ACRS-2020
PDR 5-5-83

TABLE OF CONTENTS
MINUTES OF THE
268TH ACRS MEETING
AUGUST 12-14, 1982
WASHINGTON, DC

CERTIFIED

TABLE OF CONTENTS
APPENDIXES TO MINUTES OF THE
268TH ACRS MEETING
AUGUST 12-14, 1982

TABLE OF CONTENTS (Cont.)
APPENDIXES TO MINUTES OF THE 268TH ACRS MEETING

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

August 3, 1982

SCHEDULE AND OUTLINE FOR DISCUSSION
268TH ACRS MEETING
AUGUST 12-14, 1982
WASHINGTON, D.C.

Thursday, August 12, 1982, Room 1046, 1717 H Street, NW, Washington, DC

1) 8:30 A.M. - 8:45 A.M. — ACRS Chairman's Report (Open)
  1.1) Opening statement
  1.2) Items of current interest regarding ACRS activities (PGS/RFF)

2) 8:45 A.M. - 12:30 P.M. — Grand Gulf Nuclear Station Unit 1 (Open)
  2.1) 8:45 A.M.-9:15 A.M.: Report of ACRS Subcommittee on Grand Gulf Nuclear Station Unit 1 (DO/HA)
  2.2) 9:15 A.M.-9:30 A.M.: Report of ACRS Subcommittee on Mark III Containment (MSP/PAB)
  2.3) 9:30 A.M.-12:30 P.M.: Meeting with NRC Staff and Applicant

  Portions of this session will be closed as necessary to discuss Proprietary Information related to this project.

12:30 P.M. - 1:30 P.M. — LUNCH (Note: A videotape showing of the recent TMI-2 core inspection will be shown during the lunch break.)

3) 1:30 P.M. - 1:45 P.M. — Proposed Revision of 10 CFR Part 50.46, Appendix K, ECCS Evaluation Models (Open)
  3.1) Remarks by ACRS ECCS Subcommittee Chairman (MSP/PAB)

4) 1:45 P.M. - 4:15 P.M. — Proposed NRC Nuclear Plant Severe Accident Research Plan (NUREG-0900) and Related Rulemaking (Open)
  4.1) 1:45 P.M.-2:15 P.M.: Report of ACRS Subcommittee on proposed research plan (WK/SKB/GRQ)
  4.2) 2:15 P.M.-2:45 P.M.: Report of ACRS Subcommittee Chairman on proposed NRC policy statement on severe accidents (DO/GRQ)
  4.3) 2:45 P.M.-4:15 P.M.: Meeting with NRC Staff

5) 4:15 P.M. - 5:45 P.M. Nuclear Power Plant Control Room Habitability (Open)

- 5.1) 4:15 P.M.-4:30 P.M.: Report of ACRS Subcommittee regarding control room habitability in nuclear power plants (DWM/RCT)
- 5.2) 4:30 P.M.-5:30 P.M.: Meeting with representatives of NRC staff and the nuclear industry
- 5.3) 5:30 P.M.-5:45 P.M.: Discuss proposed ACRS report/comments to NRC

6) 5:45 P.M. - 6:30 P.M. Proposed ACRS Reports to NRC (Open/Closed)

- 6.1) Discuss proposed ACRS reports to NRC regarding:
  - 6.1-1) Grand Gulf Nuclear Station Unit 1 (DO/HA) (Open)
  - 6.1-2) Ginna Nuclear Power Plant-SEP review (CPS/RKM)(Closed)

7) 6:30 P.M. - 7:00 P.M. Foreign LWR Licensing Practices (Closed)

- 7.1) Report of ACRS Subcommittee Chairman regarding practices used in regulation of foreign light-water reactors (DO)

Friday, August 13, 1982, Room 1046, 1717 H Street, NW, Washington, DC

8)	8:30 A.M. - 11:00 A.M.	Quantitative Safety Goals (Open) 8.1) 8:30 A.M.-9:00 A.M.: Report of ACRS Subcommittee on proposed NRC quantitative safety goals and proposed implementation plan (DO/JMG) 8.2) 9:00 A.M.-11:00 A.M.: Meeting with representatives of NRC Staff and the nuclear industry as appropriate
9)	11:00 A.M. - 12:00 Noon	Proposed NRC Nuclear Plant Severe Accident Research Plan (Open) 9.1) Discuss proposed ACRS report to NRC regarding NUREG-0900, Nuclear Plant Severe Accident Research Plan (Draft) (WK/SKB)
	12:00 Noon - 1:00 P.M.	LUNCH
10)	1:00 P.M. - 1:30 P.M.	Future ACRS Activities (Open) 10.1) Discuss anticipated ACRS activities (MWL) 10.2) Discuss proposed Committee activities (RFF)
11)	1:30 P.M. - 5:30 P.M.	Watts Bar Nuclear Plant Units 1 and 2 (Open) 11.1) 1:30 P.M. - 2:00 P.M.: Report of ACRS Subcommittee regarding OL request for Watts Bar Nuclear Plant (JE/SKB/GRQ) 11.2) 2:00 P.M.-5:30 P.M.: Meeting with NRC Staff and the applicant Portions of this session will be closed as necessary to discuss Proprietary Information related to this project.
12)	5:30 P.M. - 6:15 P.M.	ACRS Subcommittee Activities (Open) 12.1) Report of ACRS Subcommittee Chairman regarding consideration of seismic events in Emergency Planning (DWM/RCT)

Saturday, August 14, 1982, Room 1046, 1717 H Street, NW, Washington, DC

13) 8:30 A.M. - 11:30 A.M.

ACRS Reports to NRC (Open/Closed)

13.1) Discuss proposed ACRS reports to NRC regarding:
- 13.1-1) Watts Bar Nuclear Plant - OL (JCE/SKB/GRQ)(Open)
- 13.1-2) NUREG-0900, Nuclear Plant Severe Accident Research Plan (WK/SKB/GRQ)(Open)
- 13.1-3) High Level Radioactive Waste Disposal (10 CFR Part 60) (DWM/RCT)(Open)

13.2) Complete ACRS reports regarding:
- 13.2-1) Ginna Nuclear Power Plant (CPS/DCF)(Closed)
- 13.2-2) Grand Gulf Nuclear Plant - Outstanding OL issues (DO/HA) (Open)
- 13.2-3) Nuclear Power Plant Control Room Habitability (DWM/RCT) (Open)

14) 11:30 A.M. - 11:45 A.M.

Activities of ACRS Members (Open)

14.1) Participation of ACRS member on ANS sponsored panel to review nuclear power plant accident source term (WK/RFF)

14.2) Invitation of ACRS member to participate in panel discussion sponsored by IAEA regarding TMI-2 Impacts and Improvements (WK/RFF)

15) 11:45 A.M. - 12:00 Noon

ACRS Subcommittee Activities (Open)

15.1) Report of ACRS Subcommittee Chairman regarding preparation of proposed NRC Long Range Research Program Plan (CPS/SD)

12:00 Noon - 1:00 P.M. LUNCH

16) 1:00 P.M. - 2:30 P.M.

ACRS reports to NRC (Open/Closed)

16.1) Complete preparation of ACRS reports as needed system fill after refueling or plant startup.

NRC Inspection and Enforcement Information Notice No. 82-17 ("Overpressurization of Reactor Coolant System") was issued to other licensees informing them of these events and their potential significance.

Dated at Washington, D.C. this 30th day of July 1982.

John C. Hoyle,

Acting Secretary of the Commission.

[FR Doc. 82-21199 Filed 8-4-82; 8:45 am]

BILLING CODE 7590-01-M

Advisory Committee on Reactor Safeguards; Addition to Agenda of Meeting

In accordance with the purposes of Sections 29 and 182b. of the Atomic Energy Act (42 U.S.C. 2039, 2232b.), a meeting of the Advisory Committee on Reactor Safeguards has been scheduled for August 12-14, 1982, in Room 1046, 1717 H Street, NW., Washington, D.C. The agenda for this meeting has been changed by adding the following item:

Thursday, August 12, 1982

6:30 p.m.-7:00 p.m.: Foreign Light-Water Reactor Licensing Practices

(Closed)—The Committee will discuss information concerning the practices used in regulation of foreign light-water nuclear reactors.

This session will be closed to discuss information provided in confidence by a foreign source.

I have determined, in accordance with Subsection 10(d) Pub. L. 92-463, that it is necessary to close the portion of the meeting noted above to discuss information provided in confidence by a foreign source (5 U.S.C. 552b(c)(4)).

All other items regarding this meeting remain the same as announced in the Federal Register published Wednesday, July 28, 1982 (47 FR 32668).

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted can be obtained by a prepaid telephone call to the ACRS Executive Director, Mr. Raymond F. Fraley (telephone 202/634-3265), between 8:15 a.m. and 5:00 p.m., EDT.

Dated: July 30, 1982.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 82-21[illegible] Filed 8-4-82; 8:45 am]

BILLING CODE 7590-01-M

Advisory Committee on Reactor Safeguards, Subcommittee on Clinch River Breeder Reactor, Working Group on Structures and Materials; Meeting

The ACRS Subcommittee on Clinch River Breeder Reactor (CRBR) working Group on Structures and Materials will hold a meeting on August 18 and 19, 1982, Room 1046, 1717 H Street, NW, Washington, DC. The Subcommittee will discuss elevated temperature design, "leak before break" criteria, overall leakages, leak detection, inservice inspection, steam generator design, testing and analysis, overall structural integrity of transition joints, containment buckling analysis, and compartment analysis. Notice of this meeting was published July 20.

In accordance with the procedures outlined in the Federal Register on September 30, 1981 (46 FR 47903), oral or written statements may be presented by embers of the public, recordings will be permitted only during those portions of the meeting when a transcript is being kept, and questions may be asked only by members of the Subcommittee, its consultants, and Staff. Persons desiring to make oral statements should notify the cognizant Federal Employee as far in advance as practicable so that appropriate arrangements can be made to allow the necessary time during the meeting for such statements.

The entire meeting will be open to public attendance except for those sessions during which the Subcommittee finds it necessary to discuss proprietary, industrial security and/or Unclassified Safeguards information. One or more closed sessions may be necessary to discuss such information. (Sunshine Act Exemptions 3 and 4.) To the extent practicable, these closed sessions will be held as as to minimize inconvenience to members of the public in attendance.

The agenda for subject meeting shall be as follows:

Wednesday, August 18, 1982—8:30 a.m. until the conclusion of business

Thursday, August 19, 1982—8:30 a.m. until the conclusion of business

During the initial portion of the meeting, the Subcommittee, along with any of its consultants who may be present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittee will then hear presentations by and told discussion will representatives of the Department of Energy, NRC Staff, their consultants, and other interested persons regarding this review.

