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ACRS COMBINED ECCS/FLUID DYNAMICS SUBCOMMITTEES MEETING MINUTES
AUGUST 27, 1985
WASHINGTON, DC

PURPOSE: The purpose of the meeting is to review: (1) the status of the hydrodynamic loads issue for plants with General Electric Mark I-III containments; (2) the implementation proposal for resolution of USI A-43, "Containment Emergency Sump Performance;" (3) results of calculations of the Davis Besse feed-and-bleed capability; and, (4) ECCS-related issues of on-going concern to NRR. [Note: Item 4 was not discussed due to a lack of time.]

ATTENDEES: Principal meeting attendees included:

ACRS
D. Ward, Chairman
J. Ebersole, Member
H. Etherington, Member
I. Catton, Consultant
V. Schrock, Consultant
H. Sullivan, Consultant
T. Theofanous, Consultant
C. Tien, Consultant
P. Boehnert, Staff

Toledo Edison
T. Myers

NRC
B. Sheron
J. Kudrick
F. Eltawila
A. Serkiz
R. Jones
E. Throm

LANL
B. Boyak

B&W
B. Dunn

MEETING HIGHLIGHTS, AGREEMENTS, AND REQUESTS

1. Messrs. J. Kudrick and F. Eltawila (NRC-NRR) provided a status report on the hydrodynamic loads issues for the GE BWR Mark I-III containments. Overall, the Mark I and II Programs are basically complete. The Humphrey issues are resolved generically and

plant-specific resolution is underway for the Mark III plants. Details noted include:

- For Mark I plants, all have completed their modifications except for Browns Ferry Unit 2 which is currently shutdown for these modifications, and Hope Creek which is being handled via the OL review process.

- Details of the Mark II plant status are given on Figures 1 and 2. LaSalle has just recently completed in-plant SRV load tests per an NRC generic test requirement to demonstrate acceptable loadings for Mark II containments. Preliminary NRC evaluation shows acceptable loadings.

- Regarding the vacuum breaker issue for Mark I plants, NRC has concluded that the vacuum breaker (V/B) valve will not open during pool swell. Dr. Theofanous suggested that NRC review the results for this item, since he believes the tests run to demonstrate acceptable valve loadings were "overdriven" (excessive flow in drywell) and may not have simulated realistic conditions expected in the ring header area. This would subject the breaker to potentially large delta-P's. Further discussion resulted in Mr. Kudrick stating that NRC will investigate this point and, if further work is needed, he will report back to the Subcommittee. Mr. Ebersole suggested that Mark I-III licensees investigate the potential for installation of relief valves on these containments. Further discussions resulted in a request for detailed discussions on the aspects of the dynamic models used to calculate V/B response. Mr. Ward said the Subcommittee will develop a list of discussion points on this and related hydrodynamic loading topics for a future meeting.

- For the Mark II plants, Figures 3-5 detail the plant-specific fixes implemented. In response to Mr. Ward, Mr. Eltawila said that the modifications were deemed adequate by stress analyses, not by testing.

- The status of Generic Issue 61 (raised by J. Ebersole) was reviewed. Generic Issue 61 ("Failure of a Safety/Relief Valve Discharge Line in the Wetwell Air Space Region Combined With Failure to Reclose the SRV (Suppression Pool Bypass)"), was studied by BNL under contract to NRC. Key findings (Figure 6) show that containment spray must be initiated in order to terminate the event and assure containment integrity. Operator action (to initiate sprays) was not assumed to occur. NRC has assigned a "medium" priority to this issue.

  Proposed solutions were reviewed (Figures 7-9). Mr. Etherington noted that early experiments showed significant benefit (condensation) of pool water from steam blowing across the top of the water. NRC said they would look into this point. Mr. Ward asked if use of the ADS was considered for this issue. NRC indicated that it had not been rigorously analyzed; they said they would look into it.

  BNL is reevaluating Generic Issue 61. The study has been expanded to include the potential for V/B failure on HPCI and RCIC turbine exhaust lines. The current schedule calls for NRR review of a recommended resolution position in June 1986 with ACRS review scheduled for July 1986.

- The status of the Humphrey issues was reviewed by J. Kudrick. Mr. Humphrey (an ex-GE employee) raised a number of hydrodynamic concerns focused on the Mark III design. Figures 10-11 highlight the concerns. A Mark III Owners Group was formed to resolve the issues. In response to Dr. Theofanous,

NRC indicated that some of Mr. Humphrey's concerns had merit but there were no significant unresolved safety issues uncovered.

For the Mark I and II plants, resolution was handled either generically (Mark I) or plant-specifically (Mark II). NRC's review of the Mark I's and II's is on-going. Preliminary evaluation shows no significant safety concerns. Final evaluation will follow the Mark III reviews.

For the Mark IIIs, one remaining Humphrey issue is open - interpretation of the 1/10-scale test results. The concern is whether the 50 ft/sec pool swell velocity limit is conservative for all conditions. For River Bend, four plant-specific issues remain open: (1) submerged loads due to encroachments; (2) resonance of SRV sleeve; (3) lateral loads on discharge ends; and, (4) V/B analysis not provided. In response to Mr. Ward, Mr. Kudrick said NRC will make available the portions of plant SERs that address resolution of these and other Humphrey/Mark III issues.

