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September 27, 1983

WRITER'S DIRECT DIAL NUMBER

822-1026

Mr. Samuel J. Chilk
Secretary
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

In the Matter of
Metropolitan Edison Company
(Three Mile Island Nuclear Station, Unit No. 1)
Docket No. 50-289 (Restart)

Dear Mr. Chilk:

Please find enclosed copies of the following documents, which include information potentially relevant and material to matters under adjudication in the plant design and procedures phase of this proceeding, which is now before the Commission:

1. Letter 5211-83-219, August 15, 1983, H. D. Hukill, GPU Nuclear, to D. G. Eisenhut, NRC, Auto RC Pump Trip (NUREG 0737, II.K.3.5);

2. Letter 5211-83-232, August 23, 1983, H. D. Hukill, GPU Nuclear to J. F. Stolz, NRC, Long Term EFW Mods (NUREG 0737, II.E.1.1);

3. Letter 5211-83-250, September 7, 1983, H. D. Hukill, GPU Nuclear, to J. F. Stolz, NRC, 25°F Subcooling Margin; and

DS03

4. Letter 5211-83-243, September 9, 1983, H. D. Hukill, GPU Nuclear, to J. F. Stolz, NRC, Relief and Safety Valve Testing (NUREG 0737, II.D.1).

Respectfully submitted,

George F. Trowbridge

George F. Trowbridge
Counsel for Licensee

Enclosures
cc: Service List
GFT/lam

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE COMMISSION

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY	)	Docket No. 50-289
	)	(Restart)
(Three Mile Island Nuclear	)	
Station, Unit No. 1)	)	

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August 15, 1983
5211-83-219

SEP 2 1 1983
S.P.P. & T.

Office of Nuclear Reactor Regulation
Attn: D. G. Eisenhut, Director
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Sir:

Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
Auto RC Pump Trip (NUREG 0737, II.K.3.5)

Our letter of March 31, 1983 (5211-83-017) notified you of our plans to revise the RCP trip criterion from 1600 psig ESAS to 25°F subcooling margin. This has been accomplished. Our letter of June 8, 1983 also advised you of our intent to join the B&W Owners Group on this subject to further analyze and quantify the margins associated with the new criterion we have adopted. Enclosed is a description of the plan for the submission of the supplemental information consistent with the other B&W Owners. GPU will provide the information consistent with other B&W Owners in May, 1984.

Sincerely,

R J Toole for

H. D. Hukill
Director, TMI-1

HDH:LWH:vjf

cc: R. Conte
J. F. Stolz
J. Van Vliet
B. Sheron

PLAN FOR RESOLUTION OF TMI
ACTION ITEM II.K.3.5

"Automatic Trip of Reactor Coolant Pumps"

INTRODUCTION

The criteria for resolution of NUREG-0737, Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps", are provided in a letter from D. G. Eisenhut (NRC) to GPU Nuclear on March 4, 1983. As discussed in our letter of March 31, 1983, GPUN has revised the RCP Trip Criteria from 1600 psig ESAS to 25°F subcooling margin under 10CFR 50.59. The B&W Owners Group has been formulating a plan to demonstrate compliance with those criteria. The following represents this overall position and plan.

PLAN FOR TREATMENT OF RC PUMP OPERATION

The treatment of reactor coolant pumps during accidents and transients has received extensive attention over the past several years. THE B&W Owners Group has performed analyses evaluating the effect of a delayed RC pump trip using Appendix K assumptions during the course of a small break LOCA accident and has determined that an early trip of RC pumps is required to show conformance to 10CFR 50.46 for a range of break sizes. Therefore, to be consistent with the conservative analyses performed, it is our position that the reactor coolant pumps should be tripped if indications of a small break LOCA exist.

The B&W Owners Group and B&W maintain that it is highly desirable to maintain RC pump operation during non-LOCA events, as an aid in the mitigation of transients. Consistent with this phiolosophy, the concept of subcooling margin was chosen as an indicator for the need to trip RC pumps. It is our intention to demonstrate that this concept is consistent with our philosophy for handling RC pumps during transient conditions and complies with the intent of the criteria stated in your letter of March 4, 1983. The symptom approach of subcooling margin, developed as part of the Abnormal Transient Operating Guidelines Program, is intended to replace the present guidelines of tripping solely on the presence of a low RC pressure ESFAS signal.

It is the position of the B&W Owners Group and B&W that reactor coolant pump trip can be achieved safely and reliably by the operator. It has been determined that a loss of subcooling margin will occcur for those SBLOCAs where a pump trip is required for compliance with 10CFR 50.46.

The B&W Owners Group will undertake a program based on the above positions to demonstrate that the concept of subcooling margin is an appropriate indicator of the need to trip RC pumps, yet still allows continued RCP operation for steam generator tube ruptures (SGTR). The concept of subcooling margin will be examined for the more likely non-LOCA transients to demonstrate that under realistic conditions an indication requiring RC pump trip is unlikely.

This program is also intended to provide the justification for manual RCP trip on indication of loss of subcooling margin. Tripping on loss of subcooling margin will assure pump trip prior to the development of significant system voids. No attempt will be made to demonstrate acceptability of continued RCP operation during small break conditions. No request for an exemption of 10CFR 50.46 will be made to allow continued RCP operation during SBLOCA.

The specific plan for resolution of the RC pump trip issue is structured to address the specific criteria stated in the March 4, 1983 letter. A description of the plan, related to the criteria with which it is intended to address, follows:

I. <u>Pump Operation Criteria Which Can Result in RCP Trip During Transients and Accidents</u>

1. Setpoints for RCP Trip:

a. The RCP trip criterion, based on loss of subcooling margin, was developed with the intent of assuring that an indication for RC pump trip would occur for those SBLOCAs where pump trip was required to meet the criteria of 10CFR 50.46. A spectrum of analyses has been performed using Appendix K assumptions which demonstrate that a loss of subcooling will always occur for small breaks that have the potential to uncover the core and exceed 10 CFR 50.46 criteria if the RCPs are tripped under certain two-phase conditions. Therefore, loss of subcooling can be used as an indicator of the need for RCP trip. The actual value of the setpoint (25°F) will be verified to ensure that this indicator will allow continued forced RCS flow during realistic SGTRs up to and including the design basis SGTR - a single double ended rupture. The setpoint will also be verified to include consideration for minimizing the indication for need to trip RC pumps for more likely non-LOCA events such as a mild overcooling transients.

No partial or staggered RCP trip schemes will be considered except for the extreme case where mechanical damage to the pump is likely as this adds to increased decision making on the part of the operator during transient conditions.

b. The RCP trip criterion based on subcooling margin precludes operation of the RC pumps in a highly voided system (except for ICC conditions).

c. A primary objective of the parameter and setpoint verification is the avoidance of reactor coolant pump trip for non-LOCA events particularly SGTR. Realistic operator actions in accordance with the procedures are expected to avoid loss of subcooling and the need to trip the reactor coolant pumps for this event. Furthermore, since subcooling margin would be quickly regained following makeup or HPI initiation, without loss of natural circulation even if the operator failed to take actions to prevent RCP tripping and ESFAS actuation, restart of the pumps would be allowed. Consequently, reliance on the PORV for depressurization is unlikely.

d. The significance of primary system voiding due to flashing of hot coolant is disucssed as part of operator training. The subject void treatment is being supplemented by additional guidance on prevention, detection, and mitigation of voids. This is considered outside of the ATOG scope but will be addressed.

e. Actions following containment isolation signals will be reviewed to ensure consistency in the treatment of availability of cooling water and seal injection to prevent pump damage. Instructions for pump trip are provided in the ATOG guidelines in the unlikely event of mechanical pump damage. Crtieria for restart of RC pumps include assuring that cooling water and seal injection are available. Existing TMI-1 procedures also include the guidance.

f. Instructions for maintaining or reinitiating forced RC flow are contained in ATOG for ICC conditions.

2. Guidance for Justification of Manual RCP Trip

a. A spectrum of small break LOCAs has been analyzed for 177 and 205 FA plant types using the CRAFT2 code. Using the Appendix K evaluation techniques, there exists a combination of break sizes and RC pump trip times which result in exceeding 10 CFR 50.46 limits. For the worst break size, i.e., that size which requires the earliest pump trip, trip must occur within 2 minutes of the indication of need for pump trip. As break size decreases, more time is available for operator action. the critical time period of high void formation (>70%) when RC pump trip is not recommended, has also been determined. The critical time period for the break requiring the earliest operation action time is short (5 minutes) when pump trip could result in exceeding 10 CFR 50.46 criteria.

b. A best estimate SBLOCA analysis will be performed for each general plant type, over the spectrum of sizes determined by the conservative analyses to determine (a) the time available for a required RC pump trip, and the period of time when RC pump trip is not recommended or (b) the lack of indication for a required pump trip. If it is determined that a need for RC pump trip exists, the time for operator action will be determined and justified by comparison to ANSI Standards and operating experience. An indication of reasonable operator action time is expected to justify manual RCP trip.

3. Other Considerations

a. The level of quality of instrumentation, as described in the enclosure to the March 4, 1983 letter, used to produce the signal indicating the need for RC pump trip, will be provided by GPUN to supplement the B&WOG generic submittal for treatment of RC pumps during transients.

b. The ATOG guidelines and plant specific Emergency Operating Procedures contain criteria for the timely restart of reactor coolant pumps when conditions which will support safe pump operation exist. Table 6 of the Equipment Operator chapter of ATOG provides the conditions when RC pumps can be restarted.

c. Plant operators have been trained in their responsibility for performing RCP trip in the event of a small break LOCA. Current plant procedures (non-ATOG) require RC pump trip on 25°F subcooling margin. Instructions for plant operation are reinforced by regular requalification class and simulator training. Operators have been trained on the concept of RC pump trip on subcooling margin.

II. Pump Operation Criteria Which Will Not Result in RCP Trip During Transients and Accidents

Since it is the position of the B&WOG and B&W that the safest method for RC pump operation following SBLOCA is (manual) trip, the criteria stated in this section will not be addressed.

PLAN FOR RESOLUTION OF TMI
ACTION ITEM II.K.3.5

"Automatic Trip of Reactor Coolant Pumps"

INTRODUCTION

The criteria for resolution of NUREG-0737, Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps", are provided in a letter from D. G. Eisenhut (NRC) to GPU Nuclear on March 4, 1983. As discussed in our letter of March 31, 1983, GPUN has revised the RCP Trip Criteria from 1600 psig ESAS to 25°F subcooling margin under 10CFR 50.59. The B&W Owners Group has been formulating a plan to demonstrate compliance with those criteria. The following represents this overall position and plan.

PLAN FOR TREATMENT OF RC PUMP OPERATION

The treatment of reactor coolant pumps during accidents and transients has received extensive attention over the past several years. THE B&W Owners Group has performed analyses evaluating the effect of a delayed RC pump trip using Appendix K assumptions during the course of a small break LOCA accident and has determined that an early trip of RC pumps is required to show conformance to 10CFR 50.46 for a range of break sizes. Therefore, to be consistent with the conservative analyses performed, it is our position that the reactor coolant pumps should be tripped if indications of a small break LOCA exist.

