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FEBRUARY 4-6, 1982
WASHINGTON, DC

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APPENDIXES TO MINUTES OF THE 262ND ACRS MEETING
FEBRUARY 4-6, 1982
WASHINGTON, DC

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

Revision

January 29, 1982

SCHEDULE AND OUTLINE FOR DISCUSSION
262ND ACRS MEETING
FEBRUARY 4-6, 1982
WASHINGTON, DC

Thursday, February 4, 1982, Room 1046, 1717 H Street, NW, Washington, DC

1) 8:30 A.M. - 8:45 A.M. Chairman's Report (Open)
- 1.1) Opening Remarks
- 1.2) Report regarding matters which impact on ACRS activities
  - 1.2-1) Ginna steam generator tube failure

2) 8:45 A.M. - 11:30 A.M. Quantitative Safety Goals (Open/Closed)

Tab 2 ------
- 2.1) 8:45 A.M.-9:00 A.M.: Report of ACRS Subcommittee (DO/JMG)
- 2.2) 9:00 A.M.-10:30 A.M.: Reports of and discussion with representatives of the NRC Staff regarding proposed quantitative safety goals for nuclear facilities
- 2.3) 10:30 A.M.-11:30 A.M.: Discuss proposed ACRS interim position/action regarding quantitative safety goals

11:30 A.M. - 12:00 Noon 1.3) Proposed NRC Staff action re reactor pressue vessel liquid level instrumentation

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

3) 1:00 P.M. - 4:00 P.M. NRC Severe Accident Rulemaking (Open)

Tab 3 ------
- 3.1) 1:00 P.M.-1:30 P.M.: Combined Report of ACRS Subcommittees on Regulatory Safety Philosophy/Criteria and Class 9 Accident regarding proposed NRC policy to substitute specific standard plant rulemaking for the generic severe accident rulemaking (WK/DO/RS/JMG/SKB)
- 3.2) 1:30 P.M.-3:00 P.M.: Presentation by and discussion with representatives of the NRC Staff regarding SECY-82-1, Severe Accident Rulemaking and Related Matters
- 3.3) 3:00 P.M.-4:00 P.M.: Discuss proposed ACRS position/action

4) 4:00 P.M. - 6:30 P.M. NRC Safety Research Program Budget (Open/Closed)

4.1) Discuss proposed ACRS report to the U.S. Congress regarding the proposed NRC safety research program budget for FY 1983 (CPS et al./SD et al.)

5) 6:30 PM. - 7:00 P.M. General Discussion (Open/Closed)

Tab ---------5.1) NRC Implementation of ACRS recommendations regarding the composition of safety review committees at nuclear power stations (DAW/RKM)

Tab ---------5.2) Activities of ACRS members' participation in meetings which are not sponsored by the ACRS (HWL/RFF)

Friday, February 5, 1982, Room 1046, 1717 H Street, NW, Washington, DC

6) 8:30 A.M. - 9:30 A.M. — Nuclear Regulatory Reform (Closed)
   - 6.1) 8:30 A.M.-9:00 A.M.: Briefing by NRC Task Force Chairman regarding proposed changes in the NRC regulatory process
   - 6.2) 9:00 A.M.-9:30 A.M.: Discuss proposed ACRS comments/action

7) 9:30 A.M. - 10:30 A.M. — Discuss items for Meeting with NRC Commissioners (Open/Closed)
   - 7.1) Discuss proposed ACRS interim position/action regarding:
     - 7.1-1) Proposed Quantitative Safety Goals (DO)
     - 7.1-2) Proposed NRC policy regarding severe accident rulemaking (WK)
     - 7.1-3) Proposed Regulatory Reform (MB)

8) 10:30 A.M. - 12:00 Noon — Meeting with NRC Commissioners (Rm. 1130-H) (Open/Closed)
   - 8.1) Discuss items noted above

12:00 Noon - 1:00 P.M. — LUNCH

9) 1:00 P.M. - 2:15 P.M. — NRC Policy and Planning Guidance (Open)
   - 9.1) 1:00 P.M.-2:15 P.M.: Presentation by and discussion with the Director, OPE and the NRC Executive Director for Operations regarding NRC Policy and Planning Guidance for FY 1983-87

10) 2:15 P.M. - 2:30 P.M. — Future ACRS Activities (Open)
   - 10.1) Discuss anticipated ACRS subcommittee activities
   - 10.2) Discuss proposed ACRS activities

11) 2:30 P.M. - 5:30 P.M. — NRC Reactor Safety Research Program (Open/Closed)
   - 11.1) Discuss proposed ACRS report to the U.S. Congress regarding the proposed NRC Safety Research Program budget for FY 1983 (CPS et al./SD et al.)

12) 5:30 P.M. - 7:00 P.M.. General Discussion (Open/Closed)

12.1) 5:30 P.M.-6:30 P.M.: ACRS Reports to NRC -Discuss proposed ACRS comments/reports regarding:

12.1-1) NRC policy on the severe accident rulemaking

12.1-2) Nuclear Regulatory Reform

12.2) 6:30 P.M.-7:00 P.M.: Discuss proposed ACRS position/future action regarding:

12.2-1) Quantitative safety goals

Saturday, February 6, 1982, Room 1046, 1717 H Street, NW, Washington, DC

13) 8:30 A.M. - 9:30 A.M.. ACRS Reports to NRC - Complete discussion of ACRS comments/reports to NRC (Open/Closed)
- 13.1) NRC Policy on the severe accident rulemaking
- 13.2) Regulatory Reform

14) 9:30 A.M. - 11:00 A.M. NRC Long Range Research Program Plan (Open/Closed)
- 14.1) 9:30 A.M.-10:00 A.M.: Subcommittee report regarding proposed NRC Long Range Research Program Plan (CPS/SD)
- 14.2) 10:00 A.M.-11:00 A.M.: Discuss ACRS interim position/future action

15) 11:00 A.M. - 1:30 P.M. NRC Safety Research Program (Open/Closed)
- 15.1) Discuss proposed ACRS report to the U.S. Congress regarding the NRC Safety Research Program Budget for FY 83 (CPS et al./SD et al.)

(Note: Portions of the meeting noted above will be closed as necessary to discuss information the premature release of which would be likely to significantly frustrate proposed agency action [5 U.S.C. 552b(c)(9)(B)]; matters which relate to the personnel practices of the agency [5 U.S.C. 552b(c)(2)]; and information of a personal nature where disclosure would represent a clearly unwarranted invasion of personal privacy [5 U.S.C. 552b(c)(6)].)

Kelly

For the Nuclear Regulatory Commission.
William H. Regan, Jr.,
Chief, Siting Analysis Branch, Division of Engineering.
[FR Doc. 82-2228 Filed 1-27-82; 8:45 am]
BILLING CODE 7590-01-M

Advisory Committee on Reactor Safeguards, Subcommittees on Metal Components and Waste Management; Meeting

The ACRS Subcommittees on Metal Components and Waste Management will hold a meeting on February 12, 1982, Room 1046, 1717 H Street, NW., Washington, DC. The Subcommittees will discuss the technical aspects of proposed research efforts to predict high-level radioactive waste container long term (1000 yr.) integrity by accelerated methods as well as the technical capability of various potential contractors.

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 members 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 Designated 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 information and industrial security. One or more closed sessions may be necessary to discuss such information. (Sunshine Act Exemption 4.) To the extent practicable, these closed sessions will be held so as to minimize inconvenience to members of the public in attendance.

The agenda for subject meeting shall be as follows:

Friday, February 12, 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 the technical aspects of various proposals submitted to the NRC and the capabilities of the various organizations that submitted proposals.

The Subcommittee will then hear presentations by and hold discussions with representatives of the NRC Staff, their consultants, and other interested persons regarding this review.

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 therefor can be obtained by a prepaid telephone call to the cognizant Designated Federal Employee, Mr. Elpidio Igne (telephone 202/634-1414) between 8:15 a.m. and 5:00 p.m., EST.

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 information and industrial security. The authority for such closure is Exemption (4) to the Sunshine Act, 5 U.S.C. 552b(c)(4).

Dated: January 21, 1982.
John C. Hoyle,
Advisory Committee Management Officer.
[FR Doc. 82-2229 Filed 1-27-82; 8:45 am]
BILLING CODE 7590-01-M

Advisory Committee on Reactor Safeguards, Nuclear Regulatory Commission; Meeting

In accordance with the purposes of Sections 29 and 182b. of the Atomic Energy Act (42 U.S.C. 2039, 2232 b.), the Advisory Committee on Reactor Safeguards will hold a meeting on February 4-6, 1982, in Room 1046, 1717 H Street, NW., Washington, DC. Notice of this meeting was published in the Federal Register on January 20, 1982.

The agenda for the subject meeting will be as follows:

Thursday, February 4, 1982

8:30 a.m.-8:45 a.m.: Opening Session (Open)—The Committee will hear and discuss the report of the ACRS Chairman regarding miscellaneous matters relating to ACRS activities.

8:45 a.m.-12:00 Noon: Quantitative Safety Goals for Nuclear Power Plants (Open/Closed)—The Committee will hear and discuss the report of its Subcommittee and consultants who may be present and a presentation by representatives of the NRC Staff regarding a proposed NRC policy statement on quantitative safety goals to be used in the regulation of nuclear power plants. Representatives of the nuclear industry will present comments regarding this subject as appropriate.

1:00 p.m.-4:00 p.m.: Severe Accident Rulemaking and Related Matters (Open/Closed)—The Committee will hear and discuss the report of its Subcommittee and consultants who may be present and a report by members of the NRC staff regarding the proposed substitution of specific standard plant rulemaking proceedings for the NRC generic severe accident rulemaking. Representatives of the nuclear industry may present comments as appropriate.

4:00 p.m.-6:30 p.m.: NRC Safety Research Program (Open/Closed)—The ACRS members will discuss the proposed Committee report to the United States Congress regarding the proposed NRC safety research program budget for Fiscal Year 1983. Representatives of the NRC Staff will participate as appropriate.

Friday, February 5, 1982

8:30 a.m.-9:30 a.m.: NRC Regulatory Reform (Closed)—The ACRS will hear and discuss a report regarding activities of the NRC Regulatory Reform Task Force from the Chairman of the Task Force.

9:30 a.m.-10:30 a.m.: General Discussion (Open/Closed)—The members of the Committee will discuss interim comments of the members and/or areas needing clarification with regard to the following items scheduled for discussion with the NRC Commissioners:

- Quantitative safety goals for nuclear power plants.
- Proposed NRC policy regarding the severe accident rulemaking.
- NRC regulatory reform.

10:30 a.m.-12:00 Noon: ACRS Meeting with NRC Commissioners (Open/Closed)—The Committee will meet with the NRC Commissioners to discuss the topics noted above.

1:00 p.m.-2:15 p.m.: NRC Policy and Program Guide (Open)—The Committee will hear and discuss a presentation by NRC officials regarding the recent Policy and Program Guidance promulgated by the NRC Commissioners.

2:15 p.m.-2:30 p.m.: Future ACRS Activities (Open)—The Committee will discuss proposed and anticipated subcommittee and full Committee activity.

2:30 p.m.-5:30 p.m.: NRC Safety Research (Open/Closed)—The ACRS members will discuss the proposed Committee report to the United States Congress regarding the proposed NRC safety research program budget for Fiscal Year 1983. Representatives of the NRC Staff will participate as appropriate.

5:30 p.m.-6:15 p.m.: Reports of ACRS Subcommittees (Open)—The Committee will hear and discuss the reports of ACRS Subcommittee chairmen with respect to activities related to quality assurance deficiencies at the Zimmer

Nuclear Power Station and interpretation by the NRC Staff of ACRS recommendations regarding the composition of licensee's safety review committees.

Saturday, February 6, 1982

8:30 A.M.–10:30 A.M.: NRC Safety Research Program (Open/Closed)—The ACRS members will discuss the proposed Committee report to the United States Congress regarding the proposed NRC safety research program budget for Fiscal Year 1983.

10:30 A.M.–12:30 P.M.: General Discussion (Open/Closed)—The Committee will discuss proposed ACRS comments/recommendations and additional committee action regarding topics discussed during this meeting including:

- Quantitative safety goals.
- NRC Policy regarding the severe accident rulemaking.

1:30 P.M.–3:00 P.M.: General Discussion (Open/Closed)—The Committee will discuss proposed ACRS comments/recommendations and additional Committee activities regarding topics discussed during this meeting including:

- NRC Regulatory Reform.
- NRC Policy and Program Guidance.

Activities of individual members of the Committee will also be discussed.

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on September 30, 1981 (46 FR 47903). In accordance with these procedures, oral or written statements may be presented by members 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 Committee, its consultants, and Staff. Persons desiring to make oral statements should notify the ACRS Executive Director as far in advance as practicable so that appropriate arrangements can be made to allow the necessary time during the meeting for such statements. Use of still, motion picture and television cameras during this meeting may be limited to selected portions of the meeting as determined by the Chairman. Information regarding the time to be set aside for this purpose may be obtained by a telephone call to the ACRS Executive Director (R. F. Fraley) prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chariman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with the ACRS Executive Director if such rescheduling would result in major inconvenience.

I have determined in accordance with Subsection 10(d) P.L. 92–463 that it is necessary to close portions of this meeting as noted above to discuss matters which relate solely to the internal personnel rules and practices of the agency (5 U.S.C. 552b(c)(2)), information of a personal nature where disclosure would constitute unwarranted invasion of personal privacy (5 U.S.C. 552b(c)(6)) and information the premature release of which would be likely to significantly frustrate proposed agency action (5 U.S.C. 552b(c)(9)(B)).

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 therefor 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. EST.

Dated: January 22, 1982.

John C. Hoyle,

Advisory Committee Management.

[FR Doc. 82–2230 Filed 1–27–82; 8:45 am]

BILLING CODE 7590-01-M

OFFICE OF MANAGEMENT AND BUDGET

Agency Forms Under Review

January 20, 1982.

Background

When executive departments and agencies propose public use forms, reporting, or recordkeeping requirements, the Office of Management and Budget (OMB) reviews and acts on those requirements under the Paperwork Reduction Act (44 U.S.C., chapter 35). Departments and agencies use a number of techniques including public hearings to consult with the pubic on significant reporting requirements before seeking OMB approval. OMB in carrying out its responsibility under the act also considers comments on the forms and recordkeeping requirements that will affect the Public.

List of Forms Under Review

Every Monday and Thursday OMB publishes a list of the agency forms received for review since the last list was published. The list has all the entries for one agency together and grouped into new forms, revisions, extensions (burden change), extensions (no change), or reinstatements. The agency clearance officer can tell you the nature of any particular revision you are interested in. Each entry contains the following information:

The Name and telephone number of the agency clearance officer (from whom a copy of the form and supporting documents is available)
The office of the agency issuing this form
The title of the form
The agency form number, if applicable
How often the form must be filled out
Who will be required or asked to report
The standard industrial classification (SIC) codes, referring to specific respondent groups that are affected
Whether small businesses or organizations are affected
A description of the Federal budget functional category that covers the information collection
An estimate of the number of responses
An estimate of the total number of hours needed to fill our the form
An estimate of the cost to the Federal Government
An estimate of the cost to the public
The number of forms in the request for approval
An indication of whether Section 3504(h) of Pub. L. 96–511 applies
The name and telephone number of the person or office responsible for OMB review and
An abstract describing the need for and uses of the information collection.

Reporting or Recordkeeping requirements that appear to raise no significant issues are approved promptly. Our usual practice is not to take any action on proposed reporting requirements until at least ten working days after notice in the **Federal Register**, but occasionally the public interest requires more rapid action.

Comments and Questions

Copies of the proposed forms and supporting documents may be obtained from the agency clearance officer whose name and telephone number appear under the agency name. The agency clearance officer will send you a copy of the proposed form, the request for clearance (SF83), supporting statement, instructions, transmittal letters, and other documents that are submitted to OMB for review. If you experience difficulty in obtaining the information you need in reasonable time, please advise the OMB reviewer to whom the report is assigned. Comments and questions about the items on this list should be directed to the OMB reviewer or officer listed at the end of each entry.

Issue Date: 8/17/82

CERTIFIED

MINUTES OF THE
262ND ACRS MEETING
FEBRUARY 4-6, 1982
WASHINGTON, DC

The 262nd meeting of the Advisory Committee on Reactor Safeguards, held at 1717 H St. N.W., Washington, DC was convened by Chairman P. Shewmon at 8:30 a.m., Thursday, February 4, 1982.

[Note: For a list of attendees, see Appendix I. M. S. Plesset was unable to attend the meeting due to illness; H. W. Lewis was not in attendance on 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 (FACA) and the Government in the Sunshine Act (GISA), Public Laws 92-463 and 94-409, respectively. He noted that no requests had been received from members of the public to present either written or oral 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 Co., Inc., 400 Virginia Ave. S.W., Washington, DC 20024.]

I. Chairman's Report (Open to Public)

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

The Chairman informed the Committee that he did not have any specific statements to make at this time.

II. <u>Ginna Accident of January 25, 1982</u>

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

R. C. Haynes, NRC Region I, discussed offsite releases, the preliminary sequence of events which occurred, and institutional responses to the Ginna event (see Appendix IV). It was noted that the initial leak rate was of the order of 700 gallons per minute; that the reactor scrammed and the safety injection system actuated. Mentioned were certain lessons or questions which arose from the incident.

- Should the reactor coolant pumps be turned off manually in this type of accident?

- Should there be a procedure to throttle back safety injection pumps to avoid main steam relief/safety valves opening?

- Should the block valve on the atmospheric dump be open or closed during normal operation?

C. P. Siess questioned whether the Ginna plant had written operating procedures covering this type of transient including PORVs sticking open. R. C. Haynes indicated that they did have procedural criteria to train operators how to respond. It was the belief of C. Mark that declaring a site emergency was an overreaction by the licensee. R. C. Haynes explained that the plant superintendent was concerned about the potential for an offsite release when he declared an emergency on the site. He indicated that that is in accordance with NRC criteria.

III. Quantitative Safety Goals

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

A. Safety Philosophy, Technology and Criteria Subcommittee Report

D. Okrent expressed his belief, based on a discussion at the Subcommittee Meeting with S. H. Hanauer, that NRR has no current point of view for implementing the safety goal as documented in SECY 82-1, Severe Accident Rulemaking and Related Matters. W. Kerr expressed concern that SECY 82-1 did not provide a satisfactory description of how numerical guidelines were derived. He thought that some additional discussion during the OPE presentation concerning the guidance in the development of these numbers would be very useful to the Committee. M. Bender indicated that it was difficult from the writing in SECY 82-1 to interpret and implement these numbers in probabilistic terms once they are endorsed.

B. Status and Features of Proposed NRC Policy Statement on Safety Goals

F. Remick of OPE indicated that while the Commission has not approved the proposed Policy Statement on Safety Goals for Nuclear Power Plants, there is a good chance that it will be released for public comment at the meeting being held by the Commissioners today. The Commissioners have had the draft Policy Statement since mid November supported by a discussion paper which will come out as NUREG-0880. It is expected that once approved, the document will go out for a 90 day public comment period. The Commissioners are expected to sanction 3 or 4 public information meetings to be held in different parts of the country as briefings on the safety goal. F. Remick presented background information on the preliminary development of the proposed Policy Statement (see Appendix V).

F. Remick outlined the substance of the draft Safety Goal Policy Statement (see February 4 draft, Appendix VI). He explained that the policy statement will state the Commission's views on the acceptable level of risk to the public health and safety and on safety cost tradeoffs in regulatory decision making. The Policy Statement will focus on nuclear accidents. It does not deal with the risk from routine emissions from the nuclear fuel cycle, from sabotage or earthquakes or from diversion of nuclear weapons grade material.

F. Remick explained that this Commission proposal would adopt two qualitative safety goals supported by provisional numerical guidelines. The two qualitative safety goals proposed are entitled <u>Individual Risk</u> and <u>Societal Risk</u> (see Appendix VII). Two provisional guidelines are being proposed. The first guideline, which refers to prompt fatalities, limits the risk to an individual or to the population in the vicinity of a nuclear power plant from reactor accidents to a level not to exceed 0.1% of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed. The second guideline limits the risk to an individual or to the population in the area near a nuclear power plant site from cancer fatalities that might result from reactor accidents to a level not to exceed 0.1% of the sum of cancer fatality risks resulting from all other causes.

H. W. Lewis requested clarification of whether the term "should not exceed" which is in both guidelines suggests that numbers be calculated conservatively or realistically when evaluating compliance with the safety goal. F. Remick indicated that the Commission meant best estimate, not conservative.

C. Mark and H. W. Lewis expressed concern with the "anti-intellectual tone of implementation" which suggests that the proposed numerical guidelines should replace judgment with mathematical formulas rather than aid professional judgment in the decision making process. C. Mark suggested that engineering judgment should not be put aside unless you can demonstrate that a formula is correct. H. W. Lewis felt that it would be probably better to say that "judgment should not be replaced by mathematical formulas". M. Bender suggested that he would be more convinced of the validity of implementation of the safety goals if OPE would go through one PRA exercise completely to establish a frame of reference. D. Bradford of OPE indicated that there was an appendix in the discussion paper of October 1, 1981, NUREG-0880, which did explain the results of reactor risk assessment. He indicated that OPE did not follow the illustration through to completion because their objective was to establish what might constitute acceptable risks as opposed to guidelines for doing PRAs. Of import was the use of PRA with the safety goal guidelines.

F. Remick stated that OPE was not intending in the policy statement that each licensee must do a complete probabilistic risk assessment. However, D. A. Ward pointed out that it was not possible to compare a plant with the goals and guidelines without doing a full blown PRA on the plant.

D. W. Moeller was concerned that in the consideration of societal risk to life and health, genetic risks were not explicitly handled. This subject would be a prime candidate for challenge by reviewers. Similarly, it was his judgment that the safety goals concentrate on the risk from the nuclear plant itself implying that the risk from the nuclear plant far exceeds the risk from other steps in the fuel cycle. This concept could also be easily challenged. F. Remick agreed that considering the nuclear plant only was a judgment call, and was limited from a policy standpoint because OPE did not fully examine the rest of the fuel cycle in alternative risks or in the nuclear risk itself. F. Remick attempted to explain some whether the policy statement in evaluating alternative risks vs. nuclear risks was comparing accidents to accidents or operations and accidents to accidents.

H. Etherington expressed concern that the safety goal dealing with the core melt probability did not take into account factors such as containment reliability or emergency procedures which impact on the total risk from a core melt. D. A. Ward pointed out that without the guideline on core melt, licensee actions to meet the safety goal would almost entirely be trended toward mitigation rather than prevention. Inclusion of a core melt probability forces a split between prevention and mitigation. R. C. Axtmann pointed out that the $1000 per man-rem cost benefit guideline would tend to encourage licensees to build nuclear plants in areas of highest population density or to encourage population growth to get maximum benefit from the dollars spent to reduce exposure. D. Rathbun replied that an applicant would have to spend funds for "reduction credits" before he received his license in the first place if he were proposing to place his plant in a high population density area.

W. Kerr expressed skepticism in the ability of NRC to demonstrate achievement of the safety goal in dealing with such small numbers as 0.1% increase in cancer risk when these very small numbers are competing with normal variations in the environment. D. Okrent expressed specific concerns regarding omission from the safety goal of risks such as sabotage and earthquakes. H. W. Lewis and W. Kerr were particularly disturbed by the terminology referring to maximum risk to an exposed individual and how the NRC could avoid this being interpreted as the maximum risk to the most exposed individual. F. Remick attempted to clarify the situation when he defined the person at risk as the maximum of the average individual. D. Rathbun added that the guideline did not mean the worst case of individual risk, but individual risk as applied to a biologically average individual in terms of age and other risk factors. The discussion that followed attempted to clarify the terminology of the guideline.

D. Okrent explained that his interpretation of the earlier comments by H. Etherington and D. A. Ward concerning mitigation vs. prevention actually pointed toward a possible performance criterion on containment in the policy document.

F. Remick responded to certain additional questions that dealt with benefit cost trade-offs and a guideline on availability of containment function. A suggestion and questions concerning implementation of a specific provision for risk aversion were presented (see Appendix VIII). One question regarding implementation concerned the approach to take with respect to accident initiators which are more difficult to quantify. D. Okrent again questioned why seismic events and sabotage were specifically excluded, while other accident initiators were not. F. Remick suggested that the exclusions were made because Staff experts consulted during the formulation of the safety goal indicated that these items could not at this time be properly quantified. Therefore, they were excluded from the risk calculation.

C. P. Siess suggested that the quantitative safety goal be modified to compensate for the significant accident initiators which have been excluded from the calculation of risk. He pointed out that one now has an incomplete mathematical equation with the risk side not complete. M. Bender was particularly critical of the $1000 per manrem ALARA guideline, which implies that reducing manrems would reduce the likelihood of cancer by some increment. He noted that this concept did not take account of personnel exposures in the work environment. He felt that "the concept was so full of errors, inaccuracies, misjudgments and statistics having no validity" that the computational procedure would not have much worth.

H. W. Lewis expressed personal opposition to the use of ALARA in the policy statement because he believed that PRA should be used internally in constructing deterministic regulations for reactors. He explained, through an illustrative calculation exercise, that the combination of the ALARA criterion with the 0.1% societal risk criterion implied an enormous financial burden on society that is unsupportable.

C. Committee Discussion

Some Committee Members expressed concern regarding the exclusion of earthquake, sabotage, design errors and multiple human failures from common cause errors in the policy statement. C. Mark and D. A. Ward were particularly concerned that the safety goal did not take into account occupational exposure to workers at the nuclear plant which might completely overshadow the nonoccupational man-rems.

D. W. Moeller asked the Executive Director if the ACRS had clear-cut guides for how and when the Committee interacts on a policy statement that the Commission is developing. R. F. Fraley responded that there were no clear-cut guides regarding this subject. M. Bender welcomed the chance to provide comments to OPE even at this interim stage. R. F. Fraley indicated that the meeting of OPE with the Commissioners scheduled for Friday (February 5, 1982) is planned as an initial briefing and discussion of the safety goals.

S. Hanauer, NRC Staff, indicated that it was his opinion that the Staff would use the safety goal as one factor in the decision making process, utilizing whatever probabilistic risk assessment numbers are currently available at the time. R. Mattson, NRR, suggested that his Division might evaluate sample problems to test compliance with the safety goal during the public comment period. C. P. Siess questioned how an intervenor might use the safety goal in a hearing process. R. Mattson suggested that an intervenor might use the safety goal to attack and quantify a weak design point such that the Staff may be forced to use the safety goal to refute such arguments. In fact, R. Mattson thought that this would probably be the most likely place for the safety goal to be utilized - in the case of an appeal brought to the NRC. The Committee decided not to write a letter at this time regarding the safety goal policy statement.

IV. Liquid Level Instrumentation

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

R. Mattson of NRR brought the Committee's attention to a January 29, 1982 memo from W. J. Dircks to Chairman Palladino which had an attached two page enclosure entitled Additional Instrumentation for Inadequate Core Cooling of PWRs (see Appendix XIX). He explained that this attachment was a description of topics of interest to be taken up at a meeting scheduled for February 16 and 17 with representatives of the designers and manufacturers of liquid level indicators. M. Bender indicated that he had scanned this document and the attachment and felt that it did not bring forth his concern which was that the Staff should take some position to explicitly define for licensees the limitations on use of liquid level indicators. R. Mattson indicated that the Staff is requiring specific information about the performance of liquid level indicators in all three PWR Vendors' Emergency Procedure Guidelines so operators will know when they should or should not rely on them.

R. Mattson discussed errors in BWR vessel water level indication. For most operating BWRs their liquid level indicators will fail during certain depressurization transients including some design basis accidents. He indicated that the Staff is now aware of considerable work by General Electric (GE) in this area and has made sure that letters written to BWR owners caution them not to rely on these indicators under certain accident conditions and recommend appropriate cautions in their operating procedures. A design modification made to the liquid level indicators in the Edwin I. Hatch Unit 2 by GE to fix this problem is being considered for backfitting plants licensed before Hatch 2. R. Mattson indicated that the Staff's attention was drawn to the subject of level indication after noting the attitude of GE and the Owners Group regarding insulation of core exit thermocouples using PORVs as required by Regulatory Guide 1.97 and the TMI Action Plan. A presentation by GE to the NRC has caused the NRC to, in essence, abandon core exit thermocouples for boiling water reactors for the next six months.

R. Mattson and E. J. Ebersole discussed the merits of PORVs as devices for decay heat removal. J. Ebersole noted that the supplemental safety evaluation report on Palo Verde and CESSAR-80 suggests that PORVs are not required. R. Mattson noted that these supplements were withdrawn the day after the Ginna transient resulting from steam generator tube failures occurred. The value of a PORV in the case of simultaneous tube failures was also a factor in withdrawal of the supplements. R. Mattson continued that after study of the Ginna design, it can be pointed out that they were unable to keep from lifting the safety valves of the secondary side of a faulty steam generator even with PORVs.

R. Mattson indicated that analysis of the LOFT tests has indicated that the reactor coolant pumps should be turned off quickly during a LOCA, perhaps too fast for reliable operator action according to the traditional guidelines for operators. The indication is that one ought to make the reactor coolant pump trip automatic. R. Mattson mentioned that he was still not convinced that automatic coolant pump trip on PWRs was necessarily correct. Especially from analysis of the Ginna accident in which there are conflicting interpretations of the course of the accident and potential operator actions.

M. Bender expressed his concern that placing too many requirements on the reactor operators will tend to confuse them with contradictory procedures. R. Mattson indicated that the best information available on guidance for operator action is the draft ANSI Standard N660 which is meant to be applied to new plants. He suggested that it is certainly a better basis than "off the cuff" judgment.

The discussion of PORVs turned to the issue of Palo Verde and CESSAR. R. Mattson indicated that a supplemental evaluation report on this issue will not be coming from the Staff since NRC has turned the matter over to Combustion Engineering (CE). It is for CE to show the NRC why it should not add PORVs to its designs. M. Bender pointed out that the issue was not just installation of a PORV but more understanding of how fast the system has to be depressurized.

V. Severe Accident Rulemaking

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

A. Safety Philosophy, Technology and Criteria/Class 9 Subcommittee Report

W. Kerr referred to a presentation at the February 3, 1982 Subcommittee Meeting by R. Mattson concerning background on SECY-82-1, Severe Accident Rulemaki g and Related Matters. He discussed a handout at the Subcommittee Meeting which described a research program meant to deal with questions raised by the Commission concerning the severe accident problem (see Appendix X). D. Okrent added that the Subcommittee members had suggested to R. Mattson that he concentrate on developing his thinking along the lines of the memo from Chilk to Dircks (see Appendix X).

B. Discussion of SECY-82-1

R. Mattson proceeded to discuss the ten comments by the Commissioners included in the Chilk to Dircks memorandum of January 29, 1982 (see Appendix X). In explaining the first item, which refers to ensuring that conflicting or incorrect signals are not sent to industry relating to significant matters contained in the long term rulemaking proceedings, R. Mattson referred to the specific list of potential design changes on page 2 of SECY 82-1. There is a set of three possible signals the Staff might give to indicate how the items in this list should be treated in future reactor designs. In referring to the subject of filtered venting of containment, he suggested a first signal that indicates a high degree of interest by the Staff. Another signal that was discussed would say that the licensees must have this feature in their designs.

The third signal, which would apply to most of the items in the listing, would indicate that the Staff is still studying the matter and the applicant must consider this matter in the design of a plant for future application in the context of a safety goal and a probabilistic risk assessment. The applicant would submit a design with suggested features which could be evaluated using the $1000 per man-rem in the safety goal.

As an example, R. Mattson indicated that the matter of filtered venting of containment should be considered on both BWRs and PWRs and included by the applicant if cost effective in reducing risks. In referring to core retention devices, R. Mattson indicated that it was important that the Staff not give the incorrect signal so that applicants look only at magnesium oxide base mats and their cost effectiveness and discontinue the study of base mats entirely if this particular design is found not cost effective.

R. Mattson suggested that the second point in the Commission's memorandum, which refers to more specific guidance pertaining to design criteria considered necessary by the Commission, gives the Staff an opportunity to include in the revised policy statement guidance on minimum safety requirements for strong containments. He suggested that hydrogen control be tied to the strength of the containment, and in the meantime, the Staff will stick with the near-term CP rule or interim hydrogen rule.

M. Bender suggested that the Staff look into developing a siting approach that looks at the properties of sites in terms of their inherent ability to protect against accident contingencies that cannot be controlled readily. This siting approach would obviously not be very useful for existing plants.

R. Mattson indicated that decay heat removal should be considered by all licensees, and that the signal that should go out is that all unresolved safety issues are "fair game" at future CP hearings. In the case of post accident recovery plans, R. Mattson indicated that this matter is considerably less important than some others and the Staff would indicate that it is not very important. R. Mattson indicated that criteria for determining the location for placement of highly radioactive systems is very important and that would be the signal from the Staff. With respect to the item "Effects of Items at Multi-Unit Sites," the Staff will indicate that this is probably not that important a safety concern and is more of an economic problem for future designs. Therefore, it would be left to the utility to decide on such matters as common control rooms.

D. Okrent expressed concern that the Staff would have difficulty dealing with the very large variation in the past and expected in the future between different people trying to assess the same probabilistic risk. He questioned whether the approach would be viable if one had to reconcile the difference between the Staff, the applicant and the intervenor with respect to one's uncertainties in the PRA analyses.

C. RES Support of the Proposed Approach in SECY-82-1

D. Ross explained the Office of Research's intent in working with NRR to redefine the current draft of SECY 82-1. It would lead to a March subcommittee review and full Committee discussion in April. A four year $220 million research program was mentioned which D. Ross indicated would mainly address recommendations concerning the requirements for additional instrumentation - new and different instruments that could be more useful in a severe accident sequence.

D. ACRS Deliberations

W. Kerr applauded R. Mattson's recognition of the proposed rulemaking on degraded cores as a "morass". He recognized that there is a concensus in the Commission and the Committee that some new approach needs to be taken to resolve this question. W. Kerr suggested that SECY 82-1 in its original form does not give enough guidance to accomplish the task required. Whether the approach involves the rulemaking or use of the process of licensing of a reactor, it needs to be more specific to be a workable approach. M. Bender suggested a staged approach for dealing with the degraded core question, a graded approach in stages that could be related to various kinds of improvement actions. He mentioned three definite aspects of the degraded core event: (1) failure of the containment system; (2) metal/water reaction; (3) core melt.

M. Bender suggested that the Committee might write a letter that addressed whether the idea of a rule is good or not, whether it is a good idea to issue one at this time or later, whether the rule as written is properly structured to get a useful result, and what other concurrent actions should be taken while this rule is being promulgated.

D. A. Ward expressed concern as to whether there really is enough information available to use as the basis for a rule. He suggested that another framework other than rulemaking might be more appropriate. D. Okrent suggested that the Staff has not focused its own research program regarding this matter. He suggested that perhaps SECY 82-1 might be a good discussion piece but that the Staff position is premature and in need of further study. W. M. Mathis suggested that the Committee is not ready for a rulemaking, but, casting the problem as an intermediate policy for discussion purposes was a good approach. As a second comment, he suggested that related research, especially work connected with the area of fuel damage, has not been well received by the ACRS in the past.

R. Mattson asked the Committee for guidance to deal with the matter of the definition of a strong or vented containment. The Committee discussed the concept of containment with regard to the large scale fuel melt, hydrogen generation, steam release and release of radionuclides in an accident. P. G. Shewmon suggested that the different concepts and ideas generated in this discussion would form a sufficient basis for the drafting of a letter.

VI. Nuclear Regulatory Reform

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

James Tourtellotte, Chairman of the Regulatory Reform Task Force, briefed the Committee regarding legislative and procedural changes being considered by the Task Force.

With respect to nonmandatory ACRS review, J. Tourtellotte explained that although the ACRS is bound to a mandatory review by statute, that requirement does not state the level of review that is required, the depth of review, nor the scope of the review. He explained that in his opinion, the ACRS could, on its own initiative, establish priorities for review and criteria to support those priorities. Suggested was an arbitrary grading of level of review from A-Full; B-Modified; to C-Cursory (indication of no further review at this time). P. G. Shewmon suggested that by law the ACRS must then write a letter showing that the review done was consistent with whatever level ACRS had chosen. C. Mark noted that it would take the Committee considerable time to set a graded review level in particular instances and this would just add to the time of review. D. Okrent suggested that this really would not be that much of a problem because most of the reviews the Committee now does are B-level reviews, not A-level reviews. C-type reviews would be obvious special exceptions to the normal ACRS review process.

J. R. Tourtellotte suggested that the ACRS might be able to relax the stringency of review should it know that an applicant had demonstrated competence and experience in operating its existing nuclear facilities. M. Bender concurred that this was one important consideration but certainly not the only consideration in the determination of the planned level and depth of Committee review. C. P. Siess indicated that he could not visualize the Committee reducing its number of reviews or its scope of review simply because of a nonmandatory review provision unless the review schedule was tremendously burdensome. In fact, he suggested that he did not know of a licensing case where nonmandatory review would have helped alleviate the Committee's workload. J. R. Tourtellotte suggested that the grading of review scope mentioned above, even without legislation, could achieve the same result as if you had legislation changing things to a nonmandatory review.

J. R. Tourtellotte described the current legislation on regulatory reform as trended toward the concept of standardization. He listed three basic parts to the legislation.

- One-step licensing for the whole plant design prior to the beginning of construction

- A whole standardization design banked for 10 years, renewable at the end of 7 years

- Presite designation-banked sites for 10 years, also renewable at the end of a period of 7 to 9 years

J. R. Tourtellotte explained that the general problem was perceived as difficulty in constructing plants and getting them into operation in a reasonable length of time because the design of the plant actually goes on during the construction phase, the technical review and regulatory review of the NRC goes on during construction, and the hearing process goes on during construction. The objective of the legislation is to effectively remove the technical review, the design of the plant and the hearing process from the middle of the construction schedule. The legislative changes being considered take the regulatory process (regulatory review and the hearing process) out of the middle of the construction process and place them in front of it. All that would be left is the problem of inspections and tests to insure that the conceptual design is actually built and put into operation.

J. R. Tourtellotte discussed certain features of the banked site concept. He mentioned the design standardization sections of the proposed statute that allow for waiver of fees for applicants. He suggested that one important offshoot of standardization should be the development of a program of participatory design review which would actually involve the NRC Staff in the design process. He suggested that there be greater interfacing between the designer and the Staff and cited the General Electric/ architect/ engineer partnership and Westinghouse's attempts to design a whole plant island in cooperation with foreign governments. D. A. Ward indicated that he was quite skeptical of this approach of participatory whole plant design process. He suggested that this approach might require a revolution in current design, construction, and manufacturing processes.

J. R. Tourtellotte mentioned also a backfitting rule in the legislation for limiting backfitting to those items which provide an additional margin of safety which is more consistent with overall safety goals. J. R. Tourtellotte also mentioned that recommendations will be made for substantial changes in the conduct of the hearing process to make it considerably less formal and also increase the requirements to show that an actual safety problem exists as a matter to be resolved.

W. Kerr implied that delay in approval of the reactor design, because of the deliberate pace of the review and approval, might mean an obsolete plant when it was finally approved for construction by applicants. Since an applicant would be loath to make design changes which would hinder final approval of the plan, potentially, there could be a very obsolete plant going into operation although it might well be a very safe plant.

J. R. Tourtellotte mentioned several other points involved in the legislation.

- Use of regulations to implement one-step licensing in lieu of legislation

- Update of NUREG-0292, The Denton Report of 1977

- Study implementation of the participatory review concept and match design phase with review phase such that the reviewer is reviewing the appropriate part of the design phase

- Change Part 50.109, The Backfit Rule
- Divide the hearing process into two parts for better utilization of manpower - initial license cases and enforcement cases
- Take the National Environmental Policy Act (NEPA) out of the adjudicatory process
- Relax the ex parte rule

J. R. Tourtellotte indicated that it was extremely important that the ACRS attempt to implement the spirit of the legislation for nonmandatory review even without legislation. He indicated that the ACRS mandatory review requirement would be eliminated in the legislation except for standardized plants and the CRBR.

M. Bender suggested that the question of standardized plants only makes sense if there are many plants to be standardized. He suggested that standardization might impair the ACRS's ability to maintain its overview which is to protect the public health and safety. M. Bender questioned as an example, whether and how much the ACRS should be concerned about the effectiveness of the operating organization of the licensee.

VII. Meeting with NRC Commissioners

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

[Note: Chairman Nunzio Palladino and Commissioners V. Gillinsky, P. Bradford, J. Ahearne and T. Roberts were present.]

A. Steam Generator Tube Degradation

Chairman Palladino noted the possibility of forming a joint NRC/Industry Task Force to study the subject of steam generator tube degradation including a review of related research efforts. The objective of these efforts would be to ascertain interim and long-term measures which would be taken to mitigate the problem. Chairman Palladino suggested that the ACRS consider participation in the NRC portion of the task force. He indicated that the Committee would be kept informed regarding the formation of this task force.

