 PROVIDES THE NEEDED INFORMATION. O.J. FOSTERS MEMO PROVIDES AN EXPLANATION. ANY QUESTIONS, PLEASE CONTACT OJ OR MILT. MILTS EXT IS 6322
OJ'S EXT IS 1996

PRESENT SCALES ARE (ON TRIP UNITS)
0 PSIG = 14.7 PSIA
5 PSIG = 19.7 PSIA

NEW SCALES SHOULD BE 14.0 PSIA – 19.0 PS

PROPOSE REVISING DSDS (Design Spec Data Sheet) TO SHOW THIS FOR SCALES ONLY. PSIG TERMINOLOGY TO REMAIN AS IS TO AGREE WITH TECH SPEC'S.

F. Busch

O.J. Foster 8/16/83

MEMO

USE OF ABSOLUTE INSTR. FOR HIGH DRYWELL PRESS TRIPS

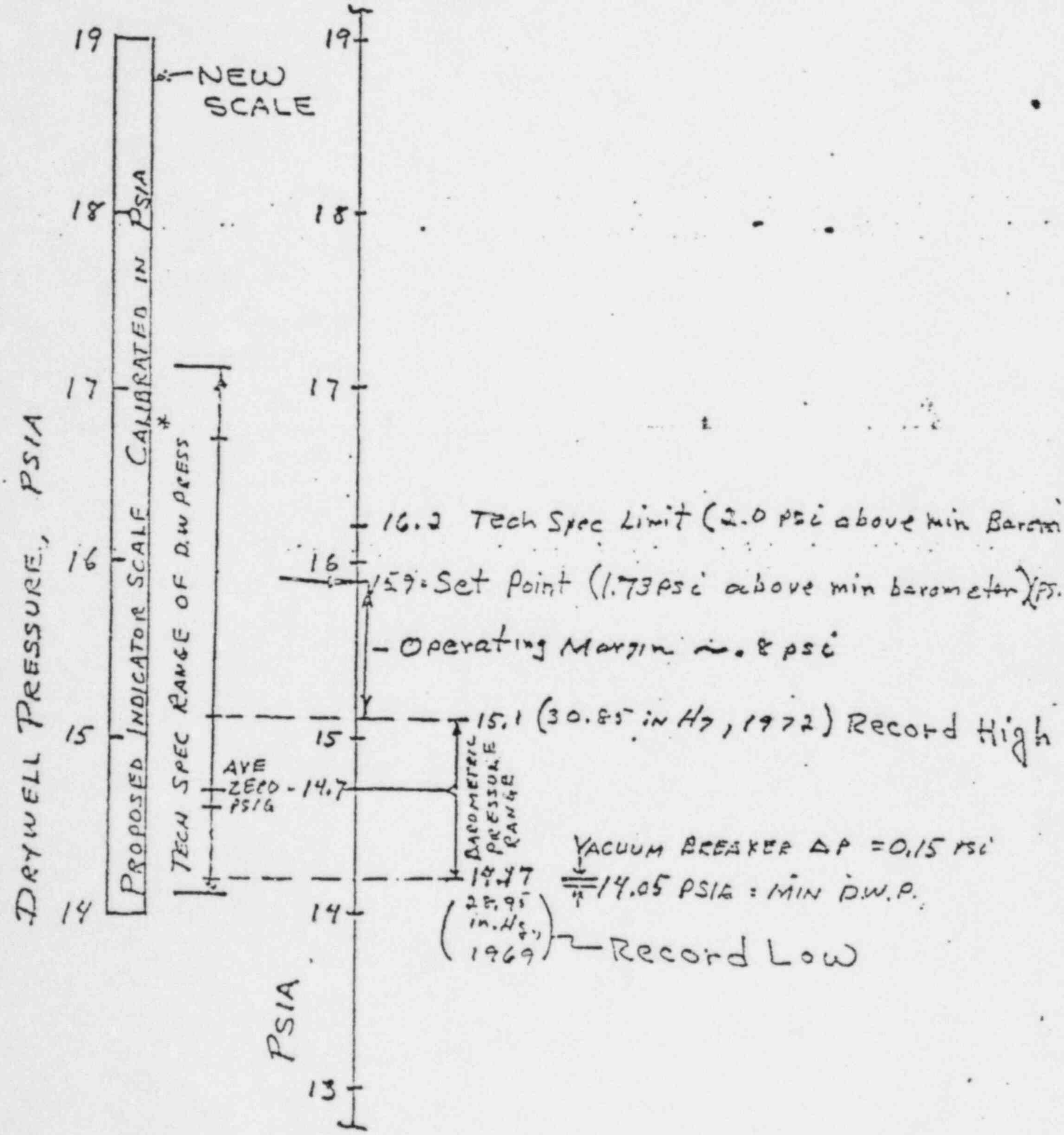
Nominal set point of this absolute press instr can be reduced to ≥ .5 psi above maximum drywell operating press., if necessary, to accommodate barometric pressure changes. This margin provides adquate spurious trip avoidance. A larger margin should be used if feasible.