The B&W Owners Group and B&W maintain that it is highly desirable to maintain RC pump operation during non-LOCA events, as an aid in the mitigation of transients. Consistent with this phiolosophy, the concept of subcooling margin was chosen as an indicator for the need to trip RC pumps. It is our intention to demonstrate that this concept is consistent with our philosophy for handling RC pumps during transient conditions and complies with the intent of the criteria stated in your letter of March 4, 1983. The symptom approach of subcooling margin, developed as part of the Abnormal Transient Operating Guidelines Program, is intended to replace the present guidelines of tripping solely on the presence of a low RC pressure ESFAS signal.

It is the position of the B&W Owners Group and B&W that reactor coolant pump trip can be achieved safely and reliably by the operator. It has been determined that a loss of subcooling margin will occcur for those SBLOCAs where a pump trip is required for compliance with 10 CFR 50.46.

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August 23, 1983

5211-83-232

Office of Nuclear Reactor Regulation
Attn: J. F. Stolz, Chief
Operating Reactor Branch No. 4
Dvision of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Sir:

Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
Long Term EFW Mods (NUREG 0737 II.E.1.1)

In response to NUREG 0737 Item II.E.1.1 and as discussed in the meeting at TMI-1 on July 11, 1983 with members of your Staff and those of mine, enclosed please find a description of the modifications to the Emergency Feedwater (EFW) System to be completed prior to startup from the Cycle 6 refueling.

The purpose of these modifications is to upgrade the EFW system to a safety grade system in order to provide increased reliability in its capability to mitigate the effects of design basis accidents when the main feedwater system is not available. These modifications will be made in accordance with the requirements of NUREG 0578 Sections 2.1.7.a and 2.1.7.b, NUREG 0737 Sections II.E.1.1 and II.E.1.2, Atomic Safety and Licensing Board (ASLB) Partial Initial Decision Section II, Subsection Q, and using the acceptance criteria of Standard Review Plan Sections 9.2.6, 10.4.9 and associated Branch Technical position ASB 10-1 as principal guidance.

The modifications being implemented as part of this upgrade include mechanical system configuration changes, mechanical (seismic) and electrical (environmental) equipment qualification upgrades, changes to the control system for EFW components and seismic upgrade of piping sections in the Main Steam, Emergency Feedwater and Main Feedwater Systems.

Sincerely,

H. D. Hukill
Director, TMI-1

HDH:LWH:vjf

cc: R. Conte, J. Van Vliet

EMERGENCY FEEDWATER SYSTEM LONG TERM
SAFETY GRADE MODIFICATIONS

I. INTRODUCTION

A. This document describes the functional, design, quality assurance, health and safety, and licensing requirements for the installation and operation of modifications to the Emergency Feedwater (EFW) System of Unit No. 1 of the Three Mile Island Nuclear Station (TMI-1).

B. The EFW System shall remain generally as presently configured with modifications to insure the addition of emergency feedwater to both OTSGs assuming a single active failure concurrent with loss of offsite power. In addition, the modified system shall be capable of providing controlled emergency feedwater flow to an intact OTSG for at least two hours without relying on alternating current (AC) power. Conversion of direct current (DC) from the station batteries to alternating current is acceptable for this application.

1. All automatic initiation features provided for the EFW system shall be retained. A new automatic EFW control system for controlling OTSG level independent of the Integrated Control System (ICS) shall be provided. In addition, the capability to manually control EFW flow and set an automatic level setpoint from the main control room shall be provided.

2. All the equipment required to initiate or control EFW or to realign the water source to the EFW pumps with the exception of valves EF-V4 and 5 shall be operable from the main control room.

3. A redundant control valve shall be installed in the flow path to each OTSG in parallel with the existing control valve. A normally open block valve shall be installed downstream of each control valve to provide additional isolation capability of EFW flow to an OTSG. A cavitating venturi has been provided in the EFW flow path to each OTSG to limit flow.

4. The installation and arrangement of cavitating venturis, control valves and block valves shall provide accessibility for plant maintenance, inservice inspection and operability of the components.

5. The installation and arrangement of electrical, instrumentation and control components shall provide testability of equipment and maintenance of electrical separation.

6. Mechanical, electrical, instrumentation and control components shall not be located in high energy line break jet zones unless they are shielded from such jets. Components shall be located such that they are not subject to damage from high energy pipe whip.

II. Mechanical Systems Requirements and Modifications

A. Requirements

1. Process piping design temperature and pressure shall be consistent with the original design basis of the EFW and related service systems as identified in GAI specification SP-5544 unless system modifications call for more stringent requirements.

2. All new piping which is part of the EFW system shall be designed, fabricated, inspected, tested and erected in accordance with ANSI B 31.1 "Power Piping Code".

3. Inspections required by ANSI B 31.1 shall be performed.

4. The seismic design criteria for the piping and support system shall be in accordance with Seismic Class I design bases as defined in GAI specifications SP-5544, item 2:15, "Plant Piping for TMI" and the TMI-1 FSAR. Seismic identification symbol shall be S-I.

5. Installation, erection and testing of all piping shall be in accordance with ASME Section XI.

6. Installed cleanness class shall be Class B in accordance with GPUNC Spec. 3050B-001.

7. All new valves and the cavitating venturis shall be designed and fabricated in accordance with ASME Section III, Class 3.

B. Modifications

1. Add Cavitating Venturis in each Once Through Steam Generator (OTSG) EFW Line. (Complete)

a. This modification has been implemented to limit the flow of EFW to a ruptured OTSG in order to ensure sufficient EFW flow to the intact OTSG and to limit the mass and energy release within the reactor building for overpressure prevention. The venturis will limit the flow to the OTSG in order to reduce excessive reactor coolant system (RCS) overcooling.

2. Provide Redundant Safety Grade EFW Control and Block Valves

   a. This is being provided to prevent a single active failure from preventing the addition of EFW to an OTSG and to ensure the capability to isolate EFW flow to a ruptured OTSG.

   b. The control valves shall have sufficient range to control the EFW flow to the OTSG(s) when the plant is being cooled and the OTSG(s) are being depressurized and the EFW flow requirement is less than that initially required.

   c. The EFW system block valves shall normally be open, and in addition, the EFW initiating signals shall also provide an open signal to the block valves. Each valve shall be provided with an electric motor operator and shall fail "as is" on loss of power. The valves shall also have remote manual operation capability from the main control room.

III. Structural Requirements and Modifications

A. Requirements

1. All components which are part of the EFW System or which are required to act in support of this system shall be qualified for Safe Shutdown Earthquake (SSE) loadings to ensure structural integrity and functional operability of active components during and after an earthquake. All existing EFW system components shall be seismically qualified by analysis or by type tests if required. The qualification of new components shall be accomplished by either analysis or testing.

2. The structural design of the EFW system modifications shall be consistent with the original design basis of the EFW system and the related service systems as

identified in the TMI-1 FSAR and GAI specifications SP-5544 and SP-5661. Where practicable, all portions of the EFW system shall be installed indoors within Seismic Class S-I aircraft-hardened structures. All portions of the system required to perform the safety function shall be designed to Seismic Class S-I requirements.

3. Portions of the EFW system located outdoors shall be designed to Seismic Class S-I requirements and shall be designed to withstand the effects of the design basis natural phenomena identified in the TMI-1 FSAR Section 2.

4. All piping and valves shall be connected and supported in such a manner that any stress due to weight, thermal effects, internal piping conditions and external environment will be within the maximum allowable stresses required by the ANSI B. 31.1 "Power Piping Code".

5. Structural steel shall be designed in accordance with AISC-70 (including latest supplements) using ASTM-A36 steel, except weld unit stresses shall be as specified in Table 9.3.2.1 of AWS D1.1, -79 "Structural Steel Welding Code".

B. Modifications

1. Upgrade the EFW pumps recirculation line from recirculation control valves (EF-V-8A/B/C) to Condensate Storage Tank (CO-T1B) to Seismic Class I requirements.

   a. This modification will ensure that failure of this piping due to a seismic event shall not occur and thus prevent depletion of the required CST inventory for the EFW function.

2. Evaluate and modify the vent stacks for safety valves MS-V22A/B and atmospheric dump valves MS-V4A/B to Seismic Class I requirements.

   a. The vent stacks for safety relief valves MS-V-22A/B and atmospheric dump valves MS-V-4A/B are routed through the Intermediate Building floors. This modification will prevent the release of main steam to the Intermediate Building as a result of vent stack failure due to a seismic event. Therefore, this modification will reduce the possibility of overpressurization in the building and protect the Emergency Feedwater system components form the exposure to the hostile environment and gravity missiles.

3. Intermediate Building Flood Protection from a Main Feedwater Line Break.

   a. This modification is being implemented to mitigate the effects of flooding due to a postulated main feedwater line break in the Intermediate Building by allowing water to flow into the tendon access gallery and portions of the alligator pit which are presently isolated. By removing the upper half of the "stop walls" in the alligator pit and opening entrance "A" and "C" to the tendon access gallery, the time required for water to flood EL. 295' in the Intermediate Building will be increased from 86 seconds to approximatey 25 minutes.

IV. <u>Electrical Requirements and Modifications</u>

A. Requirements

1. The electric power and control system shall be designed as a Class 1E system. Components of the system required to operate during a loss of all AC power (Station Blackout) shall be powered from the non-interruptable vital AC or DC buses.

2. Each train of EFW to each OTSG shall be powered from its associated power sources to facilitate safety grade initiation and control of EFW to each OTSG.

3. Electrical equipment shall be qualified in accordance with applicable sections of IEEE 323, IEEE 344, IEEE 382, and NUREG-0588 or the Division of Operating Reactor Guidelines appended to I.E. Bulletin 79-01B as appropriate.

B. Modifications

1. Provide a safey grade power supply to valves CO-V-111A/B and upgrade the cable routing for power supply to valves CO-V-14A/B to meet Seismic Class I requirements.

a. This modification shall provide the capability to isolate a damaged Condensate Storage Tank (CST) from the EFW system by closing COV-111A/B from the Main Control Room so that the intact CST will have sufficient water available for the EFW system function. Similarly, the ability to close CO-V-14A/B from the Main Control Room, will allow isolation of non-EFW functions from the CST.

These features will be used in conjunction with revised EFW plant operating procedures to close CO-V-14A/BV and CO-V-111A/B whenever there is an EFW initiation and the CST has reached the Technical Specification limit for EFW inventory.

2. Delete the existing cross connect between electrical busses that allows a control room operator to load both EFW pump motors onto a single diesel generator in order to ensure electrical separation of the busses. (Complete)

3. A review shall be conducted of the emergency power bus loadings to assure that changes in bus loadings resulting from these modifications will maintain the bus loadings within acceptable limits.

V. Instrumentation and Control Requirements and Modifications

A. Requirements

1. New control systems shall be installed to initate and regulate EFW flow. Control of EFW flow to each OTSG shall be independent of control for the other OTSG. Each control system shall be of Class 1E (safety grade) design. Electric power for the control systems shall be from safety grade uninterruptable sources.

