SAFETY EVALUATION REPORT
PLANT HATCH UNIT-2

In a letter dated October 4, 1982 from J. T. Beckham of Georgia Power Company (the licensee) to the Director of NRR, a change to Technical Specification Figures 3.2.3-1 and 3.2.3-2 for Plant Hatch unit 2 was requested. The change would increase the operating limit minimum critical power ratio (OL-MCPR) which is based upon analysis of abnormal operational occurrences.

The licensee has requested this change to correct an error detected in the reload analysis which had been submitted to the staff in March, 1982, and was used to determine the present OL-MCPR specification. General Electric Company discussed the impact of the error in a letter dated June 8, 1982 from H.C. Pfefferlen (GE) to D.G. Eisenhut (NRC). The nature of the error was such that an operational transient occurring near the end-of-cycle could result in a violation of the MCPR safety limit if the plant was operating at the technical specification limit at the time of the transient. The plant is currently operating at an OL-MCPR which reflects the corrected analysis and is more restrictive than required by the present specification. The proposed change will conservatively revise the OL-MCPR based upon corrected analyses.

In support of its proposal, the licensee has submitted supplemental reload analyses based upon the approved General Electric Company report "Generic Reload Fuel Application", NEDE-24011-P-A-2 and NEDO-24011-A-2, July 1981. Pressurization event analyses were performed using the approved (and corrected) ODYN code. The results, using the "B" option for calculating ΔCPR, indicate that the feedwater controller failure to maximum flow event is the most limiting transient. The OL-MCPR for P8x8R fuel must be ≥ 1.29, and for 8x8R fuel the OL-MCPR must be ≥ 1.27. The exact value is determined from the attached Figures 3.2.3-1 and 3.2.3-2 and depends upon measured scram times as defined in specification 3/4.2.3.

We have reviewed the Technical Specification change requested by the licensee. We note that for the limiting event, feedwater controller failure to maximum demand, credit is assumed for operation of the high water level (L8) trip and the turbine bypass system. Accordingly, we require that technical specifications be included to ensure the operability of these systems. (Attachment 1 is a sample specification which we have found acceptable for other plants).

With the addition of technical specifications for the turbine bypass system and the L8 trip, we find that the technical specification change proposed by the licensee is acceptable. This conclusion is based upon our review of the licensee's submittal which indicates that the proposed action does not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated;

(2) Create the possibility of an accident of a type different from any evaluated previously; or

(3) Involve a significant reduction in a margin of safety.

3/4.7.8 MAIN TURBINE BYPASS SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8 The main turbine bypass system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION: With the main turbine bypass system inoperable, restore the system to OPERABLE status within 2 hours or determine MCPR to be equal to or greater than the applicable MCPR limit without bypass within one hour or take the ACTION required by Specification 3.2.3.

SURVEILLANCE REQUIREMENTS

4.7.8 The main turbine bypass system shall be demonstrated OPERABLE at least once per:

a. 7 days by cycling each turbine bypass valve through at least one complete cycle of full travel, and

b. 18 months by:

1. Performing a system functional test which includes simulated automatic actuation and verifying that each automatic valve actuates to its correct position.

2. Demonstrating TURBINE BYPASS SYSTEM RESPONSE TIME to be less than or equal to 0.30 seconds.

3/4.3.8 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.8 The feedwater/main turbine trip system actuation instrumentation channels shown in Table 3.3.8-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.8-2.

APPLICABILITY: As shown in Table 3.3.8-1.

ACTION:

With a feedwater/main turbine trip system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.8-2, declare the channel inoperable and either place the inoperable channel in the tripped condition until the channel is restored to OPEABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value, or declare the associated system inoperable.

a. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 7 days or be in at least STARTUP within the next 6 hours.

b. With the number of OPERABLE channels two less than required by the Minimum OPERABLE Channels per Trip System requirement, restore at least one of the inoperable channels to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.3.8.1 Each plant system 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 4.3.9.1-1.

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

TABLE 3.3.8-1

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

FUNCTIONAL UNIT	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITIONS
a. Reactor Vessel Water Level-High, Level 8	3	1

TABLE 3.3.8-2

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

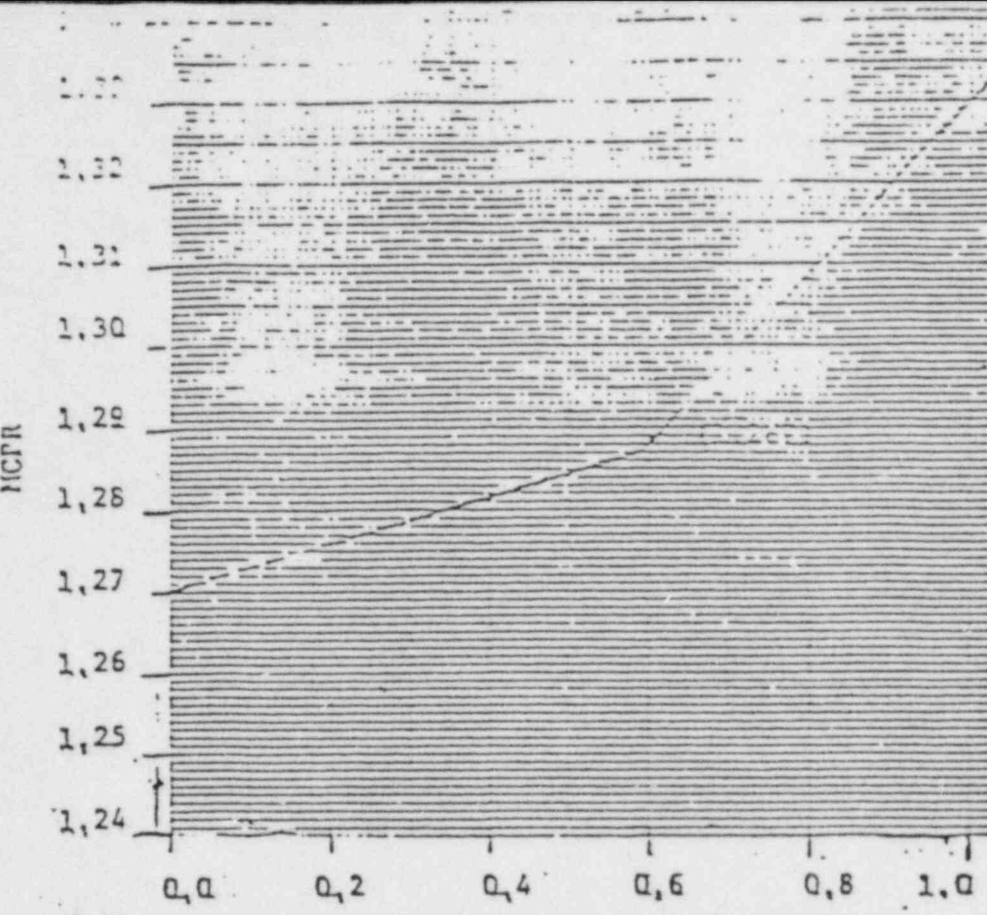
FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
a. Reactor Vessel Water Level-High, Level 8	≤ 54.5 inches*	≤ 56.0 inches

*See Bases Figure B 3/4 3-1.

TABLE 4.3.8.1-1 (Continued)

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

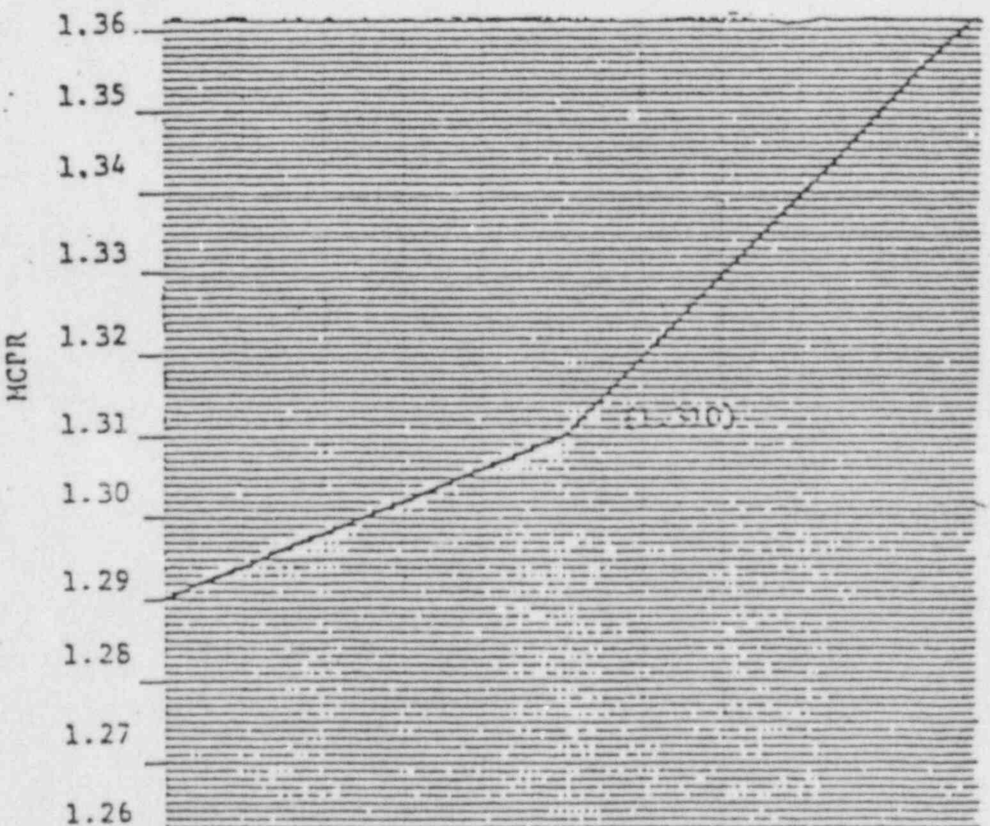
FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
a. Reactor Vessel Water Level-High, Level 8	NA	M	R	1



τ MCPR LIMIT FOR 8X8R FUEL AT RATED FLOW

FIGURE 3.2.3-1

1.287



1.310

τ MCPR LIMIT FOR P8X8R FUEL AT RATED FLOW

FIGURE 3.2.3-2

Docket File
NRC PDR
Local PDR
PRC System
NSIC
LB#2 Reading
EHylton
MHaughey
EBordenick
ACRS (16)
EJordan
NGrace
RMescott
MFliegel

Mr. C. V. Mangan
Vice President
Nuclear Licensing and Engineering
Niagara Mohawk Power Corporation
300 Erie Boulevard West
Syracuse, N.Y. 13202

Dear Mr. Mangan:

SUBJECT: NINE MILE POINT UNIT 2 PROBABLE MAXIMUM PRECIPITATION (PMP)

In your letter to Mr. A. Schwencer dated May 11, 1984, concerning the use of Hydrometeorology Reports (HMR) 51 and 52 as the basis for the PMP at Nine Mile Point Unit 2 (NMP-2), you requested that (1) the NRC request a meeting be arranged with the National Oceanographic and Atmospheric Administration (NOAA) and your staff to discuss and clarify the use of HMR 51 and 52 at the NMP-2 site and (2) the NRC Committee for the Review of Generic Requirements (CRGR) review the use of HMR's 51 and 52 as a design basis for NMP-2.

The meeting you requested with NOAA was held on May 15, 1984. During that meeting representatives of NOAA discussed why HMR's 51 and 52 are appropriate for the NMP-2 site. Translation of the Smethport, PA storm and the basis of the 0.7 ratio used to determine the one hour PMP rainfall were also discussed. During the meeting the NRC staff also discussed alternate means of dealing with potential flooding.

In our letter from Thomas M. Novak to Gerald K. Rhode dated February 3, 1984, we stated the reasons why use of HMR 51 and 52 as a design basis for NMP-2's PMP have been evaluated to be in conformance with the SRP.

As noted in our letter of February 3, 1984, if there are still objections to the use of HMR's 51 and 52 you have the right to appeal. If you do intend to appeal, it should contain a clear statement of your position along with supporting justification. The appeal process is described in Generic Letter 84-08 "Interim Procedures for NRC Management of Plant-Specific Backfitting."

If you have any questions concerning the above information, please contact the licensing project manager, Mary F. Haughey at (301) 492-7897.

Sincerely,

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

cc: See next page

*LB#2/DL	*LMEB/DE	*LB#2/DL	*AD/L/DL	D/DL
MHaughey:dh	RBallard	ASchwencer	TNovak	DEisenhut
06/ /84	06/ /84	06/ /84	06/ /84	06/10/84

*See previous concurrence

8407170140

Nine Mile Point 2

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Norman Rademacher,
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NIAGARA MOHAWK

NIAGARA MOHAWK POWER CORPORATION 300 ERIE BOULEVARD WEST SYRACUSE N.Y. 13202/TELEPHONE (315) 474-1511

May 11, 1984
(NMP2L 0050)

Mr. A. Schwencer, Chief
U. S. Nuclear Regulatory Commission
Licensing Branch No. 2
Washington, DC 20555

Dear Mr. Schwencer:

SUBJECT: Nine Mile Point - Unit 2
Docket No. 50-410

The Standard Review Plan (NUREG-0800, dated July 19, 1981) requires an analysis of the Probable Maximum Precipitation at Nine Mile Point Unit 2. FSAR Section 2.4 provides the results of the analysis performed for Unit 2. The analysis used Hydromet 33 and Corps of Engineers Engineering Manual as required by NUREG-0800. The Unit 2 design, based upon these references, prevents any local flooding at the site.

Subsequently, in Nuclear Regulatory Commission Question F240.11, it was requested that that we base the Probably Maximum Precipitation on Hydromet 51 and 52. We believe that these bases go beyond the current Standard Review Plan requirements since these reports are not referenced in the Standard Review Plan explicitly. We request the Nuclear Regulatory Commission Committee for the Review of Generic Requirements review this generic new requirement to determine if Hydromet 51 and 52 are applicable to the Nine Mile Point Unit 2 licensing basis.

Our review of Hydromet 52 indicates that the development of the Probable Maximum Precipitation curves for the Nine Mile Point Unit 2 area was heavily influenced by the Smethport, Pennsylvania storm. It is our opinion that it may be inappropriate to translate the Smethport storm to the Nine Mile Point Unit 2 site. Also, it is unclear what the basis is of the 0.7 ratio used to determine the one hour Probable Maximum Precipitation rainfall from the six hour Probable Maximum Precipitation rainfall. If Hydromet 51 and 52, as presently defined, were applicable to the Nine Mile Point Unit 2 site, our preliminary review indicates that local flooding could occur.

B001
1/0

During the period that the Committee for the Review of Generic Requirements is reviewing the applicability of Hydromet 51 and 52, we request that a meeting be arranged with the National Oceanographic and Atmospheric Administration (authors of Hydromet 52) and our technical staff to discuss and clarify these technical requirements, if applicable. We believe this to be an expedient approach in completing our evaluation of the Probable Maximum Precipitation.

Very truly yours,

CVMangan

C. V. Mangan
Vice President
Nuclear Licensing and Engineering

CVM/NLR:lf
cc: Director of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
Washington, DC 20555

NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

February 3, 1984

Docket No.: 50-410

Mr. Gerald K. Rhode
Senior Vice President
300 Erie Boulevard West
Syracuse, New York 13202

Dear Mr. Rhode:

Subject: Probable Maximum Precipitation (PMP) for Nine Mile Point Nuclear Station, Unit 2

On August 12, 1983, in a letter from A. Schwencer, we requested, in the Hydrology Section, that Niagara Mohawk Power Corporation (NMPC) provide information on the effects of PMP as determined from NOAA Hydrometeorological Reports 51 and 52. This PMP is considerably higher than the PMP rate of 8.4 inches per hour as stated in the FSAR.

Your response in Amendment 7 to the FSAR submitted December 16, 1983, indicated that your calculated rainfall rate of 8.4 inches per hour was developed from NOAA Hydrometeorological Report No. 33 which was approved at the construction permit stage.

On November 22, 1983, during a meeting on administrative matters, the NRC staff discussed with representatives of NMPC the use of Hydrometeorology Reports 51 and 52 for determining PMP. During that meeting, the NRC staff noted that Hydrometeorology Reports 51 and 52 contain updated information and more advanced methodology to determine the PMP. The staff also noted that although Hydrometeorology Report No. 33 is mentioned in the Standard Review Plan (SRP), NUREG-0800, the SRP also states that the latest methodology should be used when appropriate in determining PMP. The NRC staff further stated that the use of these later reports had been evaluated with respect to the SRP and determined to be in conformance with the SRP.

The staff's review procedures for evaluating flood levels have been and continue to be based on a PMP event. In our independent assessment of Nine Mile Point 2, we used current Corp of Engineering and National Weather Service methodology to determine the PMP depth. The analytical methods used by the staff are in accordance with generally accepted hydrological priniciples and procedures. Consideration of improvements in calculational methods is specially addressed in NUREG-0800 (SRP), Section 2.4.2 under "Review Procedures". However, NUREG-0800 provides for considerable flexibility in resolving potential flooding problems, recognizing that at the operating license stage the range of solutions may be limited by the status of plant construction. The primary focus is in assuring the capability of the plant to safely shutdown.

~~8402160646~~

Estimates of potential flooding problems, based on PMP, constitute a potential safety problem that must be addressed. In light of the information provided above, and considering the discussion held during the meeting on November 22, 1983, we believe your responses provided in Amendment 7 to the requests for additional information concerning PMP are inadequate. These responses should be revised to include a PMP rate determined in accordance with Hydrometeorology Reports 51 and 52. In order to support the licensing schedule for the SER, this information should be submitted to the NRC as an FSAR amendment no later than May 15, 1984.

If there are objections to use of Hydrometeorological Reports 51 and 52 in determining your PMP for Nine Mile Point 2 you should notify, in writing, Mr. Darrell G. Eisenhut, Director, Division of Licensing, of your desire for informal appeal meetings with the staff.

If you have any questions concerning the above information, please call the Licensing Project Manager, Mary F. Haughey at (301) 492-7897.

Sincerely,

T M Novak

Thomas M. Novak, Assistant Director
for Licensing
Division of Licensing

cc: See next page

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

December 6, 1983

Docket No.: 50-410

APPLICANT: Niagara Mohawk Power Corporation (NMPC)

FACILITY: Nine Mile Point Unit 2

SUBJECT: SUMMARY OF MEETING WITH NMPC TO DISCUSS ADMINISTRATIVE MATTERS RELATED TO NINE MILE POINT UNIT 2 (NMP-2)

On November 22, 1983, the NRC staff met with representatives from NMPC to discuss administrative matters related to NMP-2.

Among the subjects discussed were the bases for a number of requests for additional information that were requested for NMP-2. Specific examples in Seismology (determination of SSE to be used), Hydrology (use of standards to determine PMP), and Structural Engineering (information needed when codes other than those in the SRP are used) were discussed.

The applicant noted that a significant amount of work had been done in the area of seismology that was not accounted for in the requests for information. The NRC staff stated that detailed evaluations are not performed in the requests for information but are done later in the Safety Evaluation Report (SER).

The applicant stated they considered it sufficient to use Hydrometeorology Report No. 33 for determination of Probable Maximum Precipitation (PMP) as it is specifically mentioned in the Standard Review Plan (SRP) NUREG-0800. The NRC staff responded that although this standard is mentioned, the SRP also states that the latest methodology should be used when appropriate in determining PMP. Hydrometeorology reports 51 and 53 contain updated information and more advanced methodology to determine the PMP. These later reports are being used for other plants. The staff has evaluated the use of these later reports with respect to the SRP and have determined it is in conformance with the SRP.

For Category I structures, and interior structures of containment which were designed and built to ACI 318 rather than ACI 349 as referenced in the SRP, the applicant was requested to identify and justify, with respect to safety, all deviations of these structures from the applicable requirements of ACI 349 as amended by the Regulatory Guide 1.142. The NRC staff will then evaluate these deviations and justifications to assure there is no impact to the safety of this plant as a result of these deviations.

~~8312120173~~

The NRC staff noted that the applicant could make use of the appeal process to take discussions to a higher level whenever agreement could not be reached between the NRC staff and the applicant.

The applicant noted that communications between the NRC staff and the NMPC staff should remain open to provide necessary clarifications expeditiously.

A copy of the attendance list is attached.

Mary F. Haughey

Mary F. Haughey, Project Manager
Licensing Branch No. 2
Division of Licensing

Attachment:
As stated

cc w/ attachment:
See next page

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Mr. Jay Dunkleberger, Director
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Attachment

ATTENDANCE
11/22/83

Name	Organization	Title
C. Mangan	NMPC	Vice President, Nuclear Engineering & Licensing
A. Zallnick	NMPC	Licensing Manager
N. Rademacher	NMPC	Licensing Engineer
A. Schwencer	NRC	Licensing Branch Chief
M. Haughey	NRC	Licensing Project Manager
G. Lear	NRC	Chief, Structural & Geotechnical Engineering Branch
R. Ballard	NRC	Chief, Environmental and Hydrological Engineering Branch
M. Fliegel	NRC	Section Leader, Hydrology
S. Brocum	NRC	Section Leader, Geology

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

November 27, 1983

Docket No.: 50-410

APPLICANT: Niagara Mohawk Power Corporation (NMPC)

FACILITY: Nine Mile Point Unit 2

SUBJECT: SUMMARY OF MEETING WITH NMPC ON LIQUID PATHWAYS AT NINE MILE POINT UNIT 2

On November 17, 1983, the NRC staff met with representatives from NMPC and Stone & Webster Engineering Corporation to discuss Liquid Pathways at Nine Mile Point Unit 2.

During the meeting, NMPC stated that they believed that their use of Hydrometeorology Report No. 33 to determine the Probable Maximum Precipitation (PMP) is appropriate as this report is referenced in the Standard Review Plan (SRP). The NRC staff responded that while the SRP does identify Hydrometeorology Report No. 33, the SRP also notes new improvements in analytical techniques should be taken into account at the OL-stage review if significant changes in estimated flood levels result.

Liquid Pathways and Class 9 accidents were discussed. The NRC staff discussed the level of detail the analysis of liquid pathways should contain, including such things as the amount of water and fish from Lake Ontario which would be consumed. Source terms presently used by the NRC staff are taken from a postulated accident of a PWR. Final research into source terms expected for a BWR event may be available in a research report about June 1984. Ground velocity and permeability at the Nine Mile Point Unit 2 site were discussed. The De watering System at Nine Mile Point was turned off during the summer of 1983 and measurements were taken. Some of this data may be useful in calculating ground permeability at the site.

NMPC was requested to supply the Class 9, liquid pathways information requested during the acceptance review no later than early March 1984, in order to support the schedule for the Draft Environmental Statement (DES).

A list of attendees at this meeting is included as Attachment 1.

Mary F. Haughey

Mary F. Haughey, Project Manager
Licensing Branch No. 2
Division of Licensing

Attachment: As stated

cc: See next page

8312090264

Mr. Gerald K. Rhode
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Attachment 1

ATTENDANCE
11/17/83 - Liquid Pathways

Name	Organization
Mary Haughey	NRC - PM
Joe Feyder	Stone & Webster
Al Capellini	Stone & Webster
Y. C. Chang	Stone & Webster
Jim Carter	NRC/RSB
R. F. Zallnick	NMPC Licensing
N. L. Rademacher	NMPC Licensing
Mike Fliegel	NRC/EHEB
Rex Wescott	NRC/EHEB
M. S. Stocknoff	Stone & Webster
M. J. Hazzan	Stone & Webster

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

JUL 30 1984

Docket Nos.: 50-528/50-529
and 50-530

Mr. E. E. Van Brunt, Jr.
Vice President - Nuclear Projects
Arizona Public Service Company
Post Office Box 21666
Phoenix, Arizona 85036

Dear Mr. Van Brunt:

Subject: Results of Appeal Meeting Concerning the Palo Verde Alternate Shutdown Capability

The purpose of this letter is to confirm the telephone discussions we had with you on June 25, 1984 regarding the outcome of the subject appeal meeting held on May 31, 1984. The issue involved whether a source range neutron flux monitor would be required for the Palo Verde remote shutdown panel.

As we stated, your appeal has been granted based on the following design features for the Palo Verde plant and your commitment to perform a confirmatory probabilistic risk assessment (PRA) analysis for this issue;

(1) the extra worth of the control rods in comparison to other typical PWR plants,

(2) the larger coolant inventory in the primary system,

(3) the ability to measure boron concentration with the boron meter in the line downstream of the letdown isolation valve, and

(4) the remote shutdown panel instrumentation includes a log power meter and direct indication of reactor coolant system temperature and pressure.

The purpose of the PRA is to provide further confidence that a source range neutron flux monitor is not necessary to perform and control the required plant functions outside the control room for the Palo Verde plant.

~~8408130160~~

Please advise us as to when you will provide the requested PRA analysis. If you have any questions regarding this letter, you should contact Manny Licitra, the Licensing Project Manager.

Sincerely,

T M Novak

Thomas M. Novak, Assistant Director
for Licensing
Division of Licensing

cc: See next page

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

JUL 1 9 1984

Docket Nos.: 50-528/50-529
and 50-530

Applicant: Arizona Public Service Company

Facility: Palo Verde, Units 1, 2 and 3

Subject: Summary of Appeal Meeting for Palo Verde Regarding Source Range Neutron Flux Monitor for Alternate Shutdown

A meeting was held on May 31, 1984 in Bethesda, Maryland with representatives of the applicant. The meeting was held at the applicant's request to appeal the staff's position that a source range neutron flux monitor be included as part of the alternate shutdown function so as to provide a capability to directly monitor reactivity. Enclosure 1 lists the meeting attendees, and the viewgraphs used by the applicant are included as Enclosure 2. The meeting is summarized as follows:

Summary

Both the applicant and the staff summarized their positions regarding the need for a source range neutron flux monitor on the remote shutdown panel for Palo Verde.

The staff stated that Appendix R to 10 CFR Part 50 requires that the alternative shutdown capability for a nuclear plant include provisions for direct readings of the process variables necessary to perform and control certain plant functions, including the reactivity control function. To meet this requirement for a direct reading of the reactivity control function, the staff has required that a source range neutron flux monitor be provided as part of the alternative shutdown capability for PWRs.

The applicant stated that a source range neutron flux monitor is not necessary for the Palo Verde remote shutdown panel since, in the event that the control room needs to be evacuated, a criticality occurrence is very unlikely due to the following conditions:

(1) Prior to leaving the control room area, the operator will assure that the control rods are in by tripping the breakers.

~~8407300145~~

(2) The initial hot shutdown margin for Palo Verde (CESSAR System 80) with the control rods in is about 14%, which is about twice the margin for other typical PWR designs. This negative reactivity worth is sufficient to maintain the core subcritical in hot shutdown even with unborated water in the reactor.

(3) The operator will verify that makeup water to the reactor is from a borated source.

(4) In the event of an inadvertant boron dilution (addition of unborated water to the core) and assuming a maximum cooldown rate of 60°F per hour, there still would be a shutdown margin of 4% nine hours after the start of the incident.

(5) The operator will check the boron concentration in the reactor coolant by taking samples every hour starting at two hours after the control room is evacuated.

The applicant also stated that direct indication is being provided for all the other process variables (e.g., temperatures, pressure, flow rates and water levels) which can assist in determining whether a boron dilution event is occurring. Also, all other requirements for the remote shutdown panel are being met.

Based on the above, the applicant concluded that there is no apparent problem with the current Palo Verde remote shutdown panel design and that there is little, if any, benefit to including a source range neutron flux monitor on the panel (the applicant estimated it would cost about $500,000 per unit to install such a monitor). To provide added confidence in support of this conclusion, the applicant offerred to perform a PRA analysis on the issue.

Following the presentations, Mr. Novak, chairman for the appeal meeting, told the applicant that we would evaluate the information presented at the meeting. He stated that as soon as a decision is reached on the appeal, the applicant would be informed of the result.