- Addressing the remaining generic issue - possible exceedance of the 50 ft/sec pool swell limit, Mr. Kudrick noted that the 1/10-scale pool swell tests were used to confirm key assumptions of the SOLA code regarding impact loading on structures above the suppression pool. Pressure transducers and high speed cameras were used to obtain the test data. One test showed evidence of liquid impact, not froth as had been expected, leading to suspicion that the 50 ft/sec swell criteria could be exceeded. Unfortunately, the transducer measurements proved to be unusable, and NRC had to rely on the high-speed camera coverage which opens the test results to

interpretation. Plant specific tests were proposed in order to resolve this issue.

The plant specific tests showed no problems, except for the Grand Gulf encroachment configuration. NRC is proposing comparing the 1/10-scale tests with tests run in a 1/3-scale facility in order to resolve the concern here. Dr. Theofanous expressed doubt that one would be able to make any meaningful comparisons between the two sets of tests in different scaled facilities.

In response to Dr. Sullivan, NRC said they had two key concerns: (1) to assure the loads at the HCU floor are froth loads only, and (2) to assure the 50 ft/sec swell velocity is not exceeded. Mr. Etherington said that it is necessary to consider the global pool response when evaluating the loads on the encroached structures. In response to Mr. Ward, Mr. Kudrick said that if NRC decides the 50 ft/sec limit is exceeded, a structural loading reevaluation would be necessary.

The BWR Owners have argued that atypicalities in the 1/10-scale tests, due to scaling considerations, caused overdriving of the tests and resulted in atypically higher swell velocities. NRC is in essential concurrence with this point.

To resolve this issue, the Owners reevaluated the 1/3-scale data to show that the 50 ft/sec criterion was not exceeded. They have also attempted to determine the degree of conservatism (higher than expected velocities) in the 1/10-scale tests. These results will be docketed shortly. Dr. Theofanous requested the documentation associated with

verification of the SOLA code. In response to Dr. Sullivan, Mr. Kudrick said this concern would translate to a localized effect for a full scale plant in a LB LOCA situation(i.e. loadings based on ~2 feet of water column).

- Resolution of the BWR suppression pool temperature limits issue was reviewed. NRC established a 200°F limit on the pool temperature to avoid a potential situation of unstable steam condensation phenomena. Further work by GE and others shows that there is no unstable phenomena seen up to saturation. The licensees have petitioned NRC to relax the temperature limits. NRC evaluation of this effort however resulted in a low priority rating. In response to Dr. Catton, NRC said such a relaxation would be helpful in an ATWS situation, among others. The Owners plan to provide additional information in order to spur active Staff review of this relaxation proposal.

2. A. Serkiz (NRR) reviewed the resolution proposal for USI A-43 "Containment Emergency Sump Performance". NRR has issued a revised draft Regulatory Guide - Regulatory Guide 1.82, Revision 1. Mr. Etherington stated that he believes the Regulatory Guide is very detailed and perscriptive. Mr. Serkiz said it was not NRC's intent to be perscriptive - but the Appendix to the Guide is detailed in an effort to make the Guide self-contained. Mr. Ebersole recommended a specific definition of an acceptable debris stream for pumps using hydroclone separators. NRC said they are receptive to any suggestions in this area. Figure 12 details the revisions made to Regulatory Guide 1.82. Addition of Item 4 (effect of debris on pump bearings and seals) resulted from questions raised by the Subcommittee. NRC will meet with the CRGR on September 9, 1985 to review their proposed resolution actions. Since CRGR has not reviewed these actions, they are considered pre-decisional.

In closed session, NRR detailed their proposed resolution actions. Figure 13 shows the key findings. The bottom line is that NRR does not see a need for backfit on this USI (i.e. forward-fit only). NRR also noted that they are not relying on "leak-before-break" to resolve this issue. In response to Mr. Ebersole, NRR said they are not considering averted costs in their value/impact analysis in accordance with Commission policy.

NRR proposes to issue a Generic Letter to near-term OL's and OL plants suggesting that if plants are proposing to change out insulation, they evaluate the effects of using this new form of insulation vis-a-vis the USI A-43 safety issues.

Dr. Sullivan questioned the overall resolution position from the standpoint of no backfit actions. NRR indicated that the value/impact analyses do not support any backfit action. In response to a question from Mr. Ward regarding giving attention to ~6 plants that have been identified as having unique vulnerabilities vis-a-vis this USI, NRR plans no specific action for these plants. In response to further questions, Mr. Serkiz said the six or so plants considered outliers are not known by name - its a best estimate based on analyses. In response to further Subcommittee questions, Mr. Serkiz said follow-up on this item would be done by the NRC Region Offices as insulation is changed-out.

3. NRR detailed their analyses of the ability of the Davis Besse plant to cool the core via feed and bleed. Dr. Sheron led off the presentations by overviewing the operational considerations for use of feed and bleed cooling at Davis Besse. Key points noted by Dr. Sheron included:

° Feed and bleed is <u>not</u> available for loss of all AC power.

- Feed and bleed eventually produces a harsh containment environment.

- For successful feed and bleed, not only are pump and relieving capacities important, but equipment needed must be qualified to the expected environment, it must be available and operators must not be afraid or hesitant to initiate feed and bleed.

