ACRS-2336
PDR 10/7/85

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MINUTES OF THE 303rd ACRS MEETING
JULY 11-13, 1985
WASHINGTON, DC

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

Revised: July 9, 1895

SCHEDULE AND OUTLINE FOR DISCUSSION
303RD ACRS MEETING
JULY 11-13, 1985
WASHINGTON, D. C.

Thursday, July 11, 1985, Room 1046, 1717 H Street, NW, Washington, D.C.

1) 8:30 A.M. - 8:45 A.M.	Report of ACRS Chairman (Open) 1.1) Opening Statement (DAW) 1.2) Items of current interest (DAW/RFF)
2) 8:45 A.M. - 9:30 A.M.	Discuss topics for meeting with Commissioners (Open) 2.1) Consideration of seismic events in emergency planning (ACRS report dated June 10, 1985) (DWM/OSM) 2.2) Quantitative Safety Goals (status report)(DO/RPS)
3) 9:30 A.M. - 11:30 A.M.	Meeting with Commissioners (Open) 3.1) Discuss topics noted above
4) 11:30 A.M. - 12:00 Noon SEE HANDOUT 8	Briefing Regarding Steam Line Failure in Nonnuclear Power Plant (Closed) (Note: This session will be closed to discuss Proprietary Information.)
12:00 Noon - 1:00 P.M.	LUNCH
5) 1:00 P.M. - 3:00 P.M.	Quantitative Safety Goals (Open) 5.1) Discuss proposed ACRS report regarding proposed NRC Quantitative Safety Goals (DO/RPS) 5.2) Meeting with NRC Staff and invited experts, as appropriate
6) 3:00 P.M. - 4:30 P.M.	Diablo Canyon Nuclear Plant, Units 1 and 2 (Open) 6.1) Subcommittee report regarding ten-year seismic review (CPS/EGI) 6.2) Meeting with NRC Staff and the Licensee, as appropriate
7) 4:30 P.M. - 6:00 P.M.	Recent events at operating reactors (Open) 7.1) Report of ACRS Subcommittee on Reactor Operations (JCE/HA)

		7.2) Briefing by members of NRC Staff
8)	6:00 P.M. - 6:30 P.M.	Future Activities (Open)
	See Tab 4.1	4.1) Discuss anticipated ACRS meetings (MWL)
	SEE HANDOUT 4.2	4.2) Discuss proposed ACRS activities (RFF)

Friday, July 12, 1985, Room 1046, 1717 H Street, NW, Washington, D.C.

9) 8:30 A.M. - 9:30 A.M. Watts Bar Nuclear Station (Open)
- 9.1) Report of ACRS Subcommittee on quality assurance activities at the Watts Bar Nuclear Station (GAR/RKM)
- 9.2) Meeting with representatives of NRC Staff and the Applicant, as appropriate

10) 9:30 A.M. - 11:00 A.M. Quantitative Safety Goals (Open)
- 10.1) Discuss proposed ACRS report regarding proposed NRC Quantitative Safety Goals (DO/RPS)

11) 11:00 A.M. - 1:00 P.M. General Electric Standard Safety Analysis Report (GESSAR II)(Open)
- 11.1) Report of ACRS Subcommittee regarding the containment integrity for this type facility (DO/RKM)
- 11.2) Meeting with representatives of the NRC Staff and the Applicant, as appropriate

1:00 P.M. - 2:00 P.M. LUNCH

12) 2:00 P.M. - 4:00 P.M. EPA Standards for HLW Repository (Open)
- 12.1) Report of ACRS Subcommittee (DWM/OSM)
- 12.2) Meeting with representatives of the NRC Staff and the EPA, as appropriate

13) 4:00 P.M. - 5:45 P.M. ACRS Subcommittee Activity (Open)
- 13.1) Report of ACRS Subcommittees regarding:
  - 13.1-1) 4:00 P.M.-4:45 P.M.: Air Systems - NRC Staff/ANL-West Survey Control Room Habitability Practices (DWM/JOS)
  - 13.1-2) 4:45 P.M.-5:15 P.M: Long-Range Plan for NRC (MWC/JCM)

13.1-3) 5:15 P.M.-5:45 P.M.: Human Factors and Maintenance Subcommittees on Natural Aptitude Selection Procedures (GAR/DAW/JOS)

Saturday, July 13, 1985, Room 1046, 1717 H Street, NW, Washington, D.C.

14) 8:30 A.M. - 12:30 P.M. Preparation of ACRS Reports to NRC (Open)

- EPA standards for HLW repository (DWM/OSM)
- GESSAR II containment integrity (DO/RKM)
- Quantitative Safety Goals (DO/RS)
- Diablo Canyon Nuclear Plant - Ten-year seismic review (CPS/EGI)
- Indian Point - Implementation of PRA (DO/RPS)
- Provisions to preclude sabotage at nuclear power plants (JCM/JOS)

12:30 P.M. - 1:30 P.M. LUNCH

15) 1:30 P.M. - 3:30 P.M. ACRS Subcommittee Activities (Open/Closed)

15.1) Report of ECCS Subcommittee on the MIST facility (DAW/PAB)(Open)

15.2) ACRS Procedures - Proposed changes in ACRS Bylaws regarding conduct of members and procedures for revision of the ACRS Bylaws (DAW/TGM) (Open)

15.3) Reports by ACRS members regarding meeting with the RSK and the GPR (DAW, et al/TGM)(Closed)

(Note: Portions of this session will be held in closed session as required to discuss information provided in confidence by a foreign source.)

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FOIA EXEMPTION (c)(4)
PROPOSED MINUTES OF THE
303rd ACRS MEETING
JULY 11-13, 1985
WASHINGTON, D.C.

The 303rd meeting of the Advisory Committee on Reactor Safeguards, held at 1717 H Street, N.W., Washington, D.C. was convened by Chairman D. A. Ward at 8:30 a.m., Thursday, July 11, 1985.

[Note: For a list of attendees, see Appendix I. C. Michelson did not attend the meeting.]

Chairman D. A. Ward noted the existence of the published agenda for this meeting, and identified the items to be discussed. He noted that the meeting was being held in conformance with the Federal Advisory Committee Act and the Government in the Sunshine Act, Public Laws 92-463 and 94-409, respectively. He also noted that a transcript of some of the public portions of the meeting was being taken, and would be available in the NRC's Public Document Room at 1717 H Street, N.W., Washington, D.C.

[Note: Copies of the transcript taken at this meeting are also available for purchase from Ann Riley & Associates, Ltd., 1615 I Street, N.W., Suite 921, Washington, D.C. 20006.]

I. Chairman's Report (Open)

[Note: R. F. Fraley was the Designated Federal Official for this portion of the meeting.]

D. A. Ward mentioned that ACRS Member D. Okrent was awarded the NRC Distinguished Service Award for 1985. The NRC Meritorious Service Award was given to Carol Ann Rowe, Secretary to the ACRS Executive Director and a 30 Year Pin was awarded to J. C. McKinley, Branch Chief for Branch No. 1. Brief mention was made of the reorganization of upper management at TVA and problems at the Watts Bar, Sequoyah, and other TVA nuclear plants. D. A. Ward also mentioned the Idaho Loft Reactor partial meltdown experiment, which attempted to replicate the 1979 Three Mile Island accident.

II. Watts Bar Nuclear Station (Open)

[R. K. Major was the Designated Federal Official for this portion of the meeting.]

G. A. Reed indicated that a joint meeting of the Watts Bar and the Quality and Quality Assurance in Design and Construction Subcommittees was held on June 26th to discuss the quality assurance breakdown in the construction program at Watts Bar. Quality assurance problems were mentioned in the August 16, 1982 ACRS OL Report. The Committee requested that it be kept informed of the situation. He noted that TVA has invoked a major quality

assurance program including an independent contractor review of the design and construction of a typical vertical section of the plant to confirm the adequacy and safety of the as-completed plant. The Black & Veatch Engineering Company did an independent vertical slice QA review on the auxiliary feedwater system at Watts Bar Unit 1. Some 428 inconsistencies were catalogued of which 165 were due to the fact that the auxiliary feedwater system was still under construction at the time of the review. TVA's inhouse Independent Review Policy Committee and Black & Veatch decided that 115 of the inconsistencies or findings were not really deviations. There remained 148 findings which required further generic consideration.

G. A. Reed indicated that the Subcommittee had difficulty at the outset deciding whether the Watts Bar issue was one of quality assurance documentation or the quality of equipment installed in the workplace. As a result, there was a focus on the Black & Veatch presentation. The Black & Veatch representative indicated that Watts Bar Unit 1 had been designed and had been installed in accordance with TVA's licensing commitments with the possible exception of three identified unresolved issues. The first of these three issues involves problems with cable tray loadings and fire retardant coatings that were put on the cables. The second issue involves loadings on embeddment plates with respect to restraints and other attachments. The third issue deals with seismic design and the way in which the peaks were broadened on the seismic spectra.

G. A. Reed explained that TVA's Board of Directors appointed a Nuclear Safety Review Staff (NSRS). This group decided on its own to review the Black & Veatch report. The NSRS group assessment concluded that from the work done by Black & Veatch there were no issues that would preclude immediate approval for operation of Watts Bar Unit 1. He indicated that the NRC Staff at the Subcommittee meeting was not prepared to agree or disagree with TVA or its NSRS group or with the Black & Veatch report. The NRC Staff indicated they had just received a draft of an INPO Design and Construction Evaluation Report on Watts Bar but were not prepared to comment on the report or the status of release of the report. Also mentioned was TVA's solicitation of allegations from construction workers and other regular employees to be evaluated by an outside firm named Quality Technology Company. G. A. Reed cautioned that the Watts Bar QA issue be kept separate from generic problems regarding TVA management in order that a manageable review could be conducted on the quality assurance issue.

G. A. Reed called for discussion of the INPO Construction Evaluation Report. E. Adensam, NRC, indicated that the Staff has only a set of INPO field notes which are part of an exit interview dated June 20, 1985. She indicated that the Staff does not have a copy of the draft INPO Report and is not prepared to discuss the INPO field notes at this meeting. G. A. Reed noted that the INPO Report appears harsh as expected since INPO is attempting to encourage licensees to seek excellence in the workplace. TVA must now respond to this report or these criticisms will stand.

C. J. Wylie agreed that the Committee must separate the quality assurance issues from other TVA problems in order to reach some closure on the quality assurance issue. He noted that the issue involving the cable tray loading is a common problem on all construction sites where one does not give attention to the installation of cables on a continuing basis. He thought that the overfilling of the cable trays by the field installers is more of an issue of sloppy installation than a safety issue.

W. Kerr asked with which one of the issues the Committee should be concerned. G. A. Reed stated that the Committee should be concerned whether TVA met the requirements the ACRS laid down in its April 16, 1982 letter and whether the QA activity at Watts Bar is satisfactory to the ACRS. The Black & Veatch vertical slice on the auxiliary feedwater system indicated that a big problem did not exist for TVA from a quality and quality assurance standpoint. However, the new INPO report, which is a general review of quality assurance in design and construction, does seem to add further confusion with regard to the QA issue. C. P. Siess pointed out it was his belief that Black & Veatch focused on the quality of the product as well as the quality assurance paperwork problem. He indicated that the fact that they did not find any deficiencies in the design is very positive. Despite the possible paper problems, he stated that he was confident that significant deficiencies in the design are not present.

E. Adensam, NRC Licensing Branch Chief, in charge of licensing on Watts Bar, contrasted the Black & Veatch report with the detailed Staff review of it. She indicated that the Staff is satisfied that Black & Veatch fulfilled its requirements in the study, and that the findings are reasonable. This included one of three findings still under review bearing on Bulletin 79-02 which has to do with base- plate flexibility and anchor bolts.

E. Adensam noted that half of the Staff's review effort involved the TVA Independent Review Policy Committee generic review of the Black & Veatch findings to consider the implications for the rest of the plant and other safety systems that might be impacted by the Black & Veatch findings. As a result of some of the concerns that have been raised recently with regard to the report, the Staff has decided that further work is necessary. A dedicated review group composed of representatives from Region II and NRR has been established to determine if TVA did a good job in addressing the findings of the Black & Veatch Report regarding their generic applicability and corrective actions on the plant design and construction. This dedicated review group will not only close out the IDVP review, but will also address allegations received regarding the Black & Veatch review (see Appendix IV).

E. Adensam indicated that the TVA Policy Committee established a TVA Task Force to do the actual work regarding review and generic application of the Black & Veatch work. Because of allegations regarding the Black & Veatch IDVP, TVA's Nuclear Safety Review Staff looked at the Black & Veatch documents independently, the

Policy Committee report, and the work of the TVA Task Force and made recommendations to the TVA management regarding corrective actions to be taken. The Staff intends also to look at all reports and related documents that address the allegations.

E. Adensam indicated that the Staff had requested that the licensee provide additional documentation and the answer to certain questions to resolve Staff concerns prior to preparation of an SER. She mentioned a May 16, 1985 Staff letter to TVA, with two enclosures which outlined the allegations the Staff has received (see Appendix V).

G. A. Reed noted that the proliferation of committees reviewing the Black & Veatch study is making it difficult to determine who is responsible for resolving the issues and allegations. He suggested that authority for the entire review effort ought to be placed in the hands of one qualified individual. He noted that the ACRS, because of the status of the current situation, is not in a position to make any judgments at this time. F. J. Remick agreed that the ACRS could not profitably contribute to the process at this point. He pointed out that the 1982 ACRS letter asked that the Committee be kept informed and the Staff is doing just that. Certainly the ACRS is not the one to investigate allegations. He suggested that further action ought to be deferred to further developments. D. A. Ward indicated that the next step appeared to be review of the Staff conclusions after the Staff has completed its review. He asked for the Staff's schedule for completion of its review. E. Adensam indicated that the Staff has had some difficulty with scheduling because TVA has not submitted a fuel load date. In answer to a question by C. P. Siess, E. Adensam indicated that the Staff will definitely supplement the SER concerning the IDVP.

D. A. Ward then indicated that the next step appeared to be review of the Staff's SER when it is in a final draft. He thought that the decision about another Subcommittee meeting or just full Committee discussion could be determined at a later date. The Committee briefly discussed the method the Staff will use to evaluate and resolve the allegations on Watts Bar. E. Adensam indicated there is an action office in NRR that will have responsibility for resolving allegations and reporting back to the alleger if that individual can be identified. C. P. Siess noted that a more formal organized approach was taken to allegations on Diablo Canyon and Waterford 3. E. Adensam implied that if the number of allegations reached a very large, hard to manage number, a different procedure than the action office would probably be used.

III. Recent Events at Operating Nuclear Power Plants (Open)

[Note: H. Alderman was the Designated Federal Official for this portion of the meeting.]

A. Davis-Besse Loss of Main Feedwater and Auxiliary Feedwater

E. Jordan, Division Director, Emergency Preparedness and Incident Response, I&E, indicated that the NRC Staff has drafted a Commission Paper, SECY-85-208, entitled "Incident Investigation Program" which proposed to the Commission a manner for dealing with operating reactor events. It is a response to the Brookhaven study on the need for an NTSB-type board for investigating nuclear plant accidents and ACRS findings. The Staff advised the Commission of its intention to proceed, for the interim, to use this method as a response to the Davis Besse event on June 9, 1985. A multi-disciplinary team was established made up of technical experts from various NRC offices. A memorandum from the EDO identified E. Rossi as team leader, (reporting directly to the EDO) J. T. Beard of NRR, L. Bell, I&E Training Center, and W. Lanning, AEOD, as experts in the various areas that seemed appropriate for this particular investigation. This special investigative team will prepare a single report which will focus on fact finding, identify the root causes, and provide findings and conclusions. Recommendations would not be part of this report but would be developed subsequently by the program office responsible for the various areas. He indicated several steps in the investigation at Davis-Besse including the fact that statements were taken from plant personnel involved on shift at the time of the event:

- Strip chart records were reviewed
- Procedures, logs and manuals were reviewed
- Equipment involved was inspected

He noted that the program is being administered by AEOD despite the fact that E. Rossi reports directly to the EDO in terms of findings.

A. DeAgazio, NRR Project Manager for Davis-Besse, described the feedwater system at the Davis-Besse nuclear plant. He explained that the Davis-Besse plant has two once-through steam generators, two steam-driven main feedwater pumps which provide flow to either steam generator, and two turbine-driven feedwater pumps that provide auxiliary feedwater flow to either one or both of the steam generators depending on valve alignment (see Appendix VI). He noted that this plant also has a small capacity electric motor-driven startup feedwater pump normally used just during startup and latter stages of plant shutdown (the capacity of the electric motor pump is less than that of one turbine-driven auxiliary feedwater pump). [Since the characteristic curve of the electric startup feedwater pump is somewhat flat, it is difficult to say whether it has half the capacity of an auxiliary feedwater pump, but it can be assumed that it has no more than half that capacity depending upon the pressure of the steam generator.] J. C. Ebersole noted that at the time of the incident, plant management contemplated installation of additional pumps at some points in the future. A. DeAgazio indicated that there is a license condition in effect that would require that Davis-Besse provide a new startup feedwater pump in a new location with a capacity equal to one of the auxiliary

feedwater pumps. This new startup feedwater pump is not in the plant yet. It is scheduled for installation during the next refueling outage which would nominally be the spring of 1986.

A. DeAgazio indicated that just prior to the event the plant was operating at 90 percent power with one main feedwater pump on automatic operation and one main feedwater pump in manual control. The reason for this was that they were experiencing difficulties with the speed governors on the main feedwater pumps. Both of the feedwater speed governors had been replaced with new models at the last refueling outage. The initiating event was the tripping of the main feedwater pump in automatic on overspeed. The reduction in feedwater flow initiated a power run back and approximately 30 seconds later the power level was down to 78 percent. Nevertheless, the run back wasn't fast enough and a high pressure reactor trip occurred (see Appendix VII). The main steam isolation valves then tripped with the effect of stopping all steam flow from the steam generators since one main feedwater pump running in manual was deprived of steam. It coasted down on steam stored in the system beyond the isolation valves and, at approximately four minutes, it tripped. He noted that there should not have been a signal from the steam and feedwater rupture control system at that time which closed the main steam isolation valves since steam generator levels were normal. This signal provides for starting of the auxiliary feedwater pumps in the event of low steam generator water level and provides actuation of the auxiliary feedwater system in the event that all four reactor coolant pumps are lost (to promote natural circulation).

H. W. Lewis asked why the main steam isolation valves closed at this point. A. DeAgazio indicated that the Staff does not know exactly why this occurred. He noted that if there is a steam line break, both main steam lines are isolated. If there is a low steam pressure trip on the steam and feedwater rupture control system, the main steam isolation valves close on both steam generators. It does not matter which steam generator has the low pressure.

A. DeAgazio discussed the feedwater flow paths on a low steam generator water level or on a high feedwater to steam pressure differential indicative of a feedwater line break (see Appendix VI).

A. DeAgazio indicated that six minutes into the event there was an actual low level steam and feedwater actuation signal generated for steam generator number one. The operator, recognizing the steam generator water level was dropping, attempted to initiate auxiliary feedwater and not depend upon the steam and feedwater control system. He activated the Steam Feedwater Rupture Control System (SFRCS) on low pressure instead of low water level. By activating the SFRCS on low pressure, the SFRCS was signaled that both generators had experienced a steamline break or leak and the system responded, as designed, to isolate both steam generators. About one minute later, the operator recognized the error he had made and attempted to correct it but two auxiliary feedwater isolation

valves failed to open. The operators were then dispatched throughout the plant to attempt to restart these pumps. An attempt was made to get the electric motor-driven startup feedwater pump in operation and to replace some control fuses which were pulled to prevent starting of this pump against a closed suction valve. The operators were successful in restarting the startup feedwater pump and also successful in resetting the turbine-driven feedwater pumps. Thus, at 16 minutes into the event, the Davis-Besse operators had begun restoring feedwater flow to the steam generator.

A. DeAgazio indicated that, just prior to restoring feedwater flow to the steam generators, the pressure in the reactor coolant system rose to the point where the PORVs were actuated. They actuated three times, and there was indication that the PORV failed to reseat after the third actuation. There is some question whether this was an actual failure of the valve. The latest information from the site is that the valve has been disassembled and nothing abnormal found. The reason why the PORV stuck open, if it did stick open, is not known. Once the feedwater had been restored the plant entered a normal cooldown.

A. DeAgazio indicated that there were at least 13 or 14 different malfunctions or failures or unexpected occurrences. The initiating event was the main feedwater trip on overspeed (see Appendix VIII). H. W. Lewis suggested that the fact that there were nine independent failures of components in the sequence that followed the tripping of one main feedwater pump suggests that there are serious implications from this event. J. C. Ebersole suggested that this is a milestone occurrence in the context that one may have to be somewhat less optimistic about the usefulness of PRAs regarding the probability of this combination of events. H. W. Lewis noted that there is zero probability that there would be ten independent failures. He suggested waiting until all of the findings are out. H. W. Lewis hoped that the Staff's incident investigation team would be able to get at the root cause of this event.

J. C. Ebersole asked how many minutes were left before core damage would have occurred from core uncovery. B. Sheron, NRC, indicated that hand calculations were done and more sophisticated calculations with RELAP 5 will be reported to the ECCS Subcommittee on July 31st. He suggested that if the operator had taken no action whatever to turn on the makeup pumps or startup feed pumps or open a PORV, core uncovery would have occurred in about 50 minutes. The steam generators will normally dry out in a few minutes under these conditions. This scenario involves the displacement of water by steam collecting in the high points until the vent path is uncovered which is a surge line to the pressurizer. It is then just a case of boil off. It was determined that if both makeup pumps were started, the PORV opened, and the startup feedwater pump started, there was sufficient capacity to remove decay heat and the core would have stayed covered. Starting just the two makeup pumps and opening the PORV

would still have kept the core covered. Startup of one makeup pump would have uncovered the core, but would have extended the time considerably. He pointed out that the event occurred at 90 percent power and the calculational results would have been different at 100 percent power. If the plant had been operating at 100 percent power and just two makeup pumps and no startup feedwater pump had been initiated at the time of steam generator dry up (two or three minutes into the event) there would not have been core uncovery, but had the operators waited 20 minutes to initiate the makeup pumps core uncovery would have occurred.

G. A. Reed indicated that he was surprised at the estimate of 50 minutes until core uncovery. He suggested that this was because this particular B&W plant has high set steam generators versus other B&W plants with low set steam generators. One gets the drain down advantage of the loop piping on the steam generator primary side. B. Sheron indicated that the volume of the core is about the same in both cases, such that the raised loop plant does not behave much differently then the lowered loop plant in terms of time to core uncovery. Also a factor is the relative location of the surge line on the hot leg. G. A. Reed noted only one PORV on the plant had an in-series block valve. B. Sheron noted that all B&W plants have only one PORV. He indicated that the PORV, HPI or makeup system was not originally designed for feed and bleed, and the PORVs were installed in these plants to protect the lifting of safety valves. Obviously, a plant that has a higher PORV capability with more PORVs will have a much higher capability of feed and bleed.

J. C. Mark took note of the fact that except for the one operator error during the succeeding hectic 15 minutes in the control room, the operators did everything about right and very quickly. A. DeAgazio indicated that from a control room design review of the incident, the one operator error was attributed to a human engineering defect.

W. Kerr indicated that he thought the auxiliary feedwater system was an automatic start system. A. DeAgazio indicated that the auxiliary feedwater system is automatically started by the steam and feedwater rupture control system. W. Kerr asked why the operator who punched the wrong buttons had to punch any buttons at all. A. DeAgazio indicated that this is a manual action followup in most plant procedures to confirm that an automatic action has taken place. G. A. Reed agreed that it appeared that the operators did a good job except for the one error. However, he expressed concerns regarding several design errors, such as the exclusive use of steam-driven pumps as auxiliary feed pumps and a very complex valving arrangement to introduce water into one or the other of the steam generators. He also noted his concern regarding the CESSAR system-80 plants and the large number of closed valves in the valving arrangement at Davis-Besse. He noted that valves have a habit of not functioning and are often quite unreliable. The systems would be better off without closed valves from a reliability point of view. J. C. Ebersole added that the

turbine-driven pumps are cooled by ac-driven environmental cooling fans which makes them interdependent with the ac system. An electric power failure or a steam failure, or any other kind of failure, can remove these pumps from service.

E. Jordan complained that it was 36 minutes after the turbine trip that the NRC was advised with cursory information on this event. W. Kerr thought it more important that the operators look after the accident then inform the NRC Information Center. E. Jordan indicated that he had to insist on both of these items since the NRC would not have been in a position to initiate emergency response capability and make determinations as far as offsite measures, had the event gone awry. In answer to the question by G. A. Reed, E. Jordan indicated that the Davis-Besse operators met the regulatory requirements which called for a response within one hour.

B. Hatch I Stuck Open Safety Relief Valve

G. Rivenbark, NRR, indicated that on May 15, 1985, a crane passing overhead ruptured a line that provides water pressure to the charcoal filter deluge valve (the fire protection valve). This pressure which normally kept the valve closed caused the valve to open and spray water into the charcoal filters. The operator discovered the problem 15 or 20 minutes later when water began dripping into the control room through the air conditioning ducts. J. C. Ebersole asked how the crane which is out in the turbine hall got involved in this incident. G. Rivenbark indicated that the crane was passing from one turbine building to the other turbine building and passed over the control room. The floor immediately above the control room is the location of the air conditioning equipment.

G. Rivenbark indicated that as a result of the water dripping into an instrument panel in the control room, one of the SRVs opened several times and closed and finally stayed open. The operator tripped the reactor and started feedwater pumps and quickly recovered the reactor water level that had begun to drop. While the operators spent about 30 minutes attempting to close the SRV by pulling incorrectly labeled fuses, the SRV closed by itself. J. C. Ebersole noted that the placement of the deluge system with a potential for straight drop down into the control room from a water pipe is almost akin to the classical scenario of having a toilet on the ceiling which overflows after you flush it. G. Rivenbark agreed that if the drains plug up, the water will flood into the charcoal filter box and will run over and leak into the control room. J. C. Ebersole thought that the licensee should get the water ingress potential completely away from the panel board where the safety relief valve was electrically connected, not just to clean the plugged drains. G. Rivenbark indicated that the utility is looking at the possibility of removing the water altogether from the area.

C. Oyster Creek Scram Discharge Volume Isolation Valves Failure

D. Powell, I&E, indicated that the event at Oyster Creek which occurred on June 12, 1985, is of interest because it mimics an event that occurred at Hatch in August 1982 where uncontrolled leakage occurred out of the scram discharge volume drain valves. Following a scram signal on high drywell pressure while the operators tried for 38 minutes to clear the scram signal, two particular valves failed by different failure mechanisms. One valve failed to go fully shut, or was hypothesized to have been originally fully shut and to have been forced open due to the pressure build-up from the leaking valve. This valve had an improperly sized spring in the valve actuator. Reactor coolant was discharged to the reactor building drain tank. Release of steam from floor drains and blistering paint on a hot pipe caused a portion of the reactor building deluge system to activate. There was no damage to the equipment inside the reactor building due to the actuation of the deluge system or due to the steam. There was some radioactive contamination, mostly short-lived radionuclides, at the 23-foot level. The potential significance of this event was the uncontrolled leakage of radioactive coolant outside of the containment. D. Powell noted that the licensee had CRD seal high temperature alarms which respond intermittently. The problem of CRD high temperature alarms is not significant for this particular event but for later operation of the plant. Should those seals degrade due to the high temperatures, there is the possibility that they would be unable to operate properly. He noted that the failed valves were categorized as B valves under ASME Section 11 and were never leak-rate tested after installation of the backfit modifications.

J. C. Ebersole pointed out that there is controversy about the potential leakage from the scram dump volume even due to metallurgical failure of the volume proper, not to mention valve failure as being a potentially serious event. These valves should be recognized as safety-related valves. He asked if this was a standard problem for all BWRs. D. Powell indicated that probability studies done on the system show that the probability is not high for failure of the piping compared with other types of events that would lead to core damage. J. C. Ebersole indicated that the probability of piping failure was virtually zero, but the analysts ignored the presence of the valves which made an aperture through the pipe. J. C. Ebersole asked if the Staff intends to upgrade these valves. D. Powell indicated that NRR was not pursuing the matter at this point. E. Jordan indicated that the Staff will probably issue an information notice and then make a subsequent decision on whether action is needed.

J. C. Ebersole asked how the utility stopped the leakage. D. Powell indicated that they basically blew down the system by starting up the reactor water cleanup system and bled the letdown portion of that to the condenser.

D. Rancho Seco Reactor Coolant System High Point Vent Leak

H. Wong, I&E, indicated that on June 23, 1985, while in hot shutdown, the Rancho Seco plant had a 20 gallon per minute nonisolable primary coolant system leak on the high point vent system on the B steam generator. He described the RCS loop top portion of the candy cane and the additional piping which was the original design. The cause of the leak appeared to be missing supports, and fatigue failure as a result of an RCS vent line addition. Two additional pipe supports had to be modified and the addition of one cross-brace member was required. Investigation reveals that these support changes had not been performed although records stated that work had been done and inspected (see Appendix IX).

H. Wong indicated that what was important was not so much the additional pipe that was added as a TMI modification, but the removal of the spool piece in the middle which was to provide that the nitrogen system would not be contaminated during operation. The spool piece was planned for removal during operation and would be put in place for nitrogen blanket purposes during outages. The purpose of this was to transmit loads across both sides of the pipe. The dummy spool piece was designed so that it would go back during operations basically giving rigidity to the pipe so that loads would be transmitted properly. Without the spool piece, the pipe was hanging as a cantilever. In 1983 the stainless to stainless weld made at the time of the TMI 2 change cracked. P. G. Shewmon asked what kind of administrative action is likely to be taken against the person who signed off that these pipes were in place and the hangers were in proper order. H. Wong indicated that he was not aware of any licensee actions.

E. Sequoyah 2 Reactor Trip - Improper Use of Test Instrument

E. Weiss, I&E, indicated that on May 22, 1985, Sequoyah Unit 2 tripped from 100 percent power on overpower Delta T. This event demonstrates how a plant can trip following an approved procedure despite all the precautions in place to prevent maintenance or surveillance activity from causing this sort of event. An instrument technician had to take temperature readings from four protection cabinets located approximately 15 feet apart using a digital voltmeter. The voltmeter leads were incorrectly connected to the ammeter sockets in the voltmeter with the result that the internal resistance of the voltmeter was much lower than it should have been. The technician had to go to all four protection sets within three minutes. He tripped one channel and although the reactor operator in the control room noticed the trip it all happened too quickly for him to respond. The plant went down on two out of four coincidence.

E. Weiss mentioned a similar event where a digital voltmeter caused a shorting of the output transistors on a reactor protection system. An information notice was issued on that subject. The corrective actions taken for this particular event include a precaution about procedures. One must look for the proper expected value of voltage before proceeding to the next piece of equipment.

J. C. Ebersole suggested that TVA ought to outlaw multi-purpose meters for its personnel. W. Kerr thought it might be best to do this sort of testing when the plant is not operating. J. C. Ebersole noted that this was a full power thermal measurement, a thermal heat balance value which has to perform during operation.

IV. Diablo Canyon Nuclear Plants Unit 1 and 2 (Open)

[Note: E. G. Igne was the Designated Federal Official for this portion of the meeting.]

C. P. Siess explained that a requirement for a long term program to reevaluate the seismic design basis for Diablo Canyon was included as a licensed condition for Unit 1 primarily as a result of concerns expressed in a July 14, 1978, ACRS letter. The letter recommended that the seismic design of Diablo Canyon be reevaluated in about 10 years, taking into account applicable new information. The license condition required reevaluation of the geology, seismo-tectonics, earthquake magnitude, the ground motion of the site, and earthquake engineering. The Licensee, Pacific Gas & Electric Company (PG&E), issued a plan in January 1985 which the Staff has under review. The ACRS discussed the plan with the Licensee and their consultants at a subcommittee meeting near Los Angeles on March 21. The ACRS consultants viewed the plan, in general, with favor. There were no significant concerns expressed. A joint meeting of the Extreme External Phenomena and Diablo Canyon Subcommittees was held on July 10, to hear the Staff evaluation of the PG&E plan. In the several months between the time the plan was submitted and the time the Staff finished its review, the Staff submitted a list of comments, questions and suggestions to PG&E. PG&E responded to all of them. The Staff concluded that the program was acceptable and issued a draft of its evaluation for ACRS comment. ACRS consultants were particularly pleased that Dr. Slemmons of the U. S. Geological Survey would be working with the Staff. S. Brocoum, NRC GeoSciences Branch, indicated that the Staff will hear progress reports from PG&E quarterly and meet with them at a minimum every six months.

C. P. Siess indicated that the Subcommittee on July 10, found the program acceptable with one exception. The NRC has requested that PG&E assess the significance of the conclusions drawn from the seismic reevaluation studies with a probabilistic risk analysis as well as deterministic studies as necessary to assure the adequacy of seismic margins. D. Okrent has taken exception to the Licensee's proposal to do a Level 1 PRA (level of damage states only). He indicated that the Staff is satisfied with a Level 1 PRA as a basis for assessing the significance of whatever conclusions may arise regarding ground motion. He proposed that the ACRS accept the recommendations of the Subcommittee regarding the program plan and leave the question of the PRA level to a discussion with the Staff and the Licensee.

S. Israel, NRC, discussed the PRA analysis process. He indicated that the PRA analysis "front end" starts off with plant systems

analyzed in event-tree, fault-tree fashion to determine those sequences that will lead to core melt. This analysis yields various plant damage states with information about the core condition, core melt, and the condition of the containment. Of particular interest in a seismic analysis would be containment failure prior to core melt which would occur if the earthquake failed the containment. Other plant damage states of interest would be core melt without any core cooling, which usually occurs as a result of loss of all electric power, both offsite and onsite. There are other damage states dealing with transients and LOCAs which basically are early core melts (large LOCA without core injection) or later core melts due to small pipe breaks or transient situations where one loses core cooling and ultimately the core fails. Of concern is the availability of containment cooling which would be determined by an analysis of plant systems in the fault-tree process. This is basically a Level 1 PRA, a plant damage state analysis. A Level 1 PRA does not include containment failure modes. A Level 2 PRA tries to characterize what happens to the containment, given one of the plant damage states. One is concerned about where the potential containment failures occur and how they occur. Potential containment failure modes would include steam explosions (the vessel explodes and penetrates the containment), early hydrogen burns, late hydrogen burns, overpressurization failures of the containment as well as basemat melt through. The split fractions are usually characterized by a release category. A source term is then calculated for each one the release fractions. The Level 3 analysis, a consequence analysis, uses the source terms for the containment failure modes for determining offsite consequences. S. Israel explained that the Staff has found, in terms of offsite consequences, that early fatalities are dominated by containment failure prior to core melt. Those types of sequences which include containment failure prior to core melt are basically the ones associated with a seismic event where the earthquake rocks the containment away from the auxiliary building or fails the containment. One could potentially cause early fatalities from long term containment failure usually through a station blackout sequence (core melt without containment cooling) where containment pressure builds up over a period of time and the containment fails releasing radioactive materials to the environment.

S. Israel indicated that one of several reasons why the Staff did not require the Licensee to extend his analysis beyond Level 1 is the conflict between the old source term used in WASH 1400 and the new source term being developed. Another reason that the Staff did not require more than a Level 1 PRA is the fact that such analyses can become very expensive. Such analyses will require sophisticated computer calculations, phenomenological calculations in terms of containment loadings, sensitivity studies dealing with core debris coolability, as well as core-concrete interactions. He stressed that the purpose of the Diablo Canyon Study is to investigate seismic characteristics of the site as opposed to other PRA work done to determine containment potentials. A third reason the Staff does not wish to require PG&E to do more than a Level 1

PRA is the fact that traditionally the Staff has performed back-end analyses for PRAs and even though the licensees or applicants have submitted back-end analyses, the Staff does audit calculations anyway. It is like paying twice for the same work. Another reason for only requiring a Level 1 PRA is the fact that the Diablo Canyon site is a low population site, and the potential societal risks are not expected to be limiting. The thrust of the Diablo Canyon study is to deal with seismicity and geology and not resolve the ongoing source term severe accident work.

D. Okrent explained that the Diablo Canyon study would calculate accident states but would not evaluate containment failure modes except those that resulted from the earthquake itself. They would not evaluate the containment capability to withstand pressure and temperature nor would they evaluate whether the design basis, or as designed capability to withstand pressure and temperature in any way, is weakened by a severe earthquake. There would therefore be a gap in their ability to assess the likelihood of different release categories. Obviously, the containment capability is vital to this analysis.

D. Okrent pointed out that the Diablo Canyon plant has a different configuration from other plants for which PRAs have been done and unless the Diablo Canyon study proceeds beyond Level 1, the Staff will be in a poor position to make estimates of releases, of the mode of containment failure, and the frequency of containment failure release categories. He suggested that the Licensee with a better knowledge of the plant is best qualified to do the Level 2 work. D. Okrent suggested that his estimate for doing the additional work for Level 2, despite the confusion about the source term, as well as estimates for the increased analysis costs is more conservative than that calculated by the Staff. He suggested that the Staff's position is wrong and he thought that the Committee ought to recommend a Level 2 PRA for the Diablo Canyon Plant.

D. Okrent made the point that a core melt guideline is not adequate without looking at risks. It is an incomplete look even though it is relevant to know the core melt frequency. He indicated that if one could get an estimate of containment release frequency, one would have the essential information for estimating the safety of the reactor.

C. P. Siess indicated that he was in a position to draft a letter which indicated that the Committee approved the PG&E program plan and that the Committee agrees with the proposed Level 1 PRA that the Staff has accepted. Committee members could append additional comments. The letter could also recommend going beyond Level 1 to Level 2 or Level 3 for the PRA depending upon the sense of the Committee. P. G. Shewmon pointed out that the only thing missing from the Diablo analysis is containment performance. D. Okrent indicated that if you do a PRA you want to estimate the releases from the containment. You have to evaluate the capability of the containment to withstand pressure and temperature. You do that in a Level 2 and not in a Level 1 PRA.

V. General Electric Standard Safety Analysis Report (GESSAR II) (Open)

[Note: R. K. Major was the Designated Federal Official for this portion of the meeting.]

D. Okrent indicated that last month's full Committee meeting on GESSAR II discussed the subjects of containment capability and hydrogen but was not able to explore these subject in sufficient depth.

Rom Vij, GE, presented a comparison of containment design pressures, ultimate pressure capability (calculated by different architect engineers), dome configuration, containment diameter, and containment fill at the base for GESSAR-II and several other BWR plants. He indicated that GESSAR-II is a free standing steel containment backed by a concrete shield building. The design pressure is 15 psig (the ultimate pressure capability calculated to be 85 psig). The dome configuration is torispherical with a containment diameter of 120 feet. He pointed out that GE developed a methodology (given an energy dump after a break inside the drywell) to calculate pressure and temperature transients and the design pressures and temperatures on the drywell in the containment. GE has suggested an equipment layout for the containment. The actual size dimensions set for GESSAR-II are not variable. GE has not provided the detailed structural design for every containment and this accounts for some of the differences in the diameters of the Grand Gulf, Perry, River Bend, and Clinton plants. R. Vij indicated that the detailed structural design is provided for the GESSAR-II containment and therefore the architect engineer is not allowed any flexibility.

R. Vij addressed the issue of fabrication flaws in the containment. He indicated that GE's fracture mechanics group performed an analysis at the point of maximum stress in the containment for the ultimate pressure capability. It turned out to be in the knuckle region of the dome. The lower bound fracture toughness properties of the material were input to the analysis and then the plate material of the containment dome, the weldment material and the heat effected zone around the welds were followed for resultant crack formation. They picked out the most conservative value which will give the smallest flaw and it came out to be the plate material. The fracture analysis results show that a potential crack, which is three inches long and a half inch deep, can be tolerated without unstable propagation to failure. This half inch depth is more than 25 percent of the wall thickness. He contended that with controls on the welding procedures, including preheating and multi-pass to improve the impact properties of the weldings, GE does not expect to miss flaws more than about 10 percent of the wall thickness during inspections. All the welds are 100 percent radiographed on the outside where a flaw of up to two percent depth of the wall can be detected. Therefore, GE believes that undetected flaws should not compromise the calculated ultimate capability of the containment.

J. C. Ebersole inquired regarding the viability of containment penetrations under high temperatures and pressures. R. Vij indicated that this sort of detail was not considered in the fracture analysis. D. Okrent asked what the possibilities are of missing a larger flaw and the likelihood of the vessel failing at lower pressure from this flaw. R. Vij indicated that the carbon steel in the welds is considered stronger than the base metal by GE's metallurgists. He assured the Committee that there was no way that flaws of a large size could be missed, either in the inspection of base metals which are hard-rolled metal plates or during welding where there are at least 10 or 12 passes in one and three-quarter inch thick plate. The 100 percent radiography will pick up two percent angular flaws. D. Okrent asked the Staff whether, after radiographic analysis, flaws have been found larger than 10 percent of the wall thickness. J. Knight, NRC, indicated that the Staff really does not have the experience base to answer than question. But, he assumed that it is quite possible that one could initially have had flaws larger than 10 percent of the wall thickness ground out and corrected. D. Okrent indicated that he was trying to determine whether there is a significant risk of vessel failure due to the presence of larger flaws perhaps at a lower pressure. J. C. Ebersole suggested that perhaps the x-raying of welds should be a confirmatory process rather than an initial survey for finding defects. R. Vij indicated that radiography is a confirmatory process since the test results are used by insurance companies and the architect engineers, as well as the NRC. R. Klecker, NRR Division of Engineering, indicated that the Staff has made similar flaw calculations using a different analytical model and has found that one could tolerate roughly a two inch long through-wall crack or its equivalent crack area if you allow it to go a little longer. He stressed that any crack that may pre-exist should be found during an inspection even if one puts a margin of two or three on the crack size for uncertainties. J. C. Ebersole asked what would happen if the crack were to start later after fabrication. R. Klecker indicated that the Staff does not know of any mechanism that would cause a crack to grow at a later time because the containment is essentially only supporting its own weight. D. A. Ward asked if there is a high probability that a crack would be found in normal inspections. R. Klecker thought that the probability would be quite high because of the requirements of the ASME code. The allowable crack sizes are considerably smaller than what GE, BNL, and the Staff are calculating to be close to the critical crack size. He thought there was a sufficient margin so that the chances of missing a crack would be very small. The Committee discussed the detection of through-wall cracks. J. Rosenthal, NRC, indicated that the Staff calculates late containment failure (assuming no flaws) of the order of 11 to 28 hours into the event. Presuming that flaws did exist and hence the containment would fail earlier in time, one can look at the difference in consequences between an early containment failure and a late containment failure. The Staff calculates about a factor of three in persons-rems between early and late failures. He indicated that the GESSAR-II containment will have venting procedures. The containment is vented in a

period of 11 to 28 hours which allows ample time for the operators to take action.

M. Reisch, Brookhaven National Laboratory (BNL), indicated that BNL was asked by the NRC to verify structural failure analysis results presented in Appendix G of the GESSAR report, and to perform independent analyses for these structural members. The BNL deterministic studies concentrated on the torispherical steel containment, the drywell steel head, and the concrete roof slab. A reliability evaluation was then done on the torispherical containment (see Appendix XI). D. Okrent asked if extensive plastic deformation occurs, does one know the behavior of flaws that were subcritical before the plastic deformation? Are they changed in size along with the plastic deformation, or do they hold their original size? R. Klecker indicated that as one approaches plasticity, in generally tough materials the small cracks will tend to blunt first before they begin to run. In the Staff's analysis one takes the cracks just to the point where they blunt but do not extend by crack extension or unstable tearing. Therefore, if one has a small crack it would tend to open up but the sharpness of the crack itself would tend to decrease. C. P. Siess asked if the large plastic deformations are confined to the knuckle region of the torispherical shell. M. Reisch indicated affirmatively. C. P. Siess asked if BNL has looked at any way in which the plastic deformations might affect the integrity of large penetrations such as the equipment hatch. M. Reisch indicated that BNL did not check the capacity of large openings in the containment building, such as the equipment or personnel hatches, since it was assumed that they were reinforced and should have a higher capacity. J. Rosenthal indicated that the Staff thinks that the hatches are not structurally stronger than the containment.

J. Rosenthal stated that the Staff is concerned about thermal effects on seals and whether the times to containment failure are very long. If the wetwell fails, a relatively benign release can be expected. There is a lot of margin between the wetwell failure and the drywell and the head of the drywell should a crack exist. If there is no flaw in the containment, then the containment may never fail. If there is a flaw, the containment may fail early. If it fails early relieving the pressure, you don't fail the drywell or the pool. The Staff is rather confident that one will maintain the drywell and pool integrity. W. Kerr asked if, from these analyses, it is possible to make a statement that, if containment failure occurs, it will occur as predicted by this analysis or is it likely to occur at a penetration or some other location. J. Rosenthal indicated that the Staff was far more concerned about deflagrations causing failures of penetration seals due to the thermal environment rather than containment failure occurring at penetrations due to pressures or forces on the penetrations caused by distortions in the structure.

R. Vij explained that the GESSAR-II drywell is a pressure boundary for larger pressures only. This volume is in direct communication through 120 vents below the water surface contained within the

drywell wall. When one has a very sharp transient the only resistance you have to this free communication is about 5 psi or so. The drywell boundary is not challenged at all in a low pressure phenomenon. During any slow pressure build up within the drywell or within the containment, the drywell head, ceiling, walls, or containment penetrations are not challenged at all beyond 5 or 6 psi. R. Vij indicated that the drywell has a capability only for handling some duty for large pressurizations such as a steam line break or a hydrogen detonation and combustion. J. C. Ebersole asked the GE design pressure for the maximum pipe break in the drywell. R. Vij indicated that it was 30 psig.

R. Vij discussed the dominant containment failure modes. These included hydrogen detonation in the containment, hydrogen combustion in the containment, hydrogen slow burning, and steam and or noncombustible gas overpressurization (see Appendix X). With a hydrogen detonation in the containment, the loads are shock wave internal pressure on the containment and external pressure on the drywell. For a local detonation the containment failure is assumed to occur above the water line. Below the water line, there is 1.75 inch thick steel plate backed by 8 feet of concrete. For a global detonation the containment failure is also assumed above the water line and drywell failure is also postulated. Therefore, both barriers are postulated to fail. For hydrogen combustion one generates internal pressure on the containment and small external pressure on the drywell. Containment failure is assumed above the water line with no drywell failure since there are no loads above 5 or 6 psig differential. No containment failure is postulated for the case of slow hydrogen burning since pressure does not exceed the capability of the containment and the drywell again only sees 5 or 6 psig differential. For the case of steam and/or noncombustible gas overpressurization there is a slow pressure generation and a small internal pressure on the drywell because of the free communication between the drywell and the containment. GE assumes containment failure above the water line at the containment ultimate pressure capability (85 psig) and no drywell failure since there are no significant loads. D. Hankins, GE, indicated that these analyses assumed that there are no ignitors. GE is postulating that hydrogen in most cases will build up to a fairly high concentration before combustion and that is why high pressures are assumed in some cases.

D. W. Moeller asked if there are vacuum breakers between the wet-well and the drywell. He asked if there are data on the failure rates or projected failure rates available. D. Hankins indicated that GE has submitted a study to the Staff on supression pool bypass which included the failure of the mechanical and motor-operated valves. There is a very low probability for the failure of both of those because there are breakers between the wetwell and drywell. D. Okrent asked if there is any way water in the suppression pool can drain out, aside from a failure such as a hole in the containment. R. Vij indicated that the ECCS suction lines have one valve on the outboard and if one assumes a break between the valve and the containment boundary, which is equivalent to

assuming a hole in the containment, such circumstances would lead to draining or a loss of a substantial part of the pool water.

R. Vij indicated that GE considered some validation methods with a structural configuration of the structures with the loads that are imposed. The stresses were calculated assuming that the highest stress points failed first. No failures were assumed where loads were significantly less than the design loads. Based on the analysis, GE concluded that the drywell structure head and personnel lock are not challenged and the suppression pool bypass due to drywell boundary failure should not be a concern.

R. Vij indicated that GE performed a very simplified analysis which assumed that the molten core burns a 6 foot deep hole in the basemat. The temperature of the molten core is assumed to be 4,000 degrees Fahrenheit. A heat conduction analysis and a linear elastic stress analysis were performed with those dimensions and the corresponding displacements. The temperature in the drywell wall at the basemat was found to be on the order of 150 degrees Fahrenheit. The deflection of the drywell wall was approximately half an inch at that point. The calculated stresses in the concrete were about 3,500 psi where the maximum compressive strength of the concrete is about 4,000 psi. In the steel, GE calculated stresses close to 40 ksi as compared to a yield strength of 46 ksi. Therefore, GE does not expect any danger to the drywell walls overall stability because of these assumptions for the molten core.

J. Rosenthal discussed the severe accident threat to the GESSAR-II containment (see Appendix XII). He indicated that the Staff's main concern is ablation of the concrete from the molten corium. He described a core melt scenario which led to the potential for loss of integrity of the pedestal beneath the reactor vessel. In answer to an inquiry by D. Okrent, he confirmed that drywell and conceivably wetwell integrity would be lost because there would be the potential for ripping out major piping that penetrates the drywell and wetwell.

J. Rosenthal listed the dominant containment failure modes considered by the NRC Staff. He indicated that the Staff considered overpressure failure due to noncondensible gas generation as well as hydrogen deflagration and detonations. He indicated that the Staff looked at structures, seals, piping penetrations and their fragility. He noted that the Staff considered neither steam explosions nor direct heating since the latter was viewed primarily as a high pressure problem. The dominant sequences on the GESSAR plant are low pressure because of the perceived reliability of the ADS (automatic depressurization system). He indicated that the Staff looked at a range of containment failure times. The late failure is at 11 hours for a wetwell failure due to a potential fabrication flaw and the early containment failure is about 150 minutes. There is a difference of about three in consequences between early and late failure. He discussed the Staff's concerns over excessive drywell/wetwell

leakage, where certain sequences model total bypass of the suppression pool via vaporization release. He indicated that the ultimate consequences of these sequences hinge on whether one bypasses the suppression pool or fails the pool. This is the main Staff concern. D. Okrent thought that the Committee ought to focus on the ablation discussion since there is a significant probability of a loss of integrity late, but nevertheless a loss of integrity of containment. There would be release of whatever radionuclides are available at that point. This contrasts with the claims that some containments stay relatively tight for a large family of accidents. H. Etherington suggested that ablation might be a rather destructive process involving chunks of concrete breaking away rather than just washing away.