Further information regarding topics to be discussed, whether the meeting has cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by prepaid telephone call to the cognizant Designated Federal Employee, Mr. Gary Quittschreiber or the Staff Engineer, Mr. Anthony Cappucci (telephone 202/634-3267) between 8:15 a.m. and 5:00 p.m., EDT.

I have determined, in accordance with Subsection 10(d) of the Federal Advisory Committee Act, that it may be necessary to close some portions of this meeting to protect proprietary, industrial security and/or Unclassified Safeguards information. The authority for such closure is Exemption (3) and (4) to the Sunshine Act, 5 U.S.C. 552b(c)(3),(4)./

Dated: July 30, 1982.

John C. Hoyle,

[FR Doc. 82-21021 Filed 8-4-82; 8:45 am]

BILLING CODE 7590-01-M

[Docket Nos. STN 50-454 OL and STN 50-455 OL]

Commonwealth Edison Co. (Byron Station, Units 1 and 2); Prehearing Conference

August 2, 1982.

Please take notice that pursuant to the order in this proceeding on July 28, 1982, a prehearing conference will commence on August 18, 1982 at 9:00 a.m. local time in the Federal Building, Room 280, 211 South Court Street, Rockford, Illinois 61101.

It is so Ordered.

Dated at Bethesda, Maryland this 2nd day of August 1982.

For the Atomic Safety and Licensing Board.

Morton B. Margulies,

Chairman, Administrative Judge.

[FR Doc. 82-21332 Filed 8-4-82; 8:45 am]

BILLING CODE 7590-01-M

[Docket No. 50-295]

Commonwealth Edison Co.; Issuance of Amendment to Facility Operating License

The Nuclear Regulatory Commission (the Commission) has issued Amendment No. 75 to Facility Operating License No. DPR-39, issued to the Commonwealth Edison Company (the licensee), which revised Technical Specifications for operation of Zion Station, Unit 1 (the facility) located in Zion, Illinois. The Amendment was was effective on June 26, 1982.

The amendment provides a one-time change to the Technical Specifications to allow one safety injection pump to be

Issue Date:

January 19, 1983

MINUTES OF THE
268TH ACRS MEETING
AUGUST 12-14, 1982
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CERTIFIED

The 268th meeting of the Advisory Committee on Reactor Safeguards, held at 1717 H Street N.W., Washington, DC, was convened by Chairman P. G. Shewmon at 8:30 a.m., Thursday, August 12, 1982.

[Note: For a list of attendees, see Appendix I. J. J. Ray and R. C. Axtmann were not present for the meeting. M. Bender, H. W. Lewis and D. Okrent were unable to attend on Saturday. M. S. Plesset was not present on Friday or Saturday.]

The Chairman 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 noted that no requests had been received from members of the public to present either oral or written statements to the Committee. 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 St. N.W., Washington, DC.

[Note: Copies of the transcript taken at this meeting are also available for purchase from the Alderson Reporting Company, Inc., 400 Virginia Ave. S.W., Washington, DC 20024.]

I. <u>Chairman's Report</u> (Open to Public)

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

The Chairman reported to the Committee that an exemption to the LWA-1 had been approved by the Commission for the Clinch River Breeder Reactor and site preparation has begun. He indicated that the Department of Energy plans to make application for an LWA-2 which will impact on some safety related features of the plant.

II. <u>Grand Gulf Nuclear Station Unit 1</u> (Open to Public)

A. <u>Report of the ACRS Subcommittee on Grand Gulf</u>

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

D. Okrent, Subcommittee Chairman, indicated that the objective of this meeting is to complete the review for an operating license for the Grand Gulf Nuclear Station. He reminded the Committee of at least one open question regarding hydrodynamic loads on structures and components above the suppression pool which had led to the ACRS characterizing its review in October, 1981 as an interim one. He mentioned the concerns of a former employee of General Electric Company (J. Humphrey) regarding several detailed questions concerning matters pertaining to the suppression pool. He indicated that the Subcommittee meeting held on August 11, 1982 had dealt with hydrogen control, management structure, technical capability, and questions concerning the single failure criterion. He indicated that some review of quality assurance and quality control was presented from I&E reports.

D. Okrent noted a question by J. Ebersole concerning pipe failure in the drywell which might lead to failure of hydraulic lines for actuation of the control rods. Such a failure might prevent safe shutdown of the facility following the original initiating pipe break. W. Kerr requested that the Applicant explain how they determine the proper source term to use for calculation of offsite doses in an actual emergency.

B. Report of ACRS Subcommittee on Mark III Containment

[Note: P. A. Boehnert was the Designated Federal Employee for this portion of the meeting.]

M. S. Plesset referred to the Fluid Dynamics Subcommittee Meeting of July 29-30, 1982 at which a large number of concerns from J. Humphrey were reviewed. He indicated that most of the concerns were of a second order, with the exception of a concern regarding the hydraulic lines for the control rods in the drywell. He noted that the Staff had approved a structural analysis for the hydraulic control units resolving the issue (see Appendix IV).

D. W. Moeller requested clarification of a statement in the subcommittee minutes referring to changes made in the Grand Gulf plant to correct deficiencies of the same type as those which led to the Browns Ferry partial failure to scram. M. S. Plesset explained that this was part of a general discussion of the interface between the designer of the nuclear island (General Electric Company), the architect engineer, and the Applicant. General requirements for the scram system design were proposed by GE and the detailed design of the installation was made by the architect engineer. A. Smith, GE, explained that GE criteria were not properly implemented by the architect engineer.

D. W. Moeller noted that GE does not audit implementation of their recommended design requirements but relies on the architect engineer and the applicant to implement them.

C. NRC Assessment of the Status of Review

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

D. Houston, NRC, presented a chronology of the Grand Gulf licensing effort from the date of the Safety Evaluation Report, September 9, 1981, to SER Supplement No. 3, issued on July 21, 1982 (see Appendix V). He noted that the second supplement to the SER supported low-power licensing and described the resolution of structural and containment loads including the LOCA loads on the hydraulic control unit (HCU) floor (see Appendix V). The third supplement presented resolution for the loads on equipment on the HCU floor and addressed resolution of hydrogen control.

D. Houston briefly reviewed the status of outstanding issues including LOCA loads. Electrical equipment qualification and containment purge have been resolved with license conditions. He pointed out that an outstanding issue concerning management capability and organization was resolved with a license condition defining the duties of the operating shift advisor, the advisor to the corporate management, the training instructors, and the corporate safety review group. In answer to a question by W. Kerr, D. Houston indicated that the Humphrey concerns about the containment (which came up after the ACRS interim review) were evaluated by the Staff to determine whether they were of major or minor concern. D. Houston reviewed each of five issues introduced since the last ACRS meeting (see last page of Appendix V).

J. McGaughy, MP&L, noted that the plant was about to go to first criticality the weekend of August 14-15 and start power testing. He indicated that MP&L is in the process of installing the vibration monitoring system for the prototype core to do vibration monitoring.

D. Hydrogen Control Presentation by NRC

C. Tinkler, Containment Systems Branch of NRC, indicated that the internal evaluation of Grand Gulf's hydrogen ignition system was performed to determine the effectiveness of the system in controlling consequences of hydrogen releases from a TMI-type degraded core accident to prevent breach of containment and allow safe shutdown (see Appendix VI). W. Kerr asked what the significance of a TMI-type degraded core accident was as contrasted with other accidents where hydrogen is released. C. Tinkler indicated that the Staff interim evaluation considered the accident sequences chosen by MP&L as the basis for evaluation of the hydrogen ignition system without consideration of other accidents such as steam release breaks and other sequences which would have net hydrogen emissions. C. Mark asked what the basis was for thinking that one knows the rate of hydrogen emission at TMI-2. C. Tinkler indicated that it is more a belief that the Staff has some idea of the upper bounds of the hydrogen release. In answer to a question by D. W. Moeller, C. Tinkler indicated that the basis for evaluating the hydrogen ignition system was the testing and analysis performed and referenced by MP&L as augmented by the Staff's confirmatory analysis and testing. He indicated that the Staff's conclusion was that the hydrogen ignition system was found adequate on an interim basis, conditional on the successful qualification of the igniter assemblies which is expected to be completed this August. The Committee discussed other topics to be explored for the final review including investigation of combustion phenomena pertinent to a Mark III containment.

S. Hobbs, MP&L, responded to a question by D. W. Moeller concerning operation of the ignition system by explaining that the system is controlled by switches in the control room and indicator lights provide direct indication that the hydrogen ignition system is in operation.

S. Hobbs briefly discussed the hydrogen system design and qualification, base case selection, equipment survivability, structural capability, and the testing program (see Appendix VII). He explained that hydrogen release rates were taken from the MARCH computer code which was used to run analyses of several cases of hydrogen burns, including a small break in the drywell, as well as a stuck open safety relief valve. The most severe thermal environment that resulted from these cases was the wetwell burn which was used as a basis for an equipment survivability program for all components regardless of where they were located. S. Hobbs indicated that MP&L is active in the Hydrogen Controls Owners Group. On a generic basis with the Owners Group, MP&L is entering into a test program to confirm the analytical assumptions that have been made in evaluating the performance of containment response resulting from burns from the hydrogen ignition system.

E. <u>BWR Experience in Operations and MP&L Technical Support Organization</u>

F. Lewis, Chairman and President of Middle South Utilities, explained the consolidation of operating companies within Middle South Utility System, such that every corporate unit in the system will have some direct involvement with nuclear. He discussed the creation of a System Nuclear Oversight Committee including its planned functions (see Appendix VIII).

J. McGaughy briefly described the MP&L corporate organization (see Appendix IX). He pointed out certain changes since the interim review in October, 1981. The Manager of Quality Assurance now reports directly to the Senior Vice-President, N. L. Stampley. There is no Project Manager for Unit 2 with the Manager of Quality Assurance having responsibilities for both of the projects. J. McGaughy updated the status with regard to the Staffing in MP&L with professional people.

C. McCoy, Plant Manager at Grand Gulf, presented details of the commercial BWR experience of Grand Gulf personnel. MP&L increased the authorized level of people in operations, chemistry and radiation protection, technical support, training, instrument and control. He added that MP&L had made significant commitments to increase inhouse capability, reduce the reliance on contractors, and reduce turnover with adequate preparation to handle attrition (see Appendix X). C. McCoy mentioned that MP&L was concentrating on the area of procedure adherence. He noted that the volume of procedures in the industry is increasing drastically, pointing out that there are nearly 7000 procedures at Grand Gulf. He indicated that the larger number of procedures has led to much stronger procedural control of the maintenance work at the plant, including both preventive and corrective maintenance.