R. Jones provided some "hand" calculations he performed of the Davis Besse feed and bleed capability based on recently received information from Toledo Edison. In response to Mr. Ward, Mr. Jones said the operators at Davis Besse had more time to initiate feed and bleed than originally thought, but NRR is concerned with the operator's reluctance to begin feed and bleed. Mr. Ward cautioned though that one needs to carefully consider use of feed and bleed given the downside consequences. NRR agreed but noted that in the Davis Besse situation, beginning feed and bleed seemed warranted.

Mr. Jones noted that NRR had previously stated to the ACRS DHRS Subcommittee that Davis Besse could not feed and bleed. Since the June 9 LOFW event, NRR reevaluated Davis Besse feed and bleed capability using hand and code (TRAC and RELAP) calculations. NRR concluded that Davis Besse has more alternate heat removal capabilities than previously believed. The key point is that the PORV relieving capacity is significantly greater than originally believed.

The results of the calculations performed by Mr. Jones (Figure 14) indicated that for no operator action core uncovery begins in ~ 60 minutes at 90% power. If one make-up pump is available, uncovery is delayed past 100 minutes. If both make-up pumps are available recovery is usually achieved if feed and bleed is initiated in a

reasonable time (20-30 minutes into the event). Calculations also show that using the make up system pumps only, the core may partially uncover but no cladding damage is expected.

The LANL calculations of the Davis Besse event were presented by B. Boyak of LANL. LANL performed this work at the behest of NRR, initially in support of the I&E investigation Team. The objectives of the calculational exercise were: (1) evaluate how the plant would have responded to permanent total loss of feedwater (time to core uncovery); (2) evaluate how the plant would have responded to feed and bleed procedures initiated at 15, 20, and 35 minutes following a loss-of-feedwater initiator; and, (3) simulate the response of the plant during the first 830 seconds of the June 9, 1985 transient. Figure 15 shows the transient cases analyzed.

Summary results noted by LANL included: (1) core uncovery was estimated to occur at about 9200s ($\sim$2.5 hours) assuming a permanent loss of feedwater to the secondary and one makeup pump supplying the primary; (2) feed and bleed was calculated to be successful if started within 20 minutes following the initiating event. The plant remained at 1800 psia, well above decay heat removal system operating pressure; (3) feed and bleed initiated at 35 minutes following the initiating event would also be successful. There are sufficient safety-grade water supplies available to keep the core covered until the primary pressure drops below the HPI shutoff head (approximately 9 hours); and, (4) the TRAC calculation simulated the global characteristics of the June 9 transient. However, there were differences in detail. Some differences were caused by modeling deficiencies. These were corrected and produced an improvement in calculated transient response.

In conclusion LANL noted that: (1) feed and bleed will be successful if begun in the Davis Besse plant within the times for

which their analyses apply (15-35 minutes); (2) LANL is confident that the TRAC code and Davis Besse plant model adequately simulates the global response of the Davis Besse plant; (3) further efforts to confirm the quality of the provided data base are desirable (plant data, feedwater system layout, and plant operation during the transient); and, (4) recommend that suggested improvements to the plant data base and model be made and that the early plant transient be recalculated. However given these corrections, LANL believes our general conclusions about feed and bleed will still apply.

E. Throm overviewed the status of RELAP calculations of the Davis Besse transient using the nuclear plant analyzer (NPA). There have been difficulties exercising the NPA for this event. Principal problems included: computer problems, mainly due to a lightning strike near the INEL facility, and lack of plant specific information in order to exercise the plant control system. Despite the above problems, most of the results shown were in fairly good agreement with the transient data, i.e., the NPA could fairly well simulate the Davis Besse plant response.

4. T. Myers (Toledo Edison) reviewed the decay heat capabilities of the Davis Besse plant via makeup/HPI cooling (Figures 16-17). He indicated that Toledo Edison believes the make-up system has sufficient flexibility and is sufficiently rugged (portions of the MU system are qualified Seismic I and it has access to a IE power bus) such that used in conjunction with HPI, the plant has a reliable source of water for feed and bleed cooling if needed.

B. Dunn (B&W) discussed calculations of the Davis Besse plant response to the June 9 event using RELAP-5/MOD-2. Figure 18 shows the assumptions used in the calculation. An analysis of operator action times to intercept a LOFW event at 100% power shows that if

the operator acts within 20 minutes, there will be no core uncovery. Operator action at 30 minutes would result in a PCT of 1400°F, but no core damage.

5. The meeting was adjourned at 7:10 p.m.

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NOTE: Additional meeting details can be obtained from a transcript of this meeting available in the NRC Public Document Room, 1717 H Street, N.W., Washington, D.C., or can be purchased from Ann Riley & Associates, Ltd., 1625 I Street, N.W., Suite 921, Washington, DC 20006, (202) 293-3950.

MARK II PLANTS

LOCA RELATED HYDRODYNAMIC LOADS

- NUREG-0808, AUGUST 1981 IDENTIFIES THE ACCEPTANCE CRITERIA.

- LASALLE, LIMERICK, SHOREHAM, WNP-2, AND NINE MILE POINT ASSESSED AGAINST THE ACCEPTANCE CRITERIA.

- SUSQUEHANNA ASSESSED AGAINST NUREG-0808 CRITERIA EXCEPT FOR THE CHUGGING AND CONDENSATION OSCILLATION LOADS.