C. Thomas, Division of Licensing, indicated that on June 27 the Commission approved the severe accident policy statement with one minor modification. That modification does not affect the GESSAR-II review. On July 3 the Commission issued a Staff requirements memorandum asking the Staff to modify the severe accident policy statement as noted and to forward it to the Office of the Secretary for publication in the Federal Register. In accordance with the provisions of the severe accident policy statement the Staff is prepared to amend the GESSAR-II FDA to permit it to be referenced in new CP and OL applications. The amendment would allow GESSAR-II to be referenced but the Staff would not issue a new CP or OL for an application that referenced GESSAR until successful completion of the severe accident review. The Staff will further amend the GESSAR-II FDA when the review is completed to allow it to be referenced for CP and OL applications issued for a fixed period of time.

C. Thomas explained that the Staff is experiencing resource problems since this is the first standard design to undergo a severe accident review under the provisions of the severe accident policy statement. For this reason, the review has taken a lot more time and resources then envisioned at the outset. NRR, in particular, has been extremely strained regarding resources to support this review. They are having a particularly difficult time regarding technical assistance and travel dollars. The GESSAR-II review is beginning to adversely impact some other NRR reviews. For that reason the Staff urges that the subcommittee and the ACRS complete its review as soon as reasonably possible. The Staff would like to complete this review with an ACRS letter at the September ACRS meeting. D. Okrent noted that there are a few issues that are still not resolved and the Staff does have concerns in the area of security or sabotage protection. He indicated that the Committee would do its best to cooperate. C. Thomas also noted NRR's pending reorganization that will probably further exacerbate the resource problem.

VI. Quantitative Safety Goals (Open)

[Note: R. P. Savio was the Designated Federal Official for this portion of the meeting.]

V. Stello, DEDROGR, indicated that he was somewhat troubled by the Committee's discussion of the safety goal issue with the Commission earlier that morning (Thursday, July 11). He indicated that the Committee continues to remind the Staff that there are issues in which they are particularly interested and on which they wish to be kept informed of the Staff's deliberations. He stated that the Staff has not evolved to a point where it is prepared to declare what its final position will be on the quantitative safety goal issue. He wanted the Committee to be aware that there is a variety of views on a number of contentious issues. He thought the Committee was unduly severe in casting the Staff's preliminary deliberations in an unfavorable light before the Commission.

V. Stello indicated that there are two areas in which advice from the Committee would be helpful: The first is the issue of setting a performance criterion. He contrasted 10^{-4}, measured by the classical PRA calculation, with a proposed 10^{-5} per reactor-year value for a melt which would penetrate the primary system. The 10^{-4} number predicts the capability of the core to sustain severe core damage. This core damage may lead to a full scale core melt where the core physically melts through the vessel. The 10^{-5} goal refers to the case where the core has already left the reactor vessel and the fission products are inside the containment. He thought it a fundamental philosophical point of whether one must ask PRA technology to take the next step of discriminating between those two kinds of calculations. The latter approach may delay the use of safety goals in any meaningful way until the Staff has made further advances in PRA technology. He indicated that he preferred to accept PRA technology as it exists today and recognize that there is some degree of conservatism that must be applied. The second area where ACRS action would be useful would be in the area of cost-benefit analysis where there is a great deal of controversy.

V. Stello stressed that the safety goal is not a mechanism to replace the current regulatory process. It is an additional element in the regulatory process and it will not replace the defense-in-depth concept. He suggested that one ought to know the results of a cost-benefit analysis when evaluating a plant modification so that if one decides to go that way there ought to be a reason for doing it. On the other hand, if the results of a PRA show that traditional methods of regulation overlook something needed for safety, then the PRA ought to supplant those traditional methods. Similarly, if the PRA clearly indicates that one ought not to do something, then the Staff ought to give that observation substantial weight.

T. Speis, NRR, summarized some of the views provided to the EDO by H. Denton. He indicated that H. Denton is not comfortable with the proposed core melt frequency guideline number. Calculations show that use of 10^{-4} per reactor-year yields a 45 percent chance of occurrence of a serious accident in the next 20 years, and a 10 percent chance of two or more such accidents. He suggested that H. Denton believes the 10^{-4} per reactor-year number is too large and

that too much reliance has been placed on perceived knowledge of fission product behavior and containment performance. H. Denton believes that there ought to be added conservatism in the performance guidelines.

T. Speis took note of some ambiguities in the Steering Group's report regarding the extent of core damage assumed with the 10^{-4} value (see Appendix XIII). W. Kerr suggested that the implication of H. Denton's message is that nuclear power plants that comply with a criterion of 10^{-5} for release of larger amounts of fission products would be safer than existing plants. V. Stello suggested that nuclear power plants would be made safer by adopting a safety goal for core melt frequency of 10^{-5} per reactor-year. Not only would the likelihood of a melt through of the reactor vessel be lower but the likelihood of severe core damage would also be lower.

F. J. Remick suggested that when the original safety goal was developed the Commission's responsibility was recognized as protection of the health and safety of the public. Its primary goals were public risk guidelines. The historical prospective on the 10^{-4} core melt guideline was that it was to be a secondary goal. He noted that the Steering Group recommends that it be elevated to the status of a primary guideline. He asked whether the Staff had given thought to submitting the 10^{-5} per reactor year number to public comment. D. A. Ward indicated that he thought what was driving the Staff to lower the core melt guideline number was not a concern about causing more offsite cancers or accidental fatalities but was the relatively high frequency of core melt which might occur under the 10^{-4} guideline (estimated by the NRC Staff to be a 45 percent chance of a core melt accident in the next 20 years).

D. Okrent indicated that, before NUREG-0880 was written, the ACRS proposed some trial criteria including a mean large scale core-melt guideline of 10^{-4} per reactor-year. This guideline clearly meant melt through of the reactor vessel. The ACRS proposal was an effort to achieve both defense-in-depth (asking for both a mean core melt frequency and a containment performance guideline given a large scale core melt). He suggested that H. Denton was expressing a lack of confidence in the current capability to predict containment performance as claimed in PRAs. V. Stello disagreed.

T. Speis indicated that NRR agrees with the Steering Group's recommendation that averted onsite losses be included (see Appendix XIII). T. Speis indicated that it is a matter of principle that all costs should be displayed.

F. Rowsome, NRC, explained that cost-benefit analysis already has a well established role in the NRC in the backfit policy in the regulatory analysis of new generic reactor safety standards and in NEPA. The safety goal area has been chosen as the arena in which the Agency will codify how it implements cost-benefit analysis (see Appendix XIV). V. Stello explained that the Staff proposes to impose plant modifications if ever it concludes that a plant is not

adequately safe. If it is a matter of compliance, the proposed rule before the Commission says that you do not do a cost-benefit analysis. W. Kerr suggested that there are exceptions to this rule such as Branch Technical Positions to which new reactors are being required to conform. V. Stello indicated that Staff's positions are embodied in the Standard Review Plan. The Staff's instructions to a utility are that if they wish to license a nuclear plant, they must show conformance to the Standard Review Plan. If they wish to deviate from the Standard Review Plan, they must tell the Staff how they expect to meet the basic underlying regulation.

F. Gillespie, Director, Division of Risk Analysis and Reactor Operations, RES, indicated that the Research position is fundamentally in support of the Steering Group's report. The Research staff does not wish that the defense-in-depth concept be lost. He thought that emphasis should be put on consideration of uncertainties in case studies done by the Staff. One does not want to lose the ability to make improvements just because a plant meets the 10^{-4} per reactor-year guideline. D. A. Ward asked if all that was meant was implementation of ALARA. F. Gillespie stated that if one can show that a fix has a significant safety benefit, even though the plant meets a particular core-melt frequency safety goal, the policy should have enough flexibility to allow the staff to show that such a significant increment in safety is in fact worth it. C. P. Siess asked if there were some de minimis level where one would not consider modification even though a significant safety benefit can be shown. F. Gillespie indicated that as long as the incremental benefit exceeded the cost it would not be considered de minimis. D. A. Ward indicated that he was not sure whether a case was being made for ALARA or unrestricted ratcheting. F. Gillespie explained that he was not prepared to set some particular de minimis cost level to cut off potential fixes.

VII. Meeting with the Commissioners (Open)

[Note: Commissioners present were N. J. Palladino, Chairman, J. K. Asselstine, F. M. Bernthal and L. W. Zech]

A. Consideration of Earthquakes in Emergency Planning

N. Palladino indicated that on December 18, 1984 the Commission issued a proposed rule which stated that earthquakes need not be considered in emergency planning.

D. W. Moeller indicated that the ACRS Subcommittees on Reactor Radiological Effects and Site Evaluation met with the NRC Staff, and representatives of FEMA, and called in a number of consultants to discuss the subject of emergency planning as related to natural phenomena with specific emphasis on earthquakes. As a result of these discussions and deliberations the Subcommittees reached certain conclusions. The Subcommittees saw no technical reason for the exclusion of earthquakes from the natural phenomena considered in offsite emergency planning for nuclear power plants. The degree to which natural phenomena should be considered has to be

plant-specific and site-specific because the frequencies and severities of these events vary over a wide range from one geographic area to another. The Subcommittees concluded that only limited consideration of earthquakes is appropriate. The major effort should be to become aware of problems and alternative approaches to their resolution. He stressed that it would be important to evaluate what might disturb normal emergency response in case an earthquake occurred at some time close to the time of an accident. The goal should be to assure that emergency plans as developed contain sufficient flexibility to cope with the potential added impact of such events. Commissioner Bernthal asked what was meant by the earthquake occurring at some time close to an accident. D. W. Moeller indicated that the ACRS considered two possible scenarios. Either the earthquake caused the accident (initiator) or the accident occurred and an earthquake happened to occur sometime close to it in time. The probability of the latter, however, is acknowledged to be extremely low.

D. W. Moeller explained that the potential impacts of earthquakes have for many years been given detailed consideration by the NRC regulatory process in the design, construction and operation of nuclear plants. Although FEMA does not consider the potential impact of earthquakes on nuclear power plant emergency planning on a formal basis, they have for some time considered the impacts on an informal basis. An item discussed by the Subcommittee and the full Committee was a proposal that it might be possible to rule out consideration of the impact of certain natural events, such as earthquakes, on the basis of their very low probability of occurrence. The Committee did not reach a conclusion, however, since the wide range of uncertainties in such probabilities compromises this approach. PRAs for several nuclear power plants indicate that earthquakes despite their low probability may be significant contributors (initiator) to the risk of core melt accidents. He also noted that FEMA is coordinating a National Earthquakes Hazards Reduction Program.

Commissioner Asselstine asked if the ACRS concern is only with the very large earthquake, larger than the safe shutdown earthquake (SSE), or does it extend to lesser earthquakes. D. Okrent indicated that studies that he had seen to date suggested that at the SSE level one would not expect serious trouble at a plant. It is only for levels somewhat above the SSE that one starts to question the possibility of consideration.

Commissioner Asselstine suggested that one should look at the complicating effects of earthquakes on the procedural aspects of emergency planning. Chairman Palladino pointed out the site specificity of the issue and that engineering judgment would have to be applied to decide whether to protect against any complicating event or situation regardless of how unlikely the event might be. Chairman Palladino thought that the frequency of the postulated natural phenomenon under question would be important.

Commissioner Asselstine suggested that one may be most concerned about a severe core damage accident with a significant release that would entail evacuation or a situation in the plant that has that potential. That scenario would require preliminary steps in preparation for such an eventuality. He suggested a sliding scale for initiators and probability. Consideration of a relatively frequent initiator would require study in great detail and minute detailed planning. The relatively frequent event with virtually no possibility of causing a nuclear accident would require consideration because of the potential simultaneous occurrence of an independent accident and the relatively frequent natural event. A relatively infrequent event with a high risk of causing a very serious accident should also be considered to a certain extent. The low probability event with very little likelihood of causing a severe accident would be discounted. He suggested that earthquakes fall in the third category since they are fairly low probability events but have the potential for being an accident initiator.

Commissioner Asselstine mentioned a July 5, 1985 memorandum from the EDO to the Commissioners and certain numbers in that memorandum (see Appendix XV). Commissioner Bernthal requested that the ACRS comment on the relative importance of the full range of natural phenomena (earthquakes, tornadoes, floods, etc.) in terms of their potential impacts on emergency planning. Such an evaluation should be based so far a practicable on a probabilistic approach.

Commissioner Asselstine suggested that the assumption that underlies the proposed rule that is out for comment is that the NRC knows enough both about the absolute probabilities of earthquakes and tornadoes and the relative probabilities of those two natural phenomena as compared with other phenomena to say that these two should be excluded from any consideration as to their impact on emergency planning. C. P. Siess suggested that the Commission is overestimating the effectiveness of its policies and regulations on the actual effectiveness of offsite emergency preparedness. He suggested that, regardless of what is in the emergency plan, the local officials will make a judgment as to what they consider natural hazards and prepare appropriately. Local officials are going to prepare most appropriately for what they consider are their natural hazards and not so much what is written into the regulations.

Commissioner Zech noted that the Commission is dealing with uncertainties, probabilities and statistics. He thought it most important for the Commission to develop the most reasonably prudent and common sense rule possible. To this end, he requested that the Committee develop a strong ACRS consensus view on this rulemaking.

D. Okrent indicated that hurricanes and earthquakes are the two natural phenomena having the most potential for disrupting large areas from the evacuation point of view and having the potential for being an initiator of a nuclear accident, rather than tornadoes.

B. Safety Goal Implementation

D. Okrent indicated that the Committee came very close to a Committee position on the issue of safety goals at the last ACRS meeting. The ACRS has not yet, however, seen a copy of a recommendation from the EDO to the Commission on this subject. The Committee has had the benefit of memoranda written by senior members of the NRC Staff such as H. Denton and R. B. Minogue and has had the benefit of the Safety Goal Steering Group report. At its Subcommittee meeting on Wednesday, July 10, the ACRS also heard the views of Lester Lave, well known economist who consults for both the NRC and the ACRS, and Philosophy Professor Douglas McLean from the University of Maryland. Both were participants in the first and second panels that reviewed NUREG-0880 in its early formulative years.

D. Okrent speculated that the ACRS will state that the NRC is not now ready to reaffirm and implement the 1983 Safety Goal Policy in its original or some slightly modified form. Progress had been made and additional effort is needed, but the form of the goals and the plan for implementation are not yet well enough developed. Note was taken of the considerable differences with the original proposal that exist among members of senior Staff. He anticipated that the Committee will emphasize in its report that greater attention be devoted to working toward an adequate core melt objective and toward the identification and use of a containment performance objective. He suggested that the Committee is concerned that the Safety Goal Policy Statement does not give sufficient emphasis to defense-in-depth. There is also concern that inappropriate reliance may be placed on benefit-cost analysis. Chairman Palladino asked why the ACRS did not choose to wait for the Staff document on the safety goal. D. A. Ward suggested that it might be useful for the ACRS to try to influence the Staff document through a letter this month. He noted that that opinion is not unanimous among the Committee members.

F. J. Remick welcomed the opportunity to present the Committee's views but thought that the Staff ought to have the opportunity to evaluate and provide their judgment regarding the just completed two year evaluation program. He thought it appropriate for the Committee to comment on the Steering Group report but his own personal preference would be against a Committee letter at this time.

D. Okrent indicated that ACRS comments will center on the Safety Goal Steering Group report. He indicated that the Committee agrees with many of its findings and conclusions. The Committee agrees that PRAs have limitations that must be understood when the results are used and the results of a PRA should normally be used in conjunction with traditional safety review methods in making regulatory decisions. The Committee agrees tentatively that the statement of the qualitative goals in the 1983 Safety Goal Policy Statement appears to be satisfactory. The Committee agrees that,

for sites where no individuals reside within a mile from the plant, an individual should be assumed to reside one mile from the site boundary (for purposes of calculation). In applying the latent cancer fatality safety goal, the Committee agrees that it is better to consider the population within ten miles rather than 50 miles as proposed in the 1983 policy statement. He noted that while the Steering Group proposed that averted onsite costs in a core melt accident be included in benefit-cost analysis, several senior Staff members have expressed concern that not to include such costs would lead to jeopardizing defense-in-depth. He recognized that inclusion of averted onsite costs and benefits is controversial and he speculated on the possible ACRS position: Benefit-cost calculations ought not necessarily be the most important criteria in decision making concerning safety and the accomplishment of defense-in-depth. He noted that L. Lave and D. McLean stated very positively that the only way to do cost-benefit analysis is to include all significant costs and all significant benefits and the 1983 policy statement was deficient in that regard. Commissioner Bernthal expressed an interest in seeing L. Lave and D. McLean's written comments. D. Okrent agreed to forward them.

Commissioner Bernthal suggested that the Committee note how the current approach in the Safety Goal Policy squares with the NRC's mandate under the Atomic Energy Act. Chairman Palladino suggested that the Committee take account of the nuclear industry's point of view. D. Okrent acknowledged that there are many differences of opinion about the safety goal policy. He expected that the Committee will recommend that, when the Commission develops a new statement of the safety goal policy, it should state that near compliance with a mean core melt frequency of 10^{-4} per reactor-year is an NRC objective for all but a few small existing nuclear power plants and that prudence will tend to take priority over benefit-cost analysis in working toward this goal.

D. Okrent stressed that he expected the Committee to recommend development of a containment performance guideline, as well as the use of mean values instead of median when assessing core melt frequencies or doing cost-benefit calculations. He noted that the Steering Group proposed a number of detailed implementation procedures. Operating limits proposed are in some instances not sufficiently conservative and in some cases not clear. The Committee has questions regarding these proposed operating limits and wishes to discuss them in more detail with the Staff. The ACRS recommends that they not be adopted at this time.

D. Okrent indicated that the Committee will most likely give emphasis to the Commission's policy that future nuclear power plants be safer. He suggested that the ACRS may well recommend a target core melt frequency mean for future nuclear power plants of 10^{-5} per reactor year.

F. J. Remick stated that he personally did not think that the defense-in-depth concept would be compromised if used in conjunction with prudent normal judgment. He indicated that he did

not disagree with the two ACRS consultants who thought that all benefits and costs ought to be considered in a cost-benefit analysis. He questioned the Commission's authority or mandate to get into averted costs to the licensee. Commissioner Asselstine noted that the Atomic Energy Act also talks about minimizing the danger to life and property. Commissioner Bernthal said that this is clearly ALARA and ALARA has been manifestly rejected in most quarters by the U.S. Congress.

H. W. Lewis spoke of the difficulties the Staff is having regarding questions of uncertainty bound up with the issue of the interpretation of a safety goal. He noted that the Subcommittee heard a divergence of views. He suggested that the issue has not been sufficiently addressed as yet. He pointed out that the safety goal must be treated as part of a whole package which includes the backfitting rule and the Severe Accident Policy Statement. He took exception to the Staff's continued use of median versus mean values.

Commissioner Asselstine indicated that it was his sense that there are fundamental questions and disagreements within the Staff and on the Committee about some of the key issues involved in this safety goal. He wondered whether the disagreements are so significant that perhaps the Commission ought to get more directly involved now before the process works its course. D. Okrent hoped that the EDO would resolve issues in its recommendation to the Commission. Commissioner Bernthal suggested that it was his understanding that at least one school of thought in the Staff wants to move away from a definition based on core-melt to a definition based on loss of primary system integrity. C. P. Siess thought that it makes sense to consider the core out of the primary system in the containment where it can challenge the last barrier to the environment (containment). It puts some emphasis on containment performance criteria that one does not get if you do not consider various core-melt scenarios. Commissioner Bernthal suggested that one might get a cleaner analysis from a numerical point of view because of the uncertainty in core-melt phenomena. C. P. Siess noted that one would now be confronted with a calculation of where the primary coolant goes. He wondered whether this would simplify or complicate the analysis. W. Kerr suggested that it might be more difficult to calculate the loss of primary system integrity than to calculate the conditions under which core-melt is likely to occur.

Chairman Palladino indicated that this discussion has been very worthwhile. He looked forward to reports from the Committee.

VIII. EPA Standards for High Level Waste Repositories (Open)

[Note: O. S. Merrill was the Designated Federal Official for this portion of the meeting.]

D. W. Moeller indicated that, for high level radioactive waste repositories, EPA has the responsibility for developing standards (40 CFR 191) and the NRC for the development of overall performance

objectives, regulatory guides and regulations (10 CFR 60). DOE sets system guidelines (10 CFR 960). He noted that the NRC has folded in conservatisms, dose, and risk limits into its performance objectives. Both the DOE and the NRC efforts have occurred without finalized EPA Standards. DOE has gone ahead in the absence of the EPA Standards toward making plans for constructing a high level waste repository in accord with 10 CFR 60.

D. W. Moeller indicated that the NRC has been looking at the EPA standards and the discussions have been primarily directed to the resolution of jurisdictional disputes. The NRC Staff has already concurred with the draft EPA standards issued in 1983.

D. W. Moeller indicated that the Subcommittees on Waste Management and Site Evaluation met on June 18 and 19, 1985 to review Revision 6 of the EPA Standards. He noted that the Commissioners will shortly be providing comments and recommendations to the EPA on these standards. He suggested that the ACRS offer advice to the Commission as it did on the DOE Mission Plan.

D. Okrent urged that the Committee encourage the Commission to recommend that EPA officials use a risk-based approach in the development of these standards. The Subcommittee suggested that the proposed EPA Standards are unduly restrictive and may result in the rejection of some suitable sites and the expenditure of funds that might be better applied to other environmental problems. He noted that the EPA has refused to relax the Standards and the NRC has concurred. He stated that the Subcommittee review revealed that the Standards appear to be flawed. The Subcommittee believes that the Standards should be expressed in terms of dose equivalent and/or health effects limits as contrasted to radionuclide release limits, which have little relationship to health risks. The release limits given in the proposed Standards do not appear to be directly related to the proposed limitation on health effects. Since the generic environmental model was used to estimate the population doses resulting from the stated releases, the Subcommittee questions whether the resulting estimates will be applicable to specific sites selected for a repository. It also questions whether these estimates will be applicable to disposal methods other than a geologic repository.

D. Okrent indicated that the Subcommittee report lists the pitfalls of the EPA Standards and recommends that the Commission make these shortcomings known to the EPA. He suggested that it is most important to tell the Commission that its Staff should stipulate whether these Standards are practicable. E. Goldberg, OPE, noted that the Commission has no formal endorsement function or concurrence function regarding the EPA standards. D. W. Moeller pointed out that there are also problems with the confidence limits given in the proposed standards. He suggested that DOE will not be able to prove that they can meet the release limits in the standards with adequate confidence. D. Okrent expressed concern that the release limits in the EPA standards may not make it practicable to license a high level waste facility.

IX. ANL - West Survey of Control Room Habitability (Open)

[Note: J. O. Schiffgens was the Designated Federal Official for this portion of the meeting.]

D. W. Moeller indicated that the Subcommittee on Air Systems met on June 17, 1985 to discuss two NRC reports: NUREG/CR-4191 "Survey of Licensee Control Room Habitability Practices" (prepared by ANL-West, consultants to the NRC/NRR Staff), and NUREG/CR-3551, and "Safety Implications Associated with In-Plant Pressurized Gas Storage and Distribution Systems in Nuclear Power Plants" (prepared by ORNL or consultants to the NRC/AEOD Staff). With regard to NUREG/CR-3551, the consultants recommended that compressed gas cylinders ought not to be allowed in critical areas unless it can be assured that should they become missiles they would not damage critical plant equipment. They also recommended that flow controlling valves should be installed on hydrogen lines where high pressure cylinders are handled. All high pressure gas lines should be color-coded.

W. P. Gamel, NRR, explained that in June 1983 the Staff was asked to prepare a program plan to address control room habitability concerns of the ACRS. The Control Room Habitability Working Group Report, which was reviewed by the Subcommittee in November 1984, will be published as NUREG/CR-1129. The Staff's assessment of control room habitability practices, which parallels the ANL West report (NUREG/CR-4191), will be published as a supplement to NUREG/CR-1129. He explained that one recommendation of the Working Group is that efforts be increased to obtain industry feedback on control room air systems. This will be accomplished in part by increasing participation by the Staff in technical society meetings. He noted that the Staff has a new contract with ANL-West to investigate generic issues identified in the original control room habitability study and there are plans for surveying an additional 12 plants. He mentioned that progress has been slower than anticipated regarding revision of Regulatory Guide 1.52, the standard review plan, and technical specifications. G. A. Reed asked what the average cost for upgrading control room air systems would be for each nuclear plant? J. J. Hayes, NRC, indicated that the cost would be nominal because most of the upgrade would involve utility action to meet commitments that it already has in its SAR and technical specifications. W. P. Gamel added that he expected a cost of $500,000 to $1,000,000 per plant to replace isolation dampers which have been found to exhibit excessive leakage.

J. J. Hayes explained that three plants were selected for the initial assessment of control room systems, components, operations, procedures and technical specifications. Mechanical and electrical systems were reviewed as well as remote shutdown capability. NRC practices and policies and NRC licensee practices were also explored. It was concluded from the field studies that loss of ventilation and loss of air conditioning events which have occurred at operating nuclear plants should be studied further and their possible contribution to the degradation of plant safety evaluated

(see Appendix XVI). He mentioned a case of the loss of both air conditioning trains at the Calvert Cliffs Nuclear Plant where spurious signals from instrumentation and the failures of printed circuit boards were found. J. C. Ebersole suggested that a failure of printed circuit boards is a generic issue the ACRS should review by examining applicable procurement specifications. J. J. Hayes noted that LERs currently do not include instrument problems.

J. J. Hayes indicated another conclusion of the field studies was that changes to the action statements and surveillance requirements of technical specifications should be made as needed to insure that the control room HVAC system specifications provide for functioning as designed. He noted that a generic problem identified during this study was the leakage through isolation dampers or valves. As a result, leaky isolation dampers in some plants have been changed to bubble-type dampers. A problem was found during the field studies regarding appropriate laboratory conditions for the testing of charcoal filters in HVAC systems. J. C. Ebersole suggested that it appears that the control room habitability issue should be expanded to include environmental control since control room temperature limitations affect equipment as well as the operators. J. J. Hayes noted that one problem is the fact that updated FSARs for operating nuclear plants often do not reflect actual systems. This obsolesence occurs because some changes to HVAC systems can take as much as three to five years to be documented. He stated that another general conclusion of the field application studies was that the whole approach to control room HVAC systems lacks a systems approach, but is handled more on a components-approach basis.

X. Long Range Plan for NRC (Open)

[Note: J. C.McKinley was the Designated Federal Official for this portion of the meeting.]

M. W. Carbon indicated that the Subcommittee on the Long Range Plan for the NRC met on July 10 and 11, 1985 to interview invited individuals from the nuclear industry. These individuals were asked to comment on whether the NRC should have a long range plan and whether the ACRS can be of assistance in the development of such a plan. There was general agreement that the NRC should have some long range plan. R. Mattson, former NRC Staff member, and J. Ahearne, former NRC Commissioner, both suggested that the Commissioners themselves ought to get more deeply involved in the development of such a plan. Interviewees were in general agreement with the selection of prospective issues to be handled in the long range plan and thought that a plan ought to reflect the expected state of affairs for at least ten years into the future or longer.

M. W. Carbon reminded the Committee that the Commission assigned the Office of Policy Evaluation the task of developing a 5 year plan. The outline for this plan, which is based upon the current Program Planning Guidance document for the NRC, was briefly

described. The Subcommittee plans to review the outline at a future meeting.

M. W. Carbon explained that the Subcommittee concluded that there was need for an additional half-day Subcommittee meeting and an additional half-day of full Committee discussion regarding the ACRS strategy for the long range plan. He suggested that there be full Committee review of the OPE outline at the August meeting. Several Committee members expressed the opinion that it would be very difficult to provide OPE thoughtful comments on that short a schedule.

XI. Human Factors and Maintenance Subcommittees on Natural Aptitude Selection Procedures (Open)

[Note: J. O. Schiffgens was the Designated Federal Official for this portion of the meeting.]

G. A. Reed indicated that the ACRS Subcommittees on Human Factors and Maintenance Practices and Procedures met on June 18, 1985 to explore the use of natural aptitude selection procedures, tests, and evaluations. He indicated that D. Kleinke, Manager of Psychological Services at the Edison Electric Institute (EEI) discussed EEI's seven selection testing projects which include the Plant Operators Selection System (POSS), which is related to Wisconsin Electric Power tests, and the Maintenance Aptitude Selection Test (MAST).

G. A. Reed indicated that he was disappointed in Kleinke's pronouncements in that he did not make an enthusiastic case for natural aptitude selection. He was surprised that Kleinke flatly thought that mechanical aptitude could not be learned but he agreed with him. Physical abilities tests, as part of MAST, showed that one could not train someone to have mechanical ability. G. A. Reed also indicated that the Committee heard from A. Mascitti, Supervisor of Supervisory and Professional Placement, Wisconsin Electric Power. Both Kleinke and Mascitti thought that selection testing is important. No statement was made, however, as to whether it would be a good idea for NRC to draft a rule requiring natural selection testing at nuclear power plants. He indicated that he was still in favor of the promulgation of an NRC rule on natural selection. D. A. Ward indicated that he was unable to get a clear idea of how many nuclear plants use the EEI tests and how many utilities use some other kinds of tests. W. Kerr thought that the NRC is already getting too involved in nuclear plant training and he was not sure that a NRC rule requiring natural selection testing would be such a good idea.

XII. Briefing Regarding Steam Line Failure in Non-Nuclear Power Plants (Closed)

[Note: E. G. Igne was the Designated Federal Official for this portion of the meeting.]

DELETION 1

XIII. Executive Sessions (Open)

[Note: R. F. Fraley was the Designated Federal Official for this portion of the meeting.]

A. Subcommittee Assignments

1. Meeting with the NRC Commissioners

The ACRS discussed with the Commissioners the consideration of seismic events in emergency planning. Commissioner Bernthal requested that the ACRS comment on the relative importance of the full range of natural phenomena (earthquakes, tornados, floods, etc.) in terms of their potential impacts on emergency planning. Such an evaluation should be based, so far as practicable, on a probabilistic approach. Commissioner Zech requested that the Committee develop a strong ACRS consensus view on this rulemaking. Commissioner Asselstine mentioned a July 5, 1985 memorandum from the EDO to the Commissioners and certain numbers in that memorandum. He solicited ACRS comment on the numerical values proposed. These matters have been assigned to the Subcommittee on Site Evaluation, Dade Moeller, Chairman, for follow-up.

During the discussion of quantitative safety goals, the Committee committed to send to the Commission ACRS consultants' comments on safety goal policy by L. Lave and D. MacLean.

2. SALP Evaluation of Licensees

A letter from W. J. Dircks, EDO, to C. Dean, Chairman, TVA Board of Directors, dated July 3, 1985 expresses concern regarding TVA management deficiencies based on various indicators which compare TVA plants with other non-TVA nuclear units. The Committee decided to request that the Subcommittee on Human Factors consider this type of evaluation as a tool to evaluate project management at operating nuclear plants.

3. Incident Investigation Program

The Committee noted two alternate proposals for NTSB type investigations now before the Commission:

- NRC Staff Plan for improving the existing program for the investigation of significant operational events
- Use of Atomic Safety and Licensing Boards to investigate significant nuclear events

This issue was assigned to the Subcommittee on Regulatory Policy and Practices, H. W. Lewis, Chairman, for follow-up, development of an ACRS position and the preparation of draft comments for consideration for forwarding to the Commissioners.

B. ACRS Reports, Letters, and Memoranda

1. Reports, Letters, and Memoranda

ACRS Comments on Proposed NRC Safety Goal Evaluation Report

The Committee prepared a report to the Commissioners on its review of the NRC draft Safety Goal Evaluation Report dated April 1985. The ACRS expects to make further comments on the Steering Group report when the EDO has formulated recommendations to the Commission. The Committee has a range of questions on proposed operating limits and wishes to discuss these matters in detail with the NRC Staff. Additional comments by F. J. Remick, G. A. Reed, M. W. Carbon, and H. W. Lewis were appended.

2. Provisions for Protection Against Sabotage

The Committee prepared a report to the Commissioners regarding present provisions for protection against sabotage. Additional comments by G. A. Reed, D. A. Ward, H. W. Lewis, P. G. Shewmon and F. J. Remick were appended.

3. Long Term Seismic Program Plan For The Diablo Canyon Power Plant

The Committee prepared a report to the Commissioners of its review of the Long Term Seismic Program submitted by the Pacific Gas & Electric Company (Licensee) for the Diablo Canyon Power Plant. Additional comments by D. Okrent, W. Kerr, and D. A. Ward were appended.

4. EPA Standards for High-Level Radioactive Waste Disposal

The Committee prepared a report to the Commissioners which discusses the proposed "Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes" (10 CFR 191), being developed by the U.S. Environmental Protection Agency (EPA).

5. Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees

The Committee prepared a report to the Commissioners of its review of the proposed rule on "Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees" (10 CFR 30, 40, and 70).

6. Investigation of Recent Incident at the Davis Besse Nuclear Power Plant

   The Committee prepared a report to the Commissioners commending the NRC Staff for its initiative in using an incident investigation team (IIT) to investigate the recent loss of feedwater incident at the Davis Besse Nuclear Power Plant.

7. Control Room Habitability

   The Committee prepared a memorandum to the EDO regarding the report, "Survey of Licensee Control Room Habitability Practices" (NUREG/CR-4191). The ACRS wishes to be kept informed as future developments on this subject take place.

8. Materials Research

   The Committee prepared a memorandum to the EDO (Attention: Guy A. Arlotto) regarding progress made on two RES projects -- SAFT-UT at PNL and acoustic emission (AE) at Watts Bar with comments on their continued funding and use.

C. Future Schedule

1. Future Agenda

   The Committee agreed on tentative agenda items for the 304th ACRS meeting, August 8-10, 1985 (see Appendix II).

2. Future Subcommittee Activities

   A schedule of future Subcommittee activities was distributed to Members (see Appendix III).

D. Sustained Meritorious Service

G. A. Reed introduced a draft letter regarding the presentation of several American Nuclear Society (ANS) awards for Meritorious Performance in Reactor Operations. The presentations were made on June 11, 1985, in Boston as part of the ANS Annual Meeting. The Committee decided, and so informed G. A. Reed, that it would be more appropriate for him to forward the letter as a statement of his own position as a member of the public.

E. Joint Meeting of Nuclear Safety Committees

D. A. Ward suggested a joint meeting of the ACRS, the Groupe Permanent, the RSK, and a corresponding Japanese group in the U.S. in approximately one year. The Committee suggested that initial planning of the agenda begin now and that a search be undertaken to locate a source of supplemental outside funding for this project.

F. Proposed Amendments to ACRS By-Laws

The Committee approved three proposed amendments to the ACRS By-Laws. The first pertains to how amendments to the By-laws are proposed and approved. The second pertains to how a Member with a safety-related concern may solicit ACRS action and, if the member does not consider the response to be satisfactory, what additional options and Staff support may be available to him for follow-up. The third pertains to additional information on conflicts-of-interest.

G. Watts Bar Nuclear Plant

The Committee heard and discussed the report of the ACRS QA/QC Subcommittee, G. A. Reed, Acting Chairman, regarding QA/QC difficulties and proposed corrective actions regarding the IDVP of Watts Bar conducted by Black & Veatch Engineering. Several other questions regarding TVA management of its nuclear plants have recently been identified (see SECY-85-231, Proposed NRC Action with Regard to TVA, dated June 28, 1985) and TVA is in the process of reorganizing its nuclear management. The Committee decided to take no further action regarding this matter until the NRC Staff has completed their evaluation.

D. A. Ward indicated that the Committee intends to take a look at the Staff's supplement to the SER when it is in a final draft and decide if any further action is necessary. The decision whether to schedule another subcommittee meeting or just have full Committee discussion could be determined at a later date. J. C. Ebersole asked whether the Staff intends to respond to the many detailed questions submitted by Henry Meyers of the Udall Committee. The Committee expressed interest in seeing a copy of the Staff's response to the H. Meyers observations when it is available.

H. Indian Point Nuclear Plant

The members discussed a report proposed by Dr. Okrent regarding the Indian Point Special Proceeding based on the PRA for the Indian Point Nuclear Station. The Committee deferred any action on this matter until backup material can be provided and discussed by the Committee. This item has been scheduled for discussion during the August ACRS Meeting.

I. Licensing Process, Consideration of a National Academy of Nuclear Power Safety

The members discussed a suggestion by G. A. Reed that the Committee consider this subject and provide comments to the Commission regarding this proposal by Senator Moynihan since one member (Commissioner Zech) has already expressed support for a training academy of this nature. The members decided not to pursue this matter.

The 303rd ACRS Meeting was adjourned at 2:45 p.m., Saturday, July 13, 1985.

APPENDIXES
TO
MINUTES OF THE 303rd ACRS MEETING
JULY 11-13, 1985

ACRS- 2336

ATTENDEES
303rd ACRS MEETING
July 11-13, 1985

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

David A. Ward, Chairman
Harold W. Lewis, Vice-Chairman
Robert C. Axtmann
Max W. Carbon
Jesse C. Ebersole
William Kerr
Carson Mark
Carlyle Michelson
Dade W. Moeller
David Okrent
Glenn A. Reed
Forrest J. Remick
Paul G. Shewmon
Chester P. Siess
Charles J. Wylie

ACRS Staff

Raymond F. Fraley, Executive Director
M. Norman Schwartz, Technical Secretary
Herman Alderman
Paul A. Boehnert
Anthony J. Cappucci
Robert Cushman
Monideep De
Sam Duraiswamy
Medhat M. El-Zeftawy
John Flack
John T. Gilbert
James A. Jeffries
Janet Kotra
Morton W. Libarkin
Richard K. Major
John A. MacEvoy
Thomas G. McCreless
John C. McKinley
Owen S. Merrill
Austin Newsom
Sidney J.S. Parry
Gary R. Quittschreiber
Richard Savio
Stanley Schofer

NRC ATTENDEES

303RD ACRS MTG.

Thursday, July 11, 1985

OFFICE OF NUCLEAR REACTOR REGULATION

R. A. Gilbert, DL
G. Sege, DST
R. Hernan, PPAS
W. S. Hazelton, DF
S. Jsrau, DST
J. F. Stolz, DL
G. Rivenbank, DL
A. DeAgazio

OFFICE OF INSPECTION & ENFORCEMENT

R. Baer
E. Weiss
D. A. Powell

ROGR/OFFICE OF EXECUTIVE DIRECTOR FOR OPERATIONS

T. Cox

INVITED ATTENDEES

303RD ACRS MEETING

Thursday, July 11, 1985

BECHTEL POWER CORPORATION

E. Oesterle

PACIFIC GAS & ELECTRIC

R. F. Locke
L. S. Cluff
D. A. Brand
J. B. Hoch
B. Norton
R. Fray
R. F. Locke
G. G. Sarkisian
D. W. Ogden
B. S. Lew
B. Norton

PUBLIC ATTENDEES

303RD ACRS MEETING

Thursday, July 11, 1985

H. G. Hawkins, S. California Edison Comp ny
R. Borsum, Babcock & Wilcox
R. Fedini, Duquesne Light Company
P. Docherty, Westinghouse
M. Beaumont, Westinghouse
P. Higg, Atomic Industrial Forum
F. Stetson, NUS Corporation
W. Shark, Argonne National Laboratory
C. Czajkowski, Brookhaven National Laboratory
H. Specter, NY power Authority
A. J. Pressesky, American Nuclear Society
L. M. Muntzing, Dames & Moore
J. Nurmi, EPM
D. A. Brand
L. Connor, DSA
J. Berga, Electric Power Research Inst.
G. Sauter, Electric Power Research Inst.
E. Lindemann, McGraw-Hill/Inside NRC
D. Holland, General Public Utilities
K. Barnes, GPUN

NRC ATTENDEES

303RD ACRS MTG.

Friday, July 12, 1985

OFFICE OF NUCLEAR REACTOR REGULATION

E. Adensam, DL
T. J. KeKergin, DL
R. Hernan, PPAS
N. Chokshi, DE
C. Tinkler, DSI
J. Lane, DSI
C. Thomas, DL
D. Schletti, SSPB
J. Rosenthal, DSI
V. S. Parezewski, DE
M. Rubin, RRAB
J. P. Knight, DE
D. R. Muller, DSI
W. P. Gammill, DSI
J. J. Hayes, DSI
K. Dempsey, DSI
D. Persinko, LQB

ROGR/OFFICE OF EXECUTIVE DIRECTOR FOR OPERATIONS

M. Taylor

APPLICANT ATTENDEES

303RD ACRS MEETING

Friday, July 12, 1985

TENNESSEE VALLEY AUTHORITY

H. L. Jones
R. M. Pierce
J. J. Ritts
R. H. Shell

BLACK & VEATCH

W. J. Zidziunas

BROOKHAVEN NATIONAL LAB

M. Riech
S. Shema
R. Taung
J. Pires

GENERAL ELECTRIC

D. Foreman
R. Vij
R. Villa

GENERAL ELECTRIC

D. Hawkins
R. Ketchel
G. Sherwood

PUBLIC ATTENDEES

303RD ACRS MEETING

Friday, July 12, 1985

E. Lindeman, Inside NRC
R. Fedin, DuQuesne Light Company
R. Borsum, Babcock & Wilcox
R. Hubbard, MHB Associates
A. J. Pressesky, American Nuclear Society
J. Berga, Electric Power Research Inst.
J. Niman, Electric Power Research Inst.
L. Peeters, SAIC
M. Wagner, McGraw Hill

APPENDIX A
FUTURE AGENDA

AUGUST ACRS MEETING

• San Onofre Nuclear Generating Station Unit 1 -- SEP review, ACRS comments	3 hrs
• GESSAR II -- Continue ACRS review	3 hrs
• USI-A-46, Seismic Qualification of Equipment in Operating Plants -- ACRS comments	3 hrs
• Maintenance and Surveillance Program Plan -- ACRS comments	1 hr
• ACRS Subcommittee report on report of Pipe Crack study group/proposed changes in the General Design Criteria, regulations, etc. regarding piping design and the DBA	Deferred
• Assignment and makeup of ACRS Subcommittee and consultants on waste management -- Discussion among members	Deferred
• Vogtle Nuclear Plant Units 1 and 2 -- OL review, ACRS comments	4 hrs
• ECCS -- ACRS discussion regarding proposed revision of Appendix K	2 hrs - Tentative
• Report of Subcommittee on Reactor Radiological Effects regarding INPO Radiation Protection Program	Deferred
• Probabilistic Risk Assessment -- Proposed ACRS comments regarding significance/implementation of PRA results for the Indian Point Nuclear Plant and other nuclear facilities	2 hrs
• Proposed NRC Accident Source Term -- ACRS comments	Deferred
• NRC Long Range Plan -- ACRS comments on proposed plan outline	1 hr
• Regulatory Guide 1.99, Rev. 2, Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials -- ACRS comment regarding publication for public comment	Deferred to September
• Report of ACRS Subcommittee on CESSAR regarding provisions for decay heat removal in CE system 80 type plants	Deferred

•	Meeting with Director of International Programs -- Briefing regarding activities	Deferred
•	Report of Panel on ACRS Effectiveness	2 hrs
•	Response to Commissioner Asselstine's inquiry regarding adequacies/inadequacies in the NRC operator requalification procedures -- Impact on regionalization, etc.	Deferred
•	IE Inspection Program -- Briefing by IE representatives	Deferred

REVISED JUL 1 3 1985

SCHEDULE OF ACRS SUBCOMMITTEE MEETINGS

JULY

17 ATWS (BOEHNERT) - Kerr, Ebersole, Ward, Wylie. Purpose: To discuss RPS and scram breaker reliability.

18 & 19 VOGTLE, UNITS 1 & 2 (AUGUSTA, GA) (SCHIFFGENS/MCKINLEY) - Ebersole, Okrent, Reed, Wylie. Purpose: To begin review of OL application for Vogtle, Units 1 & 2.

30 JOINT WASTE MANAGEMENT AND PROCEDURES & ADMINISTRATION (MERRILL/FRALEY) - Moeller, Axtmann, Ebersole, Kerr, Mark, Remick, Shewmon, Siess, Ward. Purpose: To review the ACRS Role in the Civilian High-Level Radioactive Waste Management Program.

31 REACTOR RADIOLOGICAL EFFECTS (MERRILL) - Moeller, Axtmann, Okrent. Purpose: To review INPO's Radiation Protection Program and a recent INPO report on Excessive Personnel Radiation Exposures.

31 ECCS (BOEHNERT) - Ward, Ebersole, Etherington, Reed. Purpose: (1) To continue the review of the proposed revision to Appendix K of 10 CFR 50.46; (2) To review implementation of GE Appendix K analysis effort; (3) RCP trip issue resolution; and (4) To discuss NRR's ECCS-related issues of ongoing concern.

AUGUST

1 CLASS 9 ACCIDENTS (SAVIO/HOUSTON) - Kerr, Axtmann, Moeller, Okrent, Shewmon, Siess, Ward. Purpose: The Subcommittee will discuss with the NRC Staff and will continue the review of draft NUREG-0956, "Source Term Reassessment" and discuss a SECY paper describing regulatory initiatives related to the source term reassessment.

2 CLASS 9 ACCIDENTS (SAVIO/HOUSTON) - Kerr, Axtmann, Moeller, Okrent, Shewmon, Siess, Ward. Purpose: To discuss with the NRC and IDCOR the status of programs related to extending the results of the reference plants and how this relates to the ACRS recommended search for outliers program.

6 QUALIFICATION PROGRAM FOR SAFETY-RELATED EQUIPMENT (CAPPUCCI) - Wylie, Ebersole, Michelson, Reed, Shewmon, Siess, Ward. Purpose: To discuss NRC Staff resolution of USI A-46, "Seismic Qualification of Equipment in Operating Plants."

SCHEDULE OF ACRS SUBCOMMITTEE MEETINGS

AUGUST (CONT'D)

7	LONG RANGE PLAN FOR NRC (MCKINLEY) Carbon, Lewis, Moeller, Remick, Siess, Wylie. Purpose: The Subcommittee will continue discussions on developing comments on a long range plan for the NRC. Topics to be discussed are primarily technical issues related to the regulation of nuclear power plant safety and safety regulation over the next 5 to 10 years. The outline of a comprehensive long range plan being developed by the EDO and OPE will also be reviewed.
7	GESSAR II (MAJOR) - Okrent, Ebersole, Etherington, Mark, Michelson, Wylie. Purpose: The Subcommittee will continue its review of GESSAR II for a Final Design Approval applicable to future plants.
8 - 10	304TH ACRS MEETING
27	JOINT ECCS AND FLUID DYNAMICS (BOEHNERT) - Ward, Ebersole, Etherington, Reed. Purpose: (1) To review the status of the hydrodynamic loads issue for Mark I-III containment plants; (2) To review the AEOD report on Interfacing LOCAs; and (3) To review the USI A-43 Resolution Proposal.

SEPTEMBER

4 & 5	METAL COMPONENTS (IGNE) - Shewmon, Axtmann, Etherington, Michelson, Ward. Purpose: To review Reg. Guide 1.99, Rev. 2 and other related concerns, and to discuss the status of the NDT of piping program and the HSST program.
9 (1:00 P.M.)	REACTOR OPERATIONS (MAJOR) - Ebersole, Kerr, Michelson, Moeller, Okrent, Reed, Remick, Ward, Wylie. Purpose: To discuss recent operating experiences.
10	REGULATORY ACTIVITIES (DURAISWAMY) - Siess, Carbon, Kerr (tent.), Ward (part-time), Wylie. Purpose: To review: (1) Reg. Guide 1.23, Rev. 1, "Meteorological Measurement Programs for Nuclear Power Plants," (2) proposed Reg. Guide (Task No. IC 609-5), "Criteria for Power, Instrumentation, and Control Portions of Safety Sytems," and (3) Reg. Guide 1.105, Rev. 2, "Instrument Setpoints for Safety-Related Systems (tent.)."
10 (Closed)	WESTINGHOUSE WATER REACTORS (CAPPUCCI) - Ebersole, Etherington, Michelson, Okrent, Shewmon, Ward (part-time). Purpose: To begin the PDA review of the Westinghouse Advanced PWR (RESAR SP/90).

SCHEDULE OF ACRS SUBCOMMITTEE MEETINGS

SEPTEMBER (CONT'D)

11	LONG RANGE PLAN FOR NRC (MAJOR) - Carbon, Lewis, Moeller, Remick, Siess, Wylie. Purpose: The Subcommittee will continue discussions on developing comments on a long range plan for the NRC. Topics to be discussed are primarily technical issues related to the regulation of nuclear power plant safety and safety regulation over the next 5 to 10 years.
11	RIVER BEND (SAVIO) - Okrent, Ebersole, Shewmon. Purpose: To review Gulf States Utilities Company's application for an OL.
12 - 14	305TH ACRS MEETING
16 & 17 (Closed)	HUMAN FACTORS TOUR (RUSSELVILLE, AR) (SCHIFFGENS) - Ward, Lewis, Michelson, Moeller, Reed, Remick, Wylie. Purpose: This will be a tour and examination of ANO-1's emergency procedures (symptom based) and facilities.
23 & 24	JOINT STRUCTURAL ENGINEERING AND SEISMIC DESIGN OF PIPING (IGNE) - Siess, Ebersole, Etherington, Mark, Okrent, Shewmon. Purpose: To review the status of research programs on containment integrity, seismic margins, piping reliability, and other related matters.