B. Quantitative Safety Goals

Chairman Palladino expressed particular interest in ACRS comments pertaining to the ALARA criteria of $1000 per man-rem. C. Mark brought up a phrase on page 2 of the February 4, 1982 draft of the proposed Policy Statement which referred to "Diversion of nuclear weapons grade material". He suggested that the adjective "weapons grade" be removed from the document because of its specificity in other contexts. D. Okrent was especially concerned about the exclusion of seismic events from the current draft of the proposed Policy Statement. He indicated that from a technical point of view, this was an inappropriate exclusion because studies at CRBR, at Diablo Canyon, and other plants have found that about 90% of the risks come from seismic events. While there are large uncertainties in the methodology for evaluating seismic phenomena, they are not different from uncertainties in trying to deal with design errors or multiple human errors. C. P. Siess pointed out an ambiguity on page 2 of the February 4th draft which he interpreted as indicating that the safety goal does not deal with risks from earthquakes and sabotage, but with risks from accidents resulting from earthquakes and sabotage. He expressed his opinion that exclusion of earthquakes and sabotage along with routine emissions, diversions, and the fuel cycle from the PRA analysis does not exclude earthquakes and sabotage from being accident initiators. C. P. Siess pointed out that it was his understanding that the safety goal was set up such that a risk analysis would omit earthquakes and sabotage but be adjusted to take into account that possible initiators accounting for 90% of the risk are not included (e.g. the goal would be adjusted by a factor of 10 to include earthquake and sabotage risks).

F. Remick suggested that the objective of those statements was to identify and express the concern of the NRC for the inability to quantify earthquake and sabotage risk in a PRA.

Chairman Palladino and F. Remick recognized the inconsistency of the wording of this portion of the document. Chairman Palladino suggested that some clarification was needed to answer the questions that were raised about this matter. Commissioner Ahearne indicated that he was not comfortable with the exclusion of earthquake phenomena and he would prefer this item be struck as an exclusion from the February 4 draft. Chairman Palladino indicated that he was not sure where the phrase "or earthquakes" came up in discussions with the Staff but he acknowledged that it did not fit. Commissioner Gilinsky suggested that the Commission should "draw back and regroup its forces" regarding issuance of the safety goal for public comment.

D. W. Moeller pointed out that the safety goal does not address genetic effects. He said that the document should consider them and perhaps indicate that the latent cancers and immediate fatalities are controlling and therefore drop consideration of genetic effects. H. W. Lewis noted that it might not be appropriate for the inclusion of the ALARA principle in a quantitative safety goal. P. G. Shewmon noted that, in his opinion, discussions of the predictions of the effects of large earthquakes on plant operability is not very well developed. He suggested that the conservative approach taken in the Zion PRA might not be the most convincing to represent the probabilistic risk associated with seismic events.

D. Okrent pointed out an apparent deficiency in the document which would confuse the public. This is the fact that the goals are supposed to cover both existing plants, plants under various stages of construction, and future plants. There was no distinction made among them. He suggested that, if it were kept this way, plants at different stages of development could be easily subjected to the same kinds of changes in order to meet the safety goals rather than distinguish between existing and future plants. D. Okrent also expressed concern that, if sabotage were completely excluded from this document, this issue would not receive proper consideration or attention.

C. Proposed Policy Statement on Severe Accident Rulemaking

W. Kerr noted that the proposed policy statement on severe accidents does not deal with operating plants. He indicated that the ACRS might make some suggestions about how one might deal with operating plants and not wait for the problems to be settled with the licensing of Standard Plants. D. Okrent pointed out that he has not seen, up to this point, a convincing Staff effort to tackle the job of promulgating a severe accident rule. He suggested that the ACRS would be willing to set up a subcommittee to work with a group of suitably experienced staff people to assist in the drafting of such a rule. In addition, he added that he was referring to the rulemaking process and not the research programs that might be connected with this rulemaking.

D. Regulatory Reform Task Force

Chairman Palladino indicated that the Commission had not been properly briefed on the current status of regulatory reform and suggested that the ACRS submit written comments to the Commission. Commissioner Bradford suggested that these comments be provided before the Commission meeting the following week. M. Bender mentioned the recent Committee discussion with J. R. Tourtellotte. He pointed out that the concept of regulatory reform appeared to be centered around the idea of a very effective standardization program. Concern was expressed that experience with standardization in the U.S. and elsewhere has not been very well thought out as far as regulatory policy is concerned. Chairman Palladino suggested that the approach was primarily to create an opportunity for utilities or vendors to present standardized plant designs to the NRC. M. Bender suggested that the need to examine the plant operating organization should be considered in the development of regulatory reform regarding use of standard plants. In addition, he indicated that many of the Committee Members believe that there should be stronger interaction on significant rules by the ACRS and some interaction at the formative stage when the Staff is trying to define the scope of the rule and the reason for its promulgation. M. Bender suggested that the ACRS is entering the review process at too late a stage.

M. Bender brought up the matter of the ACRS role in this reformed regulatory process, specifically with respect to whether ACRS review should not be mandatory. He suggested that the Committee has never felt that it should be constrained in how it conducts its review action. The reason for the proposition of nonmandatory ACRS review refers to a past situation where the ACRS found itself inundated with review work and thought that a selective review approach would be more effective.

Commissioner Ahearne suggested that the ACRS should define for the Commission what it believes ought to be its role in the review process. P. G. Shewmon stated that during the discussion with J. R. Tourtellotte, the Committee decided that it was not interested in pressing the issue of optional review anymore because it wanted to stay involved in the review process. The Commissioners took note of this change in ACRS position regarding the need for a nonmandatory requirement, although H. W. Lewis did note that the Committee had not acted formally regarding this matter.

E. Ginna Nuclear Plant Steam Generator Tube Rupture on January 25, 1982

Chairman Palladino expressed the Commissioners' concern about the steam generator tube degradation problem and the potential for related accidents. He urged the Committee to submit written comments and advice regarding approaches for dealing with the

Ginna transient and its many ramifications. The Commissioners and the ACRS discussed whether the Ginna accident is a precursor of a serious generic problem and whether the subject is being adequately addressed at the present time.

Commissioners Gilinsky brought up the subject of reactor pressure vessel liquid level indicators. H. W. Lewis noted that his reservations regarding this matter were not objections to liquid level indicators per se but only to ambiguous level indicators. He explained that he supported a much simpler void meter which could help determine whether a void existed in the upper plenum rather than the proposed level indicators which will not be reliable in such a determination. M. Bender suggested that the Committee is not opposed to liquid level indicators but needs to know that they are usable and useful. This could be determined by postulating some of the scenarios in which these devices would be utilized to try to determine how the level indicators would function.

VIII. Policy and Planning Guidance, FY 1983 to 1987

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

F. Remick explained that the purpose of the Policy and Planning Guidance document is to enable the Commission to provide guidance to the Staff for establishing priorities and improving the regulatory process. He added that the document was to provide general guidance in areas of particular interest or concern to the Commission. It was not intended to be all inclusive. Mentioned were seven major themes in the document (see Appendix XI).

- Safe operation of licensed plants
- Near term licensing problems and responses
- Coordinating regulatory requirements
- Improving the licensing process
- Supporting new initiatives in nuclear waste management and the cleanup of TMI
- Improving related regulatory tools
- Safeguards

C. Mark questioned whether the Policy and Planning Guidance (PPG) addressed Commission responsibility for evaluating the technical competence of licensee management. F. Remick indicated that this was a difficult question for which he did not have an answer at this time. F. Remick proceeded to discuss the seven major themes of the document individually in some detail. Reference was made to the Task Force on Regulatory Reform under the theme of supporting new initiatives. C. Mark questioned the Staff position on one-step licensing in the PPG. F. Remick explained that it was his understanding that it related to the question of Standardized Plants and a CP/OL combination. W. M. Mathis expressed concern with the lengthy time schedules shown in the planning documents to TMI-2 cleanup and waste management. F. Remick traced the delays in the TMI-2 cleanup to limited financial resources. W. M. Mathis suggested that the financial situation would become more difficult the longer the NRC procrastinated regarding this matter.

C. P. Siess took exception to the policy statement on research at the top of page 23 of the PPG document (see Appendix XI). He suggested that the purpose stated "to assist and establish regulations for existing and future facilities" describes a research program that has been about "5% effective". He also suggested that the statement "emphasize support of the safety of operating reactors and other operating facilities" was ambiguous and not well related to the actual research program.

F. Remick summarized the major thrust of the PPG as dealing with continued vigilance over operating facilities, timely action on all regulatory decisions, resolution of safety issues in an expeditious fashion, elimination of the licensee action backlog, and improved management and simplification of the licensing process through legislative and administrative means without degradation of safety. C. P. Siess was concerned that certain ambiguities that he had found in the PPG document, such as the question about the policy on research, would necessitate a lengthy commentary for explanation to go along with the policy and planning document itself.

D. Okrent inquired as to how implementation of the safety goal would be accomplished through the policy and planning guidance. W. Dircks, EDO, explained that implementation would move down from the Commission, through the Office of the EDO, into the Office of the Directors of NRR and Research, and finally down to the Branches and Divisions. He indicated that until the safety goal was fully formulated, the Staff would probably deal with it in generalities with the primary application trending toward the Standardized Plant concept. D. Okrent noted his surprise that the Staff, which had proposed SECY 82-1, was not prepared at this meeting to inform the Committee of at least a preliminary plan as to its implementation. D. A. Ward expressed concern that CRBR and post-CRBR/NRC research has been excised from the FY 1984 - 1985 budget. W. Dircks explained that the policy and planning guide does recognize the limitations that have been placed on the budget this year and next year. He expects that there will be some residual funding for future research in this area depending upon agreements that can be reached concerning a budget request to the Office of Management and Budget.

D. W. Moeller presented several specific questions to W. Dircks which he indicated might have been directed to F. Remick. One of the questions addressed the fact that radiation protection was not addressed in the PPG, particularly occupational radiation exposure. D. W. Moeller referred to Item D on page 6 of the policy and planning guidance document which stated that NRC should require key licensee employees, including certain management and maintenance personnel, to be adequately qualified. He questioned whether management and maintenance personnel would include test personnel. F. Remick indicated that maintenance would not ordinarily include test personnel. A third question involved Item 6 on page 8 which needed clarification. This item referred to NRC working with the Federal Emergency Management Administration (FEMA) to resolve difficulties in securing the findings of offsite emergency plans for proposed nuclear plant sites in a timely fashion. F. Remick explained that this referred to the difficulty NRC was having with a timely receipt of FEMA findings. D. W. Moeller questioned whether the item under the TMI-2 cleanup, which referred to NRC working with DOE on the disposition of reactor fuel, referred to the reactor fuel from the TMI-2 plant. F. Remick concurred. Also questioned was the effect on the siting rulemaking that is in process from the preparation of the safety goal and better characterization of the radioactive source term which as indicated must precede new siting regulations. W. Dircks indicated that the Committee could expect a slowdown in the siting rulemaking to be followed by a revised version when the safety goal and source term are better characterized.

D. Okrent noted a problem with internal staff quality control and quality assurance referring to a Maine Yankee plant problem regarding the possibility of flooding in the turbine building due to failure of the circulating water piping. He suggested that a breakdown in quality control within the NRC from the point of view of timeliness caused this issue to "bounce around" from mid-1972 through the issuance by NRC of a Safety Evaluation Report on April 13, 1981 concerning the matter even though a study for the NRC by the Livermore National Laboratory had addressed the need for a fix several years earlier.

IX. NRC Safety Research Program Budget

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

The Committee discussed transfer of $1 million from the Meteorological Research budget to the area of seismic research. D. Okrent presented the case for the importance of seismic research. The importance of seismic research was supported by a reference document (see Appendix XII). On the afternoon of February 5, 1982, Leon Beratan of the Research Staff made a presentation on behalf of the Meteorological Research Program (see Appendix XIII).

D. Okrent and C. P. Siess were unconvinced that leaving the funds in the Meteorological Research Program would result in significant improvements in meteorological predictive capability.

A discussion of aspects of the LOFT Test Program took place. C. N. Kelber of the NRC Staff indicated that the severe fuel damage experiments conducted by the NRC have had peer and international review. W. Kerr felt that core melt studies should have a greater priority than fuel damage experiments. M. W. Carbon presented arguments for adding additional funds for LMFBR research, exclusive of the licensing effort on the CRBR.

X. Executive Sessions (Open to Public)

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

A. Subcommittee Assignments

1. Clinch River Breeder Reactor

William Stratton, ACRS consultant, has proposed in a recent letter, the formation of an ACRS-sponsored Task Force to review the basis for the HCDA for LMFBR's. The Committee discussed the matter and decided it would be appropriate for the CRBR Subcommittee to examine this matter with the assistance of ACRS consultants and other invited experts as appropriate. M. W. Carbon asked for guidance from his fellow members regarding the activities of the Advanced Reactor Subcommittee but time did not permit extensive discussion (see attached Request for Guidance, Appendix XIV).

B. ACRS Reports, Letters, and Memoranda

1. ACRS Report on NRC Policy on the Severe Accident Rulemaking and Related Matters

The Committee prepared a report to the Commissioners regarding the proposed approach to implementing SECY-82-1, Severe Accident Rulemaking and Related Matters, dated January 4, 1982. The ACRS also considered the Commission's comments made at the January 6, 1982 Commission briefing and contained in the memorandum, Samuel J. Chilk to William J. Dircks, dated January 29, 1982. The Committee expressed willingness to participate in the drafting of alternative approaches to resolving issues relevant to severe accidents should the Commission decide to establish an appropriate NRC working group.

2. ACRS Review and Report of the NRC Safety Research Program Budget

The Committee completed its report to the U.S. Congress regarding the proposed NRC Safety Research Program for FY-83.

3. ACRS Comments on Licensees' Safety Review Committees

The Committee prepared a memorandum to the EDO clarifying the advice which the ACRS offered in recent reports to the Commission regarding the make-up of operating license applicants' safety review committees.

C. Generic Safety Items

1. Qualification Program for Safety Related Components

The Committee agreed to a proposed briefing regarding NRC efforts to improve operational QA at nuclear facilities (see Appendix II).

2. Liquid Level Indicators

A session on liquid level indicators was tentatively scheduled for the March full Committee meeting consistent with the NRC Staff plans to resolve this issue. (Note: It now appears that this session will be deferred to the April meeting except for an interim briefing by John MacEvoy regarding performance of differential-pressure liquid level indicators.)

D. Future Schedule

1. Future Agenda

The Committee agreed on a tentative agenda for the 263rd ACRS Meeting, March 4-6, 1982 (see Appendix II).

2. Future Subcommittee Activities

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

E. H. W. Lewis Session on Emergency Planning

H. W. Lewis has been invited to participate in a tutorial session sponsored by Southern California Edison to brief local officials on emergency planning for nuclear power plants. The Committee did not endorse a proposed policy that Members should be encouraged to speak at such public service meetings but did agree to reimburse H. W. Lewis for his incurred expenses. The ACRS further decided to deal with sponsorship of attendance at public service meetings on a case-by-case basis taking note of the above tutorial session as precedent.

F. ACRS Testimony Regarding the NRC RSR Budget Before the House Committee On Interior and Insular Affairs

The Committee discussed testimony to be presented by P. G. Shewmon and C. P. Siess at the Udall Committee hearing (see Appendix XV). Background information on the LOFT research program was distributed. (see Appendix XVI). The following areas were identified as potential subjects of House Committee questioning:

- Transformation of the ACRS annual report to Congress on the NRC's Safety Research Program to a biannual report
- Assessment of significant changes in the RSR program initiated by NRC
- Review and Summary of RSR program for FY 1983
- Discussion and critique of NRC plans for LOFT
- Effect of the most significant RSR findings upon Commission standards, regulations, and regulatory guides

- Problems with steam generators at nuclear plants
- ACRS views on the Nuclear Data Link (see Appendix XVII)
- Adequacy of support of ACRS activities by NRC

Members compiled a list of particular portions of the RSR program that have impacted NRC standards, regulations, and regulatory guides. This tabulation included subjects such as the HSST program, Appendix G and Appendix H, pressurized thermal shock, ECCS Appendix K, seismic research, the CRAC computer code, and the TMI Lessons Learned. W. Kerr noted that the work on hydrogen control is, perhaps, the most demonstrable RSR program to impact NRC rules and regulations.

The 262nd Meeting of the ACRS was adjourned on Saturday, February 6, 1982 at 12:10 p.m.

APPENDIXES
TO
MINUTES OF THE 262ND ACRS MEETING
FEBRUARY 4-6, 1982

ACRS-1951

ATTENDEES
262ND ACRS MEETING
FEBRUARY 4-6, 1982

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

Paul G. Shewmon, Chairman
Jeremiah J. Ray, Vice-Chairman
Robert C. Axtmann
Myer Bender
Max W. Carbon
*Jesse Ebersole
Harold Etherington
William Kerr
Harold W. Lewis
Carson Mark
William M. Mathis
Dade W. Moeller
David Okrent
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
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

NRC ATTENDEES

262ND ACRS MEETING

Thursday, February 4, 1982

Division of Licensing

Gus Lainas, DL
Jim Lyons, DL
E. Goodwin, NRR

Analysis and Evaluation of
Operational Data

R. L. Denning

Nuclear Reactor Regulation

J. Read
S. H. Hanauer
M. C. Ernst
A. Marchese
R. Mattson
J. Meyer
J. Conran
E. Goodwin

Region I

R. C. Haynes

NRC ATTENDEES

262ND ACRS MEETING

Friday, February 5, 1982

Nuclear Reactor Regulation

E. Goodwin
F. J. Miraglia
R. Mattson
R. Jackson
S. Brocoum
I. Spickler

Inspection and Enforcement

R. Paulus
B. Zalcman

Nuclear Material Safety and Safeguards

B. Erickson
H. Smith
L. Lessler

Nuclear Regulatory Research

W. R. Ott
L. Beratan

Office of Policy Evaluation

F. Remick
D. Rathbun

PUBLIC ATTENDEES

262ND ACRS MEETING

Thursday, February 4, 1982

R. Leyse, Electric Power Research
R. Smith, Self
R. B. Borsum, Babcock & Wilcox
E. T. Murphy, Westinghouse
Joyce Nelson, Quadrex
M. White, Doub & Muntzing
J. Berga, Electric Power Research Inst.
J. Siegel, Atomic Industrial Forum
S. Filipour, ARC
F. Stetson NUS
R. J. Ross, Dames & Moore
M. A. Bauser, LNR&K
H. C. Schmidt. TXU
G. L. Wilson, TXU
J. L. Nantz, DSA

Friday, February 5, 1982

R. Leyse, Electric Power Research Inst.
L. N. Rib, LNR Associates
F. T. Murphy, Westinghouse
R. S. Boyd, KMC, Inc.
S. R. Phelps, EEI
J. Nelson, Quadrex
R. J. Ross, Dames & Moore
J. Leyse, Electric Power Research Inst.
R. Heer, ARC

APPENDIX II
FUTURE AGENDA

MARCH

Clinton Station Units 1 and 2--OL

Byron Station Units 1 and 2--OL

Waterford Station Unit 3--Outstanding OL items

~~hydrogen~~

. operating organization

NRC Long Range Research Program Plan--ACRS comments to NRC — Deferred to April

Briefing by the NRC Staff regarding Operational Quality Assurance

Update of the NRC report to the U.S. Congress regarding unresolved safety issues — Deferred to April or May

Report of the ACRS testimony regarding the NRC RSR Budget before the House Committee on Interior and Insular Affairs (Cong. M. K. Udall, Chairman)

Pilgrim Nuclear Plant--Management deficiencies ($550,000 NRC fine) and deficiencies regarding Yarway level indicators, torus damage, (gouges), and SRV performance (PGS/DCF) — Deferred

Zion/Dresden Nuclear Plants--Investigation of charges regarding deficiencies in performance of the guard force at these plants (WK/PSG/DCF) — Deferred

Subcommittee Reports

- Subcommittee on Fluid Dynamics regarding Mark III Containment development (MSP/PAB)
- Subcommittee on Indian Point Units 2 and Unit 3 regarding the subject of systems interaction (WK/DCF)
- Subcommittee on Zimmer Nuclear Station regarding Quality Assurance deficiencies to prepare a memorandum to the EDO (MB/PAB) — 45 min.
- Subcommittee on Regulatory Activities regarding Regulatory Guide 1.28, Rev. 3, Quality Assurance Program Requirements (Design and Construction and other Regulatory Guide and regulation changes (CPS/SD)

Future Agenda (Cont.)

- Subcommittee on Reactor Operations regarding proposed LER rule changes to which Commissioner Ahearne has objected (WMM/RKM)
- Subcommittee on Extreme External Phenomena regarding seismicity in the Eastern U.S. (DO/RS)

Future ACRS Activities

The RSK has accepted the ACRS' invitation to meet with the Committee in the USA and the Committee has endorsed this meeting on October 5-6, 1982, the Tuesday and Wednesday preceding the October full Committee meeting.

Subcommittee Reports

- Diablo Canyon Units 1 & 2 - seismic design deficiencies (CPS/JCM)
- Regulatory Activities - Regulatory Guide 1.23, Meteorology Measurement Programs for Nuclear Power Plants (CPS/SD)
- Generic Items - evaluation of systems interactions per ACRS memorandum to the EDO dated January 8, 1982 (MB/RS)
- Extreme Environmental Effects (seismic) - reply to Commissioner V. Gilinsky regarding proposed changes in seismic methodology per recommendations of Paul C. Jennings letter dated October 5, 1981 (DO/RS)
- AC/DC Power Systems Reliability - results of cable surveillance program at St. Lucie 1 Nuclear Plant (JJR/JMG)

REVISION

2/9/82

APPENDIX III

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

FEBRUARY

9 (pm)	Simulator Tour (Silver Spring, MD) (Major) - Kerr, Ward, Mathis. Purpose: Visit Singer-Link Corporation.
10	Tour of Westinghouse Simulator and Safety Parameter Display Demonstration (Pittsburgh, PA) (Major) - Ward, Kerr, Mathis. Purpose: The tour will include an explanation of the development of W symptom-based procedures, and the W version of the of the SPDS. The tour will also include a demonstration of the SPDS and symptom-based procedures on the W Control Room Simulator.
10	Qualification Program for Safety Related Equipment (Boehnert) - Ray, Ebersole. Purpose: To review the NRC Equipment Qualification Program Plan as outlined in SECY-81-504.
11	Reactor Radiological Effects (Alderman/McKinley) - Moeller, Shewmon, Axtmann, Ray. Purpose: To discuss occupational radiation exposure in BWRs.
12	Joint Metal Components and Waste Management (Igne/Alderman) - Shewmon, Ray, Axtmann, Moeller. Purpose: To review contractor technical capability and objectives of request for proposal on long-term performance of materials used for high-level waste packaging.
18	Zimmer Plant (Cincinnati, OH) (Boehnert) - Bender, Ebersole, Carbon. Purpose: To review QA problems associated with plant construction which resulted in $200,000 fine by NRC/I&E and to discuss plant operations.
25 & 26	Byron Station 1 & 2 (Rockford, IL) (Igne) - Shewmon, Bender (25th only), Mark, Axtmann. Purpose: Site visit (Byron, IL) and to review application for an operating license.
25 & 26	Clinton (Decatur, IL) (Savio) - Kerr, Ward (25th only), Moeller, Siess. Purpose: Site visit and to review application for an operating license.
26	Safety Philosophy, Technology, and Criteria (Griesmeyer/Quittschreiber) - Okrent, Bender, Ebersole, Mathis, Ward. Purpose: To review the proposed Systems Interaction Study for the Indian Point Nuclear Power Plant, and the NRC Systems Interaction Program.

*Conflict to be resolved

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

MARCH

3 (8:45 - 1:30 p)	Regulatory Activities (Duraiswamy) - Siess, Kerr, Ray, Bender, Carbon, Ward. Puprose: To discuss Regulatory Guide 1.28, Rev. 2, Quality Assurance Program Requirements (Design and Construction)" and proposed rule, "Accreditation of Qualification Testing Organizations".
3 (1:00 p)	Waterford (Beal/Quittschreiber) - Ward, Bender, Carbon, Ray*, Siess. Purpose: To review Waterford organization, staffing, and training programs.
3 (2:00 - 6:00)	Reactor Operations (Major) - Mathis, Ebersole, Kerr, Moeller, Okrent, Ray*. Purpose: To continue discussions with the Staff of AEOD on the proposed LER Rule, SECY-82-3.
4-6	263rd ACRS Meeting
16	Decay Heat Removal Systems (Savio) - Ward, Bender, Carbon, Ebersole, Etherington, Ray. Purpose: To review the status of Task Action Plan A-45 and PWR Decay Heat Removal Systems with the emphasis on the CESSAR System 80 standard design.
17	Human Factors (Fischer) - Ward, Bender, Lewis, Mathis, Moeller, Ray. Purpose: To review the various Safety Parameter Display System (SPDS) designs and the status of plant diagnostic systems. NUREG-0799, "Draft Criteria for Preparation of Emergency Operating Procedures," will be discussed also. Additionally, the Subcommittee will discuss with representatives from industry ACRS concerns related to the management, organization, staffing, and technical resources of utilities that operate nuclear power plants. Other areas to be discussed include: the training of Shift Technical Advisors (STAs) in the areas of plant systems and transient/accident analysis, and Senior Reactor Operator (SRO) training programs and qualification.
18 & 19	Joint Reactor Operations/R.E. Ginna (Ontario, NY) (Major/Fischer) - Mathis, Ebersole, Etherington, Shewmon, Ray, Ward, Siess*. Purpose: To discuss the 1/25/82 steam generator tube failure; Site Emergency incident and SEP review of Ginna.
19	Reliability and Probabilistic Assessment (Griesmeyer/Quittschreiber) - Okrent, Bender, Kerr**, Siess*. Purpose: To review draft Commission Policy Statement on Safety Goals.

* part-time
** Conflict to be resolved

REVISION

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SCHEDULE OF ACRS SUBCOMMITTEE MEETING

MARCH (CONT'D)

22	Structural Engineering (Albuquerque, NM) (Igne) - Siess, Bender, Ebersole, Shewmon. Purpose: To review Sandia's containment integrity program, including a visit to the Sandia structural laboratory.
23	Safeguards & Security (Albuquerque, NM) (Alderman/McKinley) - Mark**, Ray, Shewmon, Ward, Siess, Carbon (tent.), Mathis, Plesset**, Lewis (tent.). Purpose: To discuss design features in proposed standard design plants that would make sabotage by insiders more difficult.
23 & 24	WPPSS 2 (Hanford, WA) (Griesmeyer/Quittschreiber) - Bender, Ebersole, Plesset**. Purpose: To review application for an operating license.
25, 26 & 27	Advanced Reactors (Argonne, IL) (Igne/Boehnert) - Carbon, Mark**. Purpose: To continue discussion of report on LMFBR safety philosophy.
30	AC/DC Power System Reliability (Savio) - Ray, Ebersole, Kerr, Mathis, Okrent. Purpose to review the status of Task Action Plan A-44 and implementation of the recommendations of NUREG-0666, "A Probabilistic Safety Analysis of DC Power Supply Requirements for Nuclear Power Plants."
31	TMI-2 Action Plans (Major) - Mathis, Etherington*, Lewis*, Okrent*. Purpose: To review the proposed rule on 10 CFR 50, "Licensing Requirements for Pending Operating License Applications" (rule contains Basic Requirements of NUREG-0737, "Clarification of TMI Action Plan Requirements").
31	Joint Electrical Systems and ECCS (Savio/Boehnert) - Kerr, Ebersole, Plesset*, Ray, Lewis*, Bender, Etherington*. Purpose: To continue review of the NRC-and Industry-sponsored research on core water level indicator instruments and the NRC and Industry implementation of core water level indicator installation requirements.
31	Nuclear Safety Research Program (Duraiswamy) - Siess, Okrent*, Kerr*, Plesset*, Shewmon, Mark, Moeller, Ward. Purpose: To continue discussion of the NRC Long-Range Research Program Plan.
30 (p.m.) 31 (a.m.)	CRBR (Boehnert/Igne) - Carbon, Bender*, Mark. Purpose: To review the CRBR General Design Criteria.

* part-time
** Conflict to be resolved

REVISION

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

APRIL

21 & 22	Wolf Creek (near Topeka, KS) (Major/Bucci) - Ray, Axtmann, Lewis, Mark, Plesset. Purpose: Site visit and review application for an operating license.

MAY

4-5	Watts Bar (Griesmeyer/Quittschreiber) - Bender, Ebersole, Ward. Purpose: To review application for an operating license.

DATES TO BE DETERMINED

Date to Be Determined (April)	Reactor Radiological Effects (Alderman/McKinley) - Moeller, Ray, Axtmann. Purpose: To review NUREG-0833, "Environmental Impact Statement on the Siting of Nuclear Power Plants."
Date to Be Determined (May or June)	Metal Components (Igne) - Shewmon, Ward, Axtamnn, Bender, Etherington, Mathis, Plesset. Purpose: To continue review of pressurized thermal shock.
Date to Be Determined (June or July)	Joint CRBR and Site Suitability (Igne/Alderman) - Carbon Moeller, Bender, Mark, Okrent, Plesset, Shewmon, Siess, Axtmann, Ebersole, Ray. Purpose: To begin site suitability review for CRBR.
Date to Be Determined	Transportation of Radioactive Materials (Duraiswamy) - Siess, Mark, Bender. Purpose: To continue the review of the adequacy of the NRC procedures for certifying packages for transporting radioactive materials.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Feb. 9 p.m.	Simulator Tour	Kerr, Ward, Mathis Consultants: I. Catton, J. Buck, A. Debons, M. Deyserling, R. Pearson, Staff & Fellows; R. Major, P. Boehnert, D. Fischer, J. MacEvoy, K. Kirby, W. Bock, C. Ryder, T. McCreless, Fraley,

LOCATION: Singer-Link Corporation, Silver Spring, Md.

BACKGROUND:

Who proposed action: W. Kerr

Purpose: To visit Singer-Link Corporation

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

ADDITIONAL DETAILS:

This will be an afternoon trip to Singer-Link Corporation located in Silver Spring, Maryland to observe several Nuclear Power Plant Simulators under construction, possibly witness a demonstration of one, and discuss the engineering behind the simulator with employees of Singer-Link. The tour will start and end at the ACRS Offices at 1717 H Street.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
February 10, 1982	Tour of Westinghouse Simulator and Safety Parameter Display Demonstration	D. Ward, J. Buck, R. Pearson, A. Debons, R. Major, D. Fischer, W. Baldewicz, J. MacEvoy, W. Mathis, W. Kerr

LOCATION: Westinghouse Electric Corporation, Pittsburgh, PA

BACKGROUND:

Who proposed action: James W. Miller of Westinghouse has invited all interested Members to Pittsburgh for a tour, demonstration and explanation of Westinghouse emergency facilities.

PURPOSE: The tour will include an explanation of the development of Westinghouse symptom based procedures, and the Westinghouse version of the SPDS. The tour will also include a demonstration of the SPDS and symptom based procedures on the Westinghouse Control Room Simulator.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY: N/A

ADDITIONAL DETAILS: Most of those going on this trip will have made a visit to Singer-Link Corporation the previous day and will already be in Washington. On the morning of the Feb. 10th, those participating should assemble at Washington's National Airport to catch an 8:00 a.m. flight to Pittsburgh (AL-67). The group should be in Pittsburgh by 9:00 a.m. and drive to the Westinghouse presentations and demonstrations will take about 8 hours. The group should be back at the Pittsburgh Airport by 7:00 p.m. for departure flights.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
FEB. 10	Qualification Program for Safety Related Equipment	(BOEHNERT) Ray, Ebersole, Cons: Lipinski, Catton

LOCATION: Washington, D.C.
(Federal Home Loan Bank Board Conference, Room: 1700 G St, N.W.; Fifth Floor)*

BACKGROUND:

Who proposed action: J. Ray

Purpose: To review the NRC Equipment Qualification Program Plan as outlined in SECY-81-504

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SECY-81-504 plus additional material to be provided later.

*To test integrated communications/recording system.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
FEB. 11	Reactor Radiological Effects	(ALDERMAN) Moeller, Shewmon Axtmann, Ray Cons: R. Dillon T. Kassner

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: D. Moeller/P. Shewmon

Purpose: To discuss occupational radiation exposure in BWRs.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. P. Shewmon memo to D. Moeller September 25, 1981

2. SEC-81-517, unusually high occupational radiation doeses reported for power reactors operating in 1980, August 28, 1981.

3. Memo, P. Boehnert to M. Bender, October 14, 1981 , NRC I&E Action - $50,000 fine for violation of radiation exposure control requirement December 14, 1981.'

4. Evaluation of crud inventories of RLR pipings on 1100 MWe BWR Decommissioning November 1981, the Institute of Applied Energy (Japan).

5. Cost evaluation of 1100 MWe BWR decommissioning, November 1981, The Institute of Applied Energy (Japan).

6. Evaluation of Induced Activity on Decommissioning of 1100 MWe BWR November 1981, The Institute of Applied Energy (Japan.)

7. Corrosion product control, October 3, 1980 letter and attachments from R. E. Engel (G.F.) to M. Torar NRC

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SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
FEB. 12	Metal Components and Waste Management	(IGNE/ALDER...) Shewmon Axtmann, Moeller, Ray Cons: Steindler, Rodabaugh, Readey, Dillon, Kassner

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: Commission

Purpose: To review contractor technical capability and objectives of request for proposal on long-term performance of materials used for high-level waste packaging.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. Request for Proposed RS-RES-81-173, "Long Term Performance of Materials Used for High-Level Waste Packaging."
2. Contractor package consisting of documentation on technical capability and response to proposal.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
FEB. 18	Zimmer Plant	(BOEHNERT) Bender, Carbon, Ebersole

LOCATION: Cincinnati, OH

BACKGROUND:

Who proposed action: M. Bender/ACRS

Purpose: To review QA problems associated with plant construction which resulted in a $200,000 fine by NRC/I&E and to discuss plant operations.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. I&E Investigation Report (to be distributed to Committee).
2. I&E Notification of Violations and Appraisal of Fines (distributed to Committee)
3. Other pertinent documentation as it becomes available.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Feb. 25 site visit Feb. 26 scbt. mtg.	Byron Station 1 & 2	(IGNE) Shewmon, Bender (25th) Mark, and Axtmann Cons: Kassner

LOCATION: Site Visit at Byron (25th). Subcommittee meeting at the Ramada Inn in Rockford, Ill. p.m.

BACKGROUND:

Who proposed action: NRC Staff & P. Shewmon

Purpose: OL review.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

Safety Evaluation Report due 2/05/82.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Feb 25-26	Clinton	(SAVIO) Kerr, Ward (25th only) Moeller, Siess.

LOCATION: Decatur, IL
Site Visit at the Clinton site with a Subcommittee meeting near the site.

BACKGROUND:

Who proposed action:

Purpose: To review application for OL.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. Safety Evaluation Report expected to be available by February 5, 1982.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
FEB. 26	Safety Philosophy, Technology, and Criteria	(GRIESMEYER/QUITTSCHREIBER) Okrent, Bender, Ebersole, Mathis, Ward.

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: PASNY and NRR have requested that the ACRS review the proposed Systems Interaction Study for the Indian Point Nuclear Plant.

Purpose: To review the proposed Indian Point Nuclear Plant Systems Interaction Study, and the NRC Systems Interaction Program.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

To be forwarded later.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
March 3 (8:45 a - 1:30 p)	Regulatory Activities	(DURAISWAMY) Siess, Kerr, Ray, Bender, Carbon, Ward.

LOCATION: Washington, DC

BACKGROUND:

Who proposed action:

Purpose: To discuss:

1. Regulatory Guide 1.28, Rev. 3, "Quality Assurance Program Requirements (Design and Construction)" (post-comment).

2. Proposed Rule "Accreditation of Qualification Testing Organizations" (pre-comment)

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SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
March 3 (1:00 pm)	Waterford	(Beal/Quittschreiber) - Ward, Bender, Carbon, Ray, Siess Cons: Pearson, Binford

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: D. Ward

Purpose: To review Waterford organization, staffing, and training programs.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. SER Supplement dated January 1982.

A-21.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
March 3, 1982 (2-6 p.m.)	Reactor Operations	(MAJOR) Mathis, Ebersole, Kerr, Moeller, Okrent, Ray

LOCATION: Washington, D.C.

BACKGROUND:

Who proposed action: Commissioner Ahearne

Purpose: To continue discussions with the Staff of AEOD on the proposed LER Rule in SECY-82-3. See the Attached Comments from Mr. Ahearne on SECY-82-3.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. SECY-82-3

2. Commissioner Ahearne's Comments

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SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
March 16	Decay Heat Removal Systems	(Savio), Ward, Bender, Carbon, Ebersole, Etherington, Ray

LOCATION: Washington, D.C.

BACKGROUND:

Who proposed action: ACRS

Purpose: To review the status of Task Action Plan A-45 and PWR Decay Heat Removal Systems with the emphasis on the CE system 80 standard design.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

A-23

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
MARCH 17	Human Factors	(FISCHER) Ward, Bender, Lewis, Mathis, Moeller, Ray. Cons: Arnold, Buck, Debons, Keyserling, Pearson, Salvendy I. Catton

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: D. Ward

Purpose: To review the various Safety Parameter Display System (SPDS) designs and the status of plant diagnostic systems. NUREG-0799, "Draft Criteria for Preparation of Emergency Operating Procedures," will be discussed also. Additionally, the Subcommittee will discuss with representatives from industry ACRS concerns related to the management, organization, staffing, and technical resources of utilities that operate nuclear power plants. Other areas to be discussed include: the training of Shift Technical Advisors (STAs) in the areas of plant systems and transient/accident analysis, and Senior Reactor Operator (SRO) training programs and qualification.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

NUREG-0799, Draft Criteria for Preparation of Emergency Operating Procedures, dated June 1981 (for comment version).

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
March 18-19, 1982	Reactor Operations/R.E. Ginna	(Major) Mathis, Etherington, Ebersole, Shewmon, Ray, Ward, Siess.
		Consultants: I. Catton, Z. Zudan, D. Fitzsimmons, W. Lipinski

LOCATION: Ontario, New York (15 miles northeast of Rochester, New York)

BACKGROUND:

Who proposed action: W. Mathis

Purpose: The purpose of this meeting will be two fold. First the Reactor Operations Subcommittee wishes to discuss the January 25, 1982 steam generator tube failure; Site Emergency incident. Among the goals of this meeting will

be to evaluate how well the emergency preparations at Ginna served the situation and examine the operators response to the incident. Secondly, Ginna is rapidly becoming tied with Palisades as the lead SEP (Systematic Evaluation Program) plant. Once at the site, those improvements which can be observed resulting from the SEP program could be viewed. An SEP "tour" of Ginna coupled with the steam generator tube rupture review could eliminate the need for another trip to Ginna as part of the SEP review, and allow Ginna's SEP meeting to be conducted in Washington.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. Only a Preliminary Evaluation of Operator Actions for Ginna SG Tube Rupture Event is currently (1/29) available. Prior to the meeting the results of more detailed investigations should be available.

2. The SEP Safety Evaluation (SE) is currently scheduled for release in April, however, it may be possible to proceed with a plant tour to observe SEP upgrades without the SE.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
March 23-24	WPPSS 2	(Griesmeyer/Quittschreiber) Bender, Ebersole, Plesset

LOCATION: Hanford, WA

BACKGROUND:

Who proposed action: NRR

Purpose: To review application for operating license.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
March 25, 26 & 27	ADVANCED REACTORS	(IGNE/BOEHNERT) Carbon, Mark.

LOCATION: Argonne, IL

BACKGROUND:

Who proposed action: M. Carbon

Purpose: To continue discussion and preparation of safety issue and philosophy of LMFBR report to the ACRS.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
March 30	AC/DC Power Systems Reliability	(Savio), Ray, Ebersole, Kerr, Mathis, Okrent

LOCATION: Washington, D.C.

BACKGROUND:

Who proposed action: Subcommittee Chairman

Purpose: To review the Status of the NRC work on Task Action Plan A-44 and the NRR Impementation of the reccommendation of NUREG-0666.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

A-28

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
3/30 pm 3/31 am	CRBR	(BOEHNERT/IGNE) Carbon, Bender, Mark.

LOCATION: Washington, D. C.

BACKGROUND:

Who proposed action: Carbon

Purpose: To review CRBR General Design Criteria

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

The GDC for the CRBR will be sent to us by the middle of February.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
March 31, 1982	TMI-2 Action Plans	(Major), Mathis, Etherington, Lewis, Okrent

LOCATION: Washington, D.C.

BACKGROUND:

Who proposed action: W. Mathis

Purpose: to review the proposed rule on 10 CFR 50 - Licensing Requirements for Pending Operating License Applications (Rule contains the Basic Requirements of NUREG-0737, "Clarification of TMI Action Plan Requirements"). This will be the second meeting with the Staff on this rule. Public comments should have been evaluated and incorporated into the final form of the rule prior to Subcommittee meeting.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

The final form of the rule is expected to be available by mid to late February.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
March 31	Combined ECCS/Electrical Systems Subcommittee	(SAVIO/BOEHNERT) Kerr, Ebersole, Plesset, Ray, Lewis, Bender, Etherington.

LOCATION: Washington, DC

BACKGROUND:

Purpose: To continue the review of the NRC and Industry sponsored research on core water level indicator instruments and the NRC and Industry implementation of core water level indicator installation requirements.

