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

NOV 14 1978

Docket Nos: 50-329 A
50-330

APPLICANT: Consumers Power Company

FACILITY: Midland Plant, Units 1 & 2

SUBJECT: SUMMARY OF AUGUST 20, 1978 MEETING ON STEAM LINE BREAK METHODOLOGY, COLD SHUTDOWN POSITION, AND SAFEGUARDS PUMP ROOM LEAKAGE

On August 20, 1978, the NRC staff met in Bethesda, Maryland with members of Consumers Power Company (CPCO), Bechtel Associates, and the Babcock & Wilcox (B&W) Company. Attendees are listed in Enclosure 1.

The purpose of the meeting was to discuss three matters which have been the subject of requests for additional information by the staff, and in which the staff does not agree with the position taken by the applicant.

1. Staff Position on Cold Shutdown Capability

   Request 211.35 of the staff's letter of March 15, 1978 questioned the capability of the Midland plants to be taken to a cold shutdown condition using only safety grade equipment, assuming only onsite or offsite power is available, and considering a single failure. CPCO stated that full implementation of this new position by the staff on Midland is not justified on a value-impact basis: CPCO finds that the advanced construction phase and estimated nine-months schedule delay would result in substantual costs, while the high stability of the Midland grid tends to diminish the benefits to be gained from such changes. A point-by-point response to the items of the staff request was presented, as summarized in the attached handout of the draft reply (Enclosure 2). CPCO stated that the formal reply would be submitted by amendment on or about October 20, 1978.

2. Safeguards Pump Room Filters

   Several staff requests for additional information have been made regarding the need for the addition of filters to the safeguards pump rooms for Midland Plant Units 1

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and 2. These include 010.32, 312.27, 312.41, 321.1 and 321.5, and indicate that the staff requires a safety grade filter system to control offsite doses resulting from postulated pump leakage after a LOCA.

An outline of the CPCO discussion is provided in Enclosure 3. CPCO stated that the Midland design provides two ventilation systems to the rooms, but only one is safety grade and no filters are included. The design provides a slight negative pressure in the rooms during normal operation, but not after an accident.

The applicants dose analysis assumed a design basis leakage from the safeguards pumps after 24 hours of 500 ml/min which is based upon pump seal tests by the Crane Packing Company which resulted in 29 ml/min. CPCO stated that the report on these tests would be submitted for NRC staff review. CPCO also assumed Iodine carryover assumptions consistent with Standard Review Plan 15.6.5 and noted that the pump release would be liquid below 200°F. The resulting dose contribution due to pump leakage was 16 rem to the thyroid at the LPZ. CPCO finds that these low results justify the omission of filters.

The staff stated it will consider the CPCO position further, and the assumed pump leak rate in particular, based upon review of the test report, and advise CPCO of its position during November, 1978.

3. Methodology for Main Steam Line Break Analyses

Staff request 222.1 questioned several aspects of the calculational methods and computer codes used for the main steam line break (MSLB) analyses in Section 15.1.5 of the Midland FSAR. A listing of these aspects and the draft response by CPCO are given in Enclosure 4.

The analyses were performed with the TRAP-2, RADAR, and PDQ-07 computer codes. The staff noted that TRAP-2 is being reviewed as part of the Topical Report Review Program (B&W report BAW-10128), and that the code had been only conditionally approved by B&W. The RADAR and PDQ-07 codes have been previously approved by the staff.

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CPCO stated that the analyses of the MSLB accident in the Midland FSAR were performed by B&W. B&W stated that the analyses considers the effects of a stuck control rod on the gross core shutdown margin, however stuck rod effects on localized physics or thermal performance are not considered to be an analyses requirement. B&W stated that this position is based upon its interpretation of GDC-26 and -27 as summarized in Enclosure 4. B&W is preparing a topical report for staff review of its MSLB methods and anticipates submittal in the first quarter of 1979. B&W has performed three dimensional calculations on an earlier docket to determine the effect of the stuck rod and finds that the less sophisticated approach used for Midland is more conservative.

The staff stated that it will review this B&W position on stuck rod effects and advise CPCO of its position in late-October 1978.

Darl Hood

Darl Hood, Project Manager
Light Water Reactors Branch No. 4
Division of Project Management

Enclosures:
As stated

cc: See next page

Consumers Power Company

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Enclosure 1

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Attendees List

August 20, 1978

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S. Newberry
H. Daniels
O. Chopra
V. Benaroya
G. Mazetis
S. Salah
Z. Rosztoczy
P. Norian

B&W
D. Newton
J. Howard
J. Burrow
R. Reed

Consumers Power
M. Salerno
J. Zabritski

Bechtel
M. Pratt
W. Skelley
K. Prasad
J. Clements

Enclosure 2

(11 pages)

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RESPONSE TO NRC QUESTION 211.35
FOR MIDLAND UNITS 1 AND 2

The Midland design does not incorporate the ability to be taken to the cold shutdown condition using only safety-grade equipment, assuming only on-site or off-site power is available, and considering a single failure. The Midland design basis provides for the ability to achieve by safety-grade means the hot shutdown condition as described in Section 7.4 of the FSAR. As discussed in the response to Request 110.16, hot shutdown provides for an extremely stable and safe condition at which the plant can be maintained until an eventual cooldown can proceed. The modifications required to provide the capability to reach cold shutdown under the improbable conditions detailed in this question can not be justified by any tangible increase in safety. To support the contention that upgrading to provide for the capability to achieve safety-grade cold shutdown provides little additional benefit, B&W has performed brief studies to show that the risk to the public health and safety is changed insignificantly by addition of a safety-grade cold shutdown capability. This study expresses risk in terms of changes in manrem dose exposures for the hot shutdown condition as compared to the cold shutdown condition. Typical potential additional costs for upgrading current system designs have been estimated and compared to the risks (cost-risk ratio); this comparison shows that the costs outweigh the benefits and are in excess of the NRC ALARA suggested policy of $1,000/manrem.