2. The control systems shall be designed so that no single active failure will prevent delivery of the required emergency feedwater to an OTSG. Also, the probability of a single failure causing inadvertent injection of EFW into an OTSG shall be minimized.

3. The control system shall be designed to enable control of emergency feedwater for at least two hours during loss of all (on-site and off-site) alternative current (AC) power sources with the exception of the battery backed 120 VAC vital sources. During the loss of all AC

power condition for two hours, only the turbine driven emergency feedwater controls are required to be functional.

4. The design of the safety grade controls shall be in accordance with applicable sections of IEEE 308, IEEE 279 and its supplements and IEEE 379. System level manual initiation shall not be provided as recommended by IEEE-279. Instead, the system components shall be provided with a manual starting or control capability as appropriate for each component.

5. All cable routing of electrical and instrumentation shall be checked to comply with Appendix R of 10CFR50 (i.e., Fire Protection Evaluation).

6. The alligator pit flood detection system shall consist of level indication located in the alligator pit. Condenser hotwell low-low level alarm can be accomplished via the existing hotwell low-low level signals.

7. The EFW system shall receive automatic initiation signals for the following conditions:

   a. Loss of both Main Feedwater Pumps, or
   b. Loss of four (4) Reactor Coolant Pumps (RCP), or
   c. Feedwater line break as detected by high Main Steam to Feedwater differential pressure, or
   d. Low OTSG water level.

8. The EFW system block valves shall normally be open and, in addition, the EFW initiating signals shall also provide an open signal to the block valves. A control switch shall be provided for each block valve for remote operation from the control room. Direct indication of actual valve position shall also be provided in the control room.

9. The capability to manually control EFW flow from the control room shall be provided. This capability shall include features to allow independent control of each flow control valve and position indication from each control valve.

10. The capability of selecting an automatic level control setpoint shall also be provided.

11. The failure mode of the control valves shall be fail-closed on loss of either instrument air, electrical power, or control signal.

12. New steam generator level instruments external of ICS shall be provided for the following functions. Level is expressed as distance above the top of the lower tubesheet:

    a. Automatic control of EFW at 30" for the condition of at least one RCP operating and 240" for loss of all four RCP's.

    b. Initiation of EFW at a low-low OTSG water level of 18".

    c. High level alarm at 337".

    d. Low level alarm at 23".

    e. High-high level alarm to indicate OTSG overfilling. Alarm is to occur at a water level of 380".

    f. Isolation of main feedwater (MFW) on a high-high level of 370" (which is above the ICS high level limit control point of 346").

    g. Operator selected auto level setpoint for use following a LOCA.

13. In addition, the ICS shall utilize the instruments for the following purposes:

    a. OTSG level control during heat up
    b. High OTSG level limit during power operation
    c. Low OTSG level limit during power operation
    d. OTSG level control after the reactor trip.

14. The modification of the OTSG level instruments shall use the top of the lower tubesheet as a reference point and use the same measurement unit (i.e., inch). These instruments shall be compensated for process pressure and environmental temperature to aid plant startup and post trip level control.

15. Automatic EFW initiation signals for feedwater line break as detected by high main steam to feedwater differential pressure, or low OTSG water level shall be generated by using four (4) channels of level measurement and 2 out of 4 (2/4) logic for each actuation (Train A and B).

16. EFW control valve modulation shall utilize two (2) channels (one for each EFW control valve) of OTSG level measurement out of a total of four (4) channels. However, EFW initiation on low water level shall be dependent upon a 2 out of 4 (2/4) logic. Capability shall be provided to bypass this initiation from the main control room.

17. Main feedwater (MFW) control shall be performed by the existing Integrated Control System (ICS). Isolated fully compensated level signals from one (1) of the four (4) channels of level measurements shall be utilized by the ICS as described above. Main feedwater isolation upon high OTSG level shall be initiated by a 2 out of 4 (2/4) logic utilizing these same level signals. This shall be performed external of the ICS. Existing level instruments associated with ICS shall be removed.

18. Main feedwater isolation shall also be initiated on a feedwater line break utilizing a 2 out of 4 (2/4) logic based upon differential pressure between main steam and feedwater system and by the Main Steam Line Rupture Detection System (MSLRDS). The MSLRDS also utilizes a 2 out of four (2/4) logic for detection of main steam pressure below 600 psig.

19. Two (2) safety grade wide range OTSG level indications shall be provided in the control room for each OTSG.

20. A safety grade water level indication and low-low water level alarm shall be provided in the control room for each condensate storage tank.

21. All instrumentation independent of the ICS and control equipment shall be qualified for operability during a Safe Shutdown Earthquake and, when instruments are to be located in the Intermediate Building, for the environmental conditions existing in the Intermediate Building following a main steam line break.

B. Modifications

1. Deletion of the Main Steam Line Rupture Detection System (MSLRDS) Signals to the emergency feedwater control valves EF-V-30A/B. (Complete)

   The deletion of the MSLRDS signals to the EFW System improves the availability of the OTSG's as a heat sink and improves the reliability and capability of EFW flow to the OTSG(s) during loss of normal feedwater flow.

2. Provide safety grade EFW initiation and main feedwater isolation on high main steam/feedwater differential pressure.

   High main steam pressure relative to main feedwater pressure is an indication of a main feedwater line rupture. This indication along with low OTSG level) anticipates failure of the secondary heat sink due to a main feedwater failure.

3. Provide a safety grade OTSG level instrumentation and signals for main feedwater (MFW) OTSG high water level isolation and OTSG low water level initiation of the EFW system.

   The isolation of main feedwater on OTSG high water level protects against OTSG overfilling caused by failure of the feedwater control system within the Integrated Control System (ICS).

4. The control system shall be of dual setpoint design with the setpoints dependent on whether or not the reactor coolant (RC) pumps are running.

   On loss of all four (4) reactor coolant (RC) pumps, the control system shall open and control the EFW flow control valves to maintain a higher OTSG water level setpoint as required to achieve reactor natural circulation cooling within the Reactor Coolant System (RCS). If at least one RC pump is operating, the control system shall control OTSG water level to a lower setpoint sufficient for forced circulation RCS cooling.

5. Provide a safety grade automatic control system independent of the Integrated Control System (ICS) that permits the Emergency Feedwater System to control OTSG level without control interaction with the main feedwater system.

6. Upgrade the controls for the Main Steam Line Rupture Detection System to safety grade such that a single failure of the control system will not prevent isolation when required. The probability of a single failure causing inadvertent actuation shall be minimized.

   The MSLRDS shall identify a ruptured OTSG when the main steam pressure falls below 600 psig and shall then automatically isolate the main feedwater to that OTSG.

8. Provide an overspeed trip alarm in the Main Control Room for the turbine driven emergency feedwater pump (TDEFWP) EF-P-1.

   This alarm will provide indication of a loss of a portion of the EFW system.

9. Provide an "alligator pit" flood detection alarm using safety grade components and a control grade main condenser hotwell low-low level alarm in the Main Control Room.

   This modification will provide an operator with a control room alarm indicating a possible main feedwater line break.

10. Evaluate the Emergency Feedwater and Engineered Safeguard (ES) Electrical Power, Control, and Instrumentation Cables that are presently routed through the alligator pit.

    The EFW and ES electrical power, control and instrumentation cables need to be evaluated to determine their capabilty of performing their safety function after a main feedwater line break incident and subsequent alligator pit flooding.

11. A portion of the existing EFW system controls is within the ICS. This interface is being replaced with the modification as identified in previous sections. OTSG level measurements associated with the EFW system shall be provided to the ICS through suitable isolation.

VI. Miscellaneous Criteria

A. Electrical and Control Equipment Environmental Qualification

Equipment which is part of the EFW system or which is required to act in support of this system and which is located in the Intermediate Building, shall either be upgraded to be qualified for the hostile environmental conditions resulting from a Main Steam Line Break (MSLB) in this building or be replaced with qualified equipment or be relocated to an environmentally acceptable location which is otherwise suitable for their safety function.

B. Maintenance

Maintenance of valves, instrumentation and controls shall be accomplished in accordance with manufacturer's instructions and recommendations. Pipe routing and equipment location shall be selected to facilitate maintenance and be consistent with the requirements of Section I.B.

C. Surveillance and In-Service Inspection

Inservice inspection requirements of ASME B&PV Code Section XI for system design and inspection apply to the design of these modifications.

The system shall be designed to allow functional testing of all new equipment during cold shutdown conditions. It shall also be designed to allow for periodic testing in accordance with the TMI-1 Technical Specifications, Section 4.9. The design shall be consistent with requirements of the TMI-1 Technical Specifications limiting conditions for operation of the turbine cycle, Section 3.4.

D. Interfacing Systems

These modifications require interfaces with the Main Feedwater, Main Steam, Condensate, Instrument Air and Class 1E electrical systems as specifically identified in previous sections.

Changes to any of these systems shall not degrade the ability of these systems or any other plant systems to perform their design functions.

E. Testing Requirement

Adequate provisions shall be made in the design of the system modifications to allow hydrostatic testing of the piping system, calibration of instrumentation, and functional testing of the controls and alarms.

F. Quality Assurance

This modification is classified as Important to Safety. Quality Assurance requirements shall be in accordance with the "Operational Quality Assurance Plan for Three Mile Island Nuclear Station, Unit 1," with specific requirements as indicated.

G. Human Factors

Human factors reviews of the man-machine interfaces shall be performed to aid in the development of the system modifications. The interface points of type, location and arrangement of controls and display, system labelling, alarm/warning system logic, maintenance requirements, and procedural guidelines shall be reviewed and documented.

H. ALARA

The design of this sytem shall implement ALARA concepts for both the construction activities and for the operating and maintenance aspects of these modifications. The ALARA impact of these modifications on other systems and personnel access shall also be considered in the design of these modifications.

GPU Nuclear

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TELEX 84-2386
Writer's Direct Dial Number:

September 7, 1983
5211-83-250

S.P.P.&T.
SEP 21 1983

Office of Nuclear Reactor Regulation
Attn: J. F. Stolz, Chief
Operating Reactors Branch No. 4
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Sir:

Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
25°F Subcooling Margin

The purpose of this letter is to inform you of the results of recent reevaluations of instrument string error and the RCS physical configuration factor associated with the subcooling margin monitor system. In our letter of March 31, 1983 (5211-83-017), GPUN indicated that a 25°F subcooling margin (SCM) action point for RCP trip was justified based on calculations performed during accident conditions (i.e., SB LOCA) which showed that the maximum string error for pressures greater than 300 psig is -18.7°F (+21.7°F) with an assumed 5°F physical configuration factor. This factor was assumed to bound any difference between the indicated pressure at the hot leg pressure instrument and the actual pressure at the top of the hot leg.