E A Licitra

E. A. Licitra, Project Manager
Licensing Branch No. 3
Division of Licensing

Enclosures:
(1) Meeting Attendees
(2) Viewgraphs

Enclosure 1

Palo Verde Appeal Meeting

May 31, 1984

Manny Licitra	NRR/DL/LB#3
George Knighton	NRR/DL/LB#3
Tom Novak	NRR/DL/AD
Les Rubenstein	NRR/DSI/AD
Olan Parr	NRR/DSI/ASB
Jerry Wermiel	NRR/DSI/ASB
Nick Fioravante	NRR/DSI/ASB
Terry Quan	APS
Edwin E. Van Brunt, Jr.	APS
W. G. Bingham	Bechtel
S. H. Shepherd	Bechtel
R. Steve McKinney	APS
Donald R. Woodlan	Texas Utilities
Nick Baldasari	Bechtel
Charles Luguan	C-E
John Connally	C-E
George Davis	C-E
Jeff Brown	C-E
J. M. Betancourt	C-E

BACKGROUND

- HISTORY
  - STAFF/APPLICANT ACTIONS
- APPENDIX R REQUIREMENT
- ASSESSMENT OF STAFF POSITION
- PVNGS POSITION

APPENDIX R REQUIREMENT

III.L.2.d "THE PROCESS MONITORING FUNCTION SHALL BE CAPABLE OF PROVIDING DIRECT READINGS OF THE PROCESS VARIABLES NECESSARY TO PERFORM AND CONTROL THE ABOVE FUNCTIONS"

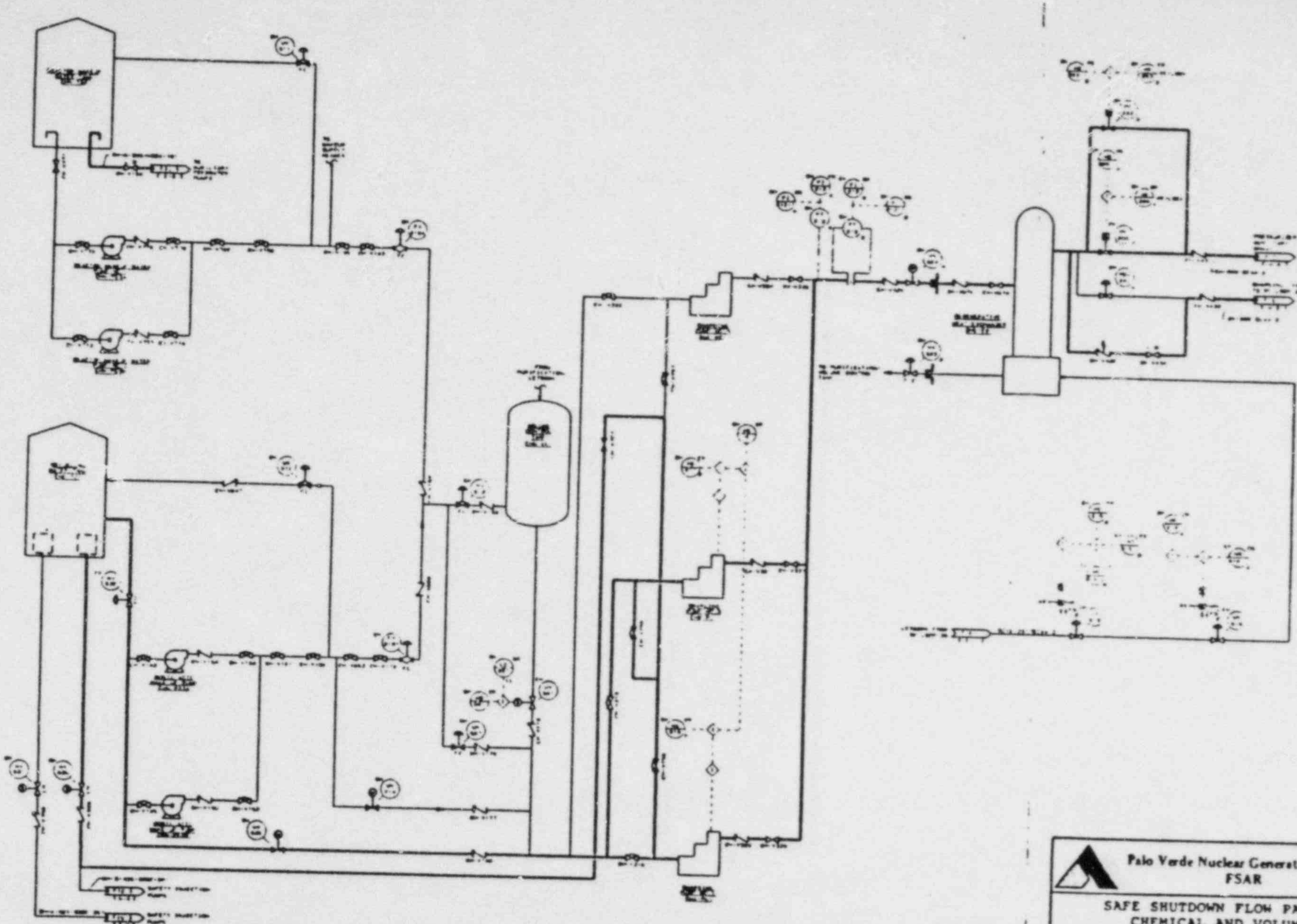
III.L.2.a "THE REACTIVITY CONTROL FUNCTION SHALL BE CAPABLE OF ACHIEVING AND MAINTAINING COLD SHUTDOWN REACTIVITY CONDITIONS"

BASIS FOR PVNGS
REMOTE SHUTDOWN PANEL

- MEET CESSAR INTERFACE
- PROVIDE DISCONNECT SWITCHES TO MEET APPENDIX R AS AGREED AT THE FIRE PROTECTION INDEPENDENT REVIEW BOARD MEETING (SEPTEMBER 1981)
- ADD T_{COLD} TO FACILITATE OPERATING PROCEDURES

DESIGN BASIS
CVCS BORATION

- AUTOMATICALLY TERMINATE CHARGING AND LETDOWN ON LOP
- WITH OR WITHOUT LOP, AN OPERATOR CAN READILY ISOLATE CHARGING AND LETDOWN
- CHARGING TO MAKEUP FOR SHRINKAGE IS EQUIVALENT TO EMERGENCY BORATION
- SHUTDOWN SEQUENCE



Palo Verde Nuclear Generating Station
FSAR

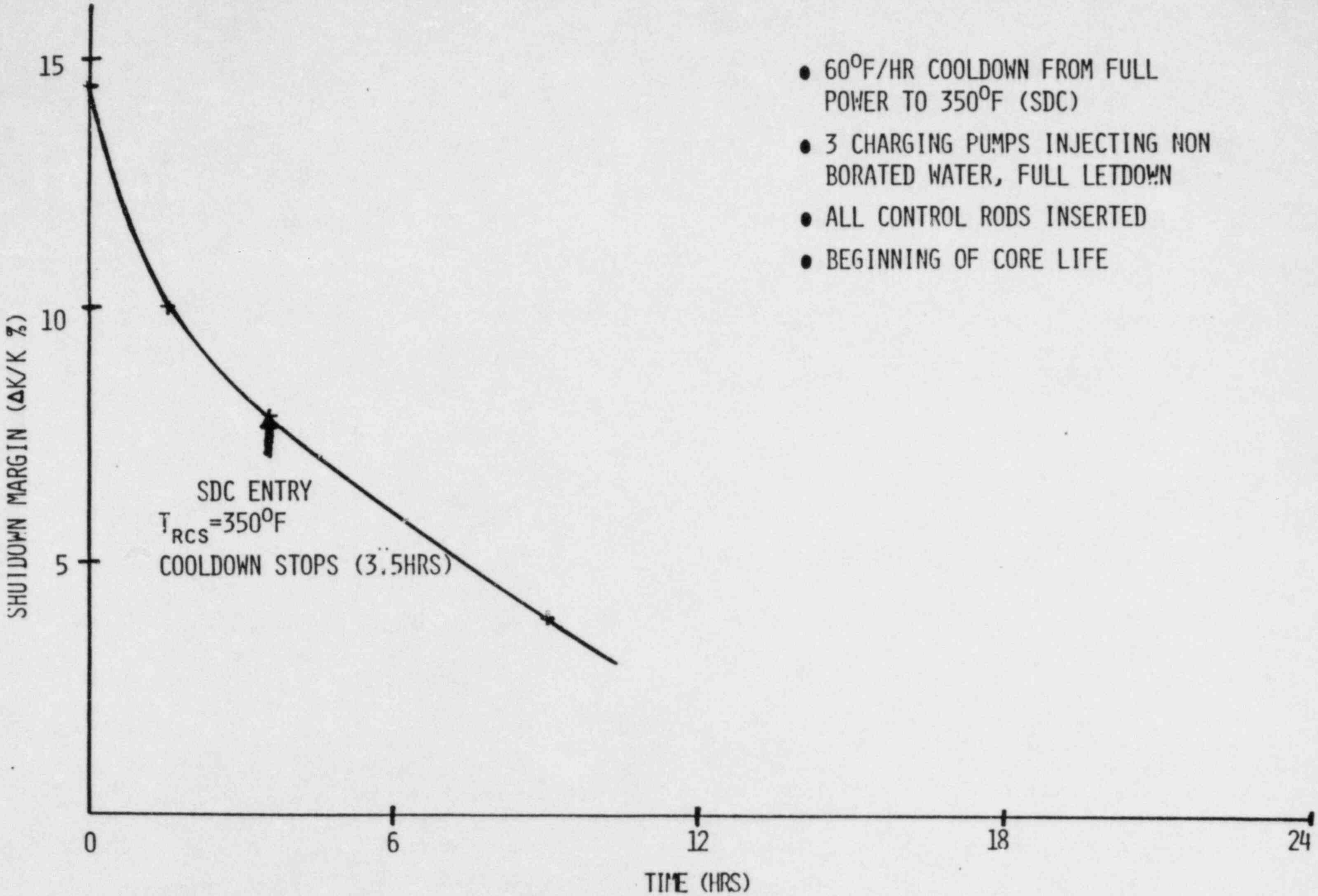
SAFE SHUTDOWN FLOW PATHS
CHEMICAL AND VOLUME
CONTROL SYSTEM

Figure 9B-41

SHUTDOWN SEQUENCE

I CONTROL ROOM FIRE
II MANUAL SCRAM
III ACTIVATE RSP - STABILIZE AT HOT STANDBY
IV BEGIN OPERATOR ROUNDS
   A. TERMINATE LETDOWN
   B. ISOLATE NON BORATED WATER SOURCES
V BEGIN COOLDOWN
   A. ALIGN BORATED SOURCE
   B. INJECT AS REQUIRED FOR SHRINKAGE
VI ACHIEVE COLD SHUTDOWN

SHUTDOWN MARGIN VERSUS TIME FROM CONTROL ROOM EVACUATION



COST

- BACKFIT COST = $1.5 MILLION (3 UNITS)

- SCOPE

  - DISCONNECT PANEL
  - CABLING
  - ELECTRONIC PROCESSOR
  - READOUT/MODIFICATION TO RSP

CONCLUSION

- NO APPARENT PROBLEM AS IS
- LITTLE, IF ANY, BENEFIT TO SFM'S
- WILLING TO DO PRA TO GIVE STAFF FURTHER CONFIDENCE IN CONCLUSION

NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

JUN 1 1 1982

Docket Nos.: 50-528/529
and 50-530

Mr. E. E. Van Brunt, Jr.
Vice President - Nuclear Projects
Arizona Public Service Company
Post Office Box 21666
Phoenix, Arizona 85036

Dear Mr. Van Brunt:

Subject: Request for Additional Information - Palo Verde Nuclear Generating Station

In the course of our review of the alternate safe shutdown capability for Palo Verde in the event of a fire (Appendix R, Section III.L), we have determined that the alternate shutdown system (remote shutdown panel) for the plant does not include the capability to monitor reactivity nor to verify adequate core cooling during natural circulation flow. The alternate shutdown system is required in the event of control room evacuation due to a fire.

Section III.L of Appendix R requires that a capability be provided for direct readings of process variables necessary to perform and control the reactivity control functions, the reactor coolant makeup functions and the reactor heat removal functions. Therefore, we require that you provide a source range neutron flux monitor and either an indication of the reactor coolant loop cold leg temperature (T_c) or reactor coolant average temperature (T_{avg}) as part of the available instrumentation for the remote shutdown panel or an alternate location which is independent of the control room.

We request that, within one week of receipt of this letter, you advise us as to when you will provide the response to the letter.

If you have any questions on this matter, please contact E. Licitra, (301) 492-7200, the Project Manager.

Sincerely,

Frank J. Miraglia

Frank J. Miraglia, Chief
Licensing Branch No. 3
Division of Licensing

cc: See next page

~~8206150424~~

Arizona Public Service Company

P.O. BOX 21666 • PHOENIX ARIZONA 85036

May 17, 1983
ANPP-23782 - WFQ/TFQ

Director of Nuclear Reactor Regulation
Licensing Branch No. 3
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Mr. George Knighton, Chief

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2 and 3
Docket Nos. STN-50-528/529/530
File: 83-056-026; G.1.01.10

Reference: A) Letter from F. J. Miraglia, NRC to E. E. Van Brunt, Jr., APS, dated June 11, 1982; Subject: Fire Protection
B) NUREG-0857, "Safety Evaluation Report related to the Operation" of Palo Verde Nuclear Generating Station Units 1, 2, and 3, dated November, 1981

Dear Mr. Knighton:

Reference A describes the results of the NRC's review of the PVNGS Remote Shutdown Panel (RSP). Contrary to 10CFR50.48, Appendix R to 10CFR50, and Section 9.5.1.9 of Reference B, the NRC staff is requiring PVNGS to meet Section III.L of 10CFR50, Appendix R. Reference A states:

> "Section III.L of Appendix R requires that a capability be provided for direct readings of process variables necessary to perform and control the reactivity control functions, the reactor coolant makeup functions and the reactor heat removal functions. Therefore, we require that you provide a source range neutron flux monitor and either an indication of the reactor coolant loop cold leg temperature (T_c) or reactor coolant average temperature (T_{avg}) as part of the available instrumenation for the remote shutdown panel or an alternate location which is independent of the control room."

Even though PVNGS is not required, nor have we committed to meet 10CFR50, Appendix R, Section III.L, we have evaluated the above stated request to include a source range neutron flux monitor (N_{source}) and either T_c or T_{ave} in the PVNGS RSP design. In order to further enhance the safety of PVNGS in the unlikely event the RSP will be needed, we will add this direct indication of T_c on the RSP even though other suitable means exist to

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determine this process variable. Installation of this instrument will be accomplished by the end of the first refueling outage for Unit 1, and prior to fuel load for Units 2 and 3. Justification for interim operation of Unit 1 without T_c on the RSP is provided below.

INTERIM OPERATION OF UNIT 1 WITHOUT T_c

The present design provides the following direct indications of core cooling; hot leg temperature (T_{hot}) and pressurizer pressure (P_{PZR}). An increase in either of these parameters would indicate to an operator inadequate core cooling at least as clearly as would T_c.

Furthermore, the operator can indirectly infer natural circulation flow through the use of a pressure-temperature curve for saturated steam. Steam generator pressure ($P_{S/G}$) is already provided on the RSP. Because of the relatively low flow under natural circulation conditions, T_c will be approximately equal to steam generator temperature ($T_{S/G}$). $T_{S/G}$, in turn, can be determined by the use of a saturated steam pressure-temperature curve, using $P_{S/G}$ (a direct RSP readout) as an entry value.

With the above indication available at the RSP, the operator is provided with sufficient instrumentation to determine adequate core cooling under natural circulation conditions. Therefore, the interim operation of Unit 1 without direct indication of T_c will not jeopardize the operator's capability to maintain the plant in a safe condition.

OPERATION WITHOUT N_{source}

In regard to N_{source}, PVNGS provides diverse indications of boron concentration and, accordingly, does not need to provide N_{source} on the RSP. PVNGS can control reactivity in accordance with 10CFR50 Appendix R without direct indication of N_{source}.

The identification of required instrumentation to perform and control reactivity is dependent upon the operating mode of the reactor. We have committed that prior to evacuation of the control room due to a fire, the operator will trip the reactor, verify that CEA's are fully inserted, and verify that core power is decreasing. Thus, the control functions of the RSP are limited to hot and cold shutdown modes.

A. HOT SHUTDOWN

The negative reactivity of the control rods alone is sufficient to maintain shutdown margin during hot shutdown. Thus, the actions taken in the control room prior to evacuation are sufficient to ensure reactivity control.

B. COLD SHUTDOWN

Chemical shim must be added to achieve shutdown margin for cold shutdown. The boron concentration of the charging flow added as shim is verified by Technical Specification every seven (7) days. Thus, it is not necessary to measure the actual neutron flux or RCS boron concentration post fire. Analysis has shown that a dilution event is impossible due to the boron content of the charging flow. However, even with the incredible assumption that boron concentrations become insufficient and the reactor becomes critical, the operator still has adequate indications of an inadvertent boron dilution event. CESSAR Section 15.4 identifies the maximum consequence of an inadvertent dilution event as not challenging fuel integrity. This meets the acceptance criteria of General Design Criterion 3 and of Section III.L of Appendix R. The difference between the Chapter 15 event and the event postulated here is that the control room alarms do not annunciate at the RSP. The RSP operator would instead react to an increase on the pressure and temperature instrumentation (T_{hot}, [T_{cold}], P_{PZR}, and $P_{S/G}$). Depending upon the severity of the control room fire, logarithmic power (N_{log}) could also be available on the RSP. The operator response is the same, namely, suspend charging and institute emergency boration procedures.

It is stressed, however, that such an event is unlikely, and there are indirect methods of shutdown margin verification available through the use of suitable procedures.

There are several methods of inferring reactivity control, of which N_{source} is one. Another method utilizes T_H, which is provided on the RSP in conjunction with RCS boron concentration. RCS boron concentration can be determined by several methods outside of the control room. These include the boronometer associated with the Post Accident Sampling System and grab sampling.

Figure 1, RCS temperature versus required boron concentration during a cooldown, illustrates how a minimum required boron concentration can be determined to maintain a given shutdown margin during a cooldown. This figure provides the operator with a valuable tool and an additional aid in maintaining reactivity control. Although shutdown margin is not necessary to verify reactivity control, it would be useful in the event of an unexpected cooldown while controlling the plant from outside the control room.

The operator has controls available locally to preclude boron dilutions and to align the charging system for boron additions, along with charging pump controls, valve control and indication is provided on the PVNGS RSP for the isolation valve between the Volume Control Tank (VCT) and the charging pumps. Since the VCT is the only low concentrated boron source that could be aligned to the RCS, the operator can verify that this source has been isolated to preclude a boron dilution. Control and indication is also provided locally for the valve between the Refueling Water Tank (RWT) and the charging pumps. This enables the operator to align a highly concentrated boron source to the RCS and assure reactivity control.

We believe the present design allows reactivity control outside of the control room, thus precluding the need for a source range neutron flux monitor.

Please contact me if you have any questions on this matter.

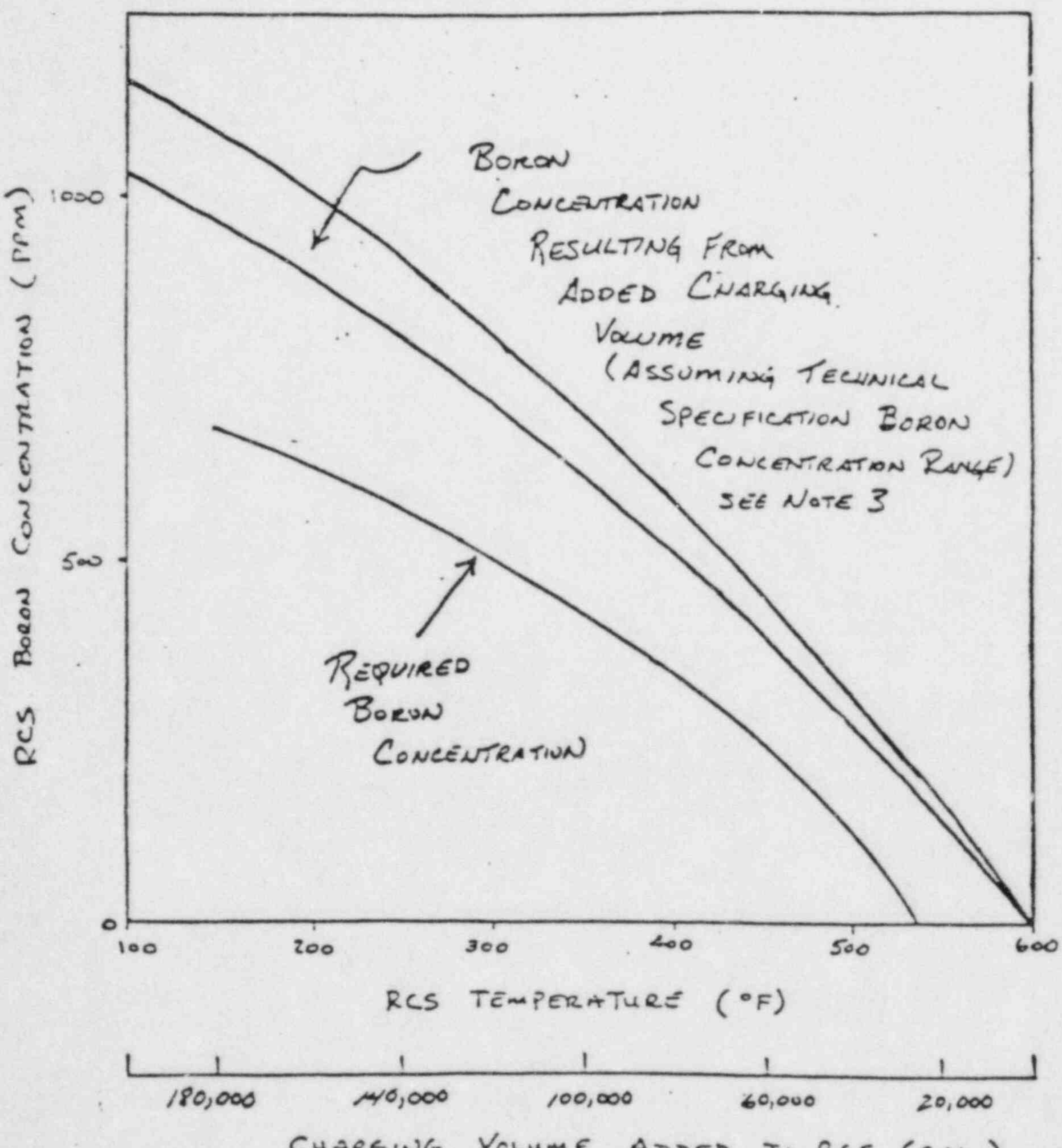
Very truly yours,

E. E. Van Brunt, Jr.
APS Vice President,
Nuclear Projects
ANPP Project Director

EEVB/TFQ/wp
Attachment

cc: E. Licitra (w/attach.)
G. Wermiel (NRC) "
A. C. Gehr "

FIGURE 1 : ILLUSTRATION OF ALLOWABLE RCS TEMPERATURE VS. REQUIRED BORON CONCENTRATION DURING A COOLDOWN



NOTE 1 : REACTIVITY CURVE CONSIDERS EQUIL. XENON, END OF CORE LIFE, WORST ROD STUCK OUT.

NOTE 2 : REACTIVITY CURVE REPRESENTS A MINIMUM BORON CONCENTRATION NECESSARY TO MAINTAIN A 2% SUBCRITICALITY.

NOTE 3: BORON CONCENTRATION OF CHARGING FLOW IS ASSUMED TO BE THE TECHNICAL SPECIFICATION REQUIREMENT OF 4000 TO 4400 PPM.

JUL 28 1983

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Docket Nos.: 50-528, 50-529
and 50-530

Mr. E. E. Van Brunt, Jr.
Vice President - Nuclear Projects
Arizona Public Service Company
Post Office Box 21666
Phoenix, Arizona 85036

Dear Mr. Van Brunt:

Subject: Source Range Flux Monitor for Palo Verde Remote Shutdown Panel

By letter dated May 17, 1983, you responded to our request to add two additional instruments to the Remote Shutdown Panel for Palo Verde for direct indication of process variables.

In your response, you state that as requested a direct indication of the reactor coolant loop cold leg temperature will be added to the Remote Shutdown Panel and you commit to complete installation of this instrument by the end of the first refueling for Unit 1 and prior to fuel load for Units 2 and 3. You also provide justification for interim operation of Unit 1 until the instrument is installed. Based on our evaluation of your submittal, we find this commitment to be acceptable.

In your response, you also state that the source range neutron flux monitor requested by us is not needed since reactivity can be controlled without direct indication of neutron source range flux. We have reviewed your response and conclude that it does not meet the staff position with regard to monitoring source range flux as shown in Enclosure 1. Therefore, we request that you revise your response to include a direct capability for monitoring source range flux from the Remote ShutdownPanel.

Please advise us as to when you plan to respond to this request. If you have any questions regarding the request, you should contact Manny Licitra, the Licensing Project Manager.

Sincerely,

Original signed by
George W. Knighton

George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing

Enclosure:
Staff Position

8308050571

Palo Verde

Mr. E. E. Van Brunt, Jr.
Vice President - Nuclear Projects
Arizona Public Service Company
P. O. Box 21666
Phoenix, Arizona 85036

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Washington, D. C. 20009

Enclosure

Staff Position

Section III.L.1 of Appendix R to 10 CFR 50 requires that alternative shutdown capability shall be able to achieve and maintain subcritical reactivity conditions in the reactor. Section III.L.2 of Appendix R to 10 CFR 50 requires provision for direct readings of the process variables necessary to perform and control the reactor shutdown function.

Among the process variables which are to be monitored are: source range flux, reactor coolant temperature, and steam generator pressure. These three have been controversial so we have set forth our basis for concluding that they are necessary in order to meet Section III.L of Appendix R.

Source Range Flux

Monitoring of core flux provides a direct indication of the reactor shutdown condition. The monitoring of other process variables would provide an inferred answer only. With regard to the fission process, changes in neutron flux provides the quickest means of assessing reactor criticality conditions. Dilution events caused by the postulated spurious operation of valves could result in power excursion which would not be readily detected by interpreting the changes in other process variables (such as reactor coolant temperature or pressure). Periodic sampling of the reactor coolant for boron concentration is considered inadequate for determining "real-time" boron requirements. Additionally, should the operators fail to detect a loss of negative reactivity in a timely manner, the capability to prevent a criticality is indeterminate since components needed for such actions may be unavailable due to fire. Thus, the provision for post fire source range flux monitoring is necessary to meet Section III.L.2 of Appendix R.

Reactor Coolant Temperatures

The reactor coolant temperatures, in conjunction with the reactor coolant system (RCS) pressure, are essential parameters necessary for plant cooldown and control and, hence, conformance with Section III.L.2 of Appendix R. The plant control elements which rely on accurate reactor coolant temperature indication are natural circulation, subcooling and pressurized thermal shock concerns.

(1) Natural Circulation: In the natural circulation mode of operation, the hot leg temperature, cold leg temperature and the difference between the hot leg and cold leg temperatures, $(T_H - T_C)$, provide indication by which natural circulation conditions can be determined. In order to verify that natural circulation has been established, normal plant procedures require the operator to use cold leg temperature T_C. It has been suggested that the saturation temperature corresponding to the secondary side steam generator pressure T_{sat}, will approximate T_C. The staff

acknowledges that such a condition can exist if natural circulation is occurring; however, the converse cannot be assumed. Cooldown is usually achieved by the operator controlling the steam generator pressure and auxiliary feedwater flow to the steam generators. Due to the inherent lag in response between the secondary and primary side, T_C cannot be inferred from T_{sat}. Natural circulation is normally determined by knowing T_H, T_C, observing that T_H and T_C are constant or decreasing, and by monitoring (T_H-T_C). Since normal control room procedures require the use of T_C in confirming natural circulation, emergency procedures should not deviate from this practice. Thus the provision for post fire cold leg temperature, T_C wide range indication is necessary for meeting Section III.L.2 of Appendix R.

(2) Upper Vessel Voiding: (Deleted)

(3) Subcooling: The bulk fluid temperature T_H provides a reliable indication of the degree of RCS subcooling when used in conjunction with the RCS pressure. T_H is also used as a means of verifying natural circulation. It has been suggested that exit core thermocouples (ECTs) provide temperature indications equivalent to T_H. ECT readings provide local temperature conditions above the core, and can give representative equivalent T_H provided the individual ECTs are judiciously selected, since ECT readings are dependent not only upon radial positioning, but also local flow rates past the ECTs. Thus, the provision for wide range ECTs is an acceptable alternate to wide range T_H loop RTDs for meeting Section III.L.2 of Appendix R, provided that the licensee demonstrates that their selection of ECTs will result in averaged temperature readings representative of T_H. Also, the licensees should demonstrate that under conditions where the reactor vessel upper head void is expanding thus, bringing higher temperature fluid into the outlet plenum and hot legs, the ECTs give a conservative indication of outlet plenum temperature.

(4) Pressurized Thermal Shock and Appendix G Considerations: T_C, in conjunction with the RCS pressure, provides a direct indication of the plant condition relative to the plant's pressure/temperature limits as it pertains to the Pressurized Thermal Shock considerations and the low temperature overpressure protection as outlined in Appendix G of 10 CFR 50. Due to the collective effect of the steam generator conditions (i.e., feedwater flow and steam generator pressure) on the primary coolant temperatures, and the inherent lag beteen the secondary and primary sides conditions especially during transient conditions, T_C may not be accurately inferred from the secondary side steam conditions.

Steam Generator Pressure

During non-power modes of operation, "control" is effected principally by adjusting secondary system parameters (the parameter usually specified by procedures is pressure) to compensate for variances in primary system performance. Maintenance of level in the steam generators may not be sufficient in itself to control the heat removal rate and thereby maintain a "hot standby" or "hot shutdown" mode, or translate from "hot shutdown" mode to "cold shutdown" mode. Improper pressure control may cause an imbalance in heat removal which could result in excessive depressurization, the result of which could be generation of an undesired bubble in the primary system (e.g., upper head for all PWRs or candy cane for B&W designs) or rapid cooldown and potential for violation of vessel pressure/temperature limits. For the monitoring of secondary system heat removal, two secondary system parameters should be known: level (inventory), and pressure. Thus, provisions for post fire steam generator pressure and level monitoring are necessary for meeting Section III.L.2 of Appendix R.

Instrumentation Guidelines

Section III.L.6 requires that, "Shutdown systems installed to ensure post-fire shutdown capability need not be designed to meet seismic Category I criteria, single failure criteria, or other design basis accident criteria, except where for required for other reasons, e.g., because of interface with or impact on existing safety systems, or because of adverse valve actions due to fire damage." Thus the monitors for the above listed parameters need not be "safety grade" in order to meet the requirements of Appendix R.

Section III.G.3 requires that, "Alternate or dedicated shutdown capability and its associated circuits, independent of cables, systems or components in the area, room or zone under consideration, shall be provided." For a postulated fire, an electrically independent monitoring capability for the above listed parameters should be provided outside the control room.

Based on the above, the revised list of instrumentation needed for PWRs is:

a) pressurizer pressure and level,
b) reactor coolant hot leg temperature or exit core thermocouples, and cold leg temperature,
c) steam generator pressure and level (wide range),
d) source range flux monitor,
e) diagnostic instrumentation for shutdown systems, and
f) level indication for all tanks used (e.g., CST).

The instrumentation needed for BWRs is unchanged.

Arizona Public Service Company

P.O. BOX 21666 • PHOENIX, ARIZONA [illegible]

November 23, 1983
ANPP-28284 - WFQ/TFQ

Director of Nuclear Reactor Regulation
Attention: Mr. George Knighton, Chief
Licensing Branch No. 3
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2 and 3
Docket Nos. STN-50-528/529/530
File: 83-056-026; G.1.01.10

Reference: (A) Letter from E. E. Van Brunt, Jr., APS, to G. W. Knighton, NRC, ANPP-23782, dated May 17, 1983.
(B) Letter from G. W. Knighton, NRC, to E. E. Van Brunt, Jr., APS, dated July 28, 1983.

Dear Mr. Knighton:

By letter, Reference (A), we responded to your request to add a source range neutron flux monitor to the PVNGS Remote Shutdown Panel (RSP). By letter, Reference (B), you stated that our response did not meet the staff position with regard to monitoring source range neutron flux. We would like to take this opportunity to restate our position on this issue.

The NRC staff concern is the loss of the reactivity control function from the Remote Shutdown Panel (RSP) due to potential boron dilution events caused by fire induced spurious operation of components. At PVNGS, the RSP would be used only when the control room becomes uninhabitable. Prior to evacuation of the control room, the operator will trip the reactor and verify that all control rods are fully inserted.

PVNGS has such an extremely high control rod worth that it is impossible to achieve a critical state at any temperature or at anytime during a fuel cycle provided that all rods are inserted. This can be further realized by noting that the HOT (564°F) Zero Power, Beginning of Cycle (BOC), clean critical all-rods-in boron level is estimated to be -364 ppm. Allowing an additional 250 ppm for cooldown to approximately 60°F, there is still a shutdown margin equivalent to -114 ppm boron. Thus, a subcritical condition is maintained, assuming no boron in the RCS, and all rods in. This indicates that the occurance of a boron dilution event does not affect the reactivity control function.

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Therefore, a source range neutron flux monitor is not required to assure reactivity control is maintained from the RSP.

Please contact me if you have any questions on this matter.

Very truly yours,

E. E. Van Brunt

E. E. Van Brunt, Jr.
APS Vice President
Nuclear Projects Management
ANPP Project Director

EEVB/TFQ/sp

cc: E. A. Licitra
G. Wermiel
A. C. Gehr

Arizona Public Service Company

P.O. BOX 21666 • PHOENIX, ARIZONA 85036

ANPP 28863 WLH/TFQ
February 14, 1984

Director of Nuclear Reactor Regulation
Attention: Mr. George Knighton, Chief
Licensing Branch No. 3
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2 and 3
Docket Nos. STN-50-528/529/530
File: 84-056-026; G.1.01.10

Reference: (A) Letter from G.W. Knighton, NRC, to E.E. Van Brunt, Jr. APS dated July 28, 1983.
(B) Letter from E.E. Van Brunt, Jr. APS, to G.W. Knighton, NRC dated November 23, 1983.

Dear Mr. Knighton:

The NRC, per Reference (A), reiterated its position with regard to the need for an independent source range neutron flux monitor at the PVNGS Remote Shutdown Panel (RSP). APS had provided a response, Reference (B). Further review has indicated a need to modify that response. This letter is also a response to Reference (A), which is to supersede our previous response.

Reference (A) states that the PVNGS design does not meet the NRC staff's position with regard to 10CFR50, Appendix R, Item II.L.2, which is to include the direct capability for monitoring source range neutron flux from the RSP. The NRC staff concern is the potential loss of reactivity control function from RSP due to potential boron dilution events caused by fire induced spurious operation of components.

APS believes that the NRC position is not justified for PVNGS because:

1. Other than Section III.G, III.J, and III.O, 10CFR50, Appendix R is not applicable to plants other than those "...licensed nuclear power electric generating stations that were operating prior to January 1, 1979..."

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2. PVNGS has committed to meet Appendix R, Sections III.G, III.J, and III.O. Compliance with Section III.L.2 is not applicable.