The following point-by-point reponse is keyed to the item numbers of NRC Request 211.35:

Response to Item 1

The DHR system flowpath from the reactor coolant system to the DHR pump suction is the single drop line. The DHR system suction isolation

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valves (DH-V11 and V12) are provided with a bypass line and manual isolation valves. A failure of an isolation valve to open would require suspension of the cooldown until the manual bypass valves are opened. To open the manual bypass valves would require containment entry. The spurious closure could also require containment entry to open the manual bypass valves. A failure of the containment building isolation valve (DH-V10) to open remotely or spurious closure would also suspend the cooldown until the valve could be opened manually. The containment building isolation valve failure would require manual action outside the control room, but not inside the containment building.

To align the DHR system for cooldown requires limited operator action outside the control room. The actions required are:

1. The operator must open the DHR pump suction cross-connect valves (DH-MV19A and B) to establish the suction flowpath.
2. The operator must reestablish power to the DHR cooler bypass valves (DH-V14A and B). These valves are electrically locked closed during normal reactor operation.

With regard to reducing the need for manual actions outside the control room to initiate the normal DHR system cooldown, the DHR system would require:

1. Replacement of manual valves (DH-MV19A and B) with either check valves or power operated valves.
2. Removal of the electrical lock on the DHR cooler bypass valves (DH-V14A and B). These valves would be insured closed during normal reactor operation by administrative control.

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To reduce the need for manual actions outside the control room to mitigate the consequences of a single active failure would require:

1. The DHR suction valves bypass isolation valves (DH-MV22A and B) be changed to power operated valves. The power and control functions for the four suction valves (DH-MV22A and B and DH-V11 and 12) must be channelized in such a manner that the failure of a bus would not impair the valves function.
2. The DHR suction containment isolation valve (DH-V10) be locked in the open position to preclude spurious closure or installation of a parallel line and valve.

The Davis-Besse Unit No 1 DHR suction cross-connect design is similar to the Midland design. The outstanding differences are that the valves correspondending to Midland DH-MV19A and B are provided with motor operators and that the Midland containment isolation valve (DH-V10) has not been incorporated in the Davis-Besse No 1 design. Incorporation of these provisions on Midland would eliminate one of the manual actions outside the control room required to align the DHR system for plant cooldown; however, the operator has at least six (6) hours to perform this action. The valves should be opened after plant cooldown with the steam generator commences, but before cooldown with the DHR system commences. Due to the magnitude of the time available to perform the action, the modification is not deemed necessary.

Response to Items 2 and 3

The Midland design provides for a single atmospheric dump valve per steam generator. These valves are air operated and although they are normally actuated automatically to open to a preset set point by high steam line

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pressure, they can be manually operated from the control room. In addition, each valve is equipped with a handwheel to allow for local manual actuation. Safety-grade atmospheric dump valves are not necessary to satisfy the Midland design basis and, therefore, are not provided. The capability to conduct a safety-grade cold shutdown assuming the most limiting single failure and loss of off-site power is not provided in the Midland design. The achievement of safety-grade hot shutdown is the Midland design basis. This condition will be maintained until repairs are made and/or off-site power is restored at which time cool-down can be conducted.

Response to Item 4

The pressurizer relief valve is seismically qualified as part of the reactor coolant pressure boundary but is not an "active component" in accordance with Regulatory Guide 1.48. Operation of this valve requires manual action which can be taken from the control room if the power supply is available. A safety-grade method to depressurize the reactor coolant system has not been provided. The hot shutdown condition will be maintained until necessary repairs are made and/or off-site power is restored after which plant cooldown will be conducted.

Response to Item 5

Boron can be added to the reactor coolant system by two different methods. The first method is to add concentrated boric acid solution to the reactor coolant system from the chemical addition system. This system uses redundant transfer pumps which can be powered from the on-site diesel generators.

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The second method is to add borated water from the BWST while bleeding reactor coolant through the letdown line of the MU&P system. The borated water is added with the safety-grade seismic Category I high-pressure injection system. A completely safety-grade, single failure proof method of providing borated water to the reactor coolant system during cooldown is not provided as this is not a design basis of the Midland Plant. The hot shutdown condition will be maintained until repairs are made and/or off-site power is restored at which time plant cooldown can be conducted.

Response to Item 6

DHR pressure relief capacity is described in FSAR Section 5.4.7.1.1.3, Revision 11. In addition, the discharge fluid is directed to the reactor building sump. A further description of relief valve design is contained in FSAR Table 5.4.10, Revision 9.