OCTOBER

8	RELIABILITY ASSURANCE (VALVES) (MAJOR) - Michelson, Ebersole, Kerr, Okrent, Reed, Ward. Purpose: To continue discussions on valve reliability. A risk perspective on valve performance will be sought. Also to be studied is the importance of valves from a safety standpoint.
9	LONG RANGE PLAN FOR NRC (MAJOR) Carbon, Lewis, Moeller, Remick, Siess, Wylie. Purpose: The Subcommittee will continue discussions on developing comments on a long range plan for the NRC. Topics to be discussed are primarily technical issues related to the regulation of nuclear power plant safety and safety regulation over the next 5 to 10 years.
10 - 12	306TH ACRS MEETING

SCHEDULE OF ACRS SUBCOMMITTEE MEETINGS

DATES TO BE DETERMINED

(August/September)	ADVANCED REACTORS (EL-ZEFTAWY) - Carbon, Siess, Mark. Purpose: To discuss the proposed policy for regulation of advanced nuclear power plants.
(August/September)	HUMAN FACTORS (SCHIFFGENS) - Ward, Reed, Remick, Wylie. Purpose: To explore methods for deciding what actions should be automated in nuclear power plant operation.
(September) (tentative)	DECAY HEAT REMOVAL SYSTEMS (BOEHNERT) - Ward, Ebersole, Etherington, Reed. Purpose: To continue the review of NRR resolution position for USI A-45.
(September/October)	ECCS (PALO ALTO, CA) (BOEHNERT) - Ward, Ebersole, Etherington. Purpose: To continue the review of the joint NRC/B&WOG/EPRI/B&W joint IST Program. A visit is planned to the EPRI Stanford Research Institute facilities supporting this Program.
(October)	JOINT REACTOR RADIOLOGICAL EFFECTS AND FIRE PROTECTION (MERRILL/ALDERMAN) - Moeller, Axtmann, Carbon, Ebersole, Michelson, Okrent, Reed, Siess, Wylie. Purpose: To review the increased N-16 radioactivity and fire protection problems in using H_2 addition to BWRs to reduce IGSCC.
(October)	ATWS (BOEHNERT) - Kerr, Ebersole, Michelson, Ward. Purpose: To continue the review of the status of ATWS Rule implementation effort and related issues.
October (tent.)	QUALITY AND QUALITY ASSURANCE IN DESIGN AND CONSTRUCTION (MAJOR) - Remick, Michelson, Okrent, Reed, Siess, Ward, Wylie. Purpose: (1) To review the final Rule on "The Important To Safety Issue, and (2) To be briefed on the "NRC Quality Assurance Program Implementation Plant."
(Fall) (tent.)	RELIABILITY & PROBABILISTIC ASSESSMENT (location to be determined) (SAVIO) - Okrent, Kerr, Ebersole, Lewis, Mark, Michelson, Siess, Ward, Wylie. Purpose: To review the probabilistic risk assessment for Millstone 3.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
JULY 17, 1985	ATWS	(BOEHNERT) Kerr, Ebersole, Ward, Wylie Cons.: Davis, Lee, Lipinski

PURPOSE: To discuss RPS and scram breaker reliability.

LOCATION: WASHINGTON, DC

BACKGROUND:

What action is requested; by what date is it needed?

To investigate RPS and scram breaker reliability; as soon as reasonably possible.

What will be done at this meeting?

To begin investigation of RPS reliability (see attached memo).

What would be the consequence of postponing this meeting?

Delay in researching this topic. It is not an urgent priority issue (per ACRS discussion at May meeting).

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
JULY 18 & 19, 1985	VOGTLE, UNITS 1 & 2	(SCHIFFGENS/MCKINLEY) Ebersole, Okrent, Reed, Wylie Cons.:

PURPOSE: To begin review of OL application for Vogtle, Units 1 & 2.

LOCATION: AUGUSTA, GA

BACKGROUND:

What action is requested; by what date is it needed?

Subcommittee OL review in time for Committee consideration at the 304th, Aug. 9-10, 1985 ACRS meeting.

What will be done at this meeting?

Review the OL application and draft a letter.

What would be the consequence of postponing this meeting?

It could be postponed -- I think. The current schedule Has the fuel load date set for September 1986.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

To be provided later. The Staff anticipates publishing the SER on, or about, June 20, 1985.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
JULY 30, 1985	JOINT WASTE MANAGEMENT AND PROCEDURES & ADMINISTRATION	(MERRILL/FRALEY) Moeller, Axtmann, Ebersole, Kerr, Mark, Remick, Shewmon, Siess, Ward

PURPOSE: To review the ACRS Role in the Civilian High-Level Radioactive Waste Management Program.

LOCATION: WASHINGTON, DC

BACKGROUND:

What action is requested; by what date is it needed?

Chairman Palladino requested (of SECY) on June 25, 1985 written ACRS comments on the Staff's proposal before he votes on this issue; no date specified.

What will be done at this meeting?

(1) Review Staff's proposal to the Commission regarding High-Level Waste Management Oversight Advisory Committee (Ref. 1);

(2) Review OPE's memorandum to the Commissioners regarding same (Ref. 2); and

(3) Prepare written comments on Staff's proposal for full Committee consideration during its 304th meeting, August 8-10, 1985.

What would be the consequence of postponing this meeting?

It would result in postponing the submission of written ACRS comments to Chairman Palladino.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. SECY-85-197, Advisory Committee for Overseeing the High-Level Radioactive Waste Repository Program, Policy Issue (Notation Vote), dated May 31, 1985
2. Memo for Commissioners from J. Zerbe, OPE, regarding OPE's review of Reference 1 (preceding), dated June 24, 1985

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
JULY 31, 1985	REACTOR RADIOLOGICAL EFFECTS	(MERRILL) Moeller, Axtmann, Okrent Cons.: To be determined

PURPOSE: To review: (1) INPO's Radiation Protection Program, (2) NRC's 2-year program evaluating the effectiveness of INPO's Radiation Protection Program, and (3) INPO's recent SOER 85-3, "Excessive Personnel Radiation Exposures."

LOCATION: WASHINGTON, DC

BACKGROUND:

What action is requested; by what date is it needed?

Dan Muller, AD/RP, suggested ACRS review of item (1) above and requested review of item (2). Item (3) was added because of its related significance and recent issuance.

What will be done at this meeting?

Briefing and review of topics named above; preparation of Subcommittee comments and recommendations.

What would be the consequence of postponing this meeting?

No adverse effects; meeting is being held during a week when Subcommittee members will already be here for other, more urgent Subcommittee meetings -- hence, better utilization of resources.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. INPO Significant Operating Experience Report (SOER) 85-3, "Excessive Personnel Radiation Exposures," dated April 30, 1985 (INPO Confidential)
2. A Status Report will be prepared before the meeting.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
JULY 31, 1985	ECCS	(BOEHNERT) Ward, Ebersole, Etherington, Reed Cons.: Catton, Schrock, Sullivan, Theofanous, Tien

PURPOSE: (1) To review Appendix K revision effort.
(2) To review implementation of GE Appendix K analysis effort.
(3) RCP Trip issue resolution.
(4) Discuss NRR's ECCS-related issues of ongoing concern.

LOCATION: WASHINGTON, DC

BACKGROUND:

What action is requested; by what date is it needed?

(1) September ACRS review.
(2) September ACRS review.

What will be done at this meeting?

Review of 1 & 2 for September ACRS. Items 3 & 4 were left over from a previous meeting.

What would be the consequence of postponing this meeting?

Loss of timely review and subsequent schedule impact on the Commission's review of issues 1 & 2 above.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

To be provided in the near future.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
AUGUST 1, 1985	CLASS 9 ACCIDENTS	(SAVIO/HOUSTON) Kerr, Axtmann, Moeller, Okrent, Shewmon, Siess, Ward

PURPOSE: To discuss with the NRC Staff and to continue the review of draft NUREG-0956, "Source Term Reassessment" and to discuss a SECY paper describing regulatory initiatives related to the source term reassessment.

LOCATION: WASHINGTON, DC

BACKGROUND:

What action is requested; by what date is it needed?

The Subcommittee has requested the Staff to provide some indication of how the source term research will be used in the regulatory regime.

What will be done at this meeting?

The NRC will provide a status of the work related to possible regulatory initiatives based on the source term work.

What would be the consequence of postponing this meeting?

The NRC Staff has recommended this date as one which would provide adequate time for ACRS comments to be incorporated in the SECY paper.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. SECY paper on Regulatory Initiatives Related to the Source Term Reassessment (to be provided).

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
AUGUST 2, 1985	CLASS 9 ACCIDENTS	(SAVIO/HOUSTON) Kerr, Axtmann, Moeller, Okrent, Shewmon, Siess, Ward Cons.: Bender, Catton, Corradini, Davis, Lee,

PURPOSE: To discuss with NRC and IDCOR the status of programs related to extending the results of the reference plants and how this relates to the ACRS recommended search for outliers program.

LOCATION: WASHINGTON, DC

BACKGROUND:

What action is requested; by what date is it needed?

ACRS requested the Staff for a status of NRR's work regarding the regulatory use of the source term work.

What will be done at this meeting?

Status report.

What would be the consequence of postponing this meeting?

The Staff is preparing a SECY paper concurrently with Source Term Reassessment to define possible regulatory initiatives related to source term. Delaying this SECY paper would delay issuance of source term work.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. NUREG-0956) To be supplied
2. SECY paper on Source Term)

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
AUGUST 6, 1985	QUALIFICATION PROGRAM FOR SAFETY-RELATED EQUIPMENT	(CAPPUCCI) Wylie, Ebersole, Michelson, Reed, Shewmon, Siess, Ward

PURPOSE: To discuss the NRC Staff resolution of USI A-46, "Seismic Qualification of Equipment in Operating Plants."

LOCATION: WASHINGTON, DC

BACKGROUND:

What action is requested; by what date is it needed?

ACRS letter on the resolution of USI A-46. 303rd ACRS (July 1985).

What will be done at this meeting?

Review the proposed resolution of USI A-46. Prepare the report to the full Committee and suggested report to Commission.

What would be the consequence of postponing this meeting?

Delay of ACRS comments.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. CRGR Resolution Package (end of May 1985 - tentative).

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
AUGUST 7, 1985	LONG RANGE PLAN FOR NRC	(MCKINLEY) Carbon, Lewis, Moeller, Remick, Siess, Wylie

PURPOSE: The Subcommittee will continue discussions on developing comments on a long range plan for the NRC. Topics under discussion are primarily technical issues related to the regulation of nuclear power plant safety and safety regulation over the next 5 to 10 years. The outline of a comprehensive long range plan being developed by the EDO and OPE will also be reviewed.

LOCATION: WASHINGTON, DC

BACKGROUND:

What action is requested; by what date is it needed?

Currently a report addressed to the Commission or as input into a parallel OPE effort is expected. The current projection is to conclude this effort in October 1985.

What will be done at this meeting?

To be determined.

What would be the consequence of postponing this meeting?

Timeliness of effort would be effected. Would become out of phase with a parallel OPE effort on LRP.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

Dr. Carbon's latest review plan for this effort is available.

A Status Report will be prepared before the meeting.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
AUGUST 7, 1985	GESSAR II	(MAJOR) Okrent, Ebersole, Etherington, Mark, Michelson, Wylie Cons.:

PURPOSE: The Subcommittee will continue its review of GESSAR II for a Final Design Approval applicable to future plants.

LOCATION: WASHINGTON, DC

BACKGROUND:

What action is requested; by what date is it needed?

Final Design Approval of GESSAR II; as soon as possible following completion of the Staff's review.

What will be done at this meeting?

Review the resolution of items in SSER 4.

What would be the consequence of postponing this meeting?

Delay the issuance of a forward looking FDA for GESSAR II.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. SSER 4 (NUREG-0979) Safety Evaluation Report related to the final design approval of the GESSAR BWR/6 Nuclear Island Design (Draft SSER #4 received in late June and distributed).

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SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
AUGUST 27, 1985	JOINT ECCS AND FLUID DYNAMICS	(BOEHNERT) Ward, Ebersole, Etherington, Reed Cons.: Catton, Schrock, Sullivan, Theofanous, Tien

PURPOSE: (1) To review the status of the hydrodynamic loads issue for Mark I-III containment plants.
(2) To review the AEOD report on Interfacing LOCAs.
(3) To review the USI A-43 Resolution Proposal.

LOCATION: WASHINGTON, DC

BACKGROUND:

What action is requested; by what date is it needed?

No specific action date needed.

What will be done at this meeting?

(1) Review the status of hydrodynamic loads issue for Mark I-III containment plants.
(2) Review AEOD report on Interfacing LOCAs.
(3) Review implementation plan for USI A-43.

What would be the consequence of postponing this meeting?

Loss of timely review of Item 3 per NRR's schedule.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
SEPTEMBER 4 & 5, 1985	METAL COMPONENTS	(IGNE) Shewmon, Axtmann, Etherington, Michelson, Ward Cons.: Kassner, Odette

PURPOSE: To review Regulatory Guide 1.99, Rev. 2, and other related concerns, and to discuss status of NDT of piping program and HSST program.

LOCATION: WASHINGTON, DC

BACKGROUND:

What action is requested; by what date is it needed?

ACRS action is requested by the NRC Staff before Reg. Guide 1.99, Rev. 2, is promulgated.

What will be done at this meeting?

The Subcommittee will review with the NRC Staff Reg. Guide 1.99, Rev. 2, and develop recommendations for ACRS comments.

What would be the consequence of postponing this meeting?

None, except that ACRS comments, if any, will not impact Reg. Guide 1.99, Rev. 2.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. Reg. Guide 1.99, Rev. 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials, due for ACRS review in late June or early July 1985.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
SEPTEMBER 9, 1985 (1:00 P.M.)	REACTOR OPERATIONS	(MAJOR) Ebersole, Kerr, Michelson, Moeller, Okrent, Reed, Remick, Ward, Wylie

PURPOSE: The Subcommittee will discuss recent operating occurrences.

LOCATION: WASHINGTON, DC

BACKGROUND:

What action is requested; by what date is it needed?

Review recent operating experience, select incidents of significance for presentation to full ACRS during September meeting.

What will be done at this meeting?

Review recent operating experience.

What would be the consequence of postponing this meeting?

Would consider events at a later date.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. Status Report to be provided.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
SEPTEMBER 10, 1985	REGULATORY ACTIVITIES	(DURAISWAMY) Siess, Kerr (tent.), Carbon, Ward (part time), Wylie

PURPOSE: To review the following:

1. Regulatory Guide 1.23, Rev. 2., "Meteorological Measurement Programs for Nuclear Power Plants" (pre-comment).
2. Proposed Regulatory Guide (Task No. IC 609-5), "Criteria for Power, Instrumentation, and Control Portions of Safety Systems" (post comment).
3. Regulatory Guide 1.105, "Instrument Setpoints for Safety-Related Systems (tent.)."

LOCATION: WASHINGTON, DC

BACKGROUND:

What action is requested; by what date is it needed?

- ACRS concurrence in the Staff's posposal to issue item 1 for public comments.
- ACRC concurrence in the Regulatory positions of item 2.

What will be done at this meeting?

See purpose.

What would be the consequence of postponing this meeting?

Delay the issuance of these Guides.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

The above mentioned Regulatory Guides are expected to be made available to the ACRS during July 1985.

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SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
SEPTEMBER 10, 1985	WESTINGHOUSE WATER REACTORS (CLOSED)	(CAPPUCCI) Ebersole, Etherington, Michelson, Okrent, Shewmon, Ward (part-time)
		Cons.: Davis

PURPOSE: To begin PDA review of Westinghouse Advanced PWR (RESAR SP/90).

LOCATION: WASHINGTON, DC

BACKGROUND:

What action is requested; by what date is it needed?

ACRS letter on PDA approval by 11/86.

What will be done at this meeting?

Begin reviewing design modules.

What would be the consequence of postponing this meeting?

Delay in the completion of ACRS PDA review.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. RESAR SP/90 Standard Plant Design (SO-601).

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
SEPTEMBER 11, 1985	LONG RANGE PLAN FOR NRC	(MAJOR) Carbon, Lewis, Moeller, Remick, Siess, Wylie

PURPOSE: The Subcommittee will continue discussions on developing comments on a long range plan for the NRC. Topics under discussion are primarily technical issues related to the regulation of nuclear power plant safety and safety regulation over the next 5 to 10 years.

LOCATION: WASHINGTON, DC

BACKGROUND:

What action is requested; by what date is it needed?

Currently a report addressed to the Commission or as input into a parallel OPE effort is expected. The current projection is to conclude this effort in October 1985.

What will be done at this meeting?

To be determined.

What would be the consequence of postponing this meeting?

Timeliness of effort would be effected. Would become out of phase with a parallel OPE effort on LRP.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

Dr. Carbon's latest review plan for this effort is available.

A Status Report will be prepared before the meeting.

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SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
SEPTEMBER 11, 1985	RIVER BEND	(SAVIO) Okrent, Ebersole, Shewmon

PURPOSE: To review Gulf States Utilities Company's application for an OL.

LOCATION: WASHINGTON, DC

BACKGROUND:

What action is requested; by what date is it needed?

Issue an ACRS "full power" OL letter; at the September ACRS meeting.

What will be done at this meeting?

Complete the Subcommittee action on the River Bend OL review in support of an ACRS review at the September ACRS meeting.

What would be the consequence of postponing this meeting?

Possible delay of River Bend full power operation.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

NRC Staff SER Supplement to be supplied at the August ACRS meeting.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
SEPTEMBER 16 & 17, 1985	HUMAN FACTORS TOUR (CLOSED)	(SCHIFFGENS) Ward, Lewis, Michelson, Moeller, Reed, Remick, Wylie

PURPOSE: This will be a tour and examination of ANO-1's emergency procedures (symptom based) and facilities. The Subcommittee wants the opportunity to examine procedures at an operating plant and see how the TMI required backfits such as SPDS interface. Up to a day and a half is expected. ANO-1 is an 850 MWe, B&W PWR.

LOCATION: ANO-1, Russellville, AR (~50 miles outside of Little Rock, AR)

BACKGROUND:

What action is requested; by what date is it needed?

Review implementation of TMI required backfits. No schedular requirements.

What will be done at this meeting?

Tour and review facilities.

What would be the consequence of postponing this meeting?

None

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. One copy of ANO-1 Emergency Operating Procedures is available for your inspection at the ACRS Office (ask J. Schiffgens for it).

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
SEPTEMBER 23 & 24, 1985	JOINT STRUCTURAL ENGINEERING AND SEISMIC DESIGN OF PIPING	(IGNE) Siess, Ebersole, Etherington, Mark, Okrent, Shewmon

PURPOSE: To review the status of research programs on containment integrity, seismic margins, piping reliability, and other related matters.

LOCATION: WASHINGTON, DC

BACKGROUND:

What action is requested; by what date is it needed?

Meeting is requested by the Subcommittee Chairman in order to keep abreast of research needs and justifications. This information will be needed to plan for ACRS comments on the Research Program and Budget.

What will be done at this meeting?

Discuss the status of the research program with NRC research and regulatory staffs in order to plan to provide ACRS comments for research program and budget reviews.

What would be the consequence of postponing this meeting?

None, except that it is required by law that the ACRS provide Congress and the Commission periodic reports on the NRC Research Program and Budget.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

A status report will be provided with pertinent background information prior to the meeting.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
OCTOBER 8, 1985	RELIABILITY ASSURANCE (VALVES)	(MAJOR) Michelson, Ebersole, Kerr, Okrent, Reed, Ward

PURPOSE: To continue discussions on valve reliability. A risk perspective on valve performance will be sought. Also to be studied is the importance of valves from a safety standpoint. A discussion with Limitorque Co. is also expected.

LOCATION: WASHINGTON, DC

BACKGROUND:

What action is requested; by what date is it needed?

No action has been requested. This is a self-initiated task.

What will be done at this meeting?

This meeting will conclude a series of three meetings designed to explore the topic of valve reliability.

What would be the consequence of postponing this meeting?

No adverse impact.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

A Status Report will be issued prior to the meeting.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
OCTOBER 9, 1985	LONG RANGE PLAN FOR NRC	(MAJOR) Carbon, Lewis, Moeller, Remick, Siess, Wylie

PURPOSE: The Subcommittee will continue discussions on developing comments a long range plan for the NRC. Topics under discussion are primarily technical issues related to the regulation of nuclear power plant safety and safety regulation over the next 5 to 10 years.

LOCATION: WASHINGTON, DC

BACKGROUND:

What action is requested; by what date is it needed?

Currently a report addressed to the Commission or as input into a parallel OPE effort is expected. The current projection is to conclude this effort in October 1985.

What will be done at this meeting?

To be determined.

What would be the consequence of postponing this meeting?

Timeliness of effort would be effected. Would become out of phase with a parallel OPE effort on LRP.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

Dr. Carbon's latest review plan for this effort is available.

A Status Report will be prepared before the meeting.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
TO BE DETERMINED (AUGUST/SEPT.)	ADVANCED REACTORS	(EL-ZEFTAWY) Carbon, Mark, Siess Cons.:

PURPOSE: To discuss the proposed policy for regulation of advanced nuclear power plants.

LOCATION: WASHINGTON, DC

BACKGROUND:

What action is requested; by what date is it needed?

ACRS comments on the proposed policy statement for advanced reactors; 306th ACRS meeting (October 1985).

What will be done at this meeting?

Review the revised version of the policy statement. Prepare the report to the full Committee and suggested report to the Commission.

What would be the consequence of postponing this meeting?

Delay of ACRS comments to the Commission.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. Letter fm R. Fraley to J. Zerbe, dated 4/15/85.
2. SECY-84-453A - Regulatory policy for advanced reactors, dated 2/26/85.

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SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
TO BE DETERMINED (AUGUST/SEPTEMBER)	HUMAN FACTORS	(SCHIFFGENS) Ward, Reed, Remick, Wylie Cons.: Gimmy

PURPOSE: To explore methods for deciding what actions should be automated in nuclear power plant operation.

LOCATION: WASHINGTON, DC

BACKGROUND:

What action is requested; by what date is it needed?

Mr. Ward asked researchers from the University of Illinois to make a presentation to the Subcommittee.

What will be done at this meeting?

What would be the consequence of postponing this meeting?

No serious consequences from postponement that I can see.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

None at this time.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
TO BE DETERMINED (SEPTEMBER) (tentative)	DECAY HEAT REMOVAL SYSTEMS	(BOEHNERT) Ward, Ebersole, Etherington, Reed Cons.: Catton, Davis

PURPOSE: To continue the review of NRR resolution position for USI A-45.

LOCATION: WASHINGTON, DC

BACKGROUND:

What action is requested; by what date is it needed?

N/A

What will be done at this meeting?

Begin review of NRR's proposed resolution position for USI A-45.

What would be the consequence of postponing this meeting?

At this time, given the "spongyness" in the schedule, no definitive answer can be given to this question.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

To be provided when available.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
TO BE DETERMINED (SEPTEMBER/OCTOBER)	ECCS	(BOEHNERT) Ward, Ebersole, Etherington, Cons.: Catton, Schrock, Sullivan, Theofanous, Tien

PURPOSE: To continue the review of the joint NRC/B&W Owners Group/EPRI/B&W joint IST and related programs.

LOCATION: PALO ALTO, CA area

BACKGROUND:

What action is requested; by what date is it needed?

No specific action date -- part of ongoing Subcommittee review of Program.

What will be done at this meeting?

Continue Program review. Key discussion topics will largely be determined based on previous Subcommittee meeting in June. Also visit EPRI SRI-supported test facilities.

What would be the consequence of postponing this meeting?

No significant impact vis-a-vis facility visits.

Uncertain on program discussion pending results of June Subcommittee meeting in Alliance, OH.

PERTINENT DOCUMENTS AND THEIR AVAILABILITY:

To be provided on a timely basis.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
TO BE DETERMINED (OCTOBER)	JOINT REACTOR RADIOLOGICAL EFFECTS AND FIRE PROTECTION	(MERRILL/ALDERMAN) Moeller, Axtmann, Carbon, Ebersole, Michelson, Okrent, Reed, Siess, Wylie

PURPOSE: To review the increased N-16 radioactivity and fire protection problems in using H_2 addition to BWRs to reduce IGSCC.

LOCATION: WASHINGTON, DC

BACKGROUND:

What action is requested; by what date is it needed?

Comments to Staff. Needed by no specific date.

What will be done at this meeting?

Review of H_2 addition regarding N-16 activity and fire hazards.

What would be the consequence of postponing this meeting?

Meeting suggested by Committee members. Postponement of meeting will not have any serious consequences.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. Paper by V. Benaroya regarding subject available.
2. EPRI workshop report on subject, JAJARC #85-WH21 in ACRS office.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
TO BE DETERMINED (OCTOBER)	ATWS	(BOEHNERT) Kerr, Ebersole, Michelson, Ward Cons.: Lee, Lipinski, Davis

PURPOSE: To continue the review of the ATWS Rule implementation effort.

LOCATION: To be determined.

BACKGROUND:

What action is requested; by what date is it needed?

No specific action date.

What will be done at this meeting?

(See Purpose above)

What would be the consequence of postponing this meeting?

No significant consequences.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
TO BE DETERMINED (October)	QUALITY AND QUALITY ASSURANCE IN DESIGN AND CONSTRUCTION	(MAJOR) Remick, Michelson, Okrent, Reed, Siess, Ward, Wylie

PURPOSE: (1) To review the final rule on the, "Important to Safety Issue."
(2) To be briefed on the NRC Quality Assurance Program Implementation Plan.

LOCATION: WASHINGTON, DC

BACKGROUND:

What action is requested; by what date is it needed?

Approval of rule by IE QAB; by the August full Committee meeting.

What will be done at this meeting?

Review final rule in preparation to bring before the full Committee for comment/approval. Discussion of NRC's Quality Assurance Program Implementation Plan, for information.

What would be the consequence of postponing this meeting?

It could delay issuance of the final rule on Important to Safety Issue.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. "Quality Assurance Program Implementation Plan," SECY-85-65 is available.
2. Current (for public comment) version of the Important to Safety Rule is available. Revised version and response to public comments expected prior to meeting.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
TO BE DETERMINED (FALL) (tentative)	RELIABILITY AND PROBABILISTIC ASSESSMENT	(SAVIO)Okrent, Kerr, Ebersole, Lewis, Mark, Michelson, Siess, Ward, Wylie Cons.:

PURPOSE: To review the PRA for Millstone 3 (not an OL critical path item).

LOCATION: To be determined

BACKGROUND:

What action is requested; by what date is it needed?

Review of the Millstone 3 PRA; the meeting is to be scheduled after the completion of the NRC Staff's review of the PRA (estimated to be by the end of May 1985). There is no ACRS action date.

What will be done at this meeting?

Review of the Millstone 3 PRA for information.

What would be the consequence of postponing this meeting?

ACRS has stated that this review need not be completed prior to full power operation.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. Millstone 3 PRA (distributed).
2. NRC Staff report on the results of the NRC/LLNL review of the Millstone 3 PRA (expected by the end of May 1985).

NRR STAFF PRESENTATION TO THE

ACRS

SUBJECT: WATTS BAR IDVP AND ALLEGATIONS

DATE: JULY 12, 1985

PRESENTER: E. G. ADENSAM

PRESENTER'S TITLE/BRANCH/DIV:

CHIEF, LICENSING BRANCH NO. 4, DIVISION OF LICENSING

PRESENTER'S NRC TEL. NO.: 301-492-7831

WATTS BAR IDVP AND ALLEGATIONS

* GENERAL REQUIREMENTS OF AN IDVP

* WATTS BAR IDVP

* STAFF REVIEW

* WATTS BAR ALLEGATIONS

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GENERAL REQUIREMENTS OF AN *IDVP*

REVIEW ORGANIZATION

* COMPETENT

* INDEPENDENT

SCOPE OF REVIEW

* TAILORED TO CONCERNS

WATTS BAR IDVP

* BLACK & VEATCH

  -- AUDIT THE AFW SYSTEM OF WATTS BAR UNIT 1 TO ENSURE THAT THE SYSTEM HAS BEEN DESIGNED AND CONSTRUCTED IN ACCORDANCE WITH THE LICENSE APPLICATION AND LICENSE COMMITMENTS

* TVA

  - EVALUATE THE BLACK & VEATCH FINDINGS TO DETERMINE THEIR APPLICABILITY TO OTHER WATTS BAR SYSTEMS

STAFF REVIEW

BLACK & VEATCH EFFORT *(AFW SYSTEM)*

* B&V BOTH COMPETENT AND INDEPENDENT

* 97 FINDINGS REVIEWED OUT OF 428

* ONE FINDING STILL UNDER REVIEW
  - RESPONSE TO IE BULLETIN 79-02

TVA GENERIC EFFORT

* INSPECTION REVIEW

* DEDICATED REVIEW GROUP

OBJECTIVES OF REVIEW GROUP

1. TO DETERMINE IF THE TVA PROGRAM TO ADDRESS THE FINDINGS OF THE B&V REPORT WAS ADEQUATE WITH RESPECT TO EVALUATION OF GENERIC APPLICABILITY OF THE FINDINGS AND CORRECTIONS MADE TO THE PLANT DESIGN AND CONSTRUCTION

2. TO ADDRESS RECENT ALLEGATIONS REGARDING IDVP

DEDICATED REVIEW GROUP
REVIEW OF WATTS BAR IDVP

* B&V REPORTS

* TVA POLICY COMMITTEE REPORT AND RELATED DOCUMENTS

* NSRS REPORTS AND RELATED DOCUMENTS

* SITE VISIT

* PREPARE SER

TYPES OF CONCERNS RAISED

* GENERAL

* MOST PREVIOUSLY IDENTIFIED IN NSRS REPORTS, NCR's, CDR's, OTHER TVA CORRESPONDENCE, AND INSPECTION REPORTS

* CONCERNS REGARDING B&V

  - CLOSEOUT OF 500 ITEMS
  - ONLY ONE CONSTRUCTION SPECIFICATION LOOKED AT BY B&V
  - B&V DID NOT KNOW HOW THE PLANT WAS BUILT
  - B&V COMPARISON MADE WITH REGARD TO DESIGN CRITERIA NOT REGULATORY CRITERIA

* OTHER TYPES OF ISSUES:

  - MATERIAL TRACEABILITY
  - RECORDS OF TOTAL LOADS
  - STRUCTURAL STEEL WELD REQUIREMENTS
  - CABLE PROBLEMS
  - PROCUREMENT PROBLEMS
  - VOLTAGE REGULATION FOR BUSES
  - UNISTRUT USE
  - INADEQUATE D/G LOAD MARGINS

REVIEW PLAN

* ASSIGN RESPONSIBLE ORGANIZATION

* SCREEN FOR SAFETY SIGNIFICANCE AND IMPACT ON LICENSING

* REVIEW RESPONSES TO MAY 16, AND MAY 30, 1985 LETTERS

* REVIEW AND EVALUATE ISSUE

* PREPARE SER OR INSPECTION REPORT

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

May 16, 1985

Docket Nos: 50-390, 50-391
and 50-438, 50-439

Mr. H. G. Parris
Manager of Power
Tennessee Valley Authority
500A Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Dear Mr. Parris:

Subject: Concerns Regarding TVA Construction Sites

Enclosure 1 lists eleven concerns about your Watts Bar facility that have been communicated to the NRC. We ask that you review these concerns and take appropriate steps to assure that your programs and implementation of those programs in these areas are adequate to meet applicable requirements and to support safe operation of the facility. Furthermore, we ask that you address any generic implications of these issues. We recognize that some of these concerns are not very specific. However, that lack of specificity should not lead you to assume there is no basis for concern. Your review of these matters should be broad enough for you to certify the safety significance of these concerns. Pursuant to Section 182 of the Atomic Energy Act of 1954, as amended, we ask that you provide the results of your review as soon as possible to assist us in our evaluation of these concerns. Enclosure 2 lists a number of questions that we have regarding these concerns. Please provide us with your response as soon as possible.

We also ask that you identify any outstanding cases currently under review by TVA's Office of the General Counsel regarding employee harassment, reprisals or intimidation.

We recently received some additional concerns (see Enclosure 3) regarding both your Watts Bar and Bellefonte facilities. You should review these to determine that no new issues related to safe plant operation have been identified. We regret some omissions occur, but this is how they were received by us.

I suggest we meet as soon as you are prepared to discuss your schedule for responding to this letter. Should you have any questions on this matter, please refer them to E. Adensam of my staff on FTS 492-7831.

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

Sincerely,

Hugh L. Thompson, Jr., Director
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc: See next page

WATTS BAR

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Chattanooga, Tennessee 37401

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General Counsel
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Mr. D. Checcet
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Mr. Ralph Shell
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Mr. Donald L. Williams, Jr.
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Resident Inspector/Watts Bar NPS
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Regional Administrator
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101 Marietta Street, N.W., Suite 2900
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Mr. David Ellis
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Mr. Mark J. Burzynski
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Watts Bar NP
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BELLEFONTE

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ENCLOSURE 1

CONCERNS RELATED TO WATTS BAR

1. A concern has been expressed that there is no material control of ASME bolts smaller than 1"; and, therefore, the bolts < 1" are mixed up and no one knows where the good ones are.

2. A concern has been expressed that electrical hangers have been modified after their initial inspection and not reinspected.

3. A concern has been expressed that field modifications have been implemented on components, piping, supports, structures, and embedments that resulted in no accurate records of total loads on these elements.

4. A concern has been expressed that the cumulative effect of tolerances has not been factored into the design and drawings, especially with respect to hanger location.

5. Several concerns have been raised regarding the Independent Design Verification Program conducted by Black & Veatch. These are:

   a) a concern regarding the close out of about 500 items,

   b) a concern that only one construction specification was looked at by Black & Veatch in their review,

   c) a concern that Black & Veatch did not know how the plant was actually built, and

   d) a concern that Black & Veatch only compared the system's design and construction to its design criteria, not to the underlying regulatory criteria.

6. A concern was expressed with respect to the use of three Q lists that all differ.

7. A concern was expressed that the method of identifying NCR's and the use of the Inspection Rejection Notice (IRN) system effectively negated the NCR process. It was submitted this was so because an NCR was only generated when 1) the equipment/component/system/etc. had been previously inspected and accepted; 2) the records for that inspection were in the vault, and 3) there was a subsequent discovery that something was wrong; however, if there were a problem identified in an initial inspection an IRN is generated.

8. A concern has been expressed with respect to structural steel welding requirements in that TVA is using a different code than the code normally used by the industry for structural steel welding.

9. A concern has been expressed that in FSAR amendment #53 TVA lessened the experience requirements for the plant manager.

10. A concern has been expressed regarding weld filler material control, especially in the area of storage and issuance of materials.

11. A concern has been made that the Quality Assurance (QA) organization at construction sites lacks the independence required by NRC regulations. Also the statement was made that inadequate QA organization independence problems were identified in a Management Analysis Company (MAC) report, "Assessment of Organizational Change in the Tennessee Valley Authority Power Program and the Nuclear Quality Assurance Program."

ENCLOSURE 2

QUESTIONS ON WATTS BAR CONCERNS

1. With respect to concern 1 our initial inspection during the week of April 29, 1985, identified instances where unmarked bolts were installed in the facility on ASME components and supports. In addition, the staff also learned that two NCRs (1979 and 1981) have been issued regarding the purchasing and installation of bolts without required markings. Describe the reasons for the apparent QC breakdown in bolt control, the reason for the repeated NCR in 1981 and your evaluation of whether this occurred in other QC areas. In view of the above, what is your basis for determining compliance with criterion VIII of Appendix B to 10 CFR 50? If documentation does not exist which demonstrates compliance with this regulation, describe the process and provide sample documentation which leads you to conclude that you comply with this regulation. Demonstrate that bolts less than 1", which have been installed, comply with all applicable ASME Code requirements related to identification and control.

2. With respect to concern 2, please review your records to determine if such modifications have occurred and, if they did, what assurance you have that the modifications have been properly reinspected. This review should include applicable work requests. Verify that documentation exists to demonstrate that modified electrical hangers were designed and constructed pursuant to applicable FSAR commitments. To the extent such documentation does not exist, what is the basis for concluding that electrical hangers, as currently installed, comply with 10 CFR 50, Appendix B, Criterion X?

3. With respect to concerns 3 and 4, 50.55 Interim Report No. 1 on NRC WBNCEB8419 states that TVA's drawing series 47A050 includes several tolerances (e.g., location of concrete anchorages, movement of attachments, and modification of baseplates) such that the cumulative effect of these tolerances may result in significant increases in baseplate stresses and anchor bolt loads. Interim Report No. 1 also states that there is no evidence that these potential increases due to cumulative effects were considered in the design of various supports and that the cumulative effect of these tolerances could increase baseplate stress by 150% and anchor bolt load by 50%. Provide the TVA engineering specifications establishing acceptable dimensional installation tolerances for supports, baseplates, and anchorages. Provide the analytical bases for establishing the above procedures which demonstrate that the effect the tolerances have on the stresses on loads in interfacing components and structures will cause these values to exceed their allowable limits. Provide the process used when field modifications are made and confirm that each modification exceeding TVA engineering specification installation tolerances has been analyzed to demonstrate continued compliance with design allowable parameters. Provide sample documentation which demonstrates how this process has been used.

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4. With respect to concern 5, please review this concern to determine no new issues are raised which would impact your assurance that the Black & Veatch review was properly designed and conducted and that TVA's close out of identified open items was consistent with your licensing commitments and safe operation of the plant

   In addition, please address the following questions:

   The NRC staff can identify only one General Construction Specification (GCS-G-32) in the documents reviewed by B&V. How many other General Construction Specifications are applicable to the auxiliary feedwater system? If you identify other applicable General Construction Specifications that were not reviewed by B&V, how could TVA use B&V to support a conclusion that construction complied with the FSAR commitment? What corrective actions (i.e., design, hardware, procedural modifications) have been taken as a result of the B&V review? Does B&V agree that these actions resolve the concerns expressed in the B&V findings? What specific actions have been taken in systems other than the Auxiliary Feedwater System to determine the extent to which deviations found by B&V in the AFW system existed in other systems? How have such actions been documented?

5. With respect to concern 6, identify the documents that demonstrate your compliance with Criterion II, Appendix B, 10CFR Part 50, for maintaining a Q list from the date of the construction permit (CP).

6. With respect to concern 7, is this a proper description of the NCR process? Please verify and certify that reporting of deficiencies meets your licensing commitments and the regulations and that IRN's and NCR's are properly controlled. Is there a master file of IRN's and their resolutions? You may wish to consider having your Quality Technology Company Employee Response Team solicit employee views regarding improper use of the IRN process in lieu of the NCR process.

7. With respect to concern 8 please verify your code use for this welding to assure regulations and licensing commitments have been met. Verify your implementation of other types of welding conformed to the accepted standards. Please provide memoranda or other documents indicating problems with TVA's AWS welding program not previously provided to NRC.

8. With respect to concern 10, now does your program assure the ASME Code requirements of 10 CFR 50.55a are met and 10 CFR Part 50, Appendix B Criterion VIII traceability requirements are met. To what version of the ASME Code was TVA committed in the CP? Did this version of the ASME code require traceability of filler material to welds by heat and lot numbers? What internal or external approvals for your program were required and received?

What version of the specification GCS-G29M, Process Specification 1.M.3.2(RO) & 1.M.3.1(R7) was used prior to 1/12/83 and 1/13/83? There appears to be an inconsistency between these two process specifications in that 1.M.3.2(RO) for power boilers is more stringent than 1.M.3.1 (R7) for nuclear plants. Describe how TVA is implementing these process specifications in the current version of GCS-G29M in the field?

9. With respect to concern 11, describe the adequacy of the independence of the QA organization as it applies to Watts Bar. In addition, describe actions taken by TVA to resolve problems identified in the MAC report and actions TVA is taking with respect to the report's recommendations. Provide any analysis which has been conducted by TVA to determine the extent to which Watts Bar design and construction quality may have been compromised as a consequence of deficiencies enumerated by the MAC report. If no analysis has been conducted, do you intend to conduct such an analysis? If not, why not?

Enclosure 3

Electrical, I&C and Diesel Generators

- Electrical and I&C Regulations (Reg. Guides, NUREGs, Bulletins and Notices) have been ignored and violated to a very large degree at all plants.
  - Caused by a lack of knowledge by personnel
  - Caused by a poor attitude toward safety and regulations by personnel
  - Caused by a lack of knowledge of industry positions on regulations
- 5% voltage drop at each plant causes problems
  - Cycles diesel generators unnecessarily, degrading reliability
  - Too many plant shutdowns
  - TVA compensates by operating buses at higher than normal voltage ratings, anticipating voltage reductions, stressing equipment and components unnecessarily and reducing their lives and reliabilities
  - Inadequate voltage regulation for buses
- Diesel Generator margins inadequate
  - TVA has added DGs to BF, Sequoyah and Watts Bar
  - Each time a question is raised, TVA must conduct another study
  - TVA adds [illegible] without upgrading licensing documentation
- Diesel generator reliability problems
  - Requires reliability upgrading program
  - Requires reduction in number of starts
  - Requires much attention given to testing program
  - Requires preventative maintenance upgrading program
  - Requires more interaction with INPO and other utilities, as well as vendors, to establish resolutions to problems

- Electrical separation and physical separation of redundant wiring and cabling and for equipment and components are all inadequate at all plants
  - Detailed reviews need to be made (They are so extensive that a consultant probably should be used, providing independence from TVA)

- Environmental Qualification of electrical and I&C equipment and components is inadequate at all plants
  - Qualification was often not done
  - If done, records do not exist in many cases, resulting in requalification or replacement of items
  - Current upgrade programs needs scrutiny

- WBN - (maybe other plants) Class 1E and Non-Class 1E Batteries are unacceptably supported (no battery tie-downs)
  - Unistrut supports unacceptably used

- Human Factors engineering and/or reviews have not been implemented for control panels and stations at WBN (possibly other plants also) - Violation of intent of NUREG-0700
  - Too many poor engineering practices in this area

- Out of service tags for valves, electrical equipment, etc., at Bellefonte have been violated everywhere
  - Extremely serious personnel safety problem

- Thermal overload bypass and indication problems at WBN - probably have similar problems meeting Reg. Guide 1.97 at other plants

- There are cable ampacity problems at WBN where derating was not properly considered
  - Probably problems at other plants

- Inadequate management, control and <u>status listing</u> of a.c. and d.c. electrical loads, including diesel generator loads

- Inadequate control of or preparation of calculations for loads
- Inadequate management and control of load margins, including electrical loads and mechanical loads (heat, BHP, etc) that translate into electrical loads

° Cable tray fill criteria of 60% for I&C cables is inadequate

- National Electrical Code allows 40% and 50% on exception basis. TVA violates code
- Industry practice is 40%
- The situation is even worse with the addition of spray-on fire retardent materials which take up space in trays

° Cable pull tension monitoring is lax

° Cable bending radii problems

° Computer cable routing program inadequate and its status system is inadequate

° Cable trays are too heavily filled; cables [illegible]

° Cable megger readings are not stored as QA records, losing traceability

° Construction Test and Installation Specs (Called General Construction Specs with G-   numbers) are often incomplete and inadequate

° Electrical testing and planning inadequate

- Engineering either does not address testing or does so inadequately
- Acceptance criteria is inadequate to nonexistent

° Electrical Standards and Guides are treated as guides and are not adequately incorporated in design criteria as requirements

° Electrical design criteria, where it exists, is not complete, is vague, and in general is inadequate

- Cabling is routed outside trays, coiled on tray supports or floors, tied on sides of trays and supports, tied on bottoms of trays, etc. All this and more exists at WBN, where extremely bad cable practices exist such as the above and 90° wire bends [illegible]

- Between 400 and 500 breakers were unacceptably set at WBN. EN DES practices and attitudes concerning these were poor. The National Electrical Code and good engineering practices were violated.

- Many cable trays at WBN are full, some exceeding 100% tray capacities, and they are not identified at site or in computer status as full

- Wall penetrations of cable trays are not identified by name and/or number at WBN

- Lighting fixtures at WBN are not properly restrained and caged to prevent them from becoming missiles or swinging missiles during seismic events

- WBN - (and Possibly other plants) - Unistrut material is used to support instruments, pipes, conduit, control stations and panels, fluid piping on skids, instrument lines, CO_2 fire protection piping, fire protection water piping, lighting, etc.

  - All unacceptable use for Seismic Category 1 support

  - Items supported as such may either fail or become missiles to cause other [Illegible]

- TVA commitments in FSAR, SER, and NRC Question Responses are treated lightly and are not being met in a wide number of areas

  - Personnel do not follow regulations and commitments, and do not think they even need to report deviations or change commitments and obtain NRC acceptance

- TVA safety and licensing evaluations by EN DES (Including NEB) are inadequate and appear too much in cover up mode

- TVA personnel have attitude problems in meeting regulatory commitments

- Too many crafts and others on site at WBN

- Gross lack of knowledge of regulations and their seriousness by TVA personnel at all levels

- Lack of frequent visits to sites by Designers

- Communications problems among designers, constructers and operation personnel

- Procurement specs, drawings and vendor supplied documents not per as-built and/or as delivered configurations

  - TVA inadequately reviews vendor work

  - TVA receipt and inspection of equipment are inadequate (Example: TVA in many cases does not inspect until ready to install - not when received)

- Construction process does not always follow EN DES requirements documents or vendor requirments/instructions

  - These do not always get included on as-built documents

  - Too much after-the-fact approval

  - QC inspection is often inadequate - (It only takes a walk thru a plant such as WBN to see examples everywhere)

- Engineering (EN DES) inadequately addresses and considers operation, maintenance, testing and construction requirements and general industry practices, in the design process

  - There are no forced interactions with other utilities

  - There is no formal system to track and assign commitments for problems identified to INPO

  - There is poor tracking of NRC experience information

- Improper reporting of events at operating plants or in design/construction

  - TVA personnel are inadequately trained and not knowledgeable in what is reportable

- Lack of adequate (or any) configuration control (management) in EN DES or at sites
  - Poor interface control between systems
- Lack of traceability of design requirements
  - Standard answer is "Its TVA Practice"
- Design/installation drawings do not always represent or include design requirements
  - Design guides or standards are utilized only when designer wants to use them
  - Design guides/standards inadequate in many areas
  - These are misused - applicable parts are [Illegible]
- Material control is poor
  - Traceability of requirements, paperwork, and materials are inadequate
  - Paperwork for quality records is poor
  - Storage requirements implementation is poor
  - Handling of equipment in storage and during and after construction is poor. WBN equipment in many cases is in poor condition and filthy dirty inside and outside
  - Equipment receipt and inspection is inadequate (identified previously)
  - These problems exist at Bellefonte and WBN (probably elsewhere)
- Lack of adequate tracking for EN DES commitments and design changes
- Lack of good status system (punch lists) for completion of commitments and completion of NRC actions, and completion of work at sites. Plant construction, pre-op, etc. status is poor
- Project Engineering inadequate (or nonexistent) to incorporate TVA and industry operating [Illegible]

° Calculation Problems

- Some are not ever prepared
- Some are inadequate in scope and quality
- Some are not stored as quality records, but are destroyed
- Traceability of design requirements is impacted due to above problems
- There is inadequate interface control and control of calculations

° TVA has set up design criteria (WBN) and, after the fact, have inactivated a large percentage of criteria

° As-Built drawings and documents are nonexistent or in poor condition in many cases

° TVA does not adequately (or at all) independently verify vendor calculations or designs.

- There are no design reviews of vendor design

° TVA does not conduct independent design reviews of its work

° QA has not effectively audited the design and construction process

° Lack of coordination of effects of upcoming (near or long term) design changes with all disciplines and site construction

- inadequate evaluation of impacts (not under configuration control)

° Lack of accountability of TVA personnel and management for not following procedures, regulations, etc. and for not doing adequate and acceptable job

° Too much blame on QA for quality problems versus emphasizing and demanding an ethic to do it right the first time. Put quality into design and construction

- Commitment (action) system in TVA nonexistent
  - No action party and schedule
- Lack of effective communications and interface control among organizations with EN DES - Branches, Projects, Procurement, etc.
- Protective and defensive attitudes of NEB and various Branch/Project groups concerning problems rather than an attitude to admit [Illegible]
- Lack of proper environments and fire protection in equipment storage areas
- Lack of knowledge (on site and in EN DES) as to status of QCIRs and IRNs
- Untimely closeout of ECNs
  - Lack of knowledge of status of ECNs or designs affected

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

May 16, 1985

Docket Nos: 50-390, 50-391
and 50-438, 50-439

Mr. H. G. Parris
Manager of Power
Tennessee Valley Authority
500A Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Dear Mr. Parris:

Subject: Letter on TVA Construction Sites Dated May 16, 1985

In Mr. H. Thompson's letter to you dated May 16, 1985, "Concerns Regarding TVA Construction Sites", there was an error of omission in Question 3 of Enclosure 2. The second request which reads "Provide the analytical bases for establishing the above procedures which demonstrate that the effect the tolerances have on the stresses on loads in interfacing components and structures will cause these values to exceed their allowable limits." should read as follows:

> "Provide the analytical bases for establishing the above procedures which demonstrate that the effect the tolerances have on the stresses or loads in interfacing components and structures will not cause these values to exceed their allowable limits."

We apologize for any inconvenience and ask that you respond to the restated question above.