Since March, we have reevaluated both the physical configuration factor and the string error. These evaluations concluded that the 5°F physical configuration factor could be reduced to less than 1.3°F (Ref. 1) to account for the elevation difference from the instrument tap to the top of the hot leg. Additionally, GPUN has reviewed R[illegible] pump head degradation for two phase flow and determined that for an ind[illegible] 25°F SCM, the void fraction at the RC pump inlet is less than 5% for R[illegible]sure above 865 psig. Using a very conservative RC pump head/void frac[illegible]rrelation (Ref. 2), the head degradation is less than 10%. The instru[illegible]rror for SB LOCA was reevaluated using more conservative assumptions a[illegible] determined to be ± 22.1°F (Ref. 3 & 4) and for normal conditions was evaluated to be ± 10.3°F. These changes do not alter our conclusion that the 25°F indicated subcooling margin action point for RCP trip is appropriate, but they do modify the assumption used by the Appeal Board in ALAB-729, dated May 26, 1983. The Appeal Board agreed with the 25°F SCM action point "providing the 20°F error in the TMI-1 instrumentation is not exceeded". Our reevaluation shows that the instrument string error exceeds 20°F ... g SB LOCA conditions, but is entirely offset by the conservatism in the physical configuration factor.

Consequently, our conclusion that the 25°F action point is appropriate remains valid.

Sincerely,

H. D. Hukill
Director, TMI-1

HDH:LWH:vjf

cc: R. Conte
J. Van Vliet

Ref.
1. GPUN Calculation 1101x-5450-015 (Attached)
2. EPRI Report (NP2578) "Two Phase Performance of Scale Models of a Primary Coolant Pump", dated 9/82, p. 6-12
3. GPUN Calculation 11014-322B-009, Rev. 2 & 3 (Attached)
4. GPUN Calculation C-1101-655-5350-001 (Attached)

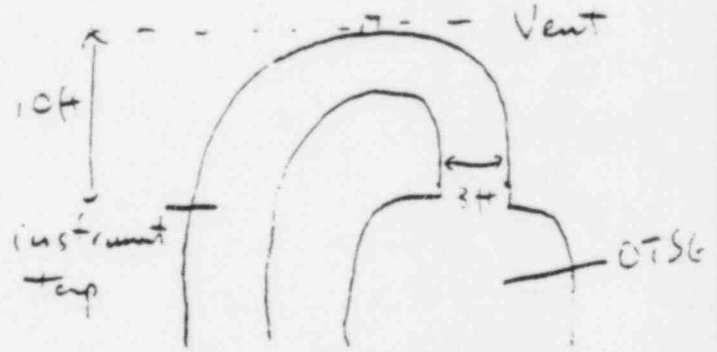
GPU Nuclear

SUBJECT Saturation Margin Monitor Adjustment ~~in~~ To Elevation Difference Between the Pressure Tap and Top of ~~Hot~~ leg.

CALC. NO. 1101x-5450-015
SHEET NO. 1 OF 3
DATE 7/29/83
COMP. BY/DATE ZCP 7/29/83
CHK'D. BY/DATE MAA 8/4

Problem statement: To determine the maximum adjustment to SCM (subcooling margin) measured at the hot leg as relative to the top of the candy cane.

The distance from the instrument tap to the top of the candy cane is 10 ft. the hot leg diameter is 3 ft.



Pressure drop from the instrument tap to the top of candy cane is composed of friction loss (Fanning) and ~~also~~ gravity, ie

$$\Delta P = 3.36\times10^{-6} f l \frac{W^2 v_f}{d^5} + \rho_f g H$$

A000 0016 12-80

GPU Nuclear

SUBJECT

CALC. NO. 1101X-5450-015
SHEET NO 2 OF 3
DATE 7/29/83
COMP BY/DATE 7/29/83 KCP
CHK'D BY/DATE MAH 8/4

where the friction term is given by the CRANE Handbook on "Flow of Fluids" Eq. 3-5

with $W = 71 \times 10^6$ lb/hr (TMI-1 Cycle 4 data)

$v_f = 0.01889$ ft³/lbm @ 300 psia (lowest pressure to operate RCP

or 0.02669 ft³/lbm @ 2200 psia normal operating press.

$d = 36$ in

$l = 10$ ft.

Table A-25 of Crane gave $f = 0.0188$ for 36" pipe turbulent flow. $Re = \frac{D v \rho}{\mu} = \frac{3 \times W \times \rho}{3 \times \frac{\pi \times 3^2}{4} \times \rho \times 0.1 \times 3600} = 8.4 \times 10^6$

$\mu = 0.1$ centipoise from A-3

$$\Delta P = \frac{3.36 \times 10^{-6} \times 0.0188 \times 10 \times (71 \times 10^6)^2 \times 0.02669}{(36)^5} + \frac{10}{.02669 \times 144}$$

$= 1.4 + 2.6$

$= 4.0$ psi @ 2200 psia

$$\Delta P = \frac{3.36 \times 10^{-6} \times 0.0188 \times 10 \times (71 \times 10^6)^2 \times .01889}{(36)^5} + \frac{10}{.01889 \times 144}$$

$= 0.99 + 3.6$

$= 4.6$ psi @ 300 psia

A000 0016 12-80

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CALC NO. 1101X-5450-015
SHEET NO. 3 OF 3
SUBJECT
DATE 7/29/83
COMP BY/DATE KCP
CHK'D BY/DATE MAA, 8/4

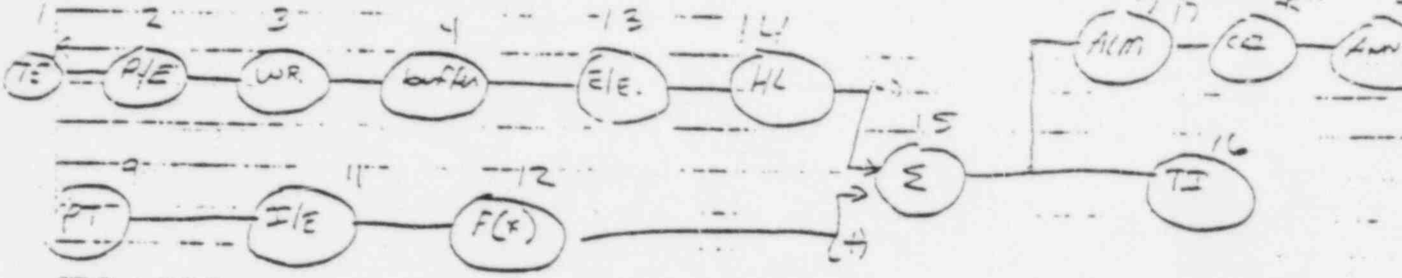
From ASME Steam Table the Tsat difference for $\Delta P = 4.6$ psi is about 1.3°F around P equal to 300 psia (lowest pressure to operate RCP). and down to 0.3°F at pressure about 2100 psia. Therefore the maximum correction will be about 1.3°F

A000 0016 12-80

Nuclear

CALC NO. [illegible] 3278-00[illegible]
SHEET NO. .... OF ..[illegible]....
DATE 6-17-83
COMP. BY/DATE [illegible]
CHK'D. BY/DATE [illegible]

SUBJECT Sat Margin Error Analysis
SL LOCA



DEVICES 1, 2, 3, 4 BMCo in NNI cabinets

13, 14, 9, 11, For in Signal Cond cabinets
12, 15, 17

Listed accuracy effects are upper range limits. These are conservatively assumed equal to 3 sigma (σ)

± statistical
+, − systematic

Device 1+2

	URange	% Analysis	
$A_{1,2}$ accuracy	± .15	± .10	(2σ)
$L_{1,2}$ linearity	± .15	± .10	(2σ)
$V_{1,2}$ supply voltage	± .02	± .007	(1σ)
$T_{1,2}$ amb. temp	± .1	± .033	(1σ)

NOTE:
Supply voltage effects corr. to voltage output affects much poorer than power supply regulation limited bus regulation. ∴ 1σ values utilized

Ambient temperature effects corr. to 50°F temperature swings. Expected ambient temp. swings are signif. less. Redundant HVAC is also provided. ∴ 1σ values utilized

Device 3

A_3 accuracy	±.15	±.10	(2σ)
L_3 linearity	±.15	±.10	(2σ)
V_3 supply voltage	±.002	±.001	(1σ)
T_3 amb temp	±.1	±.033	(1σ)
R_3 repeat.	±.001	± .001	(2σ)

Device 4

A_4 accuracy	±.15	±.10	(2σ)
L_4 linearity	±.15	±.10	(2σ)
T_4 amb. temp	± .002	±.001	(1σ)
V_4 supply voltage	± .1	± .033	(1σ)

GPU Nuclear

SUBJECT ....

CALC. NO. ....
SHEET NO. 2 OF ....
DATE ....
COMP. BY/DATE ....
CHK'D. BY/DATE ....

String error devices 1, 2, 3, 4

$$error_{1,2,3,4} = \sqrt{(\pm A_{1,2})^2 + (\pm L_{1,2}) + (\pm(V_{1,2} + V_3 + V_4))^2 + (\pm(T_{1,2} + T_3 + T_4))^2 + (\pm A_3)^2 + (\pm A_4)^2 + (\pm L_3)^2 + (\pm L_4)^2 + (\pm R_3)^2}$$

$$= \sqrt{(.1)^2 + (.1)^2 + (\underbrace{.007 + .001 + .033}_{.041})^2 + (\underbrace{.033 + .033 + .001}_{.067})^2 + (.1)^2 + (.1)^2 + (.1)^2 + (.1)^2 + (.001)^2}$$

$$error_{1,2,3,4} = \pm .257\ \%$$

DEVICE 13 Uncert/2-σ

A_{13} acc.	$\pm .5$	$\pm .333$	(2σ)
L_{13} lin.	$\pm .5$	$\pm .333$	(2σ)
V_{13} supp. volt.	$\pm .25$	$\pm .083$	(1σ)
T_{13} amb. temp.	$\pm .5$	$\pm .167$	(1σ)

DEVICE 14

A_{14}	$\pm .5$	$\pm .333$	(2σ)
R_{14}	$\pm .1$	$\pm .067$	(2σ)
V_{14}	$\pm .5$	$\pm .167$	(1σ)
T_{14}	$\pm .5$	$\pm .167$	(1σ)

string error devices 13,14

$$error_{13,14} = \sqrt{(\pm A_{13})^2 + (\pm A_{14})^2 + (L_{13})^2 + (R_{14})^2 + (\pm(V_{13}+V_{14}))^2 + (\pm(T_{13}+T_{14}))^2}$$

$$= \sqrt{(.333)^2 + (.333)^2 + (.333)^2 + (.067)^2 + (.083 + .167)^2 + (.167 + .167)^2}$$

$$error_{13,14} = \pm .715\%$$

$$error_{1,2,3,4,13,14} = \sqrt{(\pm error_{1,2,3,4})^2 + (\pm error_{13,14})^2}$$

$$= \sqrt{(.257)^2 + (.715)^2}$$

$$error_{1-14} = \pm .760\%$$

DEVICE 9

	U Range		
T/P_9 (environment - accident)	±5	±3.33	(2σ)
R_9 radiation	±5	±1.67	(1σ)

DEVICE 11

A_{11}	±.5	±.333	(2σ)
L_{11}	±.5	±.333	(2σ)
V_{11}	±.25	±.083	(1σ)
T_{11}	±.5	±.167	(1σ)