Response to Item 7

No test as described will be conducted. Single failure natural circulation cooldown is not a design basis of the Midland Plant. The hot shutdown condition will be maintained until repairs are made and/or off-site power is restored at which time plant cooldown can be conducted.

Response to Item 8

No procedures for cooling down using natural circulation will be provided as this operation is not a design basis of the Midland Plant.

Response to Item 9

As detailed in our response to Request 010.34, an adequate seismic Category I source will be available.

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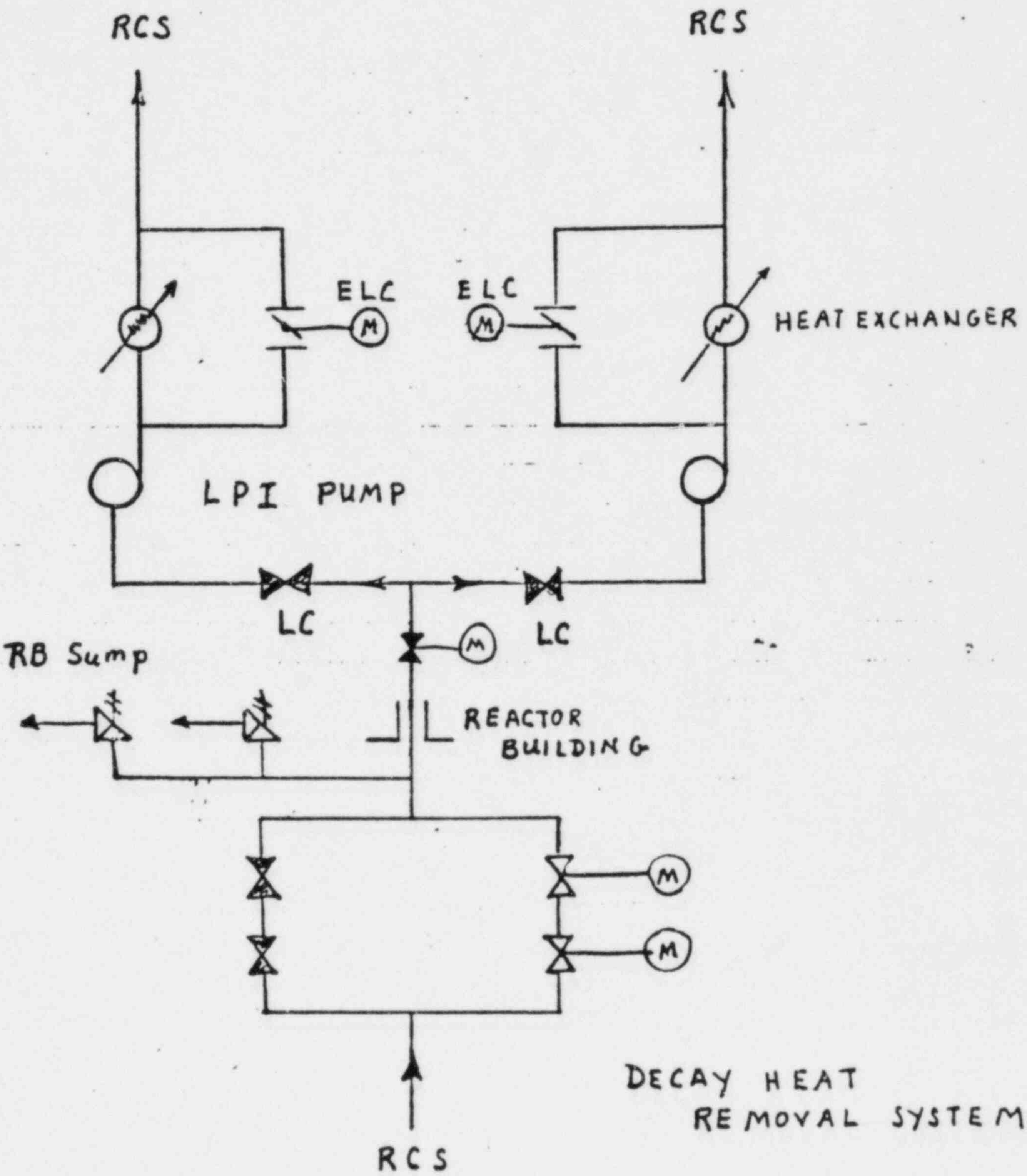

DECAY HEAT
REMOVAL SYSTEM

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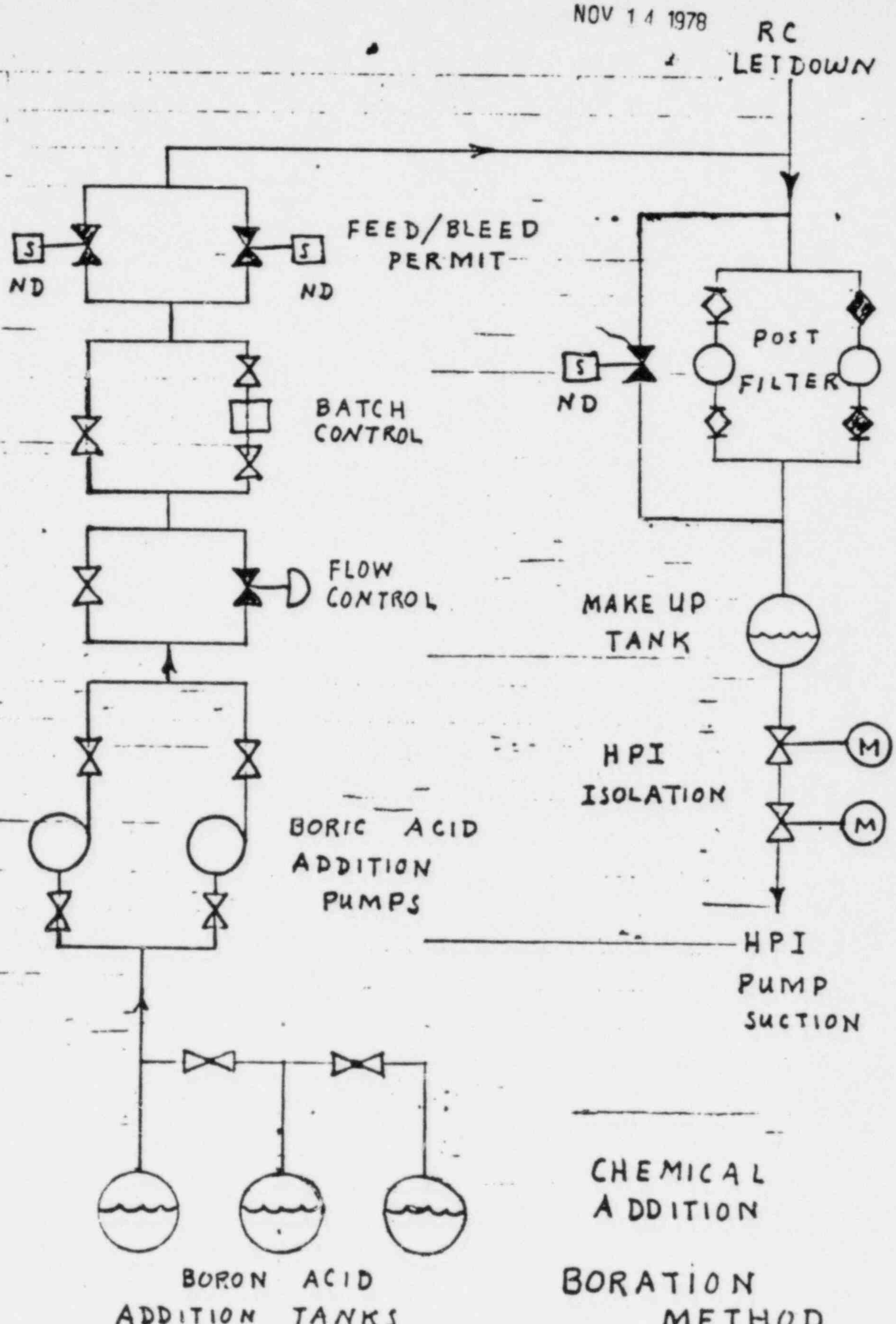

CHEMICAL ADDITION

BORATION METHOD

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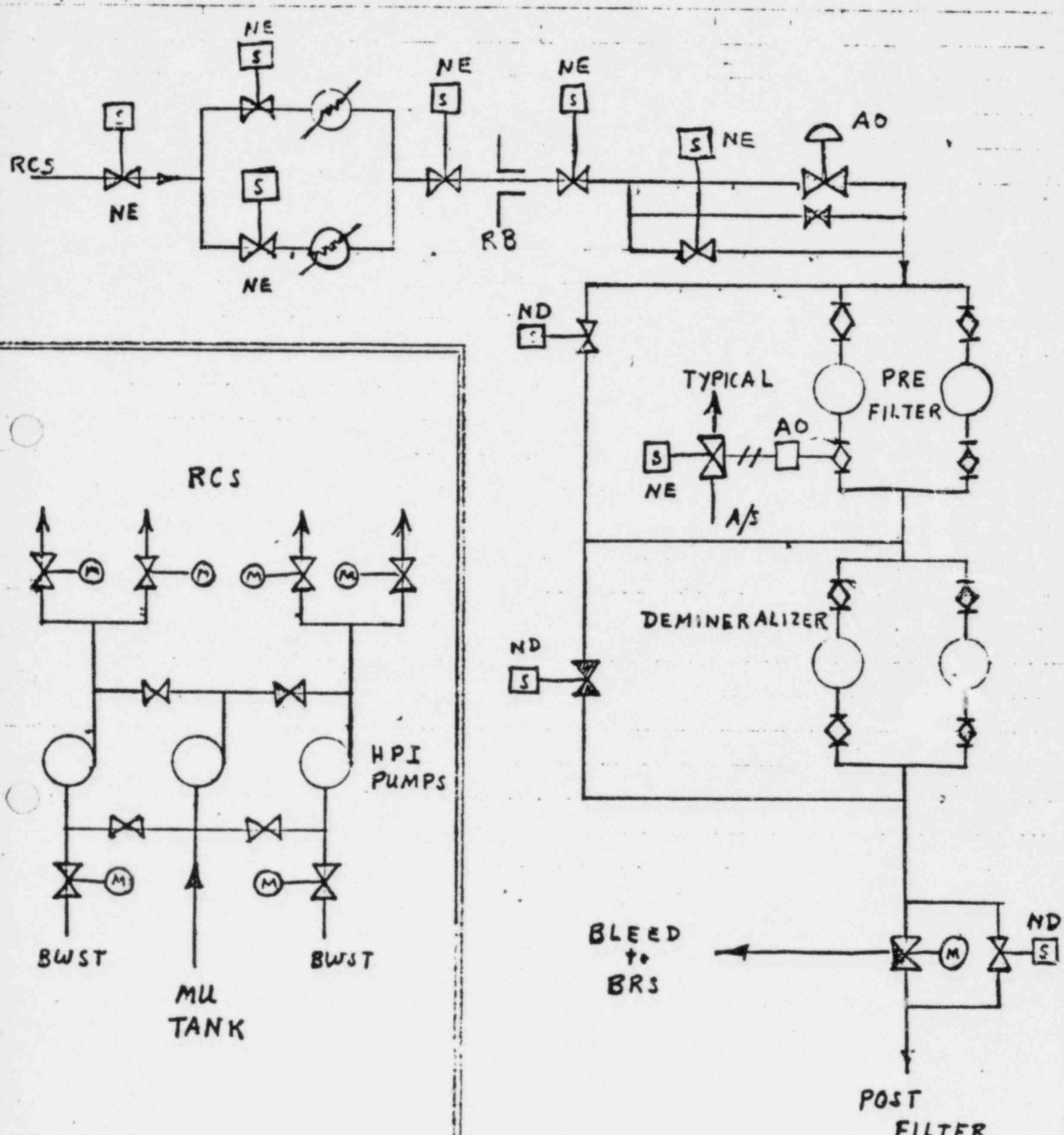

FEED /BLEED
BORATION METHOD

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HEAT REMOVAL

ONE ATMOSPHERIC DUMP VALVE PER STEAM GENERATOR

AIR OPERATED
QUALITY GROUP B
NON-ACTIVE PER REG GUIDE 1.48
LOCAL HANDWHEEL

AUXILIARY FEEDWATER SYSTEM

SINGLE ACTIVE FAILURE
SIESMIC CATEGORY I
LOSS OF OFF-SITE POWER

AUXILIARY FEEDWATER SUPPLY

SEISMIC CATEGORY I SOURCE

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BORATION METHODS

1. FEED AND BLEED

    HPI SYSTEM

        SINGLE ACTIVE FAILURE
        SIESMIC CATEGORY I
        LOSS OF OFF-SITE POWER

    MU&P SYSTEM LETDOWN

        QUALITY GROUP B AND C

2. CONCENTRATED BORIC ACID

    CHEMICAL ADDITION SYSTEM

        REDUNDANT
        DEISEL POWER AVAILABLE TO TRANSFER PUMPS

    HPI SYSTEM

        SINGLE ACTIVE FAILURE
        SEISMIC CATEGORY I
        LOSS OF OFF-SITE POWER

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DEPRESSURIZATION

PRESSURIZER ELECTRIC OPERATED RELIEF VALVE

QUALITY GROUP A
NON-ACTIVE PER REG GUIDE 1.48

Enclosure 3
(1 page)

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Engineered Safety Features Recirculation Loop Leakage

And Dose Assessment

I. Summary of Existing Design

A. Review of equipment location

B. HVAC design

C. Pump design

II. Assumption/Bases for Existing Design and Current Analysis

A. Time of postulated seal failure

Bases: 1) Periodic testing and inspection in accordance with technical specifications

2) Previous dockets: DB-1, ANO-2

3) Crane Packing Co. tests

B. Source terms

Bases: 1) Conservative BWST value assumed

2) SRP Plan 15.6.5

C. Pump seal leakage rate

Basis: Crane Packing Co. tests

D. Undetected leakage time

Basis: 1) ESF room and instrumentation design

2) Credit not taken for area radiation monitors

E. Radioactive release

Basis: 1) SRP Plan 15.6.5, Regulatory Guide 1.4

F. Isolation time

Basis: 1) 24-hour time frame for required actions

III. Summary and Questions

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Enclosure 4
(10 pages)

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222.1 ADDITIONAL INFORMATION IS REQUIRED FOR THE STEAM LINE BREAK ACCIDENTS IN SECTION 15.1.5 OF YOUR FSAR:

Item I: Provide a discussion of the calculational methods use, including all the codes used.

esponse: Subsection 15.1.5.32 discusses briefly the calculational methods used in the steam line break analysis including the computer codes used. Detailed information concerning the computer codes is contained in the referenced topical reports.

Item 2: Provide a detailed flow diagram for the primary and secondary systems identifying all the components considered.

es nse: The requested information is provided in the SLB supplement.

Item 3: Describe how the initial and transient power distributions were calculated. Provide the initial and transient power distributions used in these analyses.

esponse: The initial and transient power distributions are discussed in Item 4.

Item 4: Describe in detail how the thermal-hydraulic effects were evaluated, including calculations for DNBR.

esponse: The Thermal Hydraulic evaluation of the Midland Steam Line Break Accident used the closed channel transient RADAR computer code (BAW-10069A). The input to RADAR was calculated using the TRAP code and consists of system flow, core power, system pressure, and coolant core inlet temperature as a function of time. Two subchannels were modeled into the RADAR code. This first sub-channel represented an average core channel and was used to calculate the core pressure drop as a function of time based on the system flow transient. The core pressure drop, as calculated by RADAR, agreed with the steadystate codes in which the entire core was modeled. The core pressure drop versus time was applied across a second channel to calculate the minimum DNBR, clad temperature, etc. This second channel assumed the worst Nuclear, Thermal, and Mechanical conditions exist simultaneously. Since the core pressure drop versus time is the only dirving force for coolant flow in the hot channel, any increased heating, voiding, etc only decreases the calculated channel flowrate. The maximum design conditions are

represented by the following assumptions:

a. Design peaking conditions are used (1.78 radial x local and 1.[illegible] chopped cosine axial flux shape) to provide a conservative estimate of the hot subchannel transient response.

b. Core pressure and inlet temperature uncertainties are applied in the most conservative manner (-65 psi, + 2 F).

c. A 5% core flow maldistribution penalty is applied to the hot assembly.

d. The hot fuel assembly is assumed to have a reduced peripheral flow area due to the proximity of adjacent fuel assemblies.

e. Hot channel factors are applied to the hottest subchannels. These include:

1. F_a, which reduces the subchannel flow area
2. F_q'', which increases the local heat flux
3. F_q, which increases the heat output of the hottest fuel rod

Item 5: Provide transient axial and radial power distributions for each case analyzed. Describe how these peaking factors were considered in the thermal-hydraulic calculations. How were these peaking factors calculated?

Response: The design peaking factors and their bases are contained in Section 4. Item 4 discusses how these peaking factors were considered in the thermal-hydraulics calculations.

Item 6: Provide all the time dependent reactivity feedbacks during the accident (provide for all the cases analyzed).

Response: The requested information is provided in the SLB supplement

Item 7: Provide nuclear and thermal-hydraulic analyses for the first 15 seconds for both BOL and EOL conditions from full power.

Response: The requested information is provided in the SLB supplement.

Item 8: For the high pressure safety injection system and the flow of borated water from core reflood tanks, describe the flow path into the core, the method of evaluating the time for these fluids to reach the center of the core and the method for determining the resultant reactivity feedback.

Response: No credit was taken for HPI or core flood tanks in the DNB analyses. In the over-cooling analyses, a 25 second delay was assumed in the TRAP analyses to account for HPI pump startup and transit time of the boron to the reactor vessel downcomer. The transit time from the downcomer to the reactor core is calculated by the TRAP computer code. The resultant reactivity feedback is determined in TRAP2 by calculating the boron concentration in each flow path and then converting it to reactivity.

Item 9: Describe in detail how the coolant flow reduction in the hot channel is evaluated. Discuss the potential for coolant flow blockage due to fuel swelling.

Response: The core pressure drop versus time is the only driving force for coolant flow in the RADAR hot channel. Any increase in heating, channel voiding, etc., decreases the calculated hot channel flow-rate (see response to Item 4). No appreciable fuel swelling would be expected prior to the onset of film boiling. After the onset of film boiling, the fuel is assumed to be failed.

Item 10: Describe in detail how the time dependent pressure drop in the fuel channel was calculated.

Response: The RADAR hot channel pressure drop transient matches the core average pressure drop transient (see Item 4 response).

Item 11: Provide a plot of the core coolant density and average fuel temperature for the time period from zero to 15 seconds.

Response: The requested information is provided in the SLB supplement.

Item 12: Describe how the peaking factors in the hot channel were determined.

Response: Although no case specific peaking factors were generated for this analysis, the maximum design radial, local, and axial peaking factors were applied to the hot sub-channel for each steam line break transient analyzed. (See response to item 4)

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B&W Position on Appropriate Stuck Rod Design Bases for MSLB Analyses

Position

In evaluating the consequences of a design basis main stream line break (MSLB), B&W will consider the effect of a stuck rod on gross core shutdown margin. Stuck rod effects on localized physics or thermal performance are not an analysis requirement.

Bases

The B&W position for stuck rod assumptions in accident analyses is based on general Design Criteria 27.

GDC 27 states:

> "The reactivity control systems shall be designed to have a combined capability . . . of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained."

> It is noted that this criteria places one requirement (capability to cool the core) on accident design bases, with a stuck rod.

GDC 26 also addresses stuck rod assumptions.

GDC 26 states:

> "Two independent reactivity control systems . . . shall be provided . . . and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded."

> It is noted that there is no requirement placed by this criterion on accident design bases with a stuck rod.

B&W finds it significant that GDC 27 places no requirement on "specified acceptable fuel design limits" for accident studies using a stuck rod assumption, such as found explicitly in GDC 26 for operational occurrences. A MSLB is a very improbable event; a stuck rod is a very improbable event; a stuck rod is not a consequence of a MSLB. The intent of GDC 27 would appear to be only for the purposes of providing an added level of conservatism to assure an uncontrollable and serious event (core melt-down) will not occur.

Implementation of GDC 27 would require that reactivity control systems include adequate shutdown margin to accommodate a stuck rod and prevent core meltdown during a postulated accident, with no DNBR restrictions. If DNBR limits are in consideration as an evaluation criteria, they are implicitly excluded from stuck rod accident studies by GDC 27. Other criteria may impose a DNBR limit for accident analyses without the assumption of a stuck rod.

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INTRODUCTION

MODELS AND CODES

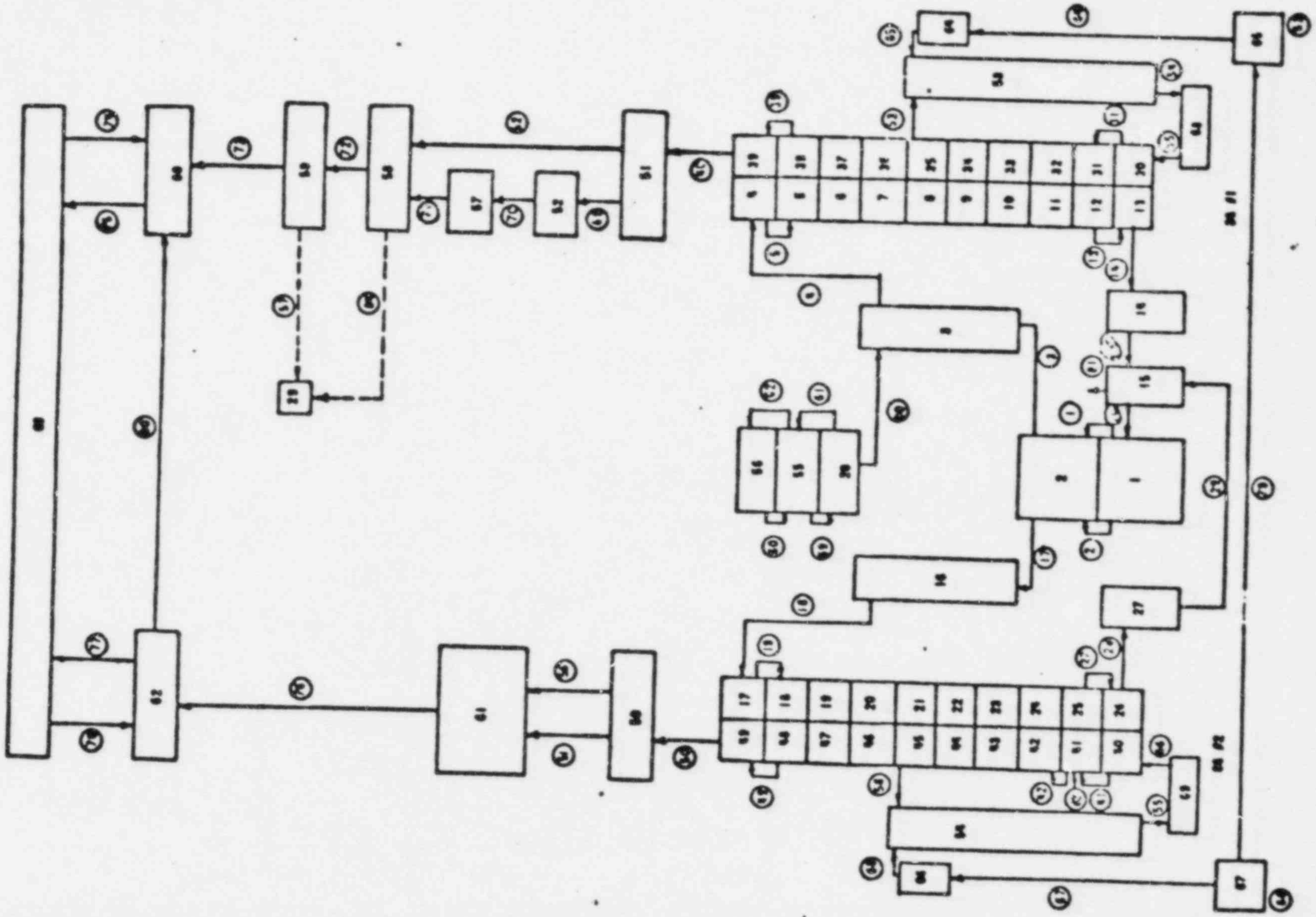
SYSTEM RESPONSE

T-H RESULTS

CONCLUSIONS

COMPUTER CODES USED

TRAP2	BAW-10128, AUGUST, 1976
RADAR	BAW-10069A, REV. 1, OCTOBER, 1974
PDQ07	BAW-10117A, JANUARY, 1977



STEAMLINE BREAK SPECTRUM SUPPLEMENT
NODING DIAGRAM

TABLE 12

DNB RESULTS

Break (Ft²)	Type(1)	Power Level Units	Time of Loop(2) (sec)	Trip Time (sec)	Trip Type	Minimum DNB	Time DNB drops below 1.3	% Core in DNB (3)
12.21	36" DER	102%	0.0	0.0	Pump Monitors	<1.0	1.4	1.25 BOL
12.21	36" DER	102%	3.0	1.466	Low RC Press	<1.0	1.6	NC BOL
12.21	36" DER	102%	0.0	0.0	Pump Monitors	<1.0	1.3	1.25 BOL
12.21	36" DER	102%	None	1.466	Low RC Press	<1.0	1.6	NC BOL
12.21	36" DER	102%	None	4.385	High Flux	1.59	-	NA BOL Process Steam
12.21	36" DER	102%	4.28 (worst flux)	4.28	Pump Monitors	1.86	-	NA BOL Process Steam
12.21	36" DER	80%	0.0	0.0	Pump Monitors	1.04	1.65	NC BOL
6.28	26" DER	102%	0.0	0.0	Pump Monitors	1.22	2.05	NC BOL
3.14 Ft²	Split	102%	4.48 (worst flux)	4.48	Pump Monitors	2.14	-	NA BOL
3.14Ft²	Split	102%	0.0	0.0	Pump Monitors	2.10	-	NA BOL

1. DE = double-ended
2. Loss of offsite power
3. NC = not calculated but less severe than case 1
NA = not applicable - no DNB

(This is Table 15D-13 in FSAR)

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T-H DESIGN CONDITION

1. Maximum design value of $F_{\Delta H}$ (radial and local peaking)
2. Maximum design value of F_z
3. Maximum errors on core pressure and inlet temperature
4. Every channel is assumed to have nominal pressure drop.
5. Isothermal flow reduction penalty of 5%
6. Reduced peripheral flow area
7. Hot channel factors of:
   a. Reduced flow area ($F_A < 1.0$)
   b. Greatest heat output (F_q'')
   c. Increase fuel pin power rating ($F_q > 1.0$)

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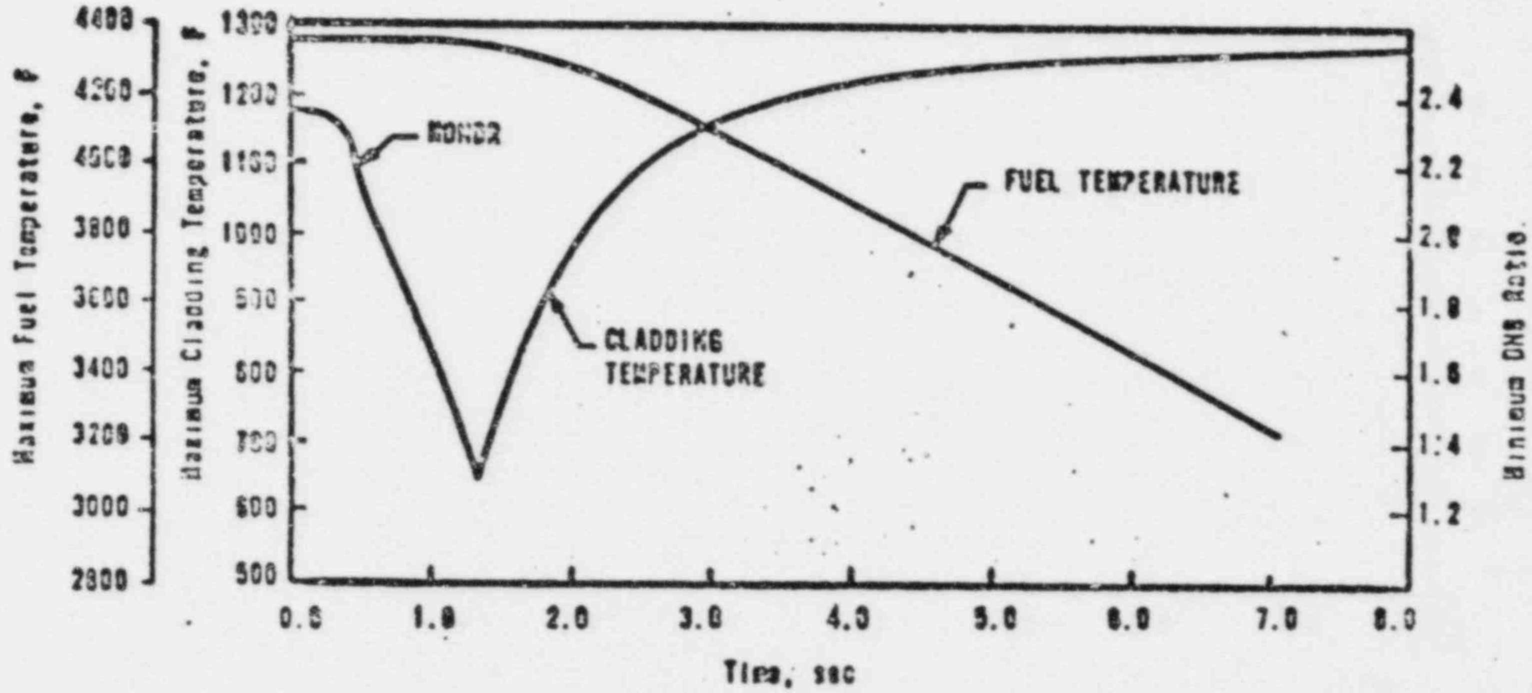

38" PER SLB BOL CONDITIONS