D. Okrent asked the NRC Staff for its appraisal of the adequacy of BWR operating experience at Grand Gulf. R. Benedict, Licensing Qualifications Branch, indicated that the Staff does not have any particular concern with regard to the manning of the shifts. He indicated that earlier problems have been concerned more with the operating expertise in middle and upper management concerns which have been assuaged by the consultants and contractors that MP&L has hired. He noted that loss of the Assistant Plant Manager by Grand Gulf represented the loss of a major portion of the BWR operating experience in their Plant Operations Department.

F. <u>Possible Effects of LOCA on Hydraulic Lines Affecting Scram Capabilities</u>

D. Terao, NRC Mechanical Engineering Branch, explained the concern to be a control rod drive (CRD) piping bundle routed very close to high energy reactor recirculating piping. A pipe break in the recirculating piping might impair the scram function by damaging the CRD piping from jet impingement forces or pipe whip. Grand Gulf proposed to address the fatigue loads and the high stresses in the piping in accord with Branch Technical Position NED 3-1. He noted an MP&L letter submitted on April 27, 1982 which stated the results of their analyses regarding this matter (see Appendix XI). Based upon their analysis, the Staff closed the issue. D. Okrent asked whether the Staff had evaluated an accident involving a pipe rupture or medium to large LOCA and loss of a sufficient number of CRD lines to prevent shutdown of the plant if it were reflooded. D. Novak, NRC Staff, indicated that the logic applied by the Staff is that if there is sufficient time for operator action, then the likelihood of an accident progressing to a point where one is unable to drive in a sufficient number of control rods to shutdown the reactor is low enough that it need not be specifically analyzed. In other words, he continued, it is reasonable to assume that proper actions would be taken such that the scenario could be aborted early in the event.

J. Richardson, MP&L, indicated that if one severed all of the CRD insert and withdrawal lines and relied only on reactor pressure, you would insert the control rods within three to four seconds. When J. Ebersole reminded him that the use of reactor pressure depended upon the characteristics of the LOCA, J. Richardson indicated that the reactor would be above 1000 lbs. pressure for the first 5 seconds and above 600 lbs. for a considerably longer length of time, quite sufficient to insert the rods. J. Ebersole expressed concern that inserting the control rods by use of a degrading system pressure could be a random effect, with some rods effectively going in while others would not because of decay of the reactor pressure. The ultimate consequence of reflooding would involve the reactor going critical again.

J. Ebersole indicated that the answer he had received from the Applicant was not sufficient. M. Bender suggested that the probability of all the circumstances in this event coming together is sufficiently low so that one would not have to be concerned. D. Okrent proposed that the Committee develop a memorandum to the Staff asking that they look at this accident probabilistically.

G. Proposed Venting of Containment in the Event of Buildup of Pressure as a Result of a Severe Accident

D. Okrent referred the Committee to a June 15, 1982 letter response from MP&L to the NRC Staff regarding containment venting (see Appendix XII). S. Hobbs presented a chronology of post-accident containment venting (see Appendix VII). He indicated that the emergency procedure guidelines, developed largely prior to the containment ultimate capacity analyses, included the option to allow containment venting at design pressure. The emergency procedures would allow the operator and the shift technical advisor and shift advisor the option under the appropriate circumstances of venting at pressures in excess of 15 lbs.

H. Committee Discussion

W. Kerr asked the Applicant for methods used during the course of an accident to determine the source term to predict possible offsite doses. K. McCoy mentioned a Grand Gulf emergency procedure called Dose Assessment which directed the operators at the plant to determine the releases and to make recommendations to the state and local governments for protective actions. The preferred source term is an accident monitoring system that reads the gas release rate and iodine release rate in the standby gas treatment system discharge. If this source term is not available, the source term is based on a study in the FSAR which assumes 100% release of noble gases and 25% release of iodine from the core in the worst case with a .35% containment volume per day release. Since the first 100 seconds of release is not processed through the standby gas treatment, the worst case source term is used initially. W. Kerr expressed concern that MP&L could not be certain that a leak from the containment to the outside would go through the gas treatment system stack. J. McGaughy indicated that the containment is completely surrounded by this system and that there was no path from the containment directly to the outside without passing through the standby gas treatment system. He indicated that all of the release points are monitored with a high rate monitoring system. In addition, he indicated that there is a high radiation monitoring team taking samples around the plant and in the plume to check out the monitors. K. McCoy indicated that current procedures call for a more accurate source term now because MP&L has high-range accident monitoring in the exhaust. J. Richardson, MP&L, indicated that from his knowledge most of the LERs and failures regarding hydrogen monitors refer to hydrogen monitoring systems that BWRs have to monitor offgas systems and involve a complicated chemical process. The monitor referred to in the hydrogen control presentation is a thermal conductivity device which is much more reliable and has not been associated with many failures. D. W. Moeller asked the NRC Staff to provide a response regarding the reason why MP&L is using a different kind of monitoring instrument for hydrogen accident monitoring within the containment than in the offgas system.

M. S. Plesset requested that the Applicant provide a copy of the analysis being performed concerning the effect of intrusion into the air space above the wetwell when the rising bubble is breaking through.

J. Ebersole asked the Applicant for a status report on consideration of the cleanliness specifications for the fluid conditions for its RHR core spray pumps, especially regarding the seals and bearings. He asked what MP&L has done to assure that these pumps and seals will run for months during a post-accident cooldown situation. H. Townsend, GE, indicated that hydrocyclone-type filters are used to take out the large particles that might be present in the RHR flows. He added that the RHR pumps are deepwell submersible-type pumps normally designed for irrigation-type service under conditions of grit and sand-type flows. J. Ebersole questioned whether GE had ascertained the effectiveness of these hydrocyclone filters to remove the contaminants, particularly whether these contaminants are heavier than water so they can be removed by centrifugal force or lighter than water and whether or not they can be removed by the filters. He asked if GE had compiled a contaminant list and suggested that these hydrocyclone filters might be an ultimate filter which would act as collectors of whatever contaminants were in the stream and suddenly release these contaminants directly to the intake of the RHR pumps.

T. Novak, NRC Staff, indicated that the Staff does not have a generic solution to this problem but addresses it on a case by case basis. M. Bender suggested that the NRC Staff investigate this issue of the recirculation of contaminants.

J. Ebersole asked MP&L if there are any automatic electrical transfers that challenge the last critical supply source in the 1E d.c. systems. J. Richardson indicated that there are no transfer type devices in the d.c. system from one supply to the other. J. Ebersole asked if major electrical boards have multiple d.c. buses inside them which have been condemned because of their potential for cascading to a terminal failure if the d.c. system fails.

III. OL Review of Watts Bar Nuclear Plant Units 1 and 2 (Open to Public)

[Note: J. R. Quittschreiber was the Designated Federal Employee for this portion of the meeting.]

A. Report of ACRS Subcommittee

J. Ebersole, Chairman of the Watts Bar Subcommittee, pointed out that there are many similarities between Watts Bar and TVA Sequoyah Plant. He indicated that even though the designs are essentially the same with some improvements at Watts Bar, there are some differences that should be reviewed including a serious quality assurance breakdown, principally in the construction and in design areas at Watts Bar, which resulted in a significant number of deficiency reports. He indicated that the plant is located essentially in the same seismic area as Sequoyah. It has quite similar hydrology problems. Watts Bar has a flooding-type design which resembles that of Sequoyah which permits the plant to be flooded after a prescribed interval of time which they think they can forecast. J. Ebersole pointed out a particular aspect of the plant which should be of interest to the ACRS, the use of cement mortar to line the coolant water system piping and the change in small diameter pipe from carbon steel to stainless steel. He pointed out that mortar-lined pipes represent a potential cascade failure in the event of seismic shocks or other mechanical upsets with sudden entrainment of debris which could cause degradation in the performance of piping systems.

B. TVA Presentation of Site and Plant Description

L. Mills, Manager of Nuclear Licensing, TVA, read a prepared presentation (see Appendix XIII). D. Ormsby, Licensing Project Engineer, Watts Bar, discussed similarities between the Sequoyah and Watts Bar designs (see Appendix XIV). He indicated that most differences between the two plants were either site specific or the result of the fact that two or three years separated the design phases. The design philosophy was the same in both plants but in some instances more current technology was used for Watts Bar. D. Ormsby cited examples of differences between the two plants such as increased primary system flow rate and differences in maximum heat flux for Watts Bar vs. Sequoyah. He noted an increase in turbine generator and gross electrical output for Watts Bar due to increased equipment and system efficiency, besides the difference in steam generators - the Westinghouse D steam generator for Watts Bar and the Model 51 in Sequoyah. In answer to a question by C. Mark, D. Ormsby noted that there is more qualification documentation regarding the PORVs for the Watts Bar than currently exists for the Sequoyah valves such that the expectation would be that the Watts Bar PORVs are at least as good or better than the Sequoyah PORVs.

R. Graves, TVA, addressed the question of interdependency between the Watts Bar Hydro Plant and the Watts Bar Nuclear Plant. He indicated that the Watts Bar Hydro Plant feeds power into the 161 Kv grid which is the source of two circuits that supply offsite power to the shutdown boards inside the Watts Bar Nuclear Plant. The shutdown boards inside the Plant each have a diesel generator as an additional source of power. He added that the 161 Kv grid is also interconnected with the entire TVA grid, and Watts Bar Hydro is not essential for the reliable operation of the shutdown system in the Watts Bar Nuclear Plant.

W. Cottle indicated that the Athens-Sequoyah line going out of Watts Bar is one of TVA's primary feeders. If this line is down, TVA has implementing procedures for special preference from the hydro plant in terms of splitting the bus and dedicating that power basically to the Watts Bar unit to prevent a failure in the hydro switchyard from further degrading loss of offsite power.

C. Outstanding Issues Presented by the NRC Staff

T. Kenyon, NRC Staff, presented the licensing status of the Watts Bar Plant (see Appendix XV). He noted only some of the Watts Bar open items were open items on Sequoyah. Some of the open items are very site specific to Watts Bar and did not come up on Sequoyah, such as the potential for liquefaction under the ERCW pipelines. Some of them are due to the normal evolution of the licensing process since it has been about two years since the Sequoyah Unit 1 was licensed. J. Ebersole expressed interest in the open items referring to the potential for liquefaction under the ERCW pipelines and Class-1A electrical conduit where these critical waterlines and the corresponding electrical services are in an earth-filled causeway surrounded by steel pilings retained by deadmen supports to tension bars.