   [APS did advise the NRC (in the Fire Protection Independent Design Review, held February 25, 1981, and subsequently in the Fire Protection Evaluation Report, Amendment 3) that "PVNGS alternative shutdown capability provides the functions" that APS considers as meeting Appendix R, Section III.L. These evaluations contemplated that the features described in 3.B, C, and D below were adequate to meet the NRC position.]

3. There are sufficient design features and procedural guidance in the existing design for PVNGS to comply with the requirements of Criterion 3 of Appendix A of 10CFR50, which is the applicable licensing standard. Specifically the PVNGS design incorporates the following features:

   A. Only in the event of control room evacuation is alternate shutdown capability from the remote shutdown panel required.

   B. When evacuation of the control room becomes necessary, the control room operator manually trips the reactor, verifies power is decreasing and all rods are inserted.

   C. By procedure and by the proposed PVNGS Technical Specifications, the operator is required to maintain a shutdown margin of 6% $\Delta K/K$, in modes 3 and 4, hot standby and hot shutdown respectively, and 4% $\Delta K/K$ in mode 5, cold shutdown. Upon control room evacuation these margins will be verified by sampling of the reactor coolant system to monitor boron concentration, at least once per hour.

   D. The Reactor Makeup Water Tank is the only source of unborated water, which could lead to a boron dilution event. This tank is isolated from the charging pumps prior to the cooldown of the RCS. This assures RCS makeup will be from the Refueling Water Tank (RWT). The RWT, which has a Technical Specification requirement of 4000 to 4400 ppm boron concentration, provides water to the charging pumps via a gravity feed path or, alternatively, via the boric acid makeup pumps (if non-IE electrical power is available).

Based upon the preceding discussion, APS believes that the current design is adequate and that a backfit of a source range monitor at the remote shutdown panel is not required to assure reactivity control is maintained while shutting down the plant at that station.

If you do not accept the APS position as stated above, please arrange for appeal meeting at your earliest convenience.

Very truly yours,

E.E. Van Brunt Jr

E. E. Van Brunt, Jr.
APS Vice President, Nuclear
ANPP Project Director

EEVB/TFQ:pt

cc: E.A. Licitra
G. Wermiel
A.C. Gehr

TOLEDO
EDISON

RICHARD P. CROUSE
Vice President
Nuclear
(419) 259-5221

Docket No. 50-346

License No. NPF-3

Serial No. 1049

May 10, 1984

Director of Nuclear Reactor Regulation
Attention: Mr. John F. Stolz
Operating Reactor Branch Nc
Division of Operating Reactors
United States Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Stolz:

On March 26, 1984 (Serial No. 1036) Toledo Edison requested on Appeal meeting on the NRC proposed Technical Specifications concerning the Auxiliary Feedwater System. This was in response to your letter dated February 21, 1984 (Log No. 1455). Toledo Edison has re-evaluated its request for the appeal meeting and hereby withdrawals that request. We will submit requested Technical Specification by June 30, 1984 for the Davis-Besse Nuclear Power Station Unit No. 1.

Very truly yours,

RPC:GAB:lah

cc: DB-1 NRC Resident Inspector

A001
1/0

TOLEDO EDISON

RICHARD P. CROUSE
Vice President
Nuclear
(419) 259-5221

Docket No. 50-346

License No. NPF-3

Serial No. 1036

March 26, 1984

Director of Nuclear Reactor Regulation
Attention: Mr. John F. Stolz
Operating Reactor Branch No. 4
Division of Operating Reactors
United States Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Stolz:

This is in response to your letter dated February 21, 1984 (Log No. 1455) concerning Safety Evaluation Report (SER) addressing TMI Task Action Plan NUREG 0737 Item II.E.1.1 for Davis-Besse Nuclear Power Station Unit No. 1 Auxiliary Feedwater (AFW) System. The SER closed out five open items, but three items remain open and you requested proposed Technical Specifications for the open items listed below:

<u>Item 1</u> Proposed Technical Specifications which would require that all local manual valves in the auxiliary feedwater pumps suction and discharge lines are locked in the open position and that the locked open position of these valves would be verified on a monthly basis.

<u>Item 2</u> That your letter dated June 15, 1983, (Serial No. 956) be supplemented with proposed Technical Specifications which would require a flow verification test of the AFW system to put water into the Steam Generators after each extended cold shutdown.

<u>Item 3</u> Proposed Technical Specifications which would require that a dedicated individual who would be in communication with the Control Room to be stationed at the manual valves of the AFW system when conducting periodic tests of the AFW system which require local manual realignment of valves to conduct the periodic tests of the AFW system.

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The AFW system (II.E.1.1) has been discussed between Toledo Edison and the NRC staff for the past two years. The above open items were subject to verbal agreement and proposed Technical Specifications for item two were submitted on June 15, 1983 (Serial No. 956). Also discussed and mutually agreed upon were items one and three for which no submittal was required. We have implemented the submittal resulting from our discussions concerning the AFW system, but now your letter requests us to negate that verbal agreement.

Your letter requests Toledo Edison to either submit the proposed Technical Specifications or to request an appeal meeting. Toledo Edison hereby requests an appeal meeting.

Very truly yours,

RPC:GAB:lrh

cc: DB1 Resident Inspector

cj a/22

RANCHO SECO NUCLEAR GENERATING STATION UNIT 1

Effect of Internally Generated Missiles on the Auxiliary Feedwater System Outside Containment

Christine F. Kelton

Prepared By: C. Kelton

Taj M. Khan/V Arora

Reviewed by: V. Arora/T. Khan

DATE: 4-27-84

David Abbott

Approved by: D. Abbott
Supervising Mechanical Engineer
Sacramento Municipal Utility District

SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT ☐ 6201 S Street, Box 15830, Sacramento, California 95813; (916) 452-3211

May 3, 1984

DIRECTOR OF NUCLEAR REACTOR REGULATION
ATTENTION JOHN F STOLZ CHIEF
OPERATING REACTORS BRANCH 4
U S NUCLEAR REGULATORY COMMISSION
WASHINGTON D C 20555

DOCKET 50-312
RANCHO SECO NUCLEAR GENERATING STATION
UNIT NO 1
AUXILIARY FEEDWATER SYSTEM (AFWS) UPGRADE REVIEW
NUREG 0737 ITEM II.E.1.1

The District committed in our June 3, 1983 letter, to a walkdown of the AFWS which would determine any areas where a single, internally generated missle could disable both AFWS trains. The District has completed an analysis, and the attached report shows that a single internally generated missle can not disable both AFWS trains.

We, therefore, conclude that adequate protection is provided and that a walkdown, per se, is no longer required. If we can provide any additional information, please advise.

John J Mattimoe

John J. Mattimoe
General Manager
and Chief Engineer

Attachment

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P PDR

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TABLE OF CONTENTS

APPENDICES

I. SUMMARY

The purpose of this study was to evaluate the effects of postulated internally generated missiles on the Auxiliary Feedwater System (AFWS) components located outside containment. Missile protection is required if a single missile can simultaneously damage components of both AFWS trains.

AFWS components located in the yard area and auxiliary building include mechanical and electrical equipment, and instrumentation and controls. High energy piping, rotating equipment, and compressed gas storage systems were considered as credible missile sources.

There were no cases where a single missile could simultaneously impact both AFWS trains. It was therefore concluded that the AFWS is adequately protected from the effects of internally generated missiles and no additional protection is required.

II. PURPOSE

This study was prepared for the Rancho Seco Nuclear Generating Station Unit 1, to evaluate the effects of postulated internally generated missiles on the Auxiliary Feedwater System (AFWS) essential components located outside the containment building. The study was based on the Design Criteria attached to this report as Appendix 5.

III. SCOPE

The scope of this study is defined as follows:

A. Only those portions of the AFWS that are quality Class I and are located outside the containment building are considered. Internal missile protection for AFWS components inside containment is documented in the Updated Safety Analysis Report, Section 5.1.2.1.3.

B. Turbine missiles or externally generated missiles are not considered.

C. Secondary missiles or ricochet targets are not considered.

D. Future modifications to the AFWS as part of the EFIC package scheduled for installation during the 1986 outage will be evaluated for missile considerations at a later date and are not included here.

E. Gravity missiles are not considered.

F. Missile protection is required only if a single missile can simultaneously disable both trains of the AFWS. Damage to one train of the AFWS is permitted.

IV. DESCRIPTION

A. Auxiliary Feedwater System Protection Philosophy

The AFWS is shown schematically in Figure 1. Protection of the AFWS from internally generated missiles is limited to those Class I components which are required to mitigate the consequences of an accident, prevent a significant uncontrolled release of radiation, or place the plant in a cold shutdown condition.

The Class I components are as follows:

1. Mechanical equipment including the pumps, turbine driver, valves, and associated piping.
2. Instrumentation/control components used for indication, monitoring and control, and associated tubing.
3. Class 1E electrical components used for signal transmission and for powering and control of the mechanical and instrumentation components described above.

Piping and instrumentation diagrams, logic diagrams, and elementary drawings were used to identify the Class I components described above. A schematic of the Class 1E electrical components is shown on Figure 3.

This study has been limited to existing components, or components for which installation drawings have been released for construction. Future additions to the AFWS, such as the EFIC modifications, will be evaluated for missile protection as part of Rancho Seco's standard plant review procedure.

B. Methods of Analysis

The missile study was performed in two steps. The first step was to identify the physical location of all the Class I AFWS components using the piping area, and conduit and tray drawings for the yard areas and auxiliary building. The second step was to identify and locate all the potential missile sources in the vicinity of the AFWS: high energy piping, rotating equipment, and compressed gas storage system components.

1. High Energy Piping: Piping area drawings, piping and instrumentation diagrams (P&ID's), and physical inspection techniques were used for the location of all high energy lines(1) running through the yard areas. The line designation list (Ref. VII. F) was used to define the normal operating conditions. Lines which did not meet the high energy criteria were considered moderate energy with not enough energy to generate destructive missiles and were excluded from any furthur consideration. Lines that qualified as high energy because the temperature was greater than 200°F but where the pressure was less than 10 psig were also not considered to generate destructive missiles and were classified as moderate energy for the purposes of this study. These lines are identified by a double asterisk in the tabulation of moderate energy lines which are attached to this report as Appendix 2. Portions of the AFWS

(1) High energy fluid systems are defined as those pressurized systems or portions thereof in which the normal operating pressure or temperature exceed 275 psig and 200°F, respectively, for more than 2% of the time it operates during normal plant conditions.

which are not pressurized during normal plant conditions are also excluded from consideration as a high energy missile source and are included in Appendix 2.

High energy fluid system components that were considered as credible missile sources include:

a. Valve bonnets, stems, and body drain plugs

b. Temperature and pressure instrumentation connections

c. Welded dead-end flanges and caps

d. Vents, drains, and test connections

A tabulation of the missile sources by line number was compiled using the yard area piping drawings, P&ID's, and physical inspection. (This tabulation is included in this report as Appendix 1). The exclusions listed in Section II.C of the Design Criteria (Appendix 5) were then used to generate a list of design missiles. The missiles were assumed to eject in the direction of the applied force. The target or impactee for each of these design missiles was determined by physical inspection.

2. Rotating Equipment: General plant equipment arrangement drawings and the yard area piping drawings were used for the identification of all rotating equipment located in the yard area. Some of these are included in the AFWS piping layout shown on Figures 2 and 2A. Missiles from impeller fragmentation were assumed to eject in the plane of rotation of the impeller centerline. Targets of these missile sources were identified by physical inspection and are tabulated in Appendix 4.

3. Compressed Gas Storage: Compressed gas storage systems located in the yard area were identified on the piping drawings and equipment location drawings. Components were assumed to eject in the direction of the applied force with targets identified by physical inspection.

V. ANALYSIS

A. Yard Area

1. High Energy Piping Missiles: From the list of design missiles originating from high energy piping, a physical inspection was performed to determine the potential targets located within the direction of these design missiles. In most cases, these design missiles were found to hit either a concrete floor or ceiling, steel beams or columns, another section of the same line that ejected the missile, a walkway grating, or nothing at all (the missile would eject into free space and would not impact anything until it returned to earth as a gravity missile). Where the target was a walkway grating, consideration was given to the targets located on the other side of the grating for any possible damage. No consideration was given to secondary missiles or ricochet missiles. In only one instance, a capped weldolet on the main feedwater line to Steam Generator E-205A was determined to impact the valve positioner on FV-20527, the flow control valve on the auxiliary feedwater line to Steam Generator E-205A. Since the valve is related to only one AFWS train, no further analysis was performed. Electrical and instrumentation components of either AFWS train were not impacted by any missiles generated by high energy piping in the yard area.

2. Rotating Equipment: All of the pumps in the yard area were evaluated for AFWS missile impact and are tabulated in Appendix 4.

Four of the pumps, the Nuclear Service Cooling Water Pumps A & B (P-482A and P-4828), the Miscellaneous Water Hold-Up Tank Pump (P-983), and the Low Pressure Injection Header Warming Pump (P-251) are physically oriented such that there are no AFWS components within the plane of rotation at the impeller centerline.

The Component Cooling Water Pump P-462A is shielded from AFWS impact by its redundant pump, P-462B. Pump P-4623, however, could impact the AFWS at two points: line 31827-6"-GB and conduit M11249. The piping target is the AFWS common test line which is located above ground outside the missile shield. The point of impact of a missile generated by P-462B would be on non-seismically qualified piping downstream of FV-31855 as shown on Figure 1. The valve FV-31855 is a "fail-closed" valve which is normally closed except during AFWS testing. Damage to the line downstream of this valve will not compromise the ability of the AFWS to perform its safety function. The electrical target of P-462B is conduit M11249 which contains one of the redundant channels of cables for the AFW pump (P-318, P-319) bearing heaters. From the electrical schematic, Figure 3, it can be seen that the bearing heaters for both AFW pumps are powered independently by both electrical channels. The cable for the other channel is located in cable tray L11AD1, which runs parallel to M11249 on the other side of the overhead pipeway. It is conceivable that this cable tray could also be impacted by a missile from P-462B, even though it is some distance away.

However, it is not possible for both cables to be impacted by the same missile simultaneously because of their locations relative to the pump (A single impeller fragment ejected radially in the plane of the impeller cannot impact two parallel overhead cables located 20 feet apart at the same elevation).

The Demineralized Reactor Coolant Storage Tank Pumps P-622A and P-622B are partially shielded by structural columns sup,-rting an overhead pipe rack. There is a possibility, however, that cable tray L11AD1 could be impacted. As noted above, however, the cable for the AFWS pump bearing heaters that runs through this cable tray is a redundant cable. A missile ejected from P-622A or P-622B cannot simultaneously impact the cable tray and conduit M11249 containing the redundant cable.

The Spent Fuel Coolant Pump, P-272, is shielded on both sides by concrete support pillars and there are no AFWS components overhead. Spent Fuel Coolant Pump P-274 is shielded on one side by a concrete support pillar. Radiation monitor R-15018 is immediately adjacent to the pump on the opposite side and effectively shields the pump from AFW line 31823-6"-DB2 to Steam Generator E-205A.

In reviewing the effects of missiles generated by rotating equipment, it should be noted that this analysis has assumed that the pump casings are actually penetrated by impeller fragments that have enough residual energy to be destructive missiles. Even though it is unlikely that this would happen, it has been postulated to avoid the lengthy calculations that would be required to prove that the pump casings are not penetrated and that no missiles are ejected.

C. Compressed Gas Storage

Two compressed gas storage systems were considered as possible missile sources. The first was the Nitrogen System. The components associated with the nitrogen storage system are located outside the yard area (west of the turbine laydown area) and would be prevented from reaching the yard area by the missile protection wall separating the yard area and the equipment laydown areas The second system considered was the carbon dioxide system used for fire protection. The CO_2 storage tank is located in the yard area behind the grade level emergency personnel hatch of the reactor containment building. The personnel hatch was regarded as an intervening structure between any CO_2 storage tank generated missiles and AFWS components. It should also be noted here that the main shutoff valve at the CO_2 storage tank is normally closed so the CO_2 header that runs through the yard area was classified as moderate energy piping based on the 2% operating time criteria. The CO_2 header was therefore not considered as a source of missiles.

2. Auxiliary Building

Inside the auxiliary building there are no mechanical or instrumentation components of the AFWS. The electrical components, however, were reviewed for missile protection. The location of the high energy piping in the auxiliary building was determined using the HELB analysis. The plant general equipment arrangement drawings were used to identify the rooms containing rotating equipment or compressed gas storage systems. The auxiliary building plan drawings were marked up to show the rooms containing high energy piping, rotating equipment, or compressed gas storage systems and are included in Appendix 3. It was found that there are no rooms where the two AFWS electrical trains are routed together that also contain a missile source. In general, the two electrical trains were physically separated from one another, and the electrical equipment areas were separated from mechanical equipment areas. The areas south and west of the main corridor above grade in the auxiliary building are where the electrical relay cabinets, control panels, motor control centers and computer equipment are housed, and there are no missile sources in these areas. Therefore, there is no AFWS impact from missiles generated in the auxiliary building.

VI. CONCLUSION

In reviewing all the postulated internally generated missiles from the various sources (high energy piping, rotating equipment, and compressed gas storage), it was found that there are no missiles which can simultaneously impact components of both AFWS trains. Therefore, the AFWS is considered adequately protected from the effect of internally generated missiles and no additional protection is required. The AFWS will perform its design function of supplying emergency feedwater to the steam generators to remove reactor decay heat during all design basis internally generated missile events.

VII. REFERENCES

A. USNRC Standard Review Plan 3.5.1.1, "Internally Generated Missiles (outside containment)", NUREG-0800, Rev. 2, July 1981.

B. General Design Criterion 4, "Environmental and Missile Design Bases", Appendix A to 10 CFR 50.

C. Rancho Seco Unit 1, "Updated Safety Analysis Report", Section 5.1.2.1.3.

D. Letter from John F. Stolz of NRC to R. J. Rodriguez of SMUD, titled "Status of Auxiliary Feedwater Upgrade Review", dated September 26, 1983.

E. Rancho Seco Unit 1, Calculation No. M21.30-363, titled "HELB Analysis for AFWS", dated June 14, 1983, Revision 0. by Bechtel Power Corporation, Job No. 12334.

F. Line Designation List M-853, Rev. 8.

G. Master Equipment List dated October 25, 1981.

H. Cable Raceway Tracking System.

I. General Arrangement Drawings

   1. M-300, Rev. 9

   2. M-301, Rev. 6

   3. M-302, Rev. 11

   4. M-303, Rev. 7

J. Yard Area Piping Drawings

   1. M-162, Rev. 9

   2. M-163, Rev. 13

   3. M-187, Rev. 7

   4. M-188, Rev. 8

   5. M-189, Rev. 9

6. M-190, Rev. 4

7. M-198, Rev. 9

8. M-199, Rev. 7

9. M-200, Rev. 4

10. M-204, Rev. 8

11. M-205, Rev. 5

12. M-329, Rev. 9

13. M-331, Rev. 8

K. Piping and Instrumentation Diagrams

1. M-530, Sheet 1, Rev. 3

2. M-530, Sheet 2, Rev. 4 and DCN's 7 and 10

3. M-530, Sheet 3, Rev. 6

4. M-532, Rev. 20 and DCN's 42, 46, 48, 49, and 50

5. M-533, Sheet 1, Rev. 5

6. M-533, Sheet 2, Rev. 4

7. M-533, Sheet 3, Rev. 2

8. M-537, Sheet 1, Rev. 7

9. M-537, Sheet 2, Rev. 1

10. M-537, Sheet 3, Rev. 4

L. Electrical Conduit and Tray Location Drawings

1. E-703, Sheet 1, Rev. 22

2. E-709, Sheet 2, Rev. 2

3. E-710, Sheet 2, Rev. 2

4. E-711, Sheet 2, Rev. 2

5. E-712, Sheet 2, Rev. 3

6. E-719, Rev. 15

7. E-720, Rev. 17

8. E-721, Rev. 16

9. E-735, Sheet 2, Rev. 3

10. E-739, Sheet 1, Rev. 21 and DCN 62

11. E-739, Sheet 2, Rev. 4

12. E-740, Sheet 1, Rev. 18 and DCN 45

13. E-740, Sheet 2, Rev. 3

14. E-741, Sheet 1A, Rev. 6

15. E-741, Sheet 2, Rev. 5 and DCN 12

16. E-742, Sheet 2, Rev. 3

17. E-743, Sheet 1, Rev. 30

18. E-743, Sheet 2, Rev. 6

19. E-744, Sheet 1, Rev. 23

20. E-744, Sheet 3, Rev. 5

21. E-744, Sheet 3A, Rev. 5

22. E-745, Sheet 2, Rev. 6

23. E-745, Sheet 3, Rev. 0

24. E-747, Sheet 1, Rev. 13

25. E-749, Rev. 24

26. E-751, Rev. 24

M. Electrical Elementary and Wiring Diagrams

1. E-104, Sheet 2, Rev. 8

2. E-104, Sheet 3, Rev. 10

3. E-105, Sheet 1, Rev. 5

4. E-105, Sheet 2, Rev. 6

5. E-107, Sheet 4, Rev. 0

6. E-205, Sheet 20F, Rev. 0

7. E-304, Sheet 32, Rev. 17

8. E-307, Sheet 25, Rev. 0

9. E-323, Sheet 11, Rev. 0

10. E-323, Sheet 13, Rev. 0

11. E-323, Sheet 14, Rev. 0

12. E-323, Sheet 15, Rev. 0

13. E-342, Sheet 1, Rev. 23

14. E-342, Sheet 29, Rev. 29

15. E-342, Sheet 32, Rev. 2

16. E-342, Sheet 56A, Rev. 1

N. Mechanical Equipment Vendor Drawings - Pumps

1. M5.06-1

2. M5.06-2

3. M29.03-31

4. M29.03-32

5. M29.03-34

6. M29.03-40

7. M29.03-43

8. M29.03-43A

9. M29.03-44

10. M29.03-51

O. Mechanical Equipment Vendor Drawings - Valves

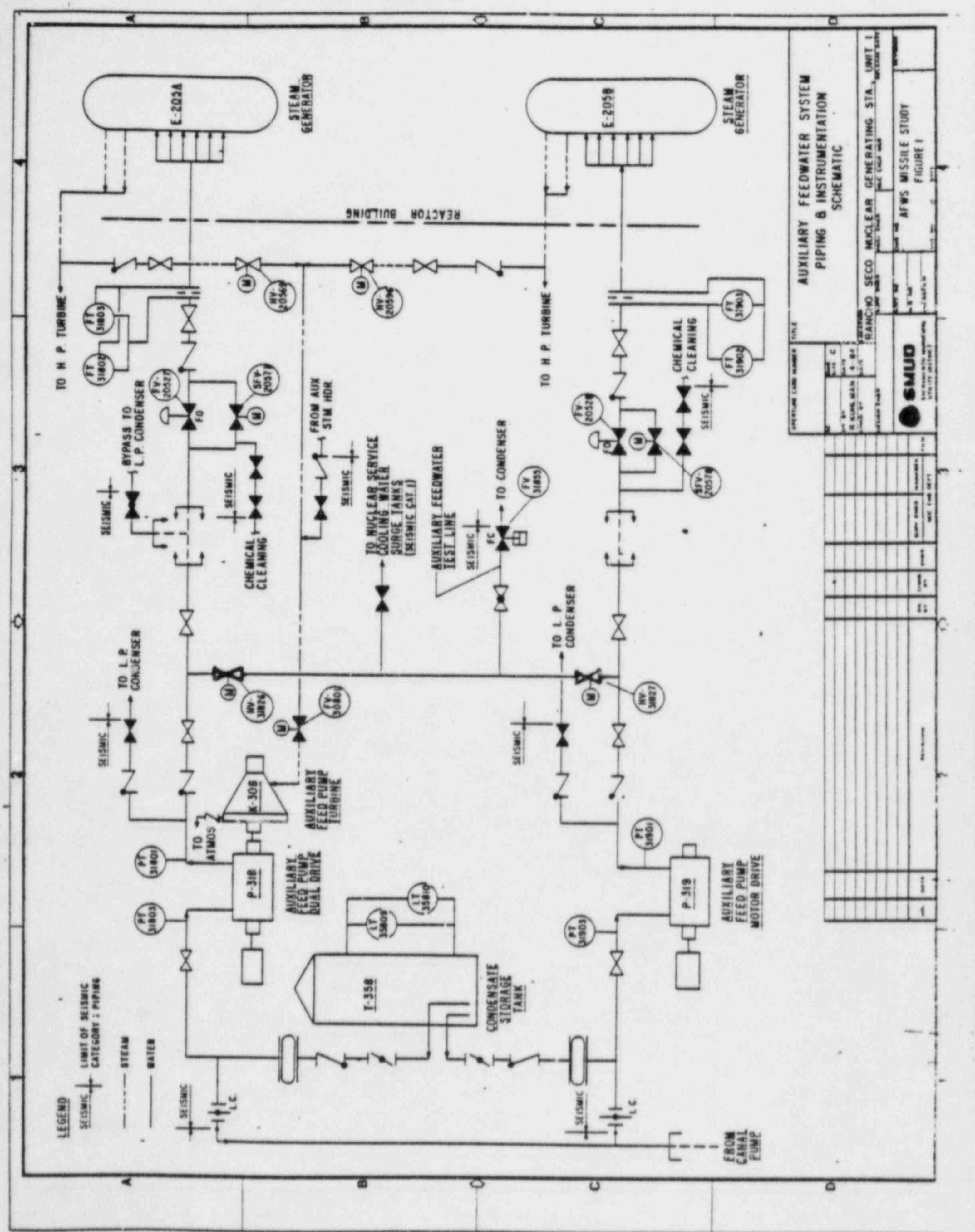
1. M19.02-338

2. M19.02-348

3. M19.05.1-1

4. M22.02-8
5. M22.02-14
6. M22.02-15
7. M22.02-16
8. M22.05-53
9. M22.05-55
10. M22.05-115
11. M22.06-23
12. M22.06-42
13. M22.06-47
14. M22.14-12
15. M22.30-28
16. N06.03-17
17. N06.03-18
18. N21.01-94

FIGURE 1



AUXILIARY FEEDWATER SYSTEM
PIPING & INSTRUMENTATION
SCHEMATIC
RANCHO SECO NUCLEAR GENERATING STA. UNIT 1
AFWS MISSILE STUDY
FIGURE I
SMUD
STEAM GENERATOR
E-205A
E-205B
REACTOR BUILDING
TO H. P. TURBINE
BYPASS TO L.P. CONDENSER
CHEMICAL CLEANING
FROM AUX STM HDR
TO NUCLEAR SERVICE COOLING WATER SURGE TANKS (SEISMIC CAT I)
AUXILIARY FEEDWATER TEST LINE
TO CONDENSER
TO L.P. CONDENSER
SEISMIC
AUXILIARY FEED PUMP TURBINE
K-308
TO ATMOS
P-318
AUXILIARY FEED PUMP DUAL DRIVE
P-319
AUXILIARY FEED PUMP MOTOR DRIVE
T-358
CONDENSATE STORAGE TANK
FROM CANAL PUMP
LEGEND
LIMIT OF SEISMIC CATEGORY I PIPING
STEAM
WATER