PERTINENT PUBLICATIONS:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
MARCH 31	Nuclear Safety Research Program	(DURAISWAMY) Siess, Okrent, Kerr, Plesset, Shewmon, Mark, Moeller, Ward

LOCATION: Washington, DC

BACKGROUND:

Who proposed action:

Purpose: To continue discussion of the NRC Long-Range Research Program Plan.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

Final Draft of the Long-Range Research Plan is expected to be made available to the ACRS in the middle of March.

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SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
April 21-22, 1982	Wolf Creek Station	(RKM/DRB) Ray, Axtmann, Lewis, Mark, Plesset CONSULTANTS: J.C. Maxwell

LOCATION: Site (listed below)

BACKGROUND:

Who proposed action: Staff and ACRS

Purpose: To visit the site and to review the application for an operating license. (WC is - 50 mi. south of Topeka, Kansas)

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

The plant safety evaluation report is due on April 7, 1982.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
May 4 & 5	Watts Bar	(Griesmeyer/Quittschreiber) - Bender, Ebersole, Ward

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: NRR

Purpose: To review application for operating license.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SER due 4/5/82.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
April 1982	Reactor Radiological Effects	(ALDERMAN/McKINLEY) Moeller, Ray, Axtmann

LOCATION: Washington, D.C.

BACKGROUND:

Who proposed action: D. Moeller

Purpose: Review NUREG-0833 "Environmental Impact statement on the siting of nuclear power plants" and obtain an update on the current NRC Staff thoughts on siting.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

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SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
To Be Determined (May or June)	Metal Components	(IGNE) Shewmon, Ward, Axtmann, Bender, Etherington, Mathis. Plesset

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: P. G. Shewmon

Purpose: To continue the review regarding pressurized thermal shock.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

The NRC Staff SER and guidance for continued operation documents are scheduled to be available in April or May.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
June or July	Joint CRBR and Site Suitability	(Igne/Alderman), Carbon, Moeller, Bender, Mark, Okrent, Plesset, Shewmon, Siess, Axtmann, Ebersole, and Ray. Consultants (to be determined).

LOCATION: Washington, D.C.

BACKGROUND:

Who proposed action: NRC Staff

Purpose: To begin site suitability review for CRBR.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

Site Suitability Report by the Office of Nuclear Reactor Regulation, USNRC in the matter of the Clinch River Breeder Reactor Plant, dated March 4, 1977 (to be revised in June or July).

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
To Be Determined	Transportation of Radioactive Materials	(DURAISWAMY) Siess, Mark, Bender Cons: Zudans, Langhaar, Shappert

LOCATION:

BACKGROUND:

Who proposed action:

Purpose: To continue the review of the adequacy of the NRC procedures for certifying packages for transporting radioactive materials.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

Ginna Event

of

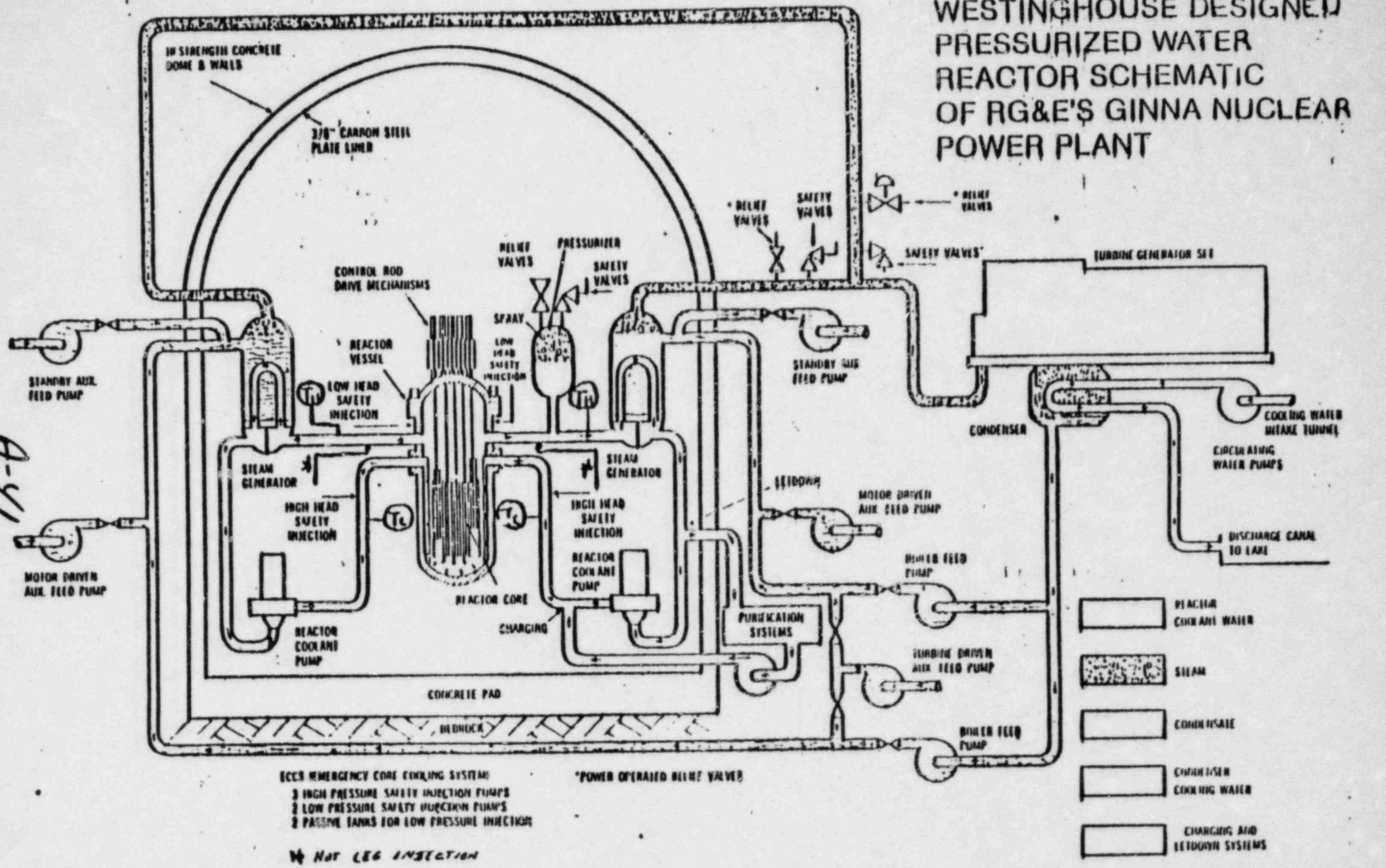
January 25, 1982

- Reactor Operations Chronology Highlights
- Radioactive Material Releases
- Institutional Responses

REACTOR OPERATIONS CHRONOLOGY HIGHLIGHTS

- RAPID DEPRESSURIZATION OF PRIMARY SYSTEM
- AUTOMATIC REACTOR SHUTDOWN AND ACTUATION OF EMERGENCY CORE COOLING SYSTEM
- MANUAL SHUTDOWN OF REACTOR COOLANT PUMPS
- ISOLATION OF SECONDARY SIDE OF FAULTED STEAM GENERATOR
- POWER OPERATED RELIEF VALVE DEPRESSURIZATION OF PRIMARY SYSTEM
- PRESSURIZER RESPONSE/SHIFT OF STEAM VOID TO REACTOR VESSEL HEAD AREA
- PRIMARY COOLANT SPILL IN CONTAINMENT
- FAULTED STEAM GENERATOR SAFETY VALVE LIFTS

WESTINGHOUSE DESIGNED PRESSURIZED WATER REACTOR SCHEMATIC OF RG&E'S GINNA NUCLEAR POWER PLANT



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EVENT CHRONOLOGY
R. E. GINNA
JANUARY 25-26, 1982

Preliminary - Subject to Revision Based on Further Data Review
January 26, 1982 2:30 PM

Time	Event
9:25 am 1/25	Charging Pump speed alarm; "B" Steam Generator (S/G) level alarm; steam flow-feed mismatch "B" S/G; air ejector radiation monitor alarm; pressurizer low pressure - 2170 psig.
9:28 am	Reactor trip on low pressure; automatic safety injection with containment isolation.
9:29 am	Pressurizer level offscale low; RCS pressure approx. 1200 psig.
9:33 am	NRC Operations Center informed via ENS. Ginna reported a reactor trip from 100% power as a result of a steam generator tube rupture. Affected S/G unknown. No release reported. Unusual event declared.
	Both Reactor Coolant Pumps manually tripped.
9:38 am	NRC Region I in phone communications with the site.
9:40 am	"B" Main Steam Isolation Valve manually closed following indication of RCS leakage into "B" S/G (increasing level and pressure); Alert declared.
9:45 am	NRC Senior Resident Inspector in the Ginna Control Room.
9:46 am	Ginna Plant Superintendent notified State; RCS pressure 1200 psig, Tavg 475.
9:53 am	"A" S/G pressure 540 psig, level 76%; B S/G pressure 826 psig, level 89%. Plant cooling down by dumping steam from "A" S/G to Main Condenser relying on natural circulation in A loop.
9:55 am	NRC Region I Incident Response Center activated.
9:57 am	Safety Injection initiation circuitry reset; instrument air for control of containment isolation valves restored.
9:58 am	TSC manned.

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10:00 am	Air ejector radiation monitor at 15000 cpm and trending down from full scale.
10:04 am	Charging pumps restarted; "B" S/G level at 100% narrow range, 400 inches wide range (almost full); RCS pressure 1300 psig; pressurizer level 18%.
10:07 am	Pressurizer PORV manually cycled to reduce primary pressure to reduce leak rate in accordance with Station Procedures; Pressurizer level 10%.
10:08 am	Pressurizer PORV manually cycled again.
10:09 am	Pressurizer PORV manually opened, unable to shut PORV; Pressurizer pressure dropped from 1300 to 800 psig; Pressurizer level increasing; Pressurizer Relief Tank (PRT) high temperature alarm; Pressurizer PORV block valve shut. Pressurizer level offscale high.
	First indication of a steam bubble in the Reactor Vessel Head.
10:10 am (about)	Safety Injection Pumps increase RCS pressure to 1300 psig.
10:18 am	Incore thermocouples indicate 458 degrees.
10:25 am	"B" S/G atmospheric relief (PORV) manually isolated as a precaution.
10:31 am	Reactor Vessel Head temperature 525° by thermocouple.
10:40 am	"B" S/G code safeties lifting (setpoint 1085 psig); Safety Injection pumps secured to reseat safeties; all charging pumps operating.
10:42 am	NRC Headquarters activated. RCS pressure 800 psig.
10:44 am	Site Emergency declared.
10:50 am	Reactor Coolant Pump seal return relief and letdown relief potentially lifted as a result of earlier containment isolation; and discharged to the PRT.
10:57 am	PRT rupture discs ruptured releasing RCS water to the "A" Containment Sump.
11:15 am (about)	One Safety Injection pump started; "B" S/G safeties lifted; RCS pressure at 1035 psig.
11:29 am	"A" Reactor Coolant Pump restarted.

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12:00 noon	Bubble drawn in Pressurizer; Pressurizer level at 80%.
12:05 pm	Established normal letdown.
12:30 pm	Cooling down at 2 degrees per hour by dumping steam from the "A" S/G through the atmospheric PORV; RCS pressure 923 psig.
2:00 pm	The two Containment Sump "A" monitors indicate 9.5 feet (approx. 11000 gal) and 5.5 feet (approx. 1900 gal), PRT at 92%.
5:00 pm	NRC Region I Incident Response Team onsite.
6:40 pm	Reestablished level in "B" S/G. Plant cooling down via single loop circulation dumping steam from "A" S/G to atmosphere. "B" S/G being cooled by feeding AFW and bleeding via the ruptured tube to the RCS.
7:05 am 1/26	RHR initiated with "A" Reactor Coolant Pump running to promote backflow circulation through the "B" loop; RCS pressure 200 psig; RCS temperature 330° (TC and core thermocouples agree).
9:50 am	Sump "A" monitors now indicate 7.5 feet (6600 gallons) and 5.5 feet (1900 gallons). (No water has been pumped out).

NOTATION BY R. C. HAYNES, REGIONAL ADMINISTRATOR, NRC REGION I: The foregoing chronology for the GINNA steam generator tube leak experienced on January 25, 1982, was prepared by the Rochester Gas & Electric Company and issued to the news media and transmitted throughout the industry via the "NOTEPAD" system. This preliminary chronology of events is subject to change as more precise information is obtained; however it is in substantial agreement with the preliminary chronology of events developed by onsite NRC personnel and issued at 9:00 a.m. on January 26, 1982.

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PRELIMINARY EVENT CHRONOLOGY R. E. GINNA JANUARY 25-26, 1982

Prepared: 9:00 am 1/26/82
By NRC personnel on-site

Time	Event
9:25 am 1/25	Charging Pump speed alarm; "B" Steam Generator (S/G) level alarm; steam flow-feed mismatch "B" S/G; air ejector radiation monitor alarm; Pressurizer low pressure-2170 psig.
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9:45 am	NRC Senior Resident Inspector in the Ginna Control Room.
9:46 am	Ginna Plant Superintendent notified State; RCS pressure 1200 psig, Tavg 475.
9:53 am	"A" S/G pressure 540 psig, level 76%; B S/G pressure 826 psig level 89%. Plant cooling down by dumping steam from "A" S/G to Main Condenser relying on natural circulation in A loop.
9:55 am	NRC Region I Incident Response Center activated.
9:57 am	Safety Injection initiation circuitry reset; instrument air for control of containment isolation valves restored.
9:58 am	TSC manned.
10:00 am	Air ejector radiation monitor at 15000 cpm and trending down from full scale.
10:04 am	Charging pumps restarted; "B" S/G level at 100% narrow range, 400 inches wide range (almost full); RCS pressure 1300 psig; Pressurizer level 18%.
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10:08 am	Pressurizer PORV manually cycled again.

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Time	Event
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10:31 am	Reactor Vessel Head temperature 525° by thermocouple.
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10:44 am	Site Emergency declared.
10:50 am	Reactor Coolant Pump seal return relief lifted as a result of earlier containment isolation; seal return relief discharged to the PRT.
10:57 am	PRT rupture discs ruptured releasing RCS water to the "A" Containment Sump.
11:15 am (about)	One Safety Injection pump started; "B" S/G safeties lifted; RCS pressure at 1035 psig.
11:29 am	"A" Reactor Coolant Pump restarted.
12:00 noon	Bubble drawn in Pressurizer; Pressurizer level at 80%.
12:05 pm	Established normal letdown.
12:30 pm	Cooling down at 2 degrees per hour by dumping steam from the "A" S/G through the atmospheric PORV; RCS pressure 923 psig.
2:00 pm	Containment Sump "A" at 9.3 feet (approx 8000 gal); PRT at 92%.
5:00 pm	NRC Region I Incident Response Team onsite.
6:40 pm	Reestablished level in "B" S/G. Plant cooling down via single loop circulation dumping steam from "A" S/G to atmosphere. "B" S/G being cooled by feeding AFW and bleeding via the ruptured tube(s) to the RCS.
7:05 pm 1/26	RHR initiated with "A" Reactor Coolant Pump running to promote backflow circulation through the "B" loop; RCS pressure 280 psig, RCS temperature 330° (TC and core thermocouples agree).

error - should read am, not pm.

NRC, Region I 1/31/82

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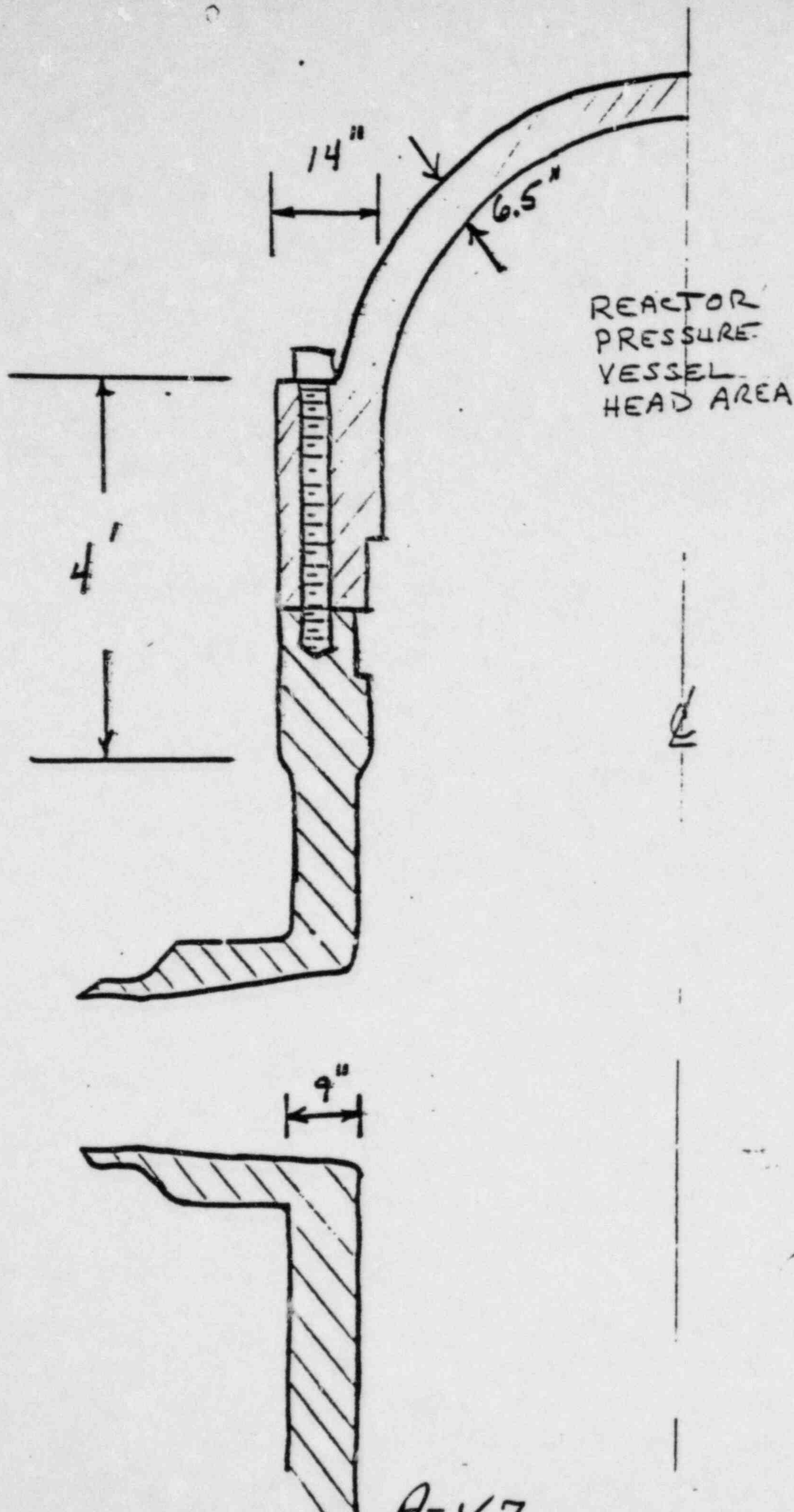

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Radioactive Material Releases

- Releases via Steam Jet Air Ejector Vent
- Releases via Main Steam Line Safety Valves
- On-site Contamination
- Off-site Contamination

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GINNA

ESTIMATE OF RELEASES

(JANUARY 25, 1982)

RELEASE POINT	ISOTOPE	ACTIVITY RELEASED
Steam Jet Air Ejector	Noble Gases	475 - 525 Ci
	I-131	0.001 - 0.002 Ci
"B" Steam Generator	Noble Gases	5 - 6 Ci
	I-131	0.015 - 0.025 Ci
	Mn-54	0.030 - 0.050 Ci
	Co-58	0.030 - 0.050 Ci
	Ba-140	0.17 - 0.30 Ci

Note: Short-lived isotopes not included.

Definition of Curie (Ci): A unit of measure of the amount of radioactivity in a material. One curie is equal to 37 billion disintegrations per second from the nuclei of atoms.

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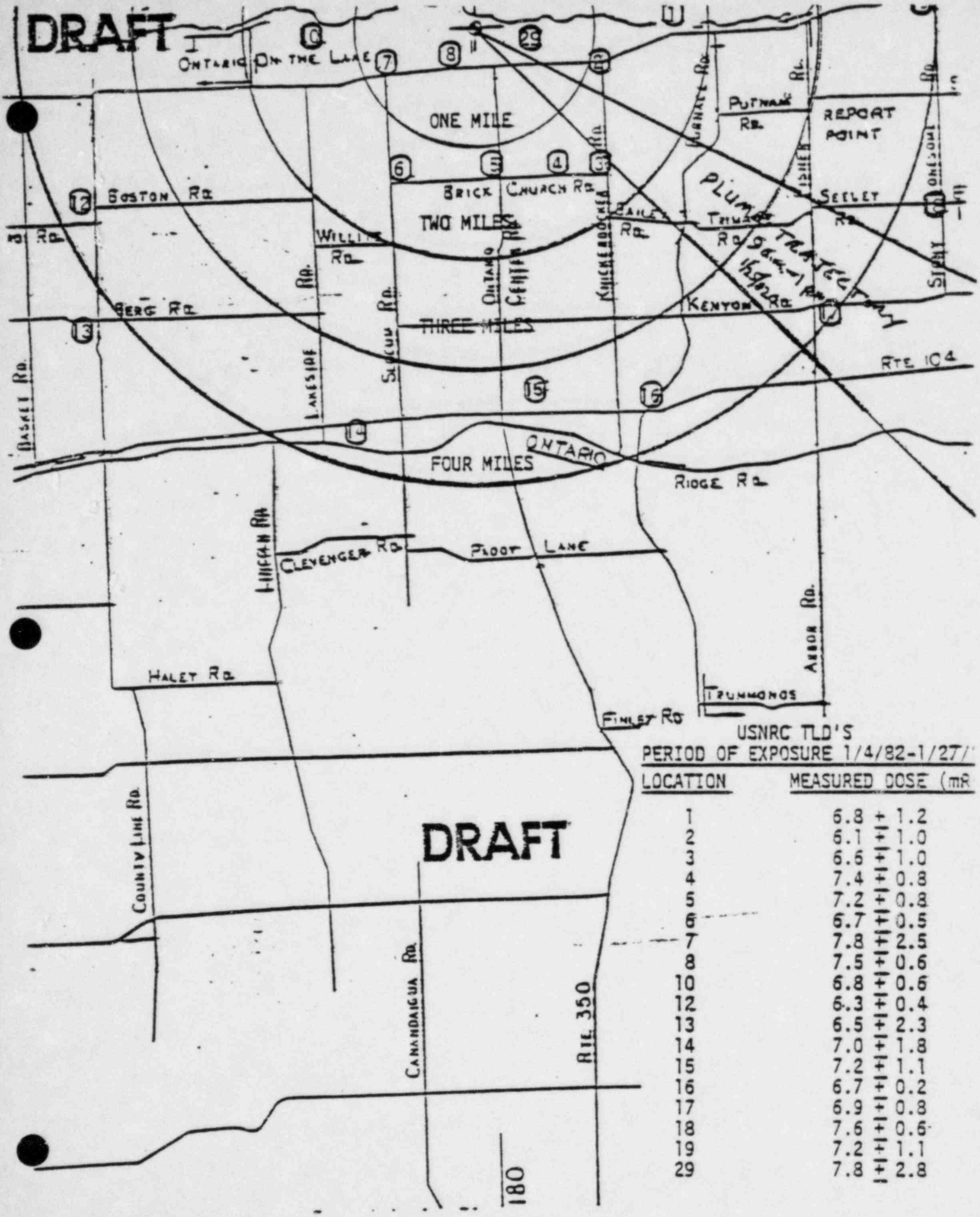
SUMMARY OF TLD ENVIRONMENTAL MONITORING AROUND GINNA - JANUARY 1982

THE ATTACHED SUMMARY SHEET CONTAINS THE RESULTS OF THE TLD MONITORING IN THE VICINITY OF GINNA FOR THE PERIOD WHICH INCLUDED THE JANUARY 25, 1982 INCIDENT. THE DOSE GIVEN IS THE GROSS DOSE MEASURED WITH NO CONTROL BADGE DOSE SUBTRACTED. THE ERROR GIVEN IS A ONE-SIGMA STATISTICAL ERROR ONLY. FOR COMPARISON, AN EXPECTED DOSE WAS CALCULATED USING DATA FOR THE FOURTH QUARTER OF 1981 AND PRO-RATING THIS DOSE FOR THE SHORTER EXPOSURE PERIOD. THE BADGES WERE IN THE FIELD FROM JANUARY 4-JANUARY 27, 1982, BUT THE EXPECTED DOSE WAS CALCULATED ON THE ASSUMPTION THAT THE BADGES WERE BEING IRRADIATED FROM THE TIME THEY WERE SENT FROM REGION 1 ON DECEMBER 22, 1981. NO DOSES WERE MEASURED WHICH WERE STATISTICALLY DIFFERENT FROM THE EXPECTED DOSES.



USNRC TLD'S

PERIOD OF EXPOSURE 1/4/82-1/27/

LOCATION	MEASURED DOSE (mR
1	6.8 ± 1.2
2	6.1 ± 1.0
3	6.6 ± 1.0
4	7.4 ± 0.8
5	7.2 ± 0.8
6	6.7 ± 0.5
7	7.8 ± 2.5
8	7.5 ± 0.6
10	6.8 ± 0.6
12	6.3 ± 0.4
13	6.5 ± 2.3
14	7.0 ± 1.8
15	7.2 ± 1.1
16	6.7 ± 0.2
17	6.9 ± 0.8
18	7.6 ± 0.6
19	7.2 ± 1.1
29	7.8 ± 2.8

PLUME TRAJECTORY BASED ON WIND DIRECTION READINGS
COLLECTED AT ONE ONSITE STATION

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STATE OF NEW YORK TLD'S

PERIOD OF EXPOSURE - 1/4/82-1/26/82

Location	Reading
Training Center	9.4 mR
West of Facility	3.9 mR

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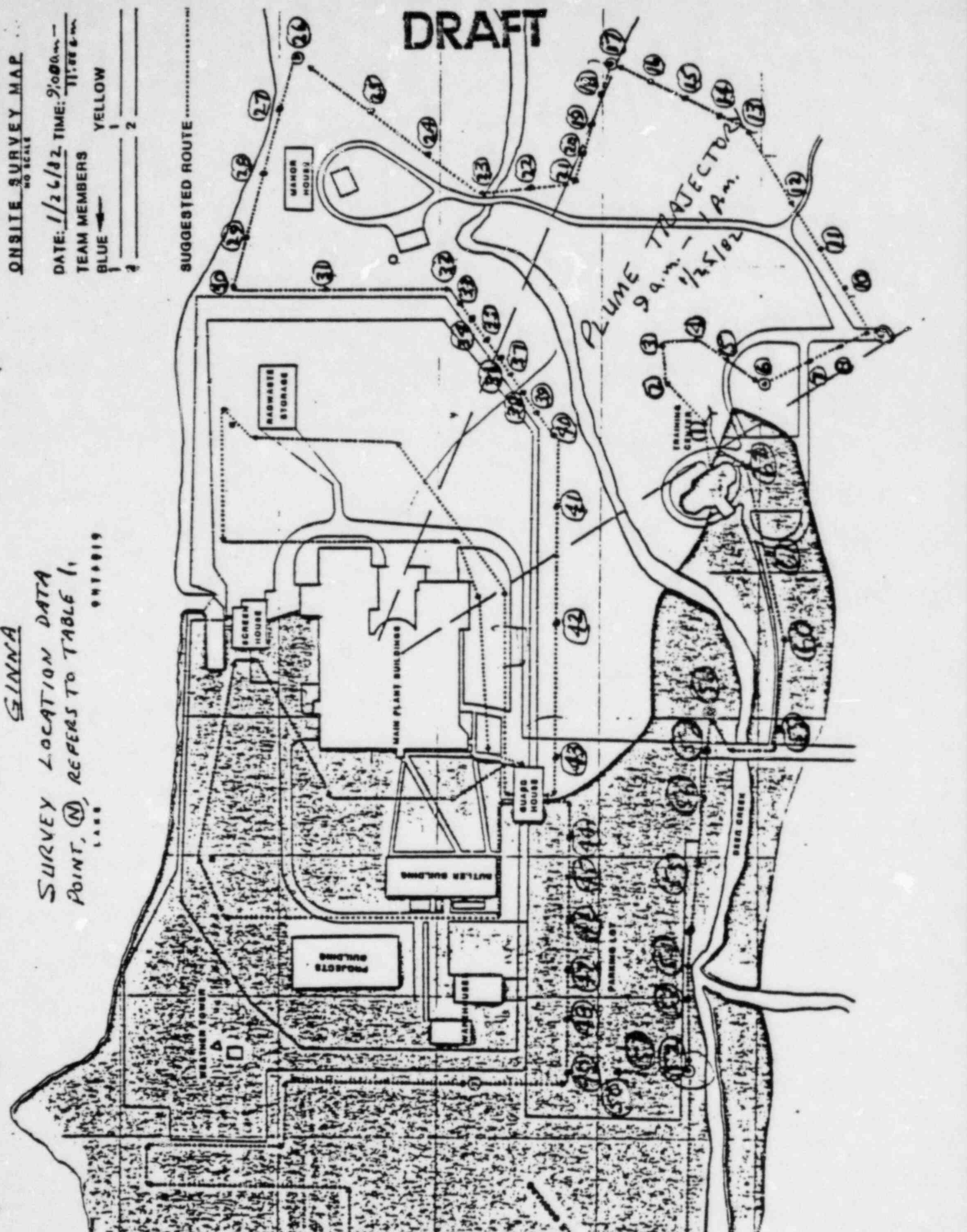

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SURVEY DONE 9:00 A.M TO 11:00 AM ON 1/26/82 WITH RM-14 WITH HP 190 PROBE, NUMBERS INCLUDE BACKGROUND - BACKGROUND ASSUMED TO BE 20 CPM (1 mr/hr ≈ 2200 cpm)

BLUE TEAM SURVEY OF ON-SITE OUTSIDE OF PLANT FENCE - READINGS TAKEN AT 1" and 3' ABOVE GROUND ALL IN CPM

Point	1" Reading	3' Reading	Point	1" Reading	3' Reading
1	700	500	27	40	10
2	1200	800	28	30	30
3	300	200	29	30	20
4	80	80	30	20	30
5	220	220	31	50	30
6	800	600	32	20	20
7	160	240	33	20	20
8	120	80	34	120	110
9	140	100	35	120	120
10	140	100	36	320	300
11	200	160	37	1000	800
12	140	80	38	1800	1300
13	80	60	39	300	300
14	60	60	40	90	180
15	40	20	41	80	80
16	40	20	42	50	30
17	30	10	43	50	40
18	30	40	44	30	40
19	40	30	45	40	40
20	20	20	46	30	10
21	60	40	47	40	20
22	80	60	48	20	20
23	40	30	49	20	10
24	30	50	50	20	40
25	20	10	51	20	20
26	30	10	52	20	10
			53	20	20
			54	40	30
			55	20	10
			56	20	10
			57	30	30
			58	10	10
			59	30	30
			60	20	20
			61	20	30
			62	20	20

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SNOW SAMPLES, GINNA

(micro-Curies/gram)

(10^{-6} Ci/gm)

ISOTOPE	TRAINING CENTER (ONSITE)	PUTNAM & FISHER RD. (OFF-SITE)
I-131	0.00009	0.0000005
I-133	0.00076	0.000004
Cs-137	0.00001	0.0000005
Cs-134	0.00007	0.0000003
Co-58	0.00011	0.000003
Cr-51	0.00006	0.000005

ANALYSIS BY NRC

ISOTOPE	TALLIES FIELD (NEAR SITE BOUNDARY)	RT. 104 & FISHER RD. (OFF-SITE
I-131	0.00001	< 0.0000001
I-133	0.00005	< 0.0000004
Cs-137	0.000001	< 0.0000001
Cs-134	0.0000008	< 0.0000001
Co-58	0.000009	0.0000003
Cr-51	0.000006	< 0.0000007

ANALYSIS BY NRC

ALL VALUES DECAY CORRECTED TO 9:26 a.m., 1/25/82

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Institutional Responses

- Utility
- NRC
- FEMA
- State/Local Agencies
- News Media

APPENDIX V
STATUS REPORT - PROPOSED POLICY STATEMENT
ON SAFETY GOALS FOR NUCLEAR POWER
PLANTS

Status Report

PROPOSED POLICY STATEMENT
ON SAFETY GOALS FOR NUCLEAR POWER PLANTS

Remarks of

Forrest J. Remick, Director
Office of Policy Evaluation

before the

Advisory Committee on Reactor Safeguards

February 4, 1982

I appreciate the opportunity to present to the ACRS a status report on the NRC safety-goal project. As you know, the Office of Policy Evaluation has submitted to the Commission for its consideration a policy paper that includes a draft proposed policy statement and a more detailed supporting report. The Commission is well along in its consideration of the paper, and I believe it will complete action on it very shortly. The expected action is issuance for public comment of the proposed policy statement, modified in accordance with Commission direction, and accompanied by a conformed discussion report. We anticipate a 90-day public comment period. The draft Federal Register Notice asks for comments on all aspects on which commenters wish to offer views as well as on some specific questions included in the Notice.

I know that the Commission will want to have the benefit of ACRS review and comment on the proposed policy statement. You may expect a request to complete your review within the 90-day period that the policy statement is out for public comment.

In developing this draft policy statement, the Commission has solicited and benefited from information and suggestions provided by workshop discussions. Two NRC-sponsored workshops have been held, the first in Palo Alto, California, on April 1-3, 1981 and the second in Harpers Ferry, West Virginia, on July 23-24. The first workshop addressed general issues involved in developing safety goals. The second workshop focused on a discussion paper which presented proposed safety goals. Both workshops featured discussion among knowledgeable persons drawn from industry, public interest groups, universities, and elsewhere, and representing a broad range of perspectives and disciplines.

In preparing the safety-goal policy paper we were aided by the ACRS's paper on An Approach to Quantitative Safety Goals for Nuclear Power Plants. We have had the benefit of discussion with the full ACRS during the early, formative stages of the project and subsequently with Dr. Okrent's Subcommittee and with several Committee members individually. NRC safety-goal workshop participation by Doctors Lewis and Okrent and Morton Libarkin's membership on the InterOffice Steering Group on Development of a Safety Goal provided further contributions from ACRS members and staff.

In arriving at a final decision on a statement of its nuclear power plant safety policy and goals, the Commission will take into consideration the comments and suggestions received from the public in response to the proposed policy statement, as well as this Committee's advice.

I shall now outline for you the substance of the proposed safety-goal policy statement as it now stands. Since Commission consideration of the statement is approaching completion, I do not believe that what will be issued for public comment -- and for your own review and advice -- will differ greatly from what I am going to describe today.

The eventual policy statement would state the Commission's views on the acceptable level of risks to public health and safety and on the safety-cost tradeoffs in regulatory decisionmaking. The proposed policy statement focuses on one matter of special public concern at the present time: nuclear power plant accidents. It does not deal with risks from routine emissions, from the nuclear fuel cycle, from sabotage or earthquakes, or from diversion of nuclear weapons-grade material.

The Commission proposal would adopt qualitative safety goals supported by provisional numerical guidelines. Two qualitative safety goals are proposed. The intent of the first is to require a level of safety such that individuals living or working near nuclear power plants should be able to go about their daily lives without special concern by virtue of their proximity to such plants. The first proposed qualitative safety goal is:

- Individual members of the public should be provided a level of protection from the consequences of nuclear power plant accidents such that no individual bears a significant additional risk to life and health.

The second proposed qualitiative goal would place a limit on the societal risks posed by reactor accidents. This proposed goal has two elements. First, the risks of accidents should be such that, when added to the risk of normal operation, the total risk to the public from an operating nuclear power plant would be comparable to or less than the risk from other viable means of generating the same quantity of electrical energy. Second, the risks of accidents should be reduced to the extent that is reasonably achievable through the application of available technology. The second proposed safety goal reads:

- Societal risks to life and health from nuclear power plant accidents should be as low as reasonably achievable and should be comparable to or less than the risks of generating electricity by viable competing technologies.

The comparative part of this goal is to be interpreted as requiring that the risks from accidents should be low enough that the total risks of nuclear power plants resulting from normal operation and accidents are comparable to or less than the total risks of the operation of competing electricity generating plants.

Now, to turn to the proposed provisional numerical guidelines. A key element in formulating a safety policy which establishes numerical guidelines is to understand both the strengths and limitations of the techniques by which one judges whether these guidelines have been met.

Since the completion of the Reactor Safety Study in 1974, further progress in developing probabilistic risk assessment and in accumulating relevant data has led to recognition that it is feasible to begin to use quantitative reactor safety guidelines for limited purposes. However, because of the sizable uncertainties still present in the methods and the gaps in the data base--essential elements needed to gauge whether the guidelines have been achieved--the quantitative guidelines should be viewed as aiming points or numerical benchmarks which are subject to revision as further improvements are made in probabilistic risk assessment. In particular, because of the present limitations in the state of the art of quantitatively estimating risks, the numerical guidelines are not substitutes for existing regulations.

For individual and societal mortality risks the following two provisional numerical guidelines are proposed:

- The risk to an individual or to the population in the vicinity of a nuclear power plant site of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.
- The risk to an individual or to the population in the area near a nuclear power plant site of cancer fatalities that might result from reactor accidents should not exceed one-tenth of one percent of the sum of cancer fatality risks resulting from all other causes.

This 0.1% ratio of the risks of nuclear power-plant accidents to the risks of accidents of non-nuclear-plant origin is intended to reflect the first qualitative goal, which would provide that no individual bear a significant additional risk. In addition, the 0.1% figure is consistent with the provision of the second qualitative safety goal, which seeks to keep risks as low as reasonably achievable. It is also consistent with the comparative provision of the second qualitative safety goal, since calculations suggest that the risk of accidents at a nuclear power plant that is consistent with the proposed numerical guidelines would compare favorably with risks of viable competing technologies. The 0.1 percent ratio to other accident risks is low enough to support an expectation that people living or working near nuclear power plants would have no special concern due to the plant's proximity.

The individual risk is taken as the estimated probability of fatality from a nuclear power-plant accident for an individual in the vicinity of the plant, including prompt deaths and delayed deaths. The individual risk limit is applied to the biologically average individual (in terms of age and other risk factors) who resides at a location within 1 mile from the plant site boundary.

In applying the numerical guideline for prompt fatalities as a population guideline, the statement proposes to define the vicinity as the area within 1 mile of the nuclear power-plant site boundary, since calculations of the consequences of major reactor accidents suggest that individuals in the population within a mile of the plant site boundary would be subject to the greatest risk of prompt death attributable to radiological causes. Beyond this distance, atmospheric dispersion and radioactive decay of the airborne radioactive materials sharply reduce the radiation exposure levels and the corresponding risk of prompt fatality.

In applying the numerical guideline for cancer fatalities as a population guideline, the statement proposes that the population considered subject to significant risk be taken as the population within 50 miles of the plant site. A substantial fraction of exposures of the population to radiation would be concentrated within this distance. This guideline would ensure that the potential increase in delayed cancer fatalities from all reactor accidents at a typical site would be no more than a small fraction of the year-to-year normal variation in the expected cancer deaths from non-nuclear causes. Moreover, the limit protecting individuals provides greater protection to the population as a whole. That is, if the guideline is met for individuals in the immediate vicinity of the plant site, the risk to persons much farther away would generally be much lower than the limit set by the guideline. Thus, compliance with the guideline applied to individuals close to the plant would generally mean that the aggregated societal risk for a 50-mile-radius area would be a number of times lower than it would be if compliance with just the guideline applied to the population as a whole were involved.

A third guideline addresses benefit-cost tradeoffs. It calls for reduction of individual and societal risks below the levels specified in the first and second numerical guidelines in accordance with the "as low as reasonably achievable" (ALARA) principle. It proposes that a guideline of $1,000 per man-rem averted be adopted for provisional use and subject to revision in the light of public comments. It reads:

- The benefit of the incremental reduction of risk below the numerical guidelines for societal mortality risks should be compared with the associated costs on the basis of $1,000 per man-rem averted.

This guideline is intended to encourage the efficient allocation of resources in safety-related activities by providing that the expected reduction in public risk that would be achieved should be commensurate with the costs of the proposed safety improvements. The benefit of an incremental reduction of risk below the numerical guidelines for societal mortality risks should be compared with the associated costs, including all reasonably quantifiable costs (e.g., design and construction of plant modifications, incremental cost of replacement power during mandated or extended outages, changes in operating procedures and manpower requirements).

Justification of proposed plant design changes or corrective actions would be related to the reduction in risk to society measured as a decrease in expected population exposure (expressed in man-rem) under accident conditions. To take into account the fact that a safety improvement would reduce the public risk during the entire remaining lifetime of a nuclear power plant, both the estimated cost of the improvement and the benefit (risk reduction) should be adjusted to reflect only the remaining years during which the plant is expected to operate (i.e., annualized).