DEVICE 12

	Urange %	
A_{12}	±.5	±.333 (2σ)
T_{12}	±.5	±.167 (1σ)
V_{12}	±.5	±.167 (1σ)
* C_{12}	±.25	±.25

*measured accuracy after calibration, added for conservatism, this is actually part of A_{12}

	published	relative to analysis
R_{12} repeat	±.1	+.1 or −.1
DB_{12} dead band	±.1	+.1 or −.1 (systematic)

DEVICE 15

A_{15}	±.5	±.333 (2σ)
R_{15}	±.1	.067 (2σ)
DB_{15}	±.1	.067 (2σ)
V_{15}	±.5	.167 (1σ)
T_{15}	±.5	.167 (1σ)

DEVICE 16

A_{16}	±.005	±.00333 (2σ)
L_{16}	±.005	±.00333 (2σ)

IGNORE — LOOP ACCURACY, INCLUDING ALARM UNIT WILL DETERMINE LOOP ACCURACY AND THUS SET POINT

DEVICE 17

A_{17}	± 1.0	±.667	(2σ)
R_{17}	±.5	±.333	(2σ)
V_{17}	±.25	±.083	(1σ)
T_{17}	±.5	±.167	(1σ)

GPU Nuclear

SUBJECT .... CALC. NO. [illegible] SHEET NO. 5 OF [illegible] DATE .... COMP. BY/DATE .... CHK'D. BY/DATE ....

$$error_{loop} = \Big[(\pm TTP_9)^2 + (\pm R_9)^2 + (\pm A_{11})^2 + (\pm L_{11})^2 + \left(\pm(V_{11} + V_{12} + V_{15} + V_{17})\right)^2 + \left(\pm(T_{11} + T_{12} + T_{15} + T_{17})\right)^2 + (\pm A_{12})^2 + (\pm C_{12})^2 + (\pm A_{15})^2 + (\pm R_{15})^2 + (\pm DB_{15})^2 + (\pm A_{17})^2 + (\pm R_{17})^2 + \left(\pm \left(\frac{800}{500}\right) error_{1-14}\right)^2 \Big]^{1/2}$$

NOTE: 800/500 accounts for diff. in temp loop span from sat. margin span.

SYSTEMATIC ERROR $\begin{cases} +R_{12} + DB_{12}, \text{ or} \\ -R_{12} - DB_{12} \end{cases}$

$$error_{loop}\ (\%) = \Big[(3.33)^2 + (1.67)^2 + (.333)^2 + (.333)^2 + (.083 + .167 + .167 + .083)^2 + (.167 + .167 + .167 + .167)^2 + (.333)^2 + (.25)^2 + (.333)^2 + (.067)^2 + (.067)^2 + (.667)^2 + (.333)^2 + \left(\frac{800}{500} \times .76\right)^2 \Big]^{1/2}$$

$$+ .2 \text{ or } - .2$$

A000 0016 12-80

GPU Nuclear

SUBJECT

CALC. NO.

SHEET NO. 2 OF 5

DATE

COMP. BY/DATE

CHK'D. BY/DATE

$$\frac{\text{error}}{\text{loop}} = \pm 4.14 \quad \begin{matrix} +.2 \\ \text{or} \\ -.2 \end{matrix} \; \%$$

ACTUAL = INDICATED + ERROR

$\therefore$ ACTUAL SUBCOOLING MARGIN = INDICATED — 4.34% (conservative) (of 500° span)

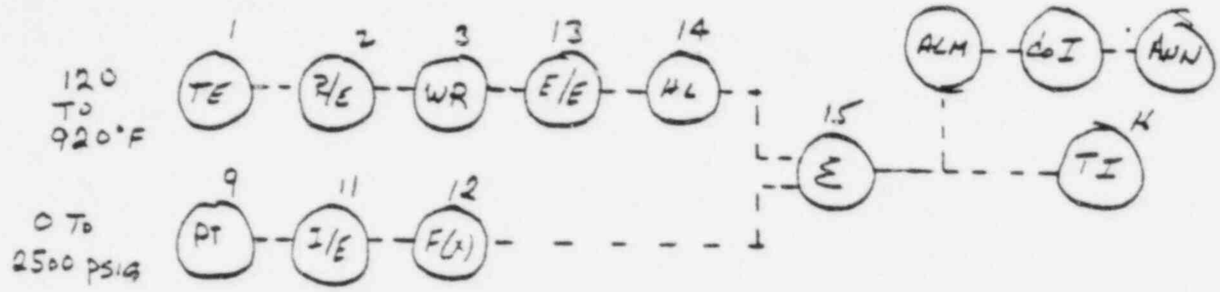
= INDICATED − 21.7° (conservative)

REV 3

GPU Nuclear

CALC. NO. 11014-322B-009
SHEET NO. 1 OF 4
DATE 8-25-83
COMP. BY/DATE [signature]
CHK'D. BY/DATE [signature]

SUBJECT TMI-1 SATURATION MARGIN MONITOR ERROR ANALYSIS SBLOCA



THIS REVISION IS REQUIRED BECAUSE SU&T WAS NOT ABLE TO ACHIEVE THE CALCULATED ACCURACY FOR MODULES 1, 2, & 3.

1. REFERENCE FQ R788 DATA SHEETS: AT FULL RANGE (920°F) THE LOOP ACCURACY AT MODULE 3 WAS WORSE CASE .63% (9.937V). BASED ON THIS DATA A MARGIN OF 33% WAS ADDED TO THE 1, 2, 3 LOOP INACCURACY TO DERIVE A 3σ ACCURACY OF ±.84%

2. THE PREVIOUS CALCULATION OF THIS LOOP INCLUDED A BUFFER AMPLIFIER BETWEEN MODULES 3 & 13 ABOVE. THIS AMPLIFIER APPEARS ON GPUN LOOP DIAGRAM 1B-660-42-015 REV 3 BUT NOT ON SECM-129-3 GAI DWG 04 4692-B210-958-RDB-1. THIS MODULE WAS NOT INSTALLED AND ITS CONTRIBUTION MAY BE NEGLECTED.

CALCULATIONS

	3σ		
MODULE 13			
A_{13} ACC	± 0.50	± 0.333 (2σ)	
L_{13} LIN	± 0.50	± 0.333 (2σ)	
V_{13} SUPP V	± 0.25	± 0.083 (1σ)	
T_{13} AMB T	± 0.50	± 0.167 (1σ)	
MODULE 14			
A_{14}	± 0.50	± .333 (2σ)	
R_{14}	± 0.10	± 0.067 (2σ)	
V_{14}	± 0.50	± 0.167 (1σ)	
T_{14}	± 0.50	± 0.167 (1σ)	

NOTE:
Supply Voltage Affects corresponding to Voltage output affects are much poorer than Power supply Regulation/Vital Bus Regulation
∴ 1σ Values Are Used

AMBIENT TEMPERATURE AFFECTS CORRESPONDING TO 50°F SWINGS EXPECTED AMBIENT SWINGS ARE LESS. REDUNDANT HVAC IS USED.
∴ 1σ VALUES ARE USED

A000 0016 12-80

STRING ERROR FOR MODULES 13 & 14

$$\text{ERROR}_{13\&14} = \left[(A_{13})^2 + (A_{14})^2 + (L_{13})^2 + (R_{14})^2 + (V_{13}+V_{14})^2 + (T_{13}+T_{14})^2\right]^{1/2}$$

$$= \left[(.333)^2 + (.333)^2 + (.333)^2 + (.067)^2 + (.083+.167)^2 + (.167+.167)^2\right]^{1/2}$$

$$= \pm 0.715\ \%.$$

$$\text{ERROR}_{1,2,3} = \pm .84\ (3\sigma) \quad \therefore \quad 2\sigma = 2/3(\pm .84) = \pm .56$$

STRING ERROR 1, 2, 3, 13 & 14

$$2\sigma = \left[(\text{ERROR}_{123})^2 + (\text{ERROR}_{13\&14})^2\right]^{1/2}$$

$$= \left[(.56)^2 + (.715)^2\right]^{1/2}$$

$$= \pm .91$$

MODULE 9

	S.L.	
T/Pq (ENVIRO. ACC)	± 5%	± 3.33 (2σ)
RADIATION	± 5%	± 1.67 (1σ)

MODULE 11

A_{11}	± 0.50	± 0.333 (2σ)
L_{11}	± 0.50	± 0.333 (2σ)
V_{11}	± 0.25	± 0.083 (1σ)
T_{11}	± 0.50	± 0.167 (1σ)

TC. -

GPU Nuclear

CALC. NO. 1101-41-3203-009
SHEET NO. 3 OF 4
DATE 8-25-83
COMP. BY/DATE
CHK'D. BY/DATE

SUBJECT

MODULE 12	U.L	
A_{12}	±.5	±.333 (2σ)
T_{12}	±.5	±.167 (1σ)
V_{12}	±.5	±.167 (1σ)
$*C_{12}$	±.25	±.25

R_{12} repeat	+.1 or -.1	
DB_{12} dead band	+.1 or -.1	

MODULE 15		
A_{15}	±.5	±.333 (2σ)
R_{15}	±.1	±.067 (2σ)
DB_{15}	±.1	±.067 (2σ)
V_{15}	±.5	±.167 (1σ)
T_{15}	±.5	±.167 (1σ)

MODULE 16		
A_{16}	±.005	±.00333 (2σ)
L_{16}	±.005	±.00333 (2σ)

MODULE 17		
A_{17}	±1.0	±.667 (2σ)
R_{17}	±.5	±.333 (2σ)
V_{17}	±.25	±.083 (1σ)
T_{17}	±.5	±.167 (1σ)

* MEASURED ACCURACY AFTER CALIBRATION, ADDED FOR CONSERVATISM, THIS IS ACTUALLY PART OF A_{12}.