D. Organization and Management

W. Cottle, Plant Superintendent at Watts Bar, detailed plant experience for key members of the Watts Bar Staff individually (see Appendix XVI). J. Ebersole asked W. Cottle for his opinion as to whether he thought the Watts Bar Plant had too high a level of automation and too small an operating complement. W. Cottle indicated that he was comfortable with the degree of automation and limited amount of manning placed on the primary plant but he indicated that at times extra people are added in the support and secondary systems area.

J. Ebersole asked W. Cottle if TVA had looked at the single failure criterion in the context of its value for preventing spurious reactor trips. W. Cottle indicated that TVA has made some significant changes on the secondary plant originated trips at the Sequoyah Unit and plan to introduce those at the Watts Bar Unit. With regard to a question by J. Ebersole concerning turbine generator trips, W. Cottle indicated that most of the improvements made have not been directly associated with inputs into the turbine trip such as seal water injection function on main feedwater pumps or loss of both main feedwater pumps which initiates the turbine trip. He indicated that TVA is basically still looking into that problem.

H. Culver, Director of TVA's Nuclear Safety Review Staff (NSRS), discussed the basic functions of the NSRS (see Appendix XVII). He indicated that the NSRS routinely makes management reviews and routine or special reviews which can include the design construction area. J. Ebersole asked whether the NSRS gets involved in assessment of the detailed mechanical aspects of the plant in its design and construction reviews. H. Culver indicated that mechanical aspects were treated only to a limited degree up to this point.

M. W. Carbon questioned whether NSRS's review of the TVA Office of Purchasing was done to insure that equipment had been purchased to the right specifications. H. Culver indicated that the basic concern was with a design intent. In answer to a second question by M. W. Carbon, H. Culver indicated that the TVA review system was developed strictly within TVA based on a demonstrated process that has been used at the NRC. In answer to a question by J. Ebersole concerning the NSRS's activities with regard to the troubled Quality Assurance Program, H. Culver indicated that the NSRS had been reviewing the Quality Assurance organization for about 18 months in response to a request by the General Manager of TVA, and had identified fragmentation in the Quality Assurance organization.

E. QA Problems and Their Resolution

G. Beasley, Manager of Quality Assurance for TVA's Office of Design and Construction (OEDC), indicated that several major problems related to quality arose since 1981, dealing with heating and ventilating systems, welding and weld inspection, transfer of systems and equipment from the construction organization to the operating organization, and quality records and traceability from inspections (see Appendix XVIII). He explained that the root causes of quality program problems involved attitude and approach toward the program itself, in some cases traceable to a lack of definition of authority and responsibility. He pointed out that half of all deficiencies had their root causes in procedures, either with people not following the procedures, procedures not adequately interpreted, or the nonexistence of a proper procedure.

G. Beasley indicated that NRC's Region II personnel recommended that TVA undertake an additional independent verification that the plant has been constructed in accordance with design, performed by an organization outside TVA, one that does not have a major dependence upon TVA for its resources. He indicated that OEDC is in the process of arranging for this independent review.

G. Beasley noted that there have been a large number of nonconformance reports for the Watts Bar Plant. He indicated that OEDC suggested that there is a low threshold for these reports which requires reporting of items that are considered marginal.

F. Seismic Margins of Safety Above SSE

J. Williams, TVA, indicated that all safety related electrical and mechanical equipment has been qualified to levels which enveloped conditions defined for its as installed configuration. He indicated that TVA's Equipment Seismic Qualification Program is in full compliance with NRC and industrial recommended procedures, guides, codes, and standards and good engineering practice (see Appendix XIX). He indicated that Equipment Qualification Reports provide a conservative demonstration that the equipment is capable of withstanding its prescribed seismic conditions. He also added that the current philosophy regarding seismic qualification throughout the industry does not require that the effort be extended to determine how much better the equipment is than it needs to be, nor does the qualification data lend itself to the extraction of such information. The Seismic Qualification Program cannot be transformed into an Equipment Reliability Program. Reevaluation effort would provide indications of margins of conservatism and qualification of specific items of equipment.

Chairman Shewmon expressed concern at the difficulty of determining margins from the qualification test. J. Williams indicated that efforts to determine margins would require more of a fragility test where equipment is qualified by both tests and analysis. In answer to another question by Chairman Shewmon, J. Williams indicated that the equipment purchased is qualified to meet TVA's own specifications, and no inquiry is made as to whether this equipment is qualified to meet more stringent specifications.

J. Williams indicated that the seismic margin of conservatism at the Sequoyah Nuclear Plant has just been addressed by a reevaluation of equipment against higher seismic levels of the site specific spectrum to demonstrate that the qualified equipment has a factor of conservatism of at least 1.5. The equipment, structures, and piping were shown to be adequate for this new site specific spectrum. J. Williams indicated that the Browns Ferry Nuclear Plant had just undergone a probabilistic risk assessment study which included the consideration of equipment qualification. He indicated that the study found that most equipment reflected large margins of conservatism beyond the prescribed seismic conditions. The weak link was shown to be relay chatter in the electrical equipment. The Watts Bar Nuclear Plant will undergo a similar study to be completed by May of 1984.

J. Ebersole indicated that the relay chatter would create a variety of transients. He suggested that while those transients might not produce damage of permanent significance, it might impede the process of safe shutdown. T. Novak, NRC, pointed out that prior to the licensing of the Trojan Nuclear Plant, review was made on the electrical components, in regards to chattering effects of electrical equipment. It was clear that the chatter would induce transients, but it was also clear that a safe shutdown was possible.

J. Williams indicated that the Browns Ferry PRA study contained a discussion of minimum factors of conservatism used in seismic design of components. These factors are applicable to the Watts Bar plants. C. P. Siess expressed interest in these minimum factors of conservatism and requested a copy of the Browns Ferry PRA study. J. Williams explained that the study will not be released to the NRC Staff until sometime in 1983, but agreed to extract portions from the report that are relevant to the ACRS discussion of seismic margins and make them available to the Committee.

G. ERCW Piping Corrosion

J. Ebersole explained that TVA experienced an extraordinary corrosion problem in certain carbon steel piping and resorted to a process of mortar lining of the critical water pipes. Of import is whether this mortar lining material will degrade over time or may potentially be loosened so that under subsequent seismic loads it will unload and plu up the process pipes.

C. Bowman, TVA, read his presentation on cement mortar lining of the essential raw cooling water system yard piping into the record (see Appendix XX). In answer to the concern expressed by J. Ebersole about a seismic event, C. Bowman indicated that during the 1971 San Fernando earthquake, within three miles of the earthquake epicenter, a 96 inch, 34 year old, above ground water line owned by the Los Angeles Department of Water, suffered both vertical and horizontal displacement due to surface acceleration, was broken from its support and accordioned but its cement mortar lining remained undamaged except for spalling at the place where the pipe was accordioned where it did spall. After citing other examples, he concluded that unless there is significant deformation of the pipe, the cement lining will remain undamaged.

D. W. Moeller asked whether the problem of corrosion could be handled by chemical control. C. Bowman indicated that there is once-through cooling at Watts Bar. He did indicate that TVA had considered a closed system where water chemistry could be controlled, but the water would have to have been deoxygenated to be assured of good chemistry control.

F. Hand, TVA, reviewed the seismic qualification of the cement mortar lining to be installed in the pipes (see Appendix XXI). He described the full-scale testing done on the cement mortar lined pipes. D. W. Moeller expressed interest in whether TVA had altered any of its filtration logic as a result of using this material. R. Pierce, TVA, indicated that TVA has not altered any filtration logic.

H. Westinghouse D-3 Steam Generator

R. Pierce, TVA, read a text regarding steam generator vibration modifications into the record. He indicated that TVA became aware of a steam generator tube wear problem due to flow-inducted vibration in November, 1981 and began working with Westinghouse. The problem will be resolved with installation of a feedwater inlet nozzle to disperse the flow of fluids. The velocity at 100% load will be dispersed like a shower head to reduce the impingement on the rows of tubes in the steam generator and cut out the vibration that is causing the wear on the steam generator tubes. Chairman Shewmon noted that there would be slightly more pressure drop in the steam generator by such a fix.

Chairman Shewmon indicated that the support plate material in the D-3 steam generator which J. Shultheis identified as carbon steel is not the latest design. The Committee discussed TVA's operating procedures for the steam generators.

R. Pierce indicated that additional modifications will be made at Sequoyah 2 as well as Watts Bar to improve water chemistry. TVA is taking the copper out of the main feedwater heater and the condenser and modifying the demineralizer. He added that TVA will be using nitrogen bubbling through the condensate storage tank. J. Ebersole asked what TVA's long-range program was for the steam generators to preclude problems after 15 or 20 years of operation which may require replacement of the steam generators. R. Pierce indicated that the long-range program consists of installation of the vibration modification and control of water chemistry to take care of steam generator denting.

I. Plant Security

In answer to a question by C. Mark, T. Canyon, NRC, indicated that the NRC Staff has just completed its review of TVA's physical security plan. He indicated that the physical security plan has been accepted with one exception. TVA wishes to designate the containment as a non-vital area during refueling or major maintenance. He indicated that the Staff will impose a license condition on this matter. L. Mills, TVA, explained that TVA has asked the NRC Staff for permission to declare the containment a nonvital area in the case of lengthy major maintenance outage work so that TVA does not have to go through the security precautions and inspect all aspects of the plant as it did during the initial fuel loading. He indicated that it is a matter of changing Staff requirements under special circumstances although it is not certain if the Staff will agree.

Chairman Shewmon requested that the NRC Staff look into their requirements for separation of safety related systems in nuclear power plants and report back to the ACRS regarding this matter. He suggested that this subject be explored with respect to plant layout with emphasis on sabotage prevention and protection of vital functions. The Committee discussed TVA's awareness of NRC Sandia reports regarding separation. R. Pierce indicated that TVA has addressed separation of safety related systems in its design. He noted that the physical layout of the Watts Bar Plant was more or less settled in the 1972-73 time frame after all the concrete work had been finished and that the plant does not have as good a separation as TVA would like. This was apparent in the course of the review TVA had with the NRC Staff. He indicated that TVA's later plants expect to have better separation, especially of redundant safety circuits.

J. Concluding Remarks

D. W. Moeller asked, who in TVA, particularly with respect to Watts Bar, keeps abreast of LERs and how they apply to the Watts Bar Plant. W. Cottle indicated that TVA has an experienced review group at its Division Central Headquarters in Chattanooga which receives input on LERs from the various publications. He indicated that they do a preliminary screening evaluation on LERs as well as other experience inputs. W. Cottle pointed out that direct copies of all Sequoyah reports are also provided at the plant.