Because of the substantial uncertainties inherent in probabilistic risk assessments of potential reactor accidents, especially in evaluation of accident consequences, the statement proposes a limitation on the probability of a core melt as a provisional guideline for NRC staff use in the course of reviewing and evaluating probabilistic risk assessments of nuclear power plants. The proposed guideline is as follows:

> The likelihood of a nuclear reactor accident that results in a large-scale core melt should normally be less than one in 10,000 per year of reactor operation.

The statement also recognizes the importance of mitigating the consequences of a core-melt accident, and calls for continued emphasis on containment, remote siting, and emergency planning as integral parts of the defense-in-depth concept.

With respect to implementation, the proposed intention is that the goals and guidelines would be used by the NRC staff in conjunction with probabilistic risk assessments and would not substitute for NRC's reactor regulations. Rather, individual licensing decisions would continue at present to be based principally on compliance with the Commission's regulations.

In all applications of the goals and guidelines, the probabilistic risk assessments, if performed, should be documented, along with the associated assumptions and uncertainties, and considered as one factor among others in the regulatory decisionmaking process. The nature and extent of the consideration given to the numerical guidelines in individual regulatory decisions would depend on the issue itself, the quality of the data base, and the reach and limits of analyses involved in the pertinent probabilistic calculations. The proposed numerical guidelines should aid professional judgment, not replace judgment with mathematical formulas.

The proposed numerical benefit-cost guideline may be used during the trial period as one consideration in deciding whether corrective measures or safety improvements should be made in plants previously approved for construction or operation. Benefits should be measured in terms of estimated annual reduction in radiological risk due to reactor accidents.

The Commission will, I believe, request the staff to develop a specific action plan for implementation of the proposed qualitative safety goals and numerical guidelines. The plan should indicate for Commission review and approval how the NRC staff plans to use the goals and guidelines in conjunction with probabilistic risk assessments.

The following pages A-68 thru A-97 has been deleted as 1 .

DELETION

VUGRAPHS

STATUS REPORT:

PROPOSED POLICY STATEMENT ON SAFETY GOALS FOR NUCLEAR POWER PLANTS

FORREST J. REMICK, DIRECTOR
OFFICE OF POLICY EVALUATION

PRESENTATION BEFORE THE
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

FEBRUARY 4, 1982

QUALITATIVE GOALS

INDIVIDUAL RISK

INDIVIDUAL MEMBERS OF THE PUBLIC SHOULD BE PROVIDED A LEVEL OF PROTECTION FROM THE CONSEQUENCES OF NUCLEAR POWER PLANT ACCIDENTS SUCH THAT NO INDIVIDUAL BEARS A SIGNIFICANT ADDITIONAL RISK TO LIFE AND HEALTH.

A-100

QUALITATIVE GOALS

SOCIETAL RISK

SOCIETAL RISKS TO LIFE AND HEALTH FROM NUCLEAR POWER PLANT ACCIDENTS SHOULD BE AS LOW AS REASONABLY ACHIEVABLE AND SHOULD BE COMPARABLE TO OR LESS THAN THE RISKS OF GENERATING ELECTRICITY BY VIABLE COMPETING TECHNOLOGIES.

A-101

PROVISIONAL NUMERICAL GUIDELINES

GUIDELINES ON INDIVIDUAL AND SOCIETAL MORTALITY RISKS

- THE RISK TO AN INDIVIDUAL OR TO THE POPULATION IN THE VICINITY OF A NUCLEAR POWER PLANT SITE OF PROMPT FATALITIES THAT MIGHT RESULT FROM REACTOR ACCIDENTS SHOULD NOT EXCEED 0.1% OF THE SUM OF PROMPT FATALITY RISKS RESULTING FROM OTHER ACCIDENTS TO WHICH MEMBERS OF THE U.S. POPULATION ARE GENERALLY EXPOSED.

- THE RISK TO AN INDIVIDUAL OR TO THE POPULATION IN THE AREA NEAR A NUCLEAR POWER PLANT SITE OF CANCER FATALITIES THAT MIGHT RESULT FROM REACTOR ACCIDENTS SHOULD NOT EXCEED 0.1% OF THE SUM OF CANCER FATALITY RISKS RESULTING FROM ALL OTHER CAUSES.

PROVISIONAL NUMERICAL GUIDELINES

Benefit-Cost Guideline

- The benefit of the incremental reduction of risk below the numerical guidelines for societal mortality risks should be compared with the associated costs on the basis of $1,000 per man-rem averted.

A-103

PROVISIONAL NUMERICAL GUIDELINES

CORE-MELT GUIDELINE

THE LIKELIHOOD OF A NUCLEAR REACTOR ACCIDENT THAT RESULTS IN A LARGE-SCALE CORE MELT SHOULD NORMALLY BE LESS THAN ONE IN 10,000 PER YEAR OF REACTOR OPERATION.

A-104

IMPLEMENTATION

RETENTION OF REGULATIONS

- THE GOALS AND GUIDELINES WOULD NOT SUBSTITUTE FOR NRC'S REACTOR REGULATIONS.

    -- LICENSING DECISIONS WOULD CONTINUE TO BE BASED PRINCIPALLY ON COMPLIANCE WITH THE COMMISSION'S REGULATIONS.

A-105

IMPLEMENTATION

ONE FACTOR AMONG OTHERS

- IN ALL APPLICATIONS OF THE GOALS AND GUIDELINES, THE PROBABILISTIC RISK ASSESSMENTS, IF PERFORMED, SHOULD BE DOCUMENTED, ALONG WITH THE ASSOCIATED ASSUMPTIONS AND UNCERTAINTIES, AND CONSIDERED AS ONE FACTOR AMONG OTHERS IN THE REGULATORY DECISION-MAKING PROCESS.

  -- THE NATURE AND EXTENT OF THE CONSIDERATION GIVEN TO THE NUMERICAL GUIDELINES IN INDIVIDUAL REGULATORY DECISIONS WOULD DEPEND ON THE ISSUE ITSELF, THE QUALITY OF THE DATA BASE, AND THE REACH AND LIMITS OF ANALYSES INVOLVED IN THE PERTINENT PROBABILISTIC CALCULATIONS.

  -- THE PROPOSED NUMERICAL GUIDELINES SHOULD AID PROFESSIONAL JUDGMENT, NOT REPLACE JUDGMENT WITH MATHEMATICAL FORMULAS.

A-106

IMPLEMENTATION

Possible Consideration in Backfit Tradeoffs

- THE BENEFIT-COST GUIDELINE MAY BE USED AS ONE CONSIDERATION IN DECIDING WHETHER CORRECTIVE MEASURES OR SAFETY IMPROVEMENTS SHOULD BE MADE IN PLANTS PREVIOUSLY APPROVED FOR CONSTRUCTION OR OPERATION.

IMPLEMENTATION

Action Plan

- The NRC staff should develop a specific ation plan for implementation of the proposed qualitative safety goals and numerical guidelines. The plan should indicate for Commission review and approval how the staff plans to use the goals and guidelines in conjunction with probablistic risk assessments.

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SUPPLEMENTARY VUGRAPHS

STATUS REPORT:

PROPOSED POLICY STATEMENT ON SAFETY GOALS FOR NUCLEAR POWER PLANTS

FORREST J. REMICK, DIRECTOR
OFFICE OF POLICY EVALUATION

PRESENTATION BEFORE THE
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

FEBRUARY 4, 1982

APPENDIX VIII
PROPOSED POLICY STATEMENT ON SAFETY
GOALS FOR NUCLEAR POWER PLANTS -
SUPPLEMENTARY VIEWGRAPHS

ADDITIONAL QUESTIONS

1. Benefit-Cost Tradeoffs

Should the benefit side of the tradeoffs include, in addition to the mortality risk reduction benefits, the economic benefit of reducing the risk of economic loss due to plant damage and contamination outside the plant?

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ADDITIONAL QUESTIONS

2. Containment Guideline

SHOULD THERE BE ADDED A NUMERICAL GUIDELINE ON AVAILABILITY OF CONTAINMENT FUNCTION, GIVEN A LARGE-SCALE CORE MELT?

ADDITIONAL QUESTIONS

3. Implementation

A. What further guidance, if any, should be given for decisions under uncertainty?

B. What further guidance, if any, should be given on resolution of possible conflicts among quantitative aspects of some issue?

C. What approach should be used with respect to accident initiators which are more difficult to quantify, such as seismic events, sabotage, multiple human errors, and design errors?

D. Should there be definition of the numerical guidelines in terms of median, mean, 90 percent confidence, etc.? If so, what should be the terms?

ADDITIONAL QUESTIONS

3. <u>Implementation (Continued)</u>

E. Should the staff action plan include further specification of a process which will lend credibility to the use of quantitative guidelines and methodology? If so, what should be the principal bases and elements of such guidance?

F. On what basis should the numerical guidelines be applied to protection of individuals? Should they be applied to the individual at greatest risk, or should they be used in terms of an average risk limit over a region near the plant? Any comments or suggestions pertaining to the present discussion of this topic (or other specifics) would be welcome.

ADDITIONAL QUESTION

4. Risk Aversion

Should there be specific provision for "risk aversion"? If so, what quantitative or other specific provision should be made?

APPENDIX IX
INSTRUMENTATION FCR DETECTION OF
INADEOUATE CORE COOLING IN
PRESSURIZED SATER REACTORS

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

JAN 29 1982

RECEIVED
ADVISORY COMMITTEE ON
REACTOR SAFEGUARDS, U.S.N.R.C.

FEB 2 1982
AM PM
7,8,9,10,11,12,1,2,3,4,5,6

MEMORANDUM FOR: Chairman Palladino

FROM: William J. Dircks, Executive Director
for Operations

SUBJECT: INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING
IN PRESSURIZED WATER REACTORS

This is in response to your memorandum of January 19 concerning our plans to address the issues considered at the Commission's meeting of January 8, 1982 on the subject instrumentation.

We have scheduled a two-day NRC/Industry meeting for mid-February. The level measurement suppliers are being asked to give presentations assessing the performance of their proposed instrumentation systems for a broad spectrum of accident scenarios. These presentations are being specifically designed for response to the issues discussed at the January 8, 1982 meeting. The vendors have been requested to address the points raised in Enclosure 1 to this memorandum. We will also invite representatives of licensed plants to participate in the exchange with the suppliers to assure adequate attention to the operational aspects of the issues that have been raised.

Subsequent to our meeting with licensees and vendors, the staff will discuss the requirements and proposed designs with the Committee to Review Generic Requirements (CRGR) of the NRC and seek that committee's guidance.

We expect that an agenda will then be established for detailed industry and staff presentations to the ACRS subcommittee and full committee in March. These presentations will reflect guidance received from the CRGR. By that time the staff's technical assistance contractor, Oak Ridge National

PES-6

Laboratory, will also have completed its review of the suppliers proposed systems. Following our discussions with the ACRS, we expect to have a recommendation for the Commission's consideration by the end of March, taking into account the reviews of the CRGR, the ACRS, our contractors and the staff.

William J. Dircks, Executive Director
for Operations

Enclosure:
Additional Instrumentation for
ICC in PWRs

cc: Commissioner Gilinsky
Commissioner Bradford
Commissioner Ahearne
Commissioner Roberts
OGC
OPE
SECY

ENCLOSURE

ADDITIONAL INSTRUMENTATION FOR INADEQUATE CORE COOLING IN PWRs

Please evaluate the capability of your existing and your proposed additional instrumentation for detection of inadequate core cooling in light of various types or classes of accident sequences and justify the sufficiency of the spectrum of accidents considered. Identify what information would aid the operators for the various accident sequences, and show how the various elements of your proposed complement of instruments contributes to supplying that information. Summarize the accident scenario presentation by describing the accident scenarios for which your proposed water level indicators will provide reliable information and those for which your water level indicators would not be useful or would give misleading or ambiguous readings. For these latter cases, what specific instructions, training or procedures would be provided to operators to prevent them from misinterpreting ambiguous indications and being misled. Include discussions of the integration with control room display of the measurements. Then explain how the symptom-oriented operating procedure guidelines will be integrated with the measurements and displays for the identified scenarios. What steps have you taken in system design and in procedure development to assure that the instruments provide complementary information and unambiguous guidance to the operators.

Discuss the design objectives for your proposed water level measurement system, and the bases of your selection and evaluation of specific instrumentation to measure water level. Summarize the other types of instrumentation you considered and the reasons they were rejected.

It has been suggested that water level indicators are superfluous to other inadequate core cooling indicators in PWRs. Please identify those parts of the accident scenarios where the information from water level indicators would be unique. What specific actions might be taken by operators because of the level measurement that would not otherwise be taken. Describe where

the water level information merely verifies other signals which are the basis for operator actions and those instances where it provides unique diagnostic information which is significant input to operating decisions. Contrast these various contributions to the potential ambiguities that the proposed level measurement systems may create. On balance, does your system help or hurt safety? Would you rely on it if you were an operator?

Discuss the quality of the information to be provided by the level monitoring instrumentation; i.e., what are the error bands under various circumstances when following the course of an accident. In particular, identify possible ranges of uncertainty when approaching core uncovery in times of rapid depressurization, rapid flow changes, reactor coolant pumps operating, ECCS pumps operating or severe core damage (flow blockage) and relate the significance of the uncertainty to interpretation and response to the event. Address the possibility and significance of circumstances where there could be an indication of water above the core while the core is actually partially uncovered or while local or global conditions of inadequate core cooling (temperature rise or sustained high temperature of the fuel) exist within the core.

Describe the procedures you recommend for implementation of an installed system, such as calibration and testing requirements, debugging, verification of displays, and operator training. When do you propose that the plant specific NRC implementation review be conducted.

Describe the development and verification testing programs for the proposed instrumentation. Discuss the results and how they have been used in the design evolution of the proposed instrumentation. Discuss conclusions from any test programs and show how the results demonstrate the capabilities and limitations of the proposed water level measurement systems.

Discuss qualification requirements and status of the final ICC monitoring instrumentation systems.

Mr. Ed Scherer
Combustion Engineering, Inc.
1000 Prospect Hill Road
Windsor, Connecticut 06095

JAN 28 1982

Dear Mr. Scherer:

The NRC staff has been reviewing the schedules and the status of your program for meeting TMI Action Item II.F.2, the requirements for inadequate core cooling measurement systems in light water reactors. Our review has involved a series of discussions with the industry, the ACRS and the Commission. The ACRS and representatives of level measurement suppliers made presentations to the Commissioners on January 8, 1982.

During the course of these discussions, a number of important questions have been raised. We have decided it is necessary to meet again to better articulate the purposes of inadequate core cooling measurements, to obtain a better understanding of the industry's general approach to these measurements, reactor water level indicators in particular, and to provide additional insight into the basis for your design selections. We invite you to meet with the staff on February 16 and 17 starting at 9:00 a.m. We are also inviting representatives of the applicable owners groups to participate in the meeting.

At a later date, you may also be asked to meet with the NRC Committee to Review Generic Requirements. It is our expectation that an agenda would then be established for detailed industry and staff presentations to the ACRS and its combined Electrical Systems and ECCS subcommittees in March.

At the meeting with the staff and other industry representatives on February 16 and 17, we request that you structure a formal presentation to address the points raised in Enclosure 1. The agenda for your presentation and others is provided in Enclosure 2. Please call L. S. Rubenstein of my staff if you have any questions regarding this subject.

Sincerely,

Original Signed by:
Roger J. Mattson

Roger J. Mattson, Director
Division of Systems Integration
Office of Nuclear Reactor Regulation

Enclosures:
1. Additional Information for ICC in PWRs.
2. Preliminary Agenda

IDENTICAL LETTERS TO:
Westinghouse - PRahe
B&W - JTaylor
NNC- LKornblith

A-119

ENCLOSURE 1

ADDITIONAL INSTRUMENTATION FOR INADEQUATE CORE COOLING IN PWRs

Please evaluate the capability of your existing and your proposed additional instrumentation for detection of inadequate core cooling in light of various types or classes of accident sequences and justify the sufficiency of the spectrum of accidents considered. Identify what information would aid the operators for the various accident sequences, and show how the various elements of your proposed complement of instruments contributes to supplying that information. Summarize the accident scenario presentation by describing the accident scenarios for which your proposed water level indicators will provide reliable information and those for which your water level indicators would not be useful or would give misleading or ambiguous readings. For these latter cases, what specific instructions, training or procedures would be provided to operators to prevent them from misinterpreting ambiguous indications and being misled. Include discussions of the integration with control room display of the measurements. Then explain how the symptom-oriented operating procedure guidelines will be integrated with the measurements and displays for the identified scenarios. What steps have you taken in system design and in procedure development to assure that the instruments provide complementary information and unambiguous guidance to the operators.

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the water level information merely verifies other signals which are the asis for operator actions and those instances where it provides unique diagnostic information which is significant input to operating decisions. Contrast these various contributions to the potential ambiguities that the proposed level measurement systems may create. On balance, does your system help or hurt safety? Would you rely on it if you were an operator?

Discuss the quality of the information to be provided by the level monitoring instrumentation; i.e., what are the error bands under various circumstances when following the course of an accident. In particular, identify possible ranges of uncertainty when approaching core uncovery in times of rapid depressurization, rapid flow changes, reactor coolant pumps operating, ECCS pumps operating or severe core damage (flow blockage) and relate the significance of the uncertainty to interpretation and response to the event. Address the possibility and significance of circumstances where there could be an indication of water above the core while the core is actually partially uncovered or while local or global conditions of inadequate core cooling (temperature rise or sustained high temperature of the fuel) exist within the core.

Describe the procedures you recommend for implementation of an installed system, such as calibration and testing requirements, debugging, verification of displays, and operator training. When do you propose that the plant specific NRC implementation review be conducted.

Describe the development and verification testing programs for the proposed instrumentation. Discuss the results and how they have been used in the design evolution of the proposed instrumentation. Discuss conclusions from any test programs and show how the results demonstrate the capabilities and limitations of the proposed water level measurement systems.

Discuss qualification requirements and status of the final ICC monitoring instrumentation systems.

IN RESPONSE REFER
TO M820106

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

January 29, 1982

OFFICE OF THE
SECRETARY

APPENDIX X
STAFF REQUIREMENTS - BRIEFING ON STATUS
AND PLAN FOR SEVERE ACCIDENT RULEMAKING

MEMORANDUM FOR: William J. Dircks, Executive Director for Operations
Forrest Remick, Director, Policy Evaluation

FROM: Samuel J. Chilk, Secretary

SUBJECT: STAFF REQUIREMENTS - BRIEFING ON STATUS AND PLAN FOR SEVERE ACCIDENT RULEMAKING (SECY-82-1), 10:05 A.M., WEDNESDAY, JANUARY 6, 1982, COMMISSIONERS' CONFERENCE ROOM, DC OFFICE (OPEN TO PUBLIC ATTENDANCE)

The Commission was briefed by staff on a new approach to the severe accident rulemaking as described in SECY-82-1. The staff expects to receive standard plant designs from industry and proposed that the Commission use rulemaking to approve the specific plant designs. The standard plants would be expected to meet or exceed requirements in the new CP/ML rule and the updated SRP. The Commission agreed to consider substituting this approach for a generic severe accident rulemaking. The staff was directed to revise the draft policy statement to address Commissioner comments including:

1. Careful consideration should be given to ensuring that conflicting or incorrect signals are not sent to industry relative to significant matters that were to be contained in the long term rulemaking proceedings, e.g., filtered vented containment, core retention devices, hydrogen control measures, thicker basemats, and other items listed in the Action Plan. To the extent feasible, include a list of these items in the policy statement and say explicitly how they are to be treated in the review of new applications.

2. More specific guidance pertaining to design criteria and minimum safety requirements considered necessary by the Commission should be specified, e.g. stronger containments.

3. Policy regarding backfitting of new requirements resulting from acceptance of standardized designs should be addressed.

4. The requirements set forth in the new CP rule should be applied to any FDA's referenced in future applications.

5. Since the safety goal is not yet in effective form, the continued reference to and reliance on it should be qualified in the proposed Policy Statement.

6. It should be emphasized that PRA is only one of several tools used in the development of safety rulemakings.

7. References to current siting policy or practice should be supplemented by noting that work is being done to further refine the policy.

8. References to one-step licensing should be qualified since the Commission has taken no action toward adoption of such a policy.

9. The following statements should be deleted from the Policy Statement:

   Page 2: "The Commission does not believe that, in the interim, this continuing research should be a deterrent to the placement of orders or the initiation of licensing reviews for new CP applications."

   Page 4: "...current generation light water reactors are estimated to be close to or below the risk levels we believe acceptable, and that..."

10. The new CP rule applies to only a narrow group of CP applications. The Policy Statement should indicate that the CP rule is at least a minimum requirement for new plants, but should not suggest that it will be sufficient for new plants.

The staff should obtain ACRS input to the Policy Statement.

The revised Policy Statement should be forwarded to the Commission for approval. (NRR/RES) (SECY SUSPENSE: 3/1/82)

The Commission also requested that, if the new policy is adopted, the staff show how:

1. they will ensure that IDCOR effort continues;

2. they will ensure that NRC research and other programs critical to this approach are continued. (NRR/RES) (SECY SUSPENSE: 3/1/82)

The staff indicated that:

- A paper containing plans for producing research information needed to confirm regulatory decisions in the severe accident area, including methodology for comparing the cost of proposed new requirements with their risk reduction, and generalized reduction in the uncertainty of PRA, will be provided to the Commission by February 25, 1982.

- Technical justification concerning the need for severe accident features on operating plants will be available from industry (IDCOR) and NRC studies of degraded core accidents in mid to late 1983.

- A revised staff estimate of the accident source term, which affects siting, emergency planning and severe accident PRA will be completed in 18 months to two years.

- A Policy Statement on safety goals should be issued in final form by July, 1982.

cc: Chairman Palladino
Commissioner Gilinsky
Commissioner Bradford
Commissioner Ahearne
Commissioner Roberts
Commission Staff Offices
ACRS
ASLBP
ASLAP
Public Document Room
DCS-016 (Phillips)

NUREG-0900
(DRAFT)

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1-28-82

NUCLEAR PLANT SEVERE ACCIDENT RESEARCH PLAN

This draft is circulated internally for review and comment. While it has had limited peer review, it may contain technical errors, and it refers to policy issues, developed in SECY 82-1, that are still unresolved by the Commission. Therefore, this draft is not in suitable form for public release, discussion or reference.

A-126

01/29/82

TABLE OF CONTENTS

1.0 INTRODUCTION

1.1 Objectives

This plan describes the coordinated research programs needed to develop a sound technical basis for Nuclear Regulatory Commission decisions concerning the ability of existing or planned nuclear power reactors to cope with severe accidents, i.e, those which involve damaged or melted fuel. It is expected that the major application of this program will be to support regulatory decisions on new standardized plants and plants in the early and mid-1980's. Also, some provision maybe needed for backfitting to operating plants, consistent with safety goal policy yet to be developed. To ensure a sound technical basis for these regulatory decisions, two categories of information will be developed: one, a manageable analysis process and models to assess benefits in terms of residual risk reduction and the accompanying costs; and, two, a base of data related to the behavior of nuclear power plant systems and components under a range of severe accident conditions, so that the risk analysis process can be applied knowledgeably.

We expect that these cateogries of information will be used for three applications to provide:

1. Technical bases for more precise appraisal of specific design and operational refinements to permit further risk reduction by enabling a clear identification of worthwhile changes (value impact) in present design or operating practices, as opposed to major redesign.

2. Confirmation of the level of safety of plants.

3. More accurate probabilistic risk assessment methods for use in regulatoy decision-making and to provide greater assurance of safety.

It is our goal to have a comprehensive base of data for regulatory decisions within four years, with significant intermediate results at the end of two years. As is normal, we expect some work to continue after four years, but at a lower overall level of activity. The plan provides for the integration of data into regulatory end products such as guides and standards.

1.2 Background

Task II.B.8, "Rulemaking Proceeding on Degraded Core Accidents," of the TMI Action Plan (NUREG-0660, May 1980) envisioned a long-term rulemaking extending beyond 1982 to establish policy, goals, and requirements related to accidents involving core damage greater than that of the present design basis. The task also included the interim steps of an Advanced Notice of Rulemaking and an Interim Rule. The Advanced Notice of Rulemaking was issued on December 2, 1980 (45 FR 65474). The Interim Rule has two parts, the first issued in effective form on December 2, 1981 (46 FR 58484), and the second issued as a proposed rule on December 23, 1981 (46 FR 62281).

The TMI Action Plan stated that the long-term rulemaking would consider several significant matters not addressed in the Interim Rule, namely:

- Use of filtered, vented containment,

- Hydrogen control measures,

- Core retention devices,
- Reexamination of design criteria for decay heat removal, and other systems,
- Post-accident recovery plans,
- Criteria for locating highly radioactive systems,
- Effects of accidents at multi-unit sites, and,
- Comprehensive review and evaluation of related guides and regulations.

After issuance of the TMI Action Plan, the rulemaking efforts involving severe accidents, siting, and emergency preparedness were coordinated. To that end, the EDO created a Degraded Core Cooling Steering Group, which functioned from period October 1980 through April 1981. Its report contained a plan recommending, among other things, several years of extensive research. That research program is now well under way. The program plan for the research work is delineated in some detail in this report.

1.3 Description of Contents

The report is organized as follows:

Chapter 2 discusses discuss the information needs and regulatory issues addressed by the plan while Chapter 3 describes the state of the art. Chapter 4 presents a brief discussion of how the detailed elements of the program are linked and estimates the schedule for production of key results, both interim and final. Chapter 5, describes each of the program elements in detail, and Chapter 6 summarizes the advantages of the approach, as well as some possible pitfalls.

In the chapters to follow we discuss a number of computer codes in varying states of development the codes discussed fall into two generic classes; those used in carrying out Probabilistic Risk Analysis, and those used in deterministic studies to develop technical specifications for regulatory guides and rules. The latter class of codes is composed of computer programs that describe operating phenomena in great detail and which are validated against experimental data. The former class is composed of codes that seek to represent the lumped effects or consequences of a series of events, in order to understand the progression of events given assumed set of faults. These codes, which are used for Probabilistic Risk Analysis are fast running, obtaining their lumped representation of effects from the more detailed, deterministic codes.

A brief glossary of codes by class is furnished in Table 1-1.

Table 1-1

Deterministic Codes	Probabilistic Analysis Codes
SCDAP - Severe Core Damage Analysis	MARCH - Models of melt down event sequences
HECTR - Hydrogen Combustion	MELCOR - Models of melt down event sequences (improved MARCH)
CORCON - Fuel/Concrete Interaction	CORRAL - Models of event sequences in containment
TRAP-MELT - Fission product release and transport in primary system	MATADOR - Improved Containment Code
CONTAIN - Detailed prediction of containment loadings	

With respect to program pace, we find in Chapter 4 that if the program elements are maintained at an even pace and if there is a good exchange of information among the elements, interim results can be produced to answer some of the needs for information well in advance of program completion. Nominally the entire program will take about four years. But the synergism induced by linking the programs and adding improved coordination and communication to existing efforts should permit the program to be focused more tightly as times goes on, thereby continually reducing the scope of and sharpening the issues remaining to be resolved.

2.0 INFORMATION NEEDS AND REGULATORY ISSUES

Resolution of the generic and specific regulatory issues will require a substantial body of organized information. We examine the information needs related to the issues and report these findings in this chapter. We then examine the state of the art in each of the areas to ascertain what we now know. The results are reported in Chapter 3. The difference between knowledge needed and knowledge on hand represents the body of technical material to be developed by the program; details of the program are described in later chapters. The budget decision units and subelements involved are: (1) Probabilistic Risk Analysis (PRA); (2) Accident Management; (3) Behavior of Damaged Fuel; (4) Fission Product Release and Transport; (5) Fuel-Melt Interaction, and (6) Accident Mitigation.

Three bodies of organized information are projected as output of the program:

1. Data for guidelines for refinements to system design, operating procedures, and instruments;

2. Verified methodology for accident load phenomena, and system responses;

3. Information for decisions on potential risk reduction add-ons and refinements.

The plan provides for transformation of these products into regulatory end products, (i.e. regulations, guides and revisions in the standard review plan).

Most current questions about severe accidents result from consideration of the accident at Three Mile Island Unit 2 (TMI-2), with some additional questions arising from other accidents with potentially serious consequences such as the Brown's Ferry fire. The need to focus on severe accidents was documented in the Reactor Safety Study (WASH-1400) but detailed technical questions were not adequately framed until the accidents provided numerous focal points of inquiry.

The accident at the TMI-2 on March 28, 1979, was a severe reactor accident. Although the accident produced virtually no offsite radiological consequences, it did great damage to the reactor and raised serious questions about the adequacy of the regulation of nuclear power plants in the United States. In the process of regulation, practice had been to test the adequacy of nuclear plant design against a set of design basis accidents that were believed to constitute a sufficient envelope of credible scenarios. System reliability was "assured" of meeting regulatory requirements by using a postulated single failure criterion in the safety analysis, quality assurance procedures, and inservice inspection and testing. The acceptability of reactor sites was tested by a hypothetical accident dose calculation that combined the most serious design basis accident with a postulated nuclear core damage and a radioactivity release level believed to represent severe accident phenomena to an adequate degree.

The TMI-2 accident challenged the validity of many of these practices. The events of the accident did not fit the envelope of design basis accidents (DBA). Events did not follow the simple binary logic postulated in the DBA in which things either worked or they failed. At TMI-2, core cooling was not completely lost but severely degraded. The core was badly damaged, but there was no significant core melting. Large quantities of hydrogen were formed, released and burnt during the prolonged core damage sequence, rather than the small amount prescribed in 50.44 of 10 CFR Part 50 as analyzed as part of the design basis analysis. Large quantities of radioactive fission products were found in the coolant water, greatly restricting the ability to circulate cooling water for safe shutdown. The released fission products so pervaded the plant that personnel access was made very difficult. The operating crew committed repeated and persistent errors, failing to diagnose the accident causes. In sum, a host of questions were raised about the adequacy of plant design and operation and of NRC regulations for dealing with severe accidents.

In particular, three key questions representative of the major concerns, were raised in the report of the President's Commission (the Kemeny report):

1. Page 15 - How can we identify and analyze the possible consequences of accidents leading to severe core damage? "Such knowledge is eseential for coping with the results of future accidents."

2. Page 15 - How can we prevent such accidents and minimize the potential

impacts on the public health and safety?

3. Page 72 - What are the consequences and probabilities of such accidents, including the consequences of meltdown?

Our aim is to see what information is needed to answer these questions. The information we seek is categorized by NRC budget decision units and subelements because of their relationship to the Long-Range Research Plan, but this categorization is otherwise arbitrary.

2.1 Probabilistic Risk Assessment

The TMI-2 accident dramatized the inadequacies of traditional regulatory treatment of severe accidents. The elements of the TMI-2 accident scenario seemed to affirm the principal factors of accident risk as described in the Reactor Safety Study (WASH-1400), which used Probabilistic Risk Assessment (PRA) to obtain as realistic as possible a description of severe accident behavior and risks. The Reactor Safety Study stopped at the risk assessment of only two plants as surrogates for the first hundred. More plant specific risk assessments are needed to develop a technical basis for regulatory decisions regarding severe accidents.

We now realize that the two Reactor Safety Study plants are not apt surrogates for the variety of plant system and containment designs which exist. We need more representative PRA models of each basic type of plant. If we are to use these models for regulatory decisions regarding severe accidents, we must assess the level of severe accident risk as well as the relative risk reduction benefits of changes in plant design or operation. The many questions raised are:

1. What are the probabilities of specific accident sequences?

2. What are the consequences of these individual accident sequences, and how do they contribute to overall risk?

3. What are the risk reduction effects of notable changes in plant design or operation?

4. What are the savings possible from averted losses?

5. What are the costs of changes to reduce risk or avert loss?

6. What are the risk control merits of current regulatory practices and how can they be improved?

The physical data necessary to apply to improved PRA techniques will be acquired by a program of physical research comprised of five technical elements. These elements correspond in general to the areas of difficulty encountered in casework such as the Zion-Indian Point Study (NUREG/CR-1409, 10, 11) and later reviews. The Zion-Indian Point Study was an initial attempt to coordinate the use of PRA and best estimate physical modeling to determine the potential value of methods for reducing residual risk from a specific set of nuclear power plants.

The five remaining elements that have been identified and now appear as budget subelements are: Severe Accident Sequence Analysis (Accident Management Guidelines); Behavior of Damaged Fuel; Fission Product Release and Transport; Fuel-Melt Interactions (Containment Failure Processes), and Accident Mitigation research. The program of physical research is narrowly defined to produce data for PRA and to be capable of defining objectives more precisely as better PRA results become available. Therefore, it is important that intermediate products be available to enable better use of PRA before the program is largely complete. The plant of physical research is designed to allow this, with major intermediate results in the second and third years.

We next address the information needs and regulatory issues associated with the five elements that make up the physical research program.

2.2 Severe Accident Sequence Analysis

The Research budget subelement addressing this particular technical area called Severe Accident Sequence Analysis (SASA). As pointed out above, the examination of the accident at TMI-2 raised a host of questions about plant operation, among other things, with respect to the tactics for dealing with severe accidents. Actually, the potential for improving accident management techniques to reduce risk was first recognized after studying operator actions during the Brown's Ferry fire. Subsequent events at TMI-2 reinforced the idea that systematic studies of accident management will yield useful guidelines for emergency procedures under multiple failure conditions. The regulatory issues raised by these considerations are:

1. Should guidelines be established for operator response during severe accidents?

2. Should there be additional instrumentation and information on requirements to assist the management of severe accidents?

3. Should the operator be required to take actions to interdict fission product transport and mitigate containment failure during severe accidents?

4. Should the regulations involving emergency response reflect emergency procedure guidelines?

5. Should the design bases for handling major fission product releases be revised, and corresponding equipment qualification standards?

The SASA program has developed a detailed program plan that is condensed within this report. This program will complete major milestones by the end of FY 1983 with respect to management of accidents to reduce the likelihood of progression of significant fuel failure. Also included in the plan are accident studies extending beyond the point of significant fuel failure scheduled for completion at the time when more comprehensive data about the behavior of cores with severely degraded cooling have been acquired in the Severe Fuel Damage research program. In general, accident management is an attempt to prevent significant core damage. Existing procedures fill this function under the single failure criterion, and SASA is attempting to extend the process to multiple failures; management can also attempt to limit damage progression, and SASA will focus on these procedures increasingly as Severe Fuel Damage data become available; and, finally, management guidelines are required for optimum use of accident mitigation features in a presumed case of large scale core melt.

2.3 Severe Fuel Damage

The Severe Fuel Damage program is the direct outgrowth of estimates of the core history during the TMI-2 accident. These estimates were made in support of the Special Inquiry Group (Rogovin Committee). The difficulties encountered in making those estimates led to the definition of a research program that would overcome these problems. Such a program is worthwhile because the estimates lead us to believe that the ability to cope with a degraded core cooling accident that might otherwise lead to a core melt accident can be markedly improved if:

1. The plant is properly instrumented and the data properly processed and presented,

2. Accident management strategies and tactics are thought out in advance, and operators are trained in their use, and

3. Core cooling equipment is properly protected against adverse environments.

The objective of the Severe Fuel Damage Program is to supply the necessary data to enable the use of such measures.

A significant extension of existing technology is required to meet information needs regarding the behavior of severely damaged fuel to address the following regulatory issues:

1. What water inventory, distribution, and flow rate is needed to cool the core corresponding to its state of damage? How do these items compare with systems in place at plants?

2. What guidelines are there to indicate the optimum method of restoring cooling so as to minimize the potential hazard to public health and safety?

3. What is the rate at which the fission products and hydrogen are being produced and transported to the containments?

The consequences of a severe accident are dependent upon the sequence of damage in the accident, and there are many paths that degraded core cooling accidents can take. The Severe Fuel Damage Program is designed to map in a rudimentary way the complex response surface defined by the damage phenomena produced by varying certain key parameters i.e., heating rate, cooling rate, steam flow, peak temperature, fuel rod burnup, bundle size, and the presence of low melting point control and structural materials. This mapping of the damage phenomena is needed to bound the range of the effects of these various parameters. Depending on the parameter, severe fuel damage states and configurations can range from fuel rods with cladding totally oxidized to ZrO_2 and geometry altered only by localized rod ballooning and rupture of the rods during the heatup; to rods with the formation, relocation, and freezing of molten cladding and liquefied fuel; to the formation of rubble beds of fuel pellet fragments, oxidized cladding fragments, solidified molten fuel, solidified liquefied fuel, and solidified spacer grid and control rod materials. Data needed on the amount and timing of the release of fission products and the generation of hydrogen are also obtained along with the information on damage progression. All the data will be used to characterize the resulting core geometry so that realistic coolability studies of such configurations can be made both in-reactor and ex-pile to answer the important question of how to maintain and manage a damaged core without further degradation and additional fission product release.

Information on the progression and character of damaged fuel and related coolability will provide needed technical bases for developing accident management guidelines and potential refinements to system design. Of particular significance is the quench with core damage state to minimize is the quench with core damage state to minimize additional damage incident to quenching. This information need not, in general, be highly detailed. The emphasis is on categorizing the state of damage and correlating the damage state with flow properties and hydrogen generation and fission product release.

Had portions of the TMI-2 reached the melt stage, the predictions consistent with PRA scenarios are that the molten fuel would have attacked the primary vessel. The existing PRA models of the attack are limited because of a lack of detailed knowledge of this process. The vessel attack by molten-core material represents the end-point of severe fuel damage and is included within the scope of the Severe Fuel Damage Program.

It is planned that major portions of data from this program will be available in FY 1983, with still more available in FY 1984. The Severe Fuel Damage program plan is provided in more detail in Section 5.4.

2.4 Fission Product Release and Transport

An observation growing out of the TMI-2 investigations and subsequent studies of better estimates of accident consequences is that the radiological source term* generated by nuclear plant severe accidents may in some cases, be very conservatively characterized by the assumptions used

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*By "source term," we mean the radioactive material in the nuclear power plant that can leak out or can be released by containment failure and thus pose a hazard to the public. Although the actual composition of the source term in an accident will depend on details of the accident, it is common practice to correlate a hypothetical composition with a given accident or set of accidents. Such a hypothetical composition is then called a "source term." The source terms used currently attempt to model in crude ways processes that transport the fission products from the fuel to the containments, and processes that tend to remove

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guides or in WASH-1400. Since both siting rules and risk evaluations depend on the technical details of the radiological source term, a research program to trace the formation of the components of the radiological source term and their transport within the primary system and containment has been established. The Fission Product Release and Transport program (FPR&T) will meet its first major milestones in FY 1983.

The Radiological Source Term Issues are:

1. What is the composition and magnitude of the radiological source term corresponding to each of several dominant accident sequences?

2. What design features significantly affect the composition and magnitude of the radiological source term, and

3. To what extent should these details of the source term components be reflected in equipment qualification, plant design (shielding) siting and emergency procedure regulations?

2.5 Fuel Melt Interaction (Containment Failure Process)

In risk analysis, another major concern arising is that radioactivity, as characterized by the radiological source term, might be released early in an accident sequence as a result of containment failure. The Zion-Indian Point Study encountered major problems in determining the likelihood of early failuree and found that the provision of engineered safety features such as vented-filtered containment which might mitigate such failures must depend on the details of how a molten core would attack the vessel and subsequently the containment. Processes that serve to attenuate fission products suspended in the containment are slower to develop, however, at the same time there are processes (e.g., increased containment pressure) that may threaten long-term containment capability. The details of these processes are important to proper classification of the release category for PRA, which is used in siting and consequence considerations.

The regulatory issues that are considered in this context are:

1. Under what circumstances can processes such as hydrogen burning, steam explosions, and basemat attack, lead to containment failure, and

2. Are there modes of containment failure that affect the magnitude of the release of radioactive material?

The Fuel-Melt research portion of this program is designed to develop relevant data to resolve issues. Major milestones are planned to be met in FY 1983 and 1984.

2.6 Accident Mitigation

As expressed in the Commission's Construction Permit/Manufacturing License (CP/ML) rule, there is a need to anticipate features to mitigate the results of severe accidents that threaten the containment.

The Accident Mitigation element of this program supports the physical research that develops technical feasibility and engineering design criteria appropriate for such engineered plant features. An early result of the PRA effort will be a tentative ranking of such features to identify the worthwhile features, to rank them, and to thereby help organize and give priority to the study, as well as limit the scope of the work. It is expected that major milestones will be met by FY 1983 with respect to important classes of features such as hydrogen fire suppression and other milestones will be met by FY 1984.

Regulatory issues addressed by this element are:

1. What are the design criteria for features that can prevent or mitigate containment failure, and

2. What are the relative costs and benefits of such features?

3.0 STATE OF THE ART

This chapter summarizes the current capabilities for providing the information needed to resolve the issues listed in Chapter 2. To the maximum extent possible, recent reports that summarize the technology involved are used. In most cases the reports cited have received extensive peer review.

3.1 Probabilistic Risk Assessment Methodology

As a framework for discussing the current state of the art in PRA methodology, we might first summarize the various analysis steps which are performed in a risk evaluation:

1. Event trees are constructed for the possible accident sequences which are to be evaluated.
2. Fault trees are constructed for the system failures in the event trees.
3. The fault trees are Boolean-evaluated to obtain the minimal cuts sets of the fault trees and event trees.
4. The minimal cut sets are quantified to obtain the system failure probabilities, accident sequence probabilities, and core melt probabilities.