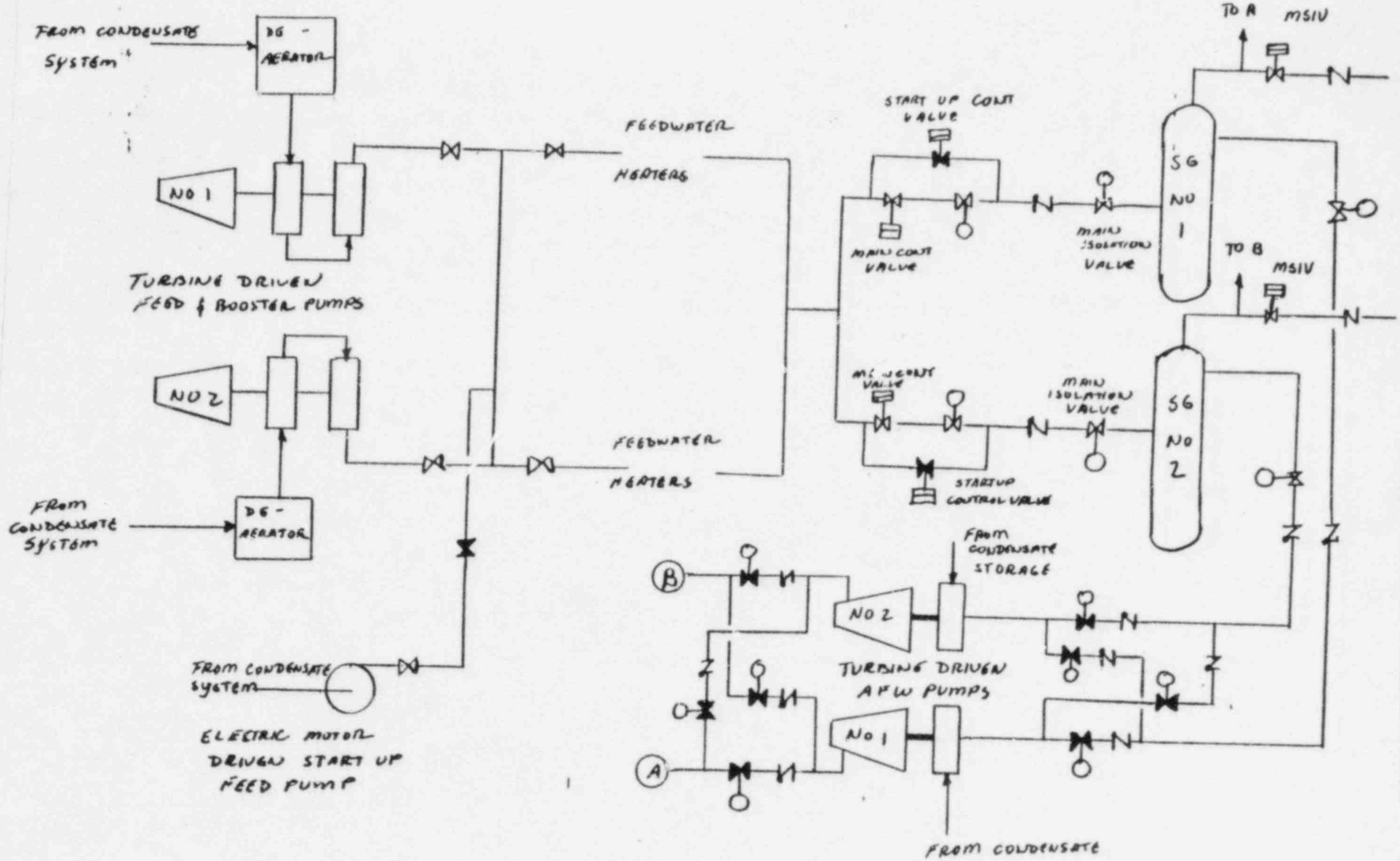
Sincerely,

Elinor G. Adensam

Elinor G. Adensam, Chief
Licensing Branch No. 4
Division of Licensing

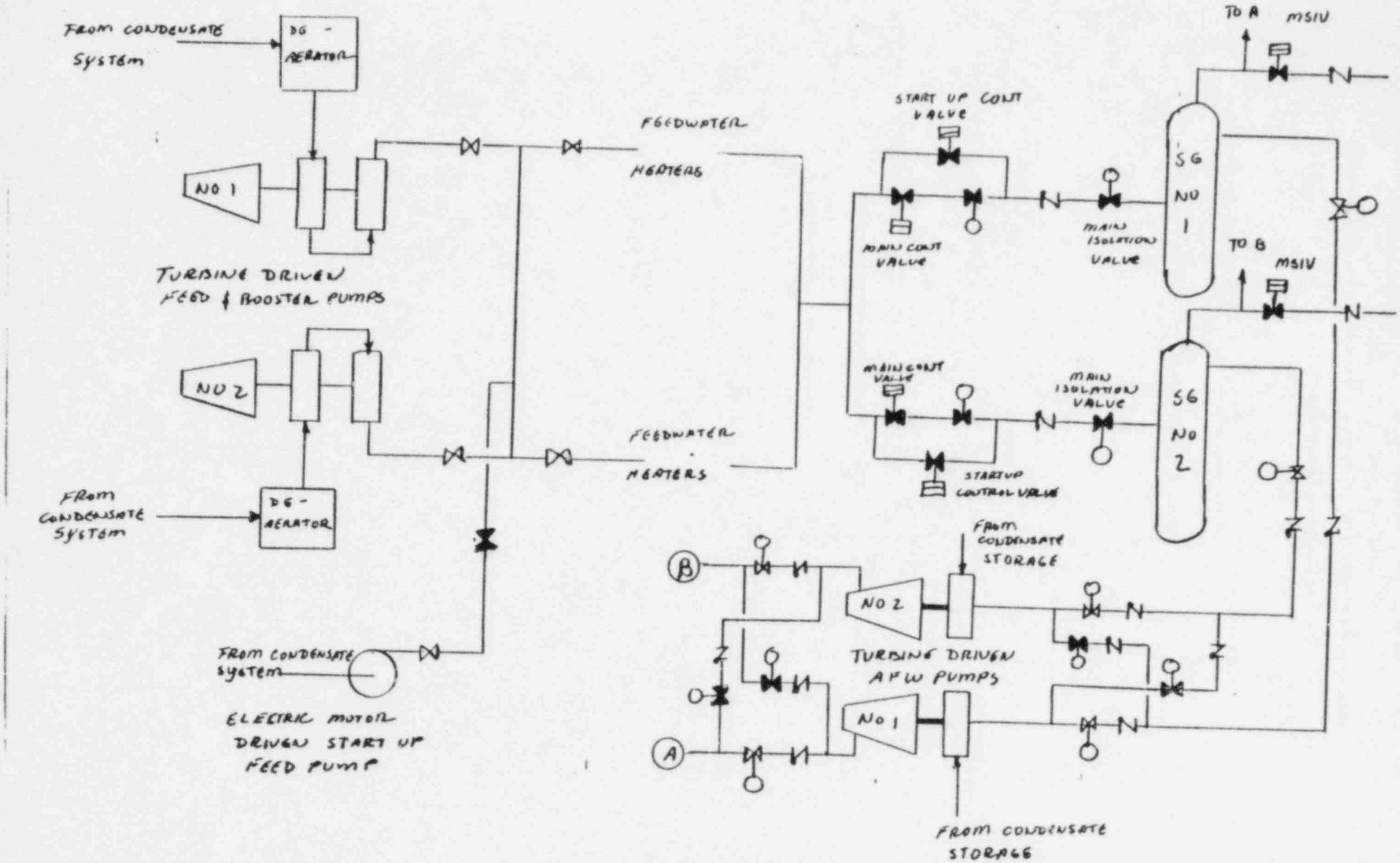
cc: See next page

A-69



DAVIS BESSE FEEDWATER SYSTEMS

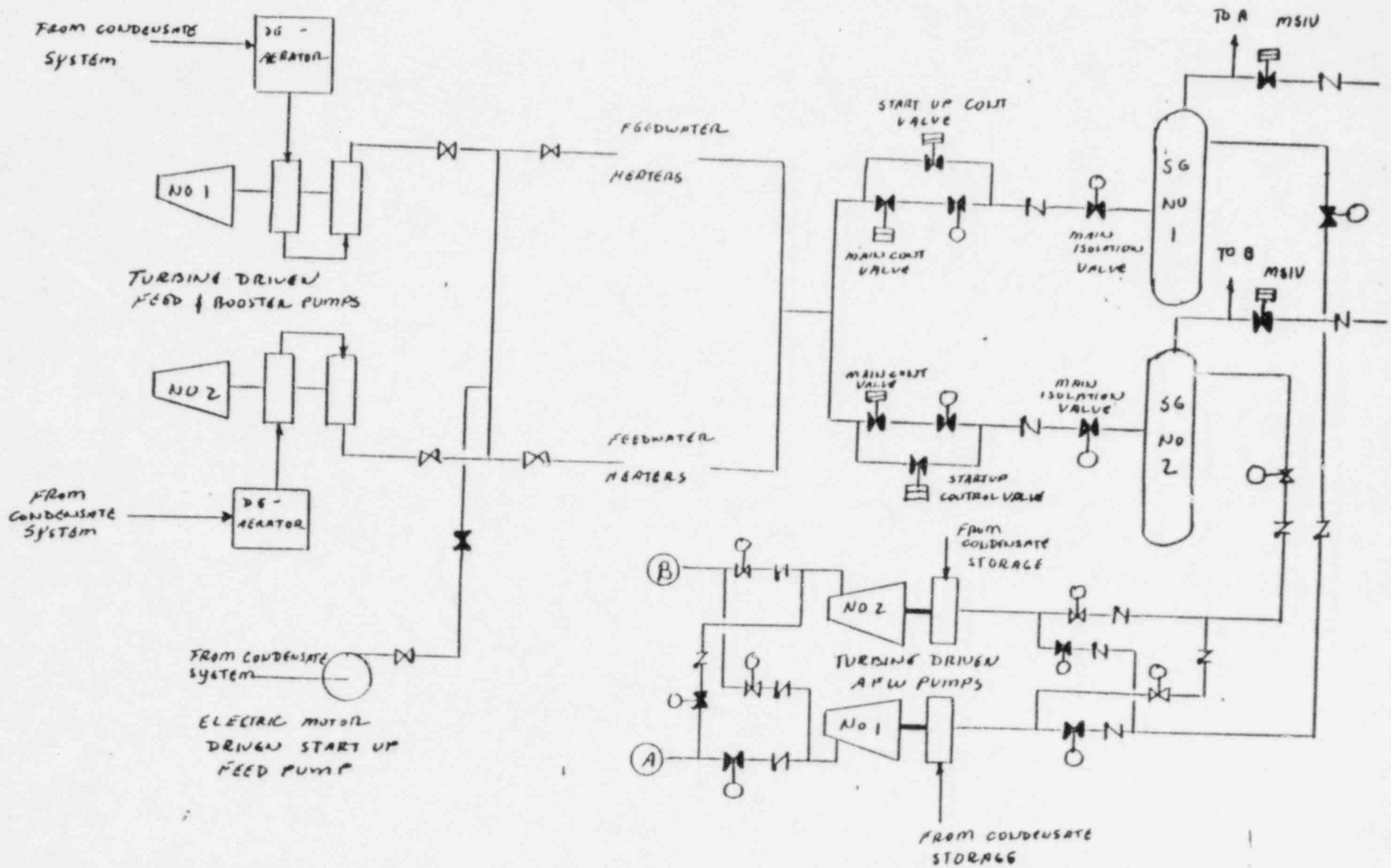
NORMAL OPERATION



DAVIS BESSE FEEDWATER SYSTEMS

AFTER SFRCS TRIP ON LOW ST GEN LEVEL OR HIGH STEAM FEED PRES DIF

A-71



DAVIS BESSE FEEDWATER SYSTEMS

AFTER SFRCS LOW PRESSURE TRIP ON ST GEN NO 1

TO: George La.

APPENDIX VII
DESCRIPTION OF DAVIS-BESSE LOSS OF MFW AND AFW

NRC Form 366A

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

APPROVED OMB NO 3150-0104 EXPIRES 8/31/85

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

Description of Occurrence: Davis-Besse Unit 1 was operating at 90 percent of full power with the No. 1 Main Feedwater Pump, MFP, operating in automatic and the No. 2 MFP in manual control. This configuration was established to limit the susceptibility of the No. 2 MFP to control problems which had previously occurred. The control problems occurred only after a reactor trip and appeared to be connected to the automatic mode of operation. This configuration, therefore, permitted automatic feedwater control during operation and offered improved availability of at least one MFP in the event of a reactor trip.

At 0135 hours, the No. 1 MFP tripped on overspeed due to an unrelated control problem. The Control Room operators increased the No. 2 MFP speed, but it did not have adequate capacity, for the existing reactor power. The reactor tripped on high Reactor Coolant System, RCS, pressure at 0135:30 hours, tripping the turbine. Reactor power was at approximately 80 percent of full power at the time of the trip.

Immediately following the trip, a spurious Steam and Feedwater Rupture Control System, SFRCS, low steam generator level full trip occurred on Channel 2, an SFRCS full trip alarm was received, and both main steam isolation valves, MSIVs, closed. An actual low steam generator level did not exist at this time. This spurious trip resulted in a partial actuation of the SFRCS components since only the MSIVs actuated. When the MSIVs closed, the main steam supply was isolated to the MFPs. The No. 2 MFP continued to supply feedwater until approximately 0140 hours at which time its discharge pressure was not high enough to supply feedwater to the steam generators. The level in the steam generators which was being maintained at the low level limit setpoint (35 inches) began to decrease. SFRCS Actuation Channel No. 1 then automatically initiated on low steam generator level, starting the No. 1 Auxiliary Feedwater Pump, AFP, to feed the No. 1 Steam Generator (see Attachment 1 for a diagram of SFRCS actuated components).

At 0141:08 hours, a Control Room operator attempted to manually initiate the SFRCS, however, he incorrectly actuated the SFRCS on low steam pressure instead of the desired low steam generator level. Therefore, each SFRCS actuation channel sensed that its respective steam generator was depressurized. SFRCS Actuation Channel No. 1 then attempted to align AFP No. 1 to feed Steam Generator No. 2. SFRCS Actuation Channel No. 2 attempted to align AFP No. 2 to feed Steam Generator No. 1. Both actuation channels closed their respective Auxiliary Feedwater Containment Isolation Valves (AF599, AF608), which prevented any auxiliary feedwater flow from reaching the steam generators. At 0141:31 hours, AFP No. 1 tripped on overspeed. At 0141:44 hours, AFP No. 2 tripped on overspeed.

At 0142:00 hours, an operator recognized the manual initiation error and reset the low pressure SFRCS buttons, and pushed the low steam generator level SFRCS manual actuation buttons. Since both SFRCS actuation channels were already tripped on low steam generator level, the SFRCS automatically began to realign the AFPs when the low pressure buttons were reset. However, the Auxiliary Feedwater Containment Isolation Valves (AF599, AF608) did not automatically open. The operators attempted to open these valves from the Control Room by operating their control switches and by reinitializing the SFRCS. These attempts failed to open the valves. Equipment

NRC FORM 366A

NRC Form 366A (9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

Operators were sent to open these valves locally, and when the valves were moved off their closed seats utilizing the manual handwheels, the motor operator responded and fully opened the valves. During this period, attempts were also being made to restart the AFPs and preparations were underway to start the motor operated Startup Feedwater Pump.

The RCS average temperature was increasing due to the lack of primary to secondary heat transfer. RCS pressure was increasing due to the decreasing density of the RCS water and increasing pressurizer level. RCS pressure increased to the Power Operated Relief Valve, PORV, setpoint (2425 psig). The PORV cycled a total of three times, relieving pressurizer pressure to the Quench Tank. Following the third opening, the PORV failed to reclose at the proper RCS pressure. The Control Room operator observed the primary plant conditions and closed the block valve on the PORV. RCS pressure was at approximately 2075 psig when the block valve closed. The Quench Tank contained the discharges from the PORV.

At approximately 0151 hours, the operators placed the Startup Feedwater Pump in operation to supply the steam generators. Steam Generator No. 1 pressure had decreased to approximately 750 psig. Steam Generator No. 1 repressurized to approximately 900 psig from the Startup Feedwater Pump. Steam Generator No. 2 had decreased to 920 psig. At 0152 hours, the No. 2 AFP was returned to service by the operators locally. Maximum RCS temperature had reached approximately 592 degrees Fahrenheit. At 0155 hours, the No. 1 AFP was returned to service locally by the operators. Control of the AFP turbines was maintained locally by an operator at the turbine trip throttle valve. At 0158 hours, RCS average temperature was restored to the normal post trip temperature. The cooldown of the RCS lowered RCS pressure to a minimum of approximately 1720 psig. Operators manually started the No. 1 High Pressure Injection, HPI, Pump in the piggyback mode (Decay Heat Pump No. 1 supplying the suction to the HPI Pump No. 1) in precautionary anticipation of the rapid cooldown. Only a slight amount of water (less than 50 gallons) needed to be injected.

Several other equipment malfunctions occurred which did not affect the physical plant response. One source range nuclear instrumentation, NI, channel was inoperable prior to the trip. The remaining source range NI channel failed to indicate properly when it was automatically energized after the trip. The display units for the Safety Parameter Display System, SPDS, were inoperable in the Control Room at the time of the trip. At 0158:40 hours, the suction of the No. 1 AFP automatically transferred from the Condensate Storage Tank, CST, to the Service Water System. The operator manually realigned the pump suction back to the CST. No significant amount of service water was added to the steam generator during the recovery from the transient. It was noticed that the pneumatic operator on one main turbine bypass valve was damaged, preventing the valve from being opened. This did not affect the post transient response of the plant.

Additional details of the plant transient and corrective actions will be provided in the restart report response to the Region III Confirmatory Action Letter (85-06). Attachment 2 provides a chronological listing of the event. This report is being submitted in compliance with paragraph 50.73(a)(2)(i), 50.73(a)(2)(iv), 50.73(a)(2)(v),

50.73(a)(2)(vi), and 50.73(a)(2)(vii). This report also satisfies the reporting requirements for a Emergency Core Cooling System Actuation Special Report, Section 6.9.2(a) of Technical Specifications. This was the fourth high pressure injection actuation cycle to date.

Designation of Apparent Cause of Occurrence: This transient was initiated when the No. 1 MFP developed control problems and tripped on overspeed. The plant tripped on high RCS pressure due to inadequate feedwater being supplied from the No. 2 MFP during the plant runback. The cause of the MFP overspeed tripping was determined to be due to a bad speed summation and valve lift reference circuit board card in the MFP control. A frequency to voltage converter chip had failed. The board is being returned to General Electric for further analysis on the root cause of the failure.

The root cause of the MSIV closure has not yet been determined. It is presently believed that the MSIVs properly responded to a momentary low level SFRCS trip. Further investigations will follow once an action plan is completed.

The cause of the SFRCS spurious trip on low steam generator level has not yet been positively determined. Troubleshooting will begin in accordance with the action plan. However, it is presently believed that the steam generator level sensing channels are sensing an extremely rapid secondary side pressure transient that occurs in the steam generator following the turbine stop valve closure on a turbine trip. These level transmitters share a common set of sensing lines with transmitters which were replaced during the 1984 Refueling Outage. Prior to the 1984 Refueling Outage, Bailey BY level transmitters were installed which have now been replaced by Rosemont Model 1153. Since these Rosemont transmitters have no significant displacement required for operation, while the Bailey BYs required a volume displacement to operate the bellows, it is postulated that the responsiveness of the sensing line and transmitter arrangement has been greatly increased by this change. This increased responsiveness allowed the SFRCS to sense the rapid secondary side pressure transients which previously were undetected. Further analysis of this condition is underway.

The cause of the incorrect manual SFRCS initiation was personnel error attributed to a poor switch layout. These SFRCS manual initiation pushbuttons had been identified in the Detailed Control Room Design Review as one of the principal items needing human engineering improvements. There are two adjacent vertical columns of buttons with five buttons in each column (see Attachment 3 for arrangement details). Each column represents one SFRCS actuation channel. To manually initiate both channels of the SFRCS for steam generator low level, the operator should have depressed the fourth button from the top in each column; instead, the two top buttons were depressed. A design change had been developed prior to this event to improve the switch layout and will be implemented during this outage.

The cause of the AFPs tripping on overspeed after initiation has not yet been positively determined. Water flashing through the nozzles of the AFP turbines is thought to be a contributor. The governor was inspected on both AFP turbines, and no contributing factors to the overspeed were seen. Further investigations and testing are planned.

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

The cause of the Auxiliary Feedwater Valves (AF599 and AF608) not opening by the motor operator was determined to be a combination of a high differential pressure and an improperly set torque switch bypass limit switch. With this torque switch bypass limit switch improperly set, the motor operator was allowed to torque out during the opening stroke. These valves are open during normal operation and were closed by the incorrect manual initiation of the SFRCS. If the AFPs had been operating at the time the valves attempted to open, the differential pressure across the valves would have been significantly lower, and the valves should have opened to allow the auxiliary feedwater flow to occur. These valves were stroked following the transient and ability of the valves to open (without significant differential pressure) was verified. Recent testing also verified that the valve operators torque out under high differential pressure with the improperly set torque switch bypass limit switch. Further investigations are in progress.

The cause of the control problems with the AFPs after the overspeed was reset is presently attributed to the difficulty in opening the trip throttle valves. No mechanical deficiencies were found while investigating the resetting of the overspeed trip device/linkage. Further investigations are in progress.

The cause of the PORV not properly reseating has not yet been positively identified. Operator observations at the time of the transient indicate that the electronic controls signal was calling for the valve to reclose. A visual inspection and disassembly of the PORV failed to identify the cause. Further investigations are in progress.

The two, independent SPDS display units were inoperable due to separate but similar failures in the data transmission system between the Control Room terminals and their respective processors. The failures are of an intermittent nature and the exact cause is still under investigation.

The cause of the source range NIs inoperability has not been positively identified. The failure of the source range NIs has been a repetitive problem at Davis-Besse with repeated investigations failing to determine the root cause. Since 1977, the boron trifluoride detectors, preamp, and cable in Containment have been replaced, along with the modules in the Reactor Protection System and a reworking of the grounding on the preamp and count rate amplifier module connections. No positive effect on the total elimination of the spiking, nor the erroneous/elevated count rate has occurred from these corrective actions. Further review is being performed on the possibility of ground loops, induced current or voltage from adjacent cables, or intermittent problems with the count rate amplifier module.

The cause of the Turbine Bypass Valve 2-2 damage has not been identified. The valve was disassembled and the actuator stem extension piece was found bent, four parts were missing, and the valve internals were found loose. Several valve parts were shipped to the vendor for further analysis. Further review of the turbine bypass valve failure is underway.

The cause of the inadvertent AFP No. 1 suction supply transfer from the Condensate Storage Tank to Service Water has not yet been determined. Testing and other investigations are currently being performed.

The cause of MS-106 apparently cycling in about one third of the expected stroked time is still under investigation.

Analysis of Occurrence: This event involved a temporary loss of feedwater to the steam generators. This event was bounded by the analyses previously performed (see Toledo Edison submittals to NRC Serial No. 506 dated May 22, 1979, and Serial No. 517 dated June 15, 1979), which analyzed a loss of all feedwater for 30 minutes following a reactor trip. These analyses showed that as long as either:

1) Auxiliary Feedwater is restored within 30 minutes of the loss of main feedwater,

OR

2) Within 30 minutes, at least one makeup pump and the PORV are available for primary cooling (feed and bleed) and the Startup Feedwater Pump is available to supply a steam generator,

fuel cladding temperatures would remain within a few degrees of saturated fluid temperature and no cladding rupture or metal water reaction would occur.

Operator interviews indicated that the shift was fully aware of the core status and were prepared to implement the "feed and bleed" core cooling method if the auxiliary feedwater was not restored. The Startup Feedwater Pump was available throughout the event and in fact was placed in service within ten minutes of the tripping of the AFPs. Auxiliary Feedwater was restored within 12 minutes of the loss of feedwater. These response times and equipment availability are well within the loss of feedwater analyses.

At no time during the event was the required subcooled margin (20 degrees Fahrenheit) lost. The reactor coolant pumps continued to operate throughout the event. The primary code safety valves were not challenged and at no time during the event did the RCS pressure or temperature exceed the allowable values. The maximum temperature reached was below the normal operating temperature for the hot leg temperature. There is no indication of any fuel cladding degradation based on the reactor coolant radiochemistry analysis.

An analysis has been performed by Babcock & Wilcox to determine if the transient adversely affected the steam generators. Conditions and components specifically analyzed include: (1) Main Feedwater Nozzles, (2) Auxiliary Feedwater Nozzles, (3) Steam Generator Tubes, (4) Tube to Shell Delta T's, and (5) Lower Tubesheets

The results show that the transient had no adverse structural effect on the steam generators.

NRC Form 366A (9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

Corrective Action: The failed circuit board will be replaced in the No. 1 MFP. As a precautionary measure, the No. 2 MFP speed control circuit will also be inspected for a similar failure.

The corrective action on the MSIV closure has not yet been determined since troubleshooting has not yet begun.

The corrective actions for the SFRCS spurious trip on low steam generator level have not yet been determined since troubleshooting has not yet begun. The proper method of manual actuation of the SFRCS buttons will be reviewed with all licensed operators. The switch layout is being modified to add additional demarkation of the actuation buttons, and to add actuation guards over the switches (see Attachment 3).

The corrective actions to be taken to prevent the AFP trip on overspeed have not yet been determined.

The torque switch bypass limit switch will be reset on the Auxiliary Feedwater Valves AF599 and AF608. Maintenance personnel will receive additional instruction, and the procedure for setting the motor operator valve limit switches will receive additional clarification. Other nuclear safety related motor operated valves at Davis-Besse will be evaluated.

The corrective actions to correct the control problems with the AFPs after the overspeed was reset have not been identified.

Corrective actions to be taken on the PORV have not yet been identified.

Corrective actions for the repair of the data transmission systems affecting the SPDS Control Room displays have not yet been identified.

The corrective actions for repair of the source range NIs have not yet been determined.

The pneumatic actuator for Turbine Bypass Valve 2-2 will be replaced. Additional corrective actions may be necessary after further investigation to determine the root cause of the failed valve actuator.

A tabulation of the causes and corrective actions determined to date is summarized in Attachment 4.

Corrective action details for the No. 1 AFP suction supply transfer from the Condensate Storage Tank to Service Water has not yet been identified.

Corrective action details for MS-106 have not yet been identified. Further investigation is in progress.

Failure Data: This is the first occurrence at Davis-Besse of a loss of both main and auxiliary feedwater.

NRC FORM 366A (9-83)

NRC Form 366A (9-83) U.S. NUCLEAR REGULATORY COMMISSION APPROVED OMB NO 3150-0104 EXPIRES 8/31/85

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

This is the first failure that has occurred at Davis-Besse on the MFP turbine electronic controllers which has caused an overspeed tripping of the pumps. A new electronic control system for main feedwater pumps was installed during the 1984 Refueling Outage.

Spurious closures of the MSIVs have occurred previously at Davis-Besse before time delays were added to the steam to feedwater pressure differential trip circuitry.

An SFRCS spurious half trip on low steam generator level has occurred on two previous trips since the 1984 Refueling Outage. Spurious trips on low steam generator level have not occurred prior to the 1984 Refueling Outage.

Incorrect manual initiation of the SFRCS has not previously occurred at Davis-Besse.

The AFPs tripping on overspeed after initiation has not previously occurred at Davis-Besse.

The Auxiliary Feedwater Valves AF599 and AF608 are normally open. One previous occurrence of one of these valves not opening with high differential pressure occurred after the March 2, 1984 reactor trip.

The operators do not normally attempt to control the AFP turbines locally. Problems with controlling these pumps do not appear to have been repetitive, however, some problems have been experienced previously with proper resetting of the trip throttle valve.

The PORV has not been challenged since the pressure setpoint was raised in 1979. Prior to 1979, several deficiencies were noted in the valve operation. In September 1977, the valve stuck in the open position, causing an overpressurization of the quench tank.

The diversity of the SPDS display sources (Ramtek and Chromatics display devices) has normally allowed at least one SPDS display to remain operable. The failure rate of these units is higher than is acceptable. Efforts are underway to increase the system reliability.

The failures of the source range NIs have been a repetitive occurrence at Davis-Besse even though exhaustive evaluations and corrective actions have been taken.

The damaged pneumatic operator on the turbine bypass valve has not previously occurred at Davis-Besse.

There have been several cases where the AFP suction inadvertently transferred from the Condensate Storage Tank to the Service Water supply.

Report No: NP-33-85-18   DVR No(s): 85-088

AGENDA FOR ACRS MEETING
JULY 11, 1985
4:30 P.M.
ROOM 1046, H STREET

RECENT SIGNIFICANT EVENTS

	Facility/Title	Event Date	Presenter/Phone	Page
1.	Hatch 1 Stuck Open SRV.	May 15, 1985	G. Rivenbark, NRR (492-7136)	2
2.	Oyster Creek Scram Discharge Volume Isolation Valves Failure	June 12, 1985	D. Powell, IE (492-8373)	7
3.	Rancho Seco - RCS High Point Vent Leak	June 23, 1985	H. Wong, IE (492-4536)	9
4.	Sequoyah Unit 2 Reactor Trip due to Improper Use of Test Instrument	May 22, 1985	E. Weiss, IE (492-9005)	11
5.	Davis-Besse loss of MFW and AFW.*	June 9, 1985	E. Jordan, IE (492-4848) A. DeAgazio, NRR (492-8945)	12

*The I.I.T. will discuss this event in detail at a later meeting.

HATCH UNIT 1 - STUCK OPEN SAFETY RELIEF VALVE OF MAY 15, 1985

(G. RIVENBARK)

- A SYSTEMS INTERACTION EVENT
  - UNIT 1 OPERATING AT FULL POWER
  - CONTROL ROOM EMERGENCY VENTILATION SYSTEM CHARCOAL FILTER DELUGE VALVE ACTUATED
  - WATER LEAKED THROUGH VENTILATION DUCTS INTO A HATCH UNIT 1 ANALOG TRANSMITTER TRIP SYSTEM (ATTS) INSTRUMENT PANEL CAUSING SRV TO OPEN
  - REACTOR MANUALLY SCRAMMED
  - FEEDWATER PUMP RECOVERS REACTOR WATER LEVEL
  - SRV CLOSED - WITHOUT OPERATOR ACTION
- CAUSE
  - LOSS OF INSTRUMENT WATER SUPPLY CAUSING DELUGE VALVE TO OPEN TOGETHER WITH PLUGGED DRAINS
  - NOT SURE HOW WATER CAUSED THE SRV TO OPEN
- ACTION
  - REPLACED ATTS POWER SUPPLY, CLEANED PLUGGED DRAINS AND INSPECTED DRAINS IN REDUNDANT FILTER UNIT
  - LICENSEE PROPOSES TO ADD CLEANOUT CHECK PROCEDURES FOR PLENUMS AND THEIR DRAINS
  - ORAB WILL DEVELOP TIA TO COORDINATE:
    - IE NOTICE
    - FURTHER INVESTIGATIVE EFFORTS
    - GENERIC REVIEW

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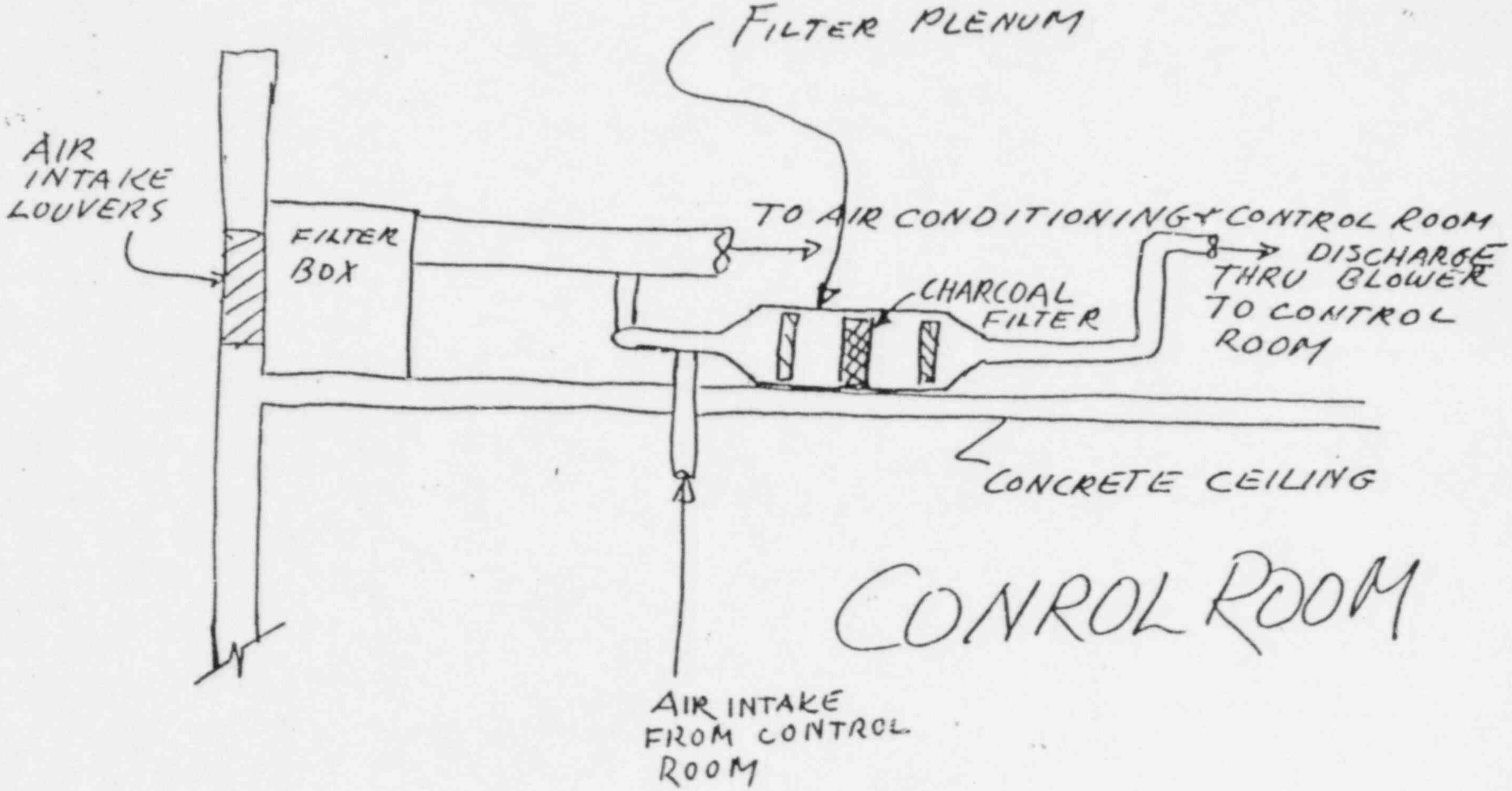
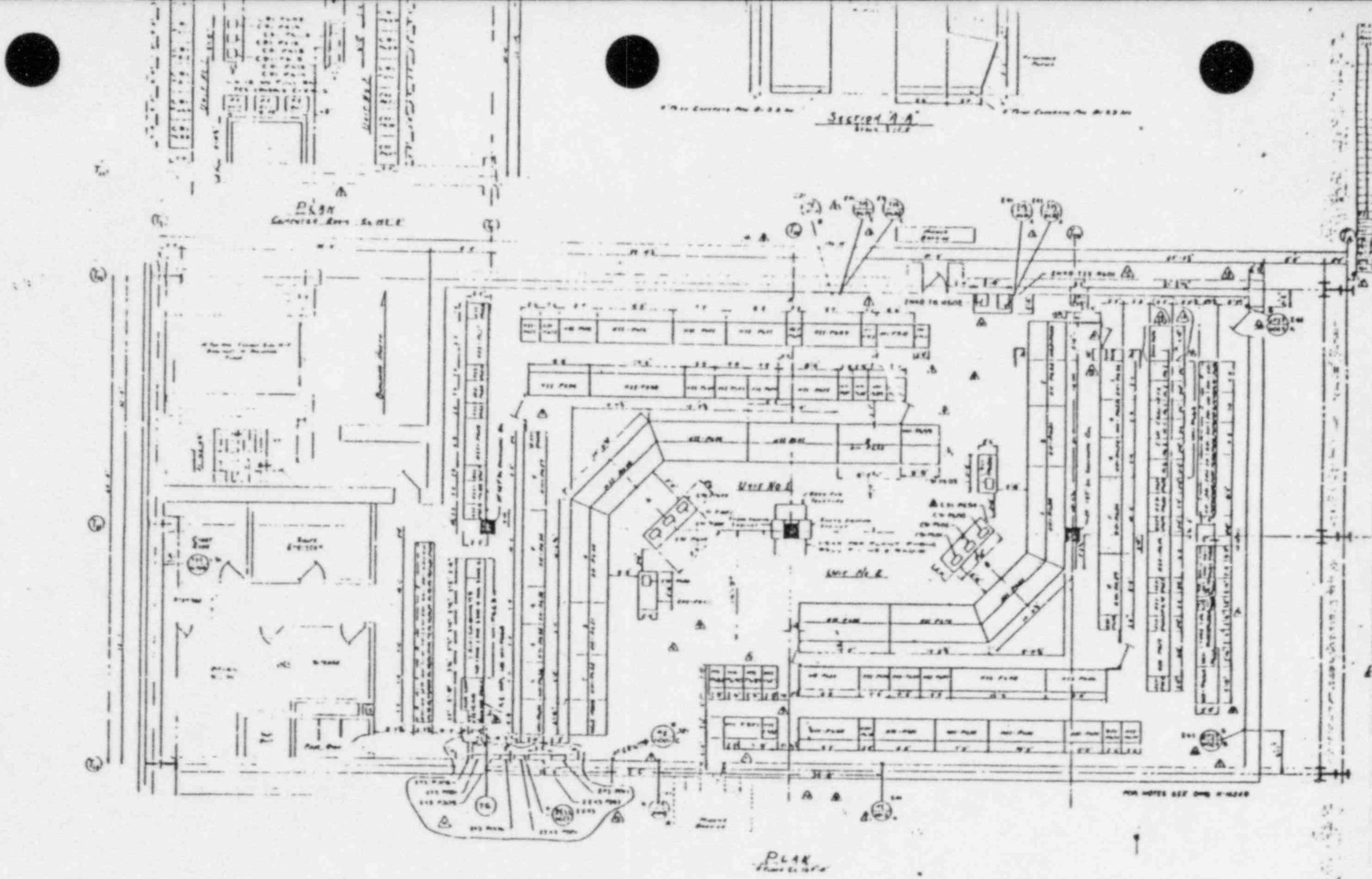

FILTER PLENUM
AIR INTAKE LOUVERS
FILTER BOX
TO AIR CONDITIONING + CONTROL ROOM
CHARCOAL FILTER
DISCHARGE THRU BLOWER TO CONTROL ROOM
CONCRETE CEILING
CONROL ROOM
AIR INTAKE FROM CONTROL ROOM

3



Georgia Power

EDWIN I. HATCH
NUCLEAR PLANT – UNIT

FIGURE 6.4-1

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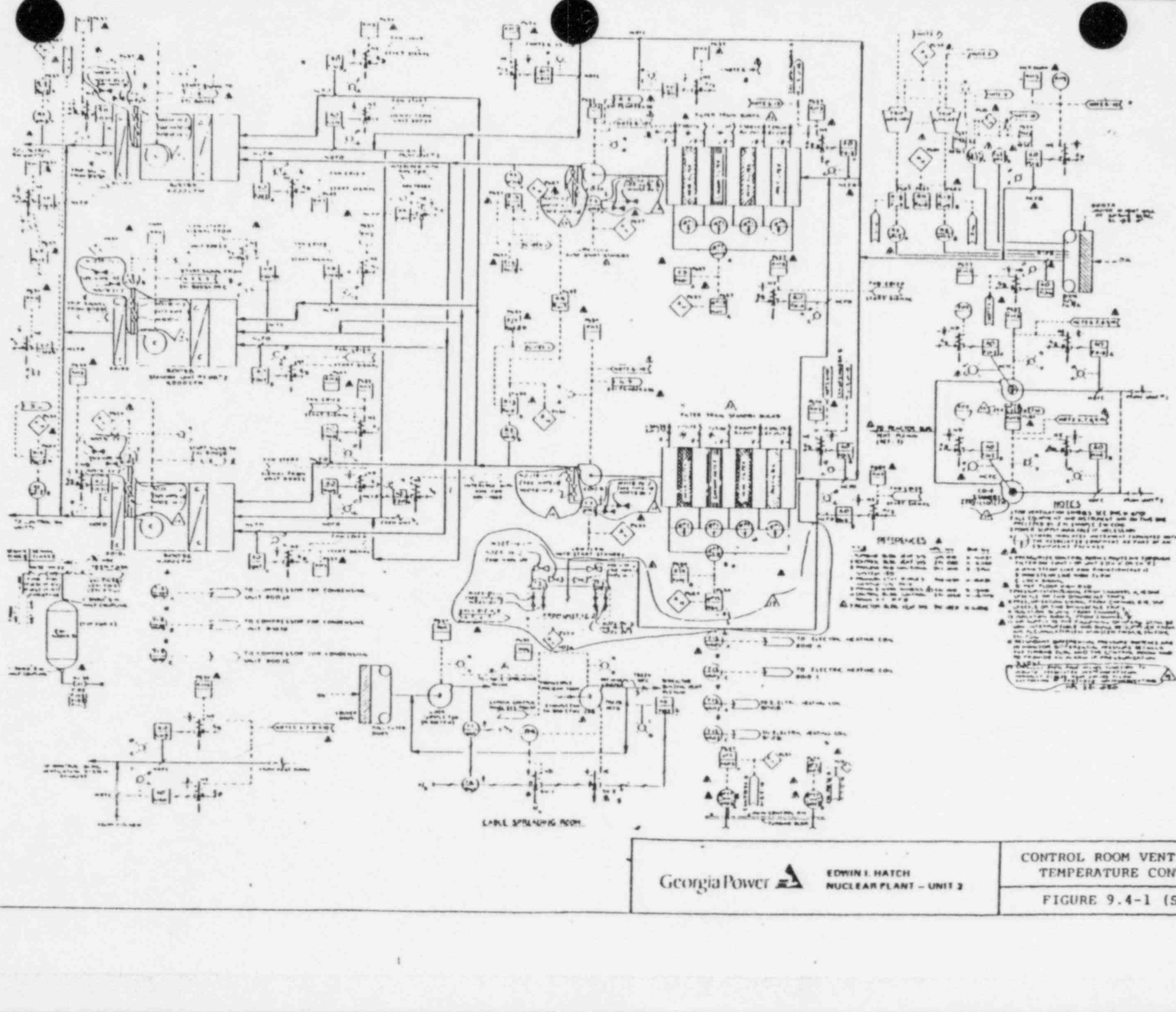

Georgia Power

EDWIN I. HATCH
NUCLEAR PLANT – UNIT 2

CONTROL ROOM VENTILATIO
TEMPERATURE CONTROL D

FIGURE 9.4-1 (SHEET]

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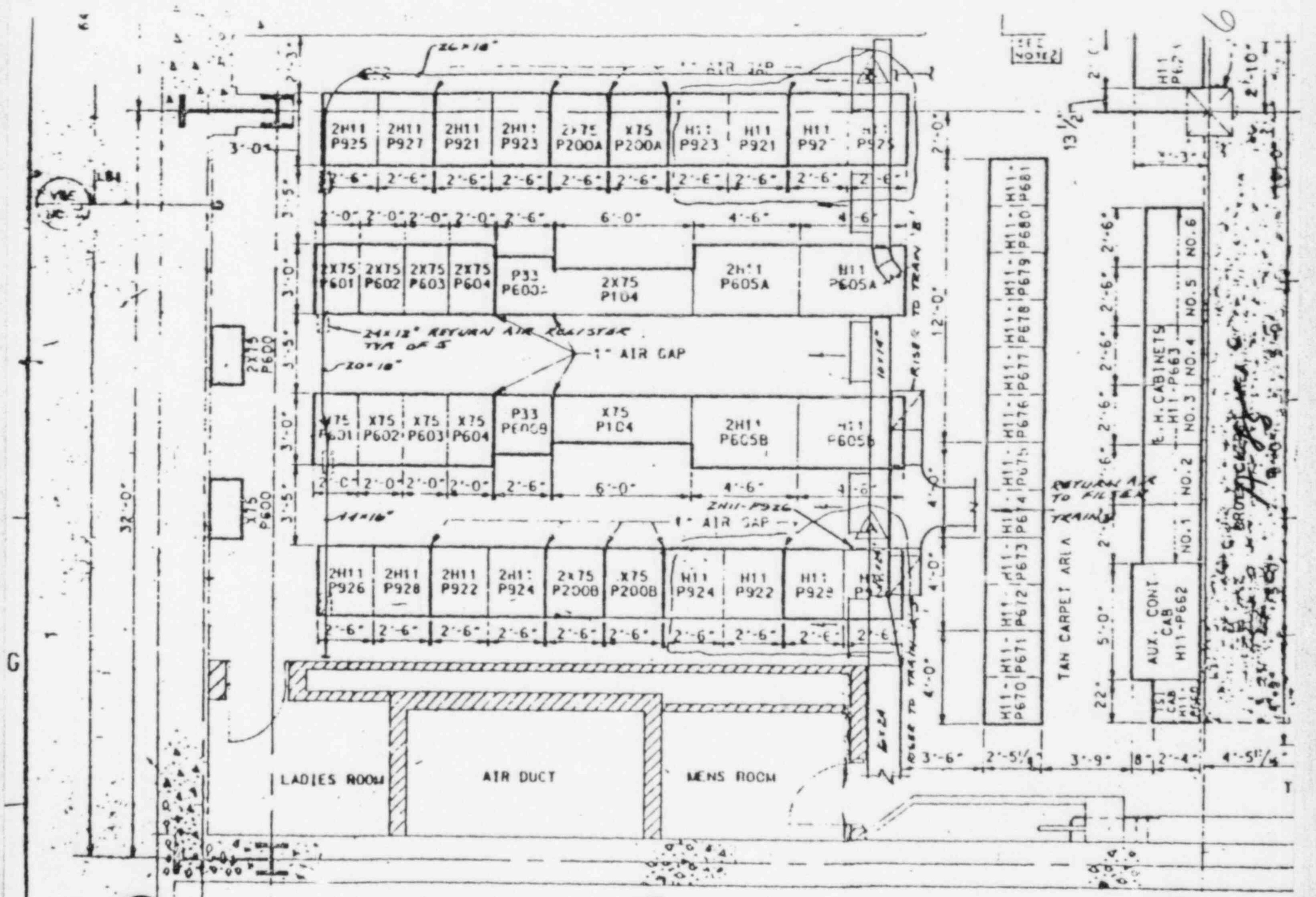
2H11 P925
2H11 P927
2H11 P921
2H11 P923
2X75 P200A
X75 P200A
H11 P923
H11 P921
1" AIR GAP
2X75 P601
2X75 P602
2X75 P603
2X75 P604
2X75 P104
2H11 P605A
H11 P605A
24x12" RETURN AIR REGISTER TYP OF 5
1" AIR GAP
RISER TO TRAIN 'B'
X75 P601
X75 P602
X75 P603
X75 P604
X75 P104
2H11 P605B
H11 P605B
2X75 P600
X75 P600
1" AIR GAP
2H11 P926
2H11 P928
2H11 P922
2H11 P924
2X75 P200B
X75 P200B
H11 P924
H11 P922
RISER TO TRAIN
RETURN AIR TO FILTER TRAINS
TAN CARPET AREA
R. H. CABINETS H11-P663
AUX. CONT CAB H11-P662
LADIES ROOM
AIR DUCT
MENS ROOM
PLAN VIEW
DUCTWORK / ATTS PANELS

OYSTER CREEK - UNCONTROLLED LEAKAGE OF REACTOR COOLANT OUTSIDE CONTAINMENT

JUNE 12, 1985 (D. POWELL, IE)

- WITH REACTOR AT 99% POWER, FAILURE OF THE ELECTRIC PRESSURE REGULATOR CAUSED A TURBINE BYPASS VALVE TO OPEN RESULTING IN A REACTOR PRESSURE DECREASE, FOLLOWED BY MSIV CLOSURE AND REACTOR SCRAM.
- SCRAM DISCHARGE VOLUME DRAIN VALVES FAILED TO FULLY SHUT/ SEAT CAUSING REACTOR COOLANT TO BE DISCHARGED TO THE REACTOR BUILDING DRAIN TANK.
- RELEASE OF STEAM FROM FLOOR DRAINS AND PAINT BLISTERING ON HOT PIPE CAUSES PORTION OF REACTOR BUILDING DELUGE SYSTEM TO ACTIVATE
- SCRAM SIGNAL NOT RESET FOR 36 MIN ALLOWING CONTINUOUS REACTOR COOLANT FLOW TO THE DRAIN TANK. CAUSE WAS 600 PSI INTERLOCK ON MSIV CLOSURE/LOSS OF CONDENSER VACUUM.
- SAFETY SIGNIFICANCE - (1) LOCA OUTSIDE CONTAINMENT, (2) POTENTIAL EQUIPMENT MALFUNCTION DUE TO FIRE DELUGE SYSTEM, (3) EXCESSIVE CRD SEAL TEMPERATURES.
- CAUSE-VALVE SPRING ON VALVE V15-134 (VELAN) VALVE UNDERSIZED
  - VALVE V15-121 (VALTAK) STROKE DISTANCE INSUFFICIENT TO TIGHTLY SEAT THE VALVE. (1/8" OPENING)
  - IMPROPER POST-INSTALLATION TESTING OF VALVES
- CORRECTIVE ACTIONS - REPLACED 400 LB SPRING WITH 1100LB SPRING, ADJUSTED VALVE STROKE DISTANCE CHECKED CRD SEALS FOR DAMAGE, CHECKED EQUIPMENT, NO DAMAGE FOUND.
- NRC FOLLOWUP ACTION - IE NOTICE IN PREPARATION.