D. W. Moeller asked another series of questions which were to be addressed by the Applicant and the NRC Staff (see Appendix XXIII).

The Committee decided that they could write a letter on the Watts Bar Plant at this meeting.

IV. Proposed NRC Nuclear Plants Severe Accident Research Plan (NUREG-0900) and Related Rulemaking) (Open to Public)

[Note: G. R. Quittschreiber was the Designated Federal Employee for this portion of the meeting.]

A. Report of ACRS Subcommittee on Proposed Research Plan

W. Kerr directed the Committee's attention to SECY-82-203, a discussion of NUREG 0900 with a number of revisions entitled Nuclear Plant Severe Accidents Research Plan. He indicated that the document labled Draft Revision 2, made available to the ACRS in early April 1982, contains plans for producing research information needed to confirm regulatory decisions in the severe accident area. This will include methodology for examining the cost of new requirements, risk reduction and generalized reduction in the uncertainty of the PRA.

At a May 28, 1982 meeting of the Subcommittee with the RES Staff, the need for better correlation of the proposed research with some approach or possible approaches to dealing with severe accidents was discussed. The Staff agreed that additional explanation of its proposed work and its relationship to an approach would be helpful, and a revised document, SECY-82-203A, Revisions to "Nuclear Plant Severe Accident Research Plan," NUREG-0900 (Draft), was submitted to the ACRS.

W. Kerr continued that the EDO transmitted a memorandum to the Commission labled SECY 82-1, Severe Accident Rulemaking and Related Matters, dated January 3, 1982 and made available to the ACRS also in January, 1982. This memorandum from the EDO proposed to deal with severe accidents in the process of licensing for Standard Plants. He pointed out that the emphasis is on Standard Plants with regulatory decisions on existing reactors which may or may not be dealt with by rulemaking. After submission of ACRS comments and questions, a revised document labled 82-1A was submitted to the Committee slightly before the August 6 meeting (see Appendix XXV).

W. Kerr summarized a few key points in SECY-82-1A. He pointed out that there is a section on filtered, vented containments which indicates that these systems or a variation of such systems should be provided on future applications for both BWRs and PWRs if these yield a cost effective reduction in risk. Studies of large reactor containment buildings indicate that the classical core retention devices are probably not cost effective regarding reduction of atmospheric release of radiation.

D. Okrent pointed out that besides SECY-82-1A, there is a second proposed policy statement on safety goals and a Draft Action Plan for implementing the Commissions proposed safety goals. He urged that the Committee give priority to this trilogy of documents at its September or October meeting.

D. Okrent posed questions concerning some of the tentative conclusions drawn by the Staff in SECY-82-1A. He pointed out that measures being taken in other countries regarding PWRs both in the area of preventing core melt and in mitigating core melt in many cases provide increased efficiency over what is in existing PWRs in this country. He suggested that before the Staff arrives at some conclusion, they should understand these other approaches and why they may or may not be relevant in the U.S. Mention was made of post-accident flooding as a possible change for a PWR. D. Okrent indicated that he thought this an interesting idea but the NRC Staff has not seriously assessed the concept or compared it to other possibilities. He suggested that some of the positions taken by the NRC Staff may not stand up technically and may not stand up politically after a change in federal administration.

D. Ross, NRC Staff, indicated that draft NUREG-0900, The Severe Accident Research Plan, consists of the main portion of SECY-82-203 with some amended pages currently undergoing final review as SECY-82-203A (see Appendix XXVI). He indicated that the general purpose of the plan is to develop generic answers or bases to determine how safe operating plants are and where and how they ought to be improved. He indicated that "how safe should plants be" will have to come from the Commission's safety goals.

D. Ross explained that the Commission will use risk assessment methods to see how safe plants are and will use different techniques to determine how to make them safer such as value-impact theory or cost analyses as well as risk reduction analyses. Detailed methods for accident evaluation would include the use of computer codes such as RELAP for the thermal hydraulics, detailed core information from codes such as SCDAP, primary system details using TRAP MELT, and details of the containment using the CONTAIN core code. Fast running methods, the so called risk codes would be the MARCH family of codes to be finally replaced by MELCOR. The Committee discussed use of the risk codes such as the MARCH MATADOR family of computer codes and differences between MATADOR and the code CORRAL.

In answer to a question by M. Bender, M. Cunningham, NRC Staff, indicated that CORRAL does not take account of specific address mechanisms for radionuclides. MATADOR has been set up to address these additional removal mechanisms to account for inert materials. M. Bender asked whether the MATADOR code gives a better and more usable representation of the behavior of a containment system as it relates to these radionuclide movements.

R. Bernero pointed out that CORRAL, an aerosol or fission product transport code tended to diverge from the apparent truth as the calculation went out in time. Since the CORRAL code is basically only gravity settling of aerosols, inclusion of other mechanisms for aerosol settling will presumably get closer to the real truth in regard to the dose model. R. Curtis pointed to the CONTAIN code as being particularly useful because it contains thermal dynamics effects. M. Bender asked whether the CONTAIN code could be retained and all of the rest of the aerosol codes deleted.

H. W. Lewis asked why a high volume unfiltered containment vent reduces the core melt frequency by a factor of 13. M. Cunningham, NRC Staff, explained that for boilers, one of the important sequences was a long-term loss of containment heat removal. In that case, the containment heated up to the point of containment failure with the ECCS working all of this time. At the time of the containment failure the ECCS was assumed to fail with some probability such that the containment failure led to the core melt. He indicated that high volume vent prevented the gross overpressure failure so that you would not get the ECCS failure. H. W. Lewis added that a factor of 13, makes that sequence more important than the total of all other sequences that would not be mitigated by the high volume core vent or containment vent. M. Cunningham indicated that this was the case. D. Okrent added that presumably the ability to get water into the reactor vessel is more reliable than the ability to take it out of the containment.

D. Ross described the Severe Accident Sequence Analysis (SASA) Program as a pure analysis program on multiple failure events.

The Committee discussed the hydrogen generation and control subelement of the program. W. Kerr was particularly interested in requirements for new plants to be designed for 100% metal water reaction. R. Bernaro explained that when you get into different accident sequences and get sufficiently different hydrogen steam source terms, one can derive almost any containment pressure and result that you want to the extent that 100% or 75% metal water reaction is manageable by igniters in one accident sequence and not in another. W. Kerr added that he understood the Staff response to mean that the Staff is not confident how to design for 100% metal reaction and needs this research in order to be able to do it. O. Bassett, NRC Staff, indicated that the Staff does not have sufficient confidence that the igniters as placed will do the job because there is not enough knowledge on how hydrogen propagates around the containment.

D. Ross discussed the development of a computer code called TRAP MELT and the intention to do fission product release and transport experiments with that code at the MARVIKEN or PBF test facility. D. Okrent noted that the question of the radioactive source term from accidents certainly warrants thought. He suggested that the Staff should be satisfied that for some scenarios there is an order or two of magnitude reduction. He suggested a systematic look in an order of magnitude way at those scenarios which are likely to lead to low releases and those scenarios likely to lead to larger releases. Identification of the cause of the scenarios would be the next step.

R. Bernero pointed out an NRC Staff memorandum to supplement NUREG-0772 which discusses relative risk reduction by looking at individual accident sequences and their characteristics. He offered to furnish copies of this memorandum to the ACRS. In answer to a question by Chairman Shewmon, R. Bernero indicated that noble gas release is not nor does not constitute a dominant offsite risk in the present accident models. Chairman Shewmon suggested that the source term will depend on choice of models and the treatment of aerosols. He noted that a lack of understanding is probably leading to a substantial overestimation of the source term in some accident scenarios. R. Bernero agreed.

D. Ross indicated that the NRC Staff has a September 9, 1982 meeting with the Commission and desires ACRS comments or concurrence on the Severe Accident Research Plan.

W. Kerr indicated that it is strongly stated in the implementation of the research plan that Staff does not yet have enough information to predict with any confidence the behavior of containments. He asked if NUREG-0900 had identified research regarding description of the behavior of the containment. J. Costello, NRC Staff, pointed to a specific section 5.8, entitled Containment Failure Mode, which attempted to define loads above design pressure that could be handled with some degree of confidence by containments. W. Kerr pointed out that containment failure due to overpressure is only one way of describing containment performance. D. Okrent noted that there are elements in the research program that relate to the question of containment performance, but they have not been laid out in a structured form even though the Staff has indicated the elements they need to know in order to derive the containment performance design objective. Having all of that information available does not answer the question as to how the Staff gets to its final objective, a reasonable index of containment performance.

The Committee agreed that it had sufficient information to prepare a letter regarding this matter.

V. Nuclear Power Plant Control Room Habitability (Open to Public)

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

A. Report of ACRS Subcommittee on Reactor Radiological Effects

D. W. Moeller indicated that the Subcommittee on Reactor Radiological Effects held a meeting on May 14, 1982 to discuss noted wide differences in the operating cababilities of control room HVAC systems in operating plants. He noted that a review of LERs pointed to a continuing increase in the number of failures of various components in control room ventilating systems of operating plants, as well as those under construction. The Subcommittee found that many occurrences of failures in control room air cleaning and air ventilating systems are not being reported in the LER system. J. Ebersole pointed out that biological habitability now involves tight thermal and environmental control because susceptible apparatus with very narrow acceptance bands of temperature performance are being used in control rooms. Instrument vulnerability must be considered as well as human habitability.

B. Meeting with Representatives of NRC Staff and Nuclear Industry

R. R. Bellamy, NRC Staff, reviewed design and testing criteria applicable to air filtration systems for control rooms. He indicated that GDC 19 of 10 CFR 50, Appendix A is the regulation the NRC Staff uses in its beginning review of control room habitability systems (see Appendix XXVII). He described Regulatory Guide 1.52, Rev. 2 which contains design testing and maintenance criteria for post-accident engineered safety features (ESF) atmosphere cleanup system air filtration and absorption units.

D. W. Moeller indicated that the likelihood of an event requiring the use of these systems does not enter into the requirement for the system and he indicated that he was not aware of the use of probabilistic calculations used in the design of control room habitability systems. K. Murphy, Division of Risk Assessment, NRC, indicated that the regulatory guides use a 10^{-6} number regarding toxic gases in terms of operator incapacitation. He added that the Staff is considering a number on the order of 10^{-5} in the frequency of operator incapacitation for toxic gas releases. H. Krug, NRC Staff, indicated that NUREG-0737 referred to Standard Review Plan 2.2.3 in which licensees are permited to show probabilistically that they do not have to provide protection for the operators if they are below the acceptance criteria which are 10^{-6} conservatively, and 10^{-7} realistically.