  PLANT UNIQUE CHUGGING AND C/O LOADS, BASED ON KWU TESTS.

SRV RELATED HYDRODYNAMIC LOADS

- NUREG-802, OCTOBER 1982, IDENTIFIES THE ACCEPTANCE CRITERIA

FIG. 1

MARK II PLANTS (CONT'D)

- T-QUENCHER

  LASALLE, SUSQUEHANNA, SHOREHAM AND NINE MILE POINT ASSESSED AGAINST THE ACCEPTANCE CRITERIA.

- X-QUENCHER

  WNP-2 USED PLANT SPECIFIC LOADS DERIVED FROM COARSO AND TOKAI-2 INPLANT TEST DATA.

- SRV - INPLANT TESTS HAVE BEEN CONDUCTED AT LASALLE. RESULTS UNDER EVALUATION.

FIG. 2

MARK II PLANTS (CONT'D)

WETWELL-TO-DRYWELL VACUUM BREAKERS

- PLANT SPECIFIC ANALYSIS

  -SUSQUEHANNA, LIMERICK, SHOREHAM (AGCO VALVES) HAVE IMPLEMENTED DESIGN CHANGES TO REDUCE THE VALVE IMPACT VELOCITIES DURING POOL SWEEL

  -THE MODIFIED VALVE WAS TESTED AT HIGHER IMPACT VELOCITIES THAN THOSE PREDICTED BY THE MODEL. THE RESULTING STRESSES WERE WITHIN CODE ALLOWABLE.

  -LASALLE GPE VALVES TESTED FOR PREDICTED OPENING AND CLOSING VELOCITIES

  MODIFICATIONS TO THE VALVES WERE INCORPORATED TO INCREASE THE MARGIN OF SAFETY

  -NMP-2 GPE VALVES RELIED ON LSCS TESTS

FIG. 3

TYPICAL MODIFICATION TO VACUUM BREAKERS

INSTALLED ON MARK II PLANTS

- REPLACE 4-BAR LINKAGE WITH A SINGLE LINK TO PROVIDE RESTORING TORQUE CHARACTERISTIC THAT REDUCES IMPACT VELOCITIES

- REPLACE THE FOLLOWING COMPONENTS WITH STRONGER MATERIAL

  SHAFT, KEYS, TURNBUCKLE AND ARM ASSEMBLY

- NEW DISC ASSEMBLY TO ACCOMMODATE CLOSING IMPACT LOADS

- NEW SHAFT BEARING CAPS

FIG. 4

- CHUGGING AND CONDENSATION OSCILLATION

  -SUSQUEHANNA, SHOREHAM, LIMERICK CAPPED THE DOWNCOMERS ON WHICH V/B ARE INSTALLED TO ELIMINATE THE CYCLING OF THE V/B DURING C/O AND CHUGGING

  -LASALLE - EXTERNAL VACUUM BREAKERS NOT AFFECTED BY C/O OR CHUGGING

  -NMP-2 V/B INSTALLED ON THE DIAPHRAM FLOOR AND, THEREFORE, ARE NOT AFFECTED BY C/O AND CHUGGING

WNP-2

- 9 V/B ASSEMBLIES; 2 VALVE DISCS IN SERIES
- ADAPT A DAMPING DEVICE (SNUBBER) TO THE V/B TO QUALITY THE VALVE FOR BOTH CHUGGING AND POOL SWELL LOADS
- DOWNCOMERS UNCAPPED
- THE SNUBBER LIMITS THE VALVE OPENING AND CLOSING IMPACT VELOCITIES TO LESS THE 2 RADIANS/SEC FOR BOTH POOL SWELL AND CHUGGING PHENOMENA. THESE VELOCITIES ARE WELL BELOW THE TESTED VALVES.

FIG. 5

KEY FINDINGS

- POSTULATED SEQUENCE CAN OCCUR AT FREQUENCY COMPARABLE TO ACCIDENT SEQUENCES NOW CONSIDERED TO BE SIGNIFICANT CONTRIBUTORS TO RISK

  * AT 100% BYPASS, RUPTURE PRESSURE EXCEEDANCE OCCURS WITHIN ABOUT 10 MINUTES OF THE INITIATING EVENT AND DESIGN PRESSURE EXCEEDANCE IN LESS THAN 5 MINUTES

  * AT 60% BYPASS (OR LESS), RUPTURE PRESSURE EXCEEDANCE OCCURS NO SOONER THAN THIRTY (30) MINUTES

  * AT 50% BYPASS (OR MORE), DESIGN PRESSURE EXCEEDANCE (70 PSIA) OCCURS WITHIN TEN (10) MINUTES

  * AT 25% BYPASS (OR LESS) DESIGN PRESSURE EXCEEDANCE OCCURS NO SOONER THAN 30 MINUTES

  * ACTUATION OF CONTAINMENT SPRAY TERMINATES ALL PRESSURE INCREASE IN CONTAINMENT

  * TO FIRST ORDER, THE ABOVE FINDINGS ARE INSENSITIVE TO:
    ** DETAILED REACTOR TRANSIENT
    ** CONTAINMENT DESIGN
    ** CONTAINMENT LEAKAGE
    ** FAILURE OF ONE VACUUM BREAKER