REV. —

GPU Nuclear

CALC NO C-1101-665-5350-0-

SHEET NO OF

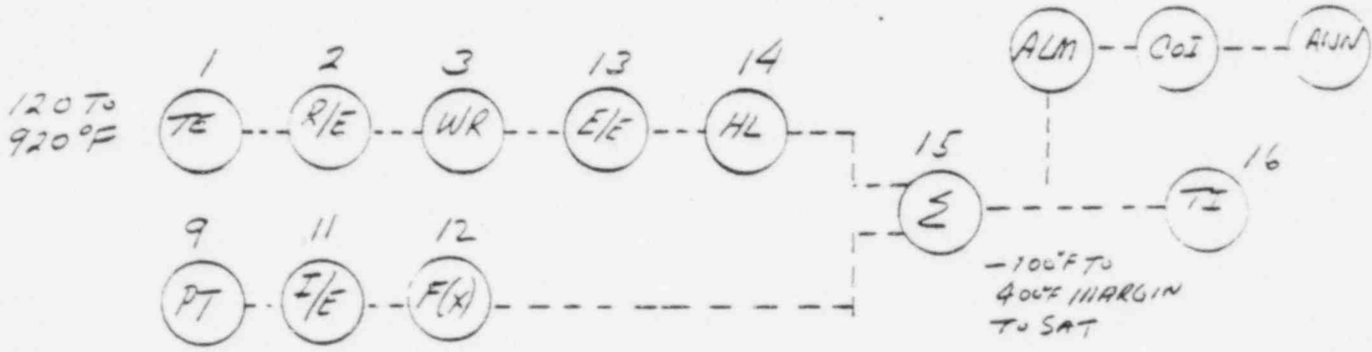
DATE 8-27-83

COMP BY/DATE

CHK'D BY/DATE

SUBJECT SATURATION MARGIN MONITOR

ERROR ANALYSIS - NON ACCIDENT



THIS REVISION IS REQUIRED BECAUSE SU&T WAS ONLY ABLE TO ACHIEVE ±0.63% ACCURACY FOR DEVICES 1, 2 & 3 (BAILEY ICS MODULES).* REV 0 PROVIDED A STRING ERROR FOR 1, 2 & 3 OF ±0.257%. IT IS APPROPRIATE TO USE THE ACTUAL ERROR OBTAINED. *THE STRING ERROR OF ±0.257 ALSO INCLUDED A BAILEY BUFFER AMPLIFIER BETWEEN DEVICES 3 & 13. ABOVE. THIS AMPLIFIER APPEARS ON GPUN LOOP DIAGRAM 1B-660-42-015 REV 3 BUT NOT ON SECM-129-3 GAI DWG C4 4692-B210-958-RUB-1. THIS MODULE WAS NOT INSTALLED. HENCE ITS CONTRIBUTION MAY BE NEGLECTED.

DEVICE 9

	UR %		
A9 ACCURACY	±0.25	±0.167 (2σ)	
S9 STABILITY	±0.25	±0.167 (2σ)	
T9 TEMPERATURE	±1.25 (100°F)	—	
O9 OVERPRESSURE	± 0.5 (4000PSI)	—	

POWER SUPPLY EFFECT NEGLIGIBLE

FOR REMAINING DEVICES, EFFECTS ARE IDENTICAL TO CALC 1101X-322II-009 REV 3

GPU Nuclear

CALC NO C-1101-665-5350-002
SHEET NO OF
DATE
COMP BY/DATE
CHK'D BY/DATE

SUBJECT

$$\text{Loop Error} = \pm\Big[(\pm A_9)^2 + (\pm S_9)^2 + (\pm T_9)^2 + (\pm O_9)^2 + (\pm A_{11})^2 + (\pm L_{11})^2 + (\pm (U_{11} + U_{12} + U_{15} + U_{17})^2 + (\pm(T_{11} + T_{12} + T_{15} + T_{17}))^2 + (\pm A_{12})^2 + (\pm C_{12})^2 + (\pm A \text{[illegible]} + (\pm R_{15})^2 + (\pm DC_{15})^2 + (\pm A_{17})^2 + (\pm K \text{[illegible]})^2 + \left(\pm\left(\tfrac{800}{500}\right) \text{Error}_{TT\,14}\right)^2\Big]^{1/2} \underbrace{\pm 0.2}_{\text{Systematic}}$$

$$= \pm\Big[(0.167)^2 + (0.167)^2 + (0.333)^2 + (0.333)^2 + (0.5)^2 + (0.666)^2 + (0.333)^2 + (0.25)^2 + (0.333)^2 + (0.067)^2 + (0.067)^2 + (0.667)^2 + (0.333)^2 + \left(\left(\tfrac{800}{500}\right) \times 0.91\right)^2\Big]^{1/2} \pm 0.2$$

$$= \pm 1.99 \pm 0.2$$

$\text{Loop Error} = \pm 2.19\%$

ACTUAL = INDICATED + ERROR

MAX ERROR $= \pm 2.19\%$ (500°F SPAN)

$= \pm 10.95°F$

ACTUAL SATURATION MARGIN = INDICATED − 10.95 °F

GPU Nuclear

GPU Nuclear Corporation
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TELEX 84-2386
Writer's Direct Dial Number:

September 9, 1983
5211-83-24?

SEP 21 1983
S.P.P.&T.

Office of Nuclear Reactor Regulation
Attn: J. F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Sir:

Three Mile Island Nuclear Station, Unit (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
Relief and Safety Valve Testing (NUREG 0737, II.D.1)

In response to your letter of July 5, 1983 and as discussed between members of your Staff and mine on July 26, 1983, enclosed please find our response to your questions. Additionally, RELAP V analysis for the 400°F subcooled water condition were transmitted to EG&G Idaho on August 5, 1983.

Sincerely,

H. D. Hukill
Vice President - TMI-1

cc: R. Conte
J. Van Vliet

RESPONSE TO NRC LETTER
DATED JULY 5, 1983

<u>RELIEF AND SAFETY VALVE TESTING</u>

Item 1. Selection of "feed and bleed" as the transient that would produce the maximum loads on the discharge piping could not be verified since no discussion of the methods or details of analyses are included in the submittal. The submittal cites the Electric Power Research Institute (EPRI) report, "Valve Inlet Fluid Conditions for Pressurizer Safety and Relief Valves for B&W 177-FA and 205-FA Plants", as justification for the selection. The cited report does not describe a transient titled "feed and bleed". The conditions identified in the submittal, subcooled water at 400°F and 2500 psig, appear to be those resulting from extended high pressure injection events. A discussion should be provided describing the methods used to select the limiting transient and clarifying the events of the transient.

Response:

The term feed and bleed (also referred to as extended HPI) is discussed in section 4.4 of the "Valve Inlet Fluid Conditions for Pressurizer Safety and Relief Valves for B&W 177 and 205 FA Plants". In this report, 3 specific limiting cases are discussed.

Case 1 - High temperature water SB LOCA less than 0.02 $ft.^2$

In this case the energy discharged through the break is not sufficient to remove core heat. No main or emergency feedwater is assumed available to sustain natural circulation on the primary side. Heat removal is accomplished by high pressure injection into the primary with discharge through the pressurizer safety/relief valves.

Case 2 - Low Temperature Water - Steam Line Break

In this case the overcooling event is intensifed by using minimum core decay heat, large uncontrolled emergency feewater flow, and no operator action to throttle or stop HPI. A minimum of 400°F subcooled water discharge resulting from the analyses was performed in response to IE Bulletin 79-05A&B.

Case 3 - Steam - Startup Rod Withdrawal

The design basis event for TMI-1 with a steam discharge is the startup rod withdrawal accident (See FSAR Section 14.1.2.2). The transient is terminated by the high pressure trip.

The report on valve inlet fluid conditions also set the limit for pressurizer surge line flow rate. As it shall be presented in the response to Item 8 in detail, the TMI-1 plant specific maximum surge line in-flow is much less than the set limit, and the corresponding EPRI test has resulted in a safety valve flow rate much greater than that the TMI unit would generate. Since only one safety valve is assumed open at a time, then the surge line in-flow is conservatively equated to safety valve discharge flow. This leads to the conclusion that both the fluid conditions and the test results are applicable to TMI-1.

Item 2. The submittal does not include a discussion of consideration of single failures after the initiating events. NUREG-0737 required selection of single failures that produce maximum loads on the safety valves. A discussion should be provided describing how the single failure considerations required by NUREG-0737 are met.

Response:

As described in the response to item 1, the bounding cases are a high and low temperature water condition.

The single failure, for the purposes of this analysis, for the high temperature water case is an assumed loss of emergency feedwater which necessitates extended HPI operation (Feed and Bleed).

The single failure, for the purposes of this analysis, for the low temperature water case is no operator action to throttle HPI during the overcooling event (additionally, EFW Flow is also considered uncontrolled).

The single failure, for the purposes of this analysis, for the steam case is no pressurizer spray capability in the pressurizer.

All other license basis events result in lower loads on the safety/relief valves and discharge piping.

Item 3. Overpressure transients will cause the pressurizer sprays to activate adding moisture to the steam volume. When the safety valves lift or the power operated relief valves (PORVs) are opened they would be passing a steam-water mixture. Was this effect considered in the analyses done to select the transients that produced maximum loads on the discharge piping?

Response:

The analyses performed on the safety/relief valves indicate that steam/water exist in the valve and down stream piping resulting from flashing of water or condensation of steam. A specific steam-water analysis was not performed. However, a steam analysis (Attachment 3) and a water analysis (Attachment 1) were performed which indicate maximum loads occur for water discharge.

The purpose of the B&W report is to document the expected range of fluid inlet conditions to which the PORV and SRV's may be subjected. The B&W report does not evaluate two phase inlet conditions.

Discharge piping design input assumptions such as: a) lower than expected inlet water temperature, b) higher than rated valve flow rate (see response to item no. 7), and c) fast opening time (see response to item no. 16), are deemed sufficient to insure conservatism in the analysis.

Item 4: The evaluation by Babcock and Wilcox (B&W) that showed up to 20% blowdown can be tolerated without any adverse effect on safety could not be verified since details of the analyses were not provided. The evaluation was based on hot leg voiding but no discussion was included to demonstrate that it is the limiting criteria. The increased blowdown would also cause a higher rise in the pressurizer level during transients that result in the safety valves lifting. No discussion is provided to demonstrate the level will not reach the discharge piping connection resulting in a transition of flow through the safety valves from steam to water-steam mixture. Details of the analyses supporting the conclusion that there will be no adverse effect on safety and details of the analyses demonstrating that the water level will not reach the discharge piping should be provided.

Response:

Attached, please find a copy of the "Pressurizer Safety Valve Maximum Allowable Blowdown" (Attachment 2), which discusses the details of the analysis.

Hot leg voiding has been identified as the primary safety concern relative to the maximum allowable PSV blowdown. The potential for hot leg voiding exists because of the lower system pressure that will be caused by larger PSV blowdown values. These lower pressures combined with transients that produce high system temperatures could result in saturation conditions and voiding that could impede natural circulation cooling.

This increase in pressurizer safety valve blowdown is not a direct safety concern. All nuclear plants are designed to accomodate Loss of Coolant Accidents including that which would result if a pressurizer safety valve sticks in the open position - i.e., an unlimited (100%) blowdown.

An additional desirable criteria would be that the pressurizer not fill for the larger PSV blowdown values. The pressurizer has a greater potential for filling because the larger blowdown values will allow large insurges during the blowdown cycle. The same EPRI tests that identified the blowdown concern also showed the safety valves were generally able to relieve two-phase fluid and water; thus, filling the pressurizer is a concern of secondary importance.

Item 5. The B&W Report on "Valve Inlet Fluid Conditions for Pressurizer Safety and Relief Valves" does not include the consequences of a reactor coolant pump shaft seizure, which is the accident which, in some plants, results in the fastest increase in and the highest peak reactor coolant system pressure. Why wasn't this considered in determining the worst case transient?

Response:

Section 14.1.2.6.3 of the TMI-1 FSAR addresses the locked rotor transient which results in a flux-flow trip. The B&W design mitigates the consequences of the accident which yields high pressures in the Westinghouse design. Attachment 2, Table 2 indicates high pressure and temperature events for B&W reactors. The "Valve Inlet Fluid Conditions for Pressurizer Safety Valves and Relief Valves for B&W 177 and 205 FA Plan B" did not include this transient since it potentially does not challenge the PORV or safety valves (15 psig above nominal).