R. Bellamy presented a schematic of a typical air cleaning system with moisture separators, a heater, a bank of HEPA filters, and charcoal adsorber (see Appendix XXVII). He discussed the basic design criteria for the system.

The Committee discussed the design and testing of HEPA filters and activated carbon filter systems. D. W. Moeller noted that the HEPA filters are no longer sent, as they used to be four years ago, to a Department of Energy laboratory. The Staff had reviewed the DOE facility data and concluded that it was no longer necessary to send those filters to the DOE labs before being sent to the site. He indicated that technical papers at the 17th Nuclear Air Cleaning Conference, nevertheless, noted a significant percentage of rejections among HEPA filters sent to DOE laboratories for performance tests. The rejections were all significant failures.

J. L. Kovach, Nuclear Consulting Services, spoke of changing Regulatory Guide 1.52 (which specifies the current requirements for testing filtered systems) to actually report test results as they are obtained and then specify the fix separately. This type of reporting of all test results instead of just the end results would give a much better history of the installation and preoperational testing of filter systems. It would also provide corroboration as to whether the current practice of 18 months between regular testing intervals is adequate or whether shorter or longer intervals would be more appropriate depending upon the type of system.

J. L. Kovach pointed out that very early generation filter systems in current operating plants have had structural problems and maintenance problems where some are almost impossible to maintain even under cold conditions. He also pointed out that systems currently installed have a chlorine adsorption capability that has nothing to do with the actual chlorine exposure in the control room and would be incapable of holding chlorine longer than a few seconds. L. Kovacs indicated that even from a radiological standpoint currently operating filter systems are greatly undersized and many of these systems are inadequate even for the undersize operation. Some of these systems leak very badly, many are located together with other filter systems for other areas of the reactor with attendant possibilities for cross contamination.

J. L. Kovach pointed out that filter systems as they exist in most of the European countries have protection capabilities significantly higher than U.S. systems mainly because of the significant conservatism used in the design of these systems and the much stronger cooperation with chemical process engineering personnel in designing the systems.

Particularly at the early stage, many of the filter adsorber trains installed in Europe were designed based on chemical and industrial experience and not solely on heating and ventilation concepts.

M. Bender asked how an NRC approved filter system differs from a European system. J. L. Kovach indicated that the main difference is the use of up to 50 centimeters of activated carbon in European systems vs. using plenums with only up to 5 centimeters of carbon in U.S. systems. The European systems have about 10 times longer residence time in the absorber. There are no other major differences.

D. W. Moeller solicited comments from J. L. Kovach concerning his experience with the testing of control room air cleaning systems where operators expressed a lack of confidence in the filter system. Operators actually fear staying in the control room in the case of a challenge to that system. J. L. Kovach pointed to instances where his consulting firm was testing a filter system and operators got upset because the habitability of the control room would deteriorate to the point of actual discomfort. This discomfort would take the form of humidity or temperature, mainly temperature.

L. Klaes, TVA, described features of main control room habitability based upon the Sequoyah Nuclear Power Plant (see Appendix XXVIII). He showed a general arrangement diagram of the Sequoyah Plant and the relationship of the control building to other major buildings on the site. The general configuration of the habitability enclosure included air intake locations and a tabulation of the main control room habitability design considerations addressed by TVA.

When L. Klaes reviewed a table of radiation sources, M. Bender referred to a question earlier by W. Kerr with regard to source terms. S. Ness, TVA, indicated use of either Regulatory Guide 1.3 or 1.4 source terms in the containment due to a loss-of-coolant accident or that inventory as the source essentially found in TID 14844 based upon containment leakage as specified in those documents. C. Mark expressed particular interest in the description of radiation monitors which activate alarms and initiate emergency operating features. D. A. Ward expressed concern regarding significant heat loads from accumulated radioactivity if there were an incident where dual systems were filtering out radioactive contaminants and one system began to leak and was shut off. L. Klaes indicated that TVA had looked at the situation and found that the heat loads are very small in these areas. He indicated that there are some sytems in other areas of the plant, such as the emergency gas treatment system that operates in the auxiliary building, that does have a potential for high heat buildup if the unit is secured. Therefore, for that system there is a recirculation mode that continues putting a small amount of air through the system.

W. Miller, Sargent and Lundy, Inc., described the control room habitability HVAC system in the LaSalle County Nuclear Plant. He indicated that the system had an air conditioning portion and an air cleaning portion, 3000 CFM for the air cleaning portion vs. 25,000 CFM for the air conditioning system. W. Miller described Sargent and Lundy's design methodology and indicated that the calculation of a bounding radiological iodine protection factor (IPF) and an estimation of hazardous chemical concentrations expected are extremely important to the kind of system to be used to meet imposed design limits. He summarized the iterative process to arrive at a final design, taking account of habitability and design leakage rate. M. Bender asked whether smoke tests were commonly used for leak testing such systems. W. Miller indicated that smoke tests are out of date and helium has been found to be a very good test for leakage as well as a simple soap bubble test especially in the case of positive pressure systems. He indicated however, that soap bubble tests will not work very well on negative pressure systems.

VI. Quantitative Safety Goals (Open to Public)

A. Report of ACRS Subcommittee on Proposed NRC Quantitative Safety Goals And Proposed Implementation Plan

[Note: J. M. Griesmeyer was the Designated Federal Employee for this portion of the meeting.]

D. Okrent explained that the purpose of the Reliability and Probabilistic Subcommittee Meeting on August 6, 1982 was to discuss the Draft Action Plan for the implementation of the proposed NRC Safety Goals, to discuss severe accident policy and its relation to the implementation of a safety goal, and to assess the status of NRC efforts to develop a safety goal. He referred to a set of questions on safety goals from F. J. Remick with special emphasis on particular questions that referred to the use of a two-level approach (i.e., design objectives and operational levels) which is part of the implementation plan. The Committee discussed Table 1 of the Draft Action Plan for implementing the proposed safety goals entitled, Implementation of ALARA Guidelines, which was part of a July 6, 1982 memorandum from the EDO to the Commission. Note was made of the use of the median estimate from probabilistic risk assessment. It was suggested that the mean might be a better indicator of risk than use of the median as proposed by the

NRC Staff. The basic principles of implementation were reviewed (see page 6 of Enclosure 1, Appendix XXX). D. Okrent noted that uncertainty in the estimates of risk will vary widely from reviewer to reviewer.

Note was made of the difficulty of incorporation of earthquakes and floods in the Quantitative Safety Goals, especially the allocation of the core melt probability to internal and external phenomena. With regard to ALARA, suggestion was made for the use of a surrogate cost figure in lieu of the $1000 per man-rem now contemplated. Chairman Shewmon thought the safety goals should be implemented through the use of examples such as actual tests with representative systems. W. Kerr noted that Sandia was calculating subsidiary probability contributions to core melt.

W. Minners, NRC Staff, indicated that safety goals would be manifested in two ways - (1) value impact, and (2) absolute goals. V. Stello, NRC Staff, expressed concern that the Staff was not convinced that the implementation plan could be used effectively without controversy occurring in a debate over numbers.

C. Mark and M. Bender expressed concern over the validity of the $1000 per man-rem ALARA cost figure in terms of delayed cancer risk. W. Kerr noted the Commission's endorsement of the ICRP position that any radioactive exposure is harmful and the Commission's commitment to mitigate any exposure.

D. A. Ward suggested the use of specific reliability requirements for individual plant systems although D. Okrent expressed some concern that the NRC was not ready to specify such numbers for the numerous systems involved in the total plant. C. Mark expressed concern for the use of core melt as a yardstick. J. Ebersole suggested that even though the NRC should be most concerned with containment reliability, it must consider a core melt even though the containment could be designed to preclude any release. H. W. Lewis argued against use of the ALARA concept as part of the licensing process, suggesting instead that the NRC should specify limits on specific plant features/performance to limit public risk to a level considered acceptable.

D. W. Moeller suggested that the Staff recognize and present ways to avoid the uncertainties associated with the implementation of the safety goals. H. Etherington questioned the correlation between the frequency/size of a core melt and the consequences of such an accident. W. Kerr suggested that there ought to be a spectrum of core melt frequencies since there are a spectrum of initiators that would cause a spectrum of core melts. D. Okrent noted that the safety goals assume that all core melts follow the same progression and all containments the same break.

D. Okrent suggested that the discussion of this matter consider the three documents before it - (1) SECY-82-1A The Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation, (2) The Action Plan to implement the Commission's proposed Safety Goal Policy Statement, and (3) the proposed Revised Safety Goals for Operation of Nuclear Power Plants as a package. C. P. Siess suggested that the safety goals should include a probability of containment failure to go along with the probability of core melt. The definition of containment performance and the use of the mean in the Draft Implementation Plan were also suggested as possible topics of discussion. The Committee agreed that it should write a letter to the Commission during the 269th ACRS Meeting regarding these issues and also to respond to the questions asked of the Commissioners by OPE.

VII. Subcommittee Report Regarding Seismic Events and Emergency Planning (Open to Public)

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

D. W. Moeller noted a position paper on emergency planning and natural hazards from the EDO entitled, Basis of Consideration of Natural Hazards in Emergency Planning (see Appendix XXXI). He indicated that he was not satisfied with the EDO's recommendation which proposed that for most sites, earthquakes need not be explicitly considered for emergency planning purposes because of the very low likelihood that an earthquake severe enough to disturb onsite or offsite planned responses will occur concurrently with or will cause a reactor accident.

D. W. Moeller cited five points of contention regarding his appraisal of the EDO's position on emergency planning and natural hazards (see Appendix XXXII). He indicated that he favored inclusion of earthquakes in emergency planning and suggested that the Subcommittee on Reactor Radiological Effects set aside sufficient time to define questions and answers regarding a response to the NRC Staff's position.

VIII. Foreign LWR Licensing Practices (Closed to Public)

[Note: J. M. Griesmeyer was the Designated Federal Employee for this portion of the meeting.]

D. Okrent presented a report to the Committee regarding information he received from foreign regulatory bodies concerning their policies on severe accidents and other safety issues and their application to the regulation of foreign light water reactors (see confidential supplement).

IX. Preparation of Proposed NRC Long-Range Research Program Plan (Open to Public)

[Note: S. Duraiswamy was the Designated Federal Employee for this portion of the meeting.]