The above four steps yield probabilities of accidents. The following five steps are additionally required to quantify the consequences of the accidents:

5. For each event tree sequence, resulting accident variables are quantified including resulting containment pressures and temperature and core conditions.
6. For each event tree sequence, the possible containment failure modes are quantified including break size and break location.
7. For each event tree sequence, the size of the radionuclide sources released to the environment are quantified including quantification of plume characteristics.
8. The source term is transported, accounting for meteorological and topographical effects to give resulting doses.

9. Finally, taking into account population distributions, the resulting doses are translated to yield quantitative health and property effects.

3.1.1 Major Uncertainties In Severe Accident Prediction

The first four steps of the PRA, which lead to calculation of core melt probability, are the measure of severe accident occurrence. The perceived bias in these steps is one of optimism, that is, an incomplete portrayal of the various causes, and therefore, of the probability of core melt. This problem of incompleteness arises from uncertainties in severe accident prediction arising principally from the treatment of common cause failures and modeling the extent of severe accidents. In addition, the limited number of PRAs so far done limits the generic applicability of the work.

3.1.1.1 Common Cause Failures

Human Interactions

This uncertainty is related to the potential for human interaction with the plant - interactions which can initiate an accident, exacerbate an accident, and prevent or mitigate an accident. While models have been under development for several years (and used in PRAs) to predict human behavior, the gross human error exhibited during the TMI-2 accident makes it clear that these predictions (and the underlying human behavior) are poorly understood. Because such human interactions are both poorly understood and have the potential for defeating the many installed systems for coping with accidents, it is believed that human interactions are a major uncertainty in severe accident prediction.

System Interactions

This uncertainty relates to the potential in nuclear plants for the failure of one system or component to result in the unforeseen failure of other equipment. The use of such techniques as "event tree" analyses in PRAs will result in the identification of many important system interactions; however, experiences in operating plants (e.g., the Rancho Seco "light bulb" event) make it clear that the present state of knowledge of such occurrences is poor. Because these potential interactions are poorly understood and have demonstrated the ability to defeat many installed systems, the issue of system interactions is considered to be a major uncertainty in the present ability to predict the likelihood of severe accidents.

External Events

This uncertainty relates to the potential for external events such as earthquakes, floods, etc. to cause severe accidents in LWRs. Again, such events have been modeled in PRAs with varying degrees of sophistication; however, the likelihood of such events to result in the gross common-cause failure of many levels of "redundant, independent" equipment is very poorly understood. (Because plant internal fires exhibit similar causal characteristics, they should also be included here, although they are not "external" events.) These events thus have the potential for causing gross common-cause failures within a plant; because they are also poorly understood, it is believed that they constitute a major uncertainty in severe accident prediction.

3.1.1.2 Extent of Severe Accidents

Virtually all risk assessments available now use system success or failure criteria that do not distinguish the extent of core damage. The systems work and there is no core damage, or the systems fail and the core melts completely. By this logic, the TMI-2 accident was not predictable. Considering the prolonged endurance of degraded cooling which the TMI-2 core suffered without undergoing full scale core melt, some have speculated that many, if not most, severe accident sequences can be terminated short of full scale core melt. A more realistic delineation of severe accident sequences between severe damage sequences and core melt sequences is needed to evaluate the relative threats of different accident sequences and the relative need for or worth of specific design features. For example, if a TMI-2 sequence of severe damage without full melt is typical of severe accident sequences, then there might well be justification for design features which cope with severe accidents by cooling such a severely damaged (but not fully molten) core. On the other hand, if almost all sequences can be expected to go to full scale core melt, then those same systems might be far less useful. Evaluation of risk or the risk reduction effectiveness of specific designs and changes requires careful accident sequence delineation. Because present capabilities to probabilistically differentiate between severe damage and gross core melt sequences are very poor, and because this differentiation can potentially result in radically different risk predictions from these accidents, this issue is considered to be a major undertainty in severe accident prediction.

3.1.1.3 Generic Applicability of PRA Predictive Results

This uncertainty relates to the applicability of PRA results for specific plants to larger, more generic, groups of plants. In this program, it is planned that the principal PRAs to be used will be the Reactor Safety Study (RSS) and the RSS Methodology Applications Program (RSSMAP). These PRAs encompass a wide diversity of NSSS and containment designs; however, these studies were of necessity based on specific plants of each type. The degree to which these plant-specific predictions of severe accident likelihood are applicable to other generally similar plants is not well known. Because this extent of typicality is not well know and has the potential to significantly alter the likelihood of severe accidents within these general plant classes, this issue is considered to be a major uncertainty in generic severe accident prediction.

3.1.2 Major Uncertainties in Severe Accident Phenomenology

Steps 5 through 9 listed at the beginning of this PRA section cover the consequences analysis portion of PRA. Here the problems are less those of completeness than of adequately modeling the severe accident phenomenology, describing the physical processes of core melt and consequent releases. The perceived bias in these steps is one of pessimism, that is, an exaggeration of the releases and consequences. This exaggeration arises principally from conscious conservatism in discounting attenuation factors where data are sparse. The discussion immediately following covers the uncertainties in severe accident phenomenology.

3.2 Severe Accident Sequence Analysis (Accident Management Guidelines

This effort uses a combination of risk assessment methods, best estimates codes and human factors methods to study the interrelationships between the man and the machine to provide the operator with guidelines for controlling the plant under accident conditions. Accident sequences that contribute significant risk in probabilistic risk assessments are studied in detail using state-of-the-art thermal hydraulics codes in the Severe Accident Sequence Analysis program. Codes such TRAC and RELAP provide a relatively precise evaluation of transients up to the start of significant core damage. Beyond that, the limitations of the phenomenological data base on fuel damage and relocation in severe accidents make credible modeling extremely difficult.

The MARCH code limitations in this and other areas are indicated in the preceding section. Details of accident sequences after the core materials penetrate the primary pressure vessel can be treated using the methods described in Section 5.7. Principal codes used are CORCON to describe the fuel-concrete reaction, and CONTAIN to predict the loads on the containment. The data base to improve this part of the accident analysis depends on the more basic work described in Sections 5.5, 5.6, 5.8 and 5.9.

Current applications of SASA provide insight into accident management guidelines so long as limitations in modeling fuel damage and relocation are recognized. These studies have analyzed small-break LOCAs, large-break LOCAs, interfacing systems LOCAs, loss of AC power, and loss of feedwater transients. The studies have evaluated numerous accident strategies for each of the sequences.

The studies have also directly supported the resolution of unresolved safety issues (USI) and the evaluation of Abnormal Transients Operator Guidelines (ATOGs).

A SASA calculation log has been established. This log will be expanded in the future to eventually become a handbook of accident signatures that can be used to improve simulator and other opreatering training programs.

The completion of the sequence analysis to date has developed a continually expanding data base of great value to other programs. It is being used to develop operator action event trees that can be used to define appropriate operator action for a variety of scenarios. This data base can also be used to evaluate the accuracy of the PRA methodology which will plan a role in the future processs of plant licensing. SASA possesses a unique capability and position in developing this data base.

3.3 Behavior of Damaged Fuel (Severe Fuel Damage)

A current assessment of the state of the art is focused in NUREG-0840, "Report of NRC Fuel Testing Task Force." This report received extensive industry and international peer review. The following is a summary of the state of the art from NUREG-0840.

3.3.1 Damage States

The knowledge of the physical and chemical state of a severely damaged core is the major prerequisite for determining the ultimate coolability of the core.

Since those scenarios which lead to core melt represent the greatest contributors to risk, the determination of core coolability at any time during a severe accident sequence will ultimately govern the risk to the public. Section 2.6 of the Executive Summary of the Reactor Safety Study (WASH-1400) states that: "The only way that potentially large amounts of radioactivity could be released is by melting the fuel in the reactor core." It goes on to state that: "Thus, for a potential accidental release of radioactivity to the environment to occur, there must be a series of sequential failures that would cause the fuel to overheat and release it radioactivity." The methodology used in WASH-1400 was based on event tree and fault tree analysis that determined the probability of failure of certain systems. After failure no allowance is given for their ultimate return to service. However, during slow accident sequences, such as that which occurred at TMI-2, recovery of such systems is possible by proper operator intervention. Such actions resulted at TMI-2 in the prevention of a massive core melt and, very low risk to the public. Therefore, when one considers the risk to the public for a given series of equipment failures, one must also consider the effect of the return to service of those systems and the proper mitigating actions of the reactor operators. In order to compute such effects on risk calculations one must know the chemical/physical/thermal state of the fuel during the sequence of events.

The current state of knowledge on severely damaged fuel is based on past experiments and analyses conducted world wide to provide a basis for understanding and evaluating core melt behavior and for risk assessment studies. The only work that was focused on the early stages of severe fuel damage (as opposed to steam explosions, core melt behavior, etc.) was performed at KfK in the Federal Republic of Germany by S. Hagen from 1976 through 1978. These experiments showed clearly that the damage state of electrically heated fuel rod simulators heated in steam to temperatures in excess of 3600°F (2255°K) will depend primarily on four major parameters, namely: (1) the final temperature reached, T_{max}; (2) the heating rate, dT/dt; (3) the rate of cooling, dQ/dt; and (4) the pressure difference between the interior of the rod and the reactor coolant, ΔP. The current knowledge can be summarized in terms of these parameters by defining "damage regimes" in terms of T_{max} and expressing the effects of the other three parameters on the phenomena that occur. The following paragraphs discuss each regime in detail by focusing on (a) the physical/chemical phenomena involved and (b) the safety issues to be addressed (if any). Figure 3-1 gives a simplified schematic illustration of the damage regimes discussed. Finally, the current state of knowledge and needs of debris coolability are discussed at the end of Regime V.

<u>Damage Regime I (T_{max} < 1700°F (1200°K); ΔP negative 100-1200 psi; any dT/dt)</u>

a. <u>Physical/Chemical Phenomena</u> - Cladding buckling, collapse, and "waisting" of the fuel stack. These phenomena were studies extensively for the NRC LOFT program and is well correlated with data. Very little additional data are needed and modeling of the effect can proceed with confidence.

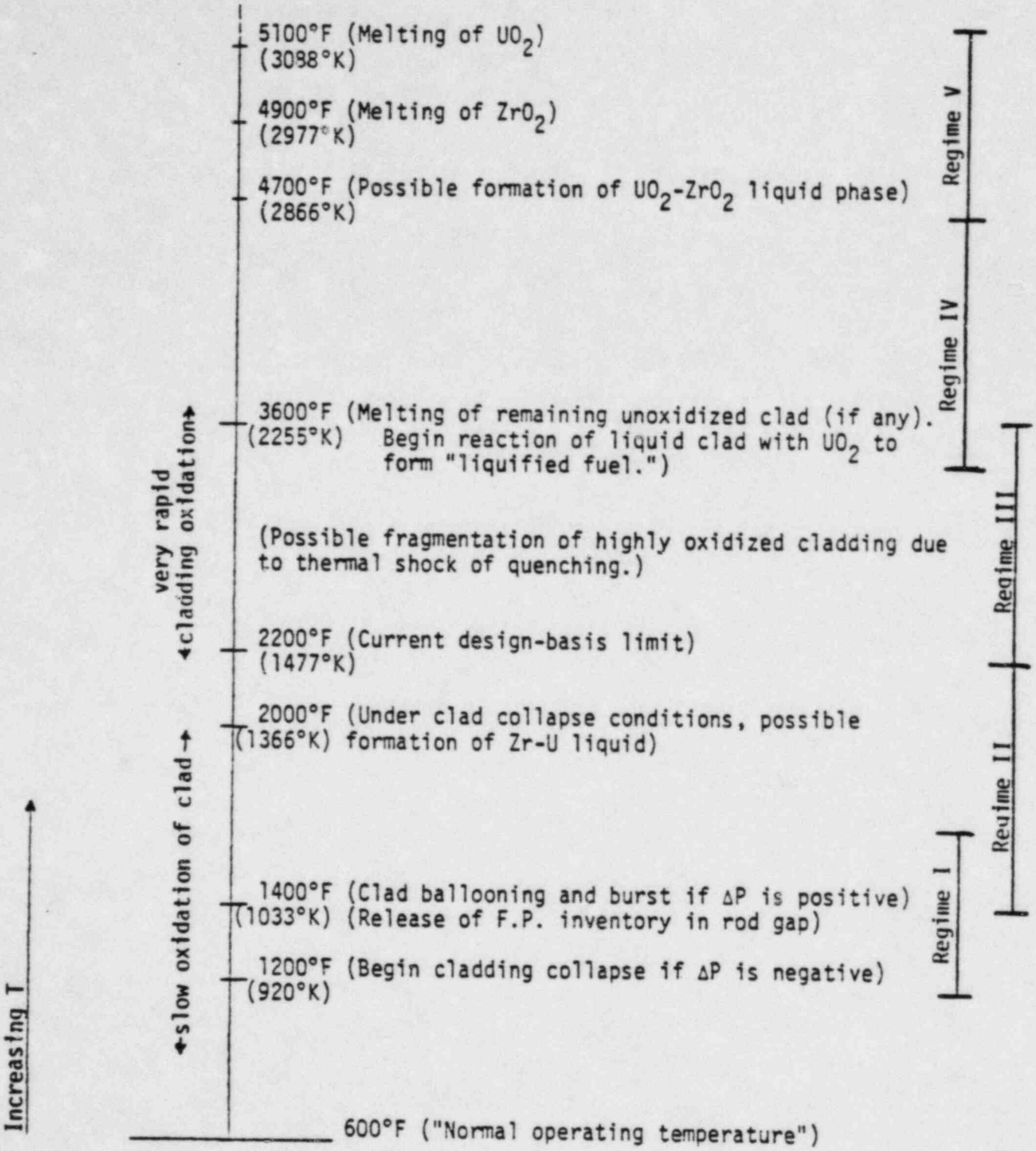

FIGURE 3-1

Schematic (not to scale) A-156

b. Safety Issues - Possible release of fission products (in-vessel) due to pellet/cladding interaction leading to stress-rupture or stress-corrosion cracking failures. Reactor behavior in this regime is covered by current licensing practice for DBA and is not considered to be "severe" fuel damage.

Damage Regime II (T_{max} <2200°F (1475°K): ΔP positive; and dT/dt)

a. Physical/Chemical Phenomena - Cladding ballooning and burst. This phenomenon has undergone extensive study in the last 10 years. Plentiful data are available and preliminary models have been developed. Final resolution of the effect on core coolability awaits completion of the NRU ballooning experiments in FY 1982 and future tests at KfK in the FRG.

b. Safety Issues - Release of the rod gap fission product inventory to the reactor coolant. Research at Oak Ridge National Laboratory (ORNL) has shown this to be a minor issue at temperatures below 2200°F (1477°K). The ballooning process may affect the coolability of the core because of the partial closure of coolant channels. If such blockage is near 100%, partial localized melting may occur. Current evidence indicates that the latter possibility is very unlikely. In any case, the programs mentioned will fully investigate the possibility. As is the case for Regime I, damage in this area is covered under current regulatory practice and is not considered in this report to be "severe."

Damage Regime III (T_{max} <3400°F (2140°K): any P; any dT/dt)

a. Physical/Chemical Phenomena - Very rapid oxidation of the Zircaloy cladding. This results in severely embrittled cladding that will fragment on reflood quenching. The embrittlement and fragmentation of highly oxidized Zircaloy has been studie extensively for the NRC at Argonne National Laboratory (ANL). The limits on the maximum time-at-temperature which will result in no fragmentation due to thermal shock from reflooding have been determined and can be used in our current models. No additional work is needed or is planned in this area. However, the oxidation kinetics of Zircaloy are not well known above 1800°K (2800°F), and high-burnup fuel may experience considerable swelling due to fission product release.

b. Safety Issues - If the accident is terminated below approximately 3400°F, the issues becomes related to the coolability of a core containing fragmented pieces of oxidized and embrittled Zircaloy-clad fuel rods. Another issue is the extent and amount of fission product release at these higher temperatures. These questions are being addressed by current experiments at ORNL as well as by the Power Burst Facility (PBF) programs.

Damage Regime IV (T_{max} <4700°F (2870°K); any ΔP; any dT/dt)

a. Physical/Chemical Phenomena - Melting of the remaining partially oxidized cladding; reaction of liquid cladding with solid UO_2 to form "liquefied fuel"; flow and refreezing of liquefied fuel to produce "candling" type (cohesive) damage and blockage; continued oxidation of liquefied fuel during flow and after refreezing. The only available data in this regime are those of Hagen at KfK where rod simulators containing a core rod of tungsten (as a heater) surrounded by annular rings of UO_2 were used. More prototypical tests using rods of standard design that are volumetrically heated by either fission or decay heat are required so that the damage and debris formation scenarios for representative fuel rods can be studied and modeled. The effect of high burnup will also be important in this regime and in Regime V below.

b. Safety Issues - The major safety issues for this regime are core coolability (i.e., can the accident be stopped?) and fission product and hydrogen release from very hot solid fuel rods, liquefied fuel, and fragmented fuel. The PBF, ACRR, and NRU programs are designed to answer these questions by the performance of core debris formation and coolability studies and the monitoring of fission product and hydrogen releases during the experiments.

Damage Regime V (T_{max} 4700°F (2870°K)

a. Physical/Chemical Phenomena - Melting of remaining UO_2 and ZrO_2; growth of the melt; motion of the melt; foaming of molten UO_2 due to fission product release; interaction of the melt with the pressure vessel sides and lower core support plate; interaction of the melt with water in the vessel; hydrogen and fission product release; explosive and non-explosive steam generation. Except for steam explosion studies currently being performed at Sandia, very little information is available on the phenomena mentioned above. Some information on melt motion may be obtainable from fast reactor experiments and models, but new information is definitely required for LWRs in this area.

3.3.2 Damaged Fuel Coolability

A primary goal of the SFD program is to determine, for each state point of severe fuel damage, whether or not the core debris is coolable and what the coolant requirements are to achieve coolability. Debris is said to be coolable if a geometry and temperature distribution have been achieved that are stable in time. The coolability approach used in the SFD program is to determine the damage state points for which the core debris is coolable by slow reflood (i.e., stagnant pool) and, for those damage state points outside this space, to determine the coolant flow velocity and pressure necessary to achieve coolability.

The most important, most easily defined and most easily measured coolability limit is the dryout heat flux limit at which liquid coolant does not reach some regions of the debris. It has been shown in fast-reactor safety experiments in the ACRR test reactor with sodium-cooled debris beds that stable temperature distributions and geometries are possible at decay-heat power levels with local dry-out in part of the debris bed. However, little is known about the available coolability margins and debris behavior between the point of local dry out and the progression into core melt. Therefore, the well-defined and relatively easy-to-measure dryout limit is the best criterion of coolability to use in reactor safety assessment and research.

A substantial data base and relatively sophisticated analytical models of dry-out coolability limits as a function of mean particle size and bed depth have been developed in the fast-reactor safety research program. The experiments have included several coolants, i.e., sodium, water, and organics, and several methods of heating, including fission heating of simulated debris in the ACRR test reactor to simulate fission product decay heating. Lipinski at Sandia has developed a relatively sophisticated first-principle model for the dryout limit of a packed unstratified debris bed that agrees well with the world data base for all the liquids tested. This model includes capillary forces and both laminar and turbulant vapor flow. The ACRR experiments with sodium-cooled debris beds have shown that the formation of vapor channels in the bed that can occur at high subcooling can increase the bed dryout limit by about a factor of five. These experiments have also shown that bed stratification with an increase in mean particule size with distance below the bed surface can decrease the bed dryout limit by a factor of five. Verified models of these phenomena and of the onset of channeling in debris beds do not yet exist.

The consensus of experts in this area is that there is a clear need to test and confirm the applicability of current coolability models to important LWR accident conditions such as scaling to high pressure the mixtures containing different sizes and shapes of debris covering a range of damaged fuel configurations, including non-ideal shapes, stratification, and deep configuration.

3.4 Fission Product Release and Transport (Radiological Source Term)

In response to issues developed by members of the technical community as a result of analyses of the TMI-2 accident, the Nuclear Regulatory Commission requested and the staff prepared NUREG-0772, "Technical Bases for Estimating Fission Product Behavior During LWR Accidents." This comprehensive report is the best current summary of capabilities in the technical area. The report was reviewed by internationally known experts and representatives of industry and DOE as well as NRC staff and contractors. The following is an excert summary of the state of the art from this report.

1. The current data base suggests that cesium iodide will be the expected predominant iodine chemical form under most postulated light water reactor accident conditions. The formation of some more volatile iodine species (e.g., elemental iodine and organic iodines), however, cannot be precluded under certain accident conditions.

2. The assumed form of iodine (either cesium iodide or elemental iodine) was not predicted to have a major influence on the estimated magnitude of iodine attenuation in the containment for severe accident sequences with early containment failure in which there is little time for natural fission

product retention mechanisms to be effective. However, the assumed chemical form of iodine can influence the predicted attenuation within the reactor coolant system, where, in general, the attenuation factor will be greater for cesium iodide than for elemental iodine (i.e., less iodine will escape into the containment).

3. A number of accident sequences were examined in this report including several core melt sequences which had been found to be the most important contributors to risk in the Reactor Safety Study (RSS). Reevaluation of fission product release from the fuel indicates that the RSS may have underpredicted the release of certain important radionuclide species during these core melt events. Mechanistic analyses of fission product transport in the containment atmosphere were in reasonable agreement with the empirically based analyses in the RSS. Predictions of the retention of radioactive material within the reactor coolant system (which was not accounted for in the RSS for most accident sequences) range from very little to substantial retention for specific accident sequences involving a water bounded reactor coolant system (e.g., TMI). In addition, for certain transient initiated core melt sequences where steam flow rates through the reactor coolant system are low and aerosol generation is high, attenuation of fission products within the reactor coolant system could be substantial as a result of agglomeration and fallout of aerosols. Consequently, for certain accident sequences considered in the RSS the release of radionuclides to the environment may have been significantly overpredicted. However, for other accident sequences (such as large or

medium size pipe break accidents) the estimated releases in this report are in approximate agreement with the RSS estimates.

There are very large uncertainties in the release rates for specific radionuclide species. Knowledge of the chemical form of the released radionuclides is however quite limited.

The transport behavior of fission products within the reactor coolant system is subject to large uncertainties resulting from limitations in the ability to predict severe accident phenomena, thermal hydraulic conditions, and the physical and chemical forms of the fission products.

In contrast, the ability to predict the behavior of fission products within the containment structure after release from the primary system is comparatively good for large volume PWR containments. Less well known is the fission product behavior within pressure suppression containments such as in boiling water reactors and in pressurized water reactor ice condenser plants where the potential attenuation of fission products within the pressure suppression pool and ice beds is subject to large uncertainties. One of the largest uncertainties associated with predicting the amount of radionuclides released to the environment during the most severe accidents (i.e., core melt accidents with containment failure) result from limitations in the ability to predict the timing, mode, and location of containment failure.

The extent to which fission product release to the environment may have been overestimated (or underestimated) in previous studies is difficult to quantify since the range of uncertainty associated with these predictions is very large as a result of the identified limitations in the data base and the early state of development and verification of the predictive methodology.

3.5 Fuel-Melt Interaction (Containment Failure Processes)

The state of the art in studying containment failure processes is defined by two recent studies of severe accidents in the Zion Aand Indian Point plants. The pertinent reports are NUREG/CR-1409,-1410, and-1411, "Report of the Zion/Indian Point Study," and in NUREG-0850, "Preliminary Assessment of the Core Melt Accidents at the Zion and Indian Point Power Plants and Strategies for Mitigating Their Effects." Both reports have received extensive peer review within the NRC staff and from contractors. Chapter Six of NUREG/CR-1410 is a summary of the current status of modeling meltdown progression and the resulting threat of the containment. These points are abstracted from that portion of the document cited, and other pertinent comments are interpolated as appropriate. In addition, the Reactor Safety Study and its successor documents, particularly the studies of different containment types have been a major source of information to define the problems and outline the scope of work.

3.5.1 Failure by Hydrogen Burning

The accident at Three Mile Island demonstrated the possibility that hydrogen can be generated in larger quantities than previously considered in NRC regulations. That is, large amounts of hydrogen can be generated in fuel damage accidents, as well as in core-melt accidents. In recent licensing actions, NRC has required that licensees install hydrogen control systems in certain small types of containment structures (i.e., ice condenser and BWR pressure suppression models Mark I and Mark II). Rulemaking is also under way to establish new hydrogen control requirements for all construction permit and operating license applicants regardless of containment types.

This failure mode is produced by the pressure loads generated on containment by the combustion of accumulated hydrogen in the containment or possibly by the generation of missiles from the detonation of packets of hydrogen. Also the burning of hydrogen could affect the operation of safety related equipment necessary for the safe isolation and shutdown of the plant. This hydrogen would be generated from the reaction of hot steam with zirconium or steel. Additional hydrogen can also be generated later in the accident by the reaction of molten-core material with concrete. Other secondary sources of hydrogen arise from the radiolytic decomposition of water and the corrosion of galvanized materials in the containment and from chemical reactions of sprays with aluminum and other organic/inorganic coatings.

Measures to control and manage accidents involving hydrogen depend on the rate and quantity of the hydrogen released; the distribution of hydrogen, air, and steam, the temperature and pressure in the containment building when the combustion occurs; and the location of the hydrogen release. As noted below, there are uncertainties in a number of these areas.

1. Source of hydrogen: Except in a steam-starved situation, the kinetics for the zirconium-steam reaction is thought to be modelled well by the Baker-Just or Cathcart-Powel models. The steam-steel reaction has a much larger uncertainty. The importance of this is dependent upon the amount of steel involved when the core slumps and when the vessel fails, releasing molten core and steel to react with water in the reactor cavity. There is a large uncertainty in the relative amounts of hydrogen generated versus steam when the core slumps, depending on the amount of core melt and oxidation prior to slump. Rapid release rates of hydrogen (100-200 lb/min) could cause problems for proposed hydrogen control systems. Additionally the generation rate of hydrogen and possibly carbon monoxide from molten core/concrete interactions has not been verified, although the rates are calculated to be on the order of 7-10 lbs/min for extended periods. Any substantial release of carbon monoxide will generate higher pressures than a corresponding burn of pure hydrogen.

2. Hydrogen release, transport, and mixing: Hydrogen can be released through a relief valve, small- or large-breakS or through the high point vent that is now required to be installed on LWRs. The rate of release is dependent on the driving force (system pressure) and the size of the break. At the onset of vessel failure, the remainder of the hydrogen formed but not previously leaked will be released. There is an uncertainty in the release rate and also in the relative amount of steam accompanying

the hydrogen release. Once released to the containment, the hydrogen mixes throughout the containment in various compartments and areas, depend on a number of factors (e.g., pressure differences, temperature gradients). Until recently, only limited work had been done on developing analytical models to calculate the transport and mixing of hydrogen in containment, and there is a need for some experimental and analytical work in this area.

3. <u>Combustion of hydrogen</u>: Different igniters or detonators lead to different deflgaration and detonation limits. In large volumes with obstructions, and particularly in pipes or ducts, deflagrations may accelerate to detonations, even in mixtures outside the Shapiro-Moffette detonation limits. There is some evidence to indicate that detonations cannot develop in atmospheres containing less than 13 percent hydrogen. A practical lean limit for the ignition of hydrogen with glowplug igniters appear to be somewhat higher than the 4 percent usually determined in laboratory ignition tests using sparks or flames to ignite tubes of gas in upward propagation. Fenwal Laboratories has achieved ignition of mixtures containing 5 percent hydrogen using glowplug igniters. In experiments at Lawrence Livermore Laboratory using a different vessel, attempts to ignite 6 percent mixtures of hydrogen in air failed, but ignition was achieved in 8 percent mixtures.

If hydrogen control by a series of small burns is contemplated, the question of interest is somewhat different from the determination of the minimum combustible limits. Instead, there is a need to know the highest concentration that might not burn under the conditions of the specific ignition system. This level determines the energy loads on the structure. The experiments indicate that the highest concentration that might not burn is about 8 percent (by volume). The use of an average concentration of 10 percent (by volume) for accident computations should provide some margin for variations in hydrogen concentrations throughout the containment building.

Not all compositions within the detonation limits necessarily detonate when ignited. Factors that enhance the probability of detonations are (1) shock waves accompanying ignition, (2) turbulence, and (3) large volumes, especially those with obstructions because they promote turbulence. A recent study of the detonation of hydrogen concludes that the absolute detonation limits (with large explosive detonators) are 13-70 percent by volume in air, rather than the 18-56 percent by volume indicated by Shapiro and Moffette. Uncertainties in the H_2/air/steam deflagration limits, questions on deflagration-to-detonation transitions, and questions on autoignition are being addressed in the NRC program on hydrogen behavior at SANDIA.

4. Efficacy of Mitigation Systems: In response to the Commission requirements on the control of hydrogen in LWR accidents various mitigation systems and schemes have been proposed. The efficacy of this system for various accident sequences, particularly for specific plant designs has not been totally demonstrated. To remove this uncertainty and to investigate

potentially better systems, there is work being sponsored by RES on mitigatio of hydrogen effects at SANDIA.

5. Equipment Survival: A hydrogen burn can potentially generate temperatures and pressures exceeding the qualification limits used to test safety grade equipment. These higher temperatures and pressures could lead to the failure of a component important in the safe isolation and shutdown of the plant. In order to remove the uncertainty and to develop improved testing methods for safety grade equipment, and in order to meet the test posed by a hydrogen burn, an analytical and experimental program on equipment survival was recently initiated. It is anticipated that questions in this area should be answered within the next two years.

3.5.2 Failure by Steam-Spike Overpressurization

This failure mode is induced by the rapid generation of steam when a mass of molten fuel drops into a cavity filled with water or when, as a result of depressurization from vessel breach, the accumulators come on and dump a large mass of water on the very hot and molten fuel. The current state of the art includes a high degree of uncertainty as to how molten fuel and structural material interaction within the pressure vessel actually cause a breach; several options are available within MARCH, and these with their attendant uncertainties are described in Chapter 6 of NUREG/CR-1410. The conclusion in that report is that two types of failure are judged most probable: (1) a catastrophic failure of the central portion of the lower head after about thirty minutes of plastic deformation, or (2) a rapid splitting in the form of a small crack at the periphery of the lower hemisphere, depending on whether or not the pressure vessel had been depressurized. Thus, there is a wide range of possible vessel failures, the most probable being at the relative extremes. The mode of failure dictates the extent to which one has to consider catastrophic mixing of a large mass of molten fuel with water in the cavity, or, for that matter, within the vessel during the initial stages of meltdown. In any event, steam will be generated, perhaps at a rapid enough rate to be called a "steam explosion." Such steam, explosion or not, constitutes a significant source of overpressure. In NUREG-0850 the estimate is made (for the Zion and Indian Point Plants) that overpressures during this event are large, about 100 psia, with uncertainties of about 20 percent, at least. A chief contributor to the uncertainty is the nature of possible hydrogen burns during this period of the transient. Estimates are that the mixture of steam and air is sufficient to suppress hydrogen burning and hence to suppress a potential source of greater overpressure. On the other hand, experiments with pouring molten fuel simulants into water show a large hydrogen burn coincident with the steam generation, so dynamic effects may be important.

3.5.3 Failure by Steam Explosion

The analysis in NUREG-0850 ascribes a low likelihood to failure by missiles generated by steam explosions, and this is presaged by work reported in NUREG/CR-1411, which predicts that if steam explosion causes a failure, it will be at the lower hemisphere and hence unlikely to cause significant missile damage. Continued caution is indicated in this area, however, because of the difficulty in extrapolating from small-scale (few kilograms) to large-scale (several thousand kilograms) mixing experiments with confidence prior to the existence of a good model of the explosion detonation process. Such models should be available in the near future.

3.5.4 Failure by Slow Overpressurization

A simple heat balance, even taking into account the slow transfer of heat through the containment, leads one to predict that unless long-term cooling is restored, the containment will eventually fail by slow overpressurization. The time of failure depends on the containment failure characteristics, the rate of gas generation during the core-concrete-coolant interaction, and the potential for hydrogen burning during the period of slow overpressurization. In the cases considered in NUREG-0850, the key uncertainty for the case of a flooded cavity was the failure characteristics of the containment that led to an estimated uncertainty of about 5 hours in the failure time. In the case of a dry cavity, the principal uncertainty is the rate of gas formation, but the failure time is in any case predicted to be late. For smaller containments, or containments of lower pressure capabilities, other sources of uncertainty might also become important.

3.5.5 Failure by Basemat Melt-through

There is some contention whether this mode of failure is inevitable, as assumed in WASH-1400, or indeed in some circumstances whether it will occur at all. In any event, the failure time is quite late. The conclusion in the Kemeny Report was that such failure would not occur, and in NUREG-0850 the conclusion is that, with a flooded cavity, basemat penetration may not occur. For convervatism, a time of failure of three days was assumed for this mode. Two major sources of uncertainty are the extent to which the interacting mass of core and concrete can be cooled by the overlying water and the rate at which hot but solidified fuel melts through concrete. In some experiments, it has been observed that water was held away from the interacting masses by a crust of material, so that establishing a coolable debris bed prior to significant attack on the concrete basemat may be prerequisite for preventing this mode of failure.

3.6 Accident Mitigation

The state of the art in mitigation systems is discussed in the Zion/Indian Point studies referenced above. Other studies in more detail have focused on one or more specific systems. Core retention systems are reviewed in "A Review of Core Retention Concepts to Light Water Reactor Containments," NUREG/CR-2155, where it is concluded that core retention systems can only reduce risk significantly if above-grade containment ruptures are prevented by another system, such as a filtered vent. However, analyses and proposed designs that show the potential effectiveness of a retainer do not adequately address all the possible thermal/hydraulic and materials uncertainties associated with the problem. Even though basemat melt-through may not be a significant source of risk, a core retention system could be designed to limit the gas generation by fuel melt interaction, thereby lessening the load on the above-grade containment. This benefit was not explored in depth in NUREG/CR-2155. Experimental work is necessary to provide information on how well the proposed concepts actually work. Analyses and experiments must be integrated to investigate the following detailed heat transfer and materials concerns before an effective core retainer can be identified:

- Distribution of core debris in the reactor cavity,
- Differentiation of melt into immiscible metallic and oxidic phases,
- Complex nature of the oxidic phase,
- Correct partitioning of heat sources,
- Radiative heat transfer to upper containment,
- Scaling of laboratory experiments to full-core situations,
- Chemical reactions,
- Formation of eutectics,
- Effects of water,
- Actual mechanisms of melt penetration,
- Formation of gases and liberation of fission products.

Vented filtered containment (VFC) was one of nine alternatives considered in "A Value-Impact Assessment of Alternative Containment Concepts" NUREG/CR-0165. VFC was judged to offer the greatest potential for reducing public risk for the least impact.

Core melt mitigation systems for the ice condenser plant were studied at INEL and reported in "Phase-2 Status Report - Core Melt Mitigation System Design for an Ice Condenser Plant" (to be published). This study concludes that hydrogen presents the dominant threat to containment, that post-accident inerting offers an attractive way to deal with this threat and presents several conceptual designs to achieve this goal.

In all of the studies done to date the unifying fact found in them all is the recognition that more research and detailed designs are needed to support any judgment whether or not to require the addition of further mitigation systems.

4. PROGRAM LOGIC, SCHEDULE, AND INTERFACES

The logic, overall schedule including major milestones and key interfaces for the Severe Accident Research Program are shown in Figure 4-1 and Table 4-1. The program structure is derived from six key Decision Units or Subelements in the RES Long-Range Research Plan (NUREG-0740), namely Reliability and Risk Assessment, Severe Accident Sequence Analysis, Behavior of Damaged Fuel, Fission Product Release and Transport, Fuel Melt, and Accident Mitigation. These Decision Units or Subelements comprise the 14 program elements that yield the three major product catagories shown in Figure 1. These major product categories provide the following types of information and technical bases for policy decision and regulatory products (regulations, reg-guides, standards, and standard review plan revisions) for severe accidents in nuclear power plants:

1. Data and guidelines for refinements to plant systems design and operating practices,
2. Verified methodology for accident loadings and system responses, and
3. Information and methodology for decision on potential risk-reduction add-ons.

Figure 4-1 shows the summary schedule in terms of major milestones (listed in Table 4-1) for each of the 14 program elements. The logic for the program is indicated by the vertical tie lines between program elements in

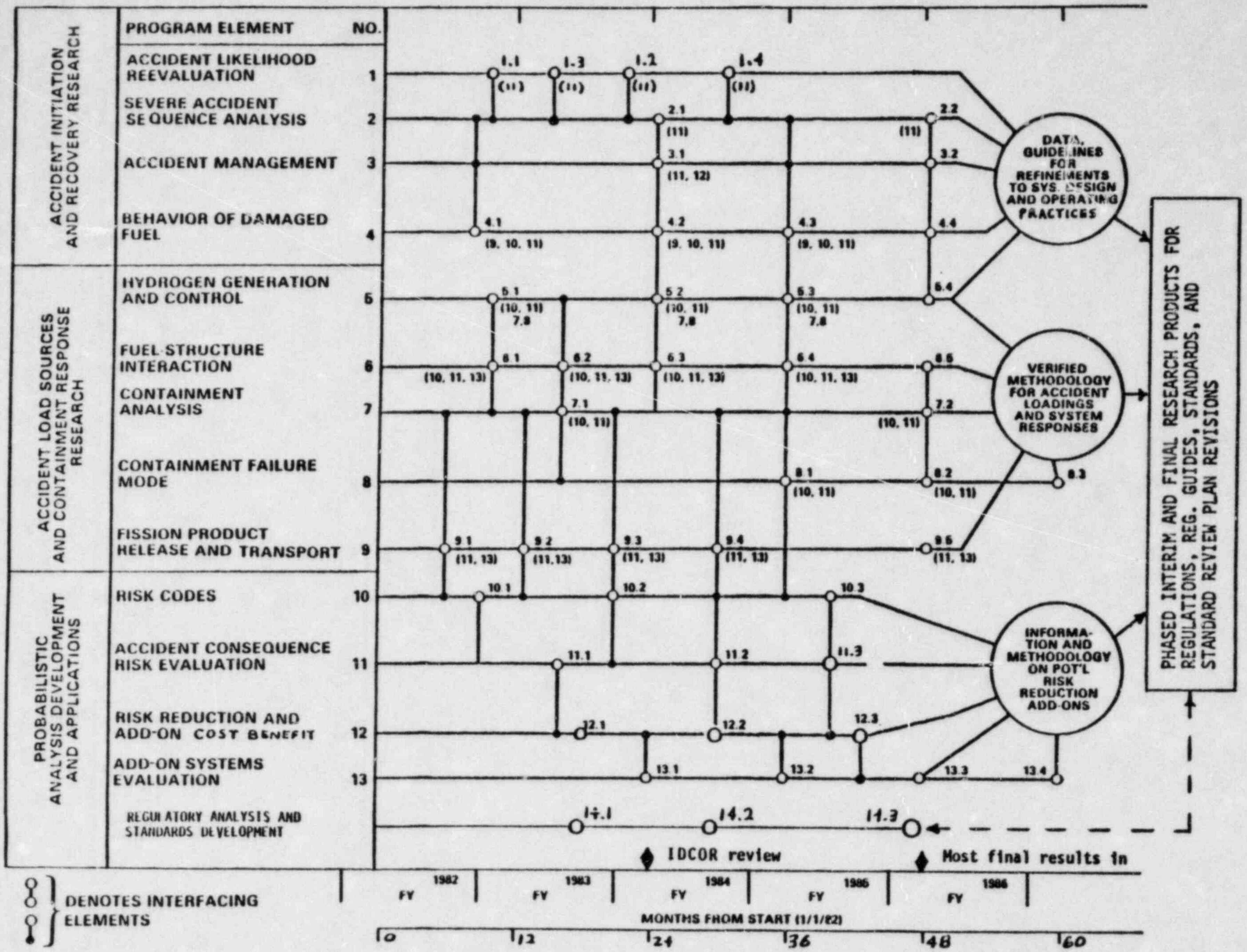

Figure 1. Severe Accident Research Program Plan

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Table 1

SEVERE ACCIDENT RESEARCH PROGRAM
MAJOR MILESTONES

MAJOR MILESTONES	DATE START = 1/1/82
1. Accident Likelihood Reevaluation	**Mo. from Start**
1.1 RSSMAP/IREP Final Report	9
1.2 Precursor Studies ... Phase 1	9
... Phase 2	9
... Final	21
1.3 Station Blackout Studies Final Report	9
USI Resolution	15
1.4 Accident Sequence Reevaluation ... Phase 1	6
... Phase 2	18
... Phase 3	30
2. Severe Accident Sequence Analysis	
2.1 Assessment of Operator Guidelines	24
2.2 Management Strategies for Severe Accidents	48
3. Accident Management	
3.1 Operating Procedure Guidelines for Recovery from Core Damage Event	24
3.2 Refinements to System Design and Operating Procedures	48
4. Behavior of Damaged Fuel	
4.1 SCDAP MOD 0 Available, First PBF Phase I Test and First ACRR Coolability Experiment	8
4.2 Complete Phase I PBF Tests and Initial ACRR Separate Effects for Damage State Coolability Criteria with SCDAP MOD 1. TMI-2 RPV Head-lift	24
4.3 SCDAP MOD 2 with Improved ACRR Phenomenological Model. Initial NRU Full-length Verification. Whole-core Analysis Available	36
4.4 Phase II PBF, ACRR, and NRU results for SCDAP Coolability and TMI-2 Data for SCDAP/Whole Core Benchmark	48
5. Hydrogen Generation and Control	
5.1 Improved Combustion Models, Preliminary Thermal Models for Equipment Survivability, Analysis of H_2 Control for Two Containment Types	12

Table 1 (Cont.)