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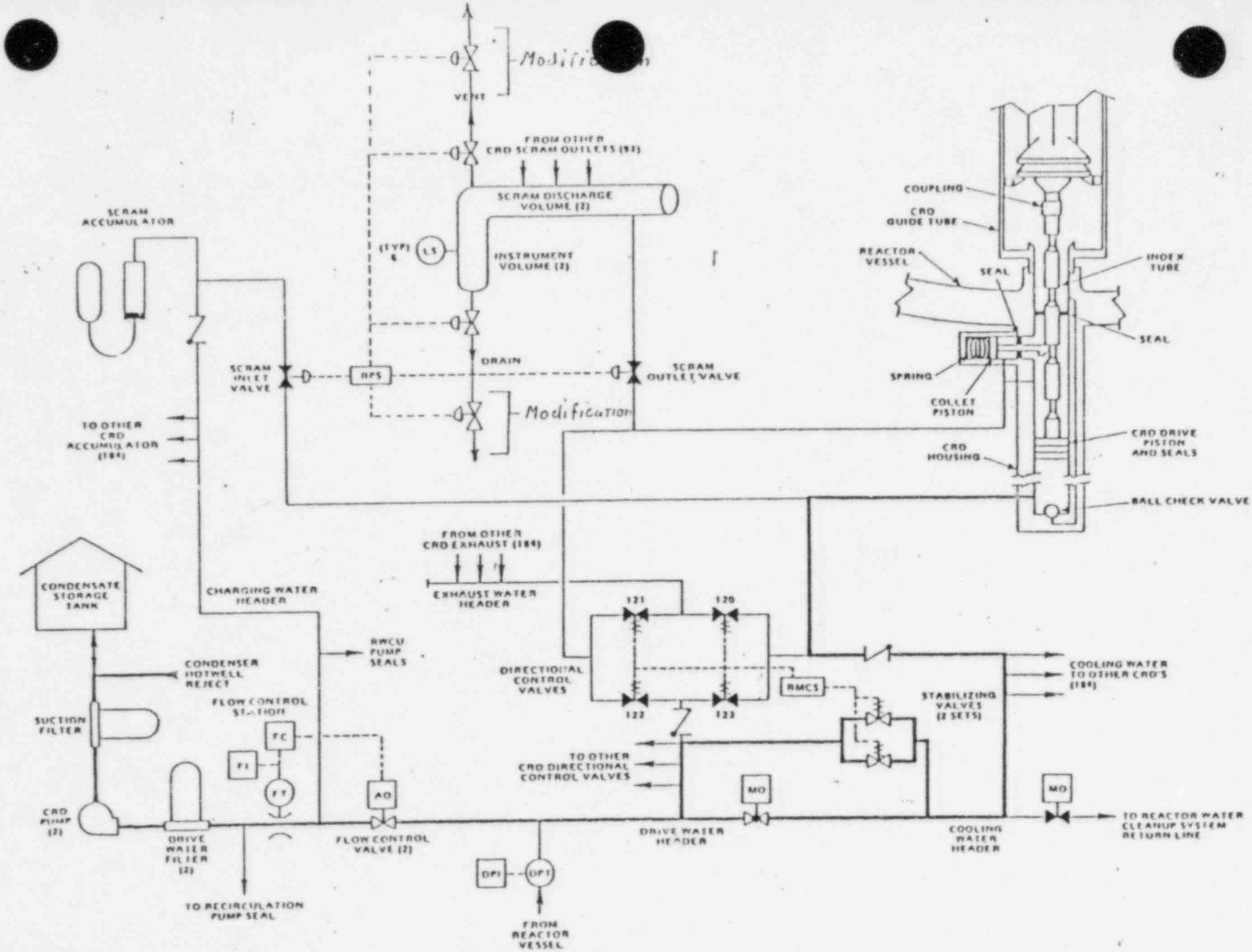

(Typical)

FIGURE 2.3-1 CONTROL ROD DRIVE HYDRAULIC SYSTEM

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RANCHO SECO - RCS HIGH POINT VENT LEAK

JUNE 23, 1985 (H. WONG, IE)

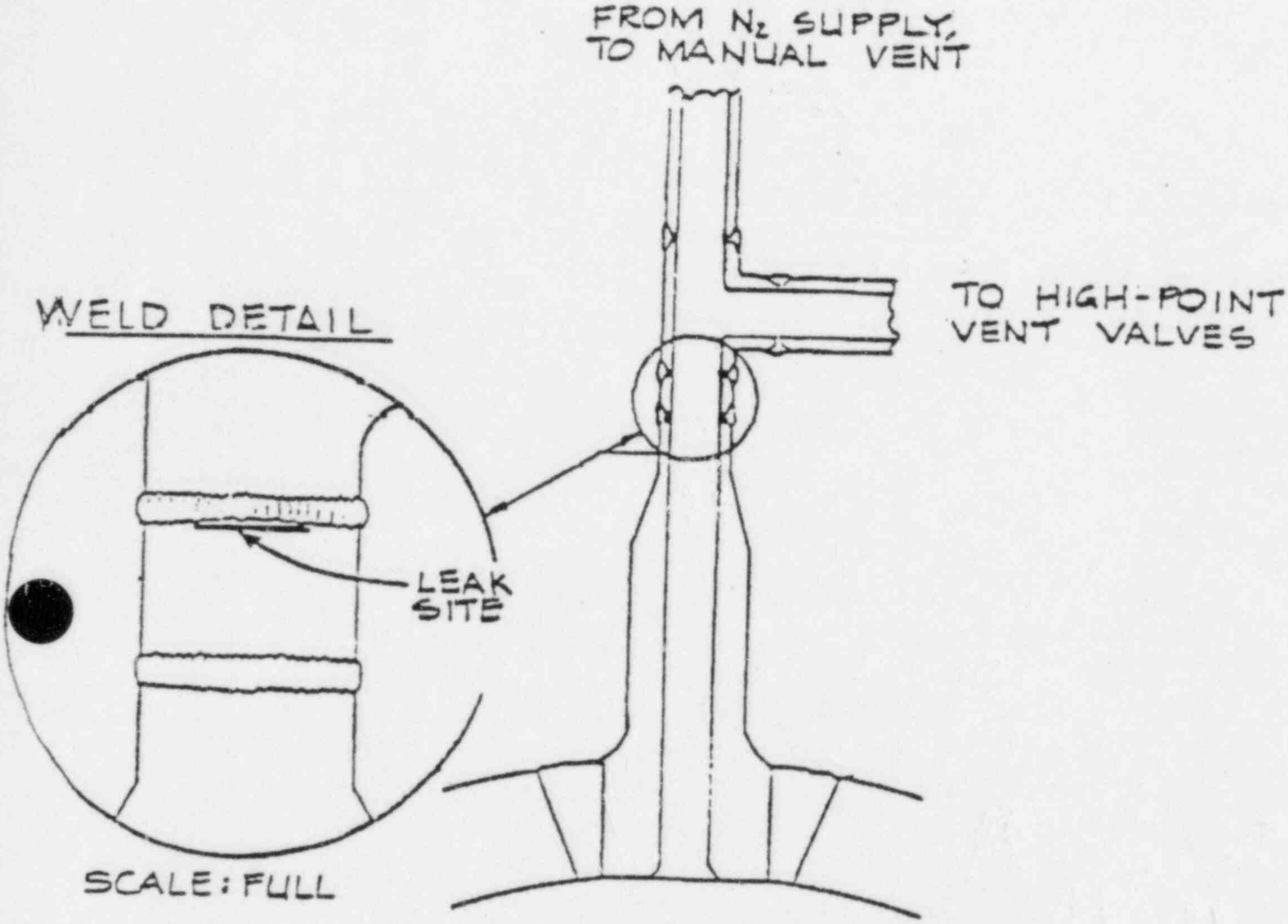
- PLANT IN HOT SHUTDOWN ~~STANDBY~~ RESTARTING FROM A REFUELING OUTAGE
- 20 GPM NON-ISOLATABLE PRIMARY COOLANT LEAK ON HIGH POINT VENT ON B STEAM GENERATOR HOT LEG
- TMI MODIFICATION INSTALLED 1983 REFUELING OUTAGE
- 120° THRU WALL LEAK AT WELD
- CAUSE APPEARS TO BE MISSING SUPPORTS AND FATIGUE FAILURE
- LICENSEE ACTIONS:

  STRESS ANALYSIS TO IDENTIFY OVERSTRESSED AREAS (BOTH HOT LEG VENTS)

  REPAIR SYSTEMS

  INSTALL SUPPORTS

  WALKDOWN TO INSPECT AND EVALUATE OTHER SYSTEMS
- REGION V, IE TEAM PARTICIPATING IN WALKDOWN.



RCS HIGH-POINT VENT
STEAM GENERATOR E-205B
SCALE: ¼" = 1"

SEQUOYAH 2 - IMPROPER USE OF TEST EQUIPMENT AND REACTOR TRIP

MAY 22, 1985 (E. WEISS, IE)

- REACTOR TRIPPED FROM 100% POWER ON OVERPOWER DELTA T
- DIGITAL VOLTMETER (DVM) USED TO MEASURE T_{HOT} AND T_{COLD}
- LEADS FROM DVM INCORRECTLY CONNECTED TO ONE RPS CABINET
- BEFORE TIME CONSTANT RECOVERS, LEADS FROM DVM INCORRECTLY CONNECTED TO 2ND RPS CABINET

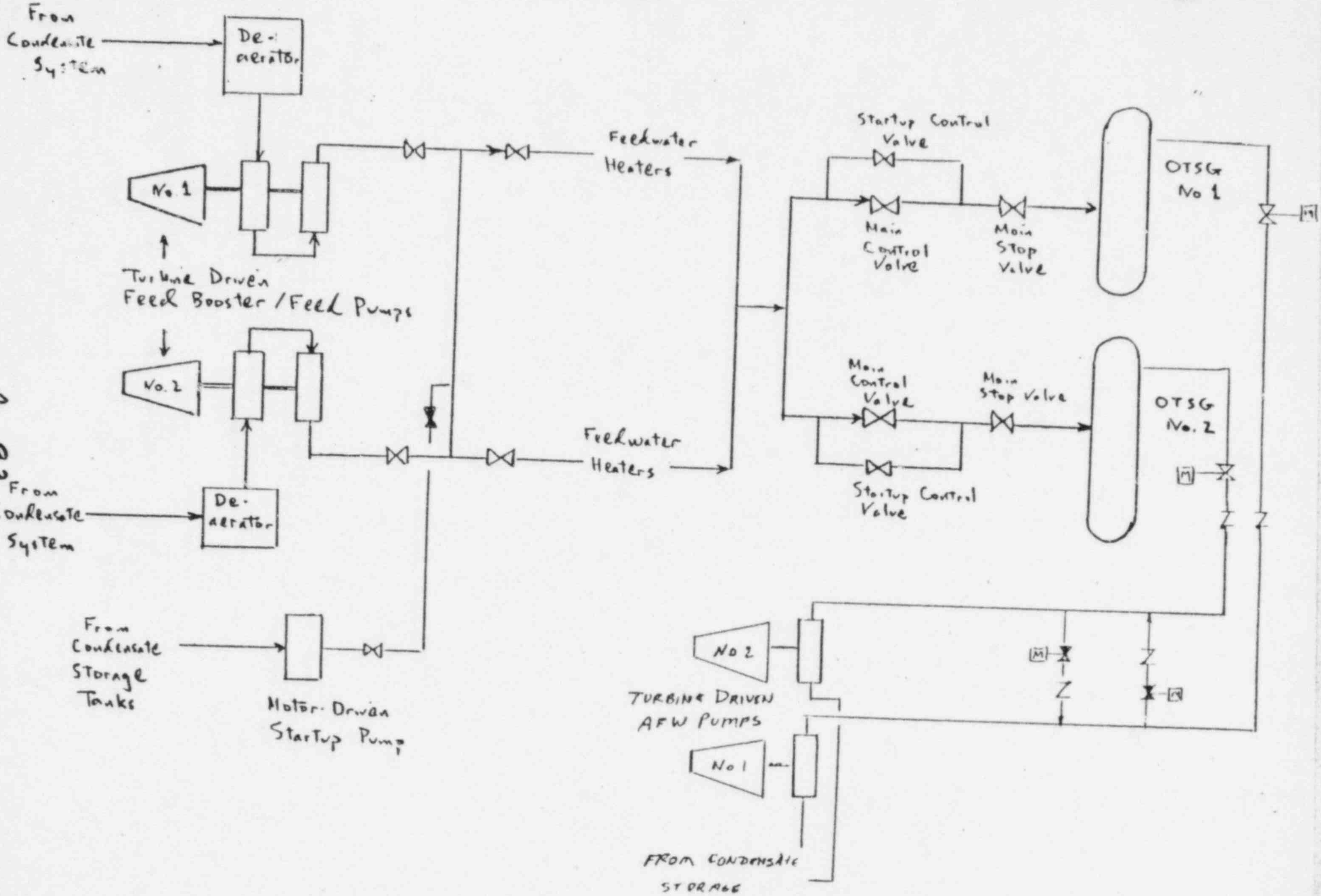
DAVIS-BESSE - LOSS OF ALL MAIN FEEDWATER AND AUXILIARY FEEDWATER

JUNE 9, 1985 (A. DEAGAZIO, NRR)

- Loss of one main feedwater pump at 90% power.
- Reactor trip at 78% power on high pressure.
- Both MSIVs close spuriously tripping remaining main feedwater pump.
- Steam generator low level starts both auxiliary feedwater pumps but both trip on overspeed.
- Operator erroneously manually trips SFRCS on low pressure
- No feedwater available for about eight minutes and steam generator levels fell to about eight inches.
- PORV cycles three times - did not reseat on third cycle, operators close block valve.
- Startup feedwater pump used to feed one steam generator.
- Operators restart auxiliary feedwater pumps and restore normal post-trip conditions.
- No indication that subcooling margin was lost or that reactor coolant activity was abnormal.
- Plant now in cold shutdown.

KNOWN EQUIPMENT FAILURES OR MALFUNCTIONS

- Main feedwater trip
- Spurious half trip of SFRCS
- MSIV closures (2)
- AFW pump trips on overspeed (2)
- AFW isolation valves fail to open (2)
- PORV failure to reseat
- SUFP valve failure to open
- AFW speed governor after reset
- Switchover to service water backup supply
- Damaged turbine bypass valve


From Condensate System
De-aerator
No. 1
Turbine Driven Feed Booster / Feed Pumps
No. 2
From Condensate System
De-aerator
Feedwater Heaters
Feedwater Heaters
Startup Control Valve
Main Control Valve
Main Stop Valve
OTSG No 1
Main Control Valve
Main Stop Valve
Startup Control Valve
OTSG No. 2
M
M
M
M
From Condensate Storage Tanks
Motor-Driven Startup Pump
No 2
TURBINE DRIVEN AFW PUMPS
No 1
FROM CONDENSATE STORAGE

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SSINS No.: 6835
IN 85-50

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

July 8, 1985

IE INFORMATION NOTICE NO. 85-50: COMPLETE LOSS OF MAIN AND AUXILIARY FEEDWATER AT A PWR DESIGNED BY BABCOCK & WILCOX

Addressees:

All nuclear power facilities holding an operating license (OL) or construction permit (CP).

Purpose:

This information notice is being provided to inform licensees of a significant reactor operating event involving the loss of main and auxiliary feedwater at a pressurized water reactor. Information in this notice is preliminary and was obtained from the special NRC fact finding team which is investigating the event. A complete report of findings will form the basis for further communications or actions related to this event. The NRC expects that recipients will review this notice for applicability to their facilities. Suggestions contained in this notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

On June 9, 1985, the Davis-Besse plant was operating at 90% power with Main Feedwater Pump 2 in manual control because problems in automatic had been experienced. A control problem with Main Feedwater Pump 1 occurred, and it tripped on overspeed. Reactor runback at 50% per minute toward 55% power was automatically initiated. Nevertheless, 30 seconds later, the reactor tripped at 80% power on high pressure in the reactor coolant system.

One second after reactor/turbine trip, one channel of the Steam and Feedwater Rupture Control System (SFRCS) was automatically initiated due to a spurious signal indicating low water level in Steam Generator 2. Both Main Steam Isolation Valves (MSIVs) closed. Three seconds after the actuation, the SFRCS automatically reset. Closing of the MSIVs isolated the turbine of the operating main feedwater pump from its source of steam. The pump continued to supply feedwater to the steam generators for a few minutes as it coasted down.

Four and a half minutes after reactor trip, water level in the steam generators began to fall from the normal post-trip level which is 35 inches. After MSIV closure, steam release to atmosphere continued to remove decay heat. One minute later, Channel 1 of SFRCS actuated when the water level in Steam Generator 1 actually reached the SFRCS setpoint at 27 inches (See Figure 1). SFRCS started Auxiliary Feedwater Pump 1 and initiated alignment of it to Steam Generator 1.

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Within seconds after automatic initiation of Channel 1 of SFRCS, the operator actuated both channels of SFRCS; however, he inadvertently actuated both SFRCS channels on low steam pressure instead of low water level. When an SFRCS channel is actuated on low steam pressure, a rupture of the steam line associated with that channel is presumed to have occurred. The SFRCS closes the steam generator isolation valves, including a valve in the auxiliary feedwater line, and aligns the auxiliary feedwater pump to the other steam generator. Because both channels had been manually actuated on low steam pressure, both steam generators were isolated from both auxiliary feedwater pumps. Five seconds after the operator's inadvertent actuation of both channels on low steam pressure, SFRCS Channel 2 received an actual low water level actuation signal. Because low pressure initiation takes precedence, alignment of the auxiliary feedwater pumps remained unchanged. At six minutes into the event as both auxiliary feedwater pumps were accelerating, they tripped on overspeed.

In summary, all main feedwater had been lost, both steam generators were isolated from feedwater and were boiling dry, all auxiliary feedwater pumps were tripped, pressure of the reactor coolant system was rising, and reactor coolant system temperature was increasing.

Within one minute after the operator's inadvertent actuation of the SFRCS on low steam pressure, the mistake had been recognized and the SFRCS had been reset. If equipment had performed in accordance with system design requirements, the operator's error might not have had a significant impact on the event. The auxiliary feedwater isolation valves should have reopened automatically, but the valves did not reopen. The operator then tried to reopen the valves from the main control panel, but the valves would not reopen. Operators were dispatched to locally start the auxiliary feedwater pumps, open the auxiliary feedwater isolation valves, start the nonsafety-related motor-driven startup feedwater pump, and valve it to the system.

Pressure and temperature in the reactor coolant system continued to rise because there was not sufficient water in the steam generators to provide an adequate heat sink. At 13 minutes after reactor trip, reactor coolant system pressure reached 2425 psig, and the Pilot Operated Relief Valve (PORV) opened three times to limit the pressure rise. On the third lift, the valve remained open. The operator closed the PORV block valve and reopened it two minutes later after the PORV had closed.

Approximately 16 to 18 minutes after reactor trip, the operators had the startup and auxiliary feedwater pumps running and the valves aligned. Water levels were beginning to rise in the steam generators. Reactor coolant temperature reached a maximum of 594° F and then started to decrease to normal. Refilling of the steam generators caused the reactor coolant system to fall to 1716 psig and about 540°F before returning to normal (See Figure 2).

At 30 minutes after reactor trip, plant conditions were essentially stable.

Discussion:

For several minutes after reactor trip, the steam generators were unable to cool the reactor coolant system adequately.

The first problem contributing to this event was the loss of all main feedwater due to closure of the MSIVs. The licensee's hypothesis, based on information from Babcock & Wilcox, is that turbine trip caused a pressure transient upstream from the turbine stop valves which caused the outputs of the redundant steam generator level instrumentation channels to oscillate widely for several seconds. The licensee believes that this caused a spurious low level actuation of SFRCS which closed the MSIVs.

Three additional problems contributed to this event by affecting the availability of both trains of the auxiliary feedwater system. The first occurred when the reactor operator pressed the wrong SFRCS buttons. The second occurred when both auxiliary feedwater pumps tripped on overspeed. The third occurred when both auxiliary feedwater isolation valves did not reopen when SFRCS was reset.

Control buttons for the SFRCS are arranged in two vertical columns. Each column of buttons controls one SFRCS channel. The operator should have pressed the fourth button from the top in each column. Instead, the operator pressed the top buttons causing isolation of both steam generators.

Both auxiliary feedwater pumps are driven by Terry turbines which tripped on overspeed early in the event. When this occurred, steam was being supplied to the turbines via crossover lines, which are longer than the normal supply lines and include long horizontal runs. The licensee believes that significant condensation may have occurred in the crossover lines. Further, the licensee believes that the quality of steam arriving at the turbines may have been affected significantly by the configuration of the crossover lines and may have caused the overspeed trips.

The auxiliary feedwater system isolation valves have Limitorque motor operators. The motor operators have torque switches which prevent overtorquing of the valves by disconnecting power to the motors. When the valves are being opened, additional torque is required to overcome friction while the gates are being unseated and while a significant pressure differential may exist across the gates. During the initial part of the opening stroke, the torque switch in the motor operator is bypassed by a bypass switch so that full motor torque is developed if necessary. The licensee believes that these bypass switches went off bypass too early. The valves did not reopen until an operator unseated them by hand.

No specific action or written response is required by this information notice. If you have any questions about this matter, please contact the Regional Administrator of the appropriate NRC regional office or this office.

E L Jordan
Edward L. Jordan, Director
Division of Emergency Preparedness
and Engineering Response
Office of Inspection and Enforcement

Technical Contact: R. W. Woodruff, IE
(301) 492-4507

Attachments:
1. Figure 1 - Steam Generator 1 Level and Pressure
2. Figure 2 - RCS Temperature and Pressure
3. List of Recently Issued IE Information Notices

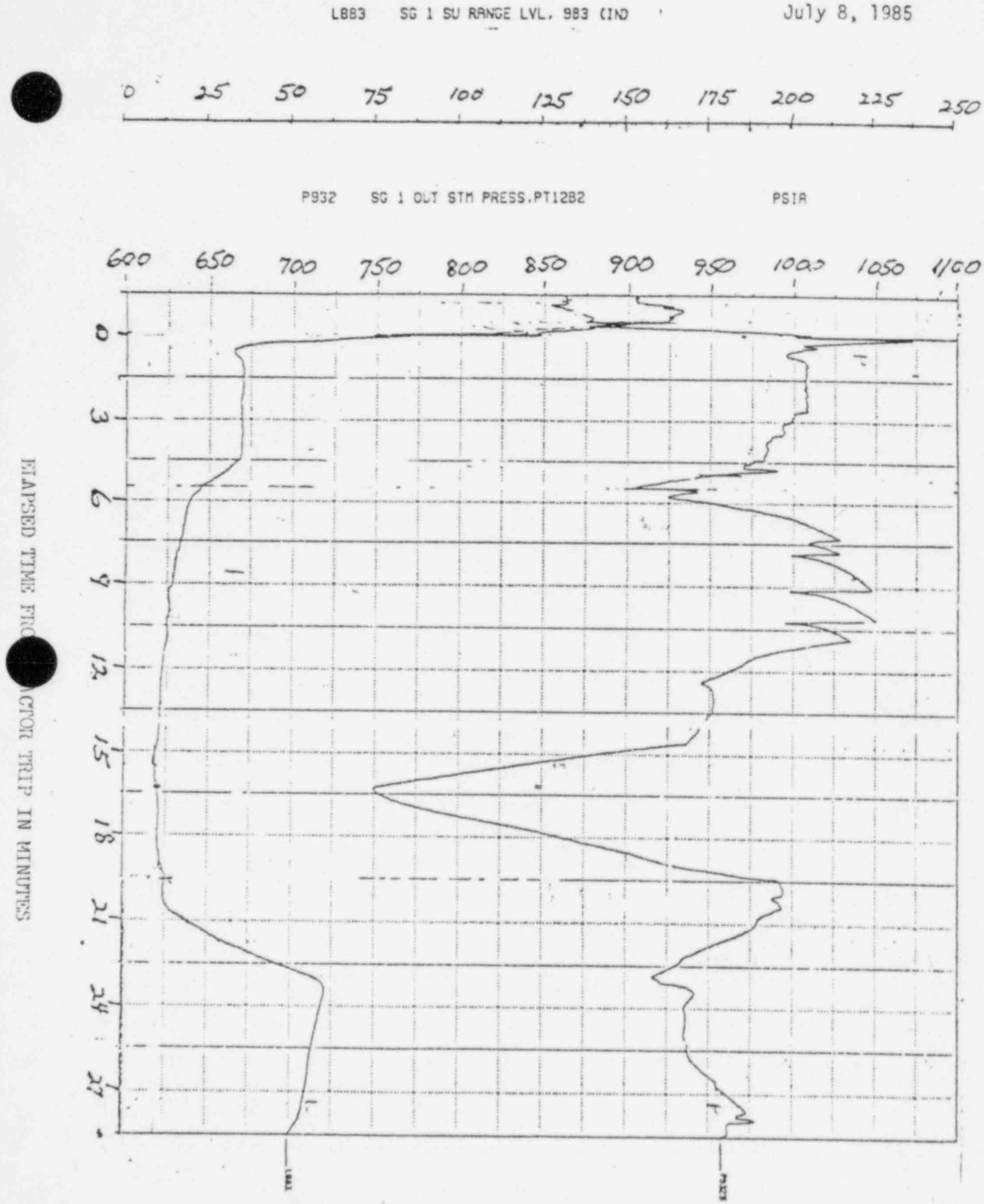

FIGURE 1: STEAM GENERATOR 1 LEVEL AND PRESSURE

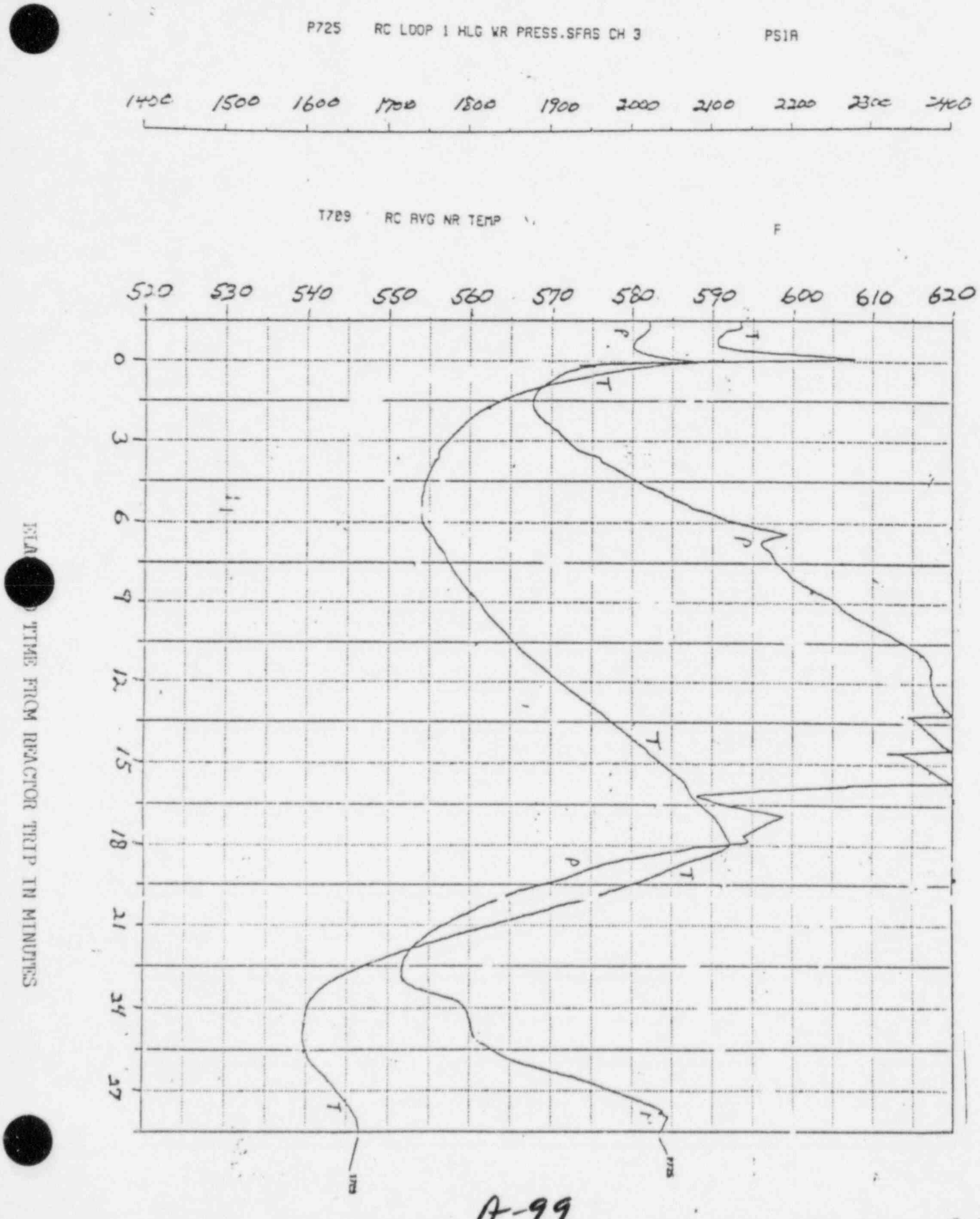

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FIGURE 2: RCS TEMPERATURE AND PRESSURE

LIST OF RECENTLY ISSUED
IE INFORMATION NOTICES

Information Notice No.	Subject	Date of Issue	Issued to
85-49	Relay Calibration Problem	7/1/85	All power reactor facilities holding an OL or CP
85-48	Respirator Users Notice: Defective Self-Contained Breathing Apparatus Air Cylinders	6/19/85	All power reactor facilities holding an OL or CP, research, and test reactor, fuel cycle and Priority 1 material licensees
85-47	Potential Effect Of Line-Induced Vibration On Certain Target Rock Solenoid-Operated Valves	6/18/85	All power reactor facilities holding an OL or CP
85-46	Clarification Of Several Aspects Of Removable Radioactive Surface Contamination Limits For Transport Packages	6/10/85	All power reactor facilities holding an OL
85-45	Potential Seismic Interaction Involving The Movable In-Core Flux Mapping System Used In Westinghouse Designed Plants	6/6/85	All power reactor facilities holding an OL or CP
85-44	Emergency Communication System Monthly Test	5/30/85	All power reactor facilities holding an OL
85-43	Radiography Events At Power Reactors	5/30/85	All power reactor facilities holding an OL or CP
85-42	Loose Phosphor In Panasonic 800 Series Badge Thermoluminescent Dosimeter (TLD) Elements	5/29/85	All power reactor facilities holding an OL or CP

OL = Operating License
CP = Construction Permit

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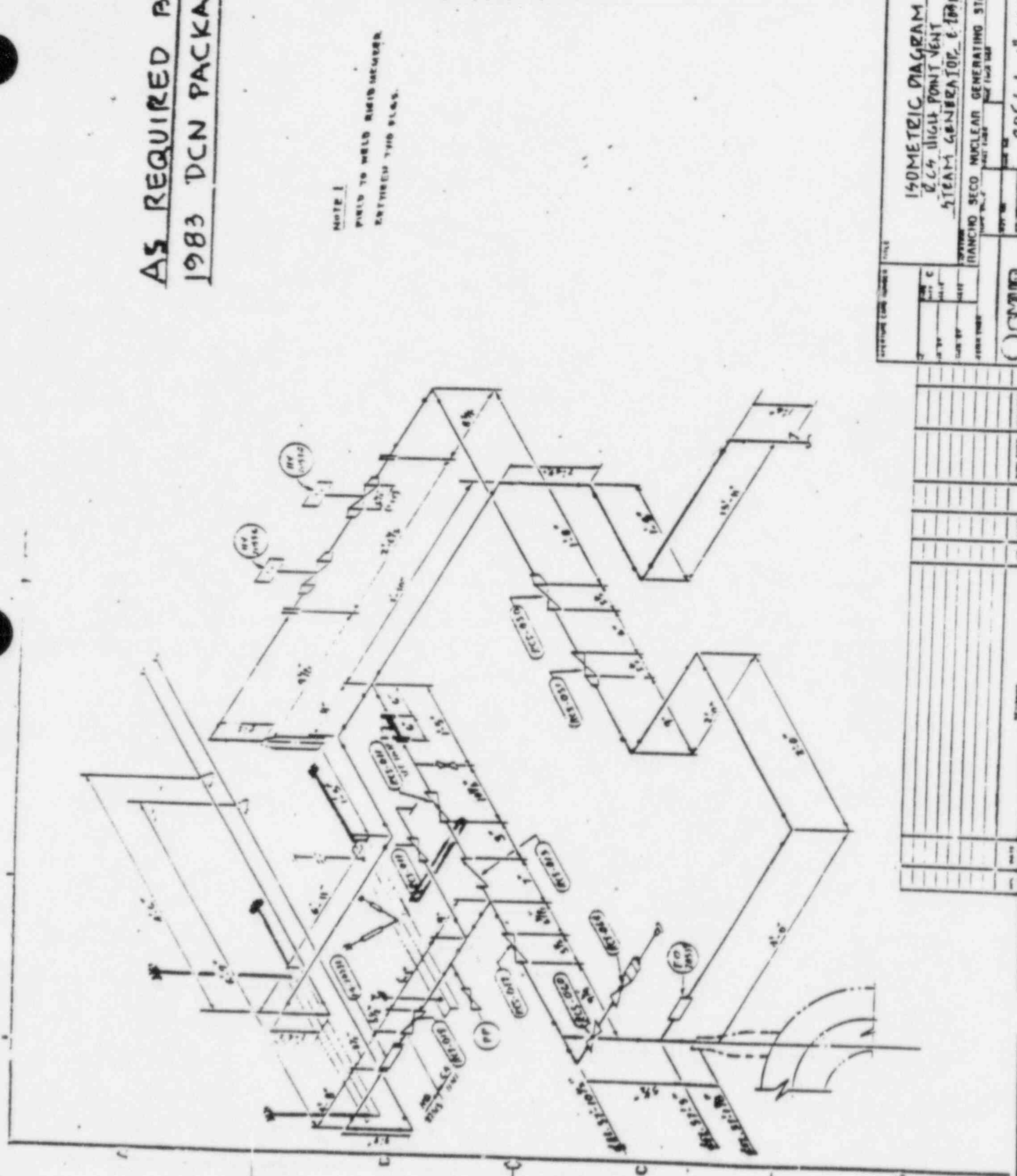
UP

AS REQUIRED BY
1983 DCN PACKAGE

APPENDIX IX
RANCHO SECO ISOMETRIC DIAGRAM RCS
HIGH POINT VENT



GESSAR II CONTAINMENT CAPABILITY

A PRESENTATION TO THE ADVISORY

COMMITTEE ON REACTOR SAFEGUARDS

WASHINGTON, D.C.

GENERAL ELECTRIC COMPANY

JULY 12, 1985

MK III CONTAINMENT DESIGN

COMPARISON

PLANT	CONT. TYPE	DES. PRESS (PSIG)	ULT. PRESS CAPAB. (PSIG)	DOME CONFIG.	CONT. DIA (FT)	CONC. FILL @ BASE
GESSAR II (238)	STEEL + CONCRETE SHIELD BLDG.	15	85	TORI-SPHERICAL	120	YES
GRAND GULF (251)	LINED REINF. CONCRETE	15	67	HEMI-SPHERICAL	123	N/A
PERRY (238)	STEEL + CONCRETE SHIELD BLDG.	15	94	TORI-SPHERICAL	120	YES
RIVER BEND (218)	STEEL + CONCRETE SHIELD BLDG.	15	90	TORI-SPHERICAL	120	YES
CLINTON (218)	LINED REINF. CONCRETE	15	95	HEMI-SPHERICAL	124	N/A

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POTENTIAL FOR UNSTABLE PROPAGATION OF AN UNDETECTED FLAW

- MAXIMUM STRESS IN THE CONTAINMENT FOR ULTIMATE PRESSURE CAPABILITY (85 PSIG) LOADINGS CONSIDERED.

- LOWER BOUND FRACTURE TOUGHNESS PROPERTIES CONSIDERED.

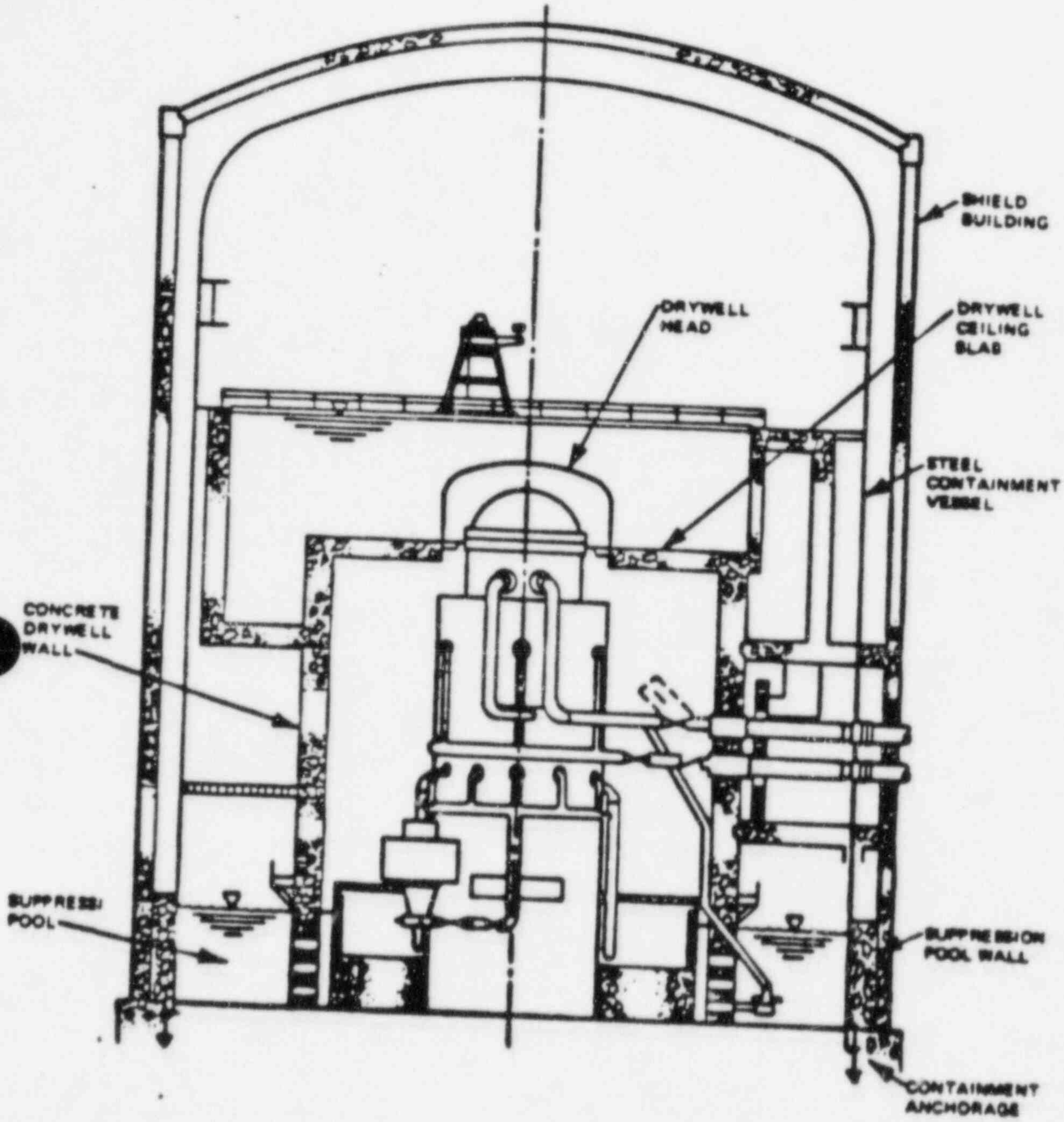
  - PLATE MATERIAL
  - WELDMENT
  - HEAT AFFECTED ZONE

- BASED ON A CONSERVATIVE FRACTURE ANALYSIS, A POTENTIAL CRACK OF UP TO 1/2 INCH DEEP (>25% OF WALL THICKNESS) AND 3 INCHES LONG CAN BE TOLERATED WITHOUT PROPAGATION TO FAILURE.

- WELDING PROCEDURES LIMIT FLAWS TO ~10% OF WALL THICKNESS.

- UNLIKELY THAT ANY WELD DEFECT SIGNIFICANTLY OVER 2% OF WALL THICKNESS AT WELD JOINTS WILL ESCAPE FULL RT OF WELDS.

UNDETECTED FLAWS WILL NOT AFFECT CALCULATED ULTIMATE PRESSURE CAPABILITY OF THE CONTAINMENT



Mark III Containment Buildings of Standard Plant

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DOMINANT CONTAINMENT FAILURE MODES

o LOADING AND FAILURE MODE

Loading	Failure Mode
1. H2 DETONATION IN CONTAINMENT o SHOCK WAVE o INTERNAL PRESSURE ON CONTAINMENT o EXTERNAL PRESSURE ON DRYWELL	1. LOCAL: CONTAINMENT FAILURE ABOVE WATER LINE GLOBAL: A) CONTAINMENT FAILURE ABOVE WATER LINE B) DRYWELL CEILING FAILURE
2. H2 COMBUSTION IN CONTAINMENT o INTERNAL PRESSURE ON CONTAINMENT o SMALL EXTERNAL PRESSURE ON DRYWELL ($\sim$5 PSIG)	2. A) CONTAINMENT FAILURE ABOVE WATER LINE B) NO DRYWELL FAILURE SINCE NO SIGNIFICANT LOADS
3. H2 SLOW BURNING o INTERNAL PRESSURE ON CONTAINMENT o SMALL EXTERNAL PRESSURE ON DRYWELL ($\sim$5 PSIG)	3. A) NO CONTAINMENT FAILURE SINCE PRESSURE IS LOW B) NO DRYWELL PRESSURE SINCE NO SIGNIFICANT LOADS
4. STEAM AND/OR NON-COMBUSTIBLE GAS OVERPRESSURIZATION o SMALL INTERNAL PRESSURE ON DRYWELL $\sim$5 PSIG o INTERNAL PRESSURE ON CONTAINMENT	4. A) CONTAINMENT FAILURE ABOVE WATER LINE AT CONTAINMENT ULTIMATE PRESSURE CAPABILITY B) NO DRYWELL FAILURE SINCE NO SIGNIFICANT LOADS

- VALIDATION OF FAILURE MODES
  - BY ANALYSIS
    - LOAD TYPES AND APPLICATION
    - CONTAINMENT AND DRYWELL STRUCTURAL CONFIGURATION
    - CALCULATED STRESSES
    - HIGHEST STRESSED POINTS ASSUMED TO FAIL FIRST
    - NO FAILURES ASSUMED WHERE LOADS ARE SIGNIFICANTLY LESS THAN THE DESIGN LOADS.

- DRYWELL STRUCTURE, HEAD, PERSONNEL LOCK NOT CHALLENGED.
- SUPPRESSION POOL BYPASS DUE TO DRYWELL BOUNDARY FAILURE SHOULD NOT BE A CONCERN.

FAILURES WITHIN DRYWELL

- FAILURES IN DRYWELL IGNORED ON THE BASIS OF LOW PROBABILITY
  - REACTOR PRESSURE VESSEL ONLY.
- RPV INSPECTION
  - AS PART OF FABRICATION
    - 100% RADIOGRAPHY
    - 100% UT OF ACCESSIBLE PORTIONS (ALL WELDS ACCESSIBLE IN BWR/6)
  - ISI
    - 25% UT WITHIN 10 YEARS
    - 100% UT IN PLANT LIFE
- CRD HOUSINGS
  - AS PART OF FABRICATION
    - 100% PENETRANT TESTING OF CRD HOUSING TO VESSEL WELDS
    - 100% UT OF CRD HOUSING TO VESSEL WELDS
  - ISI
    - EXCLUDED
    - WELD IN COMPRESSION
    - FAILURE OF ONE HOUSING PENETRATION POSTULATED

DRYWELL HEAD AND CONNECTION
DRYWELL PERSONNEL LOCK

- o DESIGNED TO ASME, SECTION III, NA-3352.
- o MATERIAL AS ALLOWED BY ASME, SECTION III, NE 2000.
- o FABRICATION AND INSTALLATION TO ASME, SECTION III, NE 4000.
- o TEST PRESSURE
  - o 30 PSIG FOR HIGH PRESSURE TEST.
  - o 3 PSIG FOR LOW PRESSURE TEST.
- o INSPECTION AND TESTING
  - o INSPECTION TO ASME SECTION III, NE 5000.
  - o REPEAT SHOP TESTS AT 30 PSIG PRESSURE FOR LEAKS.
  - o FIELD TESTS
    - o PREOPERATIONAL STRLCTURAL PROOF TEST AT 30 PSIG.
    - o INTEGRATED HIGH PRESSURE LEAK RATE TEST.
    - o INTEGRATED LOW PRESSURE LEAK RATE TEST.
    - o PERIODIC LEAK RATE TEST AT LOW PRESSURE.
    - o AFTER EACH CLOSING OF DRYWELL HEAD AND PERSONNEL LOCK, CONNECTION TESTED FOR LEAKS AT 30 PSIG.