C. P. Siess reviewed the history of communications between the ACRS and the Commission regarding the Committee's review of the proposed NRC Long-Range Research Plan (LRRP). He indicated that the Committee had not received a response to its latest letter to Chairman Palladino written on June 7, 1982 in which the Committee proposed that it no longer report formally on the LRRP. The Executive Director indicated that the Commission, when it responds to the latest ACRS letter, may ask the ACRS to continue to review the LRRP, contrary to the Committee's suggestion, as a mechanism to provide earlier ACRS input regarding the formulation of the RSR budget. C. P. Siess noted that the LRRP may include improvements in format and scope consistent with previous ACRS recommendations. C. P. Siess then addressed the upcoming ACRS 1983 report to the Congress on the NRC RSR budget proposed for 1984-85.

X. Proposed Revision of 10 CFR 50.6, Appendix K, ECCS Evaluation Models (Open to Public)

[Note: P. A. Boehnert was the Designated Federal Employee for this portion of the meeting.]

M. S. Plesset, Chairman of the ECCS Subcommittee, reported briefly regarding proposed changes in 10 CFR 50, Appendix K, ECCS Evaluation Models. He noted General Electric's (GE) proposal to allow use of the ANS 1979 decay heat curve. GE has indicated that they would not use the added margin provided by use of this curve to increase reactor power. M. S. Plesset detailed some of the benefits GE sees in the use of the new curve, including improved fuel utilization and better core power distribution.

M. S. Plesset mentioned that GE has submitted a new ECCS Evaluation Model Code which is currently under review by the NRC Staff. They contend that use of this code will provide additional margin against the 10 CFR 50.46 limits. He indicated that the Subcommittee believes that best estimate models should be used in lieu of NRC Evaluation Models for ECCS licensing requirements.

XI. Lyon Conference on LMFBR Safety

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

M. W. Carbon discussed European views on LMFBR safety as follows:

- The French and British suggest that too much emphasis has been placed on energetic, core disruptive accidents (CDAs) by the U.S.

- There was concern on the part of the French that it was difficult to determine the anticipated cost of an LMFBR. They assume that these costs would be about 25% above those for development of a light-water reactor.

- The French are disregarding or ignoring energetic CDAs in their design of their Super Phoenix.

- The Germans are taking an approach similar to that in the U.S. with regards to CDAs.

- The Europeans suggest that there is not an apparent benefit with regard to mitigating CDAs through the use of a heterogenous core (such as in the CRBR) in large LMFBRs.

- The Germans have done a probabilistic risk assessment study on their SNR 300 reactor in which they have determined that the probability of a serious release to be 10^{-8}. They contend that the biggest risk probability is for a fuel melt occurring in the storage pool.

- The German SNR 300 reactor has a dual control system consisting of solid rods and a second chain of absorbers pulled up by mechanical springs. This system should take account of seismic considerations and mitigate common mode failures.

- The British are designing a decay heat removal system for an LMFBR of commercial size to consist of four independent decay heat removal loops utilizing natural circulation in the separate loops. They calculate a realistic failure probability for the system to be 10^{-5}.

XII. Participation in NRC Staff Programs (Open to Public)

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

W. Kerr referred to a letter from C. H. Poindexter of Baltimore Gas and Electric (BG&E) to H. R. Denton of NRR regarding BG&E's decision to participate with certain conditions in the program to resolve USI A-49, Pressurized Thermal Shock, and its deferral of a decision to participate in the resolution of USI A-47, Safety Implication of Control Systems. He expressed his concern that this active participation in NRC Staff programs might be overloading BG&E and diverting resources necessary for safe plant operation. He noted that BG&E appears to be asked to participate in these generic or pilot programs primarily because of their close proximity to the headquarters office in Washington. W. Kerr suggested that the Committee give some thought to this matter, but the ACRS chose not to take any action at this time.

XIII. Executive Sessions (Open to Public)

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

A. Subcommittee Assignments

1. Reactor Operations

The Committee discussed a proposed memorandum to the EDO tentatively entitled, Suppression Pool or Containment Sump Water Contamination with Potential Adverse Effects on Post-Accident Cooling Pumps, regarding the matter of fine contaminants that may be carried into the suppression pool or containment sumps of nuclear power plants and eventually into pump bearings. The memorandum was deferred, however, and the subject was referred to the Reactor Operations Subcommittee for further action as appropriate.

2. Reactor Radiological Effects

D. W. Moeller, Chairman of the Reactor Radiological Effects Subcommittee, reported briefly concerning a proposed NRC Staff position (see memorandum from W. J. Dircks to the Commissioners dated June 22, 1982, subject: Emergency Planning and Natural Hazards) regarding consideration of seismic events in emergency planning and noted the intent of the Subcommittee to pursue this matter in a meeting with the NRC Staff.

B. ACRS Reports, Letters, and Memoranda

1. ACRS Report on the Grand Gulf Nuclear Station Unit 1

The Committee prepared a report to the Commissioners of the completion of its review of the application of Mississippi Power and Light Company (MP&L); Middle South Energy, Inc., and the South Mississippi Electric Power Association for an operating license for the Grand Gulf Nuclear Station Unit 1. The Committee concluded that, if due consideration is given to items mentioned in this report (August 18, 1982) and the recommendations contained in its interim report dated October 20, 1981, operation at full power is acceptable.

2. ACRS Report on Watts Bar Nuclear Plant Units 1 and 2

The Committee prepared a report to the Commissioners of its review of the application of the Tennessee Valley Authority (TVA) for authorization to operate the Watts Bar Nuclear Plant Units 1 and 2 and recommended that if due regard is given to the items mentioned in the body of the report, and subject to satisfactory completion of construction, staffing, and preoperational testing, the Watts Bar Nuclear Plant Units 1 and 2 can be operated at core power levels up to 3411 Mwt.

D. Okrent appended additional comments concerning a recommendation that TVA and the NRC Staff conduct studies to evaluate the margins available to accomplish safe shutdown, including long-term heat removal, following an earthquake of somewhat greater severity and lower likelihood than the safe shutdown earthquake.

3. CRS Report on the Systematic Evaluation Program Review of the R. E. Ginna Nuclear Power Plant

The Committee prepared a report to the Commissioners of its review of the Systematic Evaluation Program, Phase II, as it has been applied to the R. E. Ginna Nuclear Power Plant. The ACRS concluded the following:

a. The SEP has been carried out in such a manner that the stated objectives have been achieved for the most part for the Ginna Plant and should be achieved for the remaining plants in Phase II of the program.

b. The actions taken so far by the NRC Staff in its SEP assessment of the Ginna Nuclear Power Plant are acceptable. The Committee did note, however, that many of the decisions involved in the SEP could be made much more rationally if plant-specific PRAs were available.

c. The ACRS will defer its review of the FTOL for the Ginna Plant until the NRC Staff has completed its actions on the remaining SEP topics and the USI and TMI Action Plan items.

4. ACRS Comments on "Nuclear Plant Severe Accident Research Plan", NUREG-0900 (Draft)

The Committee prepared a report to the Commissioners of its review of the version of NUREG-0900 (Draft), Nuclear Plant Severe Accident Research Plan, which accompanied a draft of SECY-82-203A (August 1982). The Members found that neither the original nor the revised version of NUREG-0900 contains a delineation of an approach for dealing with severe accidents necessary to judge the appropriateness of the proposed research program. When referring to specific areas such as containment performance, the ACRS noted no systematic description of what information is needed or what part of the proposed program is designed to provide the information even though there are elements of the program that could contribute to more accurate specification of containment performance. Although most of the research is considered to be confirmatory by the NRC Staff, the report and associated documents do not make explicit what is to be confirmed. The Committee repeated its offer of assistance to work with the NRC Staff in developing a new approach for dealing with severe accidents.

5. Control Room Habitability

The Committee completed a report to the Commissioners regarding control room habitability at nuclear power plants including associated heating, ventilating and air conditioning (HVAC) systems as well as supporting air cleaning systems. Reported to the ACRS were deficiencies in these systems which could lead to inadequate protection of plant operating personnel in case of an accident and an erosion in the confidence that plant operators have in the anticipated response and performance of HVAC systems and associated air cleaning equipment in the event of an emergency. The Committee recommended several actions to correct the problems discussed as follows:

a. Inplementation of an improved program for testing the adequacy of air flow rates and the leak tightness of control room engineered safety feature HVAC systems

b. Laboratory or field tests conducted to obtain data for defining the proper locations of control room air intakes and evaluation of the location and performance under emergency conditions of existing control room air intakes

c. Studies to assess possible benefits of increasing the minimum thickness and number of layers in charcoal adsorption beds used in protective air cleaning systems

d. Additional members of the NRC Staff be provided technical training to evaluate control room HVAC systems

e. Reports of tests conducted by private industrial and consulting organizations on control room HVAC systems should be made available to the NRC Staff

f. NRC should reconsider its policy to eliminate the requirements for certification of HEPA filters by one of the test facilities operated by the U.S. Dept. of Energy

g. Evaluation of the degree of prescription that should be included in requirements for the design, construction, maintenance, and operation of control room habitability systems

h. Failure modes and effects analysis should be conducted on all systems related to control room habitability.

6. Proposed Regulation on Disposal of High Level Radioactive Wastes In Geologic Repositories

The Committee prepared a report to the Commissioners regarding draft regulation, Disposal of High Level Radioactive Waste in Geologic Repositories, 10 CFR 60. The Members endorsed the change in approach by the NRC Staff in which the disposal of transuranic wastes in a repository will be considered by the Commission on a case-by-case basis. The ACRS suggested that the proposed changes in the definition of the "accessible environment" is vague and would make difficult the confirmation of acceptable performance (i.e., required 1000-year groundwater travel time to the accessible environment) by the operator of a disposal facility. Reconsideration of the original definition was suggested. The Committee also noted that redefinition of the "waste package" to exclude clay backfill may make it more difficult to determine compliance with the 1000-year containment requirement.

7. Interactions with Hydraulic Lines Caused by a Loss-of-Coolent Accident

The Committee approved a memorandum from the ACRS Executive Director to the EDO regarding questions which have arisen concerning the likelihood and effects of a loss-of-coolant accident in the drywell of a BWR which causes interactions with the hydraulic lines needed for safety rod insertion in such a way as to prevent rod insertion, creating the potential for recriticality when the core is reflooded by safety injection water.

C. Generic Safety Items

1. Consideration of "Major Societal Resources" in the Siting of Nuclear Plants

Based on an interest expressed by Chairman Palladino during the 258th ACRS Meeting and a request by D. Okrent, a joint group of ACRS Staff members and ACRS fellows prepared a paper to address the subject of major societal resources in addition to demography and hydrology in the siting of nuclear plants entitled, Management of Potential Resource Losses Due to Nuclear Power Plant Accidents. The Committee discussed several alternative actions to make the document available to the Chairman and decided to transmit it as a draft document for consideration by the NRC Staff with copies to the Chairman.