MAJOR MILESTONES		Mo. from Start
5.2	Assessment of Alternative Control Methods, Improved Equipment Response Model and Improved Transport Code	24
5.3	Preliminary Assessment Flame Acceleration, other Plant-Specific Analysis	36
5.4	Large-Scale Proof Tests	48
6.	Fuel-Structure Interaction	
6.1	Large-Scale Fuel-Melt Interaction Transient tests	12
6.2	CORCON MOD 2	18
6.3	Large-Scale Melt Interaction Sustained Tests, Retrofit Retention Concepts, Melt/Concrete Aerosol Source	24
6.4	CORCON Verification Tests, Castable Concrete Tests	36
6.5	Fuel Debris-Coolant-Concrete Interaction Tests	48
7.	Containment Analysis	
7.1	Improved Version of CONTAIN	18
7.2	CONTAIN Verification	48
8.	Containment Failure Mode	
8.1	Static Pressure Loads - Steel and Concrete Containment	36
8.2	Static Loads Reinforced Concrete, Dynamic Loads - Steel and Concrete	48
8.3	Dynamic Loads - Reinforced Concrete	60
9.	Fission Product Release and Transport	
9.1	TRAP-MELT MOD 2 for RSC Transport	6
9.2	NUREG-0772 Follow-On - Reassessment of Source Term	15
9.3	NSPP Aerosol-Steam Tests, Release Rates Irradiated-Fuel to 2000½C and Melt Aerosols Source, Chemical Species - Vapor and Aqueous, TRAP-MELT MOD 3	21
9.4	ESF Severe Accident Performance, TRAP-MELT Verification for Volatile F.P. Transport	30

Table 1 (Cont.)

MAJOR MILESTONES	Mo. from Start
9.5 TRAP-MELT Aerosol Transport, Release Rates Irradiated and Simulated fuel to 2800°C	48
10. Risk Codes	
10.1 MARCH-2/MATADOR	9
10.2 Preliminary Version of MELCOR	21
10.3 Final Version of MELCO	40
11. Accident Consequence and Risk Evaluation	
11.1 Evaluations with MARCH-2/MATADOR	15
11.2 Evaluations with Preliminary Version MELCOR	27
11.3 Evaluations with Final Version MELCOR	40
12. Risk Reduction and Add-On Cost Benefit	
12.1 Integrated Risk-Cost (MARCH-2 Basis)	18
12.2 Integrated Risk-Cost (Early MELCOR)	30
12.3 Integrated Risk-Cost (Final MELCOR)	43
13. Add-On System Evaluation	
13.1 Feasibility of FVCS, Alternative Containment Cooling	24
13.2 Backfit Studies, Increased Containment Volume	36
13.3 Standardized Add-On and Improved Concepts	48
13.4 Design Criteria-Potential Add-ons	60
14. Regulatory Analysis and Standards Development	
14.1 Draft policy or regulatory options	18
14.2 Draft policy or regulatory options	30
14.3 Draft policy or regulatory options	48

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parthentheses. This logic is based on the need to transfer results among program elements for timely accomplishment of element objectives, as required for each of the three research product categories and the regulatory end-products. The program logic also provides a consistent basis for dealing with initiatives such as IDCOR. Application and integration of the research products into regulatory end-products is accomplished in Program Element 14, Regulatory Analysis and Standards Development.

The timing of the program is consistent with staff proposals made in support of FY 1983 budget submittals.

Schedule of Results

While it is anticipated that major outputs covering most outstanding problems will be available in four years, there are significant intermediate results that have a high degree of usefulness and that can be used to draw together interim assessments across the board to define some issues more narrowly, resolve others, and provide interim bases for policy considerations on severe accident regulatory requirements.

A body of knowledge exists now that constitutes the current state of the art, and well before the program described here reaches maturity and begins to wind down, there will be major additions to that body. Some major interim stages are:

End of first year: The first results of the Severe Core Damage Analysis Package (SCDAP) together with initial data from tests in PBF and ACRR to

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improve our knowledge of how to cope with fuel damage in the presence of degraded core cooling (in particular what core coolant level will, if maintained, limit further damages); an improved version of the MARCH code for risk analysis (MARCH 2); and improved hydrogen combustion models should be available. This should permit us to better define the problems in analyzing small-break accidents with respect to the role of hydrogen, whose generation will be predicted using SCDAP and whose combustion will be modeled by advanced methods. By using improved versions of MARCH, we will be able to factor these new models into an iteration of the Severe Accident Sequence Analysis as well as the risk analyses proper.

End of second year: The first integrated appraisal of all three products should be available at this time, including the first output of the value impact of possible risk reduction add-ons; guidelines for operating procedures for recovering from core damage events; integrated assessment of containment loads; hydrogen generation and control analyses; and better data on containment loads from core-concrete interaction. Key input will come from a best-estimate reassessment of the release-from-plant radio-logical source term, available early in this period. This integrated reappraisal will incorporate the assessments provided by the industry group, IDCOR, whose final report is due at this time.

A reappraisal will again be possible, during the following third year of the program, including; risk reduction re-evaluation via the MELCOR code that allows for the systematic introduction of new models and data into the risk analyses; the results of industry assessments, including early results of the NREP program, and, a much improved SCDAP model.

The interim and final research products of the program are phased to provide information and technical bases for draft policy or regulatory options developed in Program Element 14, 18, 30 and 48 months after the start of the severe accident research program.

5.0 PROGRAM ELEMENTS

This chapter contains a description of each of the fourteen program elements of this plan. Each element is described in detail using the following format:

1. Element Description
2. Technical Issues Resolved by The Element
3. Key Interfaces with Other Element
4. Background and Status
5. Plan of Work as a Function of Time

5.1 Accident Likelihood Analysis

5.1.1 Element Description

In this element a number of studies will be performed to reassess the predictions of severe accident sequences and likelihoods made in PRAs. This reassessment will be made based on the availability of new data and PRAs, the reconsiderations of previously produced event trees and accident sequences with greater emphasis on potential common-cause failure mechanisms, the investigation of possible "precursor" events in operating LWRs, and the consideration of the relative likelihood of "TMI-like" accidents distinguished from full core meltdown accidents.

5.1.2 Technical Issues Resolved by This Element

Technical issues to be addressed in this element include:

1. Development of more robust predictions of the likelihood of severe accidents in a variety of LWR design types;

2. Estimation of the contribution to risk originating in external events and sabotage;

3. Relative likelihood of "TMI-like" accidents as opposed to full core-melt accidents; and

4. Identification of important precursors to severe accidents from actual LWR operating experience.

5.1.3 Key Interfaces with Other Elements

This element has key interfaces with three elements:

1. To provide additional information to the Severe Accident Sequence Analysis element relating to the relative importance of different accident sequences;

2. To provide accident sequence likelihood information to the Accident Consequence and Risk Reevaluation element for combination with consequence analyses into risk predictions; and

3. To provide clues to the potentially serious vulnerabilities of reactor plants to severe accidents for use in the Regulatory Analysis element.

5.1.4 Background and Status

Individual programs in this element are as follows:

1. Accident Sequence Evaluation Program

   In this program, reviews are to be made of the accident sequence (event tree) evaluations in plant-specific risk assessments such as the Reactor Safety Study (RSS) and the RSS Methodology Applications Program (RSS-MAP) (see below). These risk assessments, and the reevaluated event trees from this program, will be used as the foundation from which the risk analyses of plant modifications (discussed below) will depart. The objective of the event tree reevaluation will be to consider the need for and to make, as needed, modifications to the event trees to incorporate new information and make them more appropriate for use in the value/impact analyses. More specifically, modifications will be made to differentiate between sequence variations not previously necessary, but important for the value/impact analyses; to permit differentiation between core damage and full core-melt sequences (and to assess their relative probabilities); to make modifications to account for (probabilistically) poorly understood events such as fires, sabotage, operator error, etc.; and to attempt to make the event trees more generic than originally established.

This program is now in the middle of its first iteration of sequence likelihood updating; completion of this phase is planned for mid-1982.

2. <u>Accident Sequence Precursor Program</u>

In the accident sequence precursor program, events in operating LWRs are being examined for their potential, when combined with other events, to lead to a severe accident. After an initial screening to define the more important events, estimates of the likelihood of these events resulting in a severe accident are to be made. The screening of events has now been under way for more than a year. Likelihood estimates will be developed over the next one to two years.

3. <u>Reactor Safety Study Methodology Applications Program (RSSMAP)</u>

The RSSMAP program is intended to apply the methods and insights of the Reactor Safety Study (RSS, WASH-1400) to a somewhat broader spectrum of

LWR designs. Thus, relatively limited event tree and fault tree analysis has been performed on four designs: a B&W plant; a Combustion Engineering plant; a Westinghouse four-loop plant with an ice condenser containment; and a GE BWR plant with a Mark III containment. In addition, the program has recently been amended to include the analysis of a GE plant with a Mark II containment. The final product of each plant study is a discussion of the likelihood of experiencing serious core damage, of having particular magnitudes of releases of radioactive material from the plant (i.e., the RSS "release categories") and an explanation of what types of accidents (e.g., station blackout, ATWS) contribute importantly to these likelihoods of releases.

With the exception of the newly initiated analysis of the BWR Mark II design, the final reports on these studies are either already published or to be published early in FY 1982. It is planned that the Mark II analysis will be completed by late FY 1982 or early FY 1983.

4. Interim/National Reliability Evaluation Program (IREP/NREP)

In the IREP program, PRAs are being performed on a set of plants, using the most advanced methods and data. Detailed fault trees and event trees are being generated and quantified for the purpose of yielding estimates of the likelihood of various serious accidents, an overall likelihood of severely damaging the core, and the likelihood of significant radioactive releases. While this product will provide a measure of the safety of the particular plants, it will also provide a basis for developing general PRA procedures and techniques for use on a larger scale (i.e., on all U.S. nuclear plants). The NREP program is intended to be the vehicle for the latter larger-scale effort.

The plant analyses now under way in IREP are scheduled for completion in FY 1982; NREP studies have not yet been initiated.

5. Industry PRA Reviews

In addition to the plant PRAs being performed under RSSMAP, IREP, and NREP, licensees have initiated (for varying reasons) PRAs on specific plants. It is planned that, as such PRAs become available, reviews will be undertaken and the results incorporated into the overall accident sequence likelihood reassessments being performed in this element.

6. Station Blackout Studies

Station blackout (i.e., the loss of all AC electric power at a plant) has been identified by NRC as an "Unresolved Safety Issue," because (in part) of its relatively high predicted probability in some plants of leading to high consequence accidents. As such, it has been the subject of considerable study over the past several years, including a significant amount of research as to its likelihood in operating LWRs. As this information is being compiled, it is being incorporated in this element's overall evaluation of severe accident likelihoods.

5.1.5 Plan of Work As A Function of Time

It is planned to update the element's accident sequence likelihood evaluations on roughly an annual basis (i.e., the program will be performed iteratively). The timing on the completion of these iterations and the interrelationships among the described programs are shown in Figure 5.1.

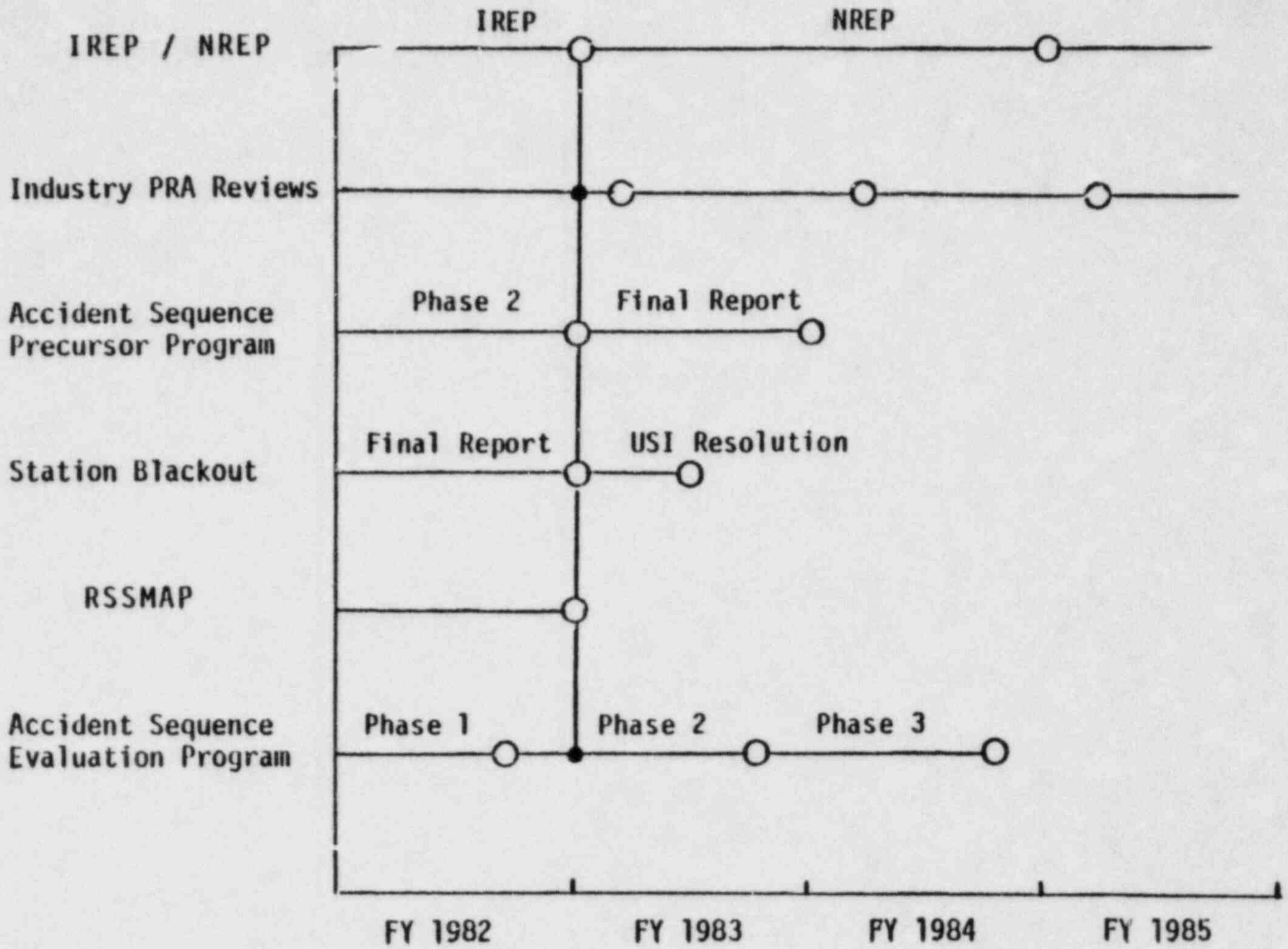

Figure 5.1 Accident Likelihood Reevaluation

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5.2 Severe Accident Sequence Analysis (SASA)

5.2.1 Element Description

This element addresses the problem of improving the understanding of reactor accidents both within and beyond the design basis in order to develop better strategies to prevent, manage, and mitigate severe accidents. Insights into issues will be gained by applying best-estimate state-of-the-art codes (e.g., RELAP, TRAC, MARCH/CORRAL, CRAC), risk assessment methodologies and plant operation procedures to several specific plants.

5.2.2 Technical Issues Resolved by This Element

The issues being addressed by this element include severe accident analysis for specific plant design, licensing and safety concerns generated by NRR Abnormal Transient Operator Guidelines (ATOG), operator instrumentation information needs, fission product release and transport, NRC unresolved safety issues such as:

1. Station Blackout,
2. Shutdown decay heat removal requirements,
3. Anticipated transient without scram (ATWS)
4. Safety implications of control systems and systems interactions in nuclear power plants,
5. Hydrogen control measures and effects of hydrogen burn on safety equipment, and
6. An increased cabability in the NRC emergency response area.

The objective of the proposed Severe Accident Sequence Analysis (SASA) program is to improve understanding of reactor accidents and of the human-machine interface during a broadened spectrum of accident sequences, including those

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within and beyond design basis limits. Particular emphasis is to be placed on the perceptions of the operator, the operator needs for information, the alternative actions the operator might take given various combinations of component failures, the effectiveness of these actions, the influence of multiple failures on plant safety system functional capabilities, the ability of degraded safety systems to be used to bring the plant to a safe shutdown condition, and the environment in which safety systems will be required to survive. Emphasis will also be placed on the recommendation for operator guidelines for severe accidents, minimum instrumentation to follow an accident, and assessment of the effects of the availability of equipment.

5.2.3 Interfaces With Other Elements

SASA will characterize sequences defined by risk assessment and provide a data base for assessment of IREP-developed methodologies. Rulemaking on plant-specific designs will define reactor accident sequences for analysis by SASA. Licensing and safety concerns generated by NRR Abnormal Transient Operating Guidelines (ATOGs) and other licensing reviews will serve to define SASA issues resulting from identified deficiencies in the symptom oriented review. The SASA program also has interfaces with the following elements: (1) fission product release and transport, (2) behavior of damaged fuel, and (3) accident management.

In the interface with fission product release and transport, analysis is in progress on the sequence describing station blackout at Brown's Ferry Unit 1. The station blackout is assumed to persist beyond the point of battery exhaustion. Without DC power, cooling water can no longer be injected, potentially leading to core meltdown and containment failure. During this sequence fission product transport paths exist as identified by SASA, that by-pass the suppression pool. In the interface with damaged fuel, analysis of a hypothetical core meltdown accident initiated by loss of offsite power for the Zion 1 PWR has been recently completed. In the interface with accident management, SASA is involved in the development of a diagnostic algorithm. This process consists of sequence definition and the development of sequence signatures, accident/transient diagnostic methods, and diagnostic software.

5.2.4 Background and Status

Small breaks, loss of AC power, large LOCAs, interfacing system LOCAs and loss of feedwater transients have been analyzed to perform pertinent evaluations of numerous accident strategies associated with these categories of sequences. The loss of AC power analyses are assisting in the resolution of the Station Blackout Unresolved Safety Issue A-44.

An in-depth analysis of the behavior of a representative Westinghouse four-loop plant (Zion I) was completed for small-break, loss of AC power, and loss of feedwater scenarios. The completion of this analysis provided valuable insight into the response of this plant design to scenarios in categories of concern to NRC licensing and research.

The analysis of the response of a BWR and the analysis of fission product noble gas and iodine transport under station blackout conditions were completed.

These analyses considered the impact of the availability and unavailability of various cooling systems. Additional analyses are in progress addressing blackout behavior using advanced thermal-hydraulic codes.

Programs allied to SASA such as Plant Status Monitoring have developed methodologies for identifying instrumentation useful in monitoring PWR and BWR status. Such methodologies interface appropriately with a program such as SASA, which encompasses the identification of operator information required to properly manage accidents and transients.

A SASA calculation log was established. This log will be expanded in the future to become a handbook of accident signatures that can be used to improve simulator and other operator training programs.

Symptom-oriented procedures have been used in SASA loss of feedwater analyses in order to assess the adequacy of these procedures.

The completion of the sequence analysis to date has developed an expanding data base of great value to other programs. This base is being used to develop operator action event trees that can be used to establish appropriate operator action for a variety of scenarios. The data base can also be used to evaluate the accuracy of PRA methodology which will likely play a role in the future process of plant licensing.

Work is in progress in FY 1982 and will continue in FY 1983 to analyze the thermal-hydraulic, fuel, and fission product transport phenomena in eight aging PWRs with a hypothesized reactor vessel break arising from thermal shock with cold repressurization, in the presence of sensitized flaw. The analyses will assess mitigative actions under taken to maintain containment integrity.

Work is in progress in FY 1982 to address the analysis needs in support of operator guidelines for responding to transients and accidents. These include:

1. Depressurization capability in CE plants without Power Operated Relief Valves (PORVs),

2. Multiple steam generators tube ruptures,

3. B&W NSSS design feature,

4. Emergency guideline development for ATWS events, and

5. High point vents.

5.2.5 Plan of Work as a Function of Time

Issues in FY 1983 and beyond addressed by SASA include:

1. Support the development of a severe accident policy for nuclear reactors by addressing system functional requirements to assess prevention and mitigation of a core melt accident in severe accidents involving multiple

system failures, evaluate equipment and system survivability in severe accident environments; and evaluate the impact of proposed plant features on severe accident sequences in which they are not primarily intended to function.

2. Support the resolution of Unresolved Safety Issue A-9, Anticipated Transient without Scram (ATWS). (This will be accomplished by assessing the effectiveness of potential procedural or plant design changes in ensuring the acceptability of the consequences of an ATWS.)

3. Provide analyses addressing the adequacy of assumptions concerning radionuclide transport and source term models.

4. Provide development and evaluations of current or future guidelines for plant emergency operating procedures to assess their usefulness for mitigating and preventing severe accidents.

5. Address plant behavior under complex transients coupled with multiple failures to define the plant behavior and to define the human machine interface required to prevent and mitigate severe accidents.

6. Provide analyses addressing general licensing issues such as depressurization of PWRs without power operated relief valves, structural integrity of reactor containments, and cold repressurization of aging plants. These analyses will address the consequences of such events.

The schedule is shown in Figure 5-2.

	FY 1982	FY 1983	FY 1984	FY 1985
Evaluation of Procedural and Plant Configuration Changes Supporting Severe Accident Rule Making		Define operator actions in preventing or mitigating the accidents	Documentation	Complete Rule Formulation and Development
Evaluation of Procedural and plant design changes for ATWS		Assessment of operator guidelines	Develop effectiveness of operator & plant design remedial options	Completion dependent upon schedule for Final Rule and Regulatory Guideline Development Schedule
Definition of Radionuclide Transport characteristics		Evaluate rate of FP movement in the plant. Estimate FP inventories and transport rates		Completion Date Dependent Upon Resolution of Degree of Conservatism of Current Models
Evaluation of Plant Abnormal and Emergency Operating Procedures	* Examples listed In Section III D. Work to address analyses needs in support of operator guidelines i) as requested by NRR and ii) as required within SASA			
Resolution of Licensing Concerns		Analyze plant grouping for e.g. ANO-1	Documentation	
Identification and Implementation of Operator Information Needs		Identify key parameters that characterize a sequence. Relate symptoms of accident sequences to emergency procedures	Develop Complete Accident signatures	Complete developing an algorithm Documentation
Characterization of Plant Behavior Under Complex Transient Conditions in Conjunction with Multiple Failures		Complete thermal- hydraulic, FP transport analyses identified by Risk Assessment Program	Define plant behavior plant enviornments, evaluate procedure and plant design effectiveness	Documentation

FP - Fission product

Figure 5.2 Severe Accident Sequence Analysis

5.3 Accident Management

5.3.1 Element Description

The objective of this program is to develop integrated strategies that combine elements of plant design and operating configuration with operator guidelines and procedures to optimize the capabilities to prevent, to arrest the progress of, or to mitigate the consequences of potentially severe accidents. To achieve this goal, we need to improve our understanding of reactor accidents and of the human-machine interface during a broad spectrum of accident sequences both within and without the design basis. This increased understanding can then use the PRA methodology for assessment of value-impact, to arrive at a basis for judgement.

This element will integrate analysis and experiments to provide the technical basis for decisions on changes in regulatory requirements on system design criteria and operating guidelines or to confirm the adequacy of established requirements. These regulatory requirements could relate to safety features, instrumentation, or administrative controls on operation.

5.3.2 Technical Issues Resolved by This Element

Technical issues to be addressed in these elements relate to instrumentation, plant design, operator guidelines and training, and the physical phenomena that must be understood to make the decisions necessary for accident management.

5.3.3 Key Interfaces With Other Elements

This element is closely tied to Elements 5.1, 5.2, 5.4 and 5.5 (See Figure 4-1). The prevention aspects of accident management depend upon a combination of accident likelihood reevaluation and severe accident sequence analysis to identify design changes such as pump cooling circuits, auxilliary power circuity, etc., plant configurations, or administration controls that can reduce the likelihood of an accident and hence reduce the need for recovery procedures or mitigation. In the event of system failures, the SASA program provides the basis to evaluate recovery procedures and guidelines. However, experimental research on the behavior of damaged fuel and hydrogen generation and control are essential parts of formulating recovery and control guidelines; for without a firm phenomenological base, the conclusions drawn from SASA studies will have an unacceptable degree of uncertainty in assessing just how to bring the plant to a safe shutdown from a severely damaged state.

5.3.4 Background and Status

Probabilistic risk assessments have continued to evolve since the two reactors were analyzed in the Reactor Safety Study (WASH-1400). Since then, risk assessments have been done under the Reactor Safety Study Methodology Applications Program (RSSMAP) and Interim Reliability Evaluation Program (IREP) as well as through industry studies. These studies have made a significant contribution to identifying the dominant contributors to risk. The SASA studies have built upon these studies to examine in detail dominant accident sequences such as small-break LOCA, large-break LOCA, interfacing systems LOCA, feedwater transients and loss of AC power. These studies have provided valuable insight into the response of specific plants to these scenarios and how operator actions and/or the restorations of systems during the course of an accident can change the scenario and its consequences. Use of these methods to systematically develop guidelines is beginning and will be further developed under this element.

5.3.5 Plan of Work as a Function of Time

This element ties together several other research elements. Within 24 months, there will be sufficient input from the other elements to provide a Phase 1 Accident Management Report. This report will provide preliminary data and guidelines for the development of procedures for the recovery from a core-damaging event.

Within 48 months, the supporting analysis and experiments will have developed substantial additional data allowing the Phase 1 report to be updated and published in final form in order to support regulatory requirements relating to safety features, instrumentation, and operating guidelines.

Figure 5-3 shows this integration of elements in the final product.

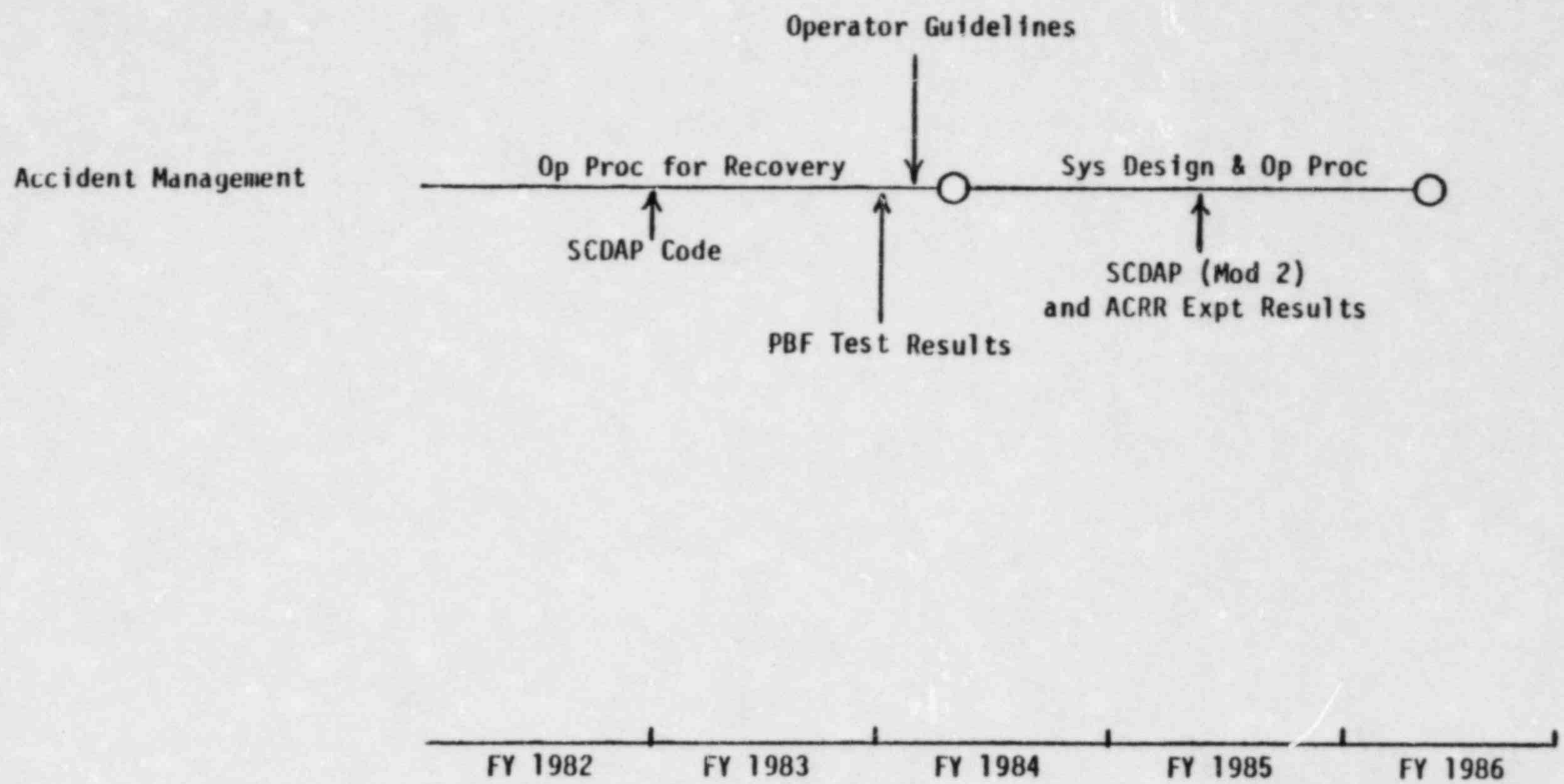

Figure 5.3 Accident Management

5.4 Behavior of Damaged Fuel

5.4.1 Element Description

The accident at TMI-2 raised many questions concerning the behavior of severely damaged LWR reactor cores. Many of these questions were formally addressed by such groups as the President's (Kemeny) Commission, the Rogovin Special Inquiry Group, the NRC Task Force on TMI-2, the Nuclear Safety Oversight Committee (NSOC), and the Advisory Committee on Reactor Safeguards (ACRS). A few of the more important questions indicated implicitly or explicitly that research should be started to:

1. Determine the general behavior of severely damaged fuel by studying its behavior in the 2200°F to 4000°F temperature range that appears to have been imposed on the TMI-2 reactor core,

2. Determine the actual hydrogen release and transport kinetics and, therefore, appropriate hydrogen mitigation features to be required,

3. Determine kinetics of fission product release and transport and their consequences,

4. Determine the consequences of hot core interactions with cooling water and its effect on reactor vessel and containment integrity,

5. Determine the coolability limits and cooling requirements of such cores at various stages of degradation,

6. Apply the newly acquired knowledge to significantly improve the calculation of perceived risk using the methodology developed in the Reactor Safety Study (WASH-1400), and

7. Apply the new information to support and confirm the NRC policy on regulatory requirements for severe accidents.

Accordingly, a four-pronged research program was initiated under this element that included (1) integral in-pile tests in the Power Burst Facility (PBF) and the NRU Reactor at Chalk River, Canada, (2) participation in the TMI-2 core examination, (3) separate effects experiments both in and out of reactor, and (4) the development of an integrated Severe Core Damage Analysis Package (SCDAP) to integrate and make useable the results of the program.

The results of this research program will be incorporated into accident-analysis systems codes and be used to provide guidelines for refinements to system design, operating procedures for improved accident management, and technical bases for improved PRA.

5.4.2 Technical Issues Resolved by This Element

Severe core damage resulting in large hydrogen and fission product releases to the containment can occur despite current regulatory procedures and engineered safety systems. However, the TMI-2 event has shown that accidents that result in core temperatures in excess of 2200°F need not result in a massive core melt or pressure vessel failure as has been conservatively assumed in the past.

If the accident is terminated below approximately 3400°F, the issue is the coolability of a core containing fragmented pieces of oxidized and embrittled zircaloy-clad fuel rods. The major safety issues at temperatures between 3400° and 4700°F are core coolability (i.e., can further core degradation be stopped?) and fission product and hydrogen release from very hot solid fuel rods, liquefied fuel, and fragmented fuel.

The formulation of regulatory policies and criteria for operating procedures to manage and mitigate the consequences of such accidents requires the development of a data base and analytical methodology ranging considerably beyond that needed for current design basis accidents. Very little data are currently available on the characteristics of severely damaged LWR cores. Information is required to determine the coolability of the core, the coolability of various types of fuel/clad debris, the nature of the thermochemical reactions that take place at high temperatures, and the extent and transport of the fission products and hydrogen released. Reliable information must be obtained from in-pile tests that closely duplicate reactor conditions such as the nuclear heat source (in liquid and solid phases), fission products, and prototypical fuel/cladding thermal and chemical reactions.

The technical objective of the Severe Fuel Damage Program is to predict, in a defined volume and time interval, as a function of temperature, heating rate, cooling rate, and pressure the following: the coolability (dry-out limits) of the fuel structure in the volume; the rate of generation of hydrogen from the fuel cladding as it interacts with water and steam; the magnitude and rate of release of fission products, and thereby the net retained decay heat source; and the overall heat transfer characteristics of the volume. These quantities can be used in a predictive way to determine the coolability of the fuel volume in any succeeding time interval, and the net release and transport of hydrogen and fission products into the primary system. In addition, we seek an overall understanding of the way in which fuel relocates as cooling is severely degraded or totally lost, so as to be able to model the attack of hot or molten fuel on the lower vessel internals. We also seek to determine a correlation, if any, between reflood rate and core uncovery time that minimizes further fuel damage from quenching.

A four-part integrated program of research (conducted in parallel) is planned to provide the needed information base of data and verified models. The first part consists of integral, multi-effects, in-pile tests in the PBF to provide early scoping data on governing phenomena, and later, in PBF and NRU, proof tests of the severe fuel damage models and codes developed in the program. The second part consists of separate-effects experiments on the governing phenomena, both in the ACRR and in the laboratory, to furnish a more specific data base for model development. An analysis package is the third part of the integrated program, including development of severe fuel damage models from the experimental data base and their integration into the severe fuel damage code, SCDAP. The fourth part is the benchmark data base to be obtained from the TMI-2 core examination.

The ultimate benefits of this program to the NRC will include:

1. An understanding SFD phenomena that are important in determining the performance requirements for engineered safety features for in-vessel termination of the accident,

2. A base of technical support for the policymaking process for severe accidents, and

3. Guidelines for the acceptance criteria used by the staff in reviewing documentation submitted by licensees.

An additional, more general benefit from the program will be to provide the necessary SFD data base and models for more accurate calculations of the true risk to the public from a given accident scenario. Detailed knowledge of degraded core behavior will allow risk analysts to include risk reduction computations resulting from recovery of previously failed safety systems and/or operator actions during a severe accident sequence. Also, it may be determined that the state of the core after a given sequence may be considerably more benign than that currently assumed because of a lack of available data on core degradation processes. It is expected that the results of this program may increase public confidence in the safety of nuclear reactors.

These benefits accrue from a program of limited scope, since the key data are the correlations between damage state, coolant flow characteristics, hydrogen generation rate, and fission product release. The basic assumption is that a categorization of damage among five or so states is sufficient for this purpose. Should that assumption turn out to be basically untenable, then the scope and objective of the program would need to be reviewed in great depth.

5.4.3 Key Interfaces with Other Elements

Key interfaces of this element with other NRC-sponsored research elements include:

1. Hydrogen Generation and Control Program - Data from the PBF test program and the resulting SCDAP assessment will provide this program with an accurate time-dependent hydrogen release rate from the core.

2. Severe Accident Sequence Analysis (SASA) - It is clear that for the SASA effort to achieve its aims, a strong and reliable data base on the response of all safety-related components of the plant is required. Both SASA and Probabilistic Risk Assessment disciplines are examples of logic exercises that produce no new data themselves, but rely on a wide ranging data base of plant physical responses under abnormal conditions. If the data base for these analytical methods does not exist, the postulated cause-effect relationships can break down and the conclusions will either lack sufficient assurance to be useful or even become untenable.

   Although it can be argued that most plant responses are either well known or calculable from first principles, the responses of the core and fuel behavior as well as fission product release and transport in severe accident scenarios constitute notable and crucial exceptions. There are at present no data which can reasonably be used to predict the extent severe core degradation, fission product behavior and responses, hydrogen generation, core support structure attack, and large melt behavior. If it is remembered that the purpose of nuclear reactor safety is to minimize exposure of the public to fission product radiation, it is clear that core degradation and meltdown accidents that inherently pose the greatest ranges of releases must be better understood for the SASA program to be complete. The Behavior of Damaged Fuel program element is an essential part in formulating recovery and control guidelines, since, without a firm phenomenological base, the conclusions drawn from SASA studies will have an unacceptable degree of uncertainty in the best approach to bring the plant to a safe shutdown from a severely damaged state.

3. Risk Codes - As the SASA program document explains, insights into issues will be gained "by applying best-estimate state-of-the-art codes such as RELAP, TRAC, MARCH-CORRAL, and CRAC...." In addition to the MARCH code, a new risk code (MELCORR) is being developed by the Office of Research. These latter codes are the only codes which address the physical behavior of the degraded core. Yet, the MARCH code was never intended as a device to predict details of degraded core behavior. It is being used for this purpose since no other codes exist, and since a reliable data base is not available as an alternative.

To test MARCH code capabilities and limitations in the context of these applications, NRC recently completed an assessment of MARCH. Results of this assessment were that the code was limited or deficient due to fundamental phenomenological uncertainties in accident modeling and to modeling simplifications. The report notes significantly that there is difficulty in validating such models due to an inadequate phenomenological data base. The new code, MELCORR, will suffer from the same lack of sufficient verification information until core behavior data are supplied by new research efforts described herein.

4. Fission Product Release and Transport Program - The PBF test series will provide some important benchmark data for the models of fission product release and transport in the reactor coolant system developed in this program element from ex-reactor experimental data.

5. Accident Management Program - The data and analyses developed in the SFD program will be applicable to accident management planning and execution. This program will furnish the information required to assess the state of the core and its coolability limit and, therefore, the proper management of such an accident, including guidelines for refinements in system design, operating procedures, and possible additional instrumentation requirements.

6. Accident Consequence and Risk Evaluation Program - Since consequence and risk calculations depend on assumptions of the core state under various

conditions, the data and analyses developed in the SFD program are necessary for accurate risk and accident consequence calculations. Data and analyses will be applied directly to further development of the MARCH code and to development and the new risk code, MELCORR.

5.4.4 Background and Status

Following a review of the events surrounding the accident at TMI-2, the NRC Special Inquiry Group (SIG), headed by Rogovin, and the President's Commission on TMI, headed by Kemeny, published findings and recommendations regarding severe accidents. It is clear from these two independent evaluations of the TMI-2 accident that the bases for licensing nuclear power plants can no longer be restricted to the design basis accidents used in the past, but must be expanded to include consideration of more severe accidents. According to the SIG report: "Modification is definitely needed in the current philosophy that there are some accidents so unlikely that reactor designs need not provide for mitigating their consequences." Further, "Reconsideration of the required 'design basis' for nuclear power plants should be initiated immediately." In addition, the President's Commission stated, "We urge strongly that research be carried out promptly to identify and analyze the possible consequences of accidents leading to severe core damage. Such knowledge is essential for coping with results of future accidents. It may also indicate weaknesses in present designs, whose correction would be important for the prevention of severe accidents." In addition to this need for research to better understand and predict the course and consequences of a degraded core event, several specific issues were raised by the SIG and the President's Commission; among them being: (1) instrumentation to diagnose plant conditions and (2) accident management and consequence mitigation. In the following paragraphs, these issues are discussed in the context of their relationship to information needs on the program on the Behavior of Damaged Fuel.

With regard to instrumentation, the SIG reports noted that "critical information was lost" because of inadequate instrumentation at Three Mile Island. Both the SIG and the President's Commission recommended that monitoring instruments and recording equipment should be qualified for the full range of accident conditions so that critical plant information would be available to the operators. Several of these recommendations were addressed by Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," and by ANSI/ANS-4.5, "Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors." Although some instrumentation requirements are specified by the guide and standard, there are presently no bases for interpreting the state of a reactor core from the information available from the instrumentation.

Accident management and consequence mitigation were identified in Recommendations D.1 and D.2, respectively, of the President's Commission. These are: (1) "equipment should be reviewed...to help them (operators) prevent accidents and cope with accidents when they occur;" and (2) "Equipment design... should be reviewed from the point of view of mitigating the consequences of accidents." The equipment available to an operator would include engineered safety features (ESFs). To respond to these recommendations, it is necessary to review and establish the need for possible refinements in system design requirements.

Such refinements cannot be established without experimental research to define the magnitudes of fission product and hydrogen releases and the conditions under which a degraded core is coolable.