FIG. 6

NRC/DST PRIORITY RANKING OF GI 61

- MEDIUM PRIORITY

  WHEN RESOURCES ARE AVAILABLE, FURTHER CONSIDERATION BE GIVEN TO THE POTENTIAL PUBLIC TO OCCUPATIONAL RISK TRADEOFFS

- PROPOSED SOLUTIONS

  - AUTOMATIC CONTAINMENT SPRAY

    NEW CONTAINMENT PENETRATION FOR SUCTION

    PUMP AND MOTOR INSTALLATION

    ELECTRIC CIRCUITRY FOR AUTOMATIC ACTUATION

    OCCUPATIONAL EXPOSURE 2100 MAN REM

    TOTAL INDUSTRY COST $36.2 MILLION

FIG 7

- INSPECT SRV DISCHARGE LINES IN WETWELL AIR SPACE

VISUAL AND/OR RADIOGRAPHIC TECHNIQUES

AT FREQUENT INTERVAL (3 TIMES IN EVERY 10 YEARS)

LOW RADIATION ENVIRONMENT, OCCUPATIONAL EXPOSURE 11,000 MAN REM

18,000 MAN WEEKS OF INSPECTION IS ESTIMATED

18,000 MAN WEEKS SUPPORT OF INSPECTION (PLANNING, REPORT WRITING, FILM READING, REPAIRS, ETC.)

TOTAL COST TO INDUSTRY $72 MILLION

- GUARD PIPES INSTALLED AROUND SRV DISCHARGE LINES IN WETWELL AIRSPACE

RESTRICTED ACCESS

PROVIDE ADEQUATE SUPPORT FOR THE IMPINGEMENT LOADS

LOW LEVEL RADIATION ENVIRONMENT

1 MONTH PER PLANT

FIG 8

NO DETAIL STUDY ON REPLACEMENT POWER COSTS, ASSUMED $216M

TOTAL COST TO INDUSTRY WOULD EXCEED $250M

- RE-ROUTING THE SRV DISCHARGE PIPING THROUGH DOWNCOMERS

NO COST/IMPACT STUDY HAS BEEN PERFORMED

FIG. 9

HUMPHREY ISSUES

1. ENCROACHMENT EFFECTS

2. SRV ANNULAR SLEEVE

3. RHR OPERATION
   - STEAM CONDENSING
   - V/B RESPONSE
   - POOL STRATIFICATION

4. POOL TEMPERATURE RESPONSE (D.W, HEAT TRANSFER, GRADIENTS)

5. STEAM BYPASS CONCERNS

6. RECOMBINER/PURGE OPERATION

7. POOL INTERACTION W/CONTAINMENT

8. FSAR/T.S. INTERACTION (WORST CASE)

9. ECCS INTERACTIONS

10/11 DRYWELL FLOODING

12. UPPER POOL DUMP SIGNAL

13. SPRAY EFFECTS ON ANALYSIS

FIG. 10

HUMPHREY ISSUES (CONT.)

14. LPCI FAILURE CAUSING LOCA

15. V/B EFFECT ON SECONDARY CONT.

16. OPERATOR ERROR DUE TO POOL TEMP. SENSORS

17. EPG GUIDE WITH UPPER POOL DUMP

18 INSULATION FAILURES

19. CHUGGING EFFECTS DUE TO UPPER POOL AND ENCROACHMENTS

20. DRYWELL LOADINGS

21. PURGE OPERATION

22. EPG FOR DEGRADED CASES MAY CONFLICT WITH DBA

FIG. 11

OVERVIEW OF REVISIONS TO RG 1.82

1) RG 1.82 HAS BEEN REVISED TO INCLUDE BOTH BWRs AND PWRs. PROVISION FOR POST-LOCA RECIRCULATION CAPABILITY IS GENERIC TO BOTH TYPES OF REACTORS.

2) THE CURRENT 50% BLOCKAGE CRITERION HAS BEEN DELETED; GUIDANCE FOR ASSESSING PLANT SPECIFIC DEBRIS BLOCKAGE EFFECTS IS PROVIDED IN APPENDIX A.

3) THE RG HAS BEEN REVISED TO REFLECT SUMP (OR SUCTION INLET) HYDRAULICS FINDINGS AND REMOVES VORTEX OBSERVATIONS AS THE BASIS TO QUANTIFY AIR INGESTION LEVELS. APPENDIX A PROVIDES CONSERVATIVE GUIDELINES FOR ESTIMATING POTENTIAL AIR INGESTION LEVELS BASED ON FULL SCALE TESTS.

4) THE RG HAS BEEN REVISED TO REQUIRE ASSESSMENT OF DEBRIS AND PARTICULATE EFFECTS ON RHR AND CSS* PUMP BEARING AND SEAL ASSEMBLIES.

FIG. 12

VY ~~RECORD~~
(CLOSED)
A SERK17

FINDINGS

LIMITED DISTRIBUTION

1) THE ESTIMATED CORE MELT FREQUENCY, CMF, FROM LOCA PLUS SUMP BLOCKAGE FOR PLANTS WITH SIGNIFICANT BLOCKAGE VULNERABILITY IS LESS THAT 3 x 10^{-5}/RX YR.