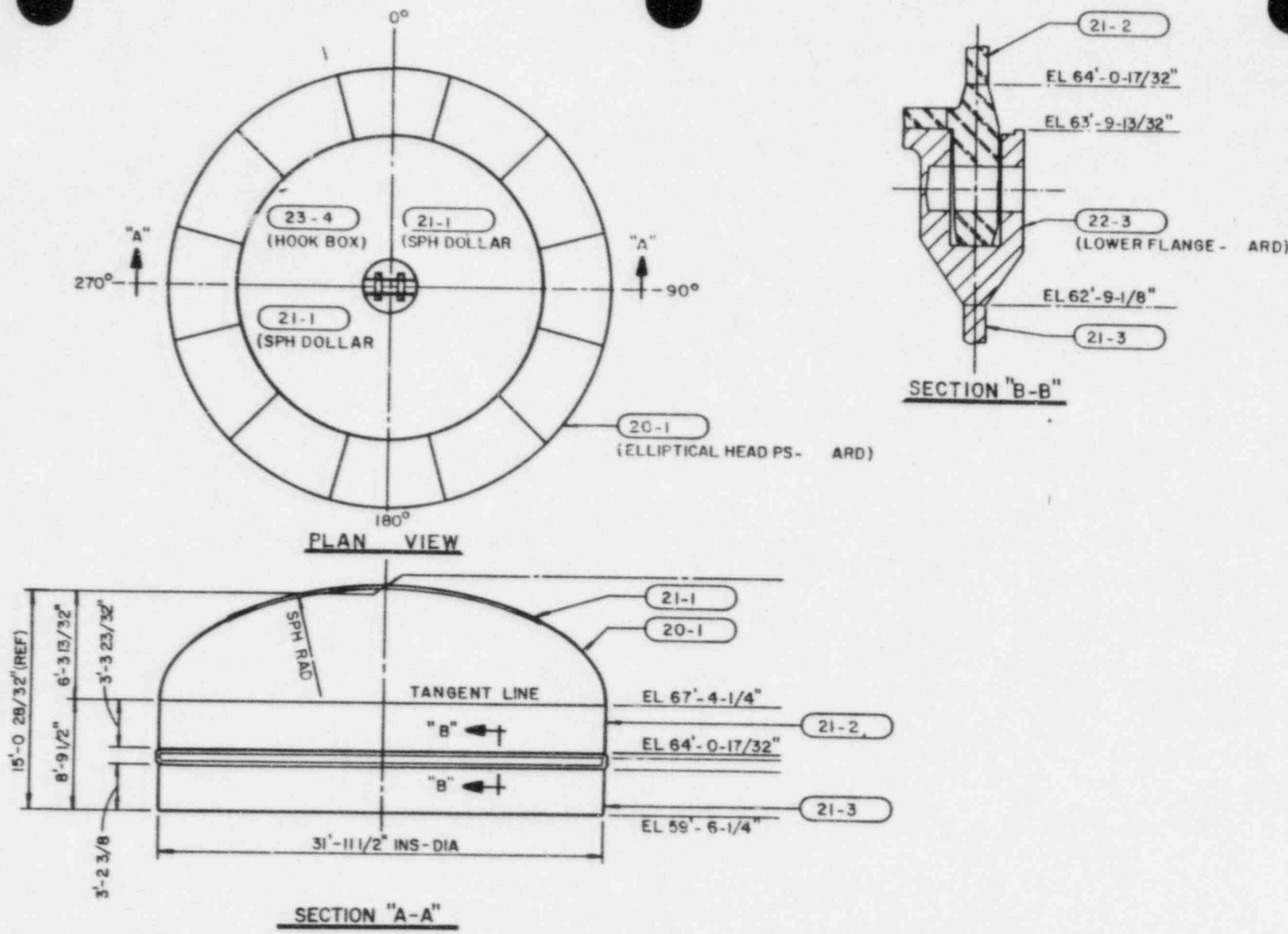


Figure 3.8-11. Drywell Head

RSVi5 - 8
JULY 1985

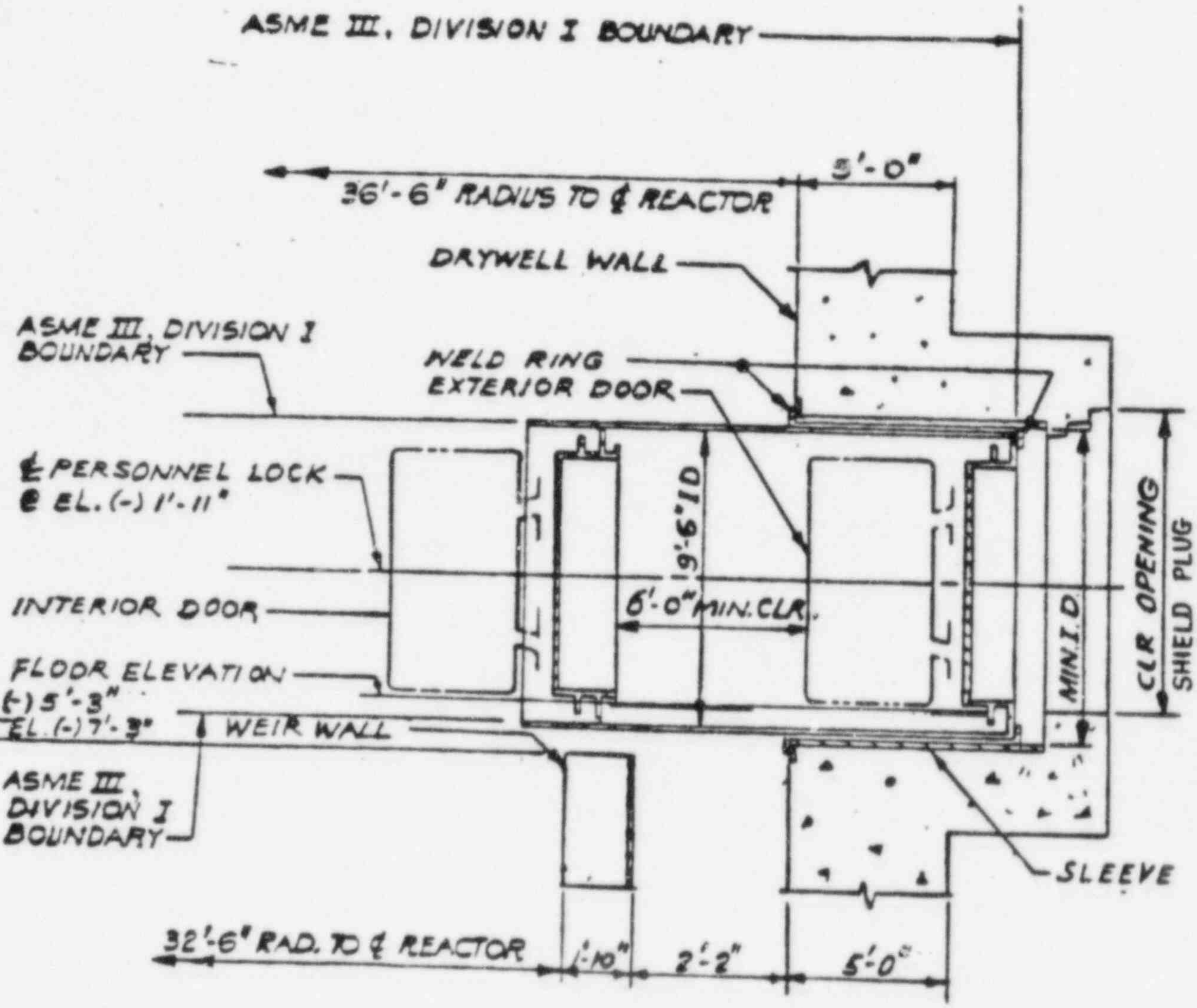


FIGURE 5 ELEVATION VIEW OF DRYWELL STRUCTURE PERSONNEL AIRLOCK

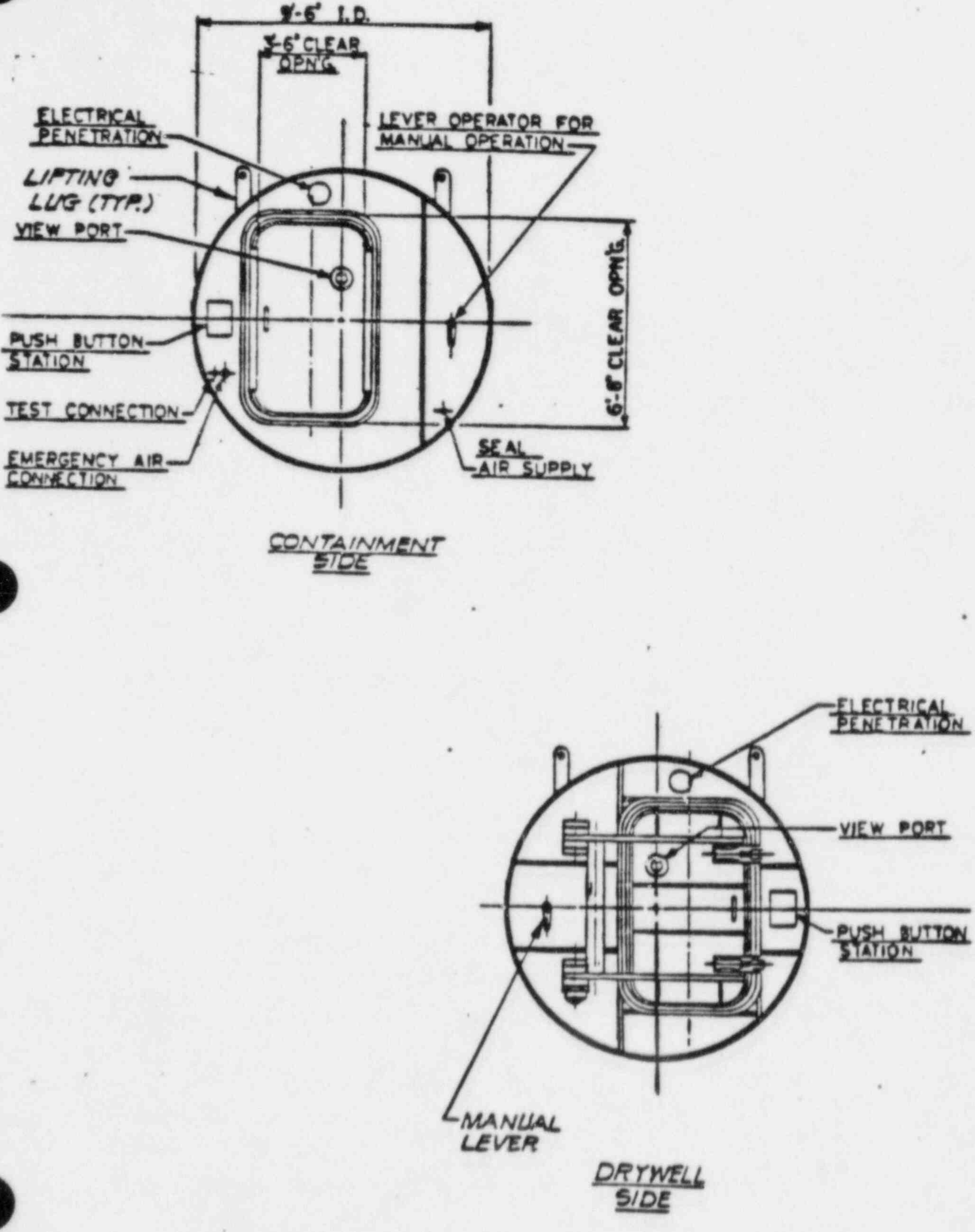


FIGURE 6 END VIEWS OF DRYWELL STRUCTURE PERSONNEL AIRLOCK

A-112

EFFECT OF MOLTEN CORE ON DRYWELL STRUCTURES

0 MOLTEN CORE ASSUMED TO BURN 6 FEET DEEP HOLE IN BASEMAT.

0 TEMPERATURE OF MOLTEN CORE ASSUMED TO BE 4000°F.

0 HEAT CONDUCTION ANALYSIS AND LINEAR ELASTIC STRESS ANALYSIS PERFORMED.

0 RESULTS

  0 TEMPERATURE IN DRYWELL WALL = ∿150°F

  0 DEFLECTION OF DRYWELL WALL = ∿0.5 INCHES

  0 STRESSES IN DRYWELL WALL AT BASEMAT

    0 ∿3500 PSI IN CONCRETE (F'C=4000 PSI)

    0 ∿40 KSI IN STEEL (FY = 46 KSI)

NO DANGER EXPECTED TO DRYWELL WALL OVERALL STABILITY

RSVI5 - II
JULY 1985

A-113

STRUCTURAL VERIFICATION STUDIES:

- TORISPHERICAL STEEL CONTAINMENT
- DRYWELL (STEEL) HEAD
- REINFORCED CONCRETE DRYWELL ROOF SLAB
- RELIABILITY EVALUATION OF STEEL CONTAINMENT

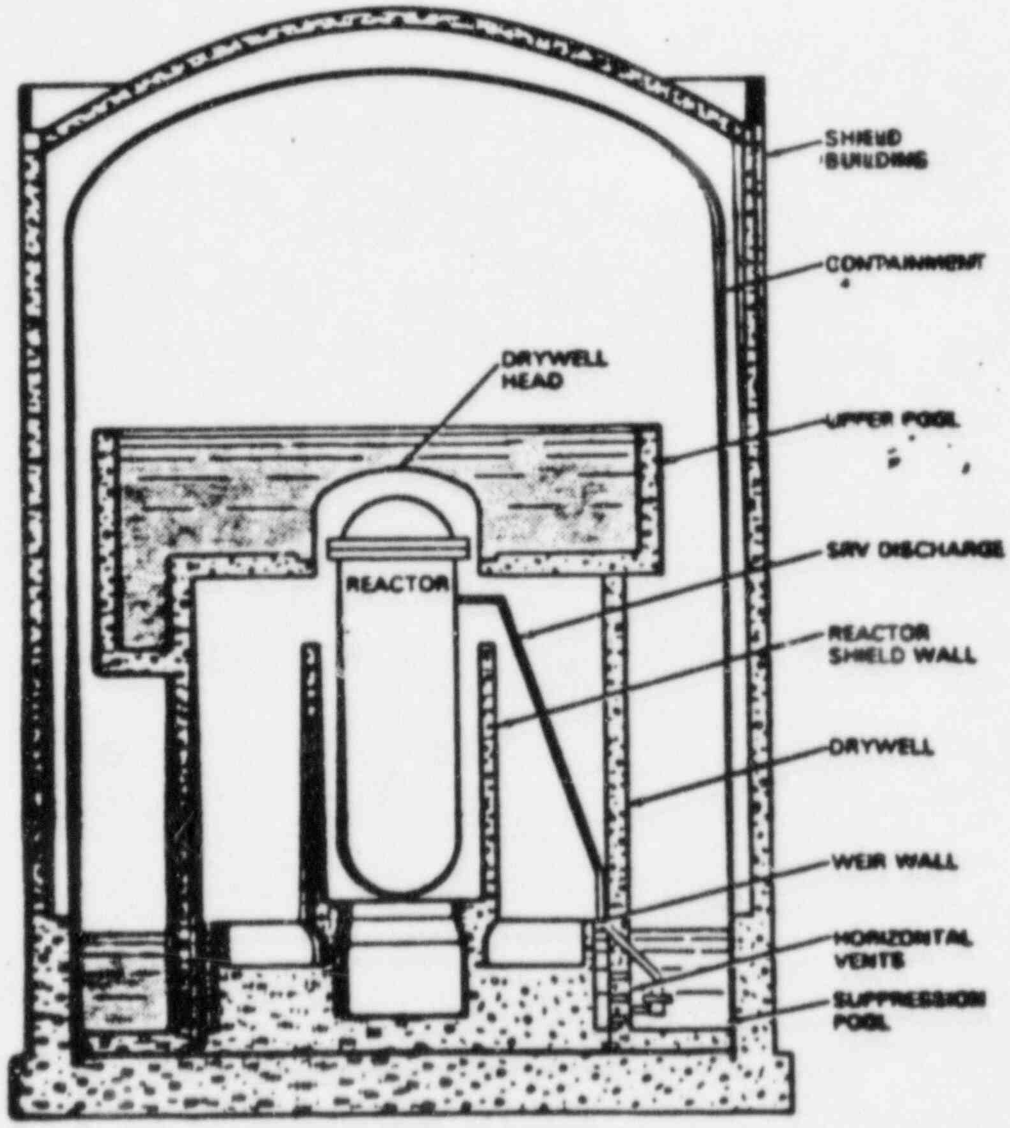


Figure 15.1 Principal features of MARK III containment

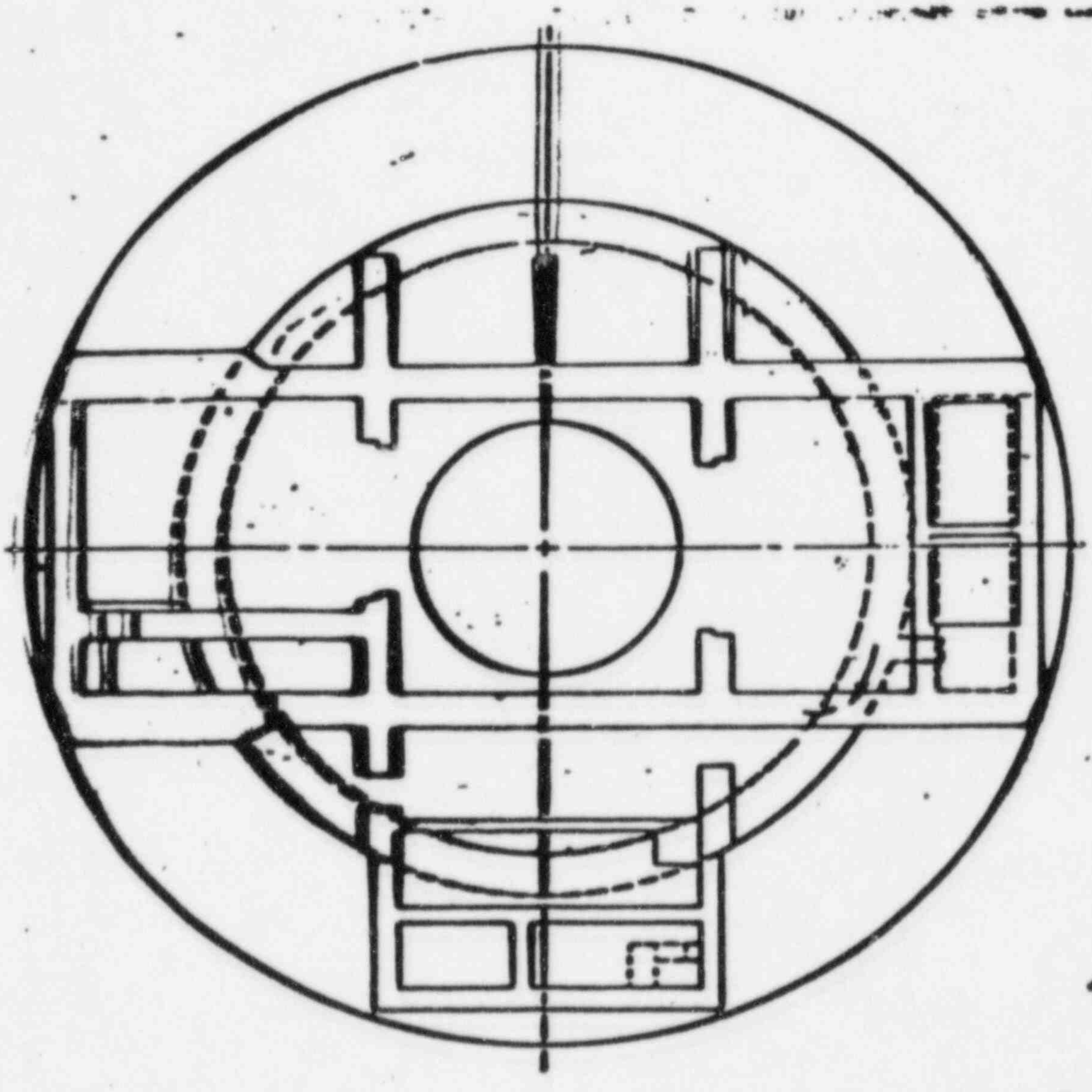
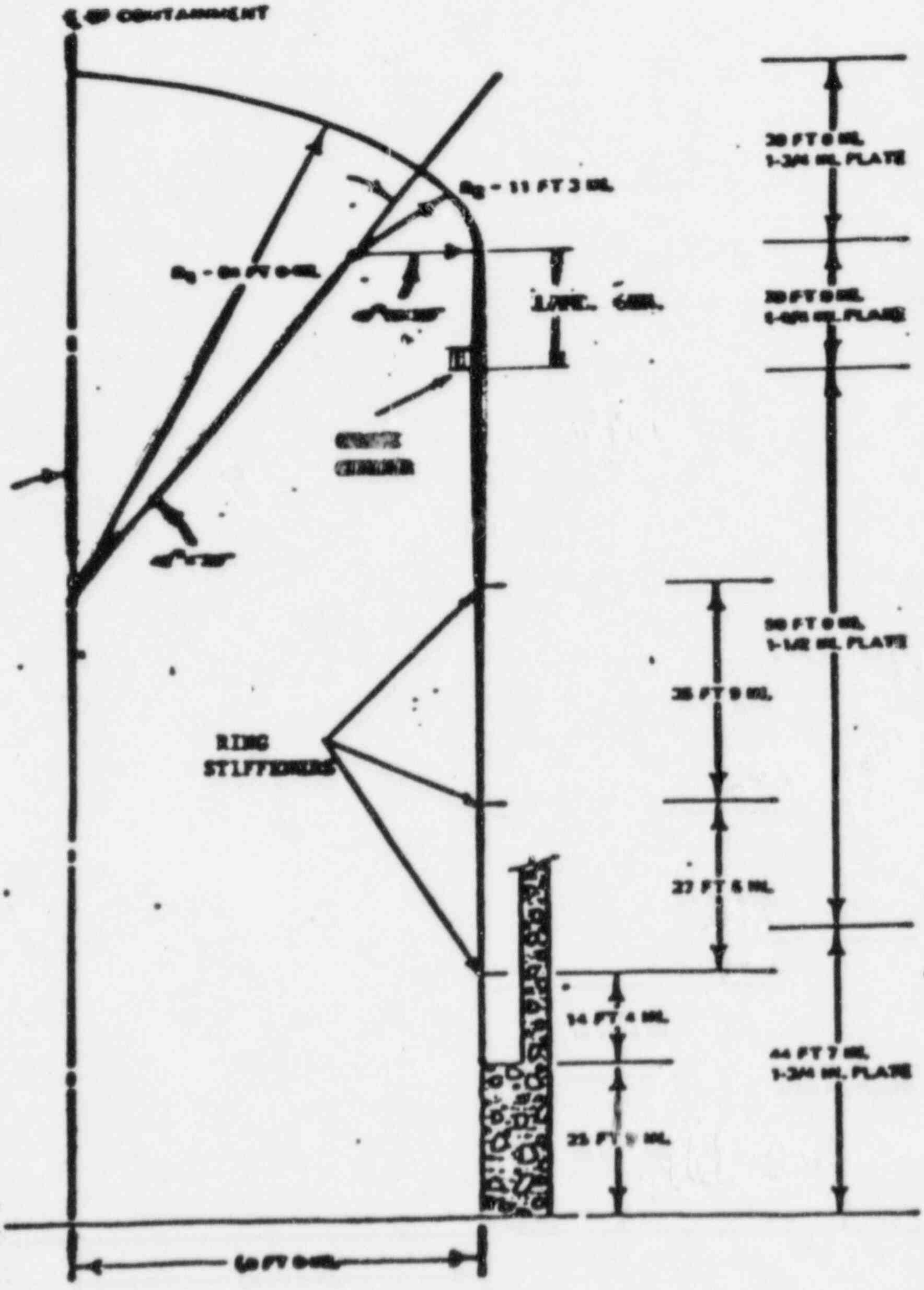


Fig. 12 - Plan View of Drywell Roof Slab



.Fig. 1 - Geometrical Details of Torispherical Steel Containment

A-117

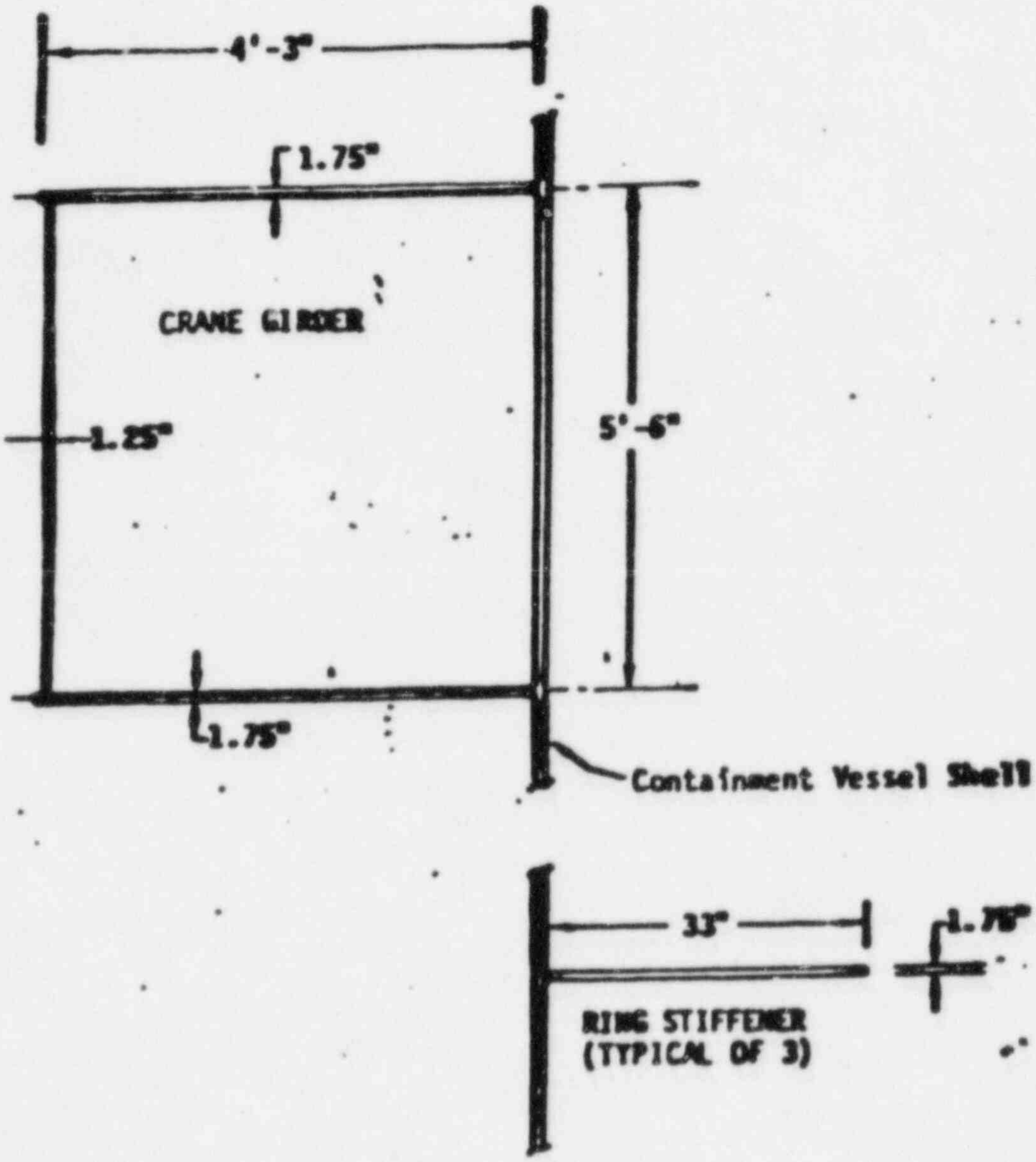


Fig. 2 - Detailed Geometry of Ring Stiffeners and Crane Girder

ANALYSIS PERFORMED FOR TORISPHERICAL CONTAINMENT

A) PLASTIC LIMIT ANALYSIS

B) SMALL DEFORMATION ELASTIC-PLASTIC FINITE-ELEMENT ANALYSIS

C) LARGE DEFORMATION ELASTIC-PLASTIC FINITE-ELEMENT ANALYSIS

D) BUCKLING EVALUATION

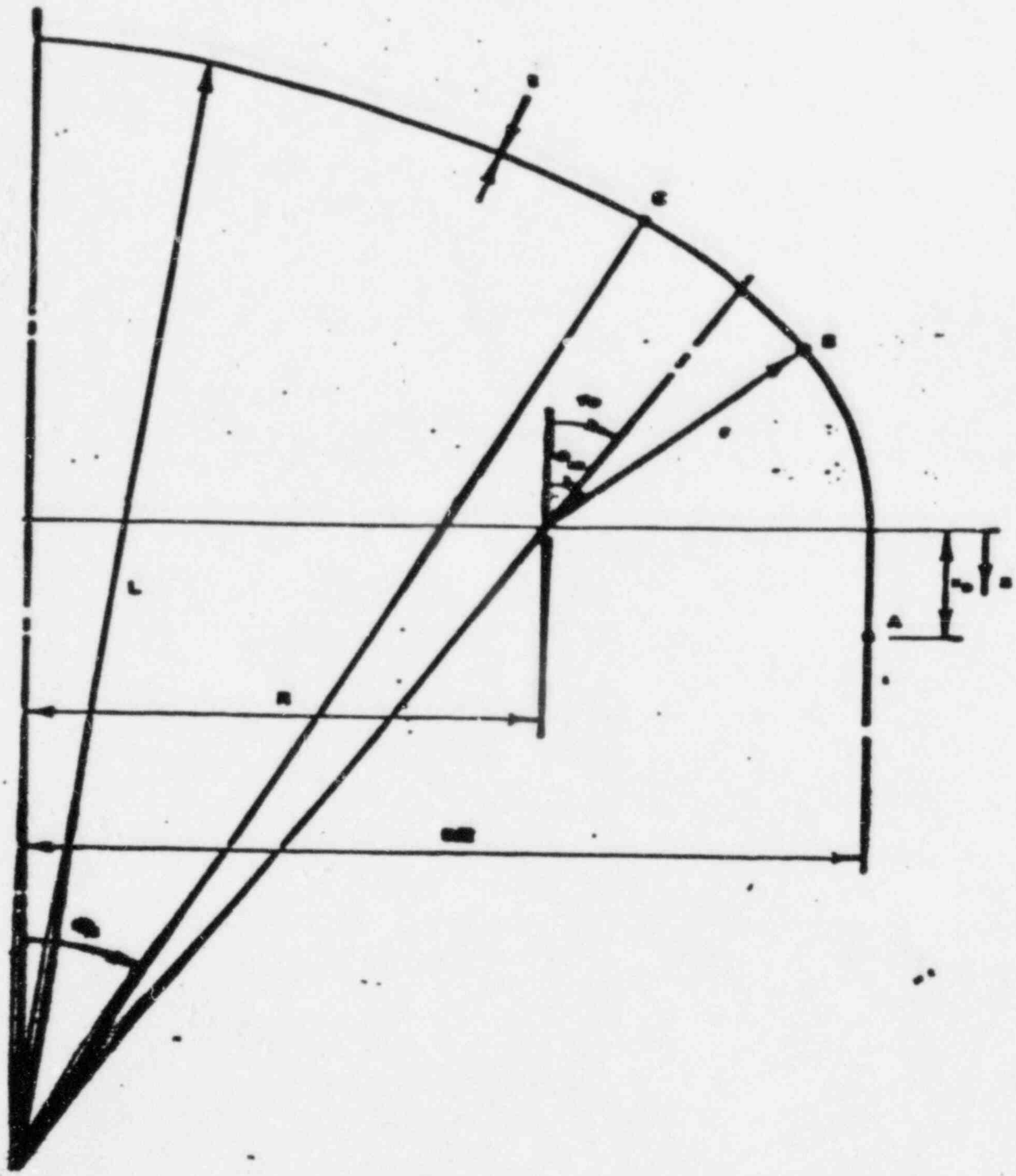


Fig. 3 - Location of Plastic Hinge Circles and Nomenclatures.

A-120

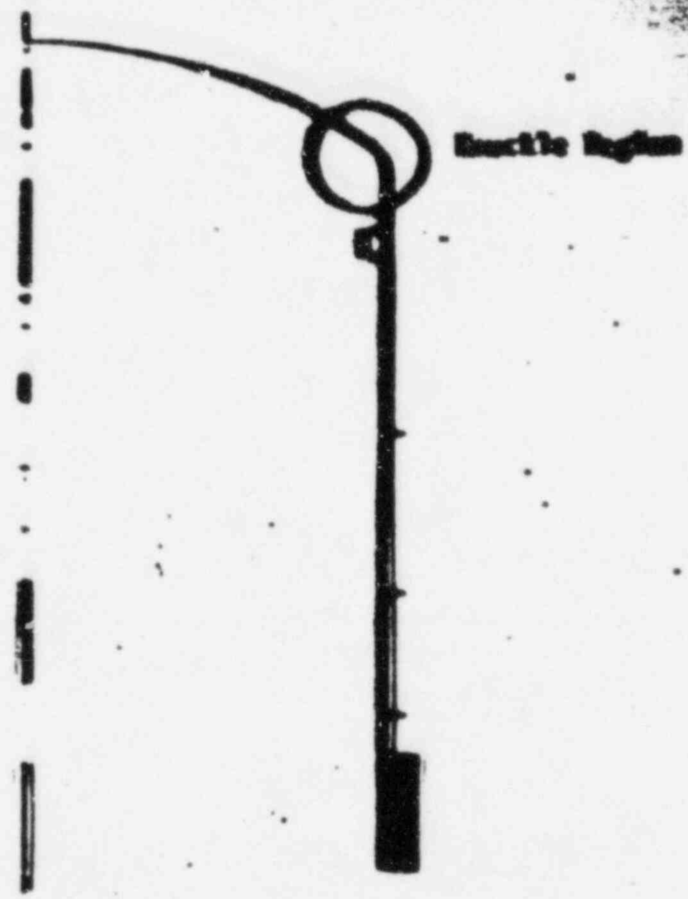


Fig. 4 - Finite Element Model for the Torispherical Containment Shell

A-21

Fig. 5 - Finite Element Grid in the Knuckle Region

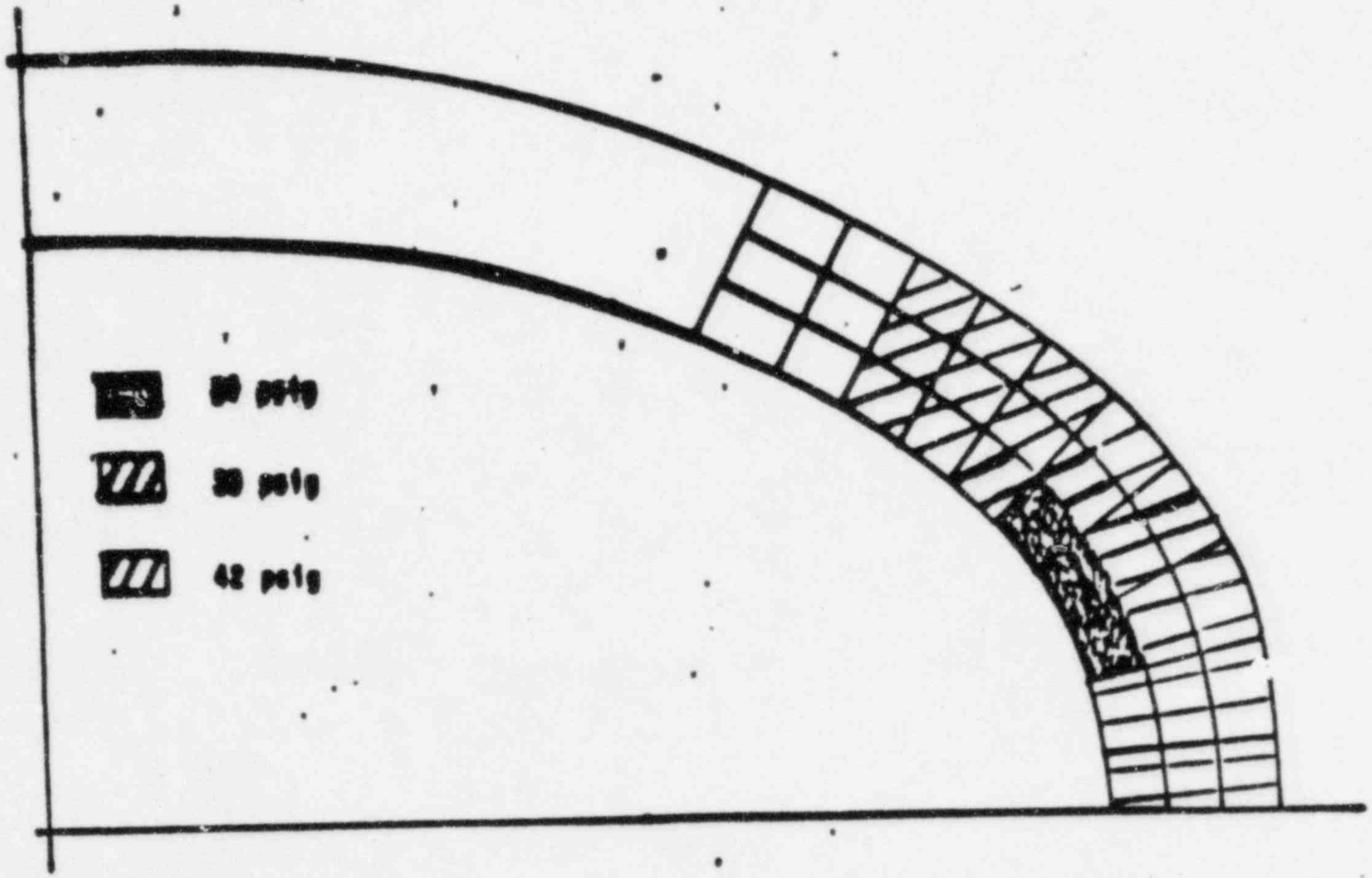


Fig. 7 - Plastic Elements in the Knuckle Region of the Containment Shell

A-122

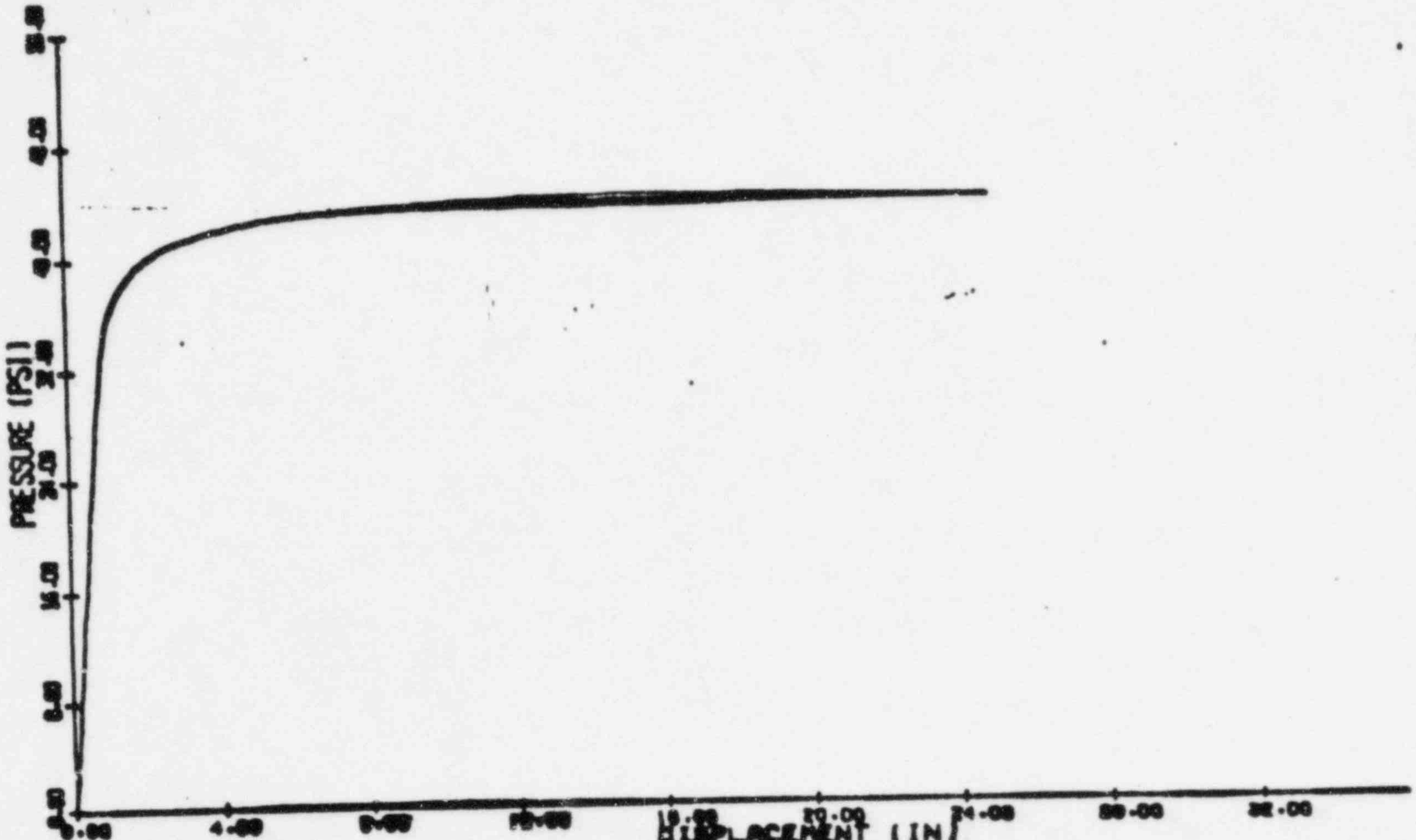


Fig. 6 - Inward Radial Displacement of Node 661 vs. Internal Pressure for Small Deformation Analysis

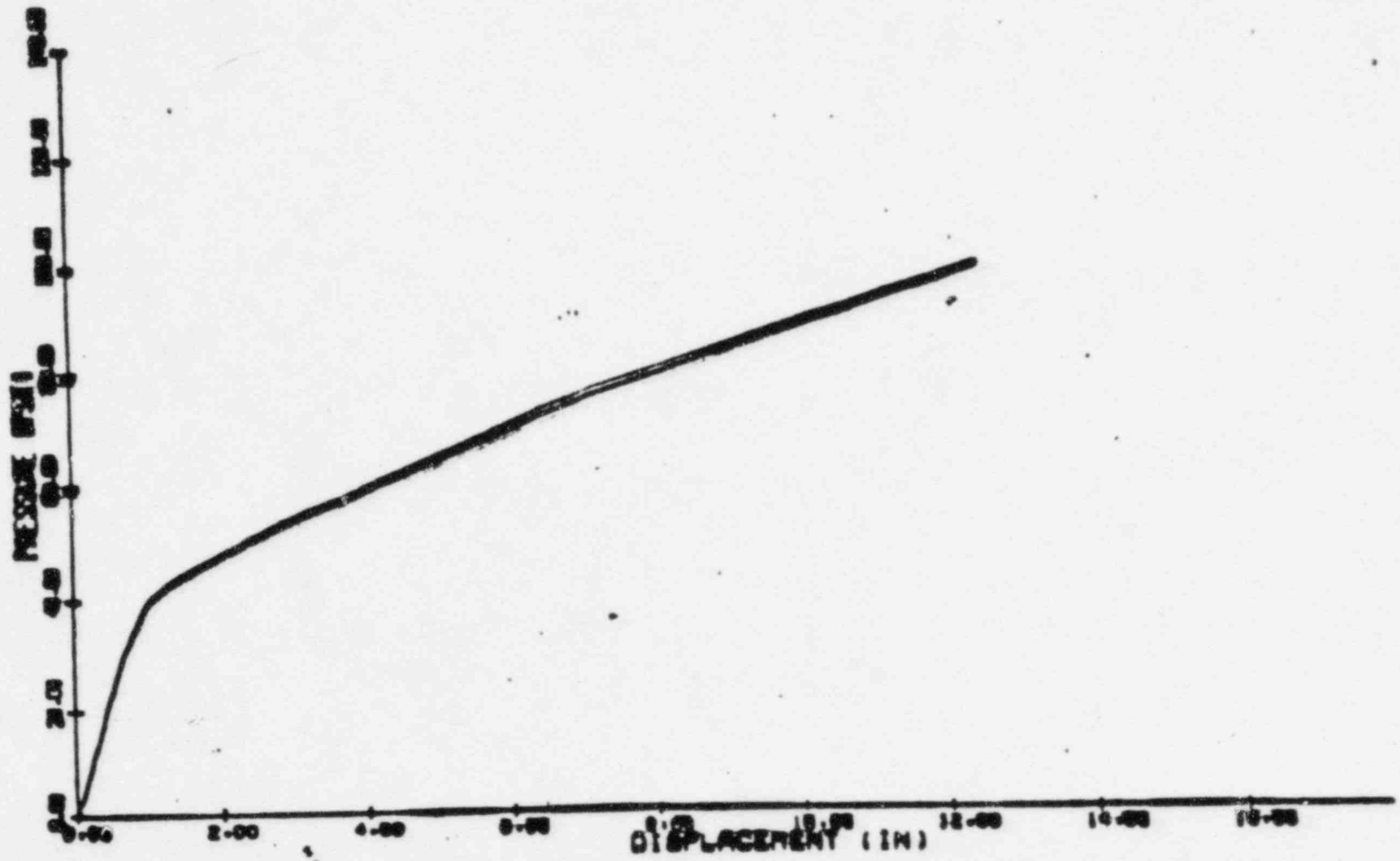


Fig. 8 - Inward Radial Displacement of Node 051 vs. Internal Pressure for Large Deformation Analysis