Upon consideration of the above issues, it should be clear that research which quantitatively addresses the consequences of severe accidents is needed. Both the President's Commission and the SIG made recommendations to this effect. Specifically, the President's Commission recommended "that continuing in-depth studies should be initiated on the probabilities and consequences (on-site and off-site) of nuclear power plant accidents, including the consequences of meltdown," and further, "that as a part of the formal safety assurance program, every accident or every new abnormal event be carefully screened, and where appropriate be rigorously investigated, to assess its implications for the existing system design, computer models of the system...management, and regulatory requirements." The SIG recommended that the design basis for nuclear power plants should be reconsidered and that one area of review must be "the magnitude of the accident, including but not limited to the severity of fuel damage and core disruption, the magnitude of release of radioactive material, and the magnitude of hydrogen generation."

5.4.5 Plan of Work as A Function of Time

1. Intergral In-Pile Tests in PBF - The major part of the program of integral in-pile tests is the Severe Fuel Damage (SFD) series in PBF. Phase 1 of the program, which is now under way, will provide integral scoping data in the temperature range 2200-2400K. The tests also will provide data on hydrogen generation and fission product release from the reactor core.

The characteristics of the severely damaged fuel will be obtained from post-test examination. This series is the foundation of the SFD research program and will form the necessary base for the in-pile and laboratory separate-effects experiments on governing phenomena as well as for the models in the integral fuel-behavior code, SCDAP.

The consequences of a severe accident are dependent upon the sequence of damage in the accident, and there are many paths that degraded core accidents can take. The PBF severe fuel damage test program is designed to map the response surface defined by the damage phenomena produced by varying certain key experimental parameters, i.e., heating rate, cooling rate, steam flow, peak temperature, fuel rod burnup, bundle size, and low melting control and structural materials. This mapping of the damage phenomena will be accomplished by attempts to bound the range of the effects of these various parameters. Depending on the parameter, the damage produced in these tests can range from fuel rods with cladding totally oxidized to ZrO_2 and geometry altered only by localized ballooning and rupture of the rods during the heatup, to rods with the formation, relocation, and freezing of molten cladding and liquefied fuel, to the formation of rubble beds of fuel pellet fragments, oxidized cladding fragments, solidified molten fuel, solidified liquefied fuel, and solidified spacer grid and control rod materials. The amount and timing of the release of fission products and the generation and transport of hydrogen are also expected to vary with the parameters mentioned above. The data from these highly-instrumented and well-controlled PBF severe fuel damage tests will be combined with the data from special separate effects experiments and the examination of the

TMI-2 core to form a complete picture of the behavior of a large LWR during a degraded core cooling event.

Current Phase 1 plans call for five 32-rod tests to be conducted: two at slow heating rates less than 0.5°C/second (to fully oxidize the cladding and therefore, preclude the formation of liquefied fuel), two at faster heating rates of about 4°C/second, and one approximating the estimated TMI-2 conditions. One of each of the slow and fast-rate heating experiments will be cooled slowly from maximum temperatures of about 2175K (1900°C, 3460°F) to preserve as much as possible the configuration existing at the maximum temperature, and the others will be quenched with reflood water with the resultant production of core debris. The detailed experimental conditions of the fifth test have not yet been specified. These tests will also verify the adequacy of the designs of the test train and the shroud that is required to contain the liquefied fuel. Moreover, the tests will produce the debris to be used for determining the expected size-ranges, compositions, permeability, and coolability of severely damaged fuel. These characteristics will be used in guiding and planning separate-effects debris coolability experiments in the ACRR. Finally, fission product and hydrogen release and transport data will be obtained from all tests and used to verify and assess SCDAP models as well as current source term analysis methodology.

There has been preliminary planning for a second phase of integral SFD testing in the PBF to explore the effects on core behavior of high-burnup fuel, control rod materials, and fuel element design. This series may also include experiments at higher temperatures and larger bundle size to explore the effects of using a decay-heat source built up by a 1-week irradiation of previously irradiated fuel rather than fission-simulation of decay heat. The larger test-bundle size would also be used to determine the effects of large arrays on blockage distribution and the effects of prototypic amounts of absorber materials. These latter tests will require a modification of the PBF to incorporate a larger test loop that would accommodate up to full-diameter 17 x 17 PWR fuel bundles.

2. Integral In-Pile Tests in the NRU Test Reactor - Subsequently integral SFD data will be available from tests in the NRU reactor at Chalk River, Canada for 21-rod, full-length fuel bundles. The data will supplement the 3-foot PBF results and permit determination of the scaling effect of a 12-foot axial length. These tests are planned to cover a wider range of accident conditions than the PBF Phase 1 tests, including higher pressure and simulation of both PWR and BWR conditions.

3. Separate-Effects Experiments on Coolability of Debris in the ACRR - The second major part of the research on severe fuel damage is a program of supplementary separate-effects phenomenological experiments on the dominant processes involved in the behavior of severely damaged fuel. A major objective of the separate-effects experiments is to determine the range of core conditions (if any) for which simple quench is not sufficient to cool the debris and terminate the accident, and to determine the cooling (pressure and flow rate) necessary for coolability under these conditions. The dry

out coolability limit is reached when liquid cannot penetrate to all points in the debris bed, because of the out flowing vapor. Considerable data and rather sophisticated analytical models of the quasi-static dry out coolability limits of debris beds of decay-heated particulate fuel debris under liquid pools have been developed in fast-reactor safety research. Beginning in late FY 1982, a series of seven LWR-specific core-debris coolability experiments will be performed in ACRR. These will be extensions of the previous LMFBR safety experiments, and the purpose of the initial experiments will be to validate, for LWR accident conditions, the current fast reactor debris-coolability models. The LWR-specific conditions that require experimental verification, in addition to the change to water coolant, are high pressure, very deep debris beds, inlet flow, and particularly the characteristics of the LWR core debris. It is known that the characteristics of the core debris are a major determinant of the dry out coolability limit under reflood conditions.

4. <u>Separate-Effects Experiments on Fuel Debris Formation and Relocation in the ACRR</u> - A program of separate-effects phenomenological experiments has also been started in ACRR on the mechanisms involved in the formation and relocation of fuel debris and on the characterization of the debris. These experiments will provide visual diagnostic data continuously in time for high-probability unprotected accident sequences, as well as debris characterization for reflood quenching at various times in the accident sequences. Data from these separate-effects experiments will be used to develop phenomenological models of the major processes for incorporation

into SCDAP. These separate-effects experiments effectively supplement the larger-scale integral Phase 1 SFD tests in PBF, and they will substantially broaden the data base for model development.

Laboratory separate-effects experiments are planned (depending on the scope of German research) to determine the thermodynamics and kinetics of the reactions between UO_2, Zircaloy, and steam. Experiments are also planned on the candling process with the ternary (U, Zr, O) liquefied fuel, and on debris formation in reflood quenching of molten fuel.

5. <u>Ex-Reactor Separate-Effects Tests</u> - Phenomenological experiments on molten fuel streaming and blockage formation, both laboratory and in-pile, will be performed to acquire a data base for the development of analytical models of streaming and blockage formation by molten fuel and by the fuel, clad, and clad-oxide liquid phases. The work includes: (1) laboratory experiments with nonreactor materials on the basic processes involved and (2) fission-heated experiments with reactor materials that can also provide continuous heating of the fuel in simple, well-characterized geometries. Similar fast-reactor-safety experiments, both laboratory and in-pile, are currently under way.

6. <u>Development of SCDAP</u> - The Severe Core Damage Analysis Package (SCDAP) computer code will predict the physical state of a light water reactor core as a function of time during various degraded core cooling accidents significantly more severe than the present design basis accidents.

The SCDAP computer code will provide a capability for analyzing fuel and core component behavior for severe accidents in an LWR core including UO_2 melt progression and ultimately reactor vessel failure. When completed SCDAP will calculate component temperatures as a function of time and axial position; fuel rod deformation, including clad ballooning and collapse; the amount and chemical forms of released fission products; oxidation of core components and the amount and axial distribution of the hydrogen generated and released; the amount and location of liquefied and resolidified material; the mass of rubble debris and the characteristics and spatial distribution of this debris; and an estimate of coolant flow blockage.

The SCDAP code will be a key product of the NRC's integrated experimental and analytical SFD research program. SCDAP will encompass much that is known and understood about the physical and chemical states of a reactor core at various points in time during a severe accident. LWR reactor utilities may use SCDAP for analysis of proposed accident management procedures and refinements in the design of engineered safety features, core instrumentation and information display systems. The NRC and its contractors will use SCDAP to evaluate licensing analyses and probabilistic risk assessment.

The modeling approach being used for SCDAP makes maximum use of existing computer models developed and assessed for LWR design-base events, and of LMFBR safety on core-debris coolability limits.

The initial version of SCDAP will be developed without any data from the planned SFD experiments. However, many of the models will be preliminary in nature and must be assessed through comparisons with experimental data. The Phase 1 PBF Severe Fuel Damage Test Series and the initial ACRR separate-effects experiments will provide the primary data base for this purpose. The SCDAP development plan is, therefore, closely coupled with the Phase 1 testing program.

The first two Phase 1 tests will provide data for assessment of the initial version of SCDAP. The first test, SFD-ST, will provide information on several unresolved questions about in-reactor chemistry at high temperatures. (Chemistry is important because it provides an important heat-source and affects subsequent fuel behavior.) Out-of-pile correlations for Zircaloy oxidation kinetics above 1850K which are based on the data of Urbanic and Hagen will be checked. Predicted oxidation kinetics in steam or steam-starved environments will be compared with measured oxide thicknesses and hydrogen release rates to see if the oxidation heating and hydrogen release rates are correctly modeled by SCDAP. The second test, SFD-1, will be used to clarify the kinetics of formation of liquefied fuel under realistic temperature distributions and its interaction with reactor materials such as Inconel grid spacers which may join liquefied U-O-Zr and affect the melt's properties significantly. Post-Irradiation Examination (PIE) of the fuel rods will provide compositions and configurations of material that has redistributed and solidified. This information will be compared with SCDAP predictions to check if the limited out-of-pile ternary phase diagrams now available are sufficiently accurate for modeling in-pile liquefaction and solidification.

The third and fourth Phase 1 tests, SFD-2 and SFD-3, will be used to assess the second version of SCDAP. The initial versions of SCDAP will use correlations based on PBF Power-Coolant-Mismatch and Reactivity Insertion Accident tests to determine debris size distribution. SFD-2 will provide initial data for size distribution under conditions that more closely match severe accident conditions. The test will also provide needed information about the composition of debris of different size, packing densities, fragmentation criteria, and size and location of flow channels. The fourth test, SFD-3, will include both liquefaction and fragmentation, and it will confirm the models of these processes developed and assessed using the two previous tests where the processes were studied separately.

Event though the Phase 1 tests will not include fuel with extensive burn-up, PIE will provide preliminary information regarding the release, transport, and relocation of fission products. PIE of the test fuel will determine the redistribution of fission products within the test assembly. As data on fission product behavior are gathered from the Phase 1 tests, the data will continually be compared with SCDAP predictions to confirm and refine the models.

The separate-effects phenomenological experiments in ACRR will also provide data and models for the development of SCDAP. Information will be obtained on the process of fuel debris formation and relocation in flowing steam, debris formation by quench at different points in severe-accident sequences, and on debris coolability limits.

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Development and assessment of the final version of SCDAP will depend on the last PBF Phase 1 test, the Phase 2 PBF tests, the ACRR separate-effect experiments, and on the TMI-2 core examination program. SFD-4 will provide a data base on fuel behavior for typical (TMI-2) heating rates for comparison with predictions by SCDAP. The Phase 2 tests will include fuel with significant prior irradiation, control rods, and tests to higher temperatures. The irradiated fuel tests are needed to confirm and extend the preliminary fission product chemistry data obtained during the Phase 1 tests. These tests are also needed to improve SCDAP modeling of fission product release rates for elements like cesium, iodine, tellurium, and tin whose release rates are not well characterized, and for in-pile fuel stresses and cracking patterns typical of severe accident conditions. Finally, use of irradiated rods allows direct comparisons of temperature and pressure as a function of time for a bundle powered by decay heat. This will be the first time such a comparison is possible with a well-instrumented bundle and will be an excellent assessment for SCDAP.

Some of the Phase 2 tests will include control rod materials in the test bundles which, because of their low melting and boiling points, may have significant effects upon damaged fuel behavior. The Phase 2 tests will provide significant data relevant to control rod behavior, and they will serve as scoping tests to indicate any major phenomena that may have been missed in SCDAP modeling. Additional data on the effects of control rod materials will come from the ACRR separate-effects experiments.

7. TMI Core Examination - The TMI-2 core examination constitutes a unique and important resource on the characteristics of severely damaged fuel. Early recovery and adequate analysis are highly desirable to provide a benchmark for research on and understanding of the behavior of severely damaged fuel, including development of the SCDAP code. The current program of the Fuel Behavior Branch includes modest support for analysis of TMI-2 core debris, but does not, of course, address the cost of the TMI-2 recovery operation.

The actual program of SFD research needed to provide a sound technical basis for accident management and licensing activities may prove to be considerably less extensive than outlined in this plan. This program was derived from our current state of knowledge on the characteristics of severely damaged fuel and on the behavior of such fuel, for which the data base and verified models are in a primitive state. It may well be that later, with data from the PBF Phase 1 scoping tests, the early ACRR phenomenological separate-effects experiments, the TMI-2 core examination, and with models developed from these data, some of the additional programs will prove unnecessary. In any case, the SFD program requirements and program plans should be and will be reexamined periodically.

A summary matrix of questions to be answered by the SFD program and the sources of information needed to provide answers is shown in Table 5-1.

The following paragraphs provide a year-by-year description of the expected program results:

In FY 1982 - The first version of the SCDAP code will be completed. The first PBF scoping test will be performed and analyzed. The first LWR core-debris coolability experiment will be conducted in the ACRR.

In FY 1983 - Two additional Phase 1 PBF tests will be performed and analyzed. The second version of the SCDAP code will be issued containing updates resulting from the first three PBF tests. Three experiments on debris formation in flowing steam and two debris coolability experiments will be conducted in ACRR. An assessment of the use of LMFBR debris-coolability models for LWR conditions will be made. Analyses of results of the NRU clad ballooning program will be completed and planning for follow-on tests will begin. Ex-pile laboratory tests on the thermodynamic and kinetic relationships between uranium, Zircaloy, and oxygen will be started.

In FY 1984 - Analyses and experiments will be completed on all Phase 1 SFD tests conducted in the PBF. Fission-product distributions, detailed analysis, and accident signatures will be reported. One or two Phase 2 PBF tests will be conducted if sufficient funding is available for planning in FY 1982 and FY 1983. The final advanced version of the SCDAP severe fuel damage code will be completed. The initial series of experiments on debris formation and relocation in flowing steam in the ACRR will be completed, and a new series on debris formation under reflood quenching will begin. Two debris-coolability experiments will be performed in ACRR. Two initial NRU follow-on tests will be completed to bridge the gap between clad ballooning data for large - and small-break LOCAs. Initial information will be available from the TMI-2 core examination.

In FY 1985 - Three Phase 2 SFD tests will be conducted in the PBF and the preliminary data analyses will be reported. Final data analyses on the Phase 1 PBF SFD tests will be completed and reported. Three SFD tests will be performed in the NRU reactor; data and analysis reports will be issued. The experiments in ACRR on debris formation and debris coolability will be completed. The SCDAP code will be assessed and kept on maintenance. Additional information from the TMI-2 core examination will be available for benchmarking SCDAP and its extensions to whole core analysis with other codes.

In FY 1986 - All SFD testing will be completed. Development of the phenomenological models from the results and of the integral SCDAP code will be completed. An assessment of the needs for higher temperature and larger-scale testing in PBF or NRU will be made at this time.

Input-Output Matrix for the Severe Fuel Damage Experiment Program*

To Answer These Questions: ⟶

We Need: ↓

Information, From ⟶	Experiments		1) For what range of severe accident conditions is the fuel debris coolable by slow reflood for in-vessel accident termination?	2) Beyond that range, what reflood velocity and pressure are necessary to cool the debris?	3) Can too rapid reflood fail the reactor vessel?	4) What are the amounts and rates of hydrogen generation?	5) What is the radiological source term out of the reactor vessel?	6) If debris cooling fails, what are the conditions of reactor vessel failure and debris entry into the containment?
	Multi.-Ef.	Sep. Ef.						
1. Clad Ballooning, Burst, and Blockage	PBF-1, 2 NRU	ACRR Lab	X	X		X	X	
2. Oxidation (Hydrogen)	PBF-1, 2 NRU	ACRR Lab	X	X		X	X	
3. Fission Product Release and Attenuation	PBF-1, 2	Lab					X	
4. Fuel Debris Characterization	PBF-1, 2	ACRR	X	X	X			X
5. Fuel Debris Relocation, Blockage	PBF-1, 2 NRU	ACRR	X	X	X			X
6. Reflood Debris Characterization	PBF-1, 2 NRU	ACRR Lab	X	X	X	X		X
7. Rapid Steam Generation and Explosion	PBF-2 LMF	ACRR Lab			X	X		X
8. Damaged Bundle Coolability	PBF-2	TBD	X	X				
9. Fuel Debris Coolability	PBF-2	ACRR Lab	X	X				X
10. Post-Dry-Out Behavior		ACRR Lab						X
11. Melt Progression	TBD							X
12. Debris Characterization at Vessel Failure	TBD	TBD						X

* The data and models from these experiments are integrated into the SCDAP severe core damage code

Table 5-1

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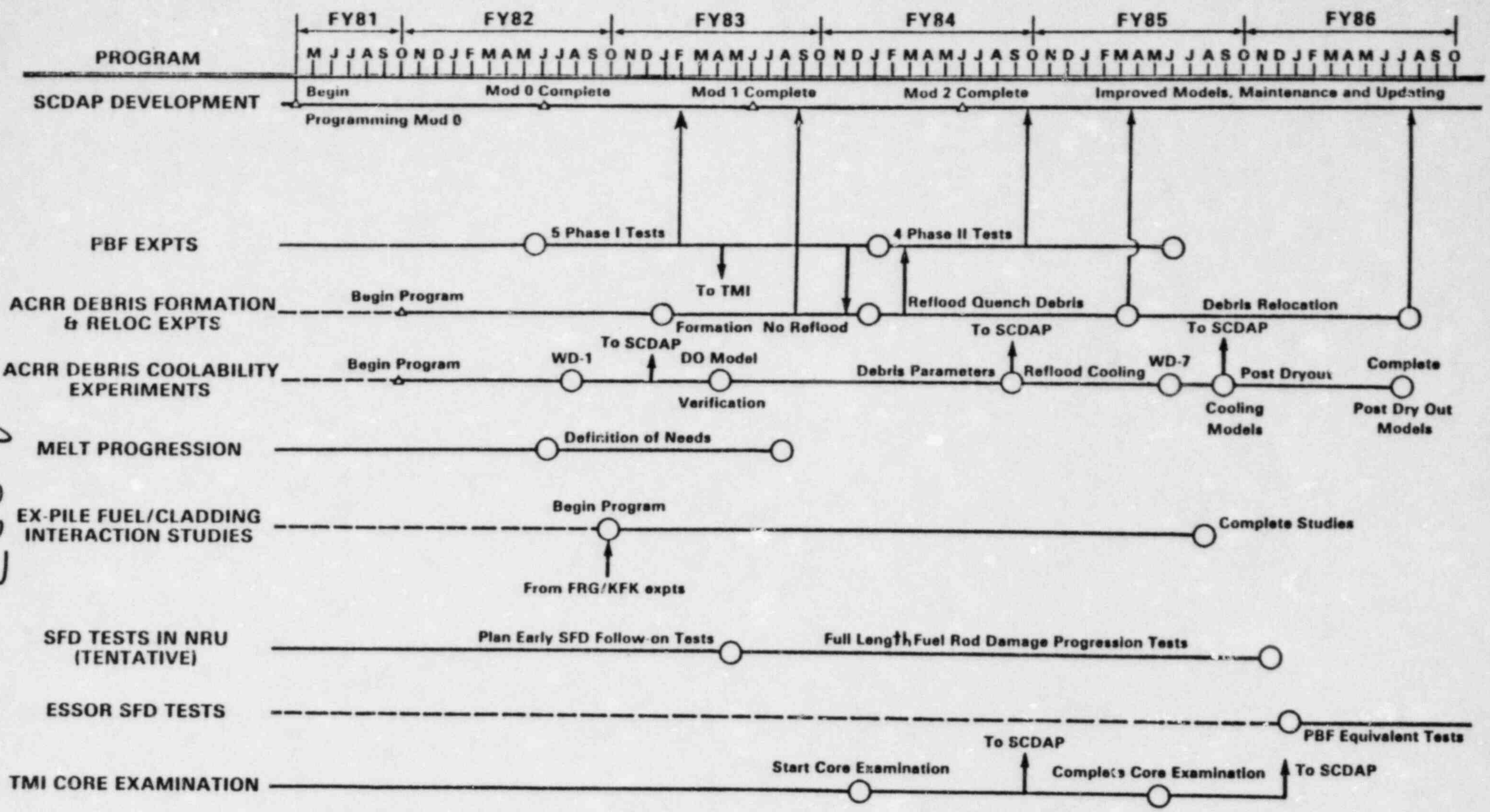

Figure 5.4 Behavior of Damaged Fuel

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5.5 Hydrogen Generation and Control

5.5.1 Element Description

Durning an accident, or as the consequences of an accident, significant quantities of hydrogen can be generated in the reactor vessel from steam-zirconium and steam-steel reactions and in the containment building from molten core-concrete interactions. The burning of this hydrogen could (1) produce loads on containment that could exceed the ultimate strength of the building or (2) cause the failure of safety-related equipment that would affect the safe operation of the plant. This program is currently providing information and analytical models to quantify this threat and to assess the efficiency of mitigation systems proposed by near-term operating license applicants and other possibly more efficient systems. This work also includes the development of an analytical model that will permit better understanding of hydrogen transport, mixing, and combustion phenomena.

5.5.2 Technical Issues Resolved by This Element

As a result of Commission requirements that plants be able to handle hydrogen releases, a number of prevention methods and mitigation schemes for hydrogen control have been proposed by utilities and vendors. An objective of the RES hydrogen program is to provide technical information on the adequacy and efficacy of these systems and also to assess the possible alternatives to these systems. In the review of accident scenarios where hydrogen is released, key technical questions arise as to the timing of the release, the amount of transport or mixing of the gases in containment, the potential for transition from simple deflagration to detonation to occur, the pressures and temperatures from hydrogen combustion, and their effects on equipment survivability.

Major technical objectives are listed below:

1. Accident analysis calculations will be made with the MARCH code for Zion, Sequoya, and Grand Gulf plants.

2. An improved multicompartment deflagration code, HECTR, will predict pressure and temperature histories during and after a hydrogen deflagration in the presence of steam and other gases.

3. Based on the accident analysis results, calculations will be performed for local regions in the containments where detonations are considered possible; the potential for missiles will be assessed.

4. A manual will be prepared and published on the behavior of hydrogen as a guide for the preparation of plant-specific operator emergency manuals.

5. A two-pronged attempt aimed at modeling accelerated flames will be initiated. One attempt is to produce a model based on the underlying physics and the available experimental data. The other will employ existing computer codes for combustion analyses.

6. The first two test series will be conducted in the Fully Instrumented Test Series (FITS) facility to investigate deflagrations and detonations in

ternary mixtures of hydrogen/air/steam. Scoping tests on the effects of aerosols on igniter performance will be performed.

7. The steam/hydrogen jet facility will be checked out and the first two test series will be performed to study autoignition, flame characteristics, and stability (including the effects of flame holders).

8. The Variable Geometry Experimental System (VGES) 16 ft. tank facility will continue to provide scoping information on combustion phenomena. Tests will address mitigation effects, flame acceleration, and direct initiation of detonation.

9. Construction of the flame acceleration facility at Sandia will be completed. Experiments will be initiated to study flame acceleration as a function of obstacle characteristics. Then experiments will begin to investigate detonations. These tests will be closely coordinated with the bench-scale tests being performed at McGill University, and both will be compared to available analytical models.

10. Safety related equipment will be procured and tested in the FITS, VEGS 16 tank and radiant heat facility to characterize the response of this equipment to hydrogen deflagration in the presence and absence of steam.

5.5.3 Key Interfaces With Other Elements*

1. Accident Management Program - Information developed from this program will be issued to assess and develop guidelines for handling hydrogen emergencies.

2. Containment Analysis - This program will provide improved burning models to be incorporated in CONTAIN and other containment analysis models.

3. Containment Failure Models - This program will provide the pressure and temperature loads placed on a containment building during a hydrogen burn. This information will be used in assessing the response of the containment to these loads.

4. Risk Code - Models developed in this program element will be incorporated, (possibly as modified versions,) into risk codes such as MARCH or MELCORR.

5. Add -on System Evaluation - This program will provide information on the efficacy of various hydrogen mitigation systems, that can be used in assessing the value of particular systems for a specific plant or containment type.

*The Electric Power Research Institute (EPRI) has a large research program on hydrogen which is complementary to the NRC effort. In many areas, and which is providing direct information in some areas (e.g., mixing, large scale effects) to the NRC program.

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5.5.4 Background and Status

1. Hydrogen Behavior Program

In this program deflagration and detonation models (named HECTR and DETON respectively) are being developed to more accurately predict temperature and pressure histories in containment during and after a hydrogen combustion. The work on deflagrations will include the effects of CO_2, CO and water fog evaporation. The detonation model will be able to predict Chapman-Jouguet pressures and temperatures characteristic of the transition through a detonation front, including increases after normal reflection. The current effort includes heat transfer mechanisms such as radiation, convection with or without condensation, conduction into surfaces, and the evaporation of sprays in the model. A first version of these codes should be available in FY 1982. The models will be improved and validated in time to provide the NRC with a technical assessment of interim deliberate ignition systems currently being proposed by several utilities.

A number of detonation calculations have been performed using the CSQ code for Sequoyah analysis (NUREG/CR-1762) and for the Zion study. The CSQ code will be used in conjunction with DETON to calculate the hydrogen combustion loads on various containments.

MARCH calculations have been performed for a number of scenarios in which hydrogen is released from the primary system. These analyses have been done for Zion (large dry PWR) and Sequoyah (ice condenser) and are currently planned for Grand Gulf (BWR MARK III). These analyses have provided useful information to NRR in their assessment of the Sequoyah interim distributed ignition system and will likewise be used in the evaluation of the hydrogen control system for Grand Gulf.

A difficult area of analysis now being addressed deals with hydrogen transport. An attempt is being made to modify or develop a code to predict the concentrations of hydrogen, air, and steam in containment as functions of position and time for hypothetical LWR accidents. The German code RALOC is being assessed to determine its present and potential ability to handle transport and mixing analysis. Currently it is planned to test the code against a series of hydrogen mixing tests in the Containment Safety Test Facility at Hanford Engineering Development Laboratory (HEDL) and sponsored by EPRI. Additionally, two other codes, COBRA and SOLA-D, will be modified and tested against the HEDL experiments. The staff is currently considering the need for combining the combustion and transport models together in a single hydrogen code.

The hydrogen program also includes experimental projects directed at determining hydrogen deflagration and detonation limits in air and steam and noting how the location and strength of the ignition source affects those limits. The experiments will determine temperature and pressure profiles as a function of time, thus providing information needed to develop and validate analytical models. Additionally, tests are being planned to understand autoignition of hydrogen, particularly as it relates to hydrogen and steam jet releases, similar to what might occur from a pipe break.

A key area of study is the work on the transition from deflagration to detonation that can occur as a flame propagates from one chamber to another, through a concentration gradient, or accelerates around structures. Flame acceleration in the upper structure of an ice condenser containment was a concern that was raised relative to the Sequoyah distributed ignition system and subsequently in the McGuire hearings. (Flame acceleration occurs when a flame front bends around a structure and begins to break up, inducing turbulence and developing more surface area and causing the flame to burn faster with higher temperatures and pressures. These phenomena can cause lean mixtures to reach temperatures and pressures exceeding the theoretical adiabatic-isochoric limits.) Some engineering scale experiments are currently being planned to mock up the upper structure of Sequoyah to assess this effect. Additionally, contractors at McGill University are performing laboratory scale tests on flame acceleration and the transition from deflagration to detonation. Analytical work in this area is somewhat behind the experimental effort. However, it is expected that as a better understanding of the phenomena involved is developed, model development will accelerate. It may be necessary to perform some large-scale testing on hydrogen combustion, particularly the area of deflagration-to-detonation transition. An assessment of this need will be performed in FY 1982.

2. <u>Hydrogen Combustion, Mitigation, and Prevention Program</u>

Work in this area is directed towards an understanding and assessment of methods to control hydrogen combustion or at least to dampen its effects. The Hydrogen Combustion, Mitigation and Prevention Program was originally a part of the Hydrogen Behavior and Control Program but was separated in order to allow more emphasis to be placed in studying the design criteria and feasibility of proposed prevention and mitigation schemes.

Experimental facilities and tests being run under the Hydrogen Behavior and Control program to characterize hydrogen deflagration and detonation also include the effects of water fogs and foams as a mitigative approach to controlling temperatures and pressures. A .05 percent volume fraction of suspended water droplets can reduce the temperature from a stoichiometric mixture of hydrogen and air from approximately 2700°K to less than 1100°K with a proportional reduction in pressure. The combination of hydrogen igniters and a water fog system appears to present a much higher level of protection in handling potential hydrogen combustion accidents for certain containments than either system alone.

Mitigation experiments on the effectiveness and feasibility of oxygen depletion, pre-inerting and post accident inerting with carbon dioxide and halons are also planned. Other prevention and mitigation schemes, such as gas turbines and high capacity recombiners, will also be assessed. Although some of this work may continue in FY 1984, most should be completed by FY 1983.

An experimental investigation of the feasibility of a deliberate flaring technique in conjunction with a high point vent as a method for controlling hydrogen release during an LWR accident will also be performed. This assessment should be completed in FY 1983 with the need for further work to be determined at that time.

3. Hydrogen Burn Equipment Survival Program

The combustion of hydrogen in containment can lead to the failure of important safety equipment or equipment necessary for plant isolation during an accident. The H_2 Burn Equipment Survival program was initiated in order to experimentally assess the effects of hydrogen combustion on equipment. This program also provides a data base in order to develop analytical models to assess equipment survivability. NRR is sponsoring the other half of this program, i.e., to develop analytical methods to calculate the effects of hydrogen combustion on equipment. The experimental facilities used in the Hydrogen Behavior and Control program will be used to actually test equipment in hydrogen burning environments; this test will begin in FY 1982. The program is phased in two parts: (1) to provide information to NRR in the short term on the effects of hydrogen combustion equipment for near-term licensing decisions and (2) to develop reliable methods to predict the response and survivability of equipment. In addition to the current facilities available for testing, the possibility of using SANDIA's Radiant Heat Transfer Facility to test larger components, if necessary, is being assessed. The test plan now covers multiple burns under deflagration conditions, and the need to perform tests under detonation conditions is also being considered.

5.5.5 Plan of Work As A Function of Time

The analyses of hydrogen accident for the three specific plants (Zion, Sequoyah, and Grand Gulf) should be completed in FY 1982. Analysis for other specific plants or standardized plant designs will be performed as requested using the deflagration code HECTR and a validated hydrogen transport code. An assessment of H_2/Air/Steam deflagration limits will be completed in early FY 1983. An experimental assessment of flame acceleration effects (except for any large-proof test whose need remains to be assessed) and preliminary analytical models should be available by the end of FY 1983. An assessment of the effects of aerosols on hydrogen control systems will be performed in FY 1983. An experimental assessment of the effects of fogs and foams on hydrogen burns will be completed by the end of FY 1982 and an initial assessment of pre-inerting, O_2 depletion and post accident CO_2 inerting by the end of FY 1982 with a final evaluation in FY 1983.

The Hydrogen Burn Equipment Survival Program will have completed a number experiments on safety related equipment during FY 1982 and during that same period will have developed some preliminary thermal models for assessing the effects of hydrogen burns on equipment. Experiments on equipment survival should be completed in FY 1984 and should result in standard methods for testing equipment under hydrogen burn conditions. Figure 5.5 lists additional milestones for this program element.

HYDROGEN GENERATION AND CONTROL

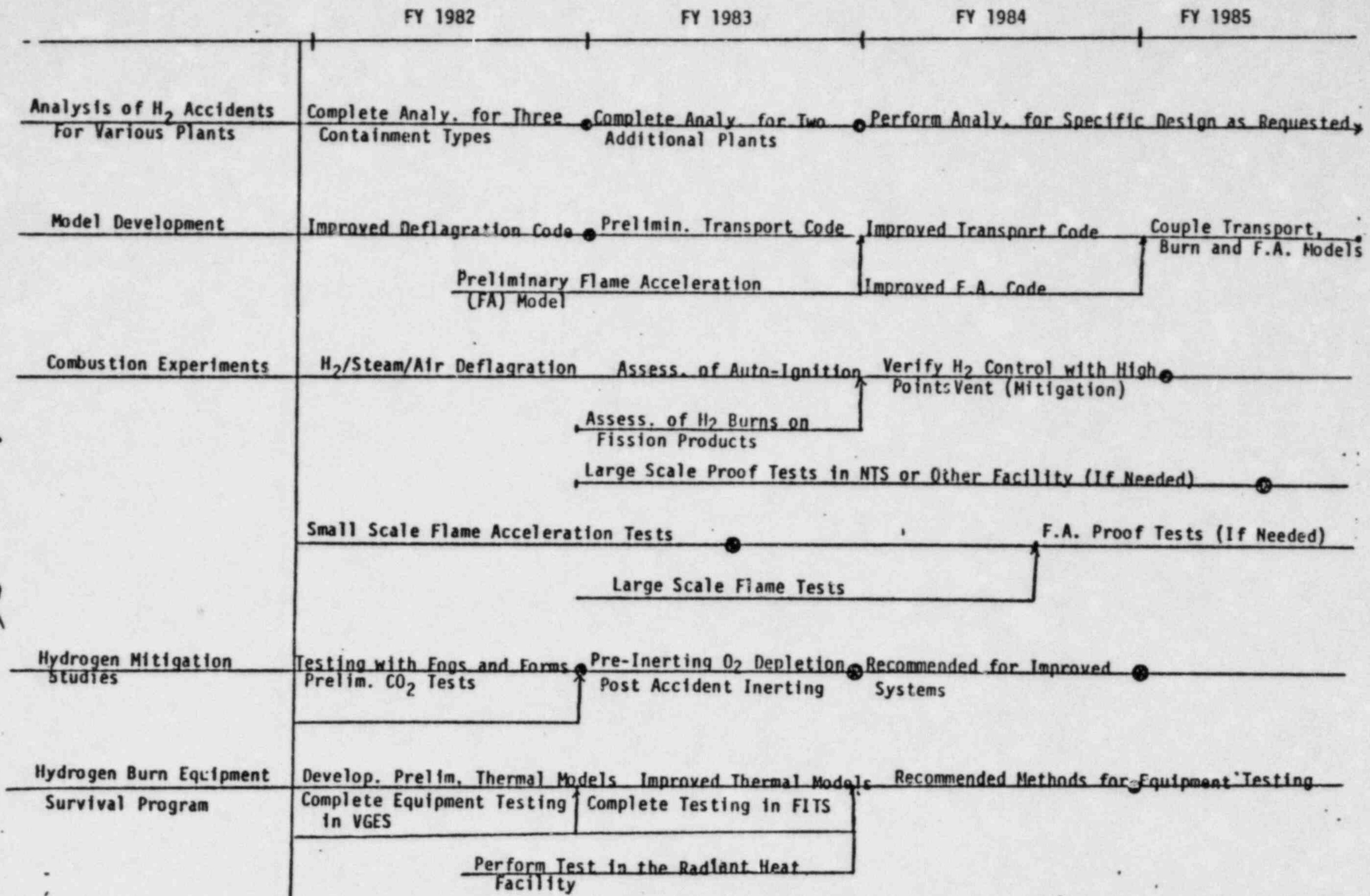

Figure 5.5 Hydrogen Generation and Control

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5.6 Fuel Structure Interaction

5.6.1 Element Description

This element addresses the interaction of fuel and other material from the primary system with the receiving component below the pressure vessel. After the fuel has escaped from the pressure vessel, the consideration of consequences of this release requires knowledge of interactions of the fuel with other material, i.e., with the basemat concrete, with water that is either present below the pressure vessel or introduced later, with concrete that is present with coolant, and with mitigation structures or devices.

Studies and experiments are conducted to evaluate the interactions with respect to penetration rates, heat generation and release, gas and aerosol release, and the rapid generation of steam with the possible formation of missiles. In turn, these quantities establish the loads on the containment for risk assessment.

5.6.2 Technical Issues Resolved by This Element

The technical issues include the interactions of core material or hot severely damaged fuel products with the internal containment environment; the interactions of fuel and water; the rapid generation of steam and the possible formation of missiles; the loads on the containment structure; the effect on instrumentation required to follow or control the accident; the source terms required for the design of mitigation systems; and the quantification and verification of parameters for analysis codes.

Steam explosions from in-vessel core-melt-water interactions have the potential to fail the reactor vessel, most probably by failure of the lower head, and also to generate missiles that would threaten the containment if there were no missile shield. Non-explosive rapid steam generation during both in-vessel and ex-vessel melt/water interaction, the "steam-spike" problem, also has the potential to fail the reactor vessel and the containment directly. The characteristics of the products as formed by water reflood of very high temperature fuel are key elements in assessing the coolability of that mixture.

5.6.3 Key Interfaces With Other Elements

The key interfaces with the elements are with Containment Analysis, the Fission Product Release and Transport, the Containment Failure Mode, and Accident Management. The core melt/concrete interaction experiments establish source term values for Fission Product Release and Transport element. In addition, these experiments establish the parameters for the core/concrete interaction model of the CORCON code, which is a module of the CONTAIN code. Concrete penetration rates, heat generation and partition, and the steam generation and work conversion efficiencies are also included. The CONTAIN code is a major tool for analytical investigation with the Accident Management Element.

5.6.4 Background Status

The scope of this task includes small-scale scoping and phenomenological experiments of thermal, mechanical, and chemical interactions of fuel above the solidus temperature, and of high temperature core debris simulants with concrete, refactory, and sacrifical materials; large-scale scoping or model-verification tests; quantification of gaseous and aerosol source terms for the interaction; heat redistribution with gaseous or aerosol sweepout; and evaluation of the effect of coolant on the fuel-mass-concrete interaction.

A large facility to study the interaction of molten fuel and structural material has been completed and the first large-scale test crucible prepared for an initial test in the second quarter of FY 1982. A second large-scale test is being assembled. A smaller facility for sustaining the heating of hot structural material and/or some material on basemat materials has been completed and the initial test has also been completed. The development of most instrumentation for the scope of testing noted above is nearing completion.

An extensive data base on the conversion of core mass thermal energy into steam-explosion mechanical work has been developed in the Fully Instrumented Test Series (FITS) facility. These results, when combined with analysis of missile generation by in-vessel steam explosions and missile failure of containment led to an early estimate that the probability of containment failure by steam explosion in an LWR meltdown accident is considerably less than the 0.01 estimate in WASH-1400, but still in a range necessitating further research. This estimate may be revised in the light of larger-scale test results. Medium-scale FITS tests have been completed with thermite-generated corium melts dropping into water. In addition, single-drop experiments on the phenomenological mechanisms involved in steam explosions have been made. These experiments have provided important information, but sufficient understanding does not yet exist to construct a mechanistic model of the thermal detonation process that would have predictive capability.

5.6.5 Plan of Work as a Function of Time

A systematic study with large-scale fuel-mass interactions with concrete and retention materials has been initiated. The first test will be conducted in the second quarter of FY 1982. A second large-scale test will be conducted in FY 1982 and orders for additional test crucibles will be placed upon completion of a satisfactory test. The intermediate-scale testing of sustained heating on basemat concrete and core interdictive materials will proceed throughout FY 1983 and 1984. The first independent results are expected in FY 1985 from the KfK Beta facility using large thermite melts.

The interdictive materials to be considered are MgO bricks and castable ceramics (notably the high alumina cements). Core coolant effects will be introduced in FY 1983-1984. Both the introduction of hot materials into coolant pools and the introduction of water onto the hot materials will be included. In the FY 1984-1985 period, tests for floating nuclear plants are expected to be included.

In addition, back-fit considerations of loose gravel beds (thoria) will be studied for sweepout and coolability.

Experiments will be initiated of non explosive rapid steam generation conditions (steam spikes) which might threaten the integrity of both the reactor vessel and the containment. Experiments in these areas will continue through FY 1985, and then terminate unless important problems requiring further work become apparent.