2) SUMP BLOCKAGE PROBABILITIES ARE BASED ON PIPE BREAK ESTIMATES DERIVED IN 1977, FOR WHICH THE DATA BASE INCLUDED PIPING FAILURES OF ALL TYPES KNOWN AT THAT TIME, INCLUDING MATERIALS NOT USED IN NUCLEAR PLANT PIPING.

3) THE LOCA PROBABILITY (OR INITIATING EVENT) IS THEREFORE OVERESTIMATED WHEN COMPARED TO CURRENT ESTIMATES OF PIPE BREAK AND BREAK TYPE, AS PREDICTED BY CURRENT FRACTURE-MECHANICS ANALYSES.

4) THE CALCULATED WIDE RANGE FOR SUMP BLOCKAGE PROBABILITY, CORE MELT PROBABILITY, OFFSITE CONSEQUENCES, AND VALUE-IMPACT RATIOS REFLECT THE RANGE OF PLANT DESIGN FEATURES AND COST ESTIMATES FOR BACKFIT ACTIONS.

5) THE CALCULATED VALUE/IMPACT RATIOS FOR ALL TYPES OF CONTAINMENT ARE MARGINAL.

6) BASED ON THE ABOVE, WE DO NOT BELIEVE THAT A BACKFIT REQUIREMENT IS JUSTIFIED.

FIG. 13

SUMMARY OF STAFF SIMPLIFIED DAVIS-BESSE CALCULATIONS

CASE NO.	POWER LEVEL	NO. OF MU PUMPS	SUFP	ACTUATION TIME	SG DRYOUT TIME MIN	SATURATION TIME MIN	CORE UNCOVERY TIME MIN	MINIMUM RCS LIQUID VOLUME CU. FT.
1	100%	0	NO	N/A	5	33	47- 57	N/A
2	90%	0	NO	N/A	4	35	54- 63	N/A
3	75%	0	NO	N/A	3	40	64- 75	N/A
4	90%	1	NO	SG DRYOUT	4	45	82-106	N/A
5	75%	1	NO	SG DRYOUT	3	55	124-105	N/A
6	100%	2	NO	SG DRYOUT	5	46	112-N/A	N/A -3265
7	100%	2	NO	20 MINUTES	5	39	93-N/A	N/A -2792
8	75%	2	NO	SG DRYOUT	3	77	N/A	4284-5919
9	100%	0	YES	20 MINUTES	5	46	120	N/A
10	100%	1	YES	SG DRYOUT	5	**	N/A	10450
11	100%	1	YES	20 MINUTES	5	91	N/A	>4284-5919
12	100%	2	YES	20 MINUTES	5	**	N/A	10450
13	90%	2	NO	10 MINUTES	4	60	N/A	3475-4950
14	90%	2	NO	30 MINUTES	4	38	N/A	2984-4419
15	100%	2	NO	30 MINUTES	5	34	83-155	N/A
16	90%	1+	NO	SG DRYOUT	4	48	108-158	N/A

NOTE 1: LIQUID VOLUME TO TOP OF CORE IS 2622 CUBIC FEET.

NOTE 2: ALL CASES ASSUME NO DEPRESSURIZATION WITH PORV.

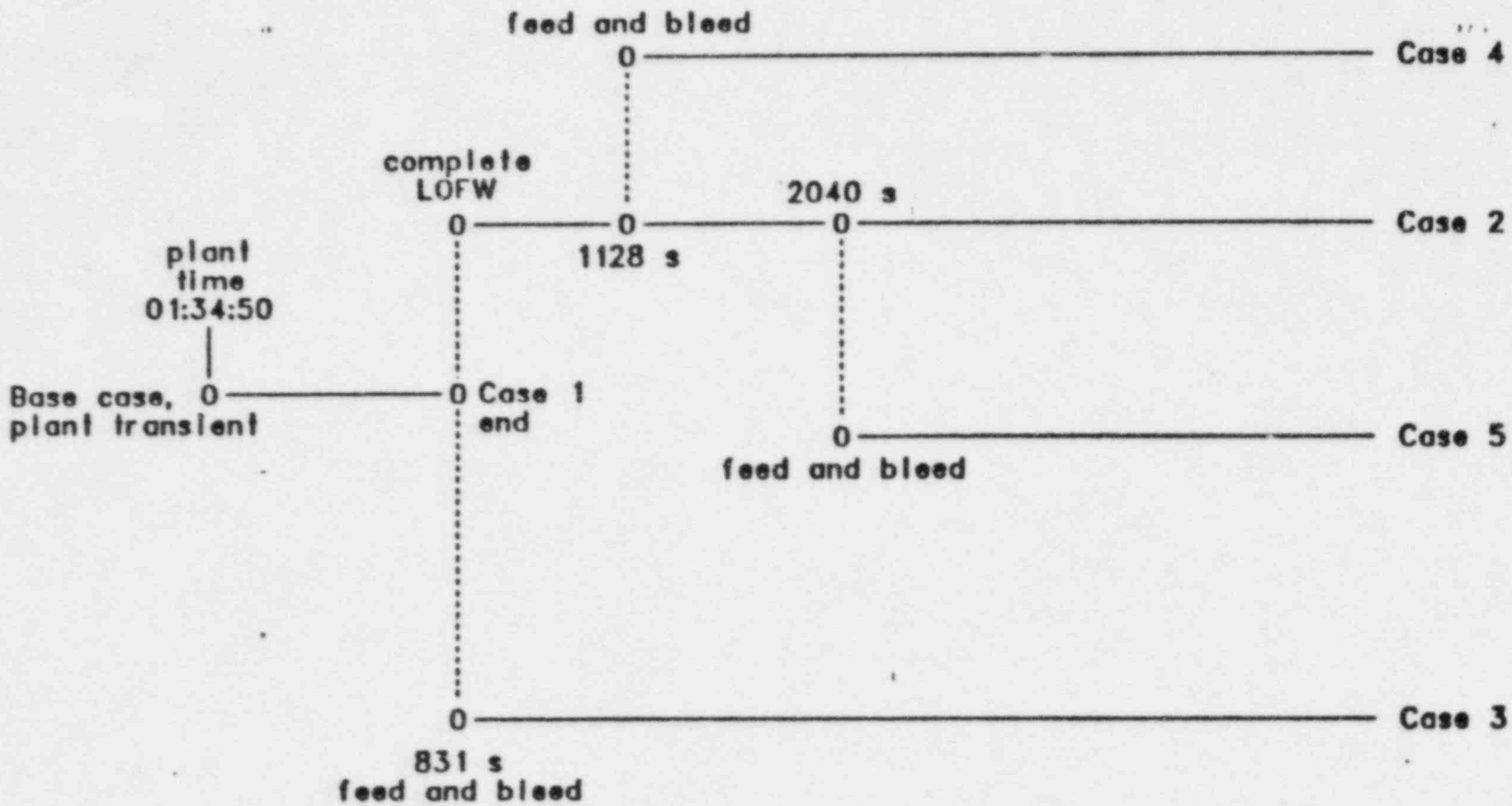
** : SYSTEM REMAINS SUBCOOLED.