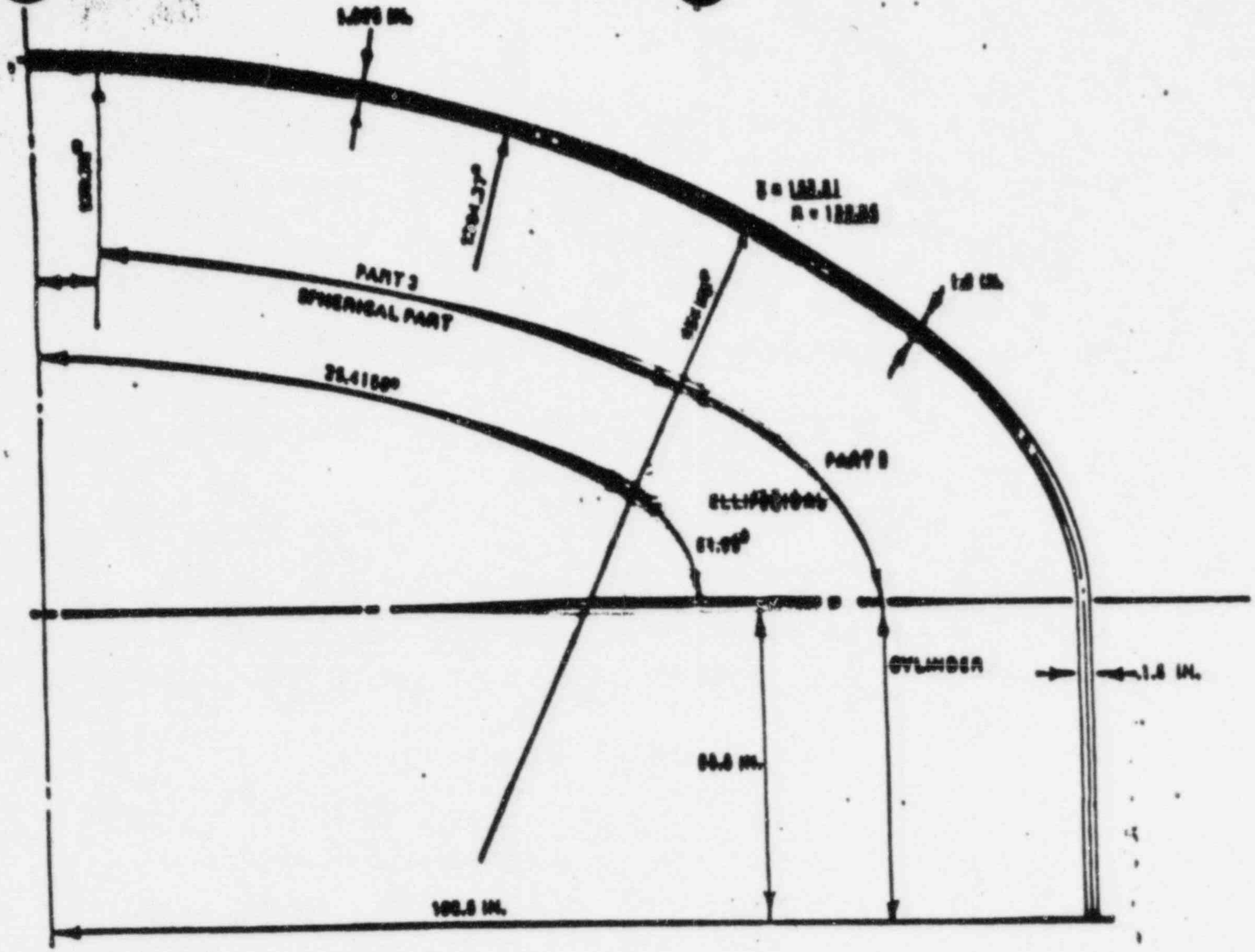


Fig. 9 - Drywell Head Geometric Details

A-125

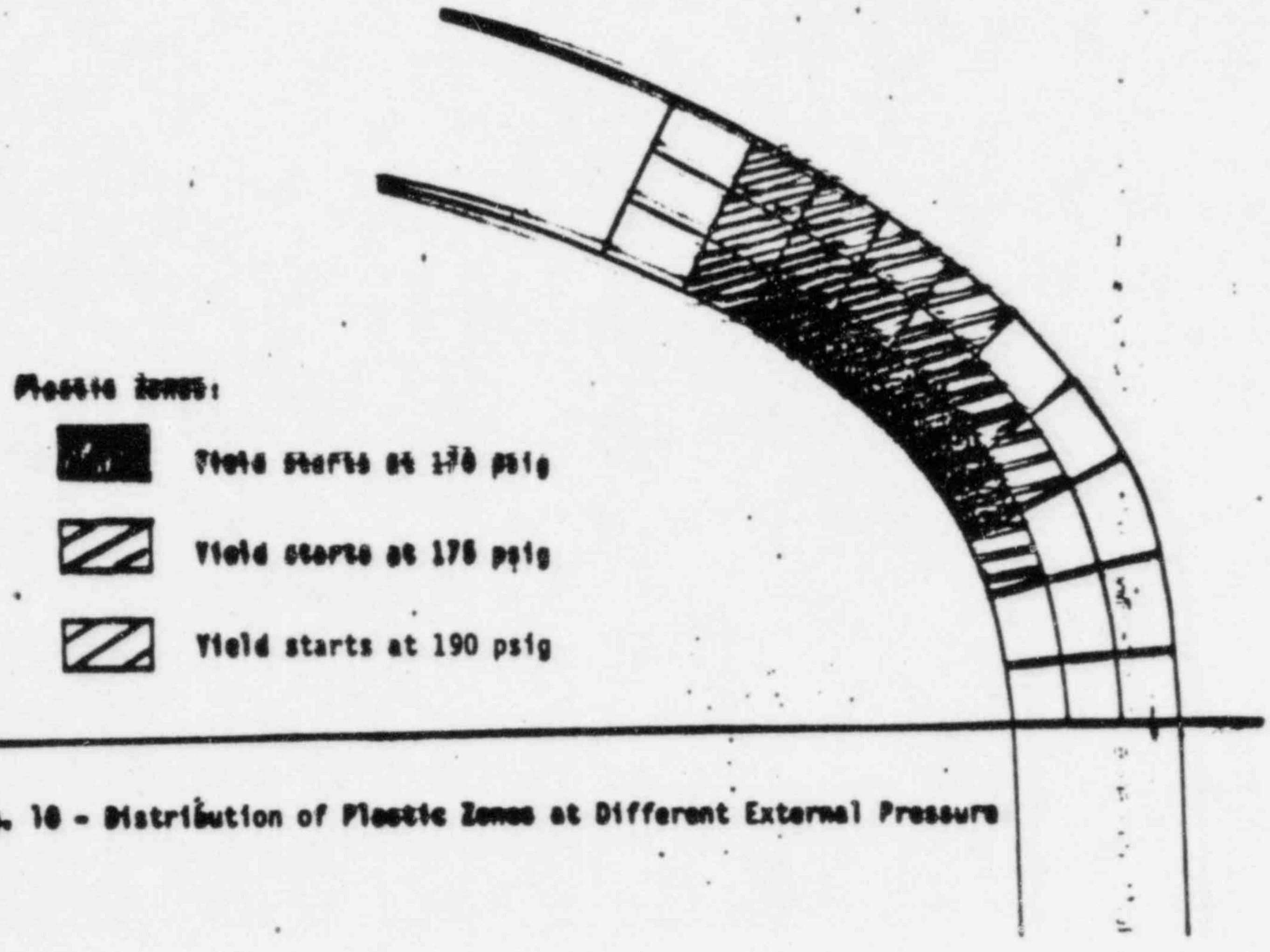


Fig. 10 - Distribution of Plastic Zones at Different External Pressure

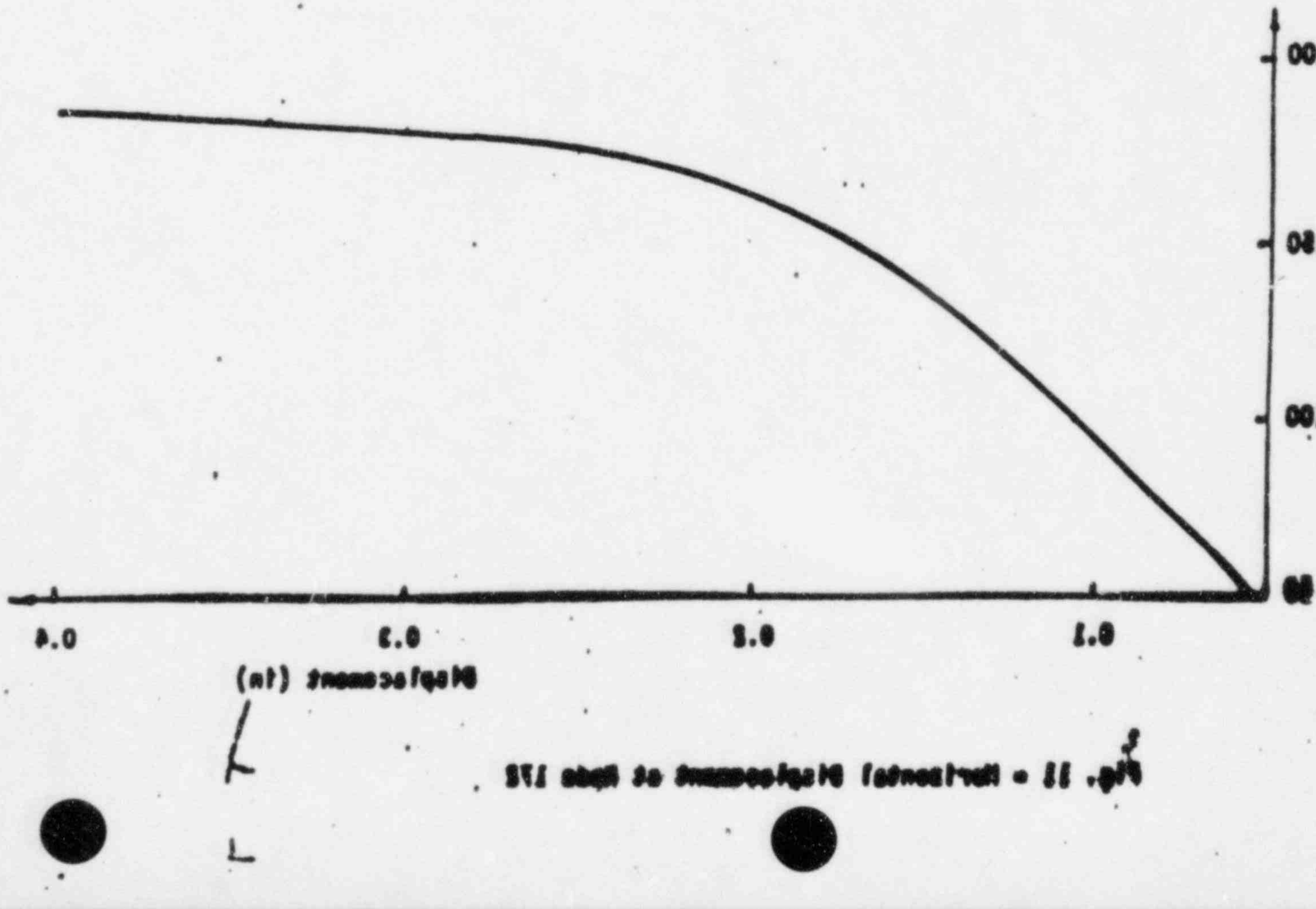


Fig. 11 - Horizontal Displacement at Node 172

A 127

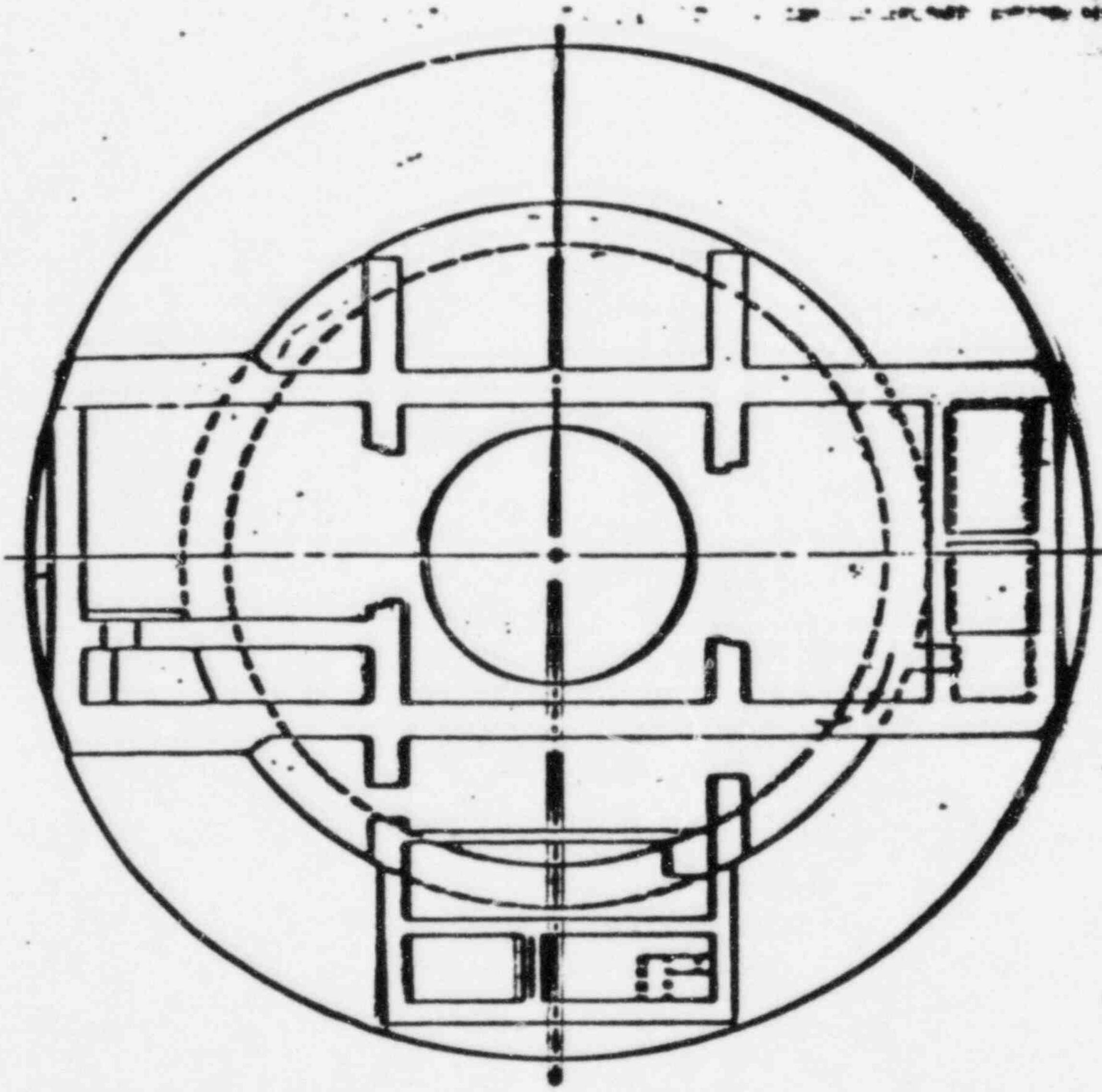


Fig. 12 - Plan View of Drywell Roof Slab

A-128

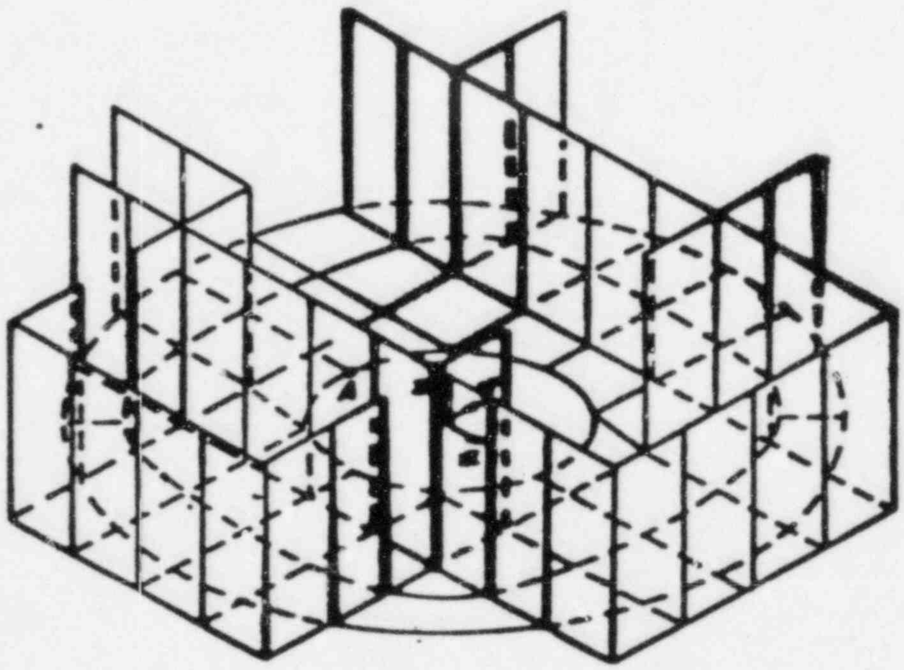


Fig. 13 - Finite Element Mesh of Drywell Roof Slab

A-129

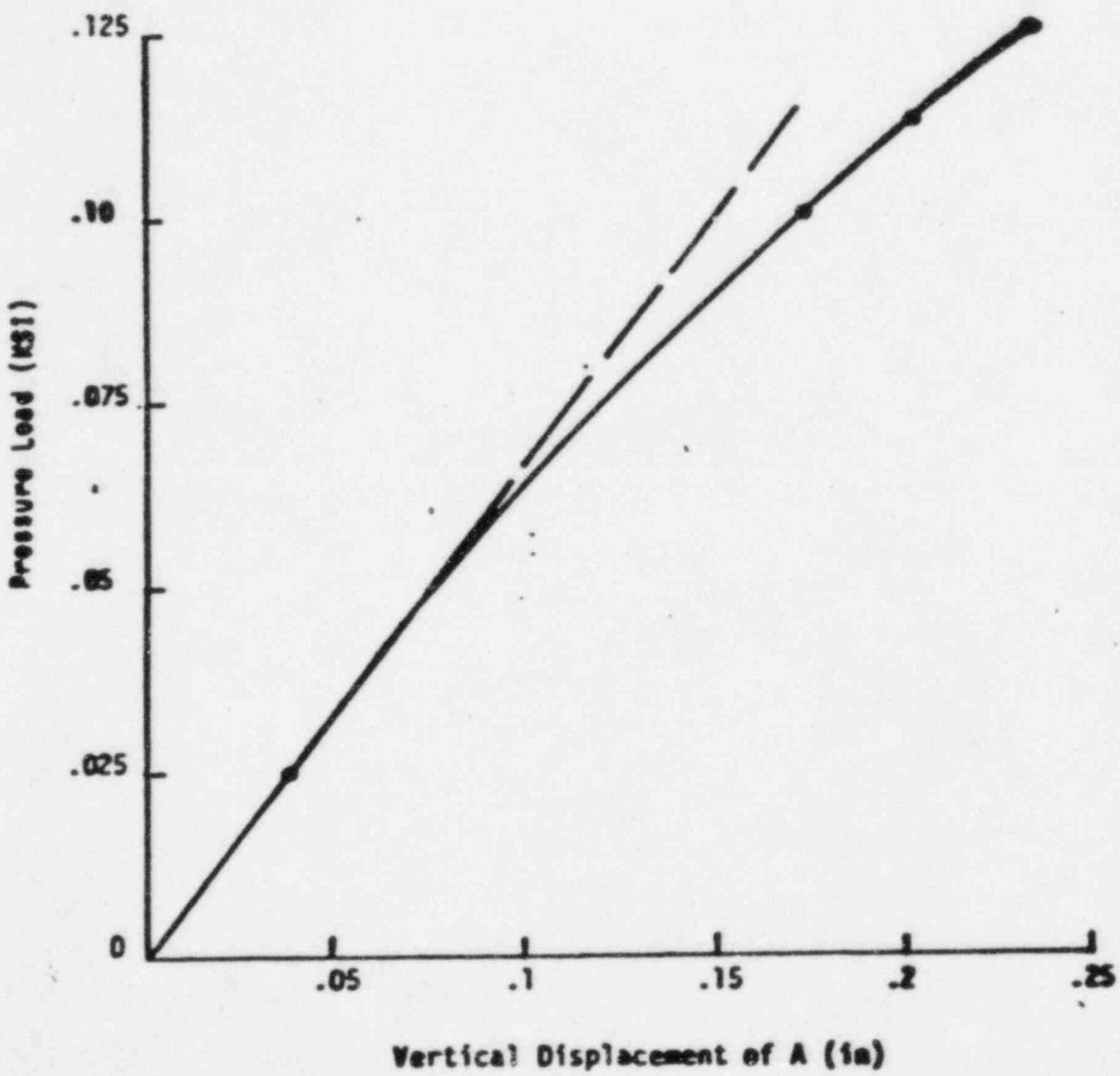
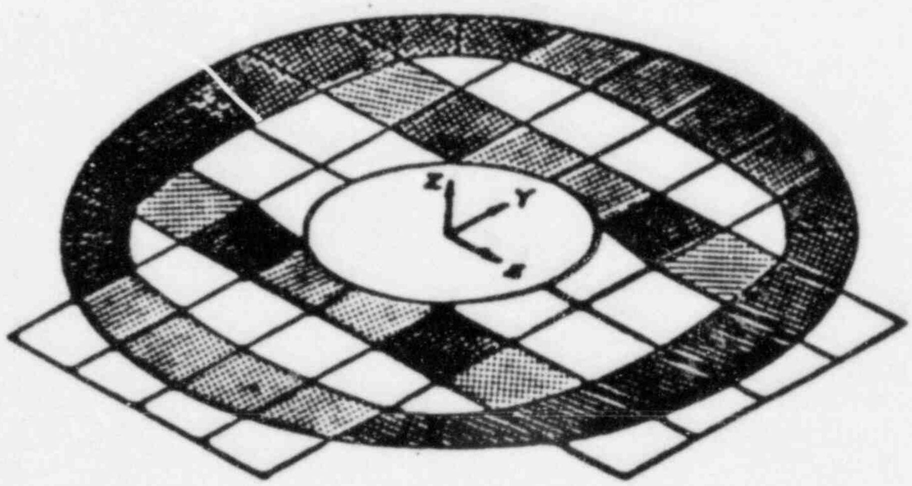


Fig. 14 - Displacement at Point A of Roof Slab vs. Pressure

A130



Cracking in Top Part

Cracking Through this Thickness

Fig. 15 - Crack Propagation for the Top Layer of Floor
(Reinforced Concrete)

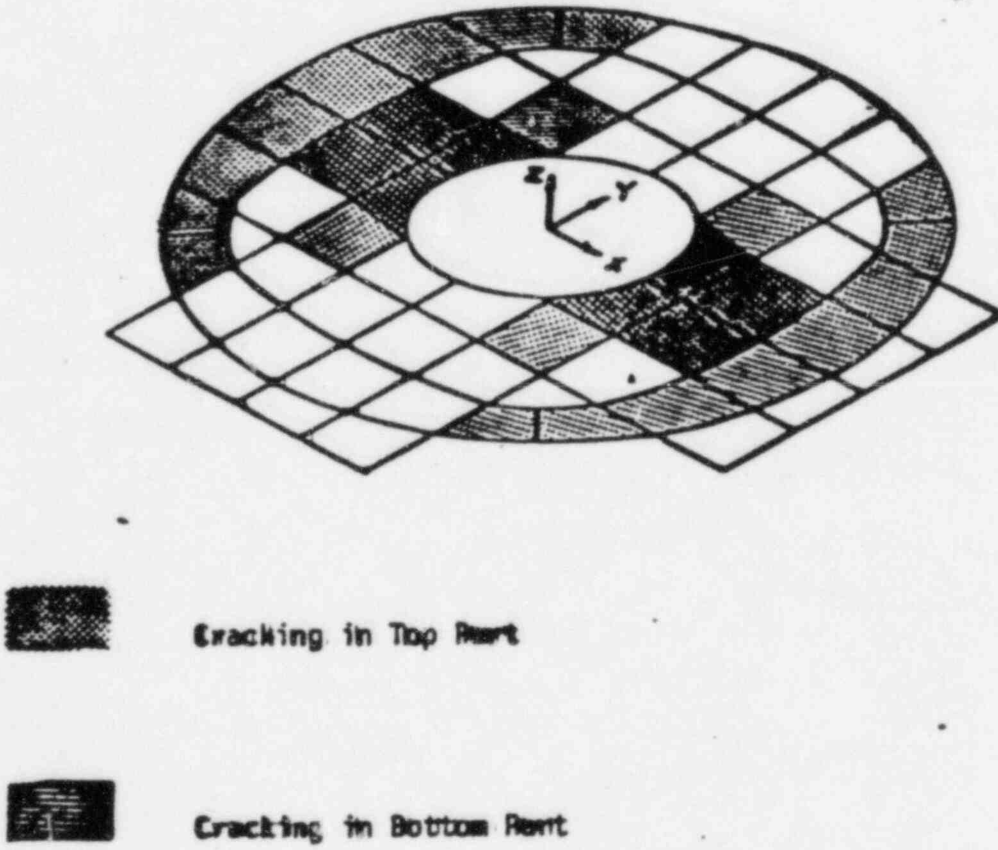
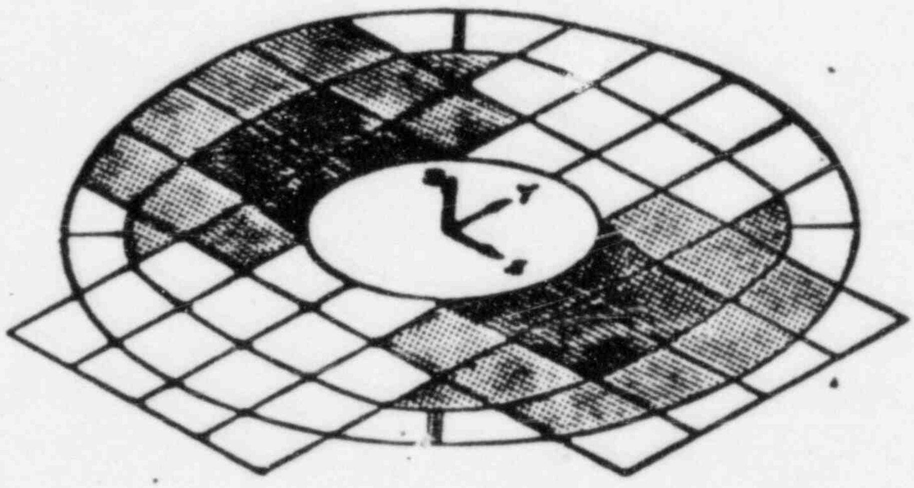


Fig. 16 - Crack Propagation for the Middle Layer of Floor (Plain Concrete)

A132



Cracking in the Bottom Part

Cracking Through this Thickness

Fig. 17 - Crack Propagation for the Bottom Layer of Floor
(Reinforced Concrete)

A 133

SUMMARY

TORISPHERICAL STEEL CONTAINMENT

(0) SHIELD-DRUCKER PLASTIC LIMIT APPROACH - 38 PSIG

(0) SMALL DEFORMATION ELASTIC-PLASTIC F.E. - 42.7 PSIG

(0) LARGE DEFORMATION ELASTIC PLASTIC F.E. - 2% STRAIN AT 100 PSIG, SLOPE NOT YET FLAT.

(0) BUCKLING FAILURE - 61.3 PSIG

DRYWELL HEAD

(0) LARGE DEFORMATION ELASTIC-PLASTIC F.E. - SECTION FULLY PLASTIC AT 190 PSIG, SLOPE NOT FLAT.

(0) LEVEL A CAPACITY (NE-3221) - 106 PSIG

(0) LEVEL A BUCKLING (NE-3133) - 75 PSIG

(0) LEVEL C CAPACITY (NE-3221) - 180 PSIG

(0) ELASTIC BUCKLING OF SPHEROIDAL PORTION - 196 PSIG

SUMMARY (CONT'D)

DRYWELL ROOF SLAB

(a) FINITE ELEMENT (NON-LINEAR) SHELL MODEL - 125 PSIG TOP AND BOTTOM 9" CRACKED, MIDDLE SECTION INTACT, REBARS ELASTIC.

(b) SIMPLIFIED SLAB CALCULATION (GE)

(c) SERVICE ALLOWABLE - 90 PSIG

(d) FACTORED ALLOWABLE - 96 PSIG

σ_{YIELD} = 36 KSI

J_{IC} = 600 IN-LB/IN^2

E = 29 X 10^6 PSI

ASSUME σ(OPERATING) = σ_Y, AND σ = 24 KSI

CRACK DEPTH/ THICKNESS	CRACK LENGTH (INCHES)
.1	67
.2	30.6
.3	20.4
.4	16.8
.5	13.4
.6	10.2
.7	9.6
.8	8.4
1.0	6.7

CRACK DEPTH/ THICKNESS	CRACK LENGTH (INCHES)
<.3	VERY LARGE
.4	100.7
.5	40.2
.6	25.2
.7	18.3
.8	14.4
.9	11.8

PROBABILISTIC ANALYSES
INPUT PARAMETERS

MATERIAL — SA 516 GR 70

MOD. OF ELASTICITY — 29×10^3 KSI (CONSTANT)

YIELD STRENGTH (MEAN)
(NORMAL DIST.) — 48.3 KSI

STANDARD DEVIATION — 3.12 KSI

__PRESSURE__

MEAN (TREATED AS A PARAMETER)

COEFFICIENT OF VARIATION — 15 AND 20 OF THE MEAN

NORMAL DISTRIBUTION

LIMIT STATE - YIELDING UNDER COMBINED STRESSES (HOOP, MERIDIONAL)

- 2% MAXIMUM PRINCIPAL STRAIN

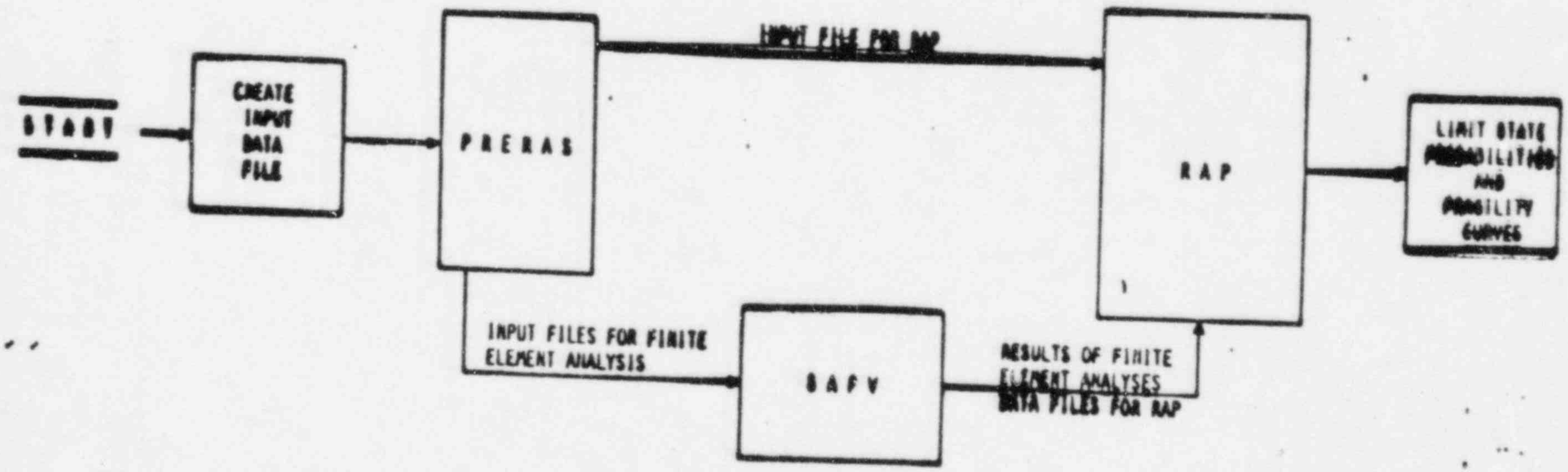


Fig. 1. Organization of RAS Program.

A-138

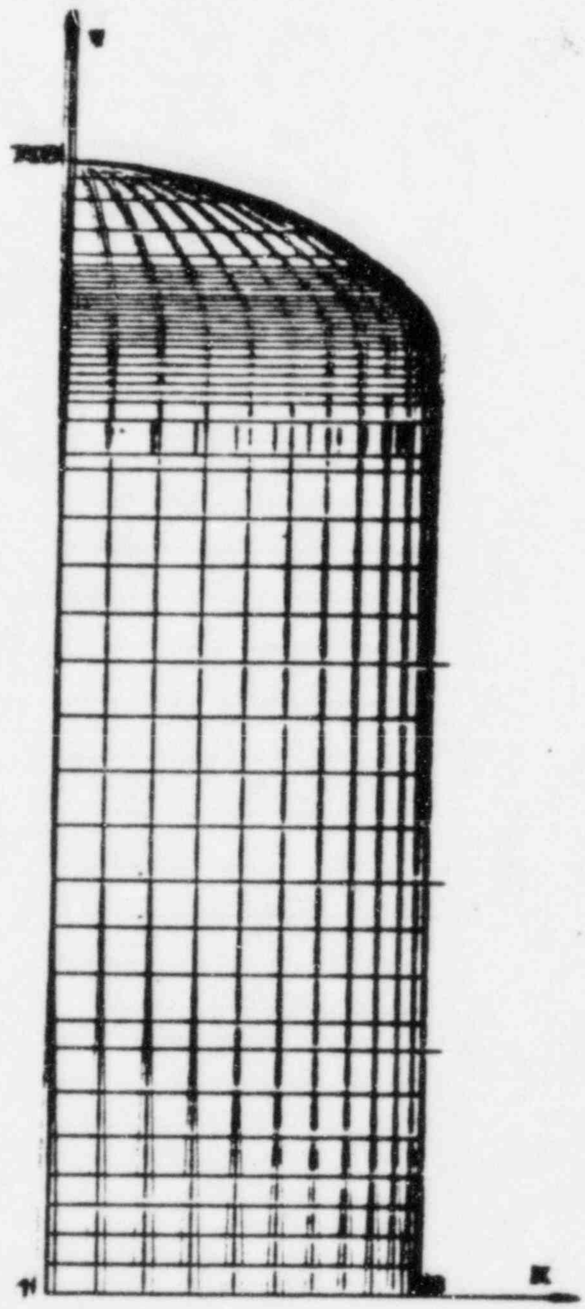


Fig. 3a - Containment Finite Element Model - Side View.

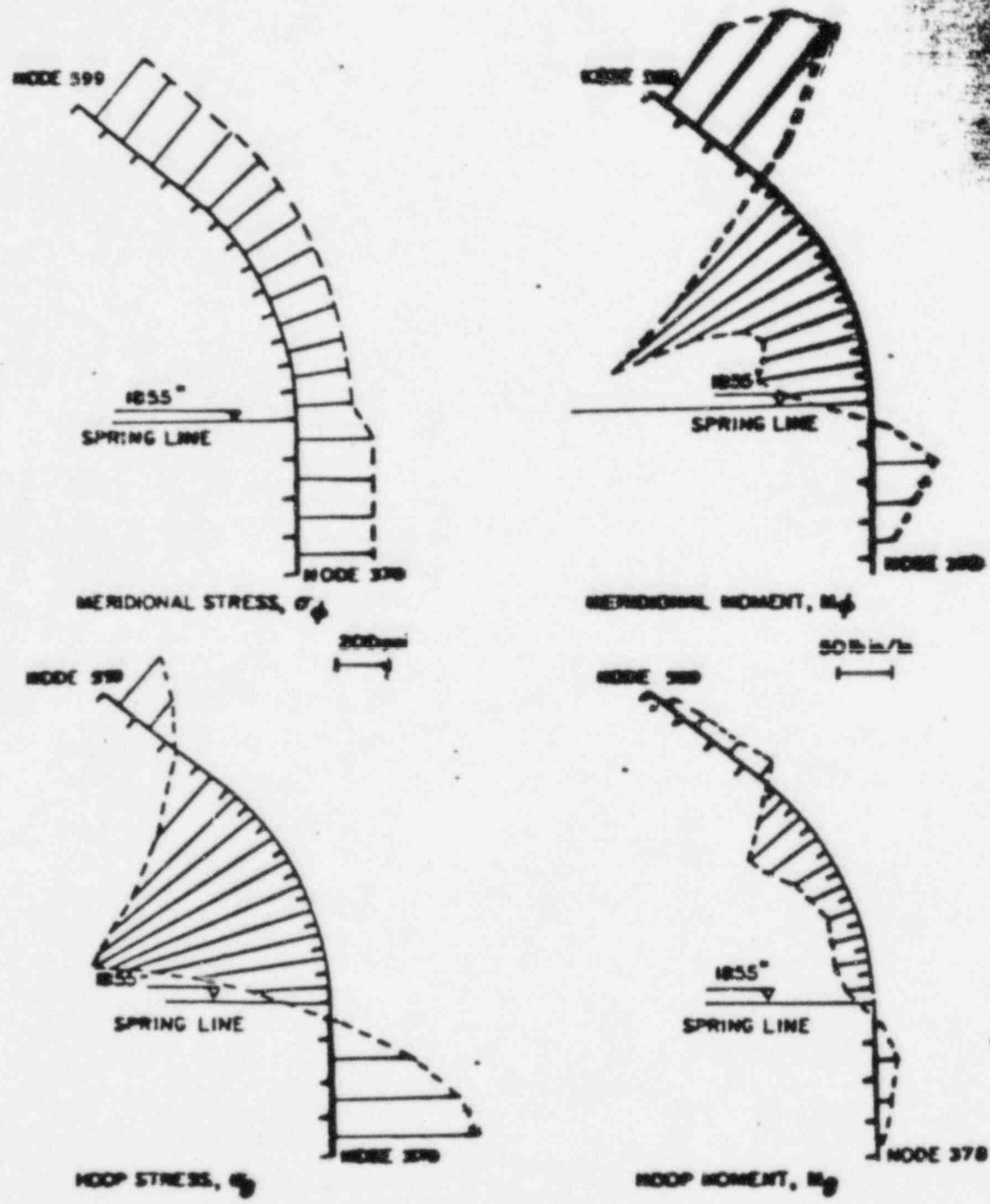


Fig. 6 - Stresses in Knuckle Region for 3 psi Internal Pressure.

A140

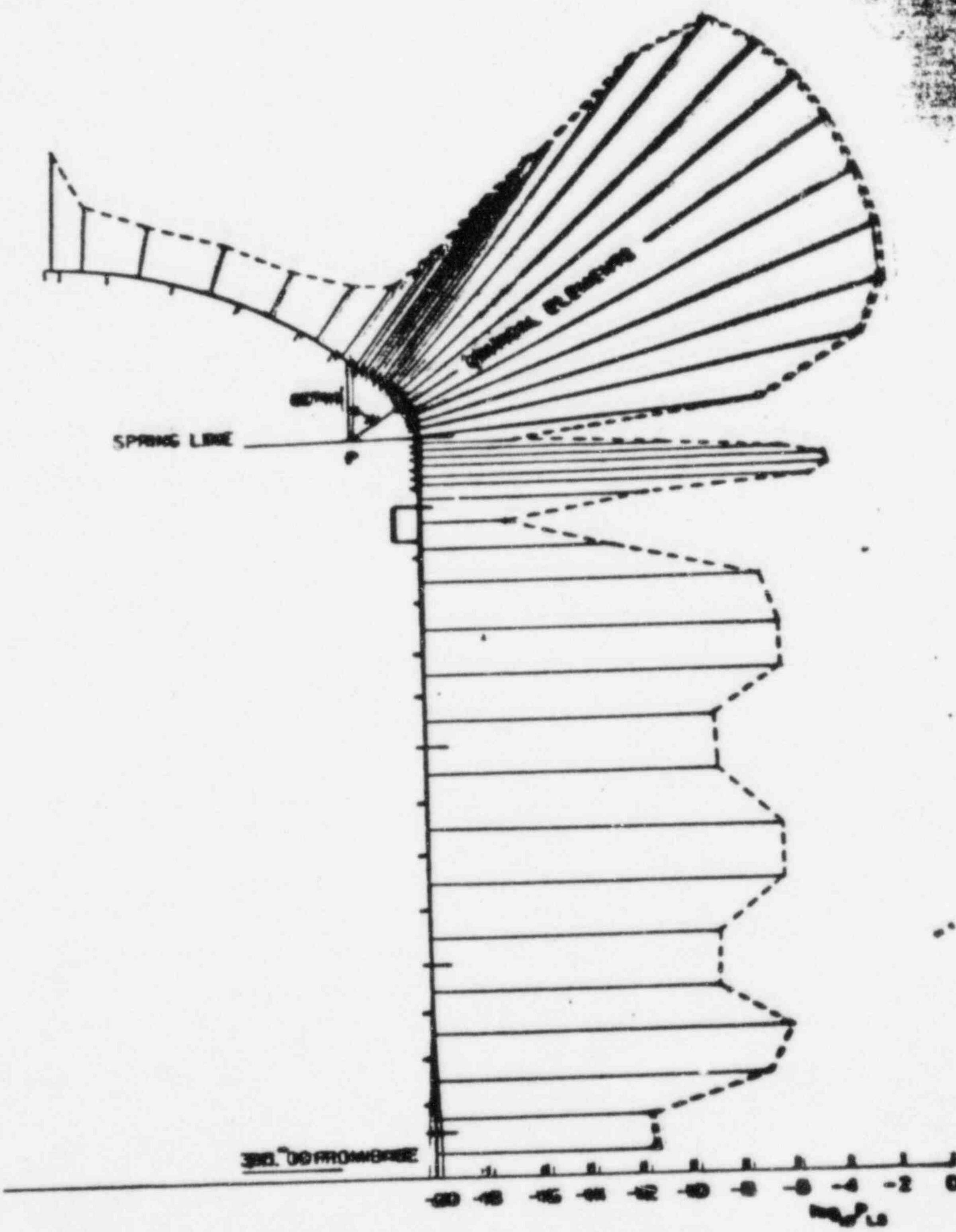


Fig. 8 - Conditional Limit State Probabilities for P_M = 45 psi and σP_M = 9 psi.

Table 3. Limit State Probabilities

Mean Pressure $\bar{P}_B$(*) (psi)	Limit State Probabilities Conditional
35	6.32×10^{-2}
40	2.19×10^{-1}
45	4.32×10^{-1}
50	6.37×10^{-1}
55	7.49×10^{-1}
60	8.42×10^{-1}
65	9.16×10^{-1}
70	9.49×10^{-1}
75	9.68×10^{-1}
80	9.80×10^{-1}
85	9.87×10^{-1}
90	9.91×10^{-1}
95	9.94×10^{-1}
100	9.95×10^{-1}

$^{*}\sigma_{P_B} = 0.2\ \bar{P}_B$

PROBABILITY OF LARGE STRAIN FAILURE

MEAN PRESSURE LOAD $\bar{p}$ (psi)	PROBABILITIES $\sigma_p = 0$	$\sigma_p = 0.15\bar{p}$	$\sigma_p = 0.2[illegible]\bar{p}$
60	$[illegible] \times 10^{-[illegible]}$	1.0×10^{-8}	6.40×10^{-6}
70	$[illegible] \times 10^{-[illegible]}$	2.86×10^{-5}	9.60×10^{-4}
80	$[illegible] \times 10^{-[illegible]}$	2.72×10^{-3}	1.66×10^{-2}
90	8.33×10^{-11}	3.71×10^{-2}	8.64×10^{-2}
100	$[illegible] \times 10^{-5}$	0.165	0.229
110	$[illegible] \times 10^{-2}$	0.382	0.410
115	[illegible]	[illegible]	0.498
120	[illegible]	[illegible]	0.580
130	[illegible]	0.773	0.714
140	1.0	[illegible]	0.811
150	1.0	[illegible]	0.876
160	1.0	[illegible]	0.918
170	1.0	[illegible]	0.946

Yield Strength: Mean, $\bar{S}_y$ = 48.3 ksi

Standard Deviation, σ_{S_y} = 3.[illegible] ksi

SUMMARY (PROBABILISTIC ANALYSES)

(0) YIELD LIMIT STATE (BNL) MEAN FAILURE PRESSURE - 46 PSIG

(0) GE PLASTIC LIMIT STATE - 52 PSIG

(0) 2% MAXIMUM PRINCIPAL STRAIN LIMIT STATE (BNL) - 115 PSIG

BROOKHAVEN NATIONAL LABORATORY bnl
ASSOCIATED UNIVERSITIES, INC. aui

GESSAR - SEVERE ACCIDENT THREAT

TO CONTAINMENT

PRESENTED TO ACRS, JULY 12, 1985

JACK ROSENTHAL - ONRR/DSI/RSB

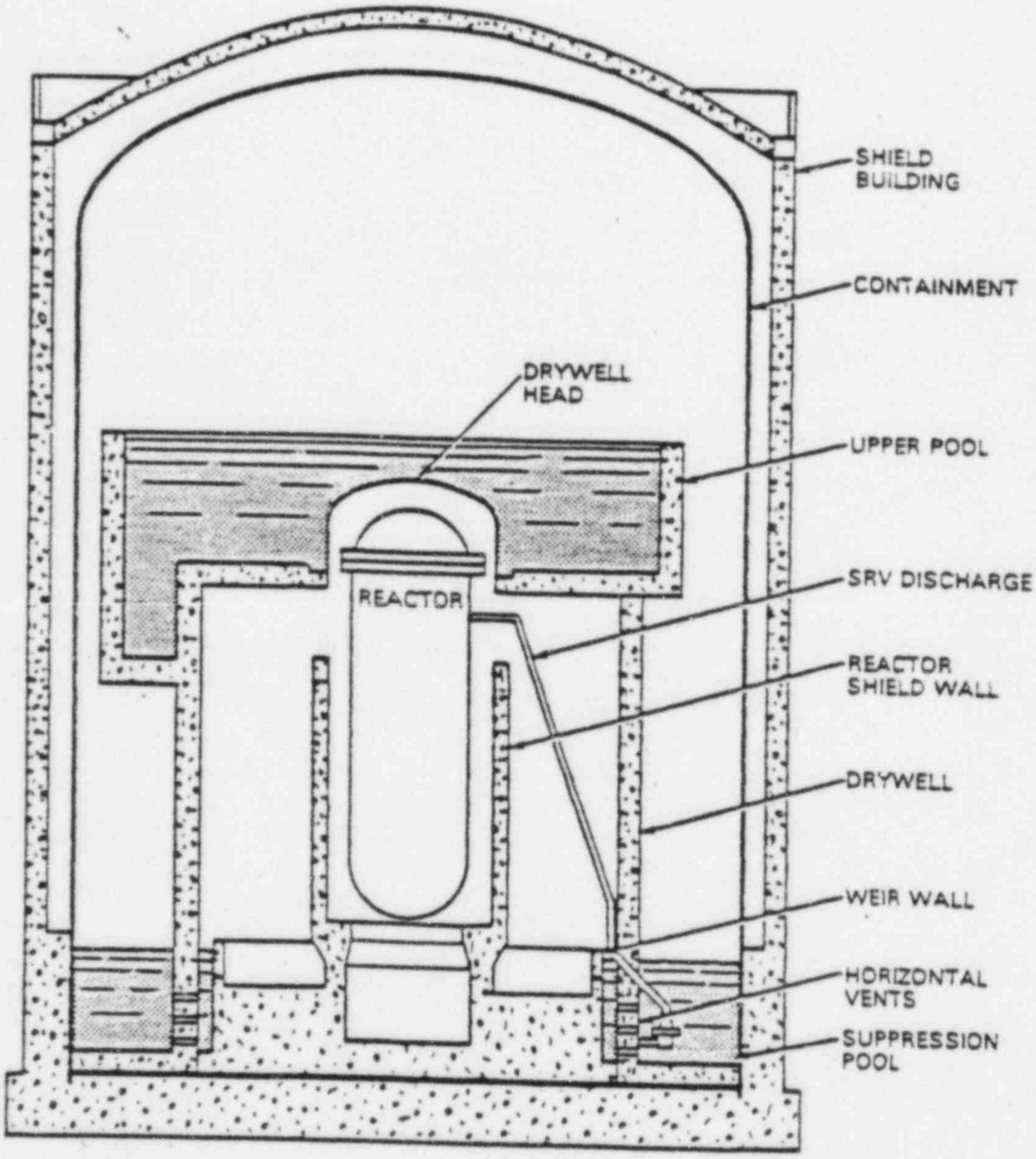


Figure 15.1 Principal features of MARK III containment

A-146

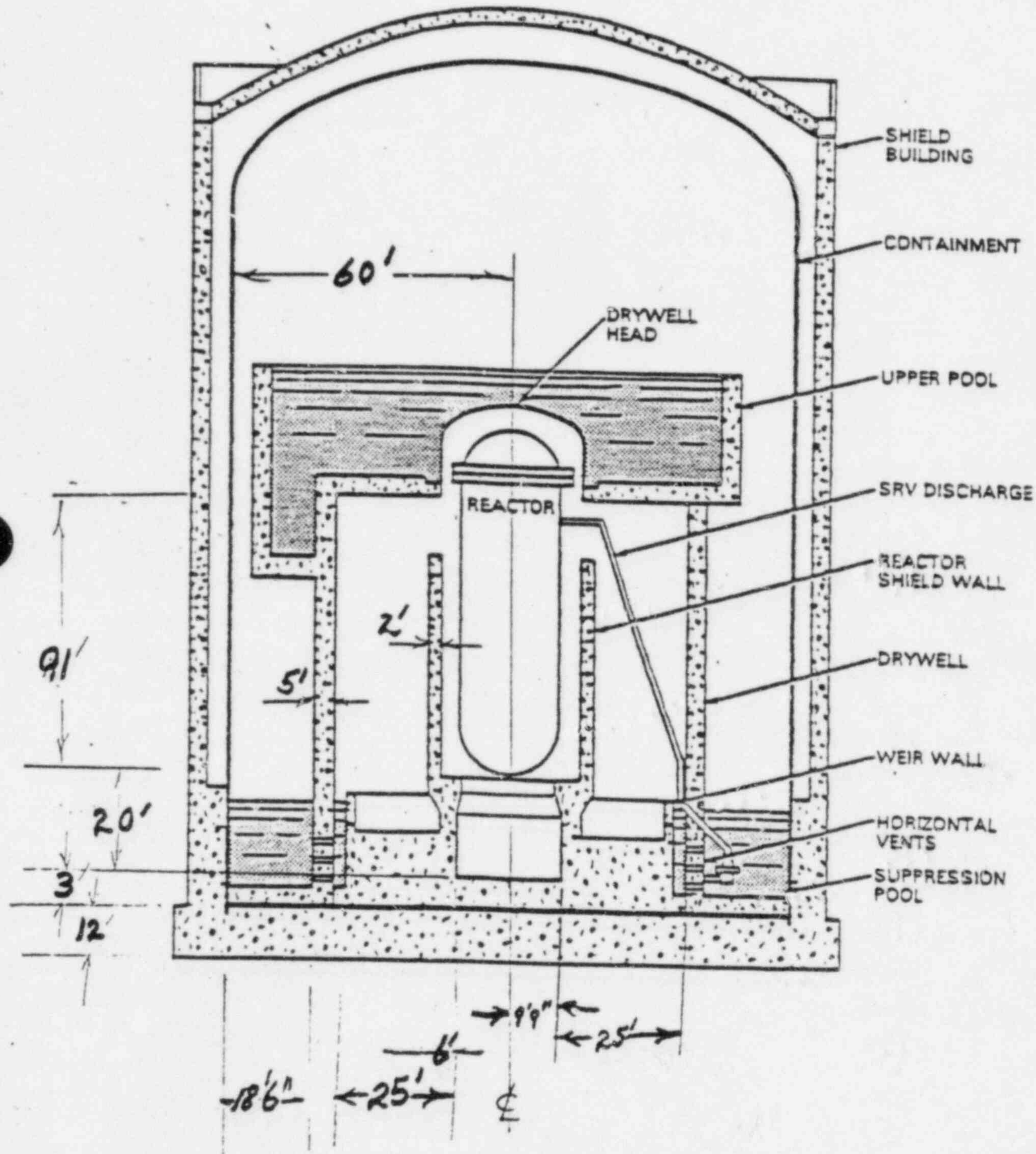
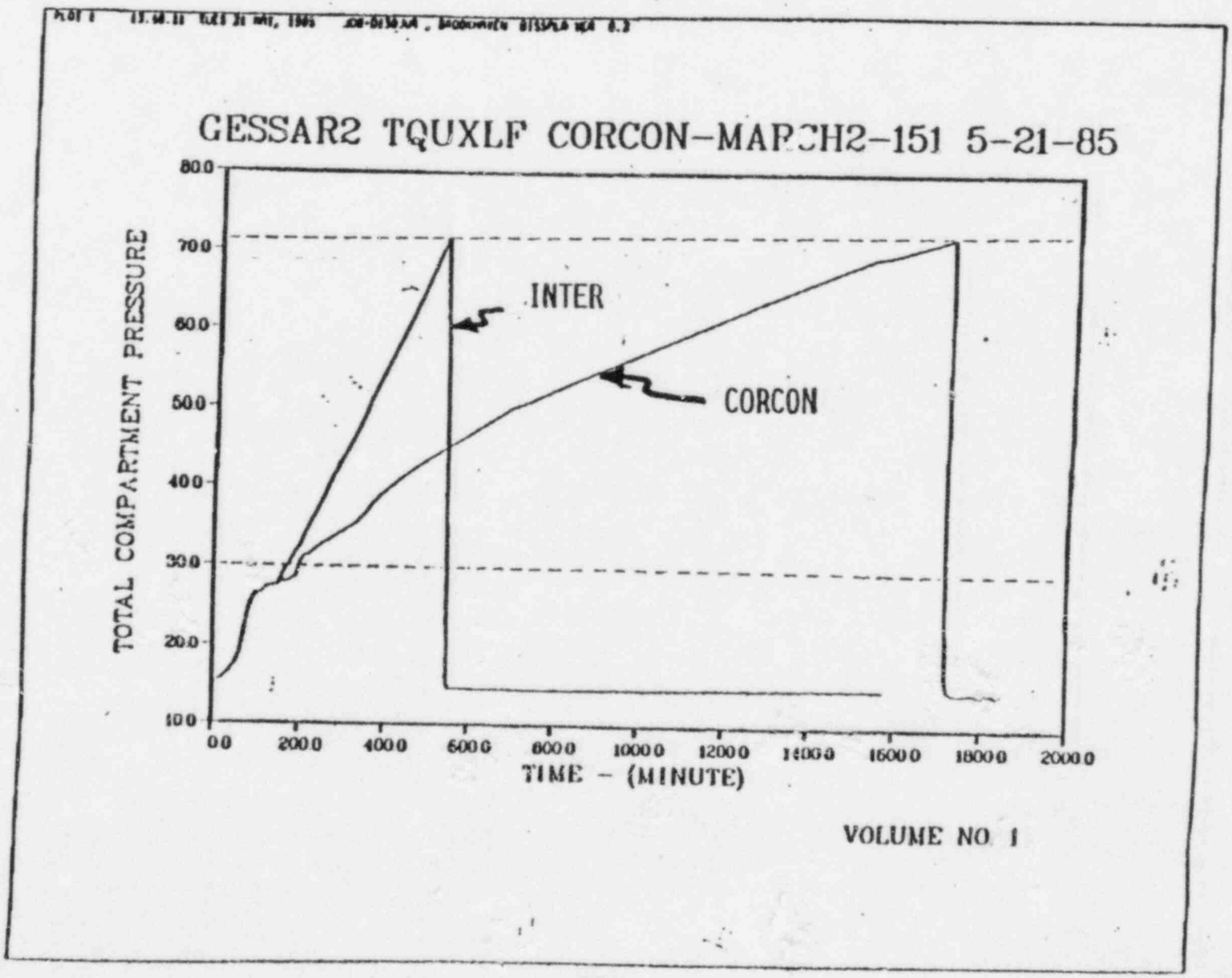


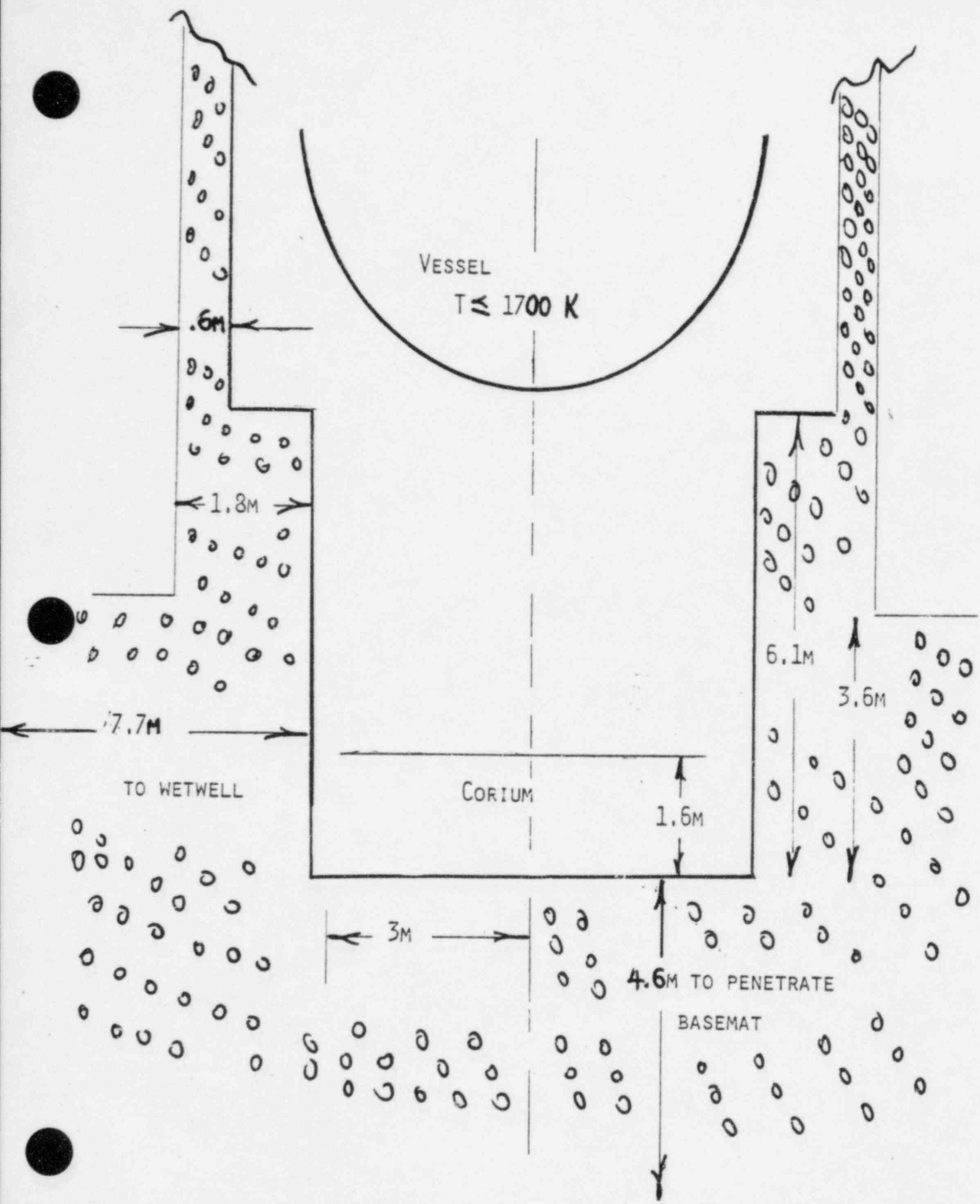
Figure 15.1 Principal features of MARK III containment