D. Future Schedule

1. Future Agenda

The Committee agreed on a tentative agenda for the 269th ACRS Meeting, September 9-11, 1982 (see Appendix II).

2. Future Subcommittee Activities

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

E. ACRS Comments regarding DOE Program Definition, Containment Integrity Function

DOE (A. Millunzi) has asked for comments by August 18, 1982 from ACRS Members regarding the proposed Program Definition Plan for an evaluation of the containment integrity function. The ACRS Executive Director informed the Committee that individual comments had been provided by M. Bender and J. Ebersole for transmittal to DOE. The Committee indicated that it would not comment as a collegial body on the DOE Program Definition Plans (8 additional plans are expected) but offered no objection to submission of comments by individual Members. The Executive Director noted that individual Member comments should be provided to the ACRS Office by August 20, 1982 for transmittal to DOE.

F. Participation of ACRS Members on American Nuclear Society Sponsored Panel to Review Nuclear Power Plant Accident Source Term

W. Kerr has been asked to serve as a member of an American Nuclear Society (ANS) ad hoc committee to study and prepare comments on the nuclear power plant source term. The Committee endorsed his request that he attend as an ACRS observer, rather than a working member of the ANS committee.

G. International Conference on Nuclear Power Experience

The Committee agreed that W. Kerr participate in a panel discussion regarding lessons learned from the TMI-2 accident at the IAEA International Conference on Nuclear Power Experience on September 13-17, 1982.

H. DOE-TVA Project on an Integrated Approach to Nuclear Power Plant Safety And Availability Performance

The Committee did not object to attendance by D. Okrent and W. Kerr at a meeting being conducted by Pickard, Lowe and Garrick, Inc. on August 10, 1982 in Washington, DC as part of a DOE sponsored effort to develop an integrated model for use by nuclear plant management regarding plant safety related changes that will take into account plant availability, economics, etc.

The 268th ACRS Meeting was adjourned at 11:30 a.m., Saturday, August 14, 1982.

ACRS-2020

APPENDIXES
TO
MINUTES OF THE 268TH ACRS MEETING
AUGUST 12-14, 1982

ATTENDEES
268TH ACRS MEETING
AUGUST 12-14, 1982

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

Paul G. Shewmon, Chairman
Myer Bender
Max W. Carbon
*Jesse Ebersole
Harold Etherington
William Kerr
Harold W. Lewis
Carson Mark
Dade W. Moeller
David Okrent
Milton S. Plesset
Forrest J. Remick
Chester P. Siess
David A. Ward

*Member Emeritus

ACRS STAFF

Raymond F. Fraley, Executive Director
Marvin C. Gaske, Assistant Executive Director
M. Norman Schwartz, Technical Secretary
Herman Alderman
William M. Baldewicz
Stuart K. Beal
Alden Bice
William M. Bock
Paul A. Boehnert
Don Bucci
Anthony J. Cappucci
Joseph Donoghue
Sam Duraiswamy
David C. Fischer
J. Michael Griesmeyer
Elpidio G. Igne
Kenneth D. Kirby
Morton W. Libarkin
John A. MacEvoy
Richard K. Major
Thomas G. McCreless
John C, McKinley
Thomas McKone
Austin Newsome
Gary R. Quittschreiber
Christopher Ryder
Richard P. Savio
Stanley Schofer
R. C. Tang

CONSULTANTS

G. L. Schott
Dave Langstaff

NRC STAFF ATTENDEES

268TH ACRS MEETING

Thursday, August 12, 1982

Nuclear Reactor Regulation

E. Goodwin
A. Schwencer
T. Novak
W. Butler, CSB
C. Tan, SEB
C. Tinkler, CSB
A. Notafrancesco, CSB
D. Terao, MEB
H. Brammer, MEB
T. Kenyon

Div. of Human Factors Safety

J. Zimolinski
R. Benedict

Region II

A. Wagner
D. Verrelli
F. Cantrell
R. Lewis

Div. of Licensing

D. Houston
K. Heitner

Nuclear Regulatory Research

A. Costello
K. Murphy
D. Ross
R. Bernero
M. Cunningham

TMI Site

R. Bellamy

A-2

NRC STAFF ATTENDEES

268TH ACRS MEETING

Friday, August 13, 1982

Nuclear Reactor Regulation

E. Goodwin
A. Buslik
J. Sharaker
P. Shewmanski, DE
T. M. Novak, DL
E. G. Adensam, DL
J. Pulsipher, DSI
C. O. Sillivan, MTEB
L. P. Crocker, DHFS
W. Minners

Nuclear Material Safety & Safeguards

R. Cramann

Region II

D. R. Quick

APPLICANT ATTENDEES

268TH ACRS MEETING

Thursday, August 12, 1982

Bechtel

E. Warfield
M. David
A. Zaccaria
J. Bizila
M. A. Harris
B. Kochis
R. Trickovic
J. Sundergill
R. Ly Beck
H. M. Brooks
J. C. Catlin, Jr.
S. D. Routh
S. K. Sen
A. T. Vience
D. H. Ashton
K. Lee
N. S. Montgomery

Middle South Services

F. Lewis
J. B. Richard

Mississippi Power & Light

J. Sundergill
J. Ceasare
J. McGaughy
C. K. McCoy
S. Hobbs
D. H. Ashton
R. W. Angle
J. Richardson
N. L. Stampley
T. E. Reaves

Bechtel Power Corporation

S. D. Routh
D. H. Ashton

General Electric Corporation

W. M. Davis
H. E. Townsend
L. F. Dale
R. L. Huang
A. R. Smith
D. K. Dennison

APPLICANT ATTENDEES

268TH ACRS MTG.

Friday, August 13, 1982

Tennessee Valley Authority

D. L. Williams
H. N. Culver
D. H. Level
M. V. Miller
M. N. Bussler
D. Ormsby
C. Bowman
E. Beasley
R. R. Reeves
J. Raulston
D. Sater
E. Merrick
W. T. Cottle
R. D. Erickson
V. A. Bianco
L. M. Mills
R. M. Pierce
J. S. G. Williams
W. L. Byrd
E. R. Ennis
F. Hand
R. Graves
L. Klaes
S. Ness

Westinghouse Electric Corporation

I. Murphy
B. Geyor
G. Buterworth
F. Cadets

PUBLIC ATTENDEES

268TH ACRS MEETING

Thursday, August 12, 1982

R. Leyse, EPRI
G. M. Fuls, Westinghouse
P. F. Collins, KMC
L. Klaes, TVA
S. Ornberg,Sargent & Lundy
E. Murphy, Westinghouse
W. H. Miller, Sargent & Lundy
R. J. Ross, Doub & Muntzing
P. A. Minson, ARC
J. E. McEwen, Jr., T
J. Neuman, Clarrion-Ledger
M. L. Ryan, McGraw-Hill
W. L. Horrell, EBASCO
J. S. Wetmire, General Public Utilities Nuclear
C. Brown, Public Utilities Fort.
S. Nass, Tennessee Valley Authority
P. Merbe, DSA
J. L. Kovoch, Nuclear Consulting Services
C. C. Wheeler, Illinois Power & Light Co.

PUBLIC ATTENDEES

268TH ACRS MTG.

Friday, August 13, 1982

J. Joostm, LECo
R. O. Sharpe, Duke power Company
M. Simons, ARC
C. Baty, Bechtel
R. G. Smith, SCP
M. L. Ryan, McGraw Hill
R. Borsum, Babcock & Wilcox
R Leyse, Electric Power Research Inst.
C. E. Ader, Stone & Webster
R. J. Ross, Dames & Moore
P. Tramblay, NUS
J. McEwen, TSI

APPENDIX A
FUTURE AGENDA

SEPTEMBER

Reactor Pressure Vessel Thermal Shock--ACRS discussion of proposed NRC Staff position regarding resolution of problems associated with repressurization of reactor pressure vessels following blowdown transients	Deferred to Oct
Naval Reactors Program Policies/Practices--Meeting with Admiral Kinnard R. McKee	1 hr
NRC Human Factors Program Plan--ACRS comments regarding proposed plan of action	1 hr
Consideration of Class 9 Accidents--ACRS discussion regarding proposed NRC Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation	
Quantitative Safety Goals--proposed Implementation Plan	
10 CFR Part 71--Packaging of Radioactive Material for Transport--ACRS comments Packages for Shipment of Radioactive Materials--ACRS comments regarding NRC procedures for review and certification	2 hrs
Nature of Installation of Liquid-Level Instrumentation--ACRS generic comments regarding the manner in which dp cells are installed, particularly in BWR's	

APPENDIX A (Cont.)

Washington Nuclear Plant Unit 2--OL	Deferred to Oct
Clinch River Breeder Reactor--Discussion of additional ACRS action prior to CP review	1/4 hr

Subcommittee Reports

Subcommittee on Reactor Operations regarding NRC enforcement policies, IE regionalization, and meeting with PROS representatives (JE/RKM)	
Subcommittee on Regulatory Activities regarding review of Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants" (CPS/SD)	1/2 hr
Meeting with NRC Commissioners	2 hrs

. Discuss proposed NRC Quantitative Safety Goals and implementation plan per ACRS report of June 9, 1982

. Discuss proposed NRC Safety Research Program Budget for FY 1984-85 per ACRS report of July 1982

. Discuss ACRS recommenation in several recent project reports regarding consideration of seismic events beyond the SSE in the design of nuclear power plants

. Discuss the proposed NRC policy statement regarding consideration of severe accidents in the regulatory process per discussion during 268th ACRS Meeting

APPENDIX A (Cont.)

Briefing by NRC Staff regarding status of PORV's in CESSAR-80 type plants--Discuss ACRS letter of April 6, 1982 (see Attachment 2)

Future Activities

Meeting of ACRS-RSK October 5-6, 1982

- Use of PRA and Quantitative Safety Goals in the design and regulation of nuclear power plants - Lead ACRS Member: D Okrent

- Recent or proposed changes in safety-related policy including items such as consideration of Class-9 accidents, design of nuclear plants - Lead ACRS Members: W. Kerr

- Recent or proposed changes in safety related technology such as use of the DEPB as the basis for limited plant features, prevention of PST, etc. - Lead ACRS Members: M. Bender, P. Shewmon

- Status of activities regarding radwaste management and disposal - Lead ACRS Member: D. Moeller

The ACRS Executive Director requested copies of papers by Members by September 17, 1982 or earlier (by September 11, 1982) if possible, to be reproduced and sent to Germany.