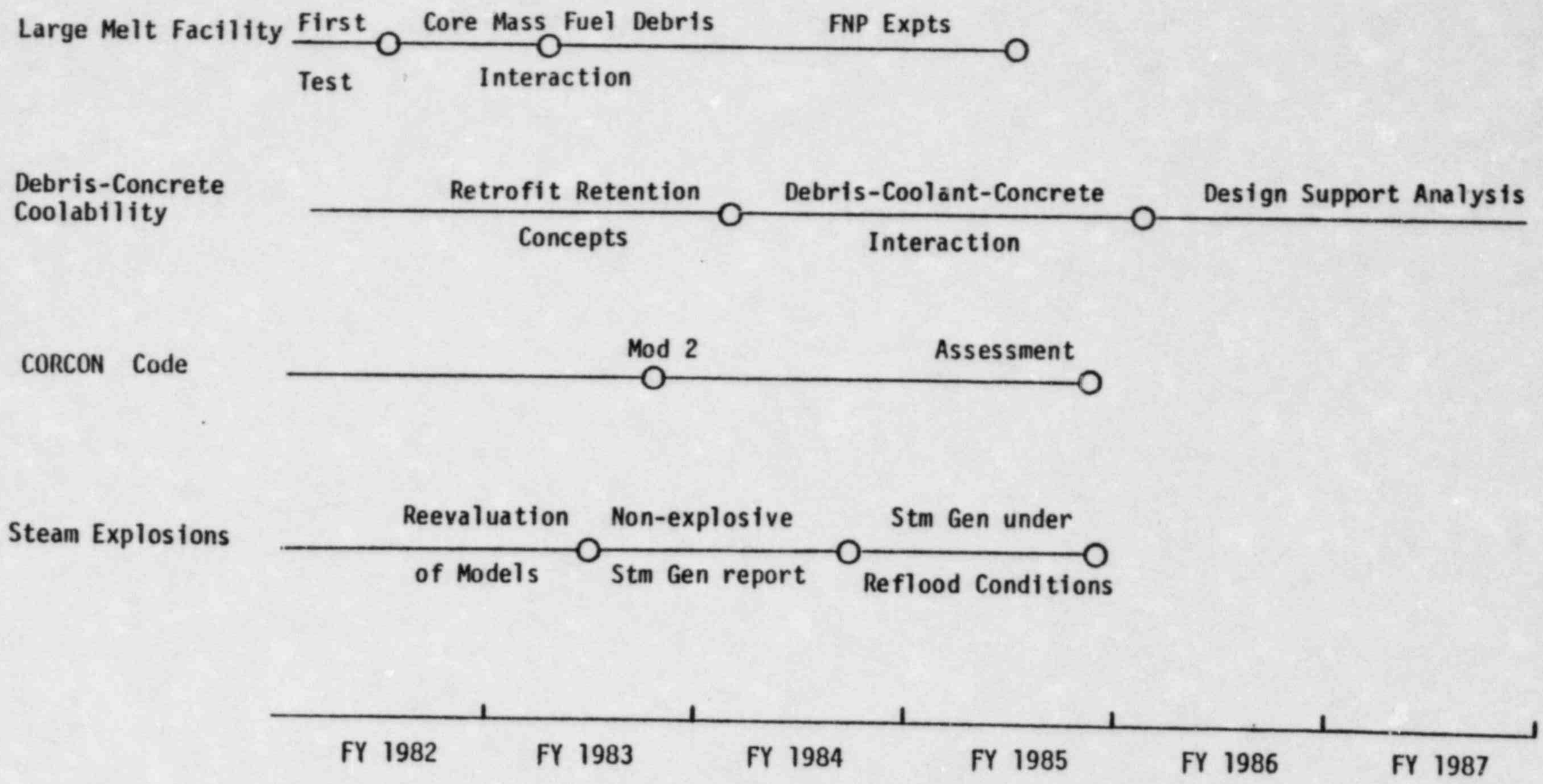

Figure 5.6 Fuel-Structure Interaction

The majority of the experiments will be done with thermitically-generated melts using the FITS facility. Some check data will be obtained with furnace-heated purely oxidic melts, with on large-scale check test (200 kg) probably performed in the LMF. Analysis of both the explosive and the non-explosive rapid-steam-generation processes, in both the dropping and reflood contact models will be continued in an attempt to develop predictive, mechanistic models.

5.7 Containment Analyses

5.7.1 Element Description

This program element is intended to provide analytical tools and phenomenological models to assess the loads that threaten the containment during a severe accident. If the primary system has failed and extensive fuel damage has occurred, the containment will be threatened by steam pressure, fission product and aerosol formation and potentially hydrogen burning, and, in extreme accidents, a threat to the basemat for molten fuel.

To do these analyses, a generalized systems code, CONTAIN, together with the necessary sub modules, is being developed that is sufficiently generic and highly flexible to be capable of handling both BWRs and PWRs with the variety of different containment designs that exist.

5.7.2 Technical Issues Resolved by This Element

The major technical issues to be addressed by the element are the quantification of loads that threaten the containment and characterization of the radiological source term that would threaten the public should the containment leak or fail. Some issues related to containment threat are:

1. Overpressurization due to steam production,
2. Overpressure due to hydrogen combustion,
3. Potential for missile production,
4. Gas production from molten fuel/concrete reactions,
5. Basemat penetration from molten fuel,
6. Coolability of core debris in reactor cavity, and
7. Performance of containment engineered safety features.

Technical issues related to source term are:

1. Fission product chemical behavior,
2. Characterization of aerosols and gases that can transport plutonium and fission products.

5.7.3 Key Interfaces with Other Elements

Because of the wide spectrum of phenomena involved, the CONTAIN program is related to a large number of other ex-vessel accident research issues, i.e., experimental projects aimed at the empirical determination of core/coolant interactions with concrete, the generation and transport of aerosols, the behavior of high-temperature debris pools, and the detailed nuclear decay properties of fission products. The containment-analysis program is thus closely interactive with many of the efforts being conducted within program elements 5.5, 5.6, 5.8 and 5.9. Also, the computational models in CONTAIN involve a variety of other analytical research projects, the purpose of which is to provide reliable predictive calculations of the various processes.

5.7.4 Background And Status

The CONTAIN code project was initiated under LMFBR-research aupsices, but the generic nature of the problem was early recognized, so that the basic code structure was designed to be independent of reactor type. One of the first problems to which CONTAIN was applied related to Zion-Indian Point Study. The importance of considering water-vapor condensation on aerosols was studied. At present, the CONTAIN code is operational on the Sandia and NRC computer systems, and a draft report of the project has been issued. The first version ofthe CORCON code has been documented as a stand-alone tool and is currently being interfaced with CONTAIN. The MAEROS code is primary submodule of CONTAIN and lacks only final refinement; a draft report has been given limited distribution. The MAEROS aerosol model is the state of the art tool for computing fission product transport via aerosol migration and will be tested against a number of other similar codes such as HAARM, QUICK and NAUA.

5.7.5 Plan of Work as a Function of Time

The CONTAIN-CORCON interface will be operational in FY 1982. Verification of the aerosol treatment will be completed as final experimental data become available. The LWR-model specifications will be completed in FY 1982 along with completion of the LWR-debris/coolant interaction model. The first complete LWR version of CONTAIN will be operational by the first part of FY 1984 and verification continued throughout FY 1984, as results of the core-concrete experiments become available.

The CORCON code will be extended to treat the conditions expected later in the accident scenario, i.e., attack on concrete by slurred pools and solid crusts of high-temperature core materials. This will be completed by the first part of FY 1984.

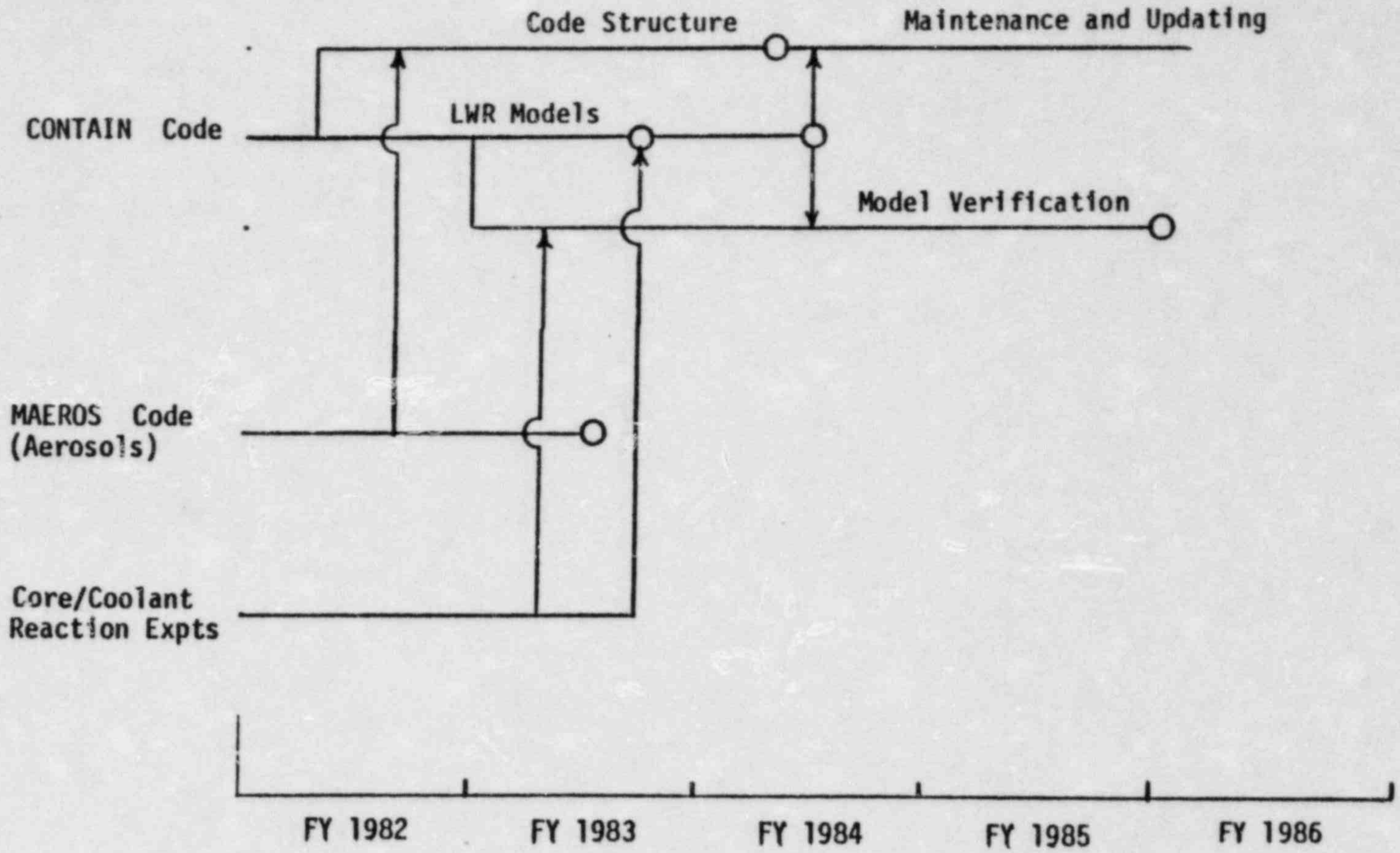

Figure 5.7 Containment Analysis

5.8 Containment Failure Mode

5.8.1 Element Description

The major source of risk to the public from the operation of nuclear power plants stems from accident scenarios that lead to a containment failure. The regulatory concern in this element is that the failure modes and associated load levels for containment structures cannot be predicted with any real confidence by state-of-the-art methods. Both assessments of the risk posed by loads outside the design basis and estimates of the effectiveness of proposed mitigative steps require an ability to predict the way in which a containment will fail.

This element does not address, however, the failure mode arising from the simple failure to isolate the containment. Both the utility and the NRC address this part of the problem through quality assurance practices, inspection and enforcement, and other administrative and management techniques. Similarly, scenarios that bypass containment via penetrations are treated in the SASA and in Accident Consequences Risk Evaluation elements.

5.8.2 Technical Issues Resolved by This Element

The associated safety question relates to the ability to predict, with high confidence the amount of load that can be sustained by a containment structure before the rate of leakage becomes unacceptable. The technical problems involve developing an ability to predict deformations for the wide array of containment types, relating deformations of containment structures to leak behavior, and determining the sensitivity of predictions to uncertainties in actual containment structures and the loads associated with accident scenarios.

The objective of this element is the development and verification of methodologies that are capable of reliably predicting the capacity of containment structures under accidental and severe environmental loadings. The reliability of any predictive method must be verified through experiments. This project contains a combined analytical and experimental effort towards the establishment of reliable predictive methods for the safety margins of containment structures under accident and severe environmental conditions. Steel and concrete containments, both prestressed and reinforced, will be studied for pressure and seismic loadings.

5.8.3 Key Interfaces With Other Elements

There is, and will continue to be, significant interaction with other NRC-sponsored programs related to the severe accident research program. Particularly close coordination will be maintained with the programs on Fuel-Structure Interaction, Hydrogen Generation and Control, and Containment Analysis. In addition, there will be interactions with the Risk Development Codes element. There will also be interaction with other U. S. programs. Contributions to the Containment Failure Mode Program Element are anticipated by the provision of analytical predictions of capacity to be compared against test results from the Electric Power Research Institute. There will be coordination with the Containment Capability Program being sponsored by the Department of Energy.

Two foreign programs have been identified as sources of information. One is the proposed test to failure of a scaled model prestressed concrete containment to be conducted in Great Britain. The other is the planned testing, on a shake table in Japan, of containment models to simulate seismic response.

5.8.4 Background and Status

Effort in FY 1982-1983 will be limited to determining containment capacity under static overpressure. The principal FY 1982 activity is experimentation involving the examination of six prototypical steel models about 1/32 the size of a containment and the design of the large prototypical steel model about 1/10 actual size. This large steel model will be used in the FY 1983 experimental effort.

Other FY 1982 activity is directed toward the understanding of behavior of steel containment structures. This effort will be used for analytical support of the experiments, the selection of analytical methods for assessment of predictive capabilities, and the comparison of the analytical predictions with the experimental results. Also included in FY 1982 is the investigation of concrete containment behavior beyond the elastic limit of its steel reinforcement and of its steel prestressing tendons. This effort will be used for the design of reinforced concrete prototypical models that will be fabricated FY 1983, and for supporting analytical investigations. A small part of the FY 1982 activity is directed toward obtaining data from the Canadian experiment on a prototypical model of the CANDU containment and interfacing with the British on their proposed experiments on a model of the SNUPPs containment. It is anticipated that the British tests will be conducted in FY 1983.

5.8.5 Plan of Work As A Function of Time

Effort in FY 1984 will concentrate on three items. One is an evaluation of analytical predictions of steel containment capacity in light of the experimental results. The second is the conduct of tests to failure, under static overpressure, of models of reinforced concrete containments. Six tests are anticipated. All models will be approximately 1/10 the size of a typical containment. The first two models will be without penetrations or seismic reinforcement. They will serve as controls and will provide data for the evaluation of two-dimensional analytical predictions of post yield behavior. The next two models will include seismic reinforcement, but no penetrations. These models will provide additional data for the calibration of two-dimensional analyses. The final two models will include seismic reinforcement and penetrations and will provide data against which three-dimensional predictions can be compared. Finally, the comparison of predictive methods for prestressed concrete containment behavior against the British data will be completed.

The planning of dynamic, unsymmetric pressure tests will begin in FY 1984. Based on results from the hydrogen combustion program and results from the static pressure test series, dynamic pressure experiments for steel and concrete containment models will be designed. These experiments will be performed in FY 1985-1987.

The following results are anticipated during FY 1984-1987 and are shown graphically in Figure 5.9.

FY 1984 Comparisons of estimates of steel containment capacities with experiments under static pressure; acceptance criteria for seismic peripheral shear values under biaxial tension.

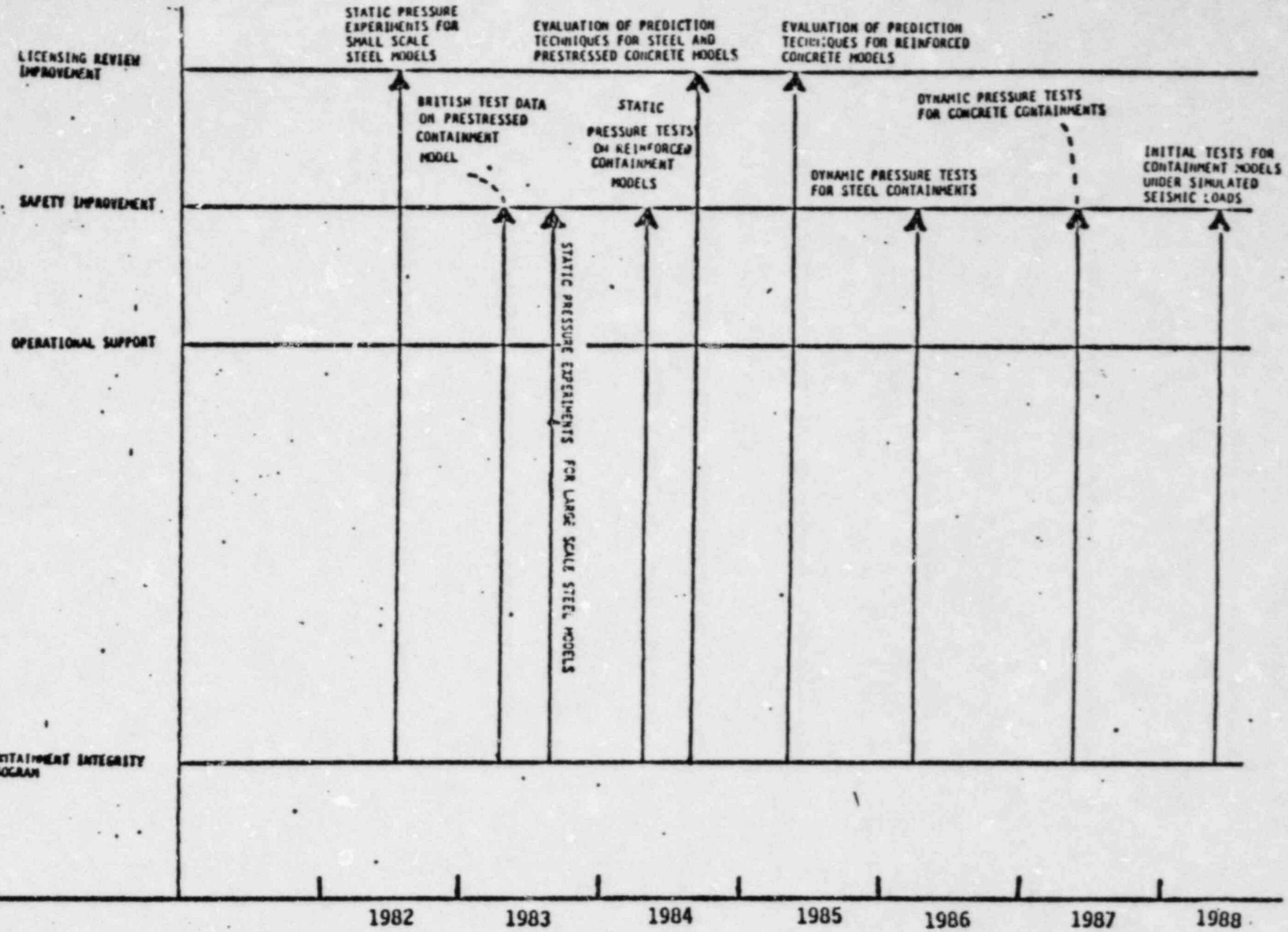

Figure 5.8 Containment Failure Mode

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FY 1985 Comparisons of predicted capacities for prestressed and reinforced concrete containments with experiments under static pressure.

FY 1986 Comparisons of predictions of steel containment capacity under dynamic pressure loads with experimental results.

FY 1987 Comparisons of predictions of capacity for reinforced and prestressed concrete containments under dynamic pressure loads with experimental results.

5.9 <u>Fission Product Release and Transport</u>

5.9.1 <u>Element Description</u>

Fission product release and transport research is directed at developing an experimental data base and models to predict the radiological source term for accident consequence assessment. This information is needed for emergency preparedness and for nuclear plant risk assessment studies, siting rulemaking actions, and for equipment qualification analysis. While a significant amount is known about fission product release and transport under controlled LOCA conditions, there are gaps in the data base relative to fission product release and transport behavior under severe core damage and core melt accident conditions.

Nuclear power reactor safety studies consistently indicate that the uncertainties associated with estimating fission product release and transport behavior are among the largest contributors to uncertainties in the risk to the public from severe accidents at nuclear power plants. This result is not surprising for two reasons: (1) offsite consequences are directly affected by the magnitude, timing, and makeup of the source term released from containment and (2) there are large uncertainties regarding the actual potential source term. The ultimate objective of this research program is to improve the quality of predictions of the potential fission product radiological source term released from containment under accident conditions.

5.9.2 Technical Issues Resolved by This Element

NUREG-0772 identified a number of key uncertainties related to estimating fission product source terms. The most important of these are:

1. Reactor Coolant Safety (RCS) aerosol and fission product behavior (experimental data for model verification),

2. RCS thermal/hydraulic conditions under core melt accident conditions,

3. Containment failure time, mode, and location (experimental data and analysis),

4. Fission product vapor phase and aqueous phase chemistry (experimental data),

5. Less volatile fission product, control material, and structural material aerosol formation rates (in-vessel and during interaction with concrete) (experimental data),

6. Aerosol behavior in condensing steam containment atmospheres (experimental data),

7. Removal of particulate fission products in water pools and ice beds (experimental data and models),

8. The effect of hydrogen combustion on fission product physical and chemical forms (experimental), and

9. Coupled models of containment fission product vapor transport, aerosol behavior, steam effects, and effects of ESFs.

The objective of these research programs is to develop a data base for assessing fission product release from the fuel and fission product transport behavior from the fuel to the environment. This research will focus on severe core damage and core melt accident conditions. The data base needs include information on the release of fission products and nonradioactive aerosols from overheated and melting fuel, the chemistry of the released fission products, aerosol formation mechanisms, the transport behavior of fission products and aerosols in the reactor coolant system and in the containment, and the effectiveness of engineered systems in mitigating fission product release under severe accident conditions.

5.9.3 Key Interfaces with Other Elements

Radiological source term and radiological source term analysis require definition of accident sequence characteristics. This need is supplied by probabilistic risk assessment studies (elements 5.1, 5.2, 5.10, 5.11) that define overall system performance and dominant accident sequences.

Fission product release and transport analysis also requires detailed information on the physical process which occur during severe accidents. Among the most important are:

1) RCS thermal hydraulic behavior,
2) Fuel heatup, melting, movement, etc. (Element 5.4)
3) Molten core/concrete interactions, (Element 5.6)
4) Molten fuel/coolant interactions, (Element 5.6)
5) Containment response to severe accident loads (i.e. failure time and mode) and (Element 5.8), and
6) Hydrogen combustion, (Element 5.5)

Major uses of the results of the fission product release and transport source term research are equipment qualification probabilistic risk assessment, definition of siting requirements, and emergency planning.

5.9.4 Background and Status

An intensive program to evaluate realistic source terms for severe LWR accident sequences was conducted during the Reactor Safety Study (RSS), WASH-1400. Because of the scarcity of applicable experimental data, large uncertainties were associated with the fission product release and transport assumptions included in the study. In fact, in certain areas, so little information was available that only bounding assumptions could be made (for example, fission product attentuation within the primary coolant system).

Beginning about 1975, several studies were initiated by the NRC to investigate the release of fission products from irradiated LWR fuel rods under severe accident conditions and to develop models for fission product transport behavior within the reactor coolant systems. These programs have provided (1) data on fission product escape from fuel rods under LOCA conditions in the temperature range 500°C to 1600°C and (2) a mechanistic model (TRAP-MELT) for fission product behavior within LWR primary coolant systems under severe accident conditions up to and including fuel meltdown.

During the reactor safety study, a relatively simple computer code (CORRAL) was developed to model the behavior of fission products in the containment atmosphere. The original CORRAL code had relatively detailed models for spray washout of iodine vapor species; however, the spray removal of particulate fission products and surface deposition of aerosols and vapor species were crudely modeled.

In the area of aerosol behavior within containment structures, significant progress has been made under the fast reactor program that is broadly applicable to all aerosol studies. Experimental programs to characterize the generation, agglomeration, and surface deposition rates of Na, UO_2, and Na/UO_2 aerosols have been conducted. The results of these experimental programs have formed the basis for a number of mechanistic aerosol behavior codes, including HAARM, ZONE, QUICK, MULTI-AEROS.

5.9.5 Plan of Work as a Function of Time

The following three sections describe specific research projects and near fuem results expected during FY 1982 and FY 1983. Figure 5-9 presents a detailed milestone schedule for these programs.

1. Fission Product Release. Research programs to investigate and quantify the release of fission products and aerosols from the fuel include:

   a. An experimental program to measure the release of fission products from commercially irradiated LWR fuel rod segments in a steam environment under elevated-temperature (1000°C-2600°C) accident conditions. First results at high temperature (2000°C) are scheduled for early FY 1982 with the higher-temperature tests (to 2600°C) to begin in early FY 1984.

   b. Experiments to investigate the release of fission products and structural material aerosols from larger bundles of fuel (.5 to 10kg) using simulated irradiated fuel (fissium) and out-of-pile heating technique (FY 1982-1983).

   c. Program to investigate the release of aerosols from molten pools of core materials interacting with reactor cavity concrete and with core retention material (ending in FY 1984).

d. Examination and analysis of samples of the TMI-2 core schedule will depend on the TMI-2 cleanup schedule.

e. Development and improvement of mechanistic models to predict the release of fission products from the fuel under accident conditions (FASTGRASS and START) during interactions of the damaged and molten fuel with residual coolant and plant structures.

f. Measurements of fission product release during Phase 1 severe fuel damage testing in the PBF reactor (FY 1982 and FY 1983).

2. Fission Product Transport

Research programs in the areas of fission product vapor and aerosol transport and deposition include:

a. Continued improvement of the TRAP-MELT code that models fission product behavior within the primary coolant system under severe accident conditions, and the coupling of the mechanistic, multi-compartment TRAP-MELT RCS code to models that predict containment fission product behavior and models for fission product (and aerosol) release from the core (ongoing, to be completed in FY 1984). Results from this program will be factored into the CONTAIN code.

b. An experimental and analytical program (thermodynamic calculations) to provide model development data for the TRAP-MELT code in the surfaces of elevated temperature fission product vapor pressures, surface deposition rates and mechanisms, and fission product chemical reactions with steam, prototypical surface materials, and other fission products (ongoing, to be completed in FY 1983, but may be extended).

c. Continuation of experimental and analytical programs to develop models for containment aerosol fission product behavior under severe accident conditions. The aerosol models will be incorporated into the TRAP-MELT, CORRAL, and the CONTAIN code to predict overall fission product transport behavior. These improved mechanistic codes will be used to benchmark simplier models in MELCOR. (To be completed in FY 1983.)

d. Modification and operation of a facility to test and verify the primary system fission product and aerosol transport codes. Tests on volatile fission product (e.g., cesium, iodine) transport will be initiated in FY 1983 and completed in FY 1984.

e. An experimental program to investigate the chemistry of various fission product species (various forms of iodine and tellurium) in aqueous reactor solutions and their liquid/vapor phase distribution under representative accident conditions.

3. Fission Product Control

Programs are planned to investigate and quantify the effectiveness of various engineered safety and mitigation features in reducing the potential fission product escape from containment. Within this area are programs to:

a. Investigate and quantify the radioiodine retention performance of impregnated activated charcoal adsorbers under accident conditions (completed in FY 1982).

b. Research on the fission product mitigation performance of engineered safety features (e.g., containment spray systems, suppression pools, ice condenser beds) under the radiological and evnironmental conditions predicted for severe core damage and core melt accidents.

c. Research on the effects of large aerosol sources (predicted for the most severe accidents) on the performance of these engineered safety features.

4. NUREG-0772 Follow-on Research

Development of updated, severe accident, release-from-plant, fission product source terms to supplement WASH-1400 estimates (completed in FY 1983).

Development of quantitative estimates of the uncertainties associated with these source term predictions and identification of the major sources of the uncertainty. (Completed inFY 1983.)

Analysis of past reactor accidents and core destructive tests for insights into fission product release and transport behavior and to compare current assumptions and models with measured releases (completed in FY 1982).

5. Longer-Term Research Program Plan (FY 1984-1988)

Fission Product Release From Overheated Fuel - Beginning in FY 1984, tests will be initiated in the high temperature fission product release program to investigate the release of fission products and aerosols from commercially irradiated fuel in the temperature range 2000°C to approximately 2600°C. The test apparatus will include techniques (laser Raman spectroscopy) for direct in situ determination of fission product chemical form. Two test series will be conducted in FY 1984, three in FY 1985, three in FY 1986, and two in FY 1987.

Reactor Coolant System (RCS) Fission Product and Aerosol Transport Tests - The tests on RCS fission product and aerosol transport will continue through FY 1985 and perhaps into FY 1986. In FY 1985 this experimental program will focus on determining the transport behavior of high density aerosols within the RCS. Tentative plans call for tests with up to 800 kg of prototypic core-melt aerosol materials.

Fission Product Transport Code (TRAP-MELT) Development - Pretest and post-test analyses of the RCS tests discussed above will be conducted with the TRAP-MELT code. Code predictions and experimental results will be compared

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and model improvements initiated (if necessary) to correct difficiencies in the code. These analyses and model development activities should continue through FY 1986. At the end of FY 1986, the TRAP-MELT code will have been tested and validated by comparison with these large-scale integral tests.

Similar analysis will be performed using the extended TRAP-MELT code, the CONTAIN code, and/or the MELCORR/MATADOR code on planned large-scale containment fission product and aerosol tests. Again these analyses should be completed by FY 1986.

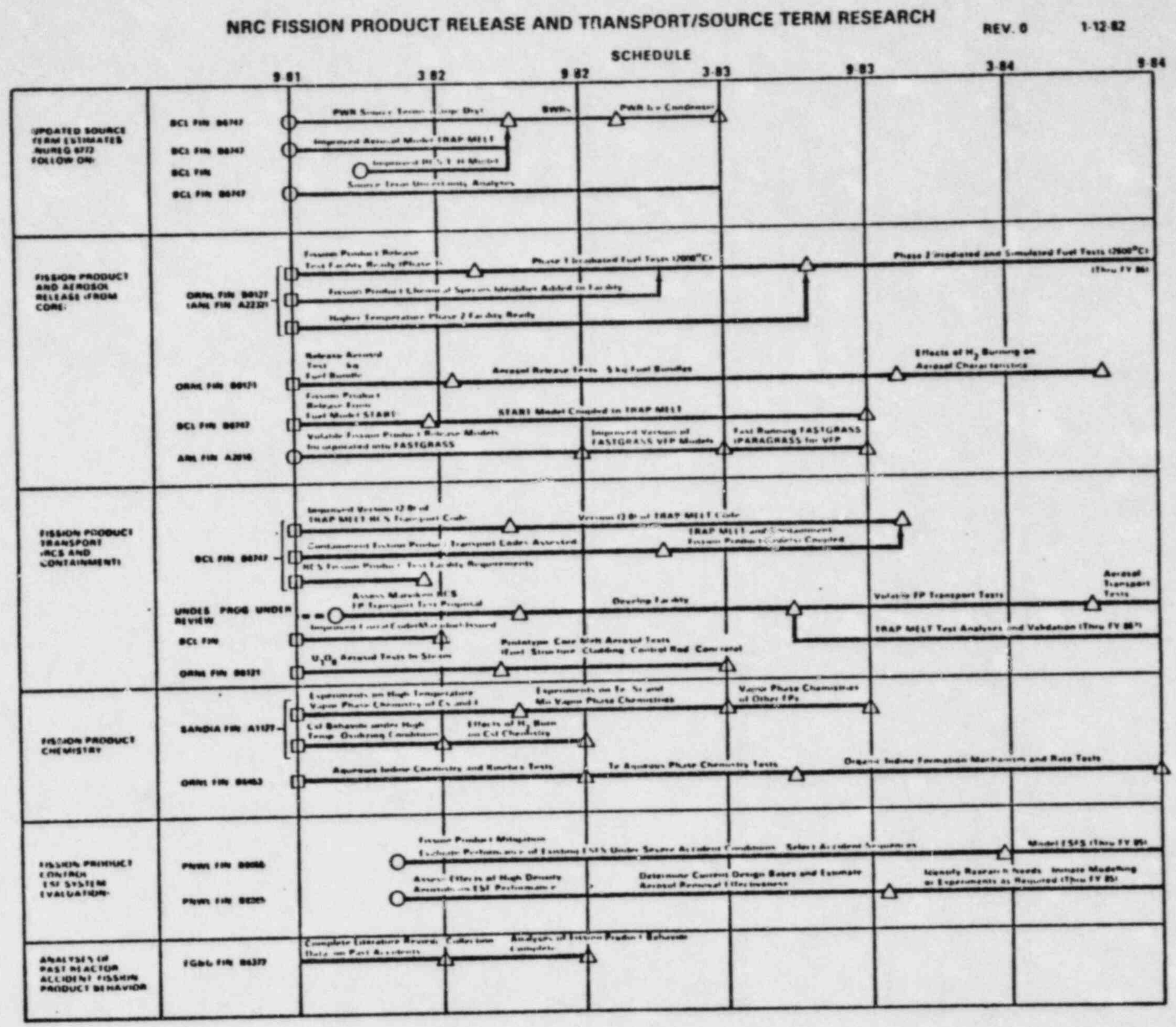

Figure 5.9 Fission Product Release and Transport

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5.10 Risk Code Development

5.10.1 Element Description

This element relates to the development of computer codes for use in PRA to analyze the phenomenological processes associated with severe accidents. Because of the need in PRA studies for the analysis of many accident sequences, these codes are to have the characteristics of being relatively simplistic and fast running. They will thus be the more approximate and quick counterparts to the more mechanistic codes being developed in parallel in other decision units. Two generations of codes are to be developed in this element; each is described below.

1. MARCH-2/MATADOR Development Program

   The MARCH-2/MATADOR development program has as its objective the short-term modification of the present code versions in order to correct known important limitations of the codes. Because of the need for improved codes on the short-time schedule of this research and other regulatory matters (e.g., plant operating license reviews), this code development will improve particular aspects of the code, but will not attempt to alter its code structure.

2. MELCORR Development Program

   Because it has been recognized that the various physical process codes used in PRA have a number of shortcomings (even with the development of MARCH-2 and MATADOR), a program to develop a unified risk code (MELCORR) is being undertaken. This effort is being initiated to correct known problems in codes such as MARCH, CORRAL/ MATADOR, and CRAC; to develop a common coding structure for ease of understanding and maintainability of the codes; and to provide the capability for unification of the codes for purposes of complete "best-estimate" consequence calculations (from initiating event to health effects and property damage) and for purposes of complete, more rigorous uncertainty analyses. The MELCORR code will thus supplement and improve upon the individual MARCH, CORRAL/ MATADOR, and CRAC codes now used in PRA.

5.10.2 Technical Issues Resolved by This Element

The MARCH and MATADOR codes have both undergone extensive review and have had a number of deficiencies identified within them. Such deficiencies have resulted in some instances for the need for numerous additional supporting calculations, sensitivity studies, etc. In the MARCH-2/MATADOR program, the more important and more readily resolvable deficiencies will be accounted for and thus will result in risk codes having additional credibility and requiring relatively fewer supplemental analyses. The longer-term MELCORR program will be used to develop a risk code structure that is much more readily adaptable. This code then would be used to incorporate new models and experimental data as they become available from elements in other decision units.

5.10.3 Key Interfaces with Other Elements

This code development will use results from various experimental programs in elements 5.4 through 5.11, during and for MELCORR development, and will provide improved capability for consequence prediction in the elements "Accident Consequence and Risk Reevaluation," and "Risk Reduction and Cost Analysis."

5.10.4 Background and Status

The risk codes now in use (MARCH 1.1, CORRAL 2, CRAC 2) had their origins in the analyses performed for the Reactor Safety Study (RSS). Following the release of the RSS, work was initiated to improve upon the initial code versions; CORRAL-2 thus became available in 1977 and MARCH 1.1 and CRAC 2 in 1981. Since 1977, work has also been under way to upgrade CORRAL-2 to account for new supporting data, to add models of certain phenomena not previously included, and to make its structure more amenable to modification. This work has led to a preliminary version of the code, now renamed "MATADOR."

5.10.5 Plan of Work as a Function of Time

Working versions of MARCH-2 and MATADOR are scheduled to be completed in July 1982 with public release by the end of FY 1982 A working version of MELCORR is planned to be available late in FY 1983, with a final version available in mid FY 1985. Figure 5-10 shows this schedule and its relationship to other research elements.

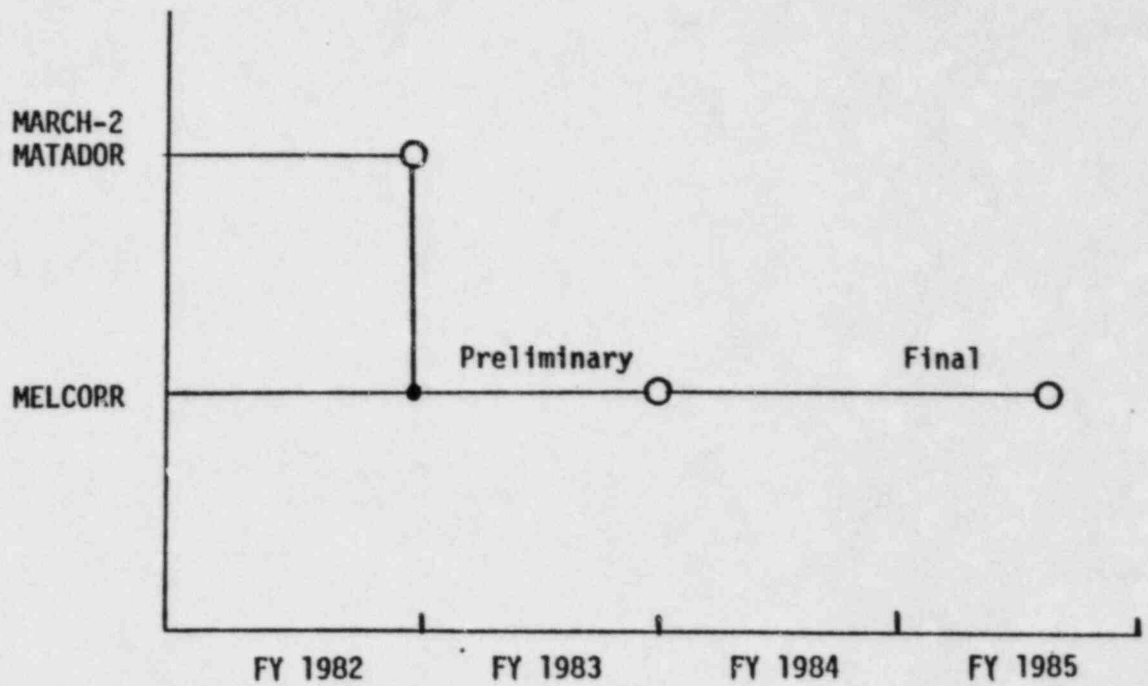

Figure 5.10 Risk Codes

5.11 Accident Consequence And Risk Reevaluation

This element relates to the application of advanced versions of risk codes for the reanalysis of the consequences of important accident sequences. That is, as the severe accident physical process codes discussed above are developed, they will be put to use in this element to reanalyze the consequences of accident sequences determined to be important in previous risk studies and the "Accident Likelihood Analysis" element. Further, as these consequence analyses are completed, they are to be combined in this element with the sequence likelihood results, resulting in the redefinition of the risk of studied plants. In this way, previously completed risk studies can be periodically updated to reflect the latest advances in accident likelihood and consequence analysis.

5.11.2 Technical Issues Resolved by This Element

The principal technical issue being resolved in this element is that of the provision of a periodically updated set of consequence and risk analyses through which best estimates of actual levels of plant risk can be determined. Such determinations of plant risk levels are important both for consideration with respect to possible safety goals and as a basis for assessing the risk reduction benefit of possible plant modifications.

5.11.3 Key Interfaces with Other Elements

This element has important interfaces with the "Accident Likelihood Analysis" and "Risk Code Development," elements drawing from them information on accident sequence likelihoods and improved analytical capabilities, respectively. In addition, the results of this element are an important input to the "Risk Reduction and Cost Analysis," element providing the periodically updated risk reevaluations described above. It also provides input to the "Regulatory Analysis and Standards Development" element by providing further evidence on the effectiveness of current requirements at limiting risk.

5.11.4 Background and Status

Present-day use of PRA in regulatory decision making relies heavily on risk studies such as the Reactor Safety Study and RSSMAP. Such PRAs do not fully represent and account for many issues that have arisen since their analyses were performed. Examples of such issues include the matters of possible reductions in fission product source terms in some accident sequences; potentially conservative treatment of certain containment threats (i.e., the "steam spike"); and the potentially important likelihood of reactor vessel failure due to pressurized thermal shock. Up until the present time, no formal mechanism for incorporating such matters into previously completed and much-used PRAs has been available. With the recent initiation of accident likelihood analysis (described above) and risk code updating (described below), the work of this element is intended to become such a mechanism.

5.11.5 Plan of Work as a Function of Time

It is planned that the performance of these consequence and risk reevaluations will be performed iteratively and at roughly one-year intervals. This schedule and its interrelationships with other elements is shown in Figure 5-11.