A-147



GESSAR2 TQUXLF CORCON-MARCH2-151 5-21-85
TOTAL COMPARTMENT PRESSURE
TIME - (MINUTE)
INTER
CORCON
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
0.0
200.0
400.0
600.0
800.0
1000.0
1200.0
1600.0
1800.0
2000.0
VOLUME NO. 1

A-148


VESSEL
T ≲ 1700 K
.6M
1.8M
7.7M
TO WETWELL
CORIUM
1.6M
6.1M
3.6M
3M
4.6M TO PENETRATE
BASEMAT

ABLATION RATES

$$\dot{q} = \rho \left[C_p(\text{CONCRETE})(T_{\text{ABLATION}} - T_{\text{INITIAL}}) + \lambda_{\text{ABLATION}} \right] \dot{x}$$

$\dot{q}$ = 10 - 20 W/CM^2 (LOWER CAVITY)

$\dot{q}$ = 12 - 3 W/CM^2 (SURROUNDING)

ρ = 2.5 GM/CM^3, C = 1 J/GM/K, λ = 240 J/GM

$\dot{x}_{\text{INITIAL}}$ = ABLATION RATE = 10-20 CM/HR (LOWER CAVITY)

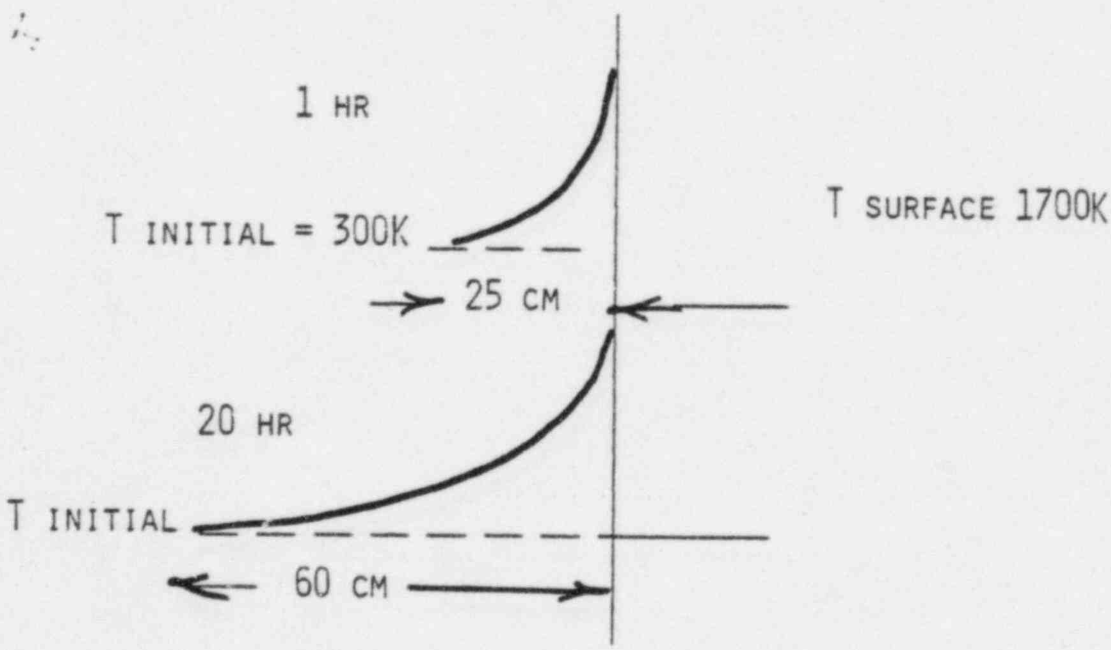
= 12-3 CM/HR (SURROUNDING)

$x_{10 \text{ HRS}}$ $\approx$ 120 CM AXIAL

$\approx$ 140 CM RADIAL, PEDESTAL INTEGRITY DOUBTFUL

THERMAL GRADIENT

$$\nabla^2 T = \alpha \, \partial T / \partial t$$



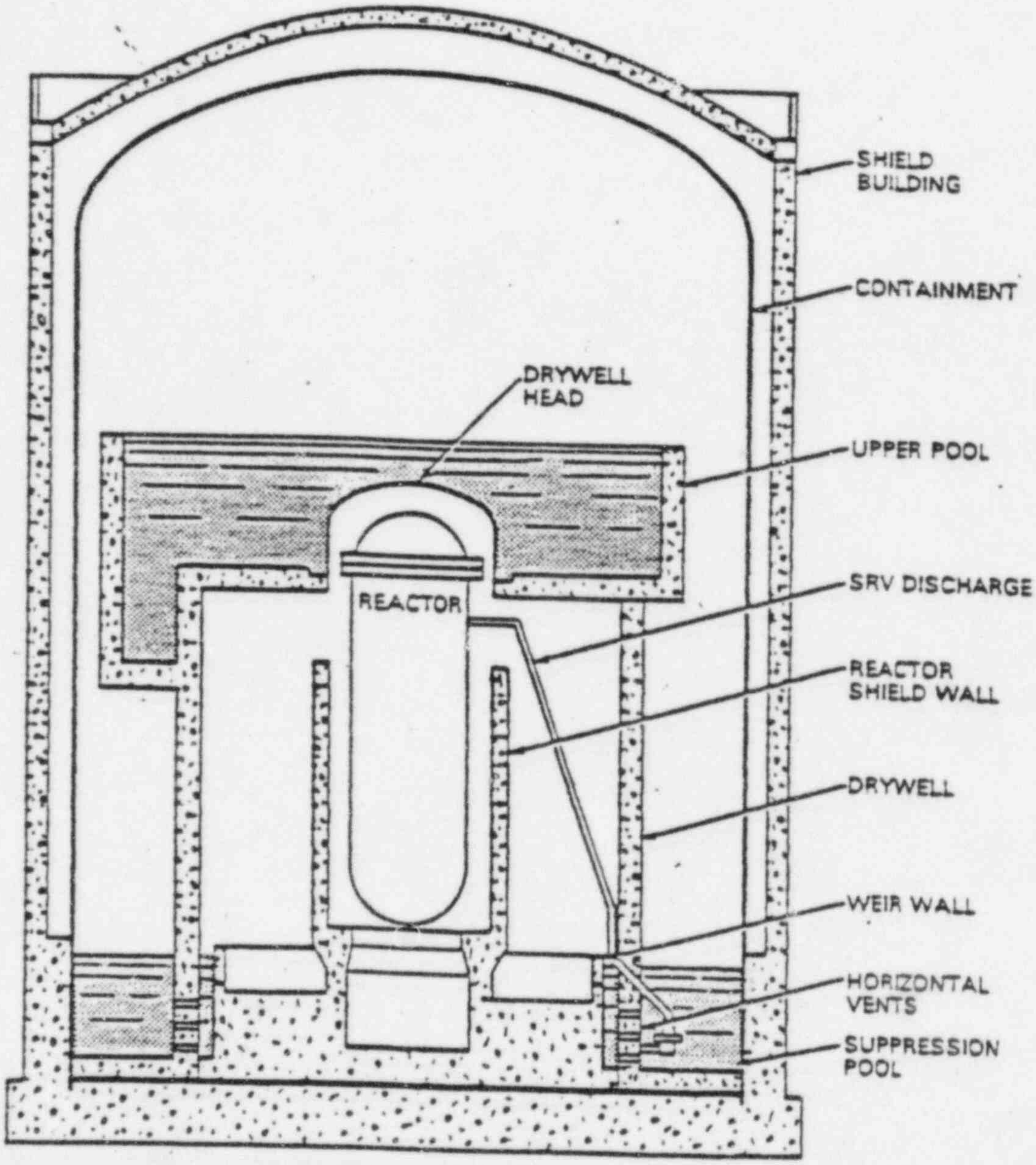


Figure 15.1 Principal features of MARK III containment

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Dominant Containment Failure Modes

- Overpressure due to non-condensible gas generation
- Hydrogen deflagration and detonations
  - failing structure, seals and piping penetrations by pressure and temperature
- Phenomenological issues not considered:
  - steam explosions
  - direct heating
- Mechanical failures not considered:
  - RPV failure
  - Double MSIV failure
  - LBLOCA with Stuck Open Vacuum Breakers
  - Vacuum Breakers Fail Closed
- Mechanical failures considered
  - 1-T-E3, 1-T-L3 releases span range of early and late containment wetwell failure due to potential fabrication flaw
- Excessive Drywell-Wetwell Leakage Considered
  - E1, E2, sequences model total bypass of pool by vaporization release
- Drywell Head Failure
  - I2, I2Q span range

Table 15.1 Conditional consequences predicted by the staff for internally initiated events and probability of occurrence with and without UPPS, per reactor year

Release category*	Early fatality	Early injury	Latent fatality	Person-rems	Probability	
					w/o UPPS	w/UPPS
1-T-L3	0	0	40	7 x E5**	3 x E-6	9 x E-7
1-T-E3	0	0.0005	200	3 x E6	8 x E-6	1 x E-6
1-T-I2Q	0	3	200	3 x E6	1 x E-5	1 x E-6
2-T-B3	0	0	300	5 x E6	4 x E-6	4 x E-7
ATWS	0	1	400	6 x E6	3 x E-6	3 x E-6
1-T-I2	0	6	500	8 x E6	3 x E-6	3 x E-7
1-SB-E1	0.006	10	600	9 x E6	1 x E-9	1 x E-9

*See definitions in Table 15.15.
**7 x E5 = 7 x 10^5.

Notes:

(1) All conditional mean consequences were calculated using the upper range BNL source term values described in SSER 2.

(2) The calculations assumed the Shippingport site, with public evacuation within 10 miles and relocation 12 hours after plume passage.

(3) Mean consequences were computed over 91 different weather conditions.

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CONSIDERATION OF HYDROGEN ISSUES

CURRENT REQUIREMENT

- QUANTITY :100% ZR - H_2O CLAD EQUIVALENT

  78000 LB. ZR

  3400 LB. H_2

- RATE :ACCEPTABLE TO STAFF

GESSAR

Containment Volume (wetwell + drywell)	= 1.4E6 cu. ft.
Mass Air (@ 80F initial)	= 1.05E5 lb.
Mass O_2	= 2.2E4 lb.
Max Mass H_2 that could burn (based on Mass O_2 and assuming complete combustion)	= 2.7E3 lb.
Corresponding Mass Zr oxidized	= 6.2E4 lb.
Fraction clad oxidized to produce sufficient H_2 to burn all O_2 in containment	= 79%
Fraction clad oxidized to produce sufficient H_2 to burn O_2 to 4% lower limit (2.2E3 lb H_2)	= 65%

GESSAR - H_2 RATES

UNMITIGATED SCENARIOS, I.E. TOTAL LOSS OF INJECTION

IN-VESSEL PRIOR TO SLUMP

FOLLOWING EPG WITH SUCCESSFUL ADS

25 LB./MIN. PEAK

500 LB. TOTAL

FAILURE OF ADS WITH 2SRV'S OPEN

85 LB./MIN. PEAK

1400-1700 LB. TOTAL

IN-VESSEL AT CORE SLUMP

~250 LB. H_2 AT ~400 LB./MIN.

EX-VESSEL CCI

2.5 TO 6 LB. H_2/MIN.

40 TO 150 LB. CO/MIN.

CORCON CALCULATIONS

HEAD FAILURE = 150 MIN.

CONT. FAILURE = 1700 MIN.

AT ASSUMED 72 PSIA

4200 LB. H_2

58000 LB. CO

99000 LB. CO_2

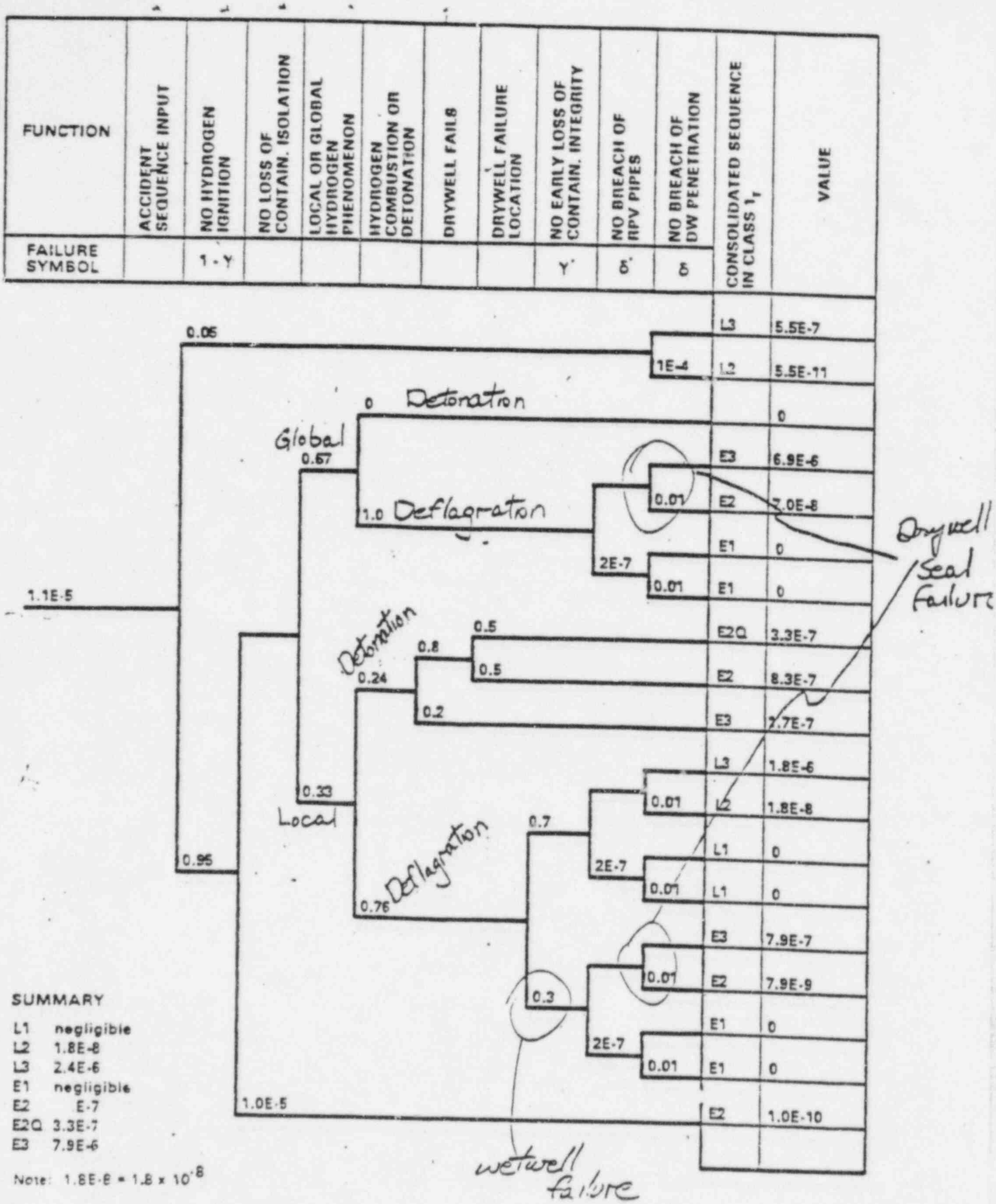


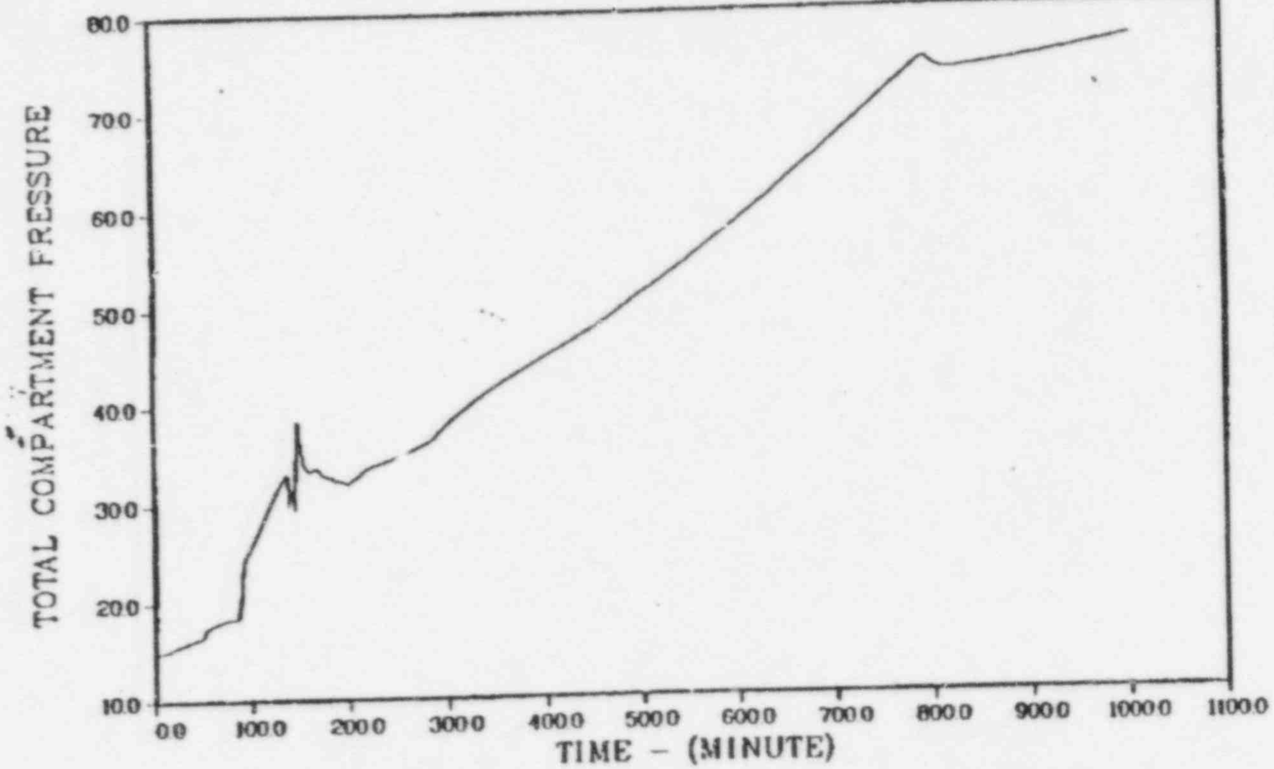
Figure 15.2 CT1-P_a best estimate containment event tree

GESSAR - H_2 THREATS TO CONTAINMENT

ADIABATIC BURN OF ALL O_2 CAN OVERPRESSURIZE
CONTAINMENT (~10 ATM)
1-T-E3 TYPE RELEASE

CONTINUOUS BURN 3000 LB. BURN OVER 1 HR. AT GRAND GULF
~40 PSIA, ~600 F

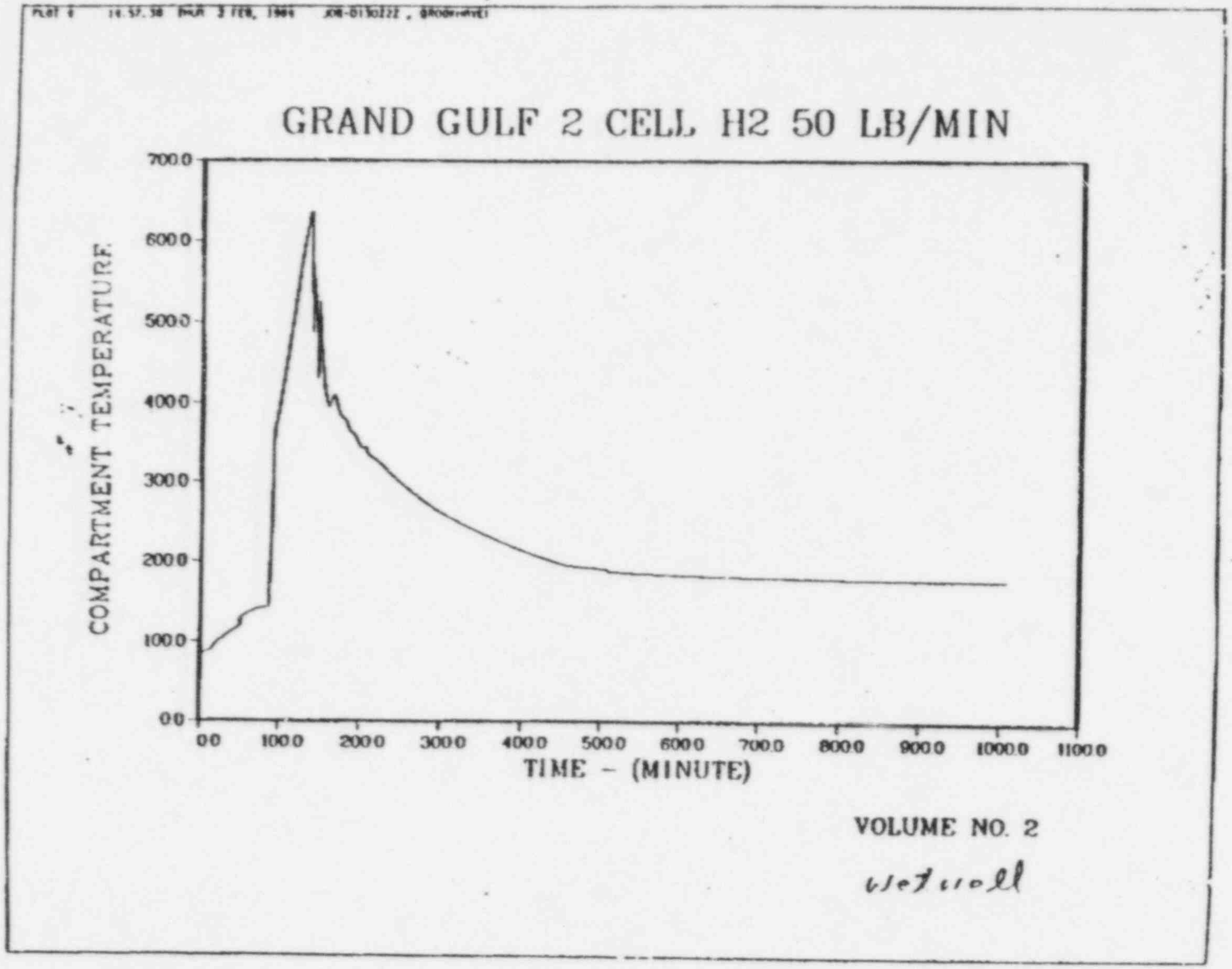
GRAND GULF 2 CELL H2 50 LB/MIN



VOLUME NO. 2

Wetwell

A-160


GRAND GULF 2 CELL H2 50 LB/MIN
COMPARTMENT TEMPERATURE
700.0
600.0
500.0
400.0
300.0
200.0
100.0
0.0
0.0
100.0
200.0
300.0
400.0
500.0
600.0
700.0
800.0
900.0
1000.0
1100.0
TIME - (MINUTE)
VOLUME NO. 2
wetwell

DEFLAGRATION

GLOBAL - NO IGNITORS

- FAILURE OF WETWELL SEAL ASSUMED UNIT PROBABILITY
- SMALL PROBABILITY OF RPV PIPE BREACH OR DRYWELL SEAL FAILURE LEADING TO E1 OR E2 RELEASE, OTHERWISE E3 RELEASE
- ABOUT FACTOR OF 3 IN PERSON-REM CONSEQUENCES

LOCAL - WETWELL SEAL MAY FAIL

- SMALL PROBABILITY OF RPV PIPE BREACH OR DRYWELL SEAL FAILURE LEADING TO E2 OR E3 RELEASE OTHERWISE L3 RELEASE
- ABOUT FACTOR OF 4 IN PERSON-REM CONSEQUENCES

- LOCAL DETONATIONS

  1-SB-E1 PORTRAYS DRYWELL AND WETWELL EARLY FAILURE

  CREDIT FOR PRIMARY SYSTEM RETENTION AND POOL SCRUBBING OF VOLATILES

  PERSON-REM CONSEQUENCES ABOUT AN ORDER OF MAGNITUDE GREATER THAN 1-T-L3

  1 - T - I2, 1 - T - I2Q PORTRAYS DRYWELL HEAD FAILURE DUE TO DETONATION SHOCK LOAD

  ABOUT FACTOR OF 3 IN PERSON-REM CONSEQUENCES DEPENDING ON FAILURE LOCATION

Hydrogen Considerations:

Optimum Ignition Sources

HCOG test program to confirm adequacy of glow plug

Power Source

Diverse Power Source recommended for ignitors

Limitations of Ignition Source

Status of HCOG considerations

NRR COMMENTS ON SAFETY GOAL EVALUATION REPORT

- THOUGHTFUL, WELL WRITTEN.

- ONE SERIOUS DIFFICULTY:

  CORE MELT FREQUENCY GUIDELINE OF 10^{-4}/RY IS TOO LENIENT:

  - 50% CHANCE OF SERIOUS REACTOR ACCIDENT NEXT 20 YRS.; 10% CHANCE OF 2 OR MORE SUCH ACCIDENTS.
  - CONTAINMENT SYSTEMS MUST FUNCTION BEYOND DESIGN CONDITIONS; TOO MUCH RELIANCE ON KNOWLEDGE OF FP BEHAVIOR, CONTAINMENT PERFORMANCE.
  - THE 10^{-4}/RY GUIDELINE NOT WELL DEFINED (I.E., WHETHER IT REPRESENTS SEVERE CORE DAMAGE OR EVEN EXTENSIVE CORE MELT WITH NO RCS PENETRATION, OR EXTENSIVE CORE MELT WITH RCS PENETRATION AND CONTAINMENT CHALLENGE).
  - PROPOSED ALTERNATIVE:

    LARGE-SCALE FUEL & FP RELEASE FROM RCS $< 10^{-5}$/RY

OTHER COMMENTS

- INCLUSION OF AVERTED ON-SITE LOSSES: NRR AGRESS
  - STRENGTHEN SUPPORTING DOCUMENTATION

- PROVISIONAL IMPLEMENTATION GUIDANCE:
  - EXCLUDES SAFETY IMPROVEMENT AT CORE MELT FREQUENCIES OF 3×10^{-5} TO 10^{-3} UNLESS MORTALITY RISK QDOs ARE NOT MET.
  - FAILS TO ADDRESS PRA OMISSIONS, BIASES, EXTENT OF DIFFERENCES BETWEEN ESTIMATES AND QDOs, ETC.

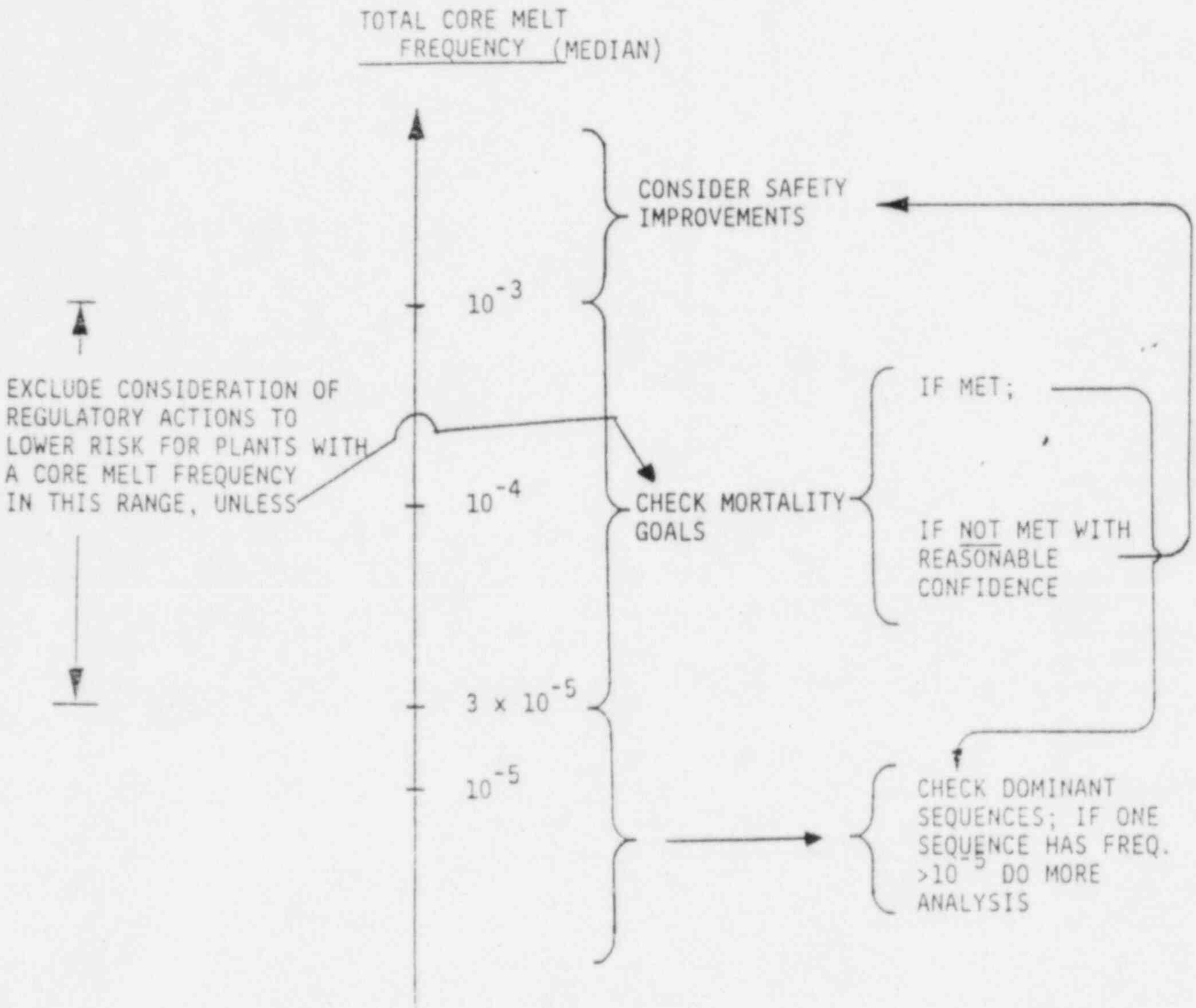
- ALARA DISMISSAL CONFLICTS WITH SEVERE ACCIDENT POLICY, STANDARDIZATION POLICY.

- COMMISSIONER ASSELSTINE'S PROPOSED SAFETY GOAL: REJECTION FLAWED.
  - GREATER EMPHASIS ON PREVENTING TMIs IS WARRANTED: REDUCE CM FREQUENCY GUIDELINE.
  - SHOULD NOT SEEM TO SHIFT PRIMARY SAFETY RESPONSIBILITY FROM LICENSEE TO NRC.

LIKELIHOOD OF CORE MELT

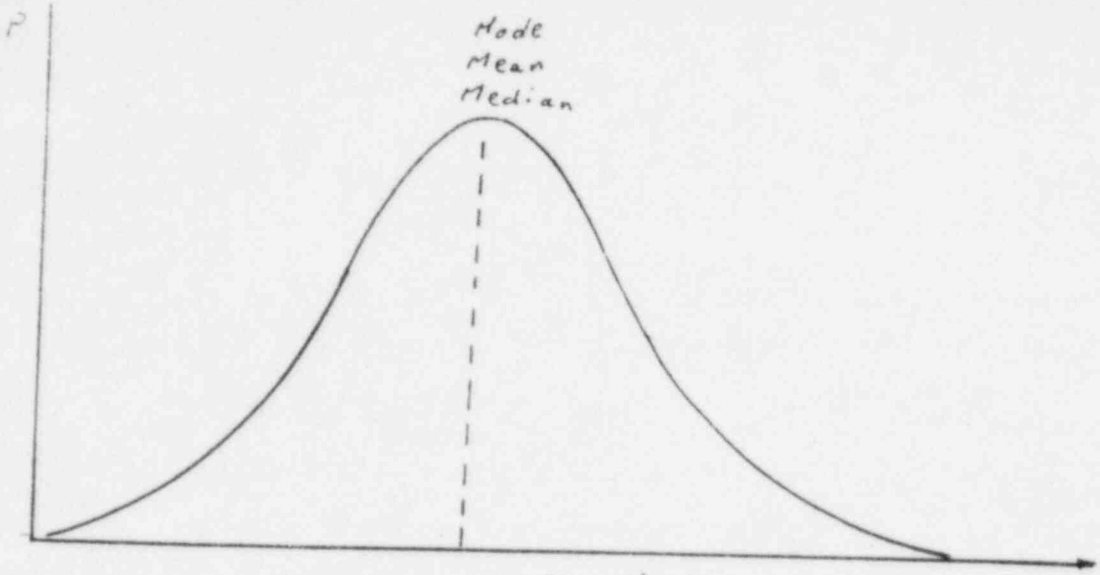
MEAN CORE MELT PROBABILITY (λ)/RY	ACCIDENT PROBABILITY IN NEXT 20 YRS IN POPULATION OF 100 PLANTS, P_A
3×10^{-4}	45%
1×10^{-4}	20%
3×10^{-5}	6%
1×10^{-5}	2%

SAFETY GOAL IMPLEMENTATION DIAGRAM

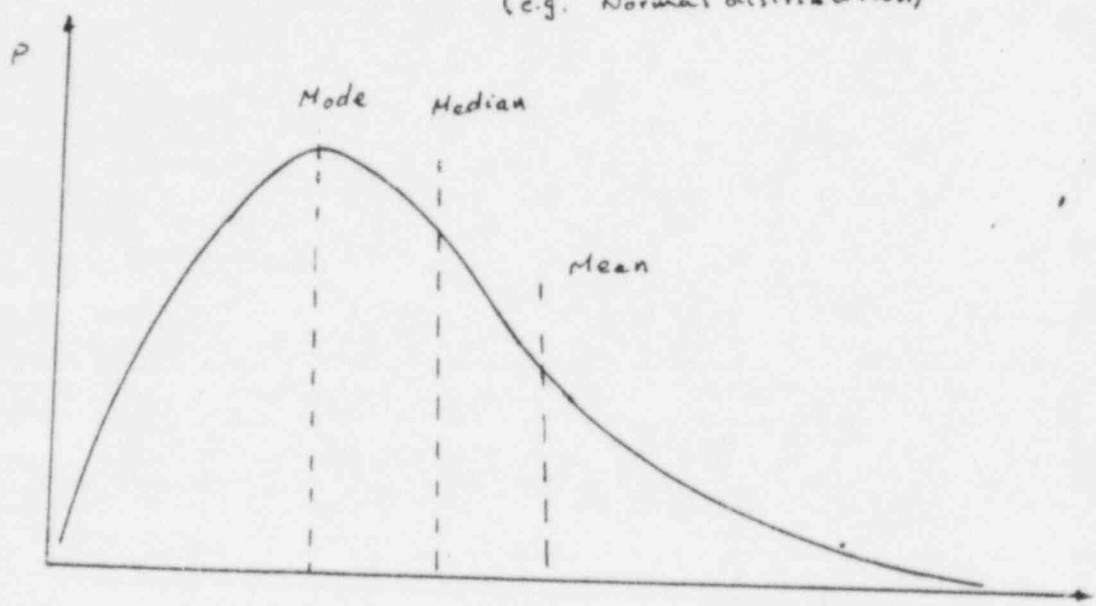


9×10^{-4}

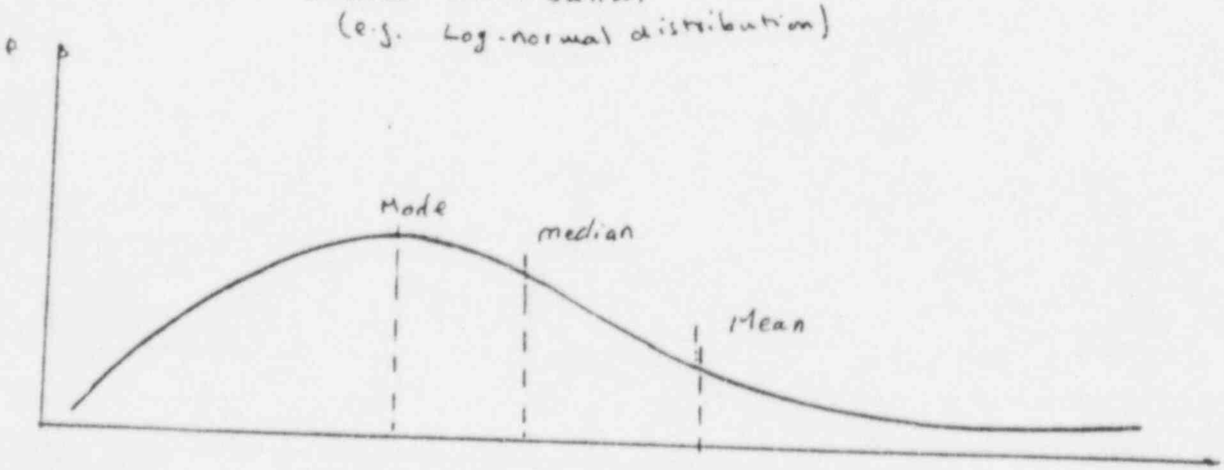
100 SEQ @ 9×10^{-6}



Symmetric Distribution
(e.g. Normal distribution)



Skewed Distribution
(e.g. Log-normal distribution)



Rowsome

NRR COMMENTS ON SAFETY GOAL EVALUATION REPORT

° THOUGHTFUL, WELL WRITTEN.

° ONE SERIOUS DIFFICULTY:

CORE MELT FREQUENCY GUIDELINE OF 10^{-4}/RY IS TOO LENIENT:

- 50% CHANCE OF SERIOUS REACTOR ACCIDENT NEXT 20 YRS.; 10% CHANCE OF 2 OR MORE SUCH ACCIDENTS.

- CONTAINMENT SYSTEMS MUST FUNCTION BEYOND DESIGN CONDITIONS; TOO MUCH RELIANCE ON KNOWLEDGE OF FP BEHAVIOR, CONTAINMENT PERFORMANCE.

- PROPOSED ALTERNATIVE:
  LARGE-SCALE FUEL & FP RELEASE FROM RCS $<10^{-5}$/RY.

° CANCER RISK GUIDELINE IS TOO LENIENT AS SURROGATE FOR SOCIETAL RISK.

- CONSIDER AN AGGREGATE SOCIETAL RISK GOAL.

° INCLUSION OF AVERTED ON-SITE LOSSES: NRR AGREES.

- STRENGTHEN SUPPORTING DOCUMENTATION.

- TREAT AS FAVORABLE COST IMPACTS, TO ARRIVE AT NET COST.

NRR COMMENTS, CONTINUED

- PROVISIONAL IMPLEMENTATION GUIDANCE: TWO PROBLEMS:

  - EXCLUDES SAFETY IMPROVEMENT AT CORE MELT FREQUENCIES OF 3×10^{-5} TO 10^{-3} UNLESS MORTALITY RISK QDOs ARE NOT MET.

  - TOO SIMPLISTIC: FAILS TO ADDRESS PRA OMISSIONS, BIASES, EXTENT OF DIFFERENCES BETWEEN ESTIMATES AND QDOs, ETC.

- ALARA DISMISSAL CONFLICTS WITH SEVERE ACCIDENT POLICY, STANDARDIZATION POLICY.

- COMMISSIONER ASSELSTINE'S PROPOSED SAFETY GOAL: REJECTION FLAWED.

  - GREATER EMPHASIS ON PREVENTING TMIs IS WARRANTED: REDUCE CM FREQUENCY GUIDELINE.

  - SHOULD NOT SEEM TO SHIFT PRIMARY SAFETY RESPONSIBILITY FROM LICENSEE TO NRC.

I DK APPENDIX XV
CONSIDERATION OF POTENTIAL COMPLICATING EFFECTS OF EARTHQUAKES ON EMERGENCY PLN.

UNITE[D STATES]
NUCLEAR REGULATORY COMMISSION SECRETARY
WASHINGTON, D. C. 20555
July 5, 1985

MEMORANDUM FOR: Chairman Palladino
Commissioner Asselstine
Commissioner Bernthal
Commissioner Zech

FROM: William J. Dircks
Executive Director for Operations

SUBJECT: CONSIDERATION OF POTENTIAL COMPLICATING EFFECTS OF EARTHQUAKES ON EMERGENCY PLANNING

The purpose of this memo is to inform you of our progress and direction in preparation of the subject final rulemaking package which I plan to submit to you by early August 1985.

On December 21, 1984, the Commission published a proposed rule change to 10 CFR Part 50 that relates to Emergency Planning and Preparedness at Production and Utilization Facilities (49 FR 49640). The proposed rule stated that neither emergency response plans nor evacuation time analyses need consider the impact of earthquakes which cause or occur proximate in time with an accidental release of radioactive material from a nuclear power reactor.

To date, 61 comment letters have been received. Twenty five (25) letters favored the promulgation of the proposed rule. The letters favoring the proposed rule were from utilities, consulting firms representing utilities, 2 private citizens and the Department of Energy.

Thirty-four (34) letters opposed promulgation of the proposed rule. Many voiced strong displeasure, shock or disbelief at the position the Commission was taking in the proposed rule change. The majority of these letters were from private citizens, and environmental groups.

Additional input was also received from Japan, France, Sweden, Germany and Taiwan, all of which stated that the potential complicating effects of earthquakes were not specifically considered in their nuclear power reactor emergency planning.

Several issues raised in the public comments (and in particular in comments from The Union of Concerned Scientists) will require substantial technical analysis prior to going forward with promulgation of a final regulation. For

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example the staff needs to: (1) assess whether there are sufficient facts to support the staff's belief that the complicating effects of earthquakes on emergency plans are adequately taken into account by the flexibility that exists in all emergency plans; (2) deal with the issue that defects in seismic design and quality assurance in construction can substantially undermine the seismic strength of plant systems and structures; (3) evaluate the limited existing information on the contribution of seismic events to overall core melt risks, recognizing that only a few PRAs assess seismic risks and the treatment entails many uncertainties; (4) deal with the question why emergency plans should not consider the complicating effects of very severe earthquakes (i.e., 2 to 4 times the SSE) whose return frequency is 10E(-4) to 10E(-5) while current emergency plans concern themselves with plant accidents whose estimated return frequency are also in this range. These complex analyses, which are underway, are not expected to be completed before late July, 1985.

After careful review of both the San Onofre and Diablo Canyon decisions regarding the complicating effects of earthquakes on emergency planning, as well as the issues identified above, the staff is considering 3 alternative approaches:

Alternative 1: Adoption of the proposed rule into a final rule with minor but important word changes, for example, "no additional emergency preparedness measures need be established to account for severe, low frequency natural phenomena than is already required in 10 CFR 50.47 and Appendix E."

Alternative 2: Leaving the issue open for adjudication on a case-by-case basis; accomplished by withdrawing the proposed rule or by requiring consideration of earthquakes.

Alternative 3: Promulgation of a final rule which clarifies the original intent of the Commission to require that emergency response plans shall assure that the following capabilities exist relative to the complicating impacts of severe, low frequency natural phenomena.

1. Ability to transport necessary personnel to the plant after the event in order to augment the original staff to cope with degraded modes of plant operation.

2. Ability to obtain damage estimates to the plant and to be able to communicate these estimates to offsite authorities. The information should be available to factor into the decisionmaking process, including recommendations for protective actions after severe, low frequency natural phenomena.

3. Emergency plans for offsite authorities should take into account various degrees and locations of damage to the plant environs. This shall be

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limited to knowing alternate routes of travel as well as establishing criteria for determining whether to shelter, relocate or to evacuate.

Having considered all of the above, as well as all comments received, past operating reactor and emergency preparedness experience, I am leaning toward a recommendation that a final rule be promulgated which would embrace the concepts of Alternative 3. This alternative would be a clarification and articulation of the Commission's original intent as to what is specifically required to assure the necessary flexibility to cope with the complicating effects of severe, low frequency natural phenomena on emergency planning.

William J. Dircks
Executive Director for Operations

cc: SECY
OGC
OPE
M. Cutchin

J Hayes

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ASSESSMENT OF FIELD APPLICATIONS OF

CONTROL ROOM HABITABILITY PRACTICES

APPENDIX XVI - ASSESSMENT OF FIELD APPLICATIONS OF CONTROL ROOM HABITABILITY PRACTICES

CONCLUSIONS

A. CONTROL ROOM SYSTEMS, COMPONENTS, OPERATIONS, PROCEDURES, TECHNICAL SPECIFICATIONS

B. NRC PRACTICES AND POLICIES AND NRC LICENSEES' PRACTICES

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A. 1 LOSS OF VENTILATION AND LOSS OF AIR CONDITIONING EVENTS, WHICH HAVE OCCURRED AT OPERATING REACTORS, SHOULD BE STUDIED AND THEIR POSSIBLE CONTRIBUTION TO THE DEGRADATION OF PLANT SAFETY SHOULD BE EVALUATED

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A. 2 CHANGES TO THE ACTION STATEMENTS AND SURVEILLANCE REQUIREMENTS OF TECHNICAL SPECIFICATIONS SHOULD BE MADE AS NEEDED TO ENSURE THAT THE CONTROL ROOM HEATING, VENTILATION, AND AIR CONDITIONING (HVAC) SYSTEMS SPECIFICATIONS PROVIDE FOR FUNCTIONING AS DESIGNED

A. 2 CHANGES TO THE ACTION STATEMENTS AND SURVEILLANCE REQUIREMENTS OF TECHNICAL SPECIFICATIONS SHOULD BE MADE AS NEEDED TO ENSURE THAT THE CONTROL ROOM HVAC SYSTEMS SPECIFICATIONS PROVIDE FOR FUNCTIONING AS DESIGNED

(1) FLOW RATE SPECIFIED FOR THE PRESSURIZATION TEST

(2) ALLOWABLE LEAKAGE RATES FOR ISOLATION VALVES OR DAMPERS

(3) ASSURANCE THAT FLOW IS NOT OCCURRING THROUGH THE ADSORBER UNIT WHEN THE SYSTEM IS SUPPOSED TO BE ISOLATED

(4) APPROPRIATE LABORATORY CONDITIONS FOR THE TESTING OF ACTIVATED CARBON

(5) SPECIFY THE LOCATIONS OF TEMPERATURE MONITORS AND WHETHER CONTROL ROOM TEMPERATURE LIMITATIONS ARE FOR HUMAN PERFORMANCE OR EQUIPMENT PERFORMANCE AS APPROPRIATE

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A. 4 ALL NRC RELATED CONTROL ROOM HABITABILITY CRITERIA AND THEIR BASES SHOULD BE INCORPORATED INTO ONE DOCUMENT TO ASSIST THE NRC STAFF AND THE APPLICANTS AND LICENSEES IN UNDERSTANDING THE BASES FOR REQUIREMENTS

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A. 5 THE CAPABILITY OF THE REMOTE SHUTDOWN FACILITIES TO BRING THE PLANTS TO COLD SHUTDOWN NEEDS TO BE DEMONSTRATED

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B. 1 THE ADVANTAGES OF CURRENT CONTROL ROOM DESIGNS ARE NOT IMMEDIATELY EVIDENT. ALTHOUGH PRESENT CONTROL ROOM DESIGNS HAVE BECOME MORE COMPLEX, THEY DO NOT NECESSARILY AFFORD THE CONTROL ROOM OPERATORS BETTER PROTECTION THAN THE SIMPLER, OLDER DESIGNS

B. 3 SOME LICENSEES ARE MAKING IMPROPER 10 CFR 50.59 EVALUATIONS

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B. 4 THE UPDATED FSAR DOES NOT REFLECT PRESENT CONTROL ROOM OPERATING SCHEMES

UPDATED FSAR's AND SYSTEM CONFIGURATION CONTROL

THE UPDATED FSAR's FOR THE OPERATING PLANTS DO NOT REFLECT THE ACTUAL AS-BUILT-SYSTEMS, NOR DO THEY ACCURATELY DESCRIBE HOW THE SYSTEMS ARE OPERATED. THE OBSOLESCENCE OF THE FSAR's APPEARS TO BE DUE TO INADEQUATE OR UNTIMELY CONFIGURATION CONTROL.

MODIFICATIONS TO BOTH THE HARDWARE AND THE METHODS OF OPERATING THE SYSTEMS HAVE BEEN MADE WITHOUT COMPLETE CONFIGURATION CONTROL. AREAS LACKING IN CONFIGURATION CONTROL INCLUDE PLANT PROCEDURES, TRAINING MATERIAL, AND PLANT DRAWINGS.

DELETION 1

pages 184- 197

ADDITIONAL DOCUMENTS PROVIDED FOR ACRS' USE

1. Memorandum, C. P. Siess to ACRS Members, Core-Melt Frequency Guideline and Cost-Benefit Analysis, July 11, 1985

2. Memorandum, A. L. Newsom to ACRS Members, NRC Appropriations Bill for FY 1986, July 11, 1985

3. Memorandum, H. W. Lewis to ACRS Members, There is No Way to Measure a Median Failure Rate, July 12, 1985