Item 6. The submittal states that the ring settings to be used for the safety valves are those that showed the most stable valve configuration during the Electric Power Research Institute (EPRI) testing. The specific ring settings, however, are not identified. The back pressures for steam flow are given in the submittal but the back pressures for flows with subcooled water at the valve inlet are not provided. The submittal does not discuss the test valve performance to verify that the valve did perform satisfactorily. The specific ring settings to be used should be provided . A comparison should be provided that demonstrates, with the specfied ring settings and appropriate back pressures, the valves will have stable operation for the Final Safey Analysis (FSAR) transients, will pass rated steam flow and will pass adequate flow to protect the primary system from over pressure for transients with subcooled water at the valve inlet.

Response:

By GPUN letter dated October 28, 1982 (82-260), GPUN informed NRC that the safety valves had been adjusted to the EPRI ring settings.

The TMI-1 plant specific ring settings are as follows:

Lower Ring	+11 notches
Middle Ring	-40 notches
Upper Ring	-48 notches

The EPRI test parameters were established to envelope the B&W, CE and Westinghouse transient conditions by using the most severe transients. Backpressures were established by EPRI using the maximum allowable backpressures per valve manufacturer requirements. A valve that operates satisfactorily at the most severe test conditions will meet plant conditions which are less severe. The TMI-1 conditions are bounded by the EPRI test conditions.

The valve ring settings are based on satisfactory performance on steam transients because the valve was designed for steam. Therefore, the ring settings for steam have to be the ring settings for water. See answer to question 8 for discussion of valve performance with subcooled water.

Item 7. The submittal describes the safey valves as Dresser Valves Model 31739A with a rated relief capacity of 317,973 lb/hr. However, the same model valve used in the EPRI test program is identified in the EPRI test report with a rated relief capacity of 297,845 lb/hr. The apparent difference in rated flow should be explained.

Response:

See second note on bottom of Table 1 of revision 1 of Gilbert report (Attachment 1) for a discussion of flow rates. Dresser Valve Model 31739A has only one orifice size which is 2.545 in.^{2}. Therefore, the only variable in the capacity equation (W=51.5KAP) is the inlet pressure. The higher the inlet pressure, the higher the flow.

W = capacity in lb/hr
K = constant = .8775 = .9 x .975
A = orifice area = 2.545
P = pressure = 2500 psig + accumulation + 14.7 psig
W = 51.5 x .8775 x 2.545 x (2500 + .03 x 2500 + 14.7) = 297,846lb/hr.
W = 51.5 x .8775 x 2.545 x (2500 + .1 x 2500 + 14.7) = 317,973lb/hr.

The higher flow rate was used for conservatism in the analysis of the discharge piping.

Item 8. The EPRI test series for the Dresser Valve Model 31739A included a test at 400°F subcooled liquid in which the valve only partially opened. The system pressure continued to accumulate and the test was terminated. The test considerations nearly duplicated the conditions for the subcooled transient selected in the submittal. Verifications should be provided to demonstrate that the valve will provide sufficient flow to relieve the pressure for the selected transient.

Response:

Under the EPRI proposed bounding conditions, namely, extended operation of three large HPI pumps (620 gpm @ 2500 psig) and maximum (1.2 times ANS) decay heat, assuming the valve inlet temperature to be 579°F, initially, the valve inlet temperature and surge line flow are calculated to be:

Time	Temperature (°F)	Flow (lbm/hr)
0	579.0	227,000
1 hr.	530.0	240,000
2 hrs.	470.1	255,000
3 hrs.	433.5	264,000

Under the realistic conditions of TMI-1, i.e., 2 HPI pumps (480 gpm @ 2500 psig) and 1.0 ANS decay heat, the fluid conditions are:

Time	Temperature (°F)	Flow (lbm/hr)
0	579.0	174.000
1 hr.	547.5	182,000
2 hrs.	496.9	192,000
3 hrs.	463.5	200,000

The EPRI water tests at 550°F and 400°F (tests 1112 and 1114) resulted in maximum flow rates of 450,000 lbm/hr and 500,000 lbm/hr, respectively. Although in the latter case, the system pressure continued to accumulate and the test was aborted at 2750 psia, the valve open flow rate excceeds that calculated for both realistic and bounding cases for TMI-1, and thus should be considered acceptable.

Review of the B&W Valve Inlet Fluid Conditions reveals that in the determination of the surge flow, B&W has incorporated conservatism by taking the sum of two terms: one for HPI injection and one for thermal expansion. Actually, as cold HPI water is mixed with hot RCS water, a contraction is resulted that reduces the net expansion owing to core decay heat.

Item 9. The submittal lists the TMI-1 power operated relief valve (PORV) as a model 31533VX-30 with a 1-3/32 in. bore. The PORV tested by EPRI was a Dresser Model 31533VX-30-2 with a bore diameter of 2-5/16 in. The effect on performance resulting from the difference in models and bore diameter should be addressed.

Response:

The EPRI Valve Selection/Justification Report discusses the differences between the various Dresser Valve models and the differences in bore diameters. The TMI-1 valves have been modified to have the -2 internals and therefore, there is no difference in operation between the TMI-1 valve and the EPRI valve. Both valves have the same size internals. The only difference between the valves is in bore diameter and this affects capacity only and that only in a minor way. The valve functions as a result of pressure ratios based on seat diameter and both valves have the same seat size (1-5/16").

Item 10. The Dresser PORV tested by EPRI failed to close and had a delayed closure for the test conditions of low temperature water followed by 550°F water. Verification should be provided to demonstrate that this performance of the valve will not have an adverse effect on the safety of the plant.

Response:

These tests were for a cold loop seal discharge followed by hot pressurizer water. This test is for plants which have a loop seal before the PORV. TMI-1 does not have a loop seal before its PORV. Therefore, this test is not applicable to TMI-1.

Item 11. NUREG 0737 requires qualification of the block valve. Specific data demonstrating qualification of the block valve is not given in the submittal. A reference is made to a report by R. C. Youndahl indicating satisfactory performance for a similar valve. The TMI-1 block valve is identified as a Velan 2-1/2 inch gate valve F9-454B-13MS with Limitorque operator SMS-00-10. The valve tested in the EPRI program was a Velan Valve, B10-30548-13MS. The EPRI tests demonstrated closure only with steam. Additional information should be provided to verify that the test valve adequately represents the TMI-1 valve and the testing with steam only provides adequate assurance that the valve will open and close satisfactory for the required plant conditions.

Response:

TMI-1 Block Valve F9-454B-13MS
EPRI Test Block Valve B10-3054B-13MS

F = flanged end	B = Butt weld ends	)No affect on representation
9 = 2-1/2"	10 = 3"	)due to similarity of valve
4 = 2500 lb rating	3 = 1500lb rating	)internals & operation
5 & 05 = conventional port gate valve, OS&Y		)
4B - vertical stem, bolted bonnet		)
13 - 316 stainless steel body		)Same for both valves
MS = 316 stellited disc & seat, 630 stem		)

The TMI-1 valve and the EPRI test valve are the same Type Velan valve (style, internal design, operation) except for size, pressure rating and valve ends which have no effect on valve operation.

The valve motor operator sizing equations use pressure to determine required operator torque outputs. The fluid and/or flow rate are not used in the sizing equations. Therefore, an operator sized to close a valve against a 250 psi pressure differential will operate on either steam or water flow.

Two Velan valves were tested during the EPRI test, both with the same model number. One was an older version of the valve and is similar to the TMI valves. The other is a newer version and is also similar to the TMI valves with regard to internals and operation. The older valve has an SB-00-15 operator and the new valve has an SMB-000-10 operator. Both valves operated satisfactorily during the testing. The older valve started with a torque switch setting of 1.7 (155ft-lbs). Supplemental tests were run down to a torque switch setting of 1.0 (82 ft-lbs). The new valve started with a torque switch setting of 1.5 (110ft-lbs). The supplemental tests were run down to a torque switch setting of 1.0 (82ft-lbs). During the tests the inlet pressures during flow were approximately 2300 psig - 2400 psig with flows of approximately 235,000 lb/hr. (The older valve's operator was originally procured for a 12" gate valve per Limitorque). There was minor seat leakage at the lowest torque switch setting, but did not affect valve operation.

The TMI-1 valve originally had a torque switch setting of 1.5 - 2.0 (60ft-lbs to 75ft-lbs). The setting was revised in 1981 to 2.75 (98ft-lbs). This revision was due to a review of the torque switch setting based on the EPRI test data and using a 2750-psi delta P. The required output torque was calculated in the same manner as the output torque calculated for the EPRI test valves. The reason why the EPRI test valves have a higher torque is because the EPRI valves are 3 in. valves. The area term of the differential pressure component is the reason for the increased required torque in the EPRI valves. Based on the Limitorque method of calculating, the TMI-1 torque switch setting is consistent with the EPRI test data.

The Limitorque method of calculating required output torque uses the differential pressure to determine the required total stem thrust. The type of fluid causing the differential pressure is not of a concern. Only the differential pressure is important.

The EPRI valves were tested in the horizontal position and the TMI-1 valve is installed in the vertical position. The difference in orientation does not affect the test results or the application of the test results to TMI-1. The valves are designed for both orientations. Also the Limitorque sizing calculation does not require the orientation. The valve disc is guided in the valve body so that internal valve forces do not affect valve operation or required stem thrust. Therefore, operator sizing is the same for both vertical and horizontal orientation.

The TMI-2 block valve is a Velan Model F9-354B-13MS, the TMI-1 valve is a Velan Model F9-454B-13MS. The only difference is that the TMI-2 valve is rated at 1500 lbs and the TMI-1 valve is rated at 2500 lbs. Both valves have the same size motor operator, SMB-00-10. Although the TMI-2 torque switch setting cannot be verified, it is assumed that it is the same as the original TMI-1 torque switch setting. This assumption is based on the fact that the TMI-1 valve was originally procured for TMI-2 and the TMI-2 valve was purchased as a direct replacement for the original valve. The valve installed in TMI-2 did operate satisfactorily during the March 29, 1979 incident. The torque switch setting is assumed to be approximately 1.5 - 2.0 (output torque of 60ft-lbs to 75ft-lbs).

Item 12. Describe what steps are being taken to remedy the recent corrosion observed on the TMI-1 PORV which has been attributed to excessively corrosive reactor coolant water. Are the valves being modified in any way to help eliminate this problem? It is our understanding that the loop seal in the safety valve inlet piping has been eliminated. Will this aggravate the corrosion of the safety valves since the valves will now be in direct contact with the pressurizer steam?

Response:

Recent corrosion problems associated with the PORV reported on March 7, 1983 (LER 83-003) and the remedy was subsequently discussed in Rev. 1 to that LER dated June 6, 1983. By eliminating the residual sulfur in the RCS through cleaning (H_2O_2) and hydrolazing, deletion of the sodium thiosulfate tank, refurbishment of the valve, frequent chemical monitoring, and valve inspection further corrosion is expected to be minimized. The safety and relief valves are not being modified as a result of this problem. The advantage of the loop seal was to reduce H_2 cutting of the valve seats. Experience at similar plants has shown this effect to be very minor.