This lowered nominal set point is needed to accommodate low barometric pressures vs the 2 PSIG Analytic limit. In order to preserve the instrument margins (accuracy, calibration and drift) for low barometric pressure conditions the nominal set point must be lowered such that the specified nominal in the set point table is not exceeded, in terms of PSIG, when the barometer is at its minimum.

For example (in round numbers) if the minimum baramoter ever expected is 14.2 psia and the specified nominal set point is 1.7 psig then a nominal set point of 14.2 psia + 1.7 = 15.9 psia should be used. This would still give a trip avoidance margin of 15.9 - 15.2 = .7 psi since it is not possible with installed equipment to hold drywell pressure above the maximum barometric pressure.

GENERAL ELECTRIC CO.
Nuclear Energy Business Operations
ENGINEERING CALCULATION SHEET

NUMBER SK MPL 8 15 83 DATE 8-15-83
SUBJECT GRAND GULF DRYWELL PRESSURE BY McSame SHEET ___ OF ___



TECHNICAL SPECIFICATION PROBLEM SHEET

Item Number: 016 Priority: 1A

______________ / ______________
Identified By Date

____
Responsible Supervisor

Tech Spec Reference: 3/4.3.8-3.3.8-2.1.b

Problem Title: Containment Press Setpoints

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other): According to the GE design spec. the CTMT Press setpoint/allowable valves should be 8.35/8.85 vs 9.0/9.2 The #'s in the present Tech Spec were provided to MP&L by GE during proof and review period. GE subsequently issued different #'s in the design spec revision (after receipt of license).

2. Safety Significance: This was identified by the surveillance review team in mid 1983; it was a low priority since procedural controls were in place to ensure compliance (calibration, function, channel checks procedures, etc.).

3. Anticipated Resolution: Tech Spec change to correct setpoint valves. Is incorporated in surveillance procedures.

4. NRC Response to Item (NRR/IE): ______________

NRC Notified: ______________ ______ / ______
Individual Notified Date Time

5. Disposition: ______________

Items Closed: (How) ______________

______ / ______
Date Time

cc: J. E. Cross
R. F. Rogers

"TECH SPEC PRIORITY"

Punchlist Item # 016

Tech Spec 3/4.3.8

Priority 1A

TO: Manager of Nuclear Plant Engineering

FROM: Chairman, Prioritization and Disposition Chairman

SUBJECT: Technical Specifications Punchlist Item # 016

PDTS:84/ 0006

DATE: 3/10/84

The subject Tech Spec item has been determined by the Disposition Committee to require Engineering support.

DETAILS: ~~Determine the correct containment spray pressure setpoint & the proper design spec to the [illegible] approved by NPE~~

Determine the correct containment spray pressure setpoint and allowable value per the latest NPE-approved GE design spec data sheet. Response should include spec revision used in addition to setpoint #'s

Please contact Joe Hendry at Extension 2675 for further information.

Please refer to the Tech Spec Punchlist item number in your response. Forward your response to G. Zinke.

J. C. Roberts
for Chairman

LLJ/JCR:swb
cc: Mr. C. L. Tyrone
Mr. J. E. Cross
Mr. D. Stonestreet
Mr. A. S. McCurdy
Mr. S. Hutchins
Mr. J. Hendry
File (Tech Spec Records)

A4/61swb1

"TECH SPEC PRIORITY"

MEMO TO: J. F. Pinto, Manager of Nuclear Plant Engineering

FROM: C. L. Tyrone, Project Manager

SUBJECT: Handling of Tech Spec Review Items

TSRO: 84/0001

DATE: March 11, 1984

This memorandum confirms our conversation of March 10, 1984. At that time, your assistance was requested in resolving discrepancies on eleven priority I items. Since then two items have been added. These items are all previously identified items which require early resolution with the NRC. A response is needed on these items by 12:30 PM on March 1?, 1984. A list is attached.

Furthermore, all items of any priority identified (or previously known) which are being handled on this program require expeditious handling. This includes areas where requests are originated from other interfacing organizations such as the Plant or Nuclear Services. In any case where conflicts regarding highest priority is not clear, I am available to provide clarification.

It is suggested that you arrange 7 day a week support in this area as it is needed and arrange for all NPE personnel who will be involved in this effort to be available (or on call) in a manner that will support the Tech Spec Review program.

CLT

SHH:sad
Attachment

cc: J. B. Richard (w/a
J. P. McGaughy (w/a)
J. F. Pinto (w/a)
J. E. Cross (w/a)
T. H. Cloninger (w/a)
H. J. Green (w/a)
R. C. Fron (w/a)
D. W. Stonestreet (w/a)

[redacted]
T. E. Reaves, Jr. (w/a)
S. M. Feith (w/a)
J. G. Cesare (w/a)
G. W. Smith (w/a)
L. R. McKay (w/a)
L. C. Burgess (w/a)
File (Tech Spec Records) (w/a)

LIST OF CURRENT PRIORITY 1
ITEMS REQUIRING NPE SUPPORT

PDTS:84/	P/L #	Date Sent
001	199	3/10/84
002	180	3/10/84
003	033	3/10/84
004	054	3/10/84
005	001	3/10/84
006	016	3/10/84
007	015	3/10/84
008	198	3/10/84
009	202	3/10/84
010	213	3/10/84
011	219	3/10/84
012	083	3/10/84
013	168	3/10/84

TECHNICAL SPECIFICATION PROBLEM SHEET

Item Number: 019 Priority: ZIA

_____ / _____ _____
Identified By Date Responsible Supervisor

Tech Spec Reference: 3/4.6.7.3.b.1

Problem Title: Drywell Purge Flowrate Definition

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other): Surveillance requires verifying a flow rate of 1000 cfm every 18 months. However, the flow rate should be considered in scfm, since the flow rate is temperature dependent. This could effect equipment enviromental qualifications.

2. Safety Significance: Changing from "cfm" to "scfm" would clarify flow requirements for surveillance tests.

3. Anticipated Resolution: Change "1000 cfm" to "1000 scfm."
Incorporate in surveillance procedure.

4. NRC Response to Item (NRR/IE): _____

NRC Notified: _____ _____ / _____
Individual Notified Date Time

5. Disposition: _____

Items Closed: (How) _____

_____ / _____
Date Time

cc: J. E. Cross
R. F. Rogers

TECHNICAL SPECIFICATION PROBLEM SHEET

Item Number: 022 Priority: 1 IC

_______________ / _______________

Identified By Date Responsible Supervisor

Tech Spec Reference: 3/4.3.4.1 (3.3.4.1)

Problem Title:

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other):
The associated action statements are not consisted with system design. For example if one trip channel is inop, then the inop channel is suppose to be tripped, this action will trip one Recire pump. If two channels are inop. (depending on channels) then the trip system may declared inop. with 72 hours time period to declare it operable.

2. Safety Significance: The action statement for a less serve condition is more restrictive and could cause undue plant shutdown transients.

3. Anticipated Resolution: Tech Spec. change to modify action statements to fit plant design. Could Use Tech Spec Position Statement

4. NRC Response to Item (NRR/IE): _______________

NRC Notified: _______________ _______ / _______

Individual Notified Date Time

5. Disposition: _______________

Items Closed: (How) _______________

_______ / _______

Date Time

cc: J. E. Cross
R. F. Rogers

SUBJECT: Technical Specification 3.3.4.1, page 3/4 3-34

No. -22

DISCUSSION: The present action statements for the anticipated transient without scram recirculation pump trip (ATWS-RPT) system instrumentation requires more stringent action to be taken for one channel being inoperable than for two. Action b for Technical Specification 3.3.4.1 requires an inoperable channel to be placed in the tripped condition when the number of operable channels are one less than required for one or both trip systems. Placing any channel in the tripped condition will trip a recirculation pump. small c Action C.2 of Technical Specification 3.3.4.1 only requires a trip system to be declared inoperable if two reactor vessel water level channels or two reactor vessel pressure channels are inoperable. With one trip system inoperable, it must be restored to operable status within 72 hours or be in at least ~~a~~ startup within the next 6 hours. All caps The action requirements should be less stringent when the number of operable channels are one less than required per trip system than when there are two less than required per trip system.

Small c's

The proposed change deletes action statements b, C.1, and C.2. The first paragraph of action C is modified to read as follows:

Strike ~~b~~.c. With the number of OPERABLE channels one or more less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip system(s) declare the affected Trip System(s) inoperable.

The action statements are re-numbered to account for the deletion of action b.

JUSTIFICATION: The following shows the logics and definitions of the terms Channel, Trip System, and Trip Function for ATWS-RPT instrumentation.

Need brackets and lines

Trip Unit	Parameter		
B21-LIS-N699A	RPV Level - Lvl 2		Trip Pump
	Remove Strike	Any	C001A
B21-PIS-N658A	RPV Press - Hi *	One	
B21-LIS-N699B	RPV Level - Lvl 2	Any	Trip Pump
B21-PIS-N658B	RPV Press - Hi	One	C001B
			TRIP FUNCTION
		TRIP SYSTEM	
B21-LIS-N699E	RPV Level - Lvl 2	Any	Trip Pump
B21-PIS-N658E	RPV Press - Hi	One	C001A
B21-LIS-N699F	RPV Level - Lvl 2	Any	Trip Pump
B21-PIS-N658F	RPV Press - Hi	One	C001B
			TRIP FUNCTION
		TRIP SYSTEM	

* One Channel (Typical of 8 shown on this page)

K20sd1

As shown above, placing any channel in the tripped condition will trip a Recirculation Pump. In order to avoid this pump trip, the proposed change to Technical Specification 3.3.4.1 is made to require a Trip System to be declared inoperable instead of tripping a channel when the number of channels is one or more less than required by the minimum operable channels per trip system requirement. Present action d or e will remain and apply when one or both trip systems are declared inoperable.

SIGNIFICANT HAZARDS CONSIDERATION:

The proposed change does not affect the requirements of present action d or e concerning one or both trip systems being inoperable for ATWS-RPT. However, this change requires that an entire trip system be declared inoperable if one or more of its channels is inoperable thus avoiding a direct recirculation pump trip from placing a channel in the tripped condition. The intent of present action statements in Technical Specification 3.3.4.1 is not to trip a recirculation pump when channels are inoperable but to allow time for restoration of the inoperable function if only one trip system is affected. This change will allow the recirculation pumps to continue in operation as the inoperable channel(s) is repaired or the plant is taken to STARTUP per present action c or d. These changes do not involve a) the reduction of safety margins, b) an increase in the probability or consequences of a previously evaluated accident, or c) the possible creation of a new or different kind of accident. Thus the proposed changes do not involve a significant hazards consideration.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.1 The anticipated transient without scram recirculation pump trip (ATWS-RPT) system instrumentation channels shown in Table 3.3.4.1-1 shall be OPERABLE with their trip setpoints set consistent with values shown in the Trip Setpoint column of Table 3.3.4.1-2.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION:

a. With an ATWS-RPT system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.1-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel setpoint adjusted consistent with the Trip Setpoint value.

~~b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within one hour.~~

b ~~c.~~ With the number of OPERABLE channels one ~~two~~ or more less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip system(s) ~~and:~~ declare the affected trip system(s) inoperable.

~~1. If the inoperable channels consist of one reactor vessel water level channel and one reactor vessel pressure channel, place both inoperable channels in the tripped condition within one hour.~~

~~2. If the inoperable channels include two reactor vessel water level channels or two reactor vessel pressure channels, declare the trip system inoperable.~~

c ~~d.~~ With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

d ~~e.~~ With both trip systems inoperable, restore at least one trip sytem to OPERABLE status within one hour or be in at least STARTUP within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.3.4.1.1 Each ATWS recirculation pump trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.4.1-1.

4.3.4.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

DEFINITIONS FOR "CHANNELS", "TRIP SYSTEMS", AND "TRIP FUNCTIONS" FOR ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION TABLE 3.3.4.1-1

Trip Unit	Parmeter	Logic	
B21-LIS-N699A	RPV Level - Lvl 2	Any One	Trip Pump C001A
B21-PIS-N658A	RPV Press - HI *		
B21-LIS-N699B	RPV Level - Lvl 2	Any One	Trip Pump Coo1B
B21-PIS-N658B	RPV Press - HI		
		TRIP SYSTEM	TRIP FUNCTION
B21-LIS-N699E	RPV Level - Lvl 2	Any One	Trip Pump Coo1A
B21-PIS-N658E	RPV Press - HI		
B21-LIS-N699F	RPV Level - Lvl 2	Any One	Trip Pump Coo1B
B21-PIS-N658F	RPV Press - HI		
		TRIP SYSTEM	TRIP FUNCTION

* One Channel (Typical of 8 shown on this page)

TECHNICAL SPECIFICATION PROBLEM SHEET

Item Number: 054 Priority: 1A

NRC(I&C plus NRR) /
Identified By Date Responsible Supervisor

Tech Spec Reference: 3/4.3.8

Problem Title: CTMT Spray Min Operable Channel

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other):
Tech Spec Table 3.3.8-1 presently requires one min. operable channel for the Containment Spray System. The NRC recommends that the number of min operable channels be changed to 2. In addition the requirement for Reactor Vessel Water Level LLL,L1 should be deleted as it really does not apply.

2. Safety Significance:

3. Anticipated Resolution: Review and determine if appropriate. Tech Spec change pending review. *Could use Tech Spec Position Statement*

4. NRC Response to Item (NRR/IE): ______

NRC Notified: ______ Individual Notified ______ / ______ Date Time

5. Disposition: ______

Items Closed: (How) ______

______ / ______
Date Time

cc: J. E. Cross
R. F. Rogers

Item No. 054

TECHNICAL SPECIFICATION PROBLEM SHEET

Priority 1A

NRC/TRC (thru NRR) 1/21/84
Identified By Date

OK
Responsible Supervisor

Tech Spec Reference: 3/4.3.8

Problem Title: CTMT Spray Min Operable Channel

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other):

~~NRC~~ Tech Spec Table 3.3.8-1 presently requires the one min operable channel for the Containment Spray System. The NRC recommends that the number of min operable channels / be changed to 2. In addition the requirement for Reactor Vessel Water Level LLL L1 should be deleted as it really does not apply

2. Safety Significance:

3. Anticipated Resolution: Review and determine if appropriate. Tech Spec Change Being Issued.

4. NRC Response to Item (NRR/IE):

NRC Notified: ________ Individual Notified ________ Date / Time

5. Disposition:

Items Closed: (How)

Date / Time

cc: J. E. Cross
R. F. Rogers

"TECH SPEC PRIORITY"

Punchlist Item # 054

Tech Spec 3/4.3.8

Priority 1A

TO: Manager of Nuclear Plant Engineering

FROM: Chairman, Prioritization and Disposition Chairman

SUBJECT: Technical Specifications Punchlist Item # 054

PDTS:84/ 0004

DATE: 3/10/84

The subject Tech Spec item has been determined by the Disposition Committee to require Engineering support.

DETAILS: Verify that the min. operable channels for Containment Spray in table 3.3.8-1 should be (2) rather than the present (1). Reference Bechtel I&C Instrumentation Review.

Please contact Joe Hendry at Extension 2675 for further information.

Please refer to the Tech Spec Punchlist item number in your response. Forward your response to ~~Joe Hendry~~ G. Zinke.

J. C. Roberts
for Chairman

LLJ/JCR:swb
cc: Mr. C. L. Tyrone
Mr. J. E. Cross
Mr. D. Stonestreet
Mr. A. S. McCurdy
Mr. S. Hutchins
Mr. J. Hendry
File (Tech Spec Records)

A4/51swb1

"TECH SPEC PRIORITY"

Punchlist Item # See ATTached

Tech Spec See ATTached

Priority See ATTached

TO: Manager of Nuclear Plant Engineering

FROM: Chairman, Prioritization and Disposition Chairman

SUBJECT: Technical Specifications Punchlist Item # See ATTached

PDTS:84/ 0014

DATE: 3/10/84

The subject Tech Spec item has been determined by the Disposition Committee to require Engineering support.

DETAILS: This letter identifies requested response dates for the following Tech Spec problems:

#199 Letter No. PDTS 84/0001	#015 Letter No. PDTS 84/0007
#180 Letter NO. PDTS 84/0002	#198 Letter No. PDTS 84/0008
#033 Letter NO. PDTS 84/0003	#202 Letter NO. PDTS 84/0009
#054 Letter NO. PDTS 84/0004	#213 Letter NO. PDTS 84/0010
#001 Letter NO. PDTS 84/0005	#219 Letter NO. PDTS 84/0011
#016 Letter NO. PDTS 84/0006	#11.8 Letter NO. PDTS 84/0013

It is requested that the responses to the above items be completed by 3/13/84

Please contact Jerry RoberTs for further information. at Extension 2695

Please refer to the Tech Spec Punchlist item number in your response. Forward your response to George ZINKE

J. C. Roberts
Chairman

LLJ/JCR:swb
cc: Mr. C. L. Tyrone
Mr. J. E. Cross
Mr. D. Stonestreet
Mr. A. S. McCurdy
Mr. S. Hutchins
Mr. J. Hendry
File (Tech Spec Records)

A4/61swb1

"TECH SPEC PRIORITY"

MEMO TO: J. F. Pinto, Manager of Nuclear Plant Engineering

FROM: C. L. Tyrone, Project Manager

SUBJECT: Handling of Tech Spec Review Items

TSRO: 84/0001

DATE: March 11, 1984

This memorandum confirms our conversation of March 10, 1984. At that time, your assistance was requested in resolving discrepancies on eleven priority 1 items. Since then two items have been added. These items are all previously identified items which require early resolution with the NRC. A response is needed on these items by 12:30 PM on March 11, 1984. A list is attached.

Furthermore, all items of any priority identified (or previously known) which are being handled on this program require expeditious handling. This includes areas where requests are originated from other interfacing organizations such as the Plant or Nuclear Services. In any case where conflicts regarding highest priority is not clear, I am available to provide clarification.

It is suggested that you arrange 7 day a week support in this area as it is needed and arrange for all NPE personnel who will be involved in this effort to be available (or on call) in a manner that will support the Tech Spec Review program.

CLT

SHH:sad
Attachment

cc: J. B. Richard (w/a
J. P. McGaughy (w/a)
J. F. Pinto (w/a)
J. E. Cross (w/a)
T. H. Cloninger (w/a)
H. J. Green (w/a)
R. C. Fron (w/a)
D. W. Stonestreet (w/a)

[redacted]
T. E. Reaves, Jr. (w/a)
S. M. Feith (w/a)
J. G. Cesare (w/a)
G. W. Smith (w/a)
L. R. McKay (w/a)
L. C. Burgess (w/a)
File (Tech Spec Records) (w/a)

LIST OF CURRENT PRIORITY 1
ITEMS REQUIRING NPE SUPPORT

PDTS:84/	P/L #	Date Sent
001	199	3/10/84
002	180	3/10/84
003	033	3/10/84
004	054	3/10/84
005	001	3/10/84
006	016	3/10/84
007	015	3/10/84
008	198	3/10/84
009	202	3/10/84
010	213	3/10/84
011	219	3/10/84
012	083	3/10/84
013	168	3/10/84

GRAND GULF UNIT 1 TECHNICAL SPECIFICATIONS
-RECOMMENDED CHANGES-

1) Safety Limit 2.1.4 Reactor Vessel Water Level
Page 2-2

Why initiate ECCS after depressurizing? Will not depressurization worsen the problem since more water will flash into steam, thereby further decreasing the water level?

D53

Recommendation:

Initiate ECCS to restore the water level to above the actual indicated fuel level and then depressurize, only if necessary.

2) Technical Specification 3.3.7.3 Meteorological Instrumentation
Page 3/4 3-64, 65

Regulatory Guide 25 requires two levels of indication of air temperature on the met tower.

Recommendation:

Add the 162 ft. level instrument to the air temperature.

3) Technical Specification 3.3.8 Plant System Actuation
Page 3/4 3-98

There is only one (1) channel required for the Containment Spray System.

Recommendation:

Require 2 channels as minimum OPERABLE.

4) Technical Specification 3.4.4 Chemistry
Page 3/4 4-12 4.4.4.C

There is presently an allowance for obtaining an in-line conductivity measurement for up to 31 days if the continuous monitor is inoperable. This doesn't state what action is to be taken after the 31 days.

C55

Recommendation:

Delete "for up to 31 days"

5) Technical Specification 3.5.1 ECCS - OPERATING
Page 3/4 5-1, 5-5, 5-6, 5-7

The normal suction flow path for the HPCS and RCIC pumps is the condensate storage tank and there is required to be an OPERABLE automatic switchover if a low-level in the CST or a high level in the suppression pool exists. The analysis assumes the water is taken from the suppression pool, however, the preferred source is the CST. There are already surveillance requirements requiring the verification of this system operation.

T22rg1

DEFINITIONS FOR
"CHANNELS", "TRIP SYSTEMS", AND "TRIP FUNCTIONS"
FOR PLANT SYSTEMS ACTUATION INSTRUMENTATION TABLE 3.3.8-1

Trip Unit	Parameter	Logic	
B21-LIS-N691A	RPV Level Low-Level 1	One Out of Two Twice	Start RHR A Open Valve F042A Starts Timer K93A
B21-LIS-N691E	RPV Level Low-Level 1		
B21-PIS-N694A	Drywell Pressure-High*	Either One (One out of two twice)	
B21-PIS-N694E	Drywell Pressure-High		
E12-PIS-N662A	Containment Pressure-High	Either One	
E12-PIS-N662C	Containment Pressure-High	All (Both)	Open Vlv. F028A Close Vlv F042A. TRIP FUNCTION
E12-K93A	Timer System A (10 MIN)		
		TRIP SYSTEM	

Trip Unit	Parameter	Logic	
B21-LIS-N691B	RPV Level Low-Level 1	One Out of Two Twice	Start RHR Pump B RHR Pump C Open F042B and F042C Starts Timer K93B
B21-LIS-N691F	RPV Level Low-Level 1		
B21-PIS-N694B	Drywell Pressure - High	Either One (One Out of Two Twice)	
B21-PIS-N694F	Drywell Pressure - High		
E12-PIS-N662D	Containment Pressure-High	Either One	
E12-PIS-N662E	Containment Pressure-High		
E12-K93B	Timer System B (10 MIN)	All (Both)	Close F042B and F028B TRIP FUNCTION
		TRIP SYSTEM	

Feedwater System/Main Turbine Trip System			
C34-LSH-K624A	RPV Level High - Level 8	Two out of Three	Trip Rx Feed Pumps, and Main Turbine TRIP FUNCTION
C34-LSH-K624B	RPV Level High - Level 8		
C34-LSH-K624C	RPV Level High - Level 8		
		TRIP SYSTEM	

* One Channels (Typical of 17 shown on this page)

NOTE: High Drywell Pressure & RPV Level Low One Out of Two Twice Logic Starts RHR Pump and Opens F042 Valves. Then the Containment Pressure High and the Timer for the System Closes F042A and B and Opens F028A and B. This logic drawing does not show RHR A & B Pump start timers, containment spray timer or manual initiation for RHR in LPCI mode or manual initiation for containment spray.

TECHNICAL SPECIFICATION PROBLEM SHEET

Item Number: 078 Priority: 1A

_____ / _____ Identified By Date

_____ Responsible Supervisor

Tech Spec Reference: 3/4.3.5 Table 3.3.5-1.a

Problem Title: RCIC Min. Operable Channels

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other):
Change minimum operable channels for RCIC Level 2 trip from 2 to 4. Since there is only one trip syste, the ACTION statement referring to two trip systems should be deleted.

2. Safety Significance:
Administrative controls heed to be in place ensure the operator knows all channels of LL2 logic are required for operability in order to ensure proper compliance.

3. Anticipated Resolution:
Change Tech Specs as described above.
Could Use Tech Spec. Position statement, why it have 1 out of 2 twice

4. NRC Response to Item (NRR/IE): _____

NRC Notified: _____ Individual Notified _____ / _____ Date Time

5. Disposition: _____

Items Closed: (How) _____

_____ / _____ Date Time

cc: J. E. Cross
R. F. Rogers

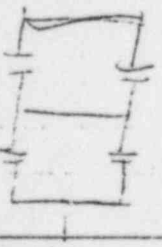
TRANSMITTAL OF PROPOSED CHANGES
TO GRAND GULF TECHNICAL SPECIFICATIONS

1. SUBJECT: Technical Specification Table 3.3.5-1, pages 3/4 3-45 and 3/4 3-46.

DISCUSSION: The Reactor Core Isolation Cooling (RCIC) system actuates on Reactor Vessel Water Level-Low Low, Level 2 with a one-out-of-two-twice logic in a single trip system. The present 2 minimum OPERABLE channels per trip system for Reactor Vessel Water Level-Low Low, Level 2 should be changed to 4 minimum OPERABLE channels per trip system. Present ACTION 50 should be changed to reflect only one trip system. ACTION 50 should be changed to read as follows:

ACTION 50 - With the number of OPERABLE channels less than required by the minimum OPERABLE channels per Trip System requirement for one trip system, place the inoperable channel(s) or that trip system in the tripped condition within one hour or declare the RCIC system inoperable.

JUSTIFICATION: RCIC initiates on low reactor water level, Level 2. The initiation logic is arranged as one trip system with four water level signals feeding a one-out-of-two-twice logic. The present requirement of 2 minimum OPERABLE channels per trip system does not ensure a success path for RCIC initiation unless the correct two channels are operable. To insure that the success path is always maintained, the minimum OPERABLE channels per trip system for Reactor Vessel Water Level-Low Low, Level 2 should be increased to four which is the number of channels installed in the plant.



Present ACTION 50 addresses two trip systems and does not reflect the "one trip system" design. New ACTION 50 addresses the "one trip system" design and matches the RCIC trip system design. New ACTION 50 allows up to two of four channels to be placed in the tripped condition before RCIC must be declared inoperable. This ACTION does not degrade system operability but is conservative since the trip system is closer to actuation with the channel(s) in the tripped condition. This Technical Specification change is proposed as followup action to MP&L's discussions with Instrumentation and Control Systems Branch on this matter as committed to in AECM-83/0519, dated September 12, 1983.

SIGNIFICANT HAZARDS CONSIDERATION:

This Technical Specification change is made to enhance RCIC system operability and to increase the number of channels required OPERABLE for actuation. The new ACTION 50 is proposed to reflect system design of only one trip system. The changes proposed are conservative when compared to present technical specification requirements. This change does not involve the reduction of safety [illegible] significant increase in the [illegible] previously evaluated is involved nor is the possibility of a new or different kind of accident from any accident previously evaluated created.

[illegible] not involve any significant hazards consideration.

TABLE 3.3.5-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

FUNCTIONAL UNITS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[a]	ACTION
a. Reactor Vessel Water Level - Low Low, Level 2	~~2~~ 4	50
b. Reactor Vessel Water Level - High, Level 8	2[b]	51
c. Condensate Storage Tank Water Level - Low	2[c]	52
d. Suppression Pool Water Level - High	2[c]	52
e. Manual Initiation	1/system[d]	53

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

(b) One trip system with two-out-of-two logic.

(c) One trip system with one-out-of-two logic.

(d) One trip system with one channel.

ITEM 1.

ITEM 1.

TABLE 3.3.5-1 (Continued)

REACTOR CORE ISOLATION COOLING SYSTEM

ACTUATION INSTRUMENTATION

ACTION 50 - ~~With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement:~~

~~a. For one trip system, place the inoperable channel in the tripped condition within one hour or declare the RCIC system inoperable.~~

~~b. For both trip systems, declare the RCIC system inoperable.~~

With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel(s) or that trip system in the tripped condition within one hour or declare the RCIC System inoperable.

ACTION 51 - With the number of OPERABLE channels less than required by the minimum OPERABLE channels per Trip System requirement, declare the RCIC system inoperable.

ACTION 52 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, place at least one inoperable channel in the tripped condition within one hour or declare the RCIC system inoperable.

ACTION 53 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, restore the inoperable channel to OPERABLE status within 8 hours or declare the RCIC system inoperable.

Item No. 78

TECHNICAL SPECIFICATION PROBLEM SHEET

Priority 1a

Identified By _______ / Date _______

Responsible Supervisor _______ OK

Tech Spec Reference: 3/4.3.5 Table 3.3.5-1.a

Problem Title: RCIC Min. Operable Channels

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other): Change minimum operable channels for RCIC Level 2 trip from 2 to 4. Since there is only one trip system, the ACTION statement referring to two trip systems should be deleted.

2. Safety Significance: Administrative controls need to be in place to ensure the operator knows all channels of LL2 logic are required for operability in order to ensure proper compliance.

3. Anticipated Resolution: Change Tech Spec as described above.

4. NRC Response to Item (NRR/IE):

NRC Notified: Denton (LECM 83/0642) — Individual Notified; 10/11/83 — Date; Time

5. Disposition:

Items Closed: (How)

Date / Time

cc: J. E. Cross
R. F. Rogers