FIGURE 2

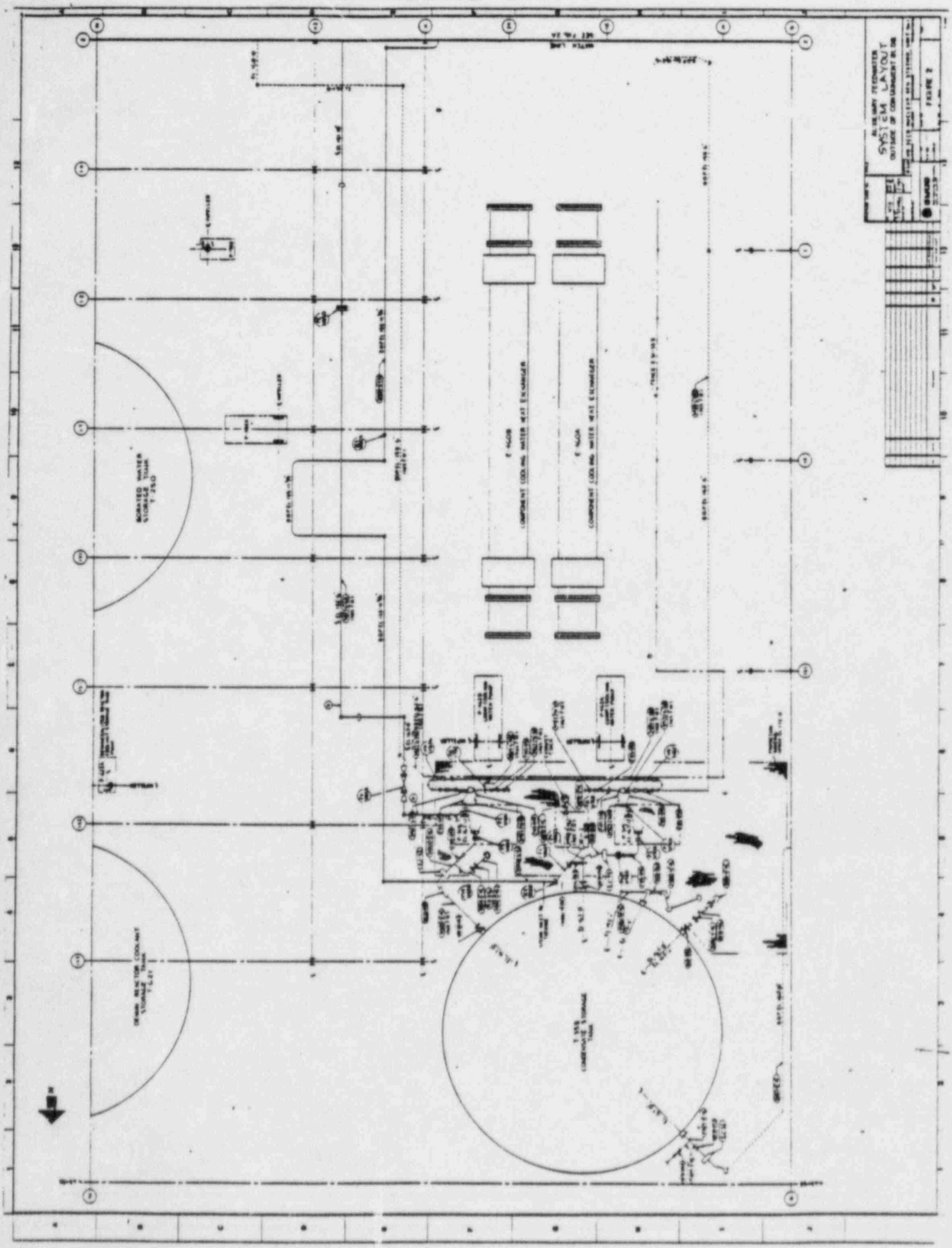


FIGURE 2A

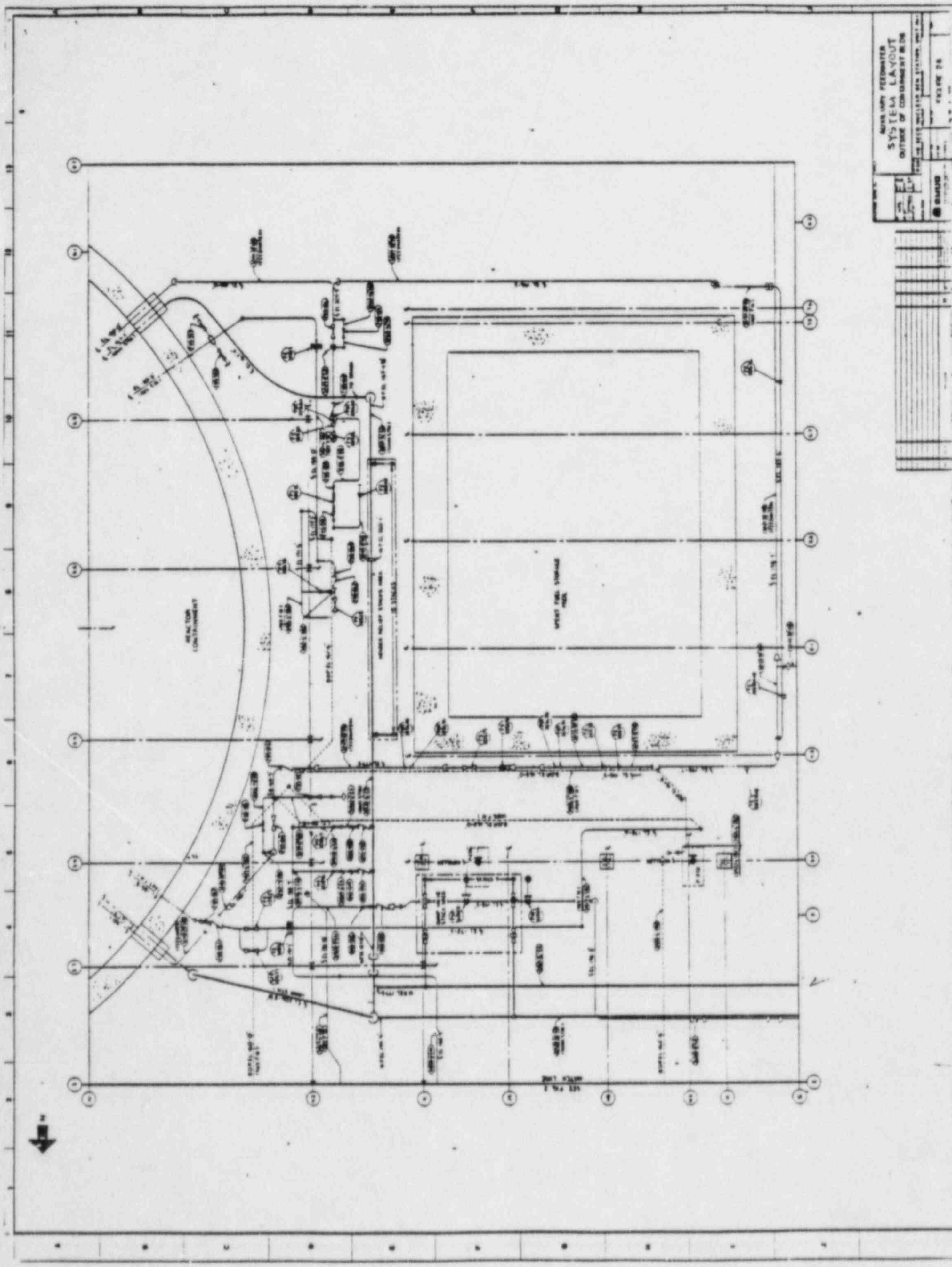
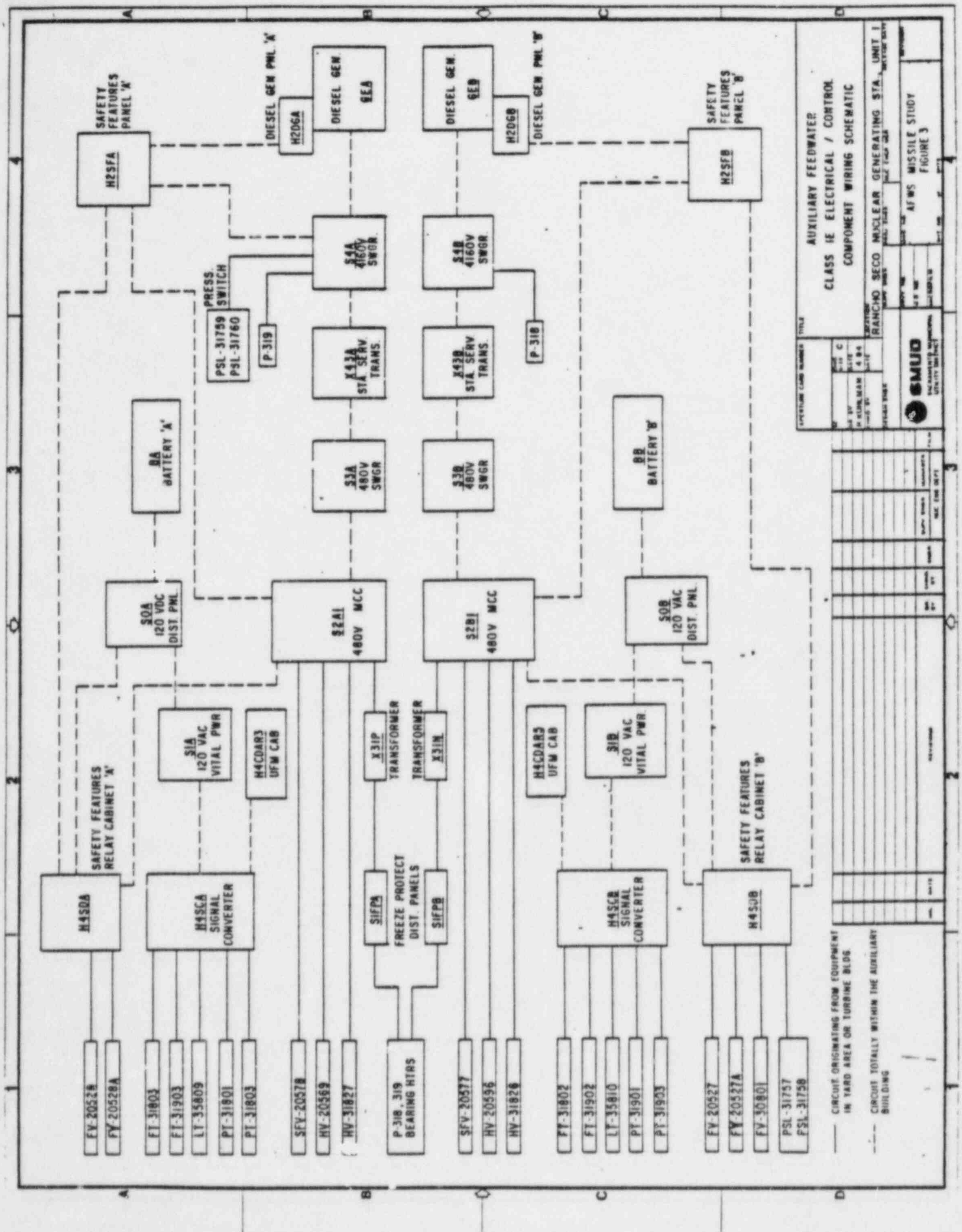


FIGURE 3


AUXILIARY FEEDWATER
CLASS IE ELECTRICAL / CONTROL
COMPONENT WIRING SCHEMATIC
RANCHO SECO NUCLEAR GENERATING STA., UNIT 1
AFWS MISSILE STUDY
FIGURE 3
SMUD
SAFETY FEATURES PANEL 'A'
H2SFA
DIESEL GEN PNL 'A'
H2DGA
DIESEL GEN.
GEA
DIESEL GEN.
GEB
DIESEL GEN PNL 'B'
H2DGB
SAFETY FEATURES PANEL 'B'
H2SFB
S4A
4160V
SWGR
S4B
4160V
SWGR
PRESS. SWITCH
PSL-31759
PSL-31760
P-319
P-318
X43A
STA. SERV.
TRANS.
X43B
STA. SERV.
TRANS.
BA
BATTERY 'A'
BB
BATTERY 'B'
S3A
480V
SWGR
S3B
480V
SWGR
S0A
120 VDC
DIST. PNL
S2A1
480V MCC
S2B1
480V MCC
S0B
120 VAC
DIST. PNL
S1A
120 VAC
VITAL PWR
H4CDAR3
UFW CAB
X31P
TRANSFORMER
X31N
TRANSFORMER
H4CDAR5
UFW CAB
S1B
120 VAC
VITAL PWR
SAFETY FEATURES RELAY CABINET 'A'
H4SDA
H4SCA
SIGNAL
CONVERTER
S1FPA
FREEZE PROTECT
DIST. PANELS
S1FPB
H4SCB
SIGNAL
CONVERTER
SAFETY FEATURES RELAY CABINET 'B'
H4SDB
FY-20528
FY-20528A
FT-31803
FT-31903
LT-35809
PT-31801
PT-31803
SFV-20578
HV-20569
HV-31827
P-318, 319
BEARING HTRS
SFV-20577
HV-20596
HV-31826
FT-31802
FT-31902
LT-35810
PT-31901
PT-31903
FV-20527
FY-20527A
FV-30801
PSL-31757
PSL-31758
CIRCUIT ORIGINATING FROM EQUIPMENT IN YARD AREA OR TURBINE BLDG
CIRCUIT TOTALLY WITHIN THE AUXILIARY BUILDING

APPENDIX 1: HIGH ENERGY PIPING AND MISSILE SOURCES

CALCULATION SHEET

APPENDIX I

CALCULATION NO. ______

SIGNATURE Charles S. Nelson DATE 4-25-84 CHECKED Louis Dale DATE 4-25-84

PROJECT Rancho Seco Unit 1 JOB NO. 12334-030

SUBJECT Aux Fw Internally Generated Missiles SHEET 1 OF 7 SHEETS

HIGH ENERGY PIPING

DESCRIPTION	LINE NO.	OPERATING P/T (psig/°F) [d]	PIPING DWG NO.	VALVES	MARK No [a]	PRESS RATING [a]	VENDOR DWG. LOG NO. [a]	BONNET MISSILE EXCLUSION [b]	BODY DRAIN PLUG AFW IMPACT [c]	OTHER POTENTIAL MISSILES	AFW IMPACT [c]	COMMENTS
Main steam from S/G A	20520-36"-EA	900/600	M-329	—	—	—				vent	none	vent valve tag MSS-531 (1")
Low point drain for line 20520-36"-EA	35067-2"-EA	900/600	M-329	HV-20570	—	2350	M22.14-12	3c	N/A			Motor operated
	20587-2"-EA	900/600	P&ID M-530	HPT-524			M1.1	3b	N/A			Bolted bonnet determined by walkdown
			"							XCV-20570	none	" " " " "
			"	HPT-525			M1.1	3b	N/A			" " " " "
			"	HPT-526			M1.1	3b	N/A			" " " " "
			"	HPT-527			M1.1	3b	N/A			" " " " "
			"	HPT-497			M1.1	3b	N/A			" " " " "
Main steam loop A atmospheric relief header	20526-36"-EA	900/600	M-329	PSV-20545	—	1050	N06.03-18	3a	N/A			
			"	PSV-20547	—	1050	"	3a	N/A			
			"	PSV-20549	—	1070	"	3a	N/A			
			"	PSV-20551	—	1070	"	3a	N/A			
			"	PSV-20553	—	1090	"	3a	N/A			
			"	PSV-20533	—	1102.5	"	3a	N/A			Pressure rating from nameplate
			"	PSV-20555	—	1090	N06.03-17	3a	N/A			
			"	PSV-20557	—	1090	"	3a	N/A			
			"	PSV-20559	—	1102.5	N06.03-18	3a	N/A			Pressure rating from nameplate
										weld cap	none	
Main steam loop A dump to HP condenser	20529-10"-EA, 20529-8"-EA	900/600	M-162, 187	—	—	—	—	—	—	—	—	No valves or other potential missiles on this line in yard area or aux bldg.

a. Valve mark number, pressure rating, and vendor drawing log number are from the Plant Equipment List dated 10-25-81, unless commented otherwise.
b. For explanation of bonnet missile exclusion categories, see App. 5 "Internally Generated Missiles Design Criteria", Section II.C
c. AFW impact determined by visual inspection during system walkdown. If valve body does not have a drain plug, AFW impact is "N/A." If missile does not impact both AFW trains, AFW impact is "none."
d. Operating P/T are from the Line Designation List M-253 Rev 8.

CALCULATION SHEET

APPENDIX I

CALCULATION NO. ______

SIGNATURE Christie ? ______ DATE 4-13-84 CHECKED L. ? DATE 4-26-84

PROJECT Rancho Seco Unit 1 JOB NO 12334-030

SUBJECT Aux FW Internally Generated Missiles SHEET 2 OF 7 SHEETS

HIGH ENERGY PIPING

DESCRIPTION	LINE NO.	OPERATING P/T (psig/°F) [d]	PIPING DWG NO	VALVES	MARK No [a]	PRESS RATING [a]	VENDOR DWG. LOG NO [a]	BONNET MISSILE EXCLUSION [b]	BODY DRAIN PLUG AFW IMPACT [c]	OTHER POTENTIAL MISSILES	AFW IMPACT [c]	COMMENTS
Main steam loop A	20531-8"-EA	900/600	M-329	MSS-017	38B	600	M22.02-16	3b	none		—	
dump to atmosphere			"	PV-20571A	—	600	N21.01-94	3c	none			Air-operated
	20540-8"-EA	900/600	M-329	MSS-019	38B	600	M22.02-16	3b	none			
										Pressure tap	none	(MSS-565) Tubing attached
			"	PV-20571C	—	600	N21.01-94	3c	none			Air operated
	20556-8"-EA	900/600	M-329	MSS-021	38B	600	M22.02-16	3b	none			
										Capped weldolet	none	Location determined by walkdown
			M-329	PV-20571B	—	600	N21.01-94	3c	none			Air-operated
Main steam loop A	30802-6"-EA	900/600	M-187	MSS-052	254	900	M22.06-23	3c	none			Pressure rating from nameplate
to AFW pump turbine			"	MSS-050	38B	600	M22.02-15	3b	none			
			"	HV-20569	185	600	M22.06-41	3b,c	none			Motor-operated
	30800-6"-EA1	900/600								4" blinded tee	none	Bolted flange, location by walkdown
										4" blinded tee	none	" " , location by walkdown
			M-204							TI-30804	none	Inside missile shield
			"							PI-30805	none	Inside missile shield
			"	FV-30801			N100	3c	none			Motor-operated
Low point drain for	30880-1½"-EA1	900/600	P&ID M-530-3	MSS-544				3b	N/A			Bolted bonnet determined by walkdown
line 30800-6"-EA1										ICV-30802	none	Steam trap w/ bolted assy, location by walkdown
				MSS-545				3b	N/A			Bolted bonnet determined by walkdown
				MSS-546				3b	N/A			" " " "

a. Valve mark number, pressure rating, and vendor drawing log number are from the Plant Equipment List dated 10-25-81, unless commented otherwise.

b. For explanation of bonnet missile exclusion categories, see App. 5 "Internally Generated Missiles Design Criteria," Section II.C

c. AFW impact determined by visual inspection during system walkdown. If valve body does not have a drain plug, AFW impact is "N/A." If missile does not impact both AFW trains, AFW impact is "none."

d. Operating P/T are from the Line Designation List M-253 Rev 8.

CALCULATION SHEET

APPENDIX I

CALCULATION NO ______

SIGNATURE [signature] DATE 4-23-84 CHECKED [signature] DATE 4-26-84

PROJECT Rancho Seco Unit 1 JOB NO 13334-030

SUBJECT Aux Fw Internally Generated Missiles SHEET 3 OF 7 SHEETS

HIGH ENERGY PIPING

DESCRIPTION	LINE NO.	OPERATING P/T (psig/°F) [d]	PIPING DWG NO	VALVES	MARK No [a]	PRESS RATING [a]	VENDOR DWG. LOG NO [a]	BONNET MISSILE EXCLUSION [b]	BODY DRAIN PLUG AFW IMPACT [c]	OTHER POTENTIAL MISSILES	AFW IMPACT [c]	COMMENTS
Main steam from	20521-36"-EA	900/600	M-329	PSV-20544	—	1050	N06.03-18	3b	N/A		—	
SG B			"	PSV-20546	—	1050	"	3b	N/A		—	
			"	PSV-20548	—	1070	"	3b	N/A			Pressure rating from nameplate
			"	PSV-20534	—	1102.5	"	3b	N/A			
			"	PSV-20550	—	1070	"	3b	N/A			
			"	PSV-20552	—	1090	"	3b	N/A			
			"	PSV-20554	—	1090	N06.03-17	3b	N/A			
			"	PSV-20556	—	1070	"	3b	N/A			
			"	PSV-20558	—	1102.5	N06.03-18	3b	N/A			
Low point drain for	35068-2"-EA	900/600	M-329	HV-20571	—	2350	M22.14-12	3c	N/A			Motor operated
line 20421-36"-EA	20576-2"-EA1	900/600	P&ID M-530	HPT-529				3b	N/A			Bolted bonnet determined by walkdown
			"							FCV-20571	none	steam trap w/ bolted assy, location by walkdown
			"	HPT-530				3b	N/A			Bolted bonnet determined by walkdown
			"	HPT-528				3b	N/A			" " " " "
			"	HPT-531				3b	N/A			" " " " "
			"	HPT-495	344	600		3b	N/A			
Main steam loop B	20530-10"EA	900/600	M-162	—	—	—	—	—	—	—	—	No valves or other potential missiles
dump to HP condenser												on this line in yard area or aux bldg
Main steam loop B	30804-6"-EA	900/600	M-187	MSS-051	234	900	[illegible]2.06-23	3a	none			Pressure rating from nameplate
to AFW pump turbine			"	MSS-049	388	600	M22.02-15	3b	none			
	20598-6"-EA1	900/600	M-187	HV-20596	385	600	M22.06-47	3b,c	none			Motor operated

a. Valve mark number, pressure rating, and vendor drawing log number are from the Plant Equipment List dated 10-25-81, unless commented otherwise.

b. For explanation of bonnet missile exclusion categories, see App. 5 "Internally Generated Missiles Design Criteria", Section II.C.

c. AFW impact determined by visual inspection during system walkdown. If valve body does not have a drain plug, AFW impact is "N/A". If missile does not impact both AFW trains, AFW impact is "none".

d. Operating P/T are from the Line Designation List M-853 Rev 8.

CALCULATION SHEET

APPENDIX I

CALCULATION NO. ____________

SIGNATURE [illegible] DATE 4-13-84 CHECKED [illegible] DATE 4/26/84

PROJECT [illegible] Unit [illegible] JOB NO. 17334-030

SUBJECT [illegible] Internally Generated Missiles SHEET 4 OF 7 SHEETS

HIGH ENERGY PIPING

DESCRIPTION	LINE NO.	OPERATING P/T [d] ([illegible])	PIPING DWG NO	VALVES	MARK No [a]	PRESS RATING [a]	VENDOR DWG LOG NO [a]	BONNET MISSILE EXCLUSION [b]	BODY DRAIN PLUG AFW IMPACT [c]	OTHER POTENTIAL MISSILES	AFW IMPACT [c]	COMMENTS
Main steam loop B	20532-8"-EA	900/600	M-329	MSS-018	38B	600	M22.02-16	3b	none		—	
dump to atmosphere			"	PV-20562A	—	600	N21.01-94	3c	none		—	Air-operated
	20558-8"EA	900/600	M-329 P&ID	MSS-020	38B	600	M22.02-16	3b	none			
										Pressure tap	none	Tubing attached (MSS-564)
			M-329	PV-20562C	—	600	N21.01-94	3c	none			Air-operated
	20519-8"EA	900/600	M-329	MSS-022	38B	600	M22.02-16	3b	none			
			"	PV-20562B	—	600	N21.01-94	3c	none			Air-operated
Auxiliary Boiler to	30809-2"-EA1	900/600	M-187	MSS-006	03B	600	M22.02-A58	3b	none			
AFW Pump Turbine			"	MSS-005	242	600	M22.02-A33A	3b	none			

a. Valve mark number, pressure rating, and vendor drawing log number are from the Plant Equipment List dated 10-25-81, unless commented otherwise.
b. For explanation of bonnet missile exclusion categories, see App. 5 "Internally Generated Missiles Design Criteria," Section II.C.
c. AFW impact determined by visual inspection during system walkdown. If valve body does not have a drain plug, AFW impact is "N/A." If missile does not impact both AFW trains, AFW impact is "none."
d. Operating P/T are from the Line Designation List M-853 Rev. 8.

CALCULATION SHEET

APPENDIX I

CALCULATION NO ________

SIGNATURE [illegible] DATE 4-23-84 CHECKED [illegible] DATE 4-26-84

PROJECT Rancho Seco Unit 1 JOB NO 17334-030

SUBJECT Aux FW Internally Generated Missiles SHEET 5 OF 7 SHEETS

HIGH ENERGY PIPING

DESCRIPTION	LINE NO.	OPERATING P/T (psig/°F) [d]	PIPING DWG NO	VALVES	MARK No [a]	PRESS RATING [a]	VENDOR DWG LOG NO [a]	BONNET MISSILE EXCLUSION [b]	BODY DRAIN PLUG AFW IMPACT [c]	OTHER POTENTIAL MISSILES	AFW IMPACT [c]	COMMENTS
Feedwater to SG A	32140-20"-DB1	1100/464	—	—	—	—	—	—	—	TI-32147	none	Location determined by walkdown. Indicator is outside reactor building; instrument tap is inside penetration.
										Capped weldolet	FV-20527	Location determined by walkdown, missile impacts Train A only, does not affect Train B.
	32120-20"-DB	1150/464	M-331,188	FWS-015	22B65	900	M22.05-53	3a	none			
										Capped weldolet	none	Location determined by walkdown
										Capped weldolet	none	
			M-331,188	FWS-013	231	900	M22.05-55	3a	none	Hinge pin cover	none	
										Capped weldolet	none	Location determined by walkdown
			M-331							PDT-20567B	none	Attached tubing, tag number verified in field
			M-331							PDT-20567A	none	" " " " " "
										capped weldolet	none	Location determined by walkdown
			M-331,187	FV-20529	225	900	M22.05-115	3a,c	none			Motor operated
										Capped weldolet	none	Location determined by walkdown
										Capped weldolet	none	
			M-331,187	FV-20525	—	900	M19.02-34B	3c	none			Air operated
										Capped weldolet	none	Location determined by walkdown
			M-331,187	FWS-132	22	900	—	3a	none	PDT-205676	none	Attached tubing (PDT)
										Capped weldolet	none	Location determined by walkdown
										PDT-20567A	none	Attached tubing
			M-331,187									" "
			" "							FE-20535	none	
												No values or other potential missiles on this line
FWS Drs Bypass	32147-1½"-DB1	1150/464	M-331,188	—	—	—	—	—	—	—	—	
Loop A	32146-1½"-DB	1150/464	"	FWS-108	135	2240	M22.02-8	3b	N/A			
FW Start Up Bypass	32139-12"-DB	355/464	M-331,188	FWS-021								
Loop A												

a. Valve mark number, pressure rating, and vendor drawing log number are from the Plant Equipment List dated 10-25-81, unless commented otherwise.

b. For explanation of bonnet missile exclusion categories, see App. 5 "Internally Generated Missiles Design Criteria" Section II.C.

c. AFW impact determined by visual inspection during system walkdown. If valve body does not have a drain plug, AFW impact is "N/A." If missile does not impact both AFW trains, AFW impact is "none."

d. Operating P/T are from the Line Designation List M-253 Rev. 8.

CALCULATION SHEET

APPENDIX I

CALCULATION NO. ______

SIGNATURE [illegible] DATE 9-13-89 CHECKED L Dechant DATE 9-26-89

PROJECT Rancho Seco Unit 1 JOB NO. 12339-030

SUBJECT [illegible] Internally Generated Missiles SHEET 6 OF 7 SHEETS

HIGH ENERGY PIPING

DESCRIPTION	LINE NO.	OPERATING P/T (psig/°F) [d]	PIPING DWG NO	VALVES	MARK No [a]	PRESS RATING [a]	VENDOR DWG LOG NO [a]	BONNET MISSILE EXCLUSION [b]	BODY DRAIN PLUG AFW IMPACT [c]	OTHER POTENTIAL MISSILES	AFW IMPACT [c]	COMMENTS
FWS-013,015 Bypass Loop A	32132-8"-DB	1150/464	M-331	FWS-019	22	900	M22.02-14	3a	none		—	
				FV-20575	—	—	M19.02-33B	3c	none		—	Air-operated
				FWS-017	22B	900	M22.02-A58	3a	none			
	32125-8"-DB	1160/464	M-331							FE-20539	none	Attached tubing
Feedwater to SG D	32141-20"-DB1	1100/464	M-331,167							TI-32148	none	Location determined by walkdown. Indicator is outside reactor bldg, instrument tap is inside penetration
										Capped weldolet	none	Location determined by walkdown
										Capped weldolet	none	" " " "
			M-331,167	FWS-016	22BGS	900	M22.05-53	3a	none			
	32124-20"-DB	1150/464	" "	FWS-014	231	900	M22.05-55	3a	none			
										Capped weldolet	none	Location determined by walkdown
			M-331							PDT-20368B	none	Attached tubing
			"							PDT-20368A	none	Attached tubing
										Capped weldolet	none	Location determined by walkdown
			M-331,187	FV-20530	225	900	M22.05-115	3a,c	none			Motor operated
			" "	FV-20526	—	900	M19.02-34B	3c	none			Air operated
										Capped weldolet	none	Location determined by walkdown
			M-331,187	FWS-133	22	900		3a	none			
			M-331							PDT-20568A	none	Attached tubing
			"							PDT-20568B	none	Attached tubing
										Capped weldolet	none	Location determined by walkdown

a. Valve mark number, pressure rating, and vendor drawing log number are from the Plant Equipment List dated 10-25-81, unless commented otherwise.
b. For explanation of bonnet missile exclusion categories, see App. 5 "Internally Generated Missiles Design Criteria" Section II.C.
c. AFW impact determined by visual inspection during system walkdown. If valve body does not have a drain plug, AFW impact is "N/A". If missile does not impact both AFW trains, AFW impact is "none".
d. Operating P/T are from the Line Designation List M-[illegible] Rev 8.

CALCULATION SHEET

APPENDIX I

CALCULATION NO. ______

SIGNATURE [illegible signature] DATE 9-23-84 CHECKED L. [illegible] DATE 9-26-84

PROJECT Rancho Seco Unit 1 JOB NO 12334-030

SUBJECT Aux Fw Internally Generated Missiles SHEET 7 OF 7 SHEETS

HIGH ENERGY PIPING

DESCRIPTION	LINE NO.	OPERATING P/T (psig/°F) [d]	PIPING DWG NO	VALVES	MARK No [a]	PRESS RATING [a]	VENDOR DWG. LOG NO [a]	BONNET MISSILE EXCLUSION [b]	BODY DRAIN PLUG AFW IMPACT [c]	OTHER POTENTIAL MISSILES	AFW IMPACT [c]	COMMENTS
FWS 016 Bypass	32169-1½"-DB1	1150/464	M-331	—	—	—	—	—	—	—	—	No valves or other potential missiles on this line.
Loop B	32168-1½"-DB1	1150/464	"	FWS-109	135	2240	M22.02-8	3b	N/A			
FW Start Up Bypass	32136-12"-DB	1150/464	M-331 (PRIO M-532, DCN 49)	FWS-022	22 865	900	M22.06-42	3a	none			
Loop B												
FWS 019,016 Bypass	32126-8"-DB	1150/464	M-331,187	FWS-020	228	900	M22.02-14	3a	none			
Loop B				FV-20576	—	—	M19.02-338	3c	none			Air operated
				FWS-018	228	900	M22.02-A38	3a	none			
										FE-20540	none	Attached tubing
Auxiliary steam header	36020-12"-GB	250/406	M-187							vent	none	1" vent valve ASC-621, location by walkdown
	36020-6"-GB	250/406	M-187,163,							6" blind flange	none	Location by walkdown
	36020-2½"-GB	250/406	187,190							drain	none	drain valve ASC-624
Aux steam header	36098-6"-GB	250/406	M-187	PSV-36012B	—	280	M19.05.1-1	3b	N/A			Bolted bonnet
pressure relief	36088-6"-GB	250/406	"	PSV-36012A	—	275	M19.05.1-1	3b	N/A			" "
Aux steam to Boric	61250-6"-GB	250/406	M-163	HV-36050	53	300	M22.30-28	3c	N/A			Motor operated; valve info from M-B72 DCN 170
Acid Concentrator	61255-4"-GB	250/406	"	HV-36051	53	300	M22.30-28	3c	N/A			Valve is motor operated; valve info from M-B72, DCN 170; line oper P/T from M-B53 DCN 135 A
Aux steam to AFW	30801-3"-GB	250/406	M-187	—	—	—	—	—	—	—	—	No values or other potential missiles on this line.
Pump Turbine												

a. Valve mark number, pressure rating, and vendor drawing log number are from the Plant Equipment List dated 10-25-81, unless commented otherwise.

b. For explanation of bonnet missile exclusion categories, see App 5 "Internally Generated Missiles Design Criteria," Section II.C.

c. AFW impact determined by visual inspection during system walkdown. If valve body does not have a drain plug, AFW impact is "N/A." If missile does not impact both AFW trains, AFW impact is "none."

d. Operating P/T are from the Line Designation List, M-??? Rev 8.

APPENDIX 2: MODERATE ENERGY PIPING

LAO 0012 8-73

CALCULATION SHEET

APPENDIX 2

CALC. NO. ____

SIGNATURE Christine J. Kelso DATE 4-26-84 CHECKED L. Destel DATE 4-26-84

PROJECT Rancho Seco Unit 1 JOB NO. 12324.030

SUBJECT Aux FW Internally Generated Missiles SHEET 1 OF 9 SHEETS

MODERATE ENERGY PIPING

Line No.	Oper. P/T (psig / °F)	Description	Piping Dwg. No.
* 20516-8"-HC	50/280	SG A Stm Dump to Atmos	M-329
* 20519-8"-HC	50/280	SG B Stm Dump to Atmos	M-329
* 20533-10"-HC	40/270	SG A Relief to Atmos	M-329
* 20534-10"-HC	40/270	SG A Relief to Atmos	M-329
* 20535-10"-HC	40/270	SG A Relief to Atmos	M-329
* 20536-10"-HC	40/270	SG A Relief to Atmos	M-329
* 20537-10"-HC	40/270	SG A Relief to Atmos	M-329
* 20538-8"-HC	40/270	SG A Relief to Atmos	M-329
* 20539-8"-HC	40/270	SG A Relief to Atmos	M-329
* 20540-10"-HC	40/270	SG A Relief to Atmos	M-329
* 20541-10"-HC	40/270	SG B Relief to Atmos	M-329
* 20542-10"-HC	40/270	SG B Relief to Atmos	M-329
* 20543-10"-HC	40/270	SG B Relief to Atmos	M-329
* 20544-10"-HC	40/270	SG B Relief to Atmos	M-329
* 20545-10"-HC	40/270	SG B Relief to Atmos	M-329
* 20546-8"-HC	40/270	SG B Relief to Atmos	M-329
* 20547-8"-HC	40/270	SG B Relief to Atmos	M-329
* 20548-10"-HC	40/270	SG B Relief to Atmos	M-329
* 20551-10"-HC	40/270	SG A Relief to Atmos	M-329
* 20557-10"-HC	40/270	SG B Relief to Atmos	M-329
* 20571-8"-HC	50/280	SG B Stm Dump to Atmos	M-329
* 20575-8"-HC	50/280	SG A Stm Dump to Atmos	M-329
* 20577-8"-HC	50/280	SG B Stm Dump to Atmos	M-329
* 20597-8"-HC	50/280	SG A Stm Dump to Atmos	M-329
25020-16"-HD	60/90	BWST to DHR Pp A	M-198,163
25020-2-1/2"-HD	60/90	BWST to DHR Pp A	M-198
25021-2-1/2"-HD	60/90	BWST to DHR Pp B	M-198
25021-16"-HD	6[illegible]/90	BWST to DHR Pp B	M-198
25022-16"-HD	60/100	SF Stg Pool to DHR Pp	M-188,187
25024-3"-HD	60/90	BWST to SF Coolant Demin Pp	M-188
25080-3"-HD	Atm/90	BWST O'flow to RC Drn Tk	M-198, 163
25081-3"-HD	Atm/90	Hose Conn on BWST	M-188

BECHTEL

CALCULATION SHEET

CALC. NO.

SIGNATURE Christine A. Wilson DATE 4-26-84 CHECKED L Dentect DATE 4-26-84

PROJECT Rancho Seco Unit 1 JOB NO. 12334-030

SUBJECT Aux FW Internally Generated Missiles SHEET 2 OF 9 SHEETS

MODERATE ENERGY PIPING

Line No.	Oper. P/T (psig / °F)	Description	Piping Dwg. No.
25083-4"-HD	60/90	BWST Port Pp Conn	M-198
25101-3"-HD	60/90	BWST to LP Inj Hdr Warming Pp	M-198
25120-2-1/2"-HD	35/90	LP Inj Hdr Warming Pp to BWST	M-198
25121-2-1/2"-HD	60/90	LP Inj Hdr Warming Pp to BWST	M-198
25123-2-1/2"-HD	60/90	LP Inj Hdr Warming Pp to BWST	M-163,190,198,162
* 26022-8"-GD	450/200	DHR Cooler to SF Stg Pool	M-162,163
26033-8"-HD	100/120	DHR Cooler to BWST	M-163,188,198
26060-18"-HE	80/145	DHR Cooler to NSCW HEX	M-163,198,188
26061-18"-HE	80/145	DHR Cooler to NSCW HEX	M-198, 201
27000-8"-HD	30/120	SF Coolant Pp to SF Cooler	M-188
27060-8"-HE	90/113	SF Cooler to CCW Pp	M-188
27020-8"-HD	30/120	SF Cooler to SF Stg Pool	M-188,163,162
27050-8"-HE	90/95	CCW HEX to SF Cooler	M-188
27200-8"-HD	20/120	SF Stg Pool to SF Coolant Pp	M-188
27400-3"-HD	30/120	SF Cooler to SF Coolant Demin Pp	M-188
27420-3"-HD	110/120	SF Clnt Demin Pp to SF Clnt Fltr	M-188,163,162,190
27620-3"-HD	150/120	SF Cool Demin to SF Stg Pool	M-190,188
27622-3"-HD	100/120	SF Clnt Demin to BWST	M-190,163,187,198
27623-3"-HD	100/120	SF Clnt Demin to SF Stg Pool	M-190,163,162
27626-3"-HD	100/120	SF Clnt Demin to BWST	M-190,163,187
27700-3"-HD	25/120	SF Stg Pool to Skimmer Pp	M-163,188
27701-3"-HD	25/120	SF Stg Pool to Skimmer Pp	M-188
27822-2-1/2"-HD	100/120	SF Pool Fltr to SF Stg Pool	M-188
* 30890-8"-HC	2/435	AFW Pp Turbine Exhaust to Atmos	M-204
* 30890-12"-HC	2/435	AFW Pp Turbine Exhaust to Atmos	M-204
* 30926-4"-EA1	900/200	SG Drn Booster Pp to Demin Area	M-162,163
* 30934-1-1/2"-DB2	25/Amb	SG Drn Booster Pp to SG A and B	M-163
31800-8"-HC	15/90	CST to AFW Pp P318	M-204
31801-8"-HC	20/Amb	Folsom S Canal Pp to AFW Pp P318	M-204
31802-8"-HE2	40/Amb	Folsom S Canal Pp to AFW Pp P318	M-204
31803-8"-HC	20/Amb	Folsom S Canal Pp to AFW Pp P319	M-204
* 31820-6"-DB2	1120/90	AFW Pp 318 to SG A	M-188

BECHTEL

CALCULATION SHEET

LAO 0013 0-73

APPENDIX 2

CALC. NO. ____

SIGNATURE Christie F. Calver DATE 4-26-84 CHECKED L. Dutach DATE 4-26-84

PROJECT Rancho Seco Unit 1 JOB NO. 12334-030

SUBJECT Aux FW Internally Generated Missiles SHEET 9 OF 9 SHEETS

MODERATE ENERGY PIPING

Line No.	Oper. P/T	Description	Piping Dwg. No.
* 31821-6"-DB	1120/90	AFW Pp 319 to SG B	M-187,162,163
31822-6"-DB	0/90	AFW Pp 318 Recirc to LP Cond	M-188
31822-2-1/2"-DB	0/90	AFW Pp 318 Recirc to LP Cond	M-204,205
* 31823-6"-DB	1150/90	AFW Pp 319 to SG A	M-204,189,188
* 31824-6"-DB2	1150/90	AFW Pp 318 Exer Line to LP Cond	M-204,198,188
* 31825-2-1/2"-DB	1150/90	AFW Pp 319 Recirc to LP Cond	M-204
* 31826-6"-DB	1150/90	AFW Pp 319 to SG B	M-187,188
* 31827-6"-GB	1150/90[1]	AFW Pp 318 Exer Line to LP Cond	M-198
* 31891-6"-DB2	1150/90	AFW Pp Crosstie	M-204
31900-8"-HC	15/90	CST to AFW Pp P319	M-204
* 31920-6"-DB	1150/90	AFW Pp 319 to SG B	M-204,205,189
31922-2-1/2"-DB2	0/90	AFW Pp 319 Recirc to LP Cond	M-204,205,189,188
31922-6"-DB	0/90	AFW Pp 319 Recirc to LP Cond	M-188
* 32291-12"-HC	127/365	2nd Pt Htr PSV to B/D Tk Relief	M-198,188,187
** 32880-20"-HC	Atm/210	B/D Tk Vent	M-198
34285-4"-HC	Atm/130	Air Ejector to Vent	M-162
** 34420-6"-HC	1/208	Gland Stm Exhauster to Atm	M-162
34500-6"-HD	30/100	Misc Wtr HU Tk to Sluice Pumps	M-188
34501-6"-HC	30/100	Misc Wtr HU Tk to Sluice Pumps	M-188
* 35620-16"-HC	15/210	Hogging Ejector to Atm	M-162
35823-12"-HC	1/90	CST to LP Condenser	M-204
35824-3"-HC	1/90	CST to LP Condenser	M-204
35824-12"-HC	1/90	CST to LP Condenser	M-204
35880-12"-HC	Atm/Amb	CST Overflow	M-204
35881-12"-HC	1/Amb	CST Overflow	M-204
35882-4"-HC	25/90	CST Overflow	M-204
35891-4"-HD	120/Amb	MU Demin to CST	M-190,204,198,188
36089-6"-HC	Atm/Amb	Aux Stm Hdr Relief to Atm	M-187
36099-6"-HC	Atm/Amb	Aux Stm Hdr Relief to Atm	M-187
40226-3"-HE2	45/125	Circ Wtr Pp CW to CCW Pp A&B	M-204
42521-24"-HE2	40/Amb	Plant CW to Gen H_2 Cooler	M-188

1. From M-853, DCN 139B

LAO 0012 6-73

CALCULATION SHEET

APPENDIX 2

CALC. NO.

SIGNATURE Christine A. Wilson DATE 4-26-84 CHECKED L. Deutsch DATE 4-26-84

PROJECT Rancho Seco Unit 1 JOB NO. 12334-030

SUBJECT Aux FW Internally Generated Missiles SHEET 4 OF 9 SHEETS

MODERATE ENERGY PIPING

Line No.	Oper. P/T (psig/°F)	Description	Piping Dwg. No.
45750-10"-HE	90/95	CCW HEX to Exciter Air Cooler	M-188
45760-10"-HE	85/125	Exciter Air Cooler to CCW Pp	M-188
46047-3"-HE	90/95	CCW HEX to Circ Wtr Pps	M-204
46050-20"-HE	95/125	CCW Pp A to CCW HEX	M-204
46051-20"-HE	95/125	CCW Pp B to CCW HEX	M-204
46053-30"-HE	40/Amb	CW Pp to CCW HEX A	M-204
46054-30"-HE	40/Amb	CW Pp to CCW HEX B	M-204
46060-18"-HE	90/99	CCW HEX to RCP	M-163,188
46060-20"-HE	90/95	CCW HEX A to RCP A	M-204,198,188
46061-20"-HE	90/95	CCW HEX B to RCP A	M-204
46062-30"-HE2	40/105	CCW HEX A to Circ Water Intake	M-204
46063-30"-HE2	40/105	CCW HEX B to Circ Water Intake	M-204
46068-2"-HE2	40/Amb	CCW HEX A Vent (tubeside)	M-204
46069-2"-HE2	40/Amb	CCW HEX B Vent (tubeside)	M-204
46076-2"-HE	90/95	CCW HEX B Vent (shellside)	M-204
46077-2"-HE	90/95	CCW HEX A Vent (shellside)	M-204
46200-18"-HE	85/125	RCP to CCW Pp	M-163,188
46200-20"-HE	85/125	RCP's to CCW Pp A Suction	M-198,204,188
46201-24"-HE	85/125	RCP's to CCW Pp B Suction	M-204
46280-6"-HE	90/125	CCW Pp A Miniflow	M-204,198
46420-6"-HE	30/Amb	CCW Surge Tk to CCW Pumps	M-163,162
46701-1-1/2"-HED	125/130	CRDM CW from Sample Station	M-163
46750-4"-HE	90/95	CCW HEX to CRD CW HEX	M-163
46755-3"-HD	125/130	CRD Units to CRD HEX	M-163
46760-4"-HE	85/124	CRD CW HEX to CCW Pp	M-163
46762-3"-HD	70/100	CRD HEX A to CRD Pp	M-163
46902-1-1/2"-HD1	70/100	CRD CW Srg Pipe to CRD CW Pp	M-163,162
46981-10"-HD	70/100	CRD CW Pp Surge Pipe	M-162,163
47081-4"-HE2	120/Amb	Fire Loop Line to W Spray Pd	M-198
47090-4"-HE2	120/Amb	Svc Wtr Fltrs to E Spray Pd	M-198,200
47091-4"-HE2	120/Amb	Svc Wtr Fltrs to W Spray Pond	M-198
47096-4"-HE2	130/Amb	Fire Loop Line to W Spray Pd	M-198

LAO 0613 6-73

CALCULATION SHEET

APPENDIX 2
CALC. NO.

SIGNATURE Christine F. Kelson DATE 4-26-84 CHECKED L. Deutsch DATE 4-26-84
PROJECT Rancho Seco Unit 1 JOB NO. 12334-030
SUBJECT Aux FW Internally Generated Missiles SHEET 5 OF 9 SHEETS

MODERATE ENERGY PIPING

Line No.	Oper. P/T (psig / °F)	Description	Piping Dwg. No.
47097-4"-HE2	130/Amb	Fire Loop Line to E Spray Pd	M-198,200
47098-4"-HE2	35/Amb	Fire Loop Line to NS Spray Pd	M-198,200
47350-10"-HE2	35/87	NSRW Pp A to DG HEX A	M-163,188,198,204
47351-10"-HE2	35/87	NSRW Pp to DG CW HEX	M-198,200,163
47360-10"-HE2	35/106	DG CW HEX to W Spray Pd	M-204,198,163,188
47361-10"-HE2	35/108	DG HEX B to E Spray Pd	M-163
47451-3"-HE2	35/87	NSRW Pp A to Chem Add Tk	M-198
47452-3"-HE2	35/87	NSRW to DG CW HEX	M-200,198
47461-3"-HE2	35/87	NSRW from Chem Add Tk to DG HEX	M-198
47462-3"-HE2	35/87	NSRW Pp to Chem Add Tk	M-198
48054-24"-HE2	35/87	NSRW Pp A to NSCW HEX	M-198
48055-24"-HE2	35/87	NSRW Pp to NSCW HEX	M-198
48060-18"-HE	60/95	NSCW HEX to NSCW Pp	M-198
48061-18"-HE	60/95	CCW HEX to RCP	M-198
48062-24"-HE2	35/109	NSCW HEX to W NS Wtr Spray Pond	M-198
48063-24"-HE2	35/105	CCW HEX to Circ Wtr Intake	M-198
48222-18"-HE	110/95	NSCW Pp to DHR Cooler	M-198,163
48223-18"-HE	110/95	NSCW Pp Disch to DHR Cooler	M-198
48400-6"-HE	110/145	NSCW Srg Tk PSV to NSCW Mx xchgr	M-163,162
48480-4"-HE	20/145	NSCW Surge Tk PSV to SRT	M-163,162
48486-1-1/2"-HE	Atm/Amb	CRD CW Srg Pipe to O'flw to SRT	M-162,163
48490-2"-HE	50/Amb	N_2 Supply to NSCW Surge Tank	M-162,163
48495-3"-HD	120/Amb	Mixed Bed Demin to NSCW Surge Tk	M-187,162,163
48750-4"-HE2	55/87	NSRW Pp to DHR Pp BOC	M-163
48760-4"-HE2	55/100	DHR Pp BOC to W Spray Pd	M-163
53520-12"-HC	1/120	RB Purge Air Supply Fan to RB	M-162
53800-2"-HD1	Atm/Amb	H_2 Purge Exhaust Blower to Vent	M-162
53802-2"-HD1	Atm/Amb	H_2 Purge Exhaust Blower to Vent	M-162
61281-1-1/2"-HD1	120/Amb	MB Demin to Boric Acid Conc/MWE	M-162
61720-3"-HD	100/120	Deborating Ion Exchanger to RCST	M-190,198,162,163
61722-1-1/2"-HD1	80/Amb	Cstc Xfer Pp to Dbrtng Ion Xchgr	M-163
62120-4"-HD	15/Amb	Demin RCST to RCST Pp A Suction	M-198

BECHTEL

CALCULATION SHEET

APPENDIX 2

CALC. NO. ______

SIGNATURE Christina F. Kelon DATE 4-26-84 CHECKED L. Tetel DATE 4-26-84

PROJECT Rancho Seco Unit 1 JOB NO. 12334-030

SUBJECT Aux FW Internally Generated Missiles SHEET 6 OF 9 SHEETS

MODERATE ENERGY PIPING

Line No.	Oper. P/T (psig/°F)	Description	Piping Dwg. No.
62180-3"-HD	Atm/Amb	RCST Overflow to BWST	M-198
62191-4"-HD	Atm/Amb	Demin RCST Drain	M-198
62220-3"-HD	100/Amb	RCST Pp to RCST Demin	M-162,163,190,198
62121-3"-HD	100/Amb	Demin RCST Pp B to RCST Demin	M-198
62121-4"-HD	15/Amb	Demin RCST to RCST Pp B Suction	M-198
68986-3"-HD	Atm/120	Ship Cask Decon Area Drn to SRT	M-162,163,187,190
68989-2"-HD	Atm/120	Emer Shwr Drn to SCD Drn	M-187,190
68990-2"-HD	Atm/120	SF Pl Area Drn to SCD Drn	M-162
69321-3"-HD	70/120	Misc Waste Cond Fltr to MWHU Tk	M-190
69322-3"-HD	70/120	Misc Waste cond fltr to MWHU Tk	M-162,163,187,188,198
71120-2"-HD	135/100	Boric Acid Filter to BWST	M-162,163,190,198
77050-3"-HE	90/95	CCW HEX to Chiller	M-162,163
77060-3"-HE	85/125	Chiller to CCW Pp Suction	M-162,163
77181-3"-HE	Atm/Amb	Chilled Wtr Stg Tk PSV to Drain	M-162
77220-2-1/2"-HE	49/55	Chlld Wtr Pp to Chlld Wtr Sply	M-162
77123-2-1/2"-HE	35/55	Chilled Wtr to Chilled Wtr Stg Tk	M-162
81580-6"-HC	1/130	Trbn LO Resv Vapor Extr to Demst	M-188,187
81691-6"-HC	0/130	Demister Vent	M-187
81780-4"-HC	0/130	Demister Vent	M-187
81782-4"-HC	0/130	Demister Vent	M-187
83780-4"-HC	0/130	Gen Brg Drn Vapor Extractor Vent	M-162
85020-24"-HE2	40/100	Gen H_2 Clr to Circ Wtr Intake Cnl	M-188
87420-4"-HC	0/130	FP Turbine LO Extr A to Demst	M-187,188
87421-4"-HC	0/130	FP Turbine LO Extr B to Demis	M-187,188
* 88680-22"-HE	Atm/750	DG A Exhaust to Atm	M-162,163
* 88681-22"-HE	Atm/750	DG B Exhaust to Atm	M-162,163
88682-24"-HE	Atm/Amb	DG Air A Intake Fltr & Slncr A	M-163
88683-24"-HE	Atm/Amb	DG Air B Intake Fltr & Slncr B	M-163
88820-2"-HC	35/100	Diesel FC Pp to DG FO Day Tk	M-163
88820-2-1/2"-HC	35/100	Diesel FO Pp to DG FO Day Tk A	M-188
88823-2"-HC	35/100	Diesel FO Pp to DG FO Day Tk B	M-163
88823-2-1/2"-HC	35/100	Diesel FO Pp to DG FO Day Tk B	M-188

BECHTEL

CALCULATION SHEET

APPENDIX 2
CALC. NO. ______

SIGNATURE Christine F. Kelson DATE 4-26-84 CHECKED L Tentsch DATE 4-26-84
PROJECT Rancho Seco Unit 1 JOB NO. 12334-030
SUBJECT Aux FW Internally Generated Missiles SHEET 7 OF 9 SHEETS

MODERATE ENERGY PIPING

Line No.	Oper. P/T	Description	Piping Dwg. No.
88825-2-1/2"-HC	35/100	Diesel FO Pp to Aux Blr FO Pp	M-188
88829-1-1/2"-HC	35/100	Diesel FO Pp to DG FO Day Tk	M-163
88830-1-1/2"-HC	1/100	Diesel FO Pp to DG FO Day Tk	M-163
88920-1-1/2"-HE	250/Amb	DG Mtr Drvn Comp A to DG	M-163
88921-1-1/2"-HE	250/Amb	DG MD Comp A to Air St Rec A	M-163
88922-1-1/2"-HE	250/Amb	DG MD Comp A to Air St Rec B	M-163
88923-1-1/2"-HE	250/Amb	DG MD Comp A to Air St Rec C	M-163
88924-1-1/2"-HE	250/Amb	AC Comp B to DG B	M-163
88925-1-1/2"-HE	250/Amb	AC Comp B to Receiver G	M-163
88926-1-1/2"-HE	250/Amb	AC Comp B to Receiver H	M-163
88927-1-1/2"-HE	250/Amb	AC Comp B to Receiver I	M-163
88928-1-1/2"-HE	Atm/Amb	DG Crankcase Drain	M-163
88931-1-1/2"-HE	Atm/Amb	DG B Crankcase Drain	M-163
89120-1-1/2"-HE	250/Amb	DG Mtr/Eng Drvn Air St to DG	M-163
89121-1-1/2"-HE	250/Amb	DG Mtr/Eng Drvn Comp A to Rec E	M-163
89122-1-1/2"-HE	250/Amb	DG Mtr/Eng Drvn Comp A to Rec D	M-163
89123-1-1/2"-HE	250/Amb	DG Mtr/Eng Drvn Comp A to Rec F	M-163
89124-1-1/2"-HE	250/Amb	DG AC&DC Mtr Drvn Comp Crosstie	M-163
89125-1-1/2"-HE	250/Amb	DG DC Mtr Drvn Comp B to DG B	M-163
89126-1-1/2"-HE	250/Amb	DG AC Mtr Drvn Comp B to Rec J	M-163
89127-1-1/2"-HE	250/Amb	DG AC Mtr Drvn Comp B to Rec K	M-163
89128-1-1/2"-HE	250/Amb	DG AC Mtr Drvn Comp B to Rec L	M-163
89129-1-1/2"-HE	250/Amb	DC & AC Crosstie to DG	M-163
89390-2"-HC	Atm/Amb	DG FO Day Tk to Diesel FO Stg Tk	M-163
89391-2"-HC	Atm/Amb	DG FO Day Tk A to Dsl FO Stg Tk	M-163
89394-2"-HC	Atm/Amb	DG FO Day Tk Vent	M-162,163
89395-2"-HC	Atm/Amb	DG FO Day Tk Vent	M-162,163
90528-1-1/2"-HE	100/Amb	SVC Air to Reactor Yd Area	M-188,190
90531-2"-HE	100/Amb	SVC Air to Aux Bldg	M-162,163
90536-2"-HE	100/Amb	SVC Air to Radwaste	M-162,163,190
90536-3"-HE	100/Amb	SVC Air to Radwaste Area	M-188,190

CALCULATION SHEET

Appendix 2
CALC. NO.

SIGNATURE Christina Kelson DATE 4-26-84 CHECKED L. Dutach DATE 4-26-84
PROJECT Rancho Seco Unit 1 JOB NO. 12334-030
SUBJECT Aux FW Internally Generated Missiles SHEET 8 OF 9 SHEETS

MODERATE ENERGY PIPING

Line No.	Oper. P/T (psig /°F)	Description	Piping Dwg. No.
91528-1-1/2"-HE2	95/Amb	Instr Air to Aux Bldg	M-188,190,162,163
91528-2"-HE2	95/Amb	Instr Air to Aux Bldg	M-188,190
92080-3"-HE	0/Amb	Generator H_2 Vent to Atm	M-187,162
92520-2"-HE	50/Amb	N_2 to Reactor Bldg	M-162,163,188,190
92525-2"-HE	2/Amb	N_2 to Turbine Area	M-162
96522-6"-HD	25/Amb	Demin Area Sump Pp to SRT	M-162,163
96720-4"-HE4	25/Amb	Acid Waste Sump Pp to SRT	M-162
98127-2"-HE2	120/Amb	Svc Wtr to Reactor Yd	M-188,190
98160-3"-HE2	120/Amb	SVC Wtr Fltr to B/D Tk	M-188
98220-3"-HE2	120/Amb	Domes Wtr Cl_2 Cntct Tk to Aux Bl	M-162,163,187,188,190
98221-2-1/2"-HE	120/Amb	Domes Wtr Cl_2 Cntct Tk to Adm Bl	M-188
98300-6"-HD	10/Amb	MWHU Tk to MWHU Tk Pp	M-198
98320-4"-HD	80/Amb	MWHU Tk Pp to Radwst Demin Hdr	M-198,162,163,190
98397-3"-HD	80/Amb	MWHU Tk Pp to SF St Pool	M-162
98399-2"-HD	80/Amb	MWHU Tk Pp to BWST	M-198
98823-2"-HD1	120/Amb	MB Demin to Lab & Sample Snks	M-162,163
99380-4"-HD	10/Amb	MWHU Tank Drain	M-198
99520-4"-HD	120/Amb	MB Demin to MWHU Tk	M-188,187,190,198
99622-8"-HE1	125/Amb	Fire Loop Line to Aux Bl @ TD	M-162,163,188,190,198
* 99820-4"-GE	300/0	CO_2 Header	M-187,188,190

* Line operates less than 2% of the time during normal plant conditions.
** Based on engineering judgement, pressure is too low for credible missile ejection.

ACRONYMS AND ABBREVIATIONS

B/D	Blowdown
BOC	Bearing Oil Cooler
BWST	Borated Water Storage Tank
CCW	Component Cooling Water
CRD	Control Rod Drive
CRDM	Control Rod Drive Mechanism
CST	Condensate Storage Tank
CW	Cooling Water
DG	Diesel Generator
DHR	Decay Heat Removal
MB	Mixed Bed
MD	Motor Driven
MWE	Miscellaneous Waste Evaporator
MWHU	Miscellaneous Water Hold-Up Tank
NSCW	Nuclear Service Cooling Water
NSRW	Nuclear Service Raw Water
Pp	Pump
RB	Reactor Building
RCP	Reactor Coolant Pump
SCD	Shipping Cask Decon
SF	Spent Fuel
SRT	Spent Regenerant Tank

APPENDIX 3: ELECTRICAL EQUIPMENT

LAO 0012 (4/73)

Appendix 3

CALCULATION SHEET

CALC. NO.

SIGNATURE Christine Kelton DATE 4-26-84 CHECKED Rlm DATE 4-26-84

PROJECT Rancho Seco Unit 1 JOB NO. 12334-030

SUBJECT Aux FW Internally Generated Missiles SHEET 1 OF 10 SHEETS

ELECTRICAL EQUIPMENT

EQUIPMENT	CABLE [a] FROM	CABLE [a] TO	SCHEME [a] CABLE No.	WIRING DIAGRAM No	CIRCUIT SCH [b] PAGE No.
P-318	S4B10	P-318	141B10		1486
P-318 Bearing Heater	H7J921	P-318 Htr	111 H1 - HC		1253
	H7J924	" "	111 H2 - HC		1264
H7J921	H7JFPA	H7J921	111 H1 - GB	E-342-32	1251
H7JFPA	S1FP-A	H7JFPA	111 H1 - FD	E-342-32	1250
S1FP-A	X31P	S1FP-A	111 H1 - FB		1250
X31P	S2A125	X31P	121A125A	E-307-25	1374
H7J924	H7JFPB	H7J924	111 H2 - FW	E-342-32	1262
H7JFPB	S1FP-B	H7JFPB	111 H2 - FD	E-342-32	1262
S1FP-B	X31N	S1FP-B	111 H2 - FB		1261
X31N	S2B128	X31N	121B128A	E-307-25	1560*
P-318 Heater	S2E162	P-318	122E162 (NON CLASS 1E)		
P-319	S4A06	P-319	141A06		1485
P-319 Bearing Heater	H7J922	P-319 Htr	111 H1 - HD		1253
	H7J925	P-319 Htr	111 H2 - HD		1264
H7J922	H7JFPA	H7J922	111 H1 - FY	E-342-32	1251
H7JFPA	S1FP-A	H7JFPA	111 H1 - FE	E-342-32	1250
H7J925	H7JFPB	H7J925	111 H2 - FY	E-342-32	1262
H7JFPB	S1FP-B	H7JFPB	111 H2 - FE	E-342-32	1261
P-319 Heater	S2C547	P-319	122C547	(NON CLASS 1E)	

a. Reference Circuit by Location, dated 1-23-84

b. Reference Circuit Schedule, Rev. 160, dated 3-26-83

* Circuit Schedule Rev. 394 dated 4-6-84

CALCULATION SHEET

Appendix 3

CALC. NO. ______

SIGNATURE Christine J. Kelson DATE 4-26-84 CHECKED RLm DATE 4-26-84

PROJECT Rancho Seco Unit 1 JOB NO. 12334-030

SUBJECT Aux FW Internally Generated Missiles SHEET 2 OF 10 SHEETS

ELECTRICAL EQUIPMENT

EQUIPMENT	CABLE [a] FROM	CABLE [a] TO	SCHEME [a] CABLE No.	WIRING DIAGRAM No.	CIRCUIT SCH [b] PAGE No.
SFV-20577	S2B160	SFV-20577	1M1B160E		547
	"	"	121B160E		1392
SFV-20578	S2A122	SFV-20578	1M1A122A		516
	"	"	121A122A		1374
FV-20527	H7J737	FV-20527	1I1F205BD		236
H7J737	H4SDB00	H7J737	1I1F205BB	E-342-29	236
FY-20527	H4IC01	FY-20527	1I2U1082B	(NON CLASS 1E)	
FY-20527A	H7J737	FY-20527A	1I1F205BC	E-342-29	236
FV-20528	H7J738	FV-20528	1I1F205AD		236
H7J738	H4SDA5	H7J738	1I1F205AB	E-342-29	236
FY-20528	H4IC01	FY-20528	1I2U1082A	(NON CLASS 1E)	
FY-20528A	H7J738	FY-20528A	1I1F205AC	E-342-29	236
FV-30801	H8SFV30801	FV-30801	1I1T308D		265
	"	"	1I1T308H		265
	"	"	1I1T308K		265
FV-30801 Limit Switch	H8SFY-30801	FV-30801 L.S.	1I1T308L		266
FY-30801 Solenoid	"	FY-30801 Sol.	1I1T308M		266
H8SFV30801	H4SDB0	H8SFV30801	1I1T308C		265
	"	"	1I1T308G		265
	"	"	1I1T308X		266

a. Reference Circuit by Location, dated 1-23-84

b. Reference Circuit Schedule, Rev. 160, dated 3-26-83

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CALCULATION SHEET

Appendix 3

CALC. NO. ____

SIGNATURE Christine F. Kelson DATE 4-26-84 CHECKED RLM DATE 4-26-84

PROJECT Rancho Seco Unit 1 JOB NO. 12334-030

SUBJECT Aux FW Internally Generated Missiles SHEET 3 OF 10 SHEETS

ELECTRICAL EQUIPMENT

EQUIPMENT	CABLE[a] FROM	CABLE[a] TO	SCHEME[a] CABLE No.	WIRING DIAGRAM No.	CIRCUIT SCH[b] PAGE No.
FV-31855 LS1	H4CDAR6	FV-31855 LS1	1Y3D318B1	NON CLASS 1E CABLES. WIRING TO BE UPGRADED	
LS2	"	FV-31855 LS2	1Y3D318A1	TO CLASS 1E DURING 1986 OUTAGE AS PART OF EFIC.	
HV-20569	S2A161	HV-20569	1M1A161B		523
	"		121A161A		1381
HV-20596	S2B137	HV-20596	1M1B137B		540
	"	"	121B137A		1389
HV-31826	H7J032	HV-31826	1M1B159C		546
	"	"	1M1B159D		546
	"	"	121B159C		1392
	"	"	121B159D		1392
H7J032	S2B159	H7J032	1M1B159B	E-342-56A	546
			121B159B	E-304-32	1392
			121B159A	E-205-20F	1392
HV-31827	S2A108	HV-31827	1M1A108B		513
	"	"	1M1A108C		513
	"	"	121A108A		1372
	"	"	121A108B		1372
FT-31802	FT-31802	H4SCB	1I1F318B1	E-323-14	239
FT-31803	FT-31803	H4SCA	1I1F318A1	E-323-11	233
FT-31902	FT-31902	H4SCB	1I1F319B1	E-323-14	239
FT-31903	FT-31903	H4SCA	1I1F319A1	E-323-11	239

a. Reference Circuit by Location, dated 1-23-84

b. Reference Circuit Schedule, Rev. 160, dated 3-26-83

BECHTEL

CALCULATION SHEET

Appendix 3

CALC. NO. ______

SIGNATURE Christine J. Kelton DATE 4-26-84 CHECKED RLM DATE 4-26-91

PROJECT Rancho Seco Unit 1 JOB NO. 12334-030

SUBJECT Aux FW Internally Generated Missiles SHEET 4 OF 10 SHEETS

ELECTRICAL EQUIPMENT

EQUIPMENT	CABLE[a] FROM	CABLE[a] TO	SCHEME[a] CABLE No.	WIRING DIAGRAM No.	CIRCUIT SCH[b] PAGE No.
LT-35809	LT-35809	H4SCA	1I1F358A1	E-323-11	239
LT-35810	LT-35810	H4SCB	1I1F358B1	E-323-13	240
PT-31801	PT-31801	H4SCA	1I1F318C1	E-323-11	239
PT-31803	PT-31803	H4SCA	1I1F318D1	E-323-11	239
PT-31901	PT-31901	H4SCB	1I1F319C1	E-323-13	239
PT-31903	PT-31903	H4SCB	1I1F319D1	E-323-13	239
PSL-31757	PSL-31757	H7J319	1I1T308P		266
H7J319	H4SD80	H7J319	1I1T308R	E-342-1	266
PSL-31758	PSL-31758	H7J319	1I1T308F	(SPARE,	NON-1E)
	PSL-31758	H7J319	1I1T308Q		266
PSL-31759	PSL-31759	S4A06	1P1A06F	(SPARE,	NON-1E)
	PSL-31759	S4A06	1P1A06L		656
PSL-31760	PSL-31760	S4A06	1P1A06G	(SPARE	NON-1E)
	PSL-31760	S4A06	1P1A06M		656

a. Reference Circuit by Location, dated 1-23-84
b. Reference Circuit Schedule, Rev. 160, dated 3-26-83

LAO 0813 6-73

BECHTEL

LAO 0613 6-73

CALCULATION SHEET

Appendix 3
CALC. NO. ______

SIGNATURE Christine F. Kelson DATE 4-26-84 CHECKED RM DATE 4-26-84

PROJECT Rancho Seco Unit 1 JOB NO. 12334-030

SUBJECT Aux FW Internally Generated Missiles SHEET 5 OF 10 SHEETS

ELECTRICAL EQUIPMENT

EQUIPMENT	CABLE [a] FROM	CABLE [a] TO	SCHEME [a] CABLE No.	WIRING DIAGRAM No.	CIRCUIT SCH [b] PAGE No.
H4SCA	S1A	H4SCA	111A03A	E-307	
	S0A10	S1A	101A10A	E-107	
	BA	S0A13	101A13A	"	
	"	"	101A13B	"	
	H4SCA	H4CDAR	1Y1X318A2	E-323-11	
	"	"	1Y1X319A2	E-323-11	
	"	"	1Y1X318C2	E-323-11	
	"	"	1Y1X318D2	E-323-11	
	"	"	1Y1X358A2	E-323-11	
H4SCB	S1B	H4SCB	111B03A	E-307	
	S0B10	S1B	101B10A	E-107	
	BB	S0B13	101B13A	"	
	"	"	101B13B	"	
	H4SCB	H4CDAR	1Y1XB318A	E-323-14	
	"	"	1Y1XB319A	E-323-14	
	"	"	1Y1X319C2	E-323-13	
	"	"	1Y1X319D2	E-323-13	
	"	"	1Y1X358B2	E-323-13	
H4SDA5	S0A01	H4SDA5	101A01A	E-107-4	
H4SDB	S0B01	H4SDB	101B01B	E-107-4	
S2A	S3A22	S2A1	131A22A	E-105-1	
S3A	X43A	S3A		E-105-1	
	S4A09	X43A	141A09A	"	
S4A	GE A	S4A	141A08A	E-104-2	
	"	"	141A08B	"	

a. Reference Circuit by Location, dated 1-23-84
b. Reference Circuit Schedule, Rev. 160, dated 3-26-83

CALCULATION SHEET

BECHTEL

LAO 0813 6-73

Appendix 3

CALC. NO. ______

SIGNATURE Christine F. Kelton DATE 4-26-84 CHECKED Rim DATE 4-26-84

PROJECT Rancho Seco Unit 1 JOB NO. 12334-030

SUBJECT Aux FW Internally Generated Missiles SHEET 6 OF 10 SHEETS

ELECTRICAL EQUIPMENT

EQUIPMENT	CABLE [a] FROM	CABLE [a] TO	SCHEME [a] CABLE No.	WIRING DIAGRAM No.	CIRCUIT SCH [b] PAGE No.
S2B	S3B22	S2B1	131B22A	E-105-2	
	"	"	131B22B	"	
S3B	X43B	S3B	131B05A	E-105-2	
	"	"	131B0[illegible]	"	
	"	"	131B05D	"	
	S4B05	X43B	141B05A	E-104-3	
	"	"	141B05B	"	
S4B	GE B	S4B	141B11A	E-104-3	
	"	"	141B11B	"	
H4SCA	H4SCA	H2PS	1I2F318B1	E-323-15	NON-CLASS 1E
	"	"	1I2F319B1	E-323-15	NON-CLASS 1E

a. Reference Circuit by Location, dated 1-23-84

b. Reference Circuit Schedule, Rev. 160, dated 3-26-83

APPENDIX 3

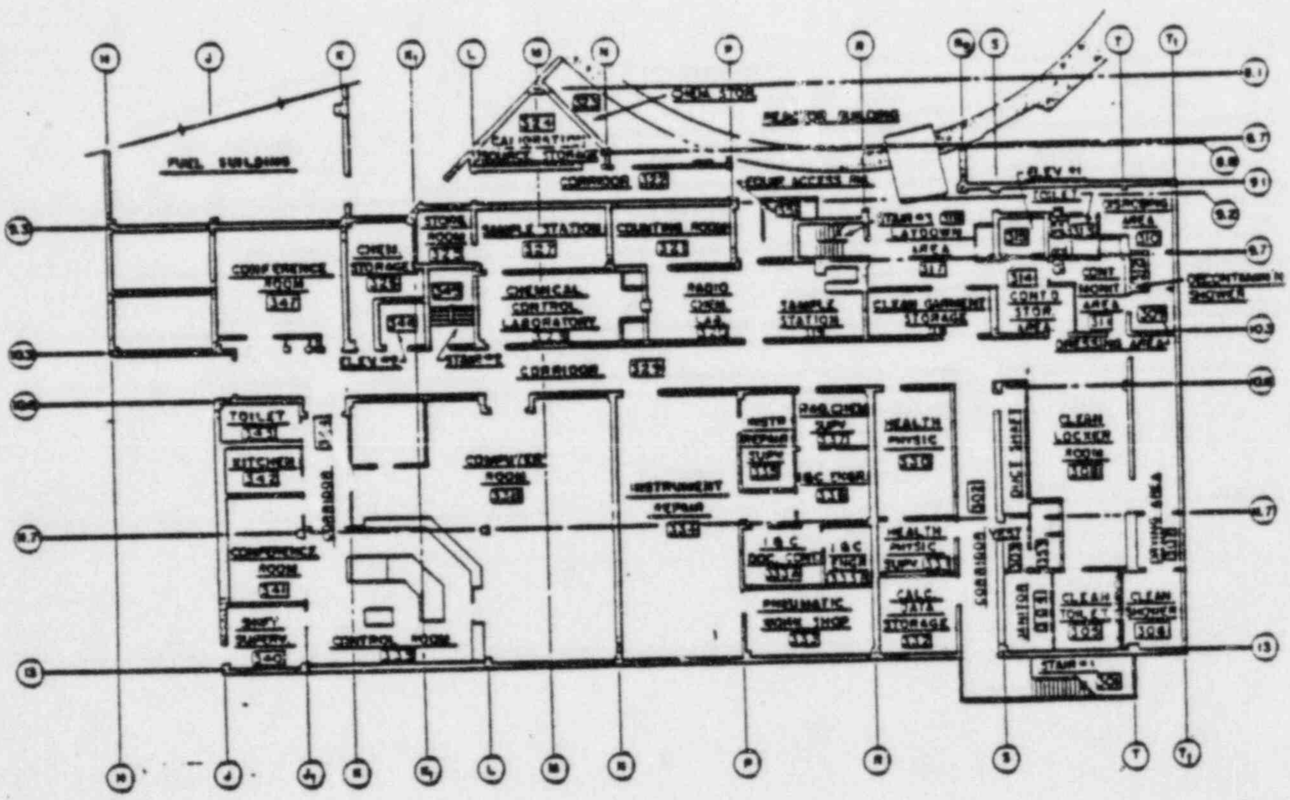
CALCULATION SHEET

CALC. NO. ____________

SIGNATURE Christine F Kelton DATE 4-26-84 CHECKED [illegible] DATE 4/26/84

PROJECT Rancho Seco Unit 1 JOB NO. 12334-030

SUBJECT Aux FW Internally Generated Missiles SHEET 7 OF 10 SHEETS



TURBINE DECK LEVEL PLAN ELEVATION 40'

0 5 10 15 20
SCALE IN FEET

ROOM CONTAINS HIGH ENERGY PIPING, ROTATING EQUIPMENT, OR COMPRESSED GAS STORAGE EQUIPMENT.

AUXILIARY BUILDING - PLAN AT EL. 40'-0"
STUDY OF INTERNALLY GENERATED MISSILES

BECHTEL

CALCULATION SHEET

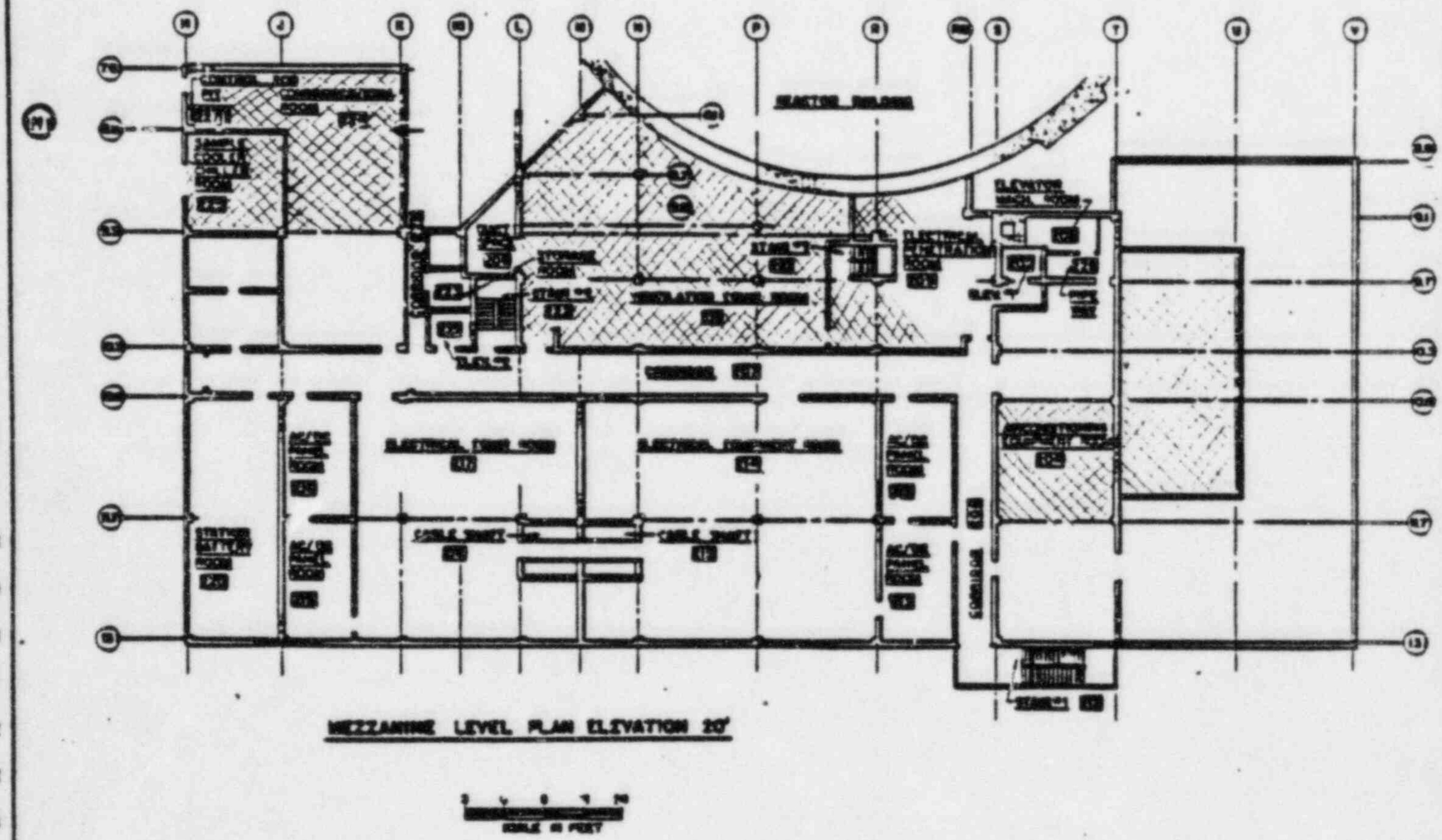
APPENDIX 3

CALC. NO. ______

SIGNATURE Christine J. Kelton DATE 4-26-84 CHECKED [illegible] DATE 4/26/84

PROJECT Rancho Seco Unit 1 JOB NO. 12334-030

SUBJECT Aux FW Internally Generated Missiles SHEET 8 OF 10 SHEETS



ROOM CONTAINS HIGH ENERGY PIPING, ROTATING EQUIPMENT, OR COMPRESSED GAS STORAGE EQUIPMENT.

AUXILIARY BUILDING - PLAN AT EL. 20'-0"

STUDY OF INTERNALLY GENERATED MISSILES

CALCULATION SHEET

BECHTEL

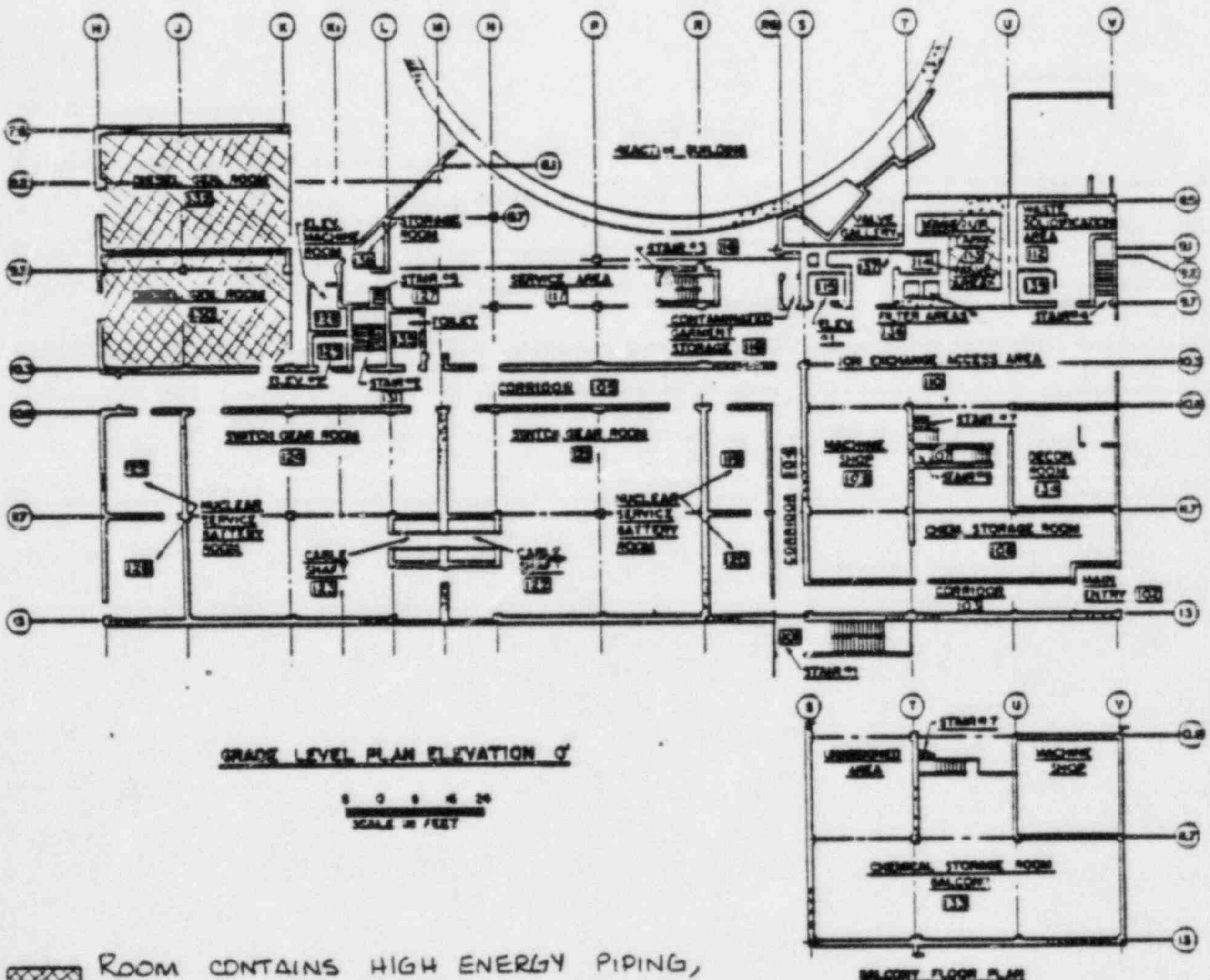
APPENDIX 3

CALC. NO. ____

SIGNATURE Christine F. Kelton DATE 4-26-84 CHECKED [illegible] DATE 4/26/84

PROJECT Rancho Seco Unit 1 JOB NO. 12334-030

SUBJECT Aux FW Internally Generated Missiles SHEET 9 OF 10 SHEETS



ROOM CONTAINS HIGH ENERGY PIPING, ROTATING EQUIPMENT, OR COMPRESSED GAS STORAGE EQUIPMENT.

AUXILIARY BUILDING - PLAN AT EL. 0'-0"

STUDY OF INTERNALLY GENERATED MISSILES

CALCULATION SHEET

APPENDIX 3

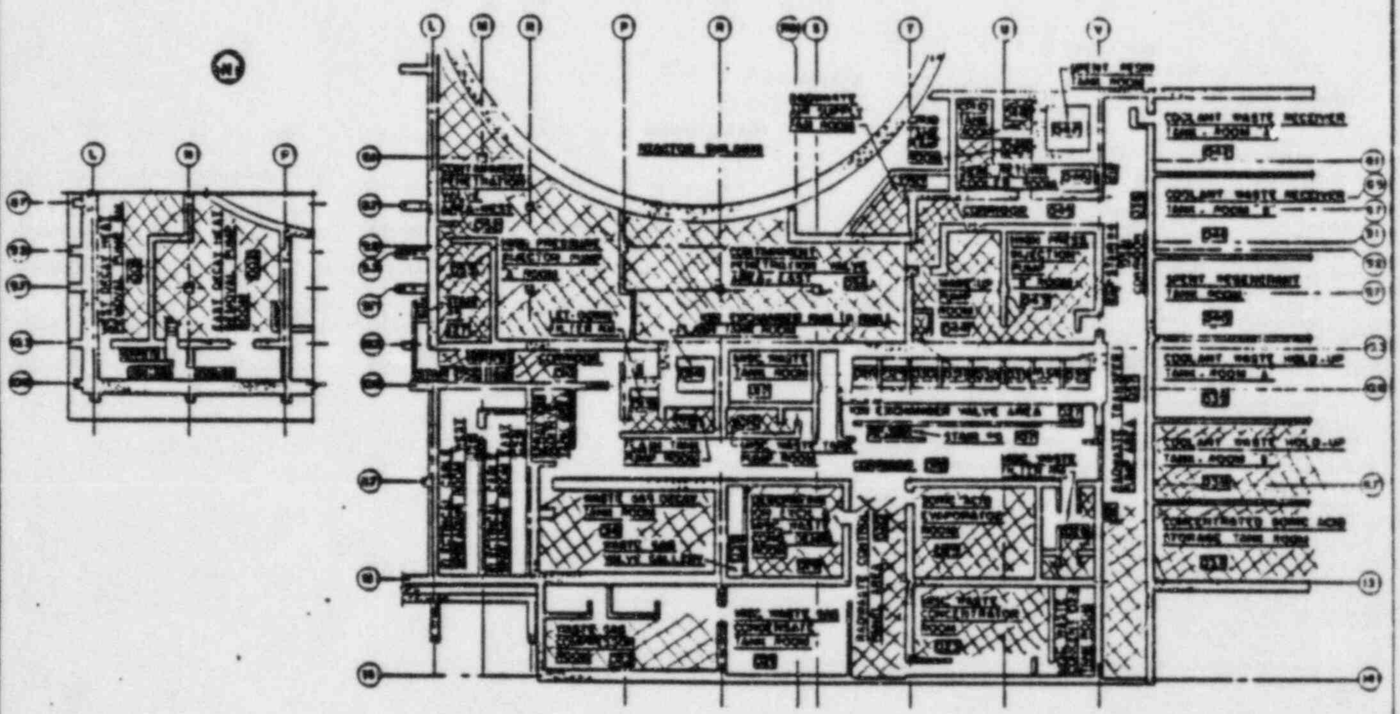
LAO 0513 8-73

CALC. NO. ______

SIGNATURE Christine Kelson DATE 4-26-84 CHECKED [illegible] DATE 4/26/84

PROJECT Rancho Seco Unit 1 JOB NO. 12334-030

SUBJECT Aux FW Internally Generated Missiles SHEET 10 OF 10 SHEETS



BASEMENT LEVEL PLAN ELEVATION (-) 20'

SCALE IN FEET

ROOM CONTAINS HIGH ENERGY PIPING, ROTATING EQUIPMENT, OR COMPRESSED GAS STORAGE EQUIPMENT

AUXILIARY BUILDING. PLAN AT EL. (-) 20'-0"

STUDY OF INTERNALLY GENERATED MISSILES

APPENDIX 4: ROTATING EQUIPMENT

CALCULATION SHEET

APPENDIX 4
CALC. NO. ______

SIGNATURE Christine J. Kelom DATE 4-26-84 CHECKED TMK DATE 4/26/84
PROJECT Rancho Seco Unit 1 JOB NO. 12334-030
SUBJECT Aux FW Internally Generated Missiles SHEET 1 OF 2 SHEETS

ROTATING EQUIPMENT

TAG NO.	DESCRIPTION	AFWS COMPONENT IN PLANE OF ROTATION	INTERVENING STRUCTURES	AFWS IMPACT	COMMENTS
P-462A	Component Cooling Water Pump A	31827-6"-GB	P-462B	NONE	MISSILES FROM P-462A ARE SHIELDED BY P-462B AND ITS ASSOCIATED PIPING.
P-462B	Component Cooling Water Pump B	31827-6"-GB	NONE	NONE	IMPACT IS DOWNSTREAM OF FV-31855 ON NON-SEISMIC PIPING
		CONDUIT M11249	NONE	NONE	CONDUIT IS REDUNDANT; OTHER CABLE IS NOT IMPACTED.
P-482A	Nuclear Service Cooling Water Pump A	NONE	N/A	NONE	
P-482B	Nuclear Service Cooling Water Pump B	NONE	N/A	NONE	
P-622A	Demineralized Reactor Coolant Storage Tank Pump A	CONDUIT L11AD1	NONE	NONE	CONDUIT IS REDUNDANT; OTHER CABLE IS NOT IMPACTED. MISSILE PARTIALLY SHIELDED BY STEEL COLUMNS.
P-622B	Demineralized Reactor Coolant Storage Tank Pump B	CONDUIT L11AD1	NONE	NONE	SAME AS P-622A
P-983	Miscellaneous Water Hold-Up Tank Pump	NONE	N/A	NONE	
P-272	Spent Fuel Coolant Pump	31823-6"-DB2	CONCRETE PILLAR @ 3.4 & H8	NONE	PILLAR COMPLETELY SHIELDS PUMP FROM AFWS PIPING.
P-274	Spent Fuel Coolant Pump	31823-6"-DB2	Radiation monitor RB-15018	NONE	RADIATION MONITOR CABINET COMPLETELY SHIELDS PUMP FROM AFWS PIPING.

CALCULATION SHEET

BECHTEL

APPENDIX 4 LAO 0513 6-73

CALC. NO.

SIGNATURE Christine A. Kelton DATE 4-26-84 CHECKED fMK DATE 4/26/84

PROJECT Rancho Seco Unit 1 JOB NO.

SUBJECT Aux FW Internally Generated Missiles SHEET 2 OF 2 SHEETS

ROTATING EQUIPMENT

TAG NO.	DESCRIPTION	AFWS COMPONENT IN PLANE OF ROTATION	INTERVENING STRUCTURES	AFWS IMPACT	COMMENTS
P-251	Low Pressure Injection Header Warming Pump	NONE	N/A	NONE	

APPENDIX 5: DESIGN CRITERIA

DESIGN CRITERIA COVER SHEET

DATE 4-27-84	SACRAMENTO MUNICIPAL UTILITY DISTRICT Rancho Seco Nuclear Generating Unit 1 Bechtel Job No. 12334	DESIGN CRITERIA	PART N/A
REVISION 0		Internally Generated Missiles	

PRINCIPAL RESPONSIBILITY Bechtel Mechanical - Design City

DESIGN CRITERIA Auxiliary Feedwater System - Criteria for Postulating Internally Generated Missiles outside Containment

QUALITY CLASS 1

SAFETY CLASS N/A

SEISMIC CATEGORY N/A

NRC REGULATORY GUIDE R.G. 1.70, Sec. 3.5, S.R.P. 3.5.1.1., S.R.P. 10.4.9

NRC GENERAL DESIGN CRITERIA 4 "Environmental and Missile Design Basis", 10CFR50 APP. "A"

REFERENCES

SAR SECTION 1.4.40

P&I DRAWINGS N/A

OTHER DRAWINGS N/A

APPROVAL SIGNATURES

[signature] 4-18-84 GL (ORIGIN DATE)	[signature] 4/18/84 GS (ORIGIN DATE)	[signature] for A.J.O. 4/27/84 MECH. CHIEF DATE
[signature] 4/27/84 PE DATE	N/A PLANT DESIGN GS DATE	[signature] 4-26-84 MECH. GS DATE
[signature] 4/27/84 ELEC. GS DATE	[signature] 4/27/84 CIVIL GS DATE	N/A CONTROL GS DATE
[signature] 4/26/84 NUC. GS DATE	N/A ARCH. GS DATE	[signature] FOR T.E. RICHARDSON 4/26/84 NUC. CHIEF DATE

DATE OF ORIGIN 4.27-84

DATE(S) OF ALL CHANGES

DESIGN CRITERIA

AUXILIARY FEEDWATER SYSTEM - CRITERIA FOR POSTULATING INTERNALLY GENERATED MISSILES OUTSIDE CONTAINMENT

RANCHO SECO UNIT 1

TABLE OF CONTENTS

I. PURPOSE AND SCOPE

This design criteria provides the guidance, scope of analysis, analysis criteria and methodology for the protection of Auxiliary Feedwater System (AFWS) components located outside containment building from the effects of postulated internally generated missiles.

The evaluation of the AFWS components (Electrical, Mechanical and Instrumentation/Controls) against the effects of internally generated missiles is limited to those components which fall into one of the following categories.

- Existing components;
- Proposed components for which installation drawings have been released for construction.

The evaluation does not consider those components that are in the planning stages, for which installation drawings have not been raleased for construction. However, evaluation shall be performed for these components whenever installation drawings become available for review.

II. MISSILE DESIGN CRITERIA

A. Definitions

1. Missiles:

   A missile shall be defined as a mass which has kinetic energy and is unrestrained.

2. Essential Components (Auxiliary Feedwater System):

   Essential components shall be identified as those which are a part of the AFWS and are required to mitigate the consequences of the accident, prevent a significant uncontrolled release of radiation, or place the plant in a cold shutdown condition. These essential AFWS components shall include electrical controls, and mechanical items. A schematic of the AFWS is shown in Figure 1.

3. High Energy Fluid Systems:

   High energy fluid systems shall include those pressurized systems or portions thereof in which the normal operating temperature exceeds 200°F or the normal operating pressure exceeds 275 psig for more than two percent of the time it operates during normal plant conditions. For the purposes of the missile analysis, portions of the AFWS, which are not pressurized during normal plant conditions, are excluded from the high energy system criteria.

4. Normal Plant Conditions:

   Normal plant conditions include reactor startup, operation at power, hot standby, or reactor cooldown to cold shutdown condition.

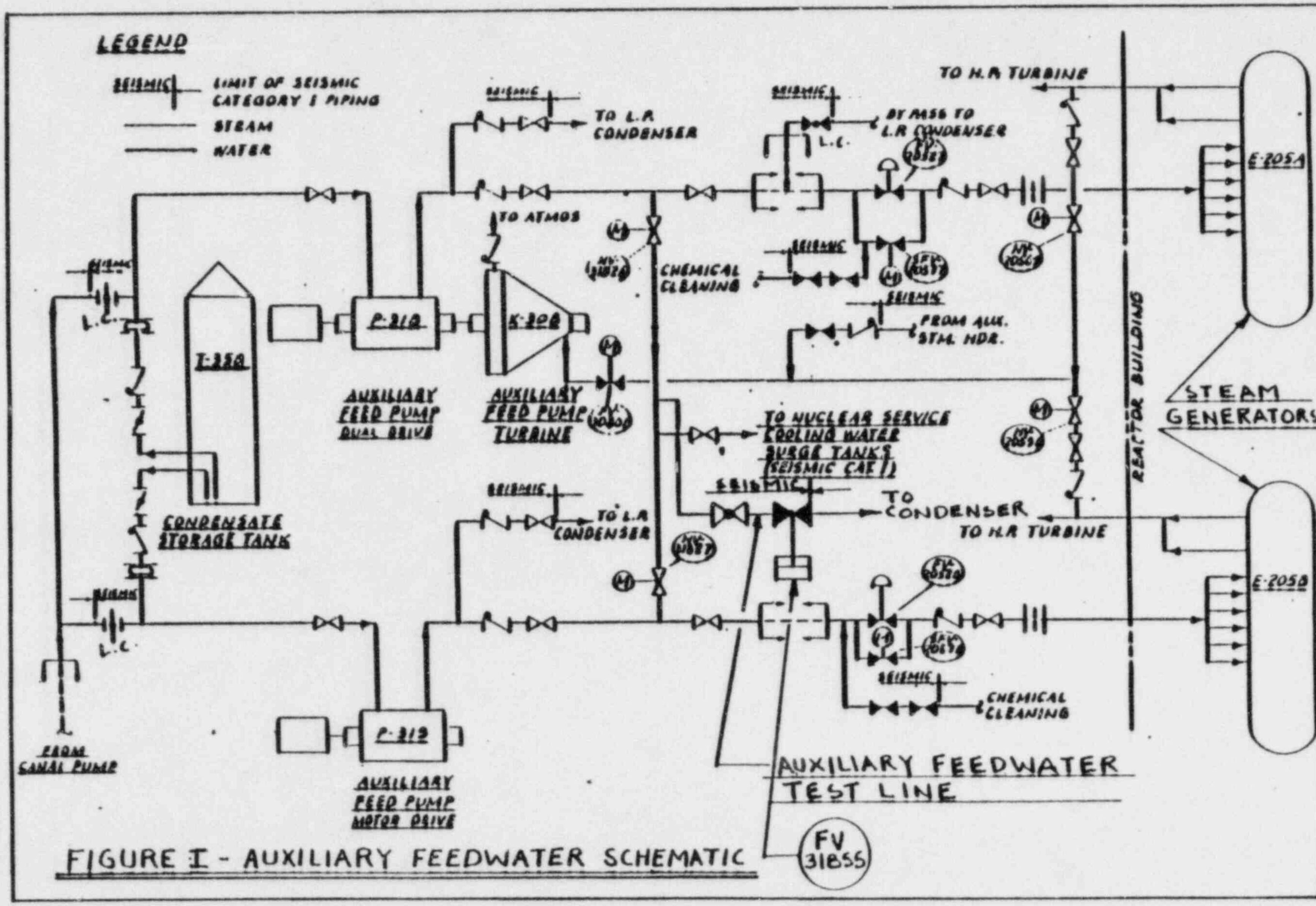


FIGURE I - AUXILIARY FEEDWATER SCHEMATIC

II. B. GENERAL

The basic philosophy of missile protection is to assure that one train of the AFWS will always be free of damage (i.e. operable) from the effects of any postulated internally generated missile sources located outside containment building. The purpose of this is to ensure that one train of the AFWS will always be available in any emergency situation to remove reactor decay heat and provide for the cooldown of the Reactor Coolant System to within the temperature and pressure limits at which the Decay Heat System can be placed in operation.

This assurance shall be provided by analyzing the effects of postulated internally generated missiles on AFWS components located outside the containment building. If the analysis indicates damage to components of both AFWS trains from a single postulated missile source, additional measures shall be taken to protect the components of at least one train from postulated missile hazards.

Only sources listed below shall be considered as credible missile sources:

1. Rotating machinery which operates during normal plant conditions.
2. High energy fluid system components including:
   a. Valve bonnets and stems except as excluded below in Item II.C.
   b. Temperature instrument wells and thimbles.
   c. Pressure instruments and connections.
   d. Welded dead-end flanges and caps.*
   e. Vents, drains and test connections.*
   f. Compressed gas storage system components.*

* - Only if failure of single circumferential weld would cause ejection.

C. EXCLUSIONS

1. Externally generated missiles such as those due to tornado winds are dealt with in Volume I of the Updated Safety Analysis Report, Section 1.5.2, and are not considered as part of this criteria document. Only missiles generated internally as a result of equipment failure are considered.
2. Analysis of turbine missiles is not included in these criteria. Turbine missiles are described in Volume VIII of Updated Safety Analysis Report, Appendix 5C.
3. Equipment which utilizes more than one retention feature in its design is not considered as capable of generating missiles.
   a. Pressure seal bonnet-type valves of ANSI B16.5, 900 psig rating and above, constructed in accordance with ASME Boiler and Pressure Vessel Code, Section III[1], have a retaining ring and yoke. Because of this dual retention feature and because of the highly conservative design of the retaining rings, pressure seal bonnets are not considered credible missiles.
   b. For valves with bolted bonnets, the bonnet-to-body bolting material limiting stresses and flange design set forth in the ASME Boiler and Pressure Vessel Code, Section III[1], prevent the valve bonnet from becoming a missile. The likelihood of complete severance of all bolts simultaneously is very remote with the result that bolted valve bonnets need not be considered as credible missiles.

1 - Or equivalent Nuclear Pump and Valve Code to which existing equipment were designed and constructed.

c. Valve stems with backseats or air, solenoid or motor-operated valve stems utilize more than one retention feature and, therefore, are not considered a missile source.

d. Components of high energy fluid piping with a nominal diameter of 1" or smaller have only a small amount of energy and are not considered as potential destructive missiles.

e. Valve body bypass lines with 1" or smaller valves are not considered as potential destructive missiles since the lines are attached to the valve body at both ends which meets the double retention feature criteria. The bypass valves themselves are excluded based on size criteria per Item d above.

f. Instrument connections with integral tubing for pressure differential transmitters or flow elements are not considered as potential missile sources since the tubing will restrain the ejected component from becoming a free missile.

4. Nuts, bolts, studs, and combinations of nuts with bolts or studs have only a small amount of energy and are not considered as potential destructive missiles.

5. This document does not consider pressure vessels and heat exchangers, as being capable of failure (producing missiles by fragmentation of the vessel casing).

6. Normally closed gate valves are not considered as potential missile sources.

7. Secondary missiles are excluded from the scope of this design criteria.

III. GENERAL MISSILE PROTECTION DESIGN CRITERIA

Where analysis indicates that internally generated missiles cannot be contained locally, the following protective measures shall be considered in order of preference listed below.

A. Where a critical target and missile source share a common area, efforts shall be made to provide adequate physical separation by relocating or reorienting either the target or the missile.

B. The use of non-destructive examination and calibration of high energy equipment to reduce the probability of missile occurrence to a level where consideration of damage is unnecessary (PRA).

C. Damage to one train of the AFWS is permitted from a potential missile source originating from another system. This is an exception to the Single Active Failure Criteria of SRP.3.5.1.1 (Ref. VI.J). Protection shall be provided if components in both trains could get damaged from a single potential missile originating from another system.

D. Where physical separation is not feasible, structural walls or specially erected barriers shall be used as protective devices .

IV. ANALYSIS CRITERIA

A. The missile analysis shall identify potential missile sources by reviewing high energy sources and rotating machinery.

B. Determination of missile characteristics (mass, velocity, geometry, trajectory and deformation characteristics) shall be made on a case-by-case basis, based on most probable point of failure. The direction of travel for missiles shall be in the direction of the applied force. For rotating machinery, targets are only considered if they are within the plane of rotation.

C. The kinetic energy of the missile shall be determined in order to calculate the ability of the missile to perforate the enclosure or restraining device and reach the target, or to penetrate concrete walls to a depth sufficient to cause spalling. This calculation shall account for any physical object which may separate the target and source. Missiles which are physically remote from the target need not be given further consideration. Methods for determination of depth of penetration, perforation, spalling, and residual velocity shall be those of BC-TOP-9A. If it is identified that barriers are required, the design of barriers for missile impact shall be performed in accordance with Civil Design Guide C2.45.

V. ANALYSIS TECHNIQUES

The missile analysis for the Auxiliary Feedwater System will be conducted as follows:

A. The first step in the missile review is the identification of all the safety-related equipment including pumps, valves, tanks, piping, instrument, instrument sensing lines and electrical cables associated with the Auxiliary Feedwater System. This can be accomplished by using the piping and instrumentation diagrams, logic diagrams and elementary drawings. Once all the safety-related components associated with the AFWS are identified, their physical location can be determined by using the piping area drawings, equipment location drawings, conduit and tray drawings, and any other applicable drawings.

B. The second step in the missile review is the identification of those areas outside containment building, where missiles from one single source can damage components of both trains of the AFWS simultaneously.

C. If the potential missiles identified in Step B above cannot be excluded based on the criteria outlined in Section II.C, then determine if the essential targets are located within the direction of these missiles. If there are no targets, then the missile analysis is complete and no protection is required. If essential targets belonging to both trains of AFWS are located within the direction of these missiles, then calculate the kinetic energy, depth of penetration, perforation thickness, spalling and residual velocity utilizing the formulas provided in BC-TOP-9A.

D. If the analysis based on the calculations performed in Step C indicates damage to components of both train simultaneously, then protection shall be provided for components of at least one train in accordance with the steps outlined in Section III of this design criteria.

E. Deviations from this procedure to reduce unnecessary work are acceptable as long as the basic philosophy of protection is not violated as described in paragraph II.B of this design criteria.

VI. REFERENCES

A. USNRC Standard Review Plan 3.5.1.1, "Internally Generated Missile (outside containment)", NUREG-0800, Rev. 2, July 1981.

B. Bechtel Power Corporation, Design Guide C2.45, "Design of Structures for Tornado Missile Impact", Revision 0, April 1982.

C. USNRC Standard Review Plan 3.5.3, "Barrier Design Procedures", NUREG-0800, Rev. 1, July 1981.

D. "Plant Design Against Missiles", American Nuclear Society, ANSI N177, April 1974.

E. BC-TOP-9A, "Design of Structures for Missile Impact, Bechtel Power Corporation, Rev. 2, September 1974.

F. General Design Criterion 4, "Environmental and Missile Design Bases", Appendix A to 10 CFR 50.

G. Regulator Guide 1.70, Section 3.5.

H. Rancho Seco Design Guide for assumptions and criteria for pipe break protection review, Revision 2, March 1983.

I. Letter from John F. Stolz of NRC to R. J. Rodriguez of SMUD on "Status of Auxiliary Feedwater Upgrade Review", dated September 26, 1983.

J. IOM from T. Khan of BPC to D. Abbott/R. Dietrich on Auxiliary Feedwater System, dated March 5, 1984.

K. Bechtel Power Corporation, SNUPPS FSAR, Section 3.5.1, Revision 1, September 1980.

L. USNRC Standard Review Plan 10.4.9, "Auxiliary Feedwater System (PWR)", NUREG-08700, Rev. 2, July 1981.

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4/16/84

SMUD

SACRAMENTO MUNICIPAL UTILITY DISTRICT ☐ 6201 S Street, Box 15830, Sacramento, California 95813; (916) 452-3211

RJR 83-733

November 7, 1983

DIRECTOR OF NUCLEAR REACTOR REGULATION
ATTENTION JOHN F STOLZ CHIEF
OPERATING REACTOR BRANCH IV
U S NUCLEAR REGULATORY COMMISSION
WASHINGTON D C 20555

DOCKET 50-312
RANCHO SECO NUCLEAR GENERATING STATION
UNIT NO 1
AUXILIARY FEEDWATER SYSTEM UPGRADE REVIEW - NUREG 0737 ITEM II.E.1.1

The Sacramento Municipal Utility District has reviewed your letter of September 26, 1983 which provided a status of your evaluation of the Rancho Seco Auxiliary Feedwater System (AFWS) Upgrade. You stated that our position regarding the protection of the AFWS from internally generated missiles was not acceptable. This letter is to inform you of our desire to appeal your decision.

The requirements of the TMI action plan, NUREG 0737, Items II.E.1.1 and 2 have been completed. In particular, the auxiliary feedwater system now meets the automatic initiation and flow indication requirements of Item II.E.1.2. The District's decision to upgrade the auxiliary feedwater system with the addition of an Emergency Feedwater Initiation and Control (EFIC) system was beyond the scope of this document. The Class I control system is intended to prevent reactor system overcooling transients and steam generator overfill. We object to the staff's provison that the upgraded system be protected against the affects of internally generated missiles in accordance with the guidelines of Standard Review Plan Sections 3.5.1.1 and 3.5.1.2. We therefore wish to appeal to NRR management in accordance with the Commission's policy statement in the Federal Register dated Wednesday, September 28, 1983, page 44173, since we feel this is a backfitting requirement. An internally generated missile study has never been performed for the Rancho Seco Nuclear Generating Station.

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We did commit, however, on June 3, 1983, to address this issue by a walkdown of the auxiliary feedwater system to identify any components which due to their location or proximity of one component to the other could be susceptible to a single internally generated missile. At such locations, we also committed to provide missile protection. We feel this action is adequate even though it is not in accordance with current guidelines.

We will remain in contact with our Project Manager, Sydney Miner, to determine the scheduling for the proper actions in this appeal process. At this time, we have requested a quotation from Bechtel Corporation for the performance of an internally generated missile study. This information will provide input to a cost benefit analysis, however, the cost of missile protection will remain an unknown since we do not intend to actually perform the study. If we can provide any further information at this time, please advise.

R. J. Rodriguez
Executive Director, Nuclear

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

September 26, 1983

Docket No. 50-312

Mr. Ronald J. Rodriguez
Executive Director, Nuclear
Sacramento Municipal Utility District
6201 S Street
Post Office Box 15830
Sacramento, California 95813

Dear Mr. Rodriguez:

SUBJECT: RANCHO SECO - STATUS OF THE AUXILIARY FEEDWATER (AFWS) UPGRADE REVIEW (NUREG-0737 ITEM II.E.1.1)

By letter dated April 7, 1983, we provided you with status report of our evaluation of the Rancho Seco upgrade AFWS. In our letter we stated that three open items remain where we require additional information to complete our review. These were: (1) protection of the AFWS from internal missiles; (2) additional protection for the condensate storage tank; and (3) pipe break analysis for all AFWS components including existing components. By letters dated May 2, 1983, June 3, 1983, and June 21, 1983, you provided additional information regrading these items. We have completed our review of the information and the results of the review are provided in the enclosed Safety Evaluation Report (SER).

As outlined in the SER we have concluded that you have satisfied our requirements for items (2) and (3) above and we, therefore, consider these items completed. For item (1) protection of the AFWS from internal missiles you proposed evaluating the effects of internal missiles and will provide missile protection where a single missile could incapacitate both AFWS trains. This is not acceptable to us. Our position is that the effects of missiles should not adversely affect the AFWS function considering a single active failure. Therefore, within 30 days of receipt of this letter, please provide in writing a commitment that you will meet the staff's position.

As discussed with Mr. Bob Dieterick of your staff, NRR procedures provide an opportunity for an appeal by a licensee to NRR management when the staff imposes new requirements on a licensee (backfit) and the licensee objects to the position. Since our internal missile position is a potential backfit requirement for Rancho Seco, you may wish to appeal our position to NRR management. If you decide to appeal to NRR management to have the staff's position modified, within 30 days or receipt of this letter, please indicate in writing that (a) you object to the staff's position; (b) you wish to appeal the staff's position to NRR management to have it modified; and (c) your proposed modification. Should you have any additional questions regarding the staff's position or the appeal process, please contact the NRR project manager.

8310110471

The reporting and/or recordkeeping requirements contained in this letter affect fewer than ten respondents, therefore, OMB clearance is not required under P.L. 96-511.

Sincerely,

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

cc w/enclosure(s):

David S. Kaplan, Secretary and General Counsel
Sacramento Municipal Utility District
6201 S Street
P. O. Box 15830
Sacramento, California 95813

Sacramento County
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SAFETY EVALUATION REPORT

RANCHO SECO - AUXILIARY FEEDWATER SYSTEM

In accordance with the requirements of Item II.E.1.1 of NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," and NUREG-0737, "Clarification of TMI Action Plan Requirements," the Licensee is required to:

(1) Perform a simplified auxiliary feedwater (AFW) system reliability analysis that uses event-tree and fault-tree logic techniques to determine the potential for AFW system failure under various loss-of-main feedwater-transient conditions. Particular emphasis is given to determine potential failures that could result from human errors, common causes, single-point vulnerabilities and test and maintenance outages.

(2) Perform a deterministic review of the AFW system using the acceptance criteria of Standard Review Plan Section 10.4.9 and associated Branch Technical Position ASB 10-1 as principal guidance; and

(3) Reevaluate the AFW system flow rate design bases and criteria.

Our evaluation of the Rancho Seco auxiliary feedwater system (AFWS) against the requirements of Item II.E.1.1 is presented in two parts. Part I is our evaluation of the proposed AFWS upgrade design against the criteria of the Standard Review Plan. Part II is our evaluation of the (1) AFWS against the criteria developed after the Three Mile Island Unit 2 accident and enumerated in NUREG-0611 and NUREG-0635, (2) the licensee's reliability analyses, and (3) the licensee's reevaluation of the design basis for the AFWS flow requirements. Our evaluation of the immediate actions required by the Commission shutdown order of May 7, 1979 was provided in a Safety Evaluation Report which was transmitted to the licensee by letter dated June 27, 1979. Nothing in our current review has resulted in a change to the conclusions expressed in the

5. General Design Criterion 44, "Cooling Water," to assure:

   a. The capability to transfer heat loads from the reactor system to a heat sink under both normal operating and accident conditions.

   b. Redundancy of components so that under accident conditions the safety function can be performed assuming a single active component failure. (This may be coincident with the loss of offsite power for certain events.)

   c. The capability to isolate components, subsystems, or piping if required so that the system safety function will be maintained.

6. General Design Criterion 45, "Inspection of Cooling Water System," as related to design provisions made to permit periodic inservice inspection of system components and equipment.

7. General Design Criterion 46, "Testing of Cooling Water System," as related to design provisions made to permit appropriate functional testing of the system and components to assure structural integrity and leak-tightness, operability and performance of active components, and capability of the integrated system to function as intended during normal, shutdown, and accident conditions.

8. Regulatory Guide 1.26, "Quality Group Classification and Standards for Water-, Steam- and Radioactive Waste Containing Components for Nuclear Power Plants," as related to the quality group classification of system components.

9. Regulatory Guide 1.29, "Seismic Design Classification," as related to seismic design classification of system components.

10. Regulatory Guide 1.62, "Manual Initiation of Protective Actions," as related to design provisions made for manual initiation of each protective action.

initiation and control. One AFWS train is served by pump P-318, a combination turbine-driven/motor-driven pump with both the turbine and electric motor on a common shaft. Either motive source can drive the pump at its rated capacity of 840 gpm at 1150 psig with a normal recirculation flow of 60 gpm. The turbine driver is used as the primary motive source for this pump and is automatically initiated. The motor driver can only be manually initiated. The pump serving the other train, pump P-319, is a motor-driven pump which has the same rated capacity and recirculation flow as pump P-318. Pump P-319 is automatically initiated and automatically loaded on the emergency bus. The discharge lines from the pumps are cross-connected by a full-flow line containing two normally-open motor operated valves in series. This cross-connect permits either pump to feed either or both steam generators. The primary water source for both AFW trains is the seismic Category I condensate storage tank. Alternative AFWS suction sources are available from the on-site reservoir and the Folsom South Canal. Piping from these alternative sources enters the cross-connect in the suction piping between locked closed manual valves. The alternative source is fed by transfer pumps from the Folsom South Canal or by gravity flow from the reservoir.

Rancho Seco is a one unit site, therefore General Design Criterion 5 is not applicable.

1. We have reviewed the licensee's submittals listed above in order to verify the acceptability of the AFWS design with respect to its classification and operating characteristics.

    a. There is not sufficient information available to ensure that the AFWS will meet all the current guidelines for the various functions of the system. Specifically, the licensee has not verified that the proposed upgraded AFWS will be protected against the effects of internally generated missiles in accordance with the guidelines of Standard Review Plan Sections 3.5.1.1 and 3.5.1.2. Therefore we cannot conclude that the AFWS meets all the minimum performance requirements of General Design

ANSI B311.1 criteria with the exception of the AFWS containment penetrations which were designed to ANSI B31.7 Class 2 criteria. These are the criteria that were in effect at the time the existing Rancho Seco AFWS was designed and they pre-date the ASME codes specified in Regulatory Guide 1.26. However, the essential design criteria of the ANSI B31.1 and B31.7 Class 2 codes are basically the same as current design criteria. Therefore, we conclude that the AFWS, as upgraded, meets the requirements of General Design Criterion 2 and the guidelines of Regulatory Guide 1.29 and meets the essential features of Regulatory Guide 1.26 and is, therefore, acceptable.

d. Provisions for AFWS testing and inspection are included in the design. Each AFW pump is equipped with a full flow recirculation line to the condensers which can be used for periodic functional testing purposes. Periodic testing of the AFWS pumps and valves is identified in the plant Technical Specifications. In addition, plant Technical Specifications require periodic inspection of all valves, including those that are locked, sealed, or otherwise secured in position. Therefore, we conclude that the AFWS meets the requirements of General Design Criteria 45 and 46 with regard to design provisions for inservice inspection and functional testing.

2. We have reviewed the AFWS design for protection against the effects of natural phenomena, pipe breaks or cracks in fluid systems outside containment, single system component failures, loss of an onsite motive power source, or loss of offsite power.

a. We have evaluated the upgraded AFWS design against the requirements of General Design Criterion 2 with respect to the structures housing the system and the system itself being capable of withstanding the effects of earthquakes. By letter dated January 14, 1983 the licensee verified that all AFWS essential components will be located either outdoors or in seismic Category I structures or will be otherwise provided with relative to turbine missile protection as defined by the guidelines of Regulatory Guide 1.115, we conclude that the upgraded system adequately meets the requirements in effect at the time of the license.*

Internally Generated Missiles

By letter dated December 8, 1982, we requested the licensee to provide information regarding the protection afforded the AFWS against the effects of internally-generated missiles. At a meeting on February 9, 1983, the licensee stated that an evaluation of internally generated missiles had never been performed for the Rancho Seco AFWS. Subsequently, in a letter dated June 3, 1983, the licensee stated that protection from internally generated missiles will be addressed by a walkdown of the AFWS trains to identify AFWS components and piping which, due to their location or their proximity of one train to the other and lack of intervening structures, could be susceptible to a single internally generated missile. At such locations, potential missiles will be evaluated and where a single missile could incapacitate both AFWS trains, missile protection will be provided. In a letter dated June 21, 1983, the licensee reiterated his position that the AFWS will only be protected against missile sources that can damage both redundant AFWS trains simultaneously. The licensee's position is not in accordance with current guidelines for internally generated missiles as discussed in Standard Review Plan Sections 3.5.1.1 and 3.5.1.2. These guidelines state that the effects of missiles should be evaluated in conjunction with a single active failure. Consequently, when the missile source is a nonsafety-related system, the effect of missile impact on one train of a safety-related

*As discussed in Section II.C of this report, the staff is currently developing a multi-plant position relative to AFWS reliability requirements for all operating reactors which could lead to modifications to increase system reliability for Ranch Seco.

Tornado Missiles

By letter dated January 14, 1983, the licensee stated that tornado missiles are not part of the design basis for the Rancho Seco plant. The licensee's Final Safety Analysis Report (FSAR) notes that the AFW pumps are protected against missiles generated by 175 mph winds and the piping and CST are protected against missiles generated by 101 mph winds. These winds are not in accordance with the wind speeds specified in current guidelines (see above). In addition, the missiles used for the FSAR analysis do not meet the current guidelines in Standard Review Plan Section 3.5.1.4. However, the Rancho Seco AFWS meets tornado missile protection criteria in effect at the time the plant was licensed.

In addition, a recent analysis of the CST indicates that this essential component is adequately protected against tornado missiles. By letters dated June 3 and June 21, 1983, the licensee stated that the lower 8 feet of the CST, being thicker than the upper portion, could withstand missiles generated by a 175 mph wind. The missile spectrum used in the reanalysis included the missiles defined in Standard Review Plan Section 3.5.1.4. Therefore, missile penetration of the lower 8 feet of the CST is not expected for local wind speeds up to 175 mph. As noted above, the probability of occurrence of a 175 mph wind is 10^{-6} per year. The protected 8 feet of CST water, in conjunction with low level and low-low level alarms for the CST, ensures adequate time for operator action to switch over to an alternate source of AFW water. We, therefore, conclude that the CST meets the requirements of General Design Criterion 4 with respect to wind generated missiles.

c. The AFWS is not used for startup and normal shutdown; therefore, it is considered a moderate energy system for pipe breaks in the AFWS. The only high-energy piping in the system is located sphere via the atmospheric dump valves. The turbine-driven pump receives main steam from connections to both main steam lines upstream of the turbine stop valves which serve as main steam isolation valves at Rancho Seco. The AFWS steam supply lines are six-inch lines each containing a check valve, a locked open manual valve and a normally open AC motor operated valve. Downstream of the motor-operated valves, the AFWS steam supply lines connect to provide a common supply to the AFW pump turbine. The common steam supply line contains a normally-closed DC motor-operated valve which opens on an emergency feedwater initiation signal. The motor-driven AFW pump also starts automatically on the emergency feedwater initiation signal. The motor driven pump will be modified to provide automatic loading on a diesel generator powered emergency bus on loss of offsite power. All of the valves associated with each pump train are normally open with the exception of four normally closed isolation valves. In each train, there will be one DC motor-operated isolation valve and one AC motor-operated isolation valve in parallel piping paths. A normally open air-operated valve will be provided in series with the DC motor operated valve and a normally open solenoid valve will be provided in series with the AC motor-operated valve. In the AFW system description (Revision 3) transmitted by the April 28, 1983 letter, the licensee stated that both motor-operated isolation valves will open on AFWS initiation and the flow to each steam generator will be controlled by the air-operated and solenoid flow control valves. The licensee will provide power to the flow control valves from two separate battery-backed buses. In addition, the air-operated valves will be equipped with seismic Category I air accumulators which will enable the valves to be operated for up to two hours following a loss of plant air supply to the valves. Final drawings for the emergency feedwater instrumentation and control (EFIC) system will be provided by the licensee no sooner than January 1984. Pending review of the final EFIC system drawings and verification of be provided with position indication in the control room. Because normally shut manual valves isolate the AFWS from nonessential systems, isolability of the AFWS is not jeopardized by active valve failure. Thus, adequate feedwater will be assured in the event of a postulated design basis accident concurrent with a single failure. We, therefore, conclude that the AFWS meets the requirements of General Design Criterion 44 with respect to the single failure criterion.

f. AFW Train A pump, P-318, is a combination turbine-driven and motor-driven pump with both a turbine and electric motor on a common shaft. AFW Train B pump, P-319, is a motor-driven pump with the same rated capacity as the Train A pump. The turbine-driven pump train provides a diverse means of assuring feedwater supply to the steam generator independent of all offsite or onsite AC power sources for at least two hours. The pump and turbine are not dependent on secondary support systems. The bearings on the pump and turbine are lubricated by slinging oil from reservoirs near the bearings. Lube oil cooling is accomplished by heat ransfer to the pumped fluid. In the AFW system description (Rev.-3) transmitted by the April 23, 1983 letter, the licensee stated that automatic actuation and control of the turbine train will be provided with battery-backed DC power. The steam admission valve to the AFW pump turbine is a DC motor-operated valve. The control power to the flow control valves in each pump train will be from redundant battery-backed buses. Control air to the air-operated flow control valves will be ensured by safety-related nitrogen accumulators. As discussed in Section I.B.2.d of this report, EFIC system drawings will be provided later by the licensee. Therefore, pending staff review of the final EFIC system drawings and verification of the adequacy of the EFIC system to automatically initiate and control the AFW flow with no AC power available, we conclude that the AFWS meets the power diversity position of BTP ASB 10-1.

i. In the Revision 3 AFW system description, the licensee stated that the AFWS function will be initiated automatically in the event of a main feedwater or main steam line rupture. Both AFW pumps will automatically start and steam generator level will be automatically controlled for the condition where main feedwater line and steam line ruptures depressurize the steam generators. Automatic isolation of AFW flow to a leaking steam generator will be provided, and a steam line break or main feedwater line break that depressurizes a steam generator will cause isolation of the main feedwater lines on the depressurized steam generator. If isolation of the steam generator main feed line does not isolate the break, AFW flow will be isolated from the leaking steam generator so that AFW flow will be provided only to the intact steam generator. By letter dated June 21, 1983, the licensee verified that the design will ensure that no single active failure in the upgraded AFWS design will prevent AFW flow from being supplied to the intact steam generator or allow AFW flow to be supplied to the leaking steam generator. However, a main steam rupture with failure of a single turbine stop valve could result in blowdown of both steam generators with consequent AFW pump runout. By letter dated November 3, 1982, the licensee was requested to evaluate this matter and propose a solution. A licensee response is expected in late 1983. This matter is being resolved as part of Multiplant Action B-69, "Main Steam Line Break with Continued Feedwater Addition," and is beyond the scope of this AFWS evaluation. The results of the B-69 evaluation could result in the need for further AFWS modifications and will, therefore, be evaluated for their impact on the AFWS. Until Multiplant Action B-69 is resolved we cannot conclude that the AFWS meets the requirements of General Design Criterion 44 with respect to its ability to transfer heat under accident conditions and provide isolation to assure system function.

j. Each AFW pump is designed to provide 100% of the flow for residual heat removal over the entire range of accidents as

PART II

INTRODUCTION AND BACKGROUND

The Three Mile Island Unit 2 (TMI-2) accident and subsequent investigations and studies highlighted the importance of the Auxiliary Feedwater System (AFWS) in the mitigation of transients and accidents. As part of our assessment of the TMI-2 accident and related implications for operating plants, we evaluated the AFW systems for all operating plants having nuclear steam supply systems (NSSS) designed by Westinghouse (NUREG-0611) or Combustion Engineering (NUREG-0635). Our evaluations of these system designs are contained in the NUREGs along with our recommendations. The objectives of the evaluation were to: (1) identify necessary changes in AFW system design or related procedures of these plants, and (2) to identify other system characteristics of the AFW systems which, on a long term basis, may require system modifications. To accomplish these objectivas, we:

(1) Reviewed plant specific AFW system designs in light of current regulatory requirements (SRP) and,

(2) Assessed the relative reliability of the AFW systems under various loss of feedwater transients (one of which was the initiating event of TMI-2) and other postulated failure conditions by determing the potential for AFW system failure due to common causes, single point vulnerabilities, and human error.

We have applied the generic results and recommendations of the above described review to the Rancho Seco auxiliary feedwater system (AFWS) design. The detailed reliability analyses submitted by the licensee were also evaluated. We also evaluated the licensee's design basis for AFWS flow requirements.

Section A of Part II is our evaluation of the present AFWS against our generic short-term recommendations. Section B is our evaluation of the proposed AFWS upgrade design against our generic long-term recommendations. Section C is our evaluation of the reliability analysis provided by the licensee for the

These modifications were approved by the staff and issued by letter dated March 27, 1981. We conclude that the Technical Specifications are in compliance with our recommendations and are, therefore, acceptable.

3. Recommendation GS-3 - "The licensee has stated that it throttles AFW system flow to avoid water hammer. The licensee should reexamine the practice of throttling AFW system flow to avoid water hammer. The licensee should verify that the AFW system will supply on demand sufficient initial flow to the necessary steam generators to assure adequate decay heat removal following loss of main feedwater flow and reactor trip from 100% power. In cases where this reevaluation results in an increase in initial AFW system flow, the licensee should provide sufficient information to demonstrate that the required initial AFW system will not result in plant damage due to water hammer."

By letter dated April 29, 1982, the licensee transmitted a Licensee Event Report, which reported damage to the AFW header in both steam generators. At a meeting on June 24, 1982, the licensee and the Babcock and Wilcox Company presented their plan to retire-in-place the existing internal AFW header and to install an external AFW header on each of the two steam generators. The new design is a modified design of the external AFW header used at several other Babcock and Wilcox designed plants. Details of the proposed design modifications were provided by a licensee letter dated August 3, 1982, and approved by the staff in a letter dated August 19, 1982. By letter dated August 13, 1982, the licensee committed to perform a water hammer test after installation of the new header arrangement. The water hammer test was performed and it was verified that no water hammer occurred. Therefore, we conclude that the design is acceptable.

4. Recommendation GS-4 - "Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant

only locked-open valves and check valves. Thus, the primary AFWS water source is always aligned to provide suction flow to the pumps. The CST is designed to seismic Category I criteria and would be available in the event of an earthquake. Although the CST is not protected against tornado wind speeds in accordance with current guidelines, it is protected against wind damage and against missiles generated by winds of speeds up to 175 miles per hour. The probability of occurrence of wind speeds greater than 175 miles per hour is sufficiently low that catastrophic loss of the CST is not expected. Therefore, we conclude that the AFW pumps are adequately protected against loss of suction flow in accordance with this short-term requirement (see also Section II.B.4 of this report).

5. Recommendation GS-5 - "The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train, independent of any alternating current power source. If manual AFW system initiation or flow control is required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling of the lube oil for the turbine-driven pump bearings may be dependent on alternating current power, design or procedural changes shall be made to eliminate this dependency as soon as possible. Until this is done, the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in a manual on-off mode until alternating current power is restored. Adequate lighting powered by direct current power sources and communications at local stations should also be provided if manual initiation and control of the AFW system is needed. (See Recommendation GL-3 for the longer-term resolution of this concern.)"

On loss of all AC power, the steam turbine-driven AFW pump will start as a result of the DC powered steam inlet valve opening. The staff by letter dated March 27, 1981. We conclude that the Technical Specifications are in compliance with our recommendations and are, therefore, acceptable.

7. Recommendation GS-7 - "The licensee should verify that the automatic start AFW system signals and associated circuitry are safety grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-5.

   (1) The design should provide for the automatic initiation of the auxiliary feedwater system flow.

   (2) The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

   (3) Testability of the initiation signals and circuits shall be a feature of the design.

   (4) The initiation signals and circuits should be powered from the emergency buses.

   (5) Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.

   (6) The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

the motor-driven AFW pump onto the existing diesel generator-supplied emergency bus. By letter dated October 5, 1982, the licensee provided additional information to verify the acceptability of this interim action. This additional information has been reviewed by the staff and the licensee's proposal has been found to be acceptable inasmuch as the licensee provided adequate assurance that the AFW pump can be automatically loaded on the diesel generator with no adverse effect to the diesel generator or safety loads.

Therefore, we conclude, that the existing initiation and control system is in conformance with the guidelines of Recommendation GS-7 and is, therefore, acceptable.

8. Additional Short Term Recommendation 1 - "The licensee should provide redundant level indication and low level alarms in the control room for the AFW system primary water supply, to allow the operator to anticipate the need to makeup water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

   For long-term, the level indication and alarms must be safety grade with redundant sensors, detectors readouts, and alarms all the way from the CST to control room, including power supplies. Circuitry equipment and power supplies are required to be Class 1E."

   As indicated in the licensee letter of December 17, 1979, the condensate storage tank level is indicated in the control room. Previously, in a letter dated June 27, 1979, the staff noted that condensate storage tank low level alarms in the control room provide 40 minutes for operator action to transfer to an alternative water source. The staff concluded that the alarms and operating procedures were adequate to assure timely transfer to an alternative source when the condensate storage tank supply is being depleted.

10. Additional Short-Term Recommendation 3 - "The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

'Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room. The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9.'"

By letter dated December 17, 1979, the licensee indicated that the existing AFWS obtains an indication of AFW flow from clamp-on ultrasonic flow meters on each AFWS train. Previously, by letter dated October 18, 1979, the licensee committee is to provide a safety-grade flow indication system in the upgraded AFWS design. In a letter dated February 26, 1980, the staff required that a procedure for performing channel functional tests of the existing AFW flow indication system be established and implemented until the upgraded AFW flow indication system is installed. By letter dated November 30, 1981, the licensee verified that these procedures are in place. By letter dated September 8, 1981, the licensee provided the design description of the upgraded AFW flow indication system. The long-term design modifications were reviewed by the staff and found acceptable. The safety-grade flow indication system will be added during the refueling outage which began in February 1983. Therefore, we conclude that the AFW flow indication system is in compliance with this recommendation and is, therefore, acceptable.

11. Additional Short-Term Recommendation 4 - "Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train, and there is only one remaining AFW train available for operation, should propose Technical Specifications to provide that a dedicated individual who is in communication

The existing AFWS contains a control-grade automatic initiation system. The proposed upgrade of this system to a safety-grade system is evaluated in Section II.B.5, "Recommendations GL-5," of this report.

2. Recommendation GL-2 - "Licensees with plant designs in which the primary AFW system water supply passes through valves in a single flow path, but the alternate AFW system water supplies connect to the AFW system pump suction piping downstream of the above valve(s); should: (a) install redundant valves parallel to the above valve(s) or (b) provide automatic opening of the valve(s) from the alternate water supply upon low pump suction pressure."

   Each AFW pump is provided with separate suction piping to the condensate storage tank. The suction piping to each pump has two locked-open manual valves and a check valve. We conclude that the AFWS suction piping is acceptable.

3. Recommendation GL-3 - "At least one AFW system pump and its associated flow path and essential instrumentation should automatically initiate AFW system flow and be capable of being operated independently of any AC power source for at least two hours. Conversion of DC power to AC power is acceptable."

   The capability of the AFWS to operate for two hours in the event of a loss of all AC power is discussed in detail in Sections I.B.2.d and I.B.2.f of this report. By letter dated January 14, 1983, the licensee verified that the flow control valves in the upgraded design will be capable of being operated for two hours independent of AC power. Therefore, we conclude that the AFWS is in compliance with this recommendation and is, therefore, acceptable.

4. Recommendation GL-4 - "Licensees having plants with unprotected normal AFW water supplies should evaluate the design of their AFW systems to determine if automatic protection of the pumps is necessary following a seismic event or a tornado. The time available

licensee letter dated September 8, 1981. Subsequent to the staff review of the Revision 1 design, a Revision 3 AFW system design with further modifications to the EFIC system, was submitted by licensee letter dated April 28, 1983. The safety-related auto-initiation system, as described in the Revision 3 design description, will be installed during the refueling outage which began in February 1983. The staff will perform a post-implementation review of the auto-initiation system upon receipt of final system drawings and design description. Pending approval of the final design, we find the safety-related automatic initiation and control system to be acceptable.

C. Auxiliary Feedwater System Reliability Evaluation

In accordance with the requirements of NUREG-0660 and NUREG-0737, the licensee has performed a reliability study of the upgraded Rancho Seco auxiliary feedwater system (AFWS). The design description of the upgraded AFWS and the reliability study for the upgraded design were provided by the licensee in letters dated September 8, 1981 and January 18, 1982, respectively.

The licensee's reliability study was performed in a manner similar to that employed in the NUREG-0611 study using generic failure rate data as modified by Rancho Seco experience. The NUREG-0611 study considered the following three transient conditions for determining the reliability of the AFWS:

1. LMFW - Loss of Main Feedwater

2. LOOP - Loss of Offsite Power/Loss of Main Feedwater

3. LOAC - Loss of all AC Power/Loss of Main Feedwater

The licensee's evaluation does not present a separate value of AFWS unavailability for each of the three transients, but rather reports

TABLE 1: AFWS UNAVAILABILITY FOR THREE TRANSIENTS CASES

		BNL Estimate, Per Demand No Operator Recovery	BNL Estimate, Per Demand Operator Recovery	Licensee Estimate, Per Demand, Operator Recovery
1.	LMFW	7.6×10^{-4}	2.6×10^{-4}	1.0×10^{-4}
2.	LOOP	1.5×10^{-3}	5×10^{-4}	3.6×10^{-4}
3.	LOAC	2.7×10^{-2}	1.3×10^{-2}	1.6×10^{-2}

TABLE 2

Dominant Contributors to AFWS Unavailability

Dual (turbine/motor) driven pump train

BNL Estimate	Licensee Estimate	Comments
Hardware^Δ		
5×10^{-4}	5×10^{-4}	Turbine/motor pump fails to start.
1×10^{-3}	5×10^{-3}	Steam turbine driver fails to start.
3.1×10^{-3}	3.9×10^{-3}	Steam admission valve fails to operate.
7×10^{-3}	$\sim3\times10^{-4}$	Actuation signal failure, per train.
5×10^{-3}	*	Steam admission valve left disabled after maintenance.
3.6×10^{-2}	*	Diesel generator B fails to start because of hardware failure (3×10^{-2}) or in maintenance (6×10^{-3}).
1×10^{-3}	8×10^{-4}	Battery fails in the loss of all AC power transient.
Maintenance		
5.8×10^{-3}	1.15×10^{-3}	Turbine pump maintenance (motor driven is not considered here).
2.1×10^{-3}		Steam admission valve maintenance.
2.1×10^{-3}		Four parallel valves are under maintanance.

*Events did not appear in the licensee fault trees.
ΔAll failure rates are in per demand basis.

TABLE 3

Dominant Failure Models

BNL Analysis	Licensee Analysis
A. Loss of Main Feedwater (LMFW) Case	
1. One pump under maintenance and hardware failure of second pump.	1. The motor driven pump (MDP) unavailability due to loss of off-site power and diesel generator A failure.
2. Failure of both actuation trains: control logic A (EFIC-A) actuates MDP and logic B (EFIC-B) actuates DDP (dual drive pump).	2. The dual drive pump (DDP) unavailability due to steam admission valve failure or hardware failure of the turbine driver.
3. Leakage from test line valve FWS-X5 can divert AFW flow and potentially dry out the steam generators.	3. Valve FWS-X5 fails to close after the test.
4. Miscalibration of all four steam generator level setpoints by the operator.	4. Miscalibration of all four steam generator level setpoints.
5. Hardware failure of both DDP and MDP.	5. Valves FWS-045 and FWS-046 fail to reopen after pump maintenance
	6. The Feed-only-good-generator (FOGG) Logic fails due to miscalibration.
B. Loss of Off-site Power (LOOP) Case	
1. Diesel generator A failure or being maintained which disables MDP train while DDP train is unavailable due to maintenance or the steam admission valve failure.	
2. Same as A.3, A.4.	
C. Loss of All AC Power	
1. Actuation channel B fails.	
2. Turbine driven pump being maintained.	
3. Steam admission valve fails to open.	
4. Local control to steam admission valve fails.	

that, until a staff position is developed regarding the need for further modifications to improve AFWS reliability, operation of Rancho Seco, with the proposed upgraded AFWS design, is acceptable.

D. Auxiliary Feedwater Flow Requirements

The design basis event originally used for sizing the auxiliary feedwater system (AFWS) is loss of main feedwater (LMFW) with a concurrent loss of offsite power (LOOP), and subsequent loss of the reactor coolant pumps. The pertinent parameters for this accident relative to the AFWS are design flowrate and required time to full AFW flow. The design values which resulted from this original (FSAR) analysis are 780 gpm deliverable to the steam generator within 40 seconds of the initiation signal. The 40 second time was chosen to allow the AFWS to inject feedwater and begin increasing steam generator level to the 50% operating range level required for natural circulation prior to completion of the reactor coolant pump coastdown. The design flowrate was selected to be equal to or greater than the decay heat generation rate at 40 seconds. As described in the licensee submittal of September 8, 1981, each AFW pump has a rated capacity of 840 gpm at 1150 psig with a normal recirculation flow of 60 gpm; thus the net flow rate to the steam generators is 780 gpm.

Following the Three Mile Island Accident, the licensee provided an additional flow rate analysis which had been provided to the licensee in a letter from the Babcock and Wilcox Company (B&W) dated May 16, 1979. This new B&W analysis indicated that at 36 seconds after reactor trip, an AFW flow rate of 760 gpm would be adequate to remove decay heat, and at 40 seconds the minimum required flow rate would decrease to 748 gpm. By letter dated February 26, 1980 and at a meeting with the licensee on February 9, 1983, the staff requested additional information from the licensee to verify that the criteria used to establish minimum AFW flow requirements would assure adequate decay heat removal. The licensee responded to this request by providing the "Rancho Seco Auxiliary Feedwater Flow Evaluation" in a letter dated November 30, 1981 and a revised minimum flow analysis by letter dated February 18, 1983. Our evaluation of the February 18, 1983 revised minimum flow analysis is provided below.

Part III

CONCLUSION

Although substantial progress has been made in the staff's evaluation of the design and operation of the existing and proposed upgrade design of the Rancho Seco auxiliary feedwater system, we cannot complete our review until the licensee provides the required additional information identified in Part I above. While the upgraded design will not meet the current reliability guidelines, the staff has not yet fully resolved the need for additional modifications to increase system reliability. A staff position in this regard is being developed for all plants whose AFWS reliability does not meet current guidelines. Should Rancho Seco be required to implement modifications to increase system reliability such as adding a third train (pump), this solution could aid in resolution of deficiencies relative to other current guidelines noted in Part I. Until these matters are resolved, it is the staff's judgement that operation with the proposed upgraded AFWS is acceptable. Also, since many of the concerns associated with the existing system have been resolved and some of the proposed system upgrades will be implemented during the current refueling outage,* it is the staff's judgment that interim operation with the existing AFWS is acceptable. The licensee should, however, provide the information requested in Part I of this report. The information needed to complete our review is identified on page 9 (internally generated missiles) above.

*The safety grade AFWS initiation system and a safety grade flow indication system will be installed during the refueling outage which began in February 1983.

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

GR

September 2, 1983

Docket No. 50-366

Licensee: Georgia Power Company

Facility: Hatch Unit 2

SUBJECT: SUMMARY OF JULY 13, 1983 MEETING WITH GEORGIA POWER COMPANY CONCERNING HIGH WATER LEVEL TRIP AND TURBINE BYPASS SYSTEM TECHNICAL SPECIFICATIONS

The staff met with Georgia Power Company (GPC) representatives in Bethesda, Maryland, on July 13, 1983 to hear and discuss GPC's arguments concerning the staff's prior request for Technical Specifications related to assuring operability of the Hatch Unit 2 high water level trip and turbine bypass systems. GPC had indicated in its earlier response to the staff request for Technical Specifications on this subject that it believed the staff's request concerned generic issues that should have been reviewed under the formal NRC system for reviewing generic requirements.

GPC representatives, together with their consultants from General Electric, expressed their views that the failures they were asked to assume (i.e. to have both the flow controller and the turbine bypass valve fail), as the basis for requiring Technical Specifications, were so improbable that they should not be considered in the moderate frequency category.

GPC's General Electric Company consultants contended that:

1. Staff approved methodology in a generic report referred to as GESTAR was used in performing the analyses of the failure of the flow controller and turbine bypass.

2. The probability that both the feedwater flow controller and the turbine bypass system would fail is extremely low and is lower than values assumed for transients.

3. The staff's Standard Review Plan does not explicitly require safety-grade equipment for mitigation of transients. The staff is treating this case as if it involved an accident rather than a transient.

4. The equipment in question was not designed for or meant to have the type of surveillance testing that the staff has requested be performed on it. They don't know how they would perform the surveillance testing if it were required. The requested Technical Specification would add to the cost and complexity of testing.

5. The reliability of the equipment in question is such that it will work in a transient (without the proposed surveillance testing).

6. It is a new staff requirement to require Technical Specifications to assure operability of nonsafety-grade equipment.

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The staff explained that it considers that GPC is assuming credit for equipment that is not classified as safety related to mitigate the consequences of an operational transient. The staff also stated that General Design Criterion (GDC) 1 requires that equipment important to safety be designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety function to be performed. And since the turbine bypass system and the high reactor water level trip are not classified as safety related, the staff has no assurance as to the reliability of the equipment and therefore considers that GDC 1 is not satisfied. The staff considers the proposed Technical Specifications and surveillance requirements sufficient to satisfy GDC 1.

The staff also stated that it is fairly clear that the issue in question is a generic issue. And it agreed to hold its previously stated requirement for these Technical Specificaitons in abeyance until it has considered the GPC arguments.

A list of meeting attendees is enclosed.

GW Rivenbark

George W. Rivenbark
Operating Reactors Branch #4
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

MEETING SUMMARY DISTRIBUTION

Licensee: Georgia Power Company

*Copies also sent to those people on service (cc) list for subject plant(s).

Docket File
NRC PDR
L PDR
ORB#4 Rdg
Project Manager -GRivenbark
JStolz
BGrimes (Emerg. Preparedness only)

OELD
NSIC
ELJordan, IE
JMTaylor, IE
ACRS (10)

NRC Meeting Participants:

RHouston
TCollins
DVassallo
TCox
WKane

July 13, 1983

MEETING WITH GEORGIA POWER COMPANY CONCERNING HIGH WATER LEVEL TRIP AND TURBINE BYPASS SYSTEM TECHNICAL SPECIFICATIONS

List of Attendees

Name	Organization
G. Rivenbark	NRC/DL ORB #4
R. W. Houston	NRC/NRR/DSI
Tim Collins	NRC/NRR/DSI/RSB
Ken Turnage	Southern Company Services
Larry K. Mathews	Southern Company Services
J. S. Charnley	General Electric
R. L. Wagne	General Electric
Tom Cox	EDO/DEDROGR
W. F. Kane	EDO/DEDROGR
J. D. Heidt	General Electric
D. L. Townley	GPC
R. D. Baker	GPC
D. B. Vassallo	NRC/DL

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

October 28, 1983

Docket No. 50-366

Mr. J. T. Beckham, Jr.
Vice President, Nuclear Generation
Georgia Power Company
P. O. Box 4545
Atlanta, Georgia 30302

Dear Mr. Beckham:

In our Safety Evaluation supporting Amendment No. 33 to the Hatch Unit 2 Operating License, we indicated that the Georgia Power Company (GPC) disagreed with the staff position that Technical Specifications for surveillance of the high water level (Level 8) turbine trip and the turbine bypass systems are required in order to assure operability of these trip and turbine bypass systems when required to mitigate a transient involving failure of the feedwater flow controller. We also stated that we would defer implementation of such Technical Specifications for 60 days following startup in order to allow time for further discussion with GPC on this subject. We subsequently met with GPC on July 13, 1983 and discussed GPC's arguments concerning the need for these Technical Specifications. At that time, we informed GPC that it appeared to us that the issue in question is generic and that we would hold our previously stated requirements for these Technical Specifications in abeyance until we had further considered the GPC arguments discussed at the meeting.

We have now concluded that this subject should be treated as a generic issue, and we plan to handle it in accordance with our internal procedures for dealing with such issues. We have also determined, based on preliminary analysis, that the risk of operating Hatch Unit 2 without Technical Specifications concerning surveillance of highwater level turbine trip or turbine bypass systems until the generic issue is resolved is small.

Accordingly, we will not require implementation, at this time, of any Technical Specifications on this issue for the Hatch Plant. We will inform you of the results of our consideration of this issue when it is finally resolved.

Sincerely,

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

cc: See next page

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

Project Manager

July 11, 1983

Docket No. 50-321

MEMORANDUM FOR: John F. Stolz, Chief, Operating Reactors Branch #4, DL

FROM: George Rivenbark, Project Manager, Operating Reactors Branch #4, DL

SUBJECT: FORTHCOMING MEETING WITH GEORGIA POWER COMPANY (GPC)

Time & Date: Wednesday, July 13, 1983
1:00pm-4:00pm

Location: Maryland National Bank Building, Rm. 1713
Bethesda, Maryland

Purpose: Discuss Georgia Power Company's arguments relative to the staffs request for Technical Specifications to assure the operability of the Hatch Unit 2 high water level trip and turbine bypass systems.

Requested Participants: NRC-GRivenbark, DVassallo, WHouston, WHodges, TCollins.

GPC-RBaker, and consultants, including GE.

George Rivenbark

George Rivenbark, Project Manager
Operating Reactors Branch #4, DL

cc:
See next page

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MEETING NOTICE DISTRIBUTION

OPERATING REACTORS BRANCH #4, DIVISION OF LICENSING

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Resident Inspector

NRC Meeting Participants:
DVassallo
WHouston
WHodges
TCollins

Tom Cox

MNBD 6113

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

JULY 6 1983

Docket No. 50-366

Mr. J. T. Beckham, Jr.
Vice President - Nuclear Generation
Georgia Power Company
P. O. Box 4545
Atlanta, Georgia 30302

Dear Mr. Beckham:

By letter dated May 12, 1983, we requested that you submit Technical Specifications (TSs) for Hatch Unit 2 to assure the operability of the high water level trip and turbine bypass systems that you assumed operable in your cycle 4 reload analysis. Your letter of May 26, 1983, pointed out a number of objections to our request for these TSs that you believed to be new staff requirements involving generic issues.

We want to assure you that we did not intentionally withhold our request for these TSs until a short time before your scheduled restart of Hatch Unit 2. Our delay in submitting this request was solely due to an oversight on our part.

We agree with your concern that a short period of time was provided for you to present your arguments and discuss this request with the staff. Therefore, we have decided to defer implementation of these TSs for a period of 60 days following startup for Cycle 4 operation. This should allow you and the staff time to consider and discuss the objections that you raised in your May 26, 1983 letter. Based on discussions between George Rivenbark of our staff and Ray Baker of Georgia Power Company, we understand that a tentative date of July 13, 1983 has been selected for a meeting to discuss this matter.

Sincerely,

DVassallo for

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

cc: See next page

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Georgia Power Company
333 Piedmont Avenue
Atlanta, Georgia 30308
Telephone 404 526-7020

Mailing Address
Post Office Box 4545
Atlanta, Georgia 30302

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Georgia Power
the southern electric system

J. T. Beckham, Jr.
Vice President and General Manager
Nuclear Generation

NED-83-303

May 26, 1983

Director of Nuclear Reactor Regulation
Attention: Mr. John F. Stolz, Chief
Operating Reactors Branch No. 4
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

NRC DOCKET 50-366
OPERATING LICENSE NPF-5
EDWIN I. HATCH NUCLEAR PLANT UNIT 2
RESPONSE TO REQUEST FOR TECHNICAL SPECIFICATIONS

Gentlemen:

Georgia Power Company (GPC) has received your letter of May 12, 1983, which requests that Georgia Power Company submit additional Technical Specifications prior to restart of Plant Hatch Unit 2 from the current refueling outage. Consequently, GPC amends its submittal of March 30, 1983, to include the attached Technical Specifications which propose additional surveillance requirements on the high reactor water level trip function (i.e., for main turbine and feedwater turbines) and which specify surveillance testing of the main turbine bypass valves.

GPC does not agree with the conclusion presented in your letter of May 12, 1983, regarding the need for additional Technical Specifications. More importantly, we are concerned over the issuance of this letter by the NRC staff for two reasons: 1) the sense of urgency implied by attaching this requirement to our cycle 4 reload and formally informing GPC of the request only 17 days before the originally scheduled startup date is not supported by an appropriate technical basis; 2) the NRC staff is apparently avoiding the established generic issues review process by withholding reload licenses on a plant-by-plant basis until additional Technical Specifications are backfitted when the stated issue is an obvious generic concern.

First, we want to point out that the issues involved have been the subject of several conversations between GPC licensing personnel, the NRC Hatch Licensing Project Manager, and other NRC staff personnel. These conversations have occurred over a period of several months, and actually began during the latter part of the previous operating cycle. At that time we stated our position that the NRC proposed requirements were more appropriately discussed in a generic review since their implementation was being imposed generically and since the proposal was in variance with

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Georgia Power

accepted licensing bases. At that time we also requested that the NRC transmit the request to GPC in writing, so that a forum for discussion of the technical merit of the request could be established. A letter was not issued by the NRC, and Unit 2 continued to the end of cycle 3 operation and into the current refueling/maintenance outage.

Second, only 17 days prior to the originally scheduled startup date for cycle 4 operation, the NRC finally chose to issue the request with the additional arbitrary requirement that it be resolved prior to unit startup. GPC objects to being informed at such a late date that the reload license will be withheld until the requested Technical Specifications are accepted by GPC. Furthermore, the technical questions involved relate to pressurization transients which historically are not limiting at beginning of cycle operation. Clearly, in light of GPC's objections to its substance, the requirement to resolve this request prior to restart and without adequate time for GPC and NRC to mutually and fully discuss the issues is inappropriate.

Third, the procedure being followed by the NRC avoids the established generic issues review process. It is inappropriate to impose backfits lacking a finding of need for substantial additional public protection. The NRC staff conclusion expressed in the May 12, 1983, letter clearly rejects the assumptions previously accepted by NRC as stated in General Electric's Generic Reload Fuel Application licensing report (NEDE-24011-P). As such, this is an issue that should be addressed by the Committee to Review Generic Requirements (CRGR) so that the nuclear industry can comment on the staff conclusions. However, we understand that these requirements have previously been imposed on at least one other operating nuclear unit during a refueling outage and on two near term operating license (NTOL) units as they approached receipt of an operating license. The staff is apparently masking generic requirements as plant-specific modifications and issuing them without the formal analysis required by the Commission. By applying this requirement on a plant-by-plant basis as a particular nuclear plant is placed in a vulnerable position, a commitment can be easily extracted due to the threat of non-approval of a reload licensing package or an operating license. GPC objects to this procedure.

GPC is ready to discuss with the staff the reasons why additional Technical Specifications are not required. However, that issue should be properly dealt with by referring the subject to the CRGR. We propose a meeting in the near future (possibly June 1983) of representatives from the NRC, GE licensing, and GPC licensing staffs to discuss the issues expressed in this letter as well as the technical merits of the proposed specifications.

Georgia Power

In the meantime, GPC proposes additional Technical Specifications in compliance with your request. The Plant Review Board and Safety Review Board have reviewed these changes and have determined that the implementation of these changes does not constitute an unreviewed safety question because these changes merely establish LCOs and codify surveillances on equipment and/or systems that were previously being serviced by plant maintenance and surveillance procedures. Since there is no change to the plant and only a codifying of surveillances, the probability of occurrence and the consequences of an accident or malfunction of equipment important to safety are not icreased above those analyzed in the FSAR. The possibility of an accident or malfunction of a different type than analyzed in the FSAR does not result, nor is the margin of safety as defined in Technical Specifications reduced due to implementation of these changes. Because we desire to resolve the technical issues regarding this submittal, we request that these Technical Specifications be made applicable to cycle 4 operation only, pending resolution of this subject.

The proposed Technical Specifications have been determined to be an amendment, as requested, to a previous submittal. As such, no fee is required.

J. T. Beckham, Jr. states that he is Vice President of Georgia Power Company and is authorized to execute this oath on behalf of Georgia Power Company, and that to the best of his knowledge and belief the facts set forth in this letter are true.

GEORGIA POWER COMPANY

By: J T Beckham Jr
J. T. Beckham, Jr.

Sworn to and subscribed before me this 26th day of May, 1983.

Mae H. Battle
Notary Public, Georgia, State at Large
My Commission Expires Sept. 20, 1983
Notary Public

DLT/mb
Enclosure
xc: J. T. Beckham, Jr.
H. C. Nix, Jr.
J. P. O'Reilly (NRC- Region II)
Senior Resident Inspector
V. Stello
J. R. Tourtellotte

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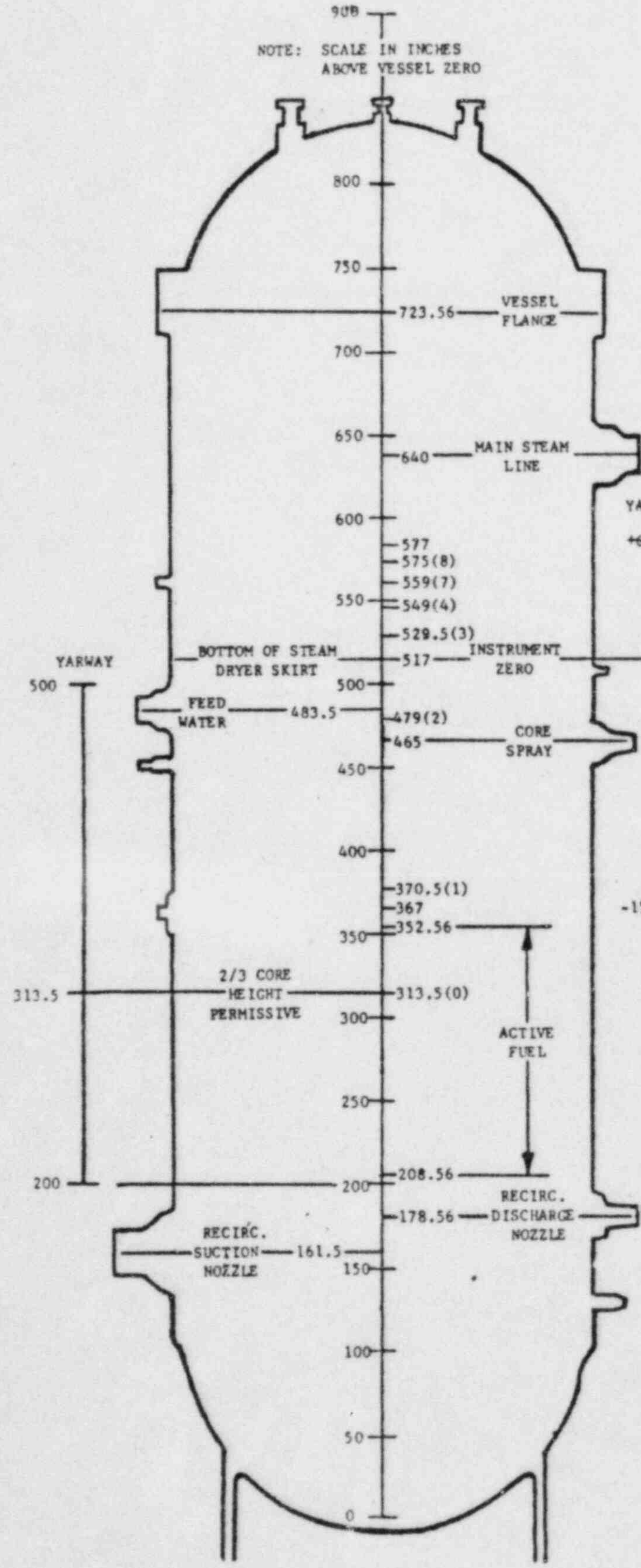
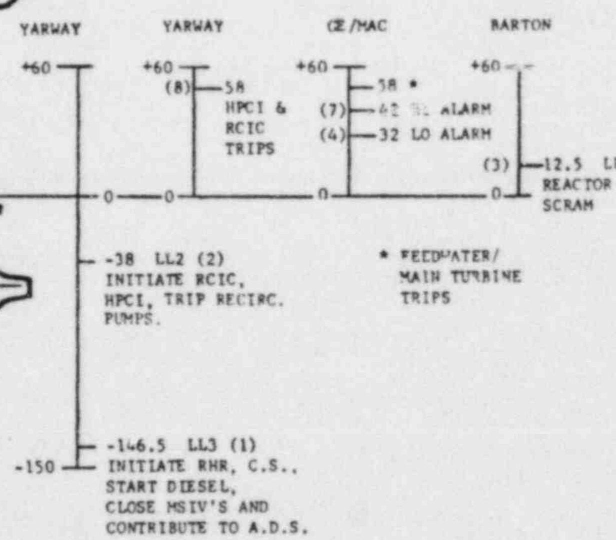
TABLE 3.3.9-1

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

FUNCTIONAL UNIT	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITIONS
a. Reactor Vessel Water Level-High, Level 8	2	1, when THERMAL POWER $\geq$ 25% RAT[E]D THERMAL POWER

WATER LEVEL NOMENCLATURE			
NO.	HEIGHT ABOVE VESSEL ZERO (INCHES)	READING	INSTRUMENT
(8)	575	+58	YARWAY
(7)	559	+42	GE/MAC
(4)	549	+32	GE/MAC
(3)	529.5	+12.5	BARTON (LL1)
(2)	479	-38	YARWAY (LL2)
(1)	370.5	-146.5	YARWAY (LL3)
(0)	313.5	+313.5	YARWAY





REACTOR VESSEL WATER LEVEL

Bases Figure B 3/4 3-1

3/4.7.9 MAIN TURBINE BYPASS SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.9 The main turbine bypass system shall be OPERABLE.

APPLICABILITY: CONDITION 1, when THERMAL POWER $\geq$ 25% RATED THERMAL POWER

ACTION: With the main turbine bypass system inoperable, restore the system to OPERABLE status within 2 hours or determine MCPR to be equal to or greater than the MCPR limit in Specifications 3.2.3 within one hour or take the ACTION required by that Specification.

SURVEILLANCE REQUIREMENTS

4.7.9 The main turbine bypass system shall be demonstrated OPERABLE at least once per:

a. 7 days by cycling each turbine bypass valve through at least one complete cycle of full travel, and

b. 18 months by:

1. Performing a system functional test which includes simulated automatic actuation and verifying that each automatic valve actuates to its correct position.

2. Demonstrating TURBINE BYPASS SYSTEM RESPONSE TIME to be less than or equal to 0.30 seconds.

ATTACHMENT 1

NRC DOCKET 50-366
OPERATING LICENSE NPF-5
EDWIN I. HATCH NUCLEAR PLANT UNIT 2
PROPOSAL FOR TECHNICAL SPECIFICATION CHANGES

The proposed change to the Technical Specification (Appendix A to Operating License NPF-5) would be incorporated as follows:

Remove Page	Insert Page
--	3/4 3-66
--	3/4 3-67
--	3/4 3-68
--	3/4 3-69
--	3/4 7-33
B 3/4 3-6	B 3/4 3-6

3/4.3.9 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.9 The feedwater/main turbine trip system actuation instrumentation channels shown in Table 3.3.9-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.9-2.

APPLICABILITY: Condition 1, when THERMAL POWER $\geq$ 25% RATED THERMAL POWER

ACTION:

With a feedwater/main turbine trip system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.9-2, declare the channel inoperable and either place the inoperable channel in the tripped condition until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value, or declare the associated system inoperable.

a. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 7 days or reduce thermal power $<$25% Rated thermal power within the next 4 hours.

b. With the number of OPERABLE channels two less than required by the Minimum OPERABLE Channels per Trip System requirement, restore at least one of the inoperable channels to OPERABLE status within 72 hours or reduce thermal power $<$ 25% rated thermal power within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.3.9.1 Each plant system 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 4.3.9.1-1.

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

TABLE 3.3.9-2

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
a. Reactor Vessel Water Level-High, Level 8	$\leq$ 58.0 Inches	$\leq$ 58.0 Inches

*See Bases Figure B 3/4 3-1.

TABLE 4.3.9.1-1 (Continued)

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
a. Reactor Vessel Water Level-High, Level 8	NA	M	R	1, when THERMAL POWER $\geq$ 25% RATED THERMAL POWER