Item 13. The submittal describes the intended modifications to mount the safety valves and the PORV on the pressurizer nozzles. This modification would significantly affect the loads on the pressurizer nozzle. The submittal does not discuss the effect of the modification and the effect of the valve discharge loads on the ASME Section III, Class 1 analysis of the pressurizer nozzles. Verification that the Section III, Class 1 stress limits are met should be provided.

Response:

The original design of the B&W NSSS provided for safety valves to be mounted directly on the pressurizer nozzle. During the construction phase, Met Ed decided to move the valves to the end of the loop seal and provided a justification for that new design. GPUN, as a result of the EPRI test results, has returned the system to its original configuration.

Item 14. The submittal states that the safety valves and PORV connections to the pressurizer are assumed as anchors. It does not mention the large displacement of the connection due to the thermal expansion of the pressurizer when heated to operating conditions. Verification should be provided that the displacement were considered in the stress analyses of the piping and pressurizer nozzles.

Response:

Pressurizer nozzle thermal growths were accounted for by using anchor movement inputs in thermal analysis (for example, vertical thermal displacement = 1.375"). Thermal movement calculations are included in Attachment 1, pp. 32-34.

Item 15. The submittal states that the valve nozzle loads at the outlet flanges imposed by the discharge piping exceeds the allowable listed in the vendor catalog for the safety valves and exceeds those shown in the previous design for the PORV. It states that the loads for both types of valves have been re-evaluated by Dresser, the vendor, and found to be acceptable. However, the acceptance criteria and details of the analyses are not given. Sufficient additional information should be provided so that the acceptability of the nozzle loads can be verified or appropriate references cited.

Response:

By letter dated November 11, 1982, GPUN was notified by Dresser Industries as to the acceptablility of the loads at the outlet flanges: the loads were combined as follows:

Normal Operation	Deadweight # + Thermal Case #
Normal Operation (safe shutdown earthquake-valve closed)	Deadweight # + 2 x OBE Seismic + Thermal Case 3
Upset Operation (valve open)	Deadweight + Thermal + Blowdown
Upset Operation (safe shutdown earthquake-valve open)	Deadweight + Thermal + Blowdown + 2 x OBE Seismic

Nozzle load information is provided on page 55 of the revised Attachment 1.

Item 16. The submittal identifies the initial conditions and valve opening times for the safety valves and PORV analyses. However, the method of handling the valve resistances is not described and the corresponding flows are not reported. Since the ASME code requires derating the safety valves to 90% of predicted flow, actual flows of 100% of rates are likely. Additional information should be provided describing considerations of safety valve derating and describing methods used to predict the flows for the safety valves and the PORV.

Response:

The valve flow areas used in the RELAP-V models were chosen so as to produce a steady state steam flow of 370,968 lbm/hr @ 2500 psig for each SRV and 116,667 lbm/hr @ 2300 psig for the PORV. These values correspond to rated flow corrected for 10% ASME derating and a 5% error. These values conservatively maximize the discharge piping analysis.

The SRV opening times used were obtained from Tables 3.1.1.b and 3.1.1.c of the EPRI Safety Valve Test Data Report corresponding to the short inlet configuration. The shortest opening times reported are 0.012 sec. and 0.043 sec for steam and water conditions, respectively. Therefore, the SRV opening characteristic used in the RELAP5 analysis was linear opening at 0.012 and 0.040 sec, respectively for the steam and water cases.

Item 17. Two valve opening sequences were considered in the submittal, the two safety valves opening simultaneously and discharging without PORV flow and the PORV discharging by itself. These sequences however, may not bound the forces for all possible valve opening sequences. The experience of EG&G Idaho indicates that maximum forces would be expected when the sequence of opening is such that the initial pressure waves from the safety valves opening reach the common junction, located 1-1/2 ft above the drain tank, simultaneously. The safety valve lines and the PORV line were apparently modeled independently; however, if the PORV is discharging with flow past the junction when the safety valves open, piping loads may be significantly affected. Additional justification should be provided to demonstrate that the sequences considered in the submittal are adequate.

Response:

The sequences used for the SRV's and PORV are conservative for discharge piping transient analysis and provide bounding loads.

(a) SRV Sequencing:

Both SRV-A and SRV-B are conservatively assumed to actuate simultaneously. The design basis loading on the SRV discharge piping is based on the 400°F subcooled water discharge case. Despite having a larger force magnitude, the initial wave spike is not the controlling load. The fluid momentum results in a later and broader peak on the discharge piping which piping analysis has determined to be the controlling case. These force time history profiles may be seen by reviewing Attachment 1 of our submittal.

The length of piping between the SRV's and the common junction is approximately 125 ft. The length of SRV-A discharge piping is approximately 6 ft. longer than that of SRV-B. Further information concerning the fluid condition within the SRV discharge lines and their common junction can be obtained by reviewing the RELAP5 computer run for 400°F subcooled water discharge included herein. Items such as nodal pressure within the discharge lines may be seen by studying the provided nodal pressure for components 7 and 17 for each RELAP5 major edit output. RELAP5 minor edit output also provides information concerning individual pipe segment forces and flow rates from the valves, to the common junction, and from the common junction in 1 millisecond intervals.

(b) PORV - SRV Sequencing:

The PORV discharge piping analysis was done separately. The resulting steady state backpressure at the common junction is 73 psia. Therefore, using a backpressure of 70 psia (RCDT rupture disc pressure rating) adequately models the SRV annalysis backpressure.

Conclusion: Based on the above, the conservative flow rates, and opening times used in the analysis, the SRV/PORV discharge piping hydraulic loading functions used are conservative.

Item 18. The adequacy of the thermal-hydraulic analyses could not be verified since sufficient detail is not provided in the submittal. To provide for a more complete evaluation, additional discussion should be provided for the rationale used in the selecting key parameters such as node spacing, time steps, valve flow area and choked flow junctions. Computer printouts of input and output for key problems should also be provided. Suggested key problems are the RELAP5 printouts for the 400°F subcooled water case for both the safety valves and the PORV.

Response:

Calculations and computer output are available for inspection at GPUN or the contractors facilities. A copy of RELAP V analysis for 400° subcooled water was transmitted to EG&G Idaho, August 5, 1983. In addition, the following general criteria were used in developing the RELAP V models:

(a) Nodal Spacing

Near the valve outlet the node size is initially restricted by the geometry of the pipe segment and are typically 0.5 ft. As downstream segments become longer, node length was sometimes increased but the volume change was always less than 50% for adjacent nodes.

Our contractors' experience (See Appendix A of our previous submittal) and sensitivity studies described in EPRI/CE Reference 1 of our previous submittal indicate this criteria is sufficient in modelling relief valve discharge transients.

(b) Time Step Size

The maximum time steps were evaluated using the Courant limit.

$$t = \frac{X}{V+C}$$

where: t = maximum time step

X = minimum nodal length

V = maximum phasic velocity

C - speed of sound

In addition, the minimum time step used was 1×10^{-10} seconds.

The maximum time step used was 1.0×10^{-4} sec.

(c) Valve Flow Rates

Valve flow rates are addressed in response to 16 above.

(d) Choking

RELAP V junctions were allowed to choke at all area changes.

Item 19. Solving the acceleration term of the momentum balance equation was used to develop a forcing function for the structural code. The experience of EG&G Idaho with this technique is that spurious data spikes will occur during water discharge transients if every RELAP 5 computational time step is used. However, if a finite time step is used the technique may not include the peak load. A discussion of the solution techniques should be provided which demonstrates the accuracy and applicability of results for water discharge transients.

Response:

The forcing functions for the structural code were calculated by solving the acceleration term of the momentum balance equation by using every RELAP V computational time step. Although this technique sometimes results in spurious force data spikes, this is not the case for most of the forces calculated. See Appendix B (Attachment 1) for force time histories for every RELAP V computation time step. No smoothing out was performed on the curves.

Item 20. Insufficient information is available to assess the structural analyses. A more complete assessment requires description to be key parameters used in the analyses such as damping, lumped mass spacing details of support models, and the integration time step.

The submittal infers that only the net unbalanced forces for the RELAP elements were used as input to the structural analysis. A discussion should be provided that describes how the axial extension from the balancing forces on each end of the elements was treated. Computer printouts of input and output for key problems should be provided. Suggested key problems are the TPIPE printouts for the 400°F subcooled water case for the safety valves lifting simultaneously and for the PORV lifting along.

Response:

The following parameters were used in the analyses.

(a) Damping Ratio = 0
How is zero damping applied?
In the direct integration method of TPIPE, the damping matrix C used is computed by

$$C = \alpha M + \beta K$$

Where M is the mass matrix, K is the stiffness matrix. The α and β are arbitrary proportional factors. The damping ratio is specified by specifying α and β in the TPIPE time history analysis inputs. Zero damping is applied by specifying $\alpha = \beta = 0$ in TPIPE inputs.

This method is derived from Eq. (13 - 24) in Dynamics of Structures, by R. W. Clough & J. Penzien, McGraw-Hill, 1975.

No damping factor was used in the dynamic analysis.

(b) Lump Mass Spacing

Lumped masses were calculated for each nodal point by TPIPE Computer Code internally. Analysis node spacings can be found from the isometric drawings to the report.

(c) Modeling

The support eccentricity has been modelled. The details are available for inspection at GPU or Contractors' facilities.

(d) Integration Time Step

Analysis time step T = 0.001 second was used in the TPIPE time history analysis for the blowdown load cases.

The forcing functions were checked between RELAP 5 result and TPIPE input to make sure that the analysis time step was acceptable and that the peak forces were accounted for.

(e) Axial Extension

The axial extension effect on piping stresses from the balancing forces on each end of the pipe elements were considered in the static pressure analysis by using peak pressure. This should cover the effects of transient pressure and momentum forces combined. Furthermore, the axial balanced forces have no effect on the support design.

(f) SRV lifting simultaneously

(See response to item 21.)

(g) Print-out of RELAP V for 400° subcooled water condition were transmitted to EG&G Idaho, August 5, 1983.

Item 21. The submittal indicates that the three piping branches were assumed to be structurally independent and that the connectons to the pressurizer and drain tank were treated as anchors. The interaction of the three branches at the junction 1-1/2 ft. above the drain tank and the flexibility of the connections would appear to have a significant effect on the response and stress level of the piping. Additional justifications for these assumptions should be provided.

Response:

The three (3) piping branches were assumed to be structurally independent for the following reasons:

(a) The interaction of the three (3) branch lines and the common header is isolated from the pressurizer connections by three (3) intermediate anchors, one on each branch.

(b) The common junction is located in a relative stiff section of the piping adjacent to the pressurizer drain tank anchor.

(c) The piping dynamic stress in the region of the common junction are very low (i.e., OBE stress 2000 psi and blowdown 1000 psi) leaving sufficient margins for the possible differences that may result from a more refined structural analysis model.