+ : 127 GPM

FIG 14

ACRS AUGUST 27, 1985

R. C. JONES, NRR/DSI/RSB



Transients analyzed

Los Alamos

FIG. 15

Decay Heat Removal by Makeup/HPI Cooling

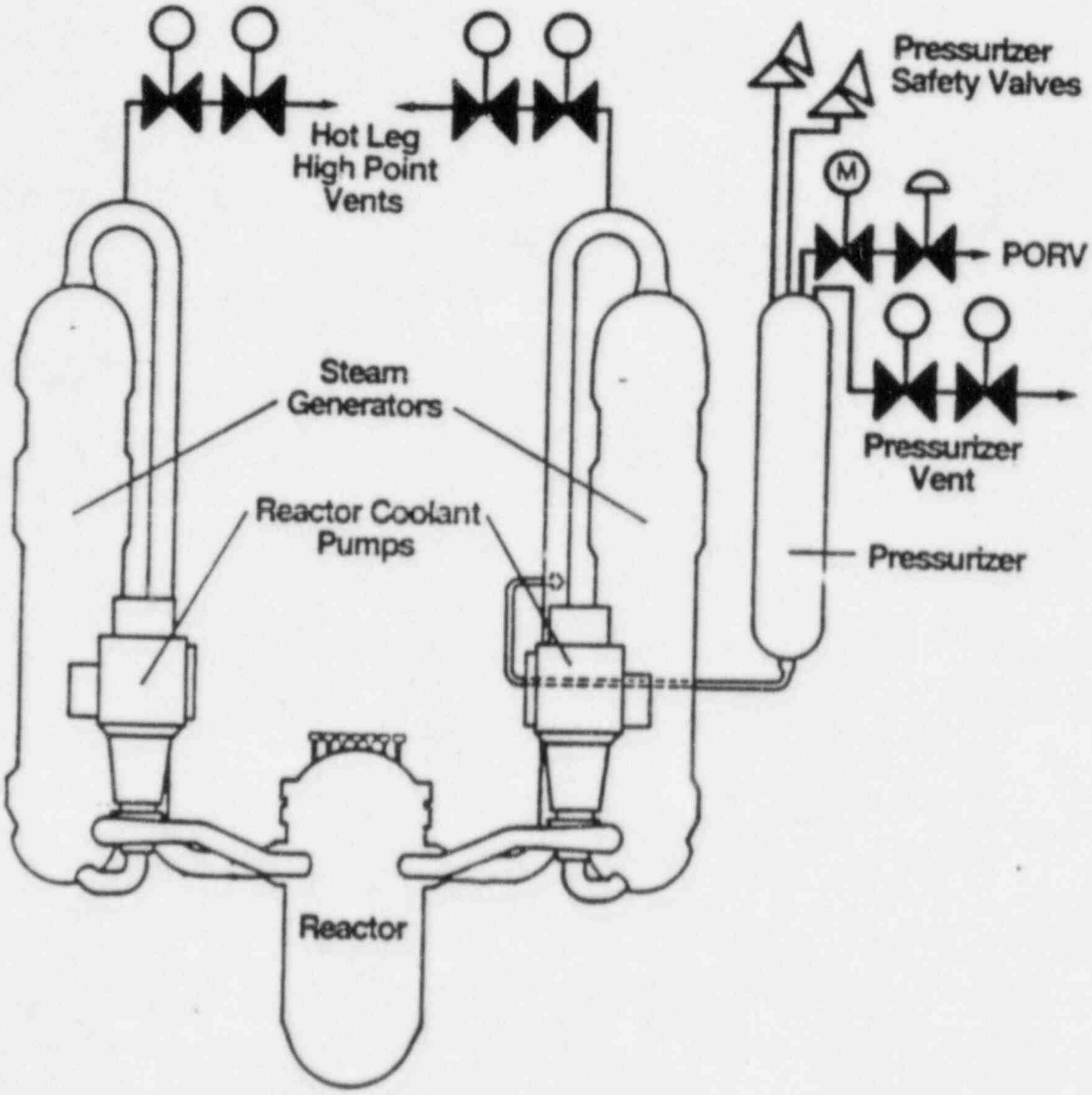


Fig. 16

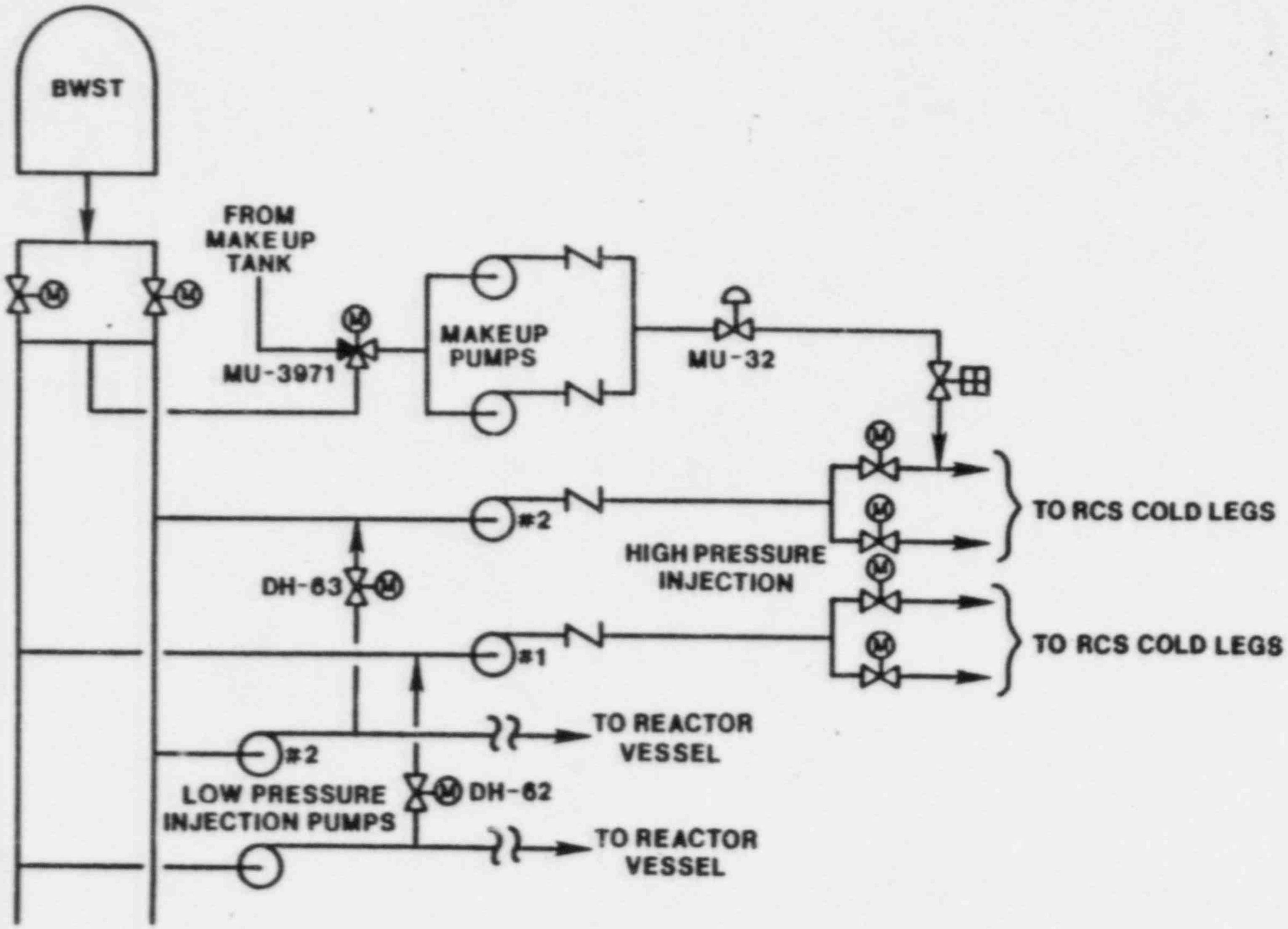


Figure 4.5 Makeup/HPI Cooling System

FIG. 17

SIMULATION OF TRANSIENT BY RELAP5/MODE2

ASSUMPTIONS

POWER LEVEL	BASE LEVEL IS 2772 MWT
DECAY HEAT	1.0 1979 A.N.S. ENVELOPE
PUMP TRIP	AT 20F SUBCOOLING
PORV FLOW	226,000 LBM./HR. STEAM AT 2500 PSIA
MAKEUP FLOW	165 GPM AT 2500 PSIA
HPI	(NOT ACTIVE IN THESE STUDIES)

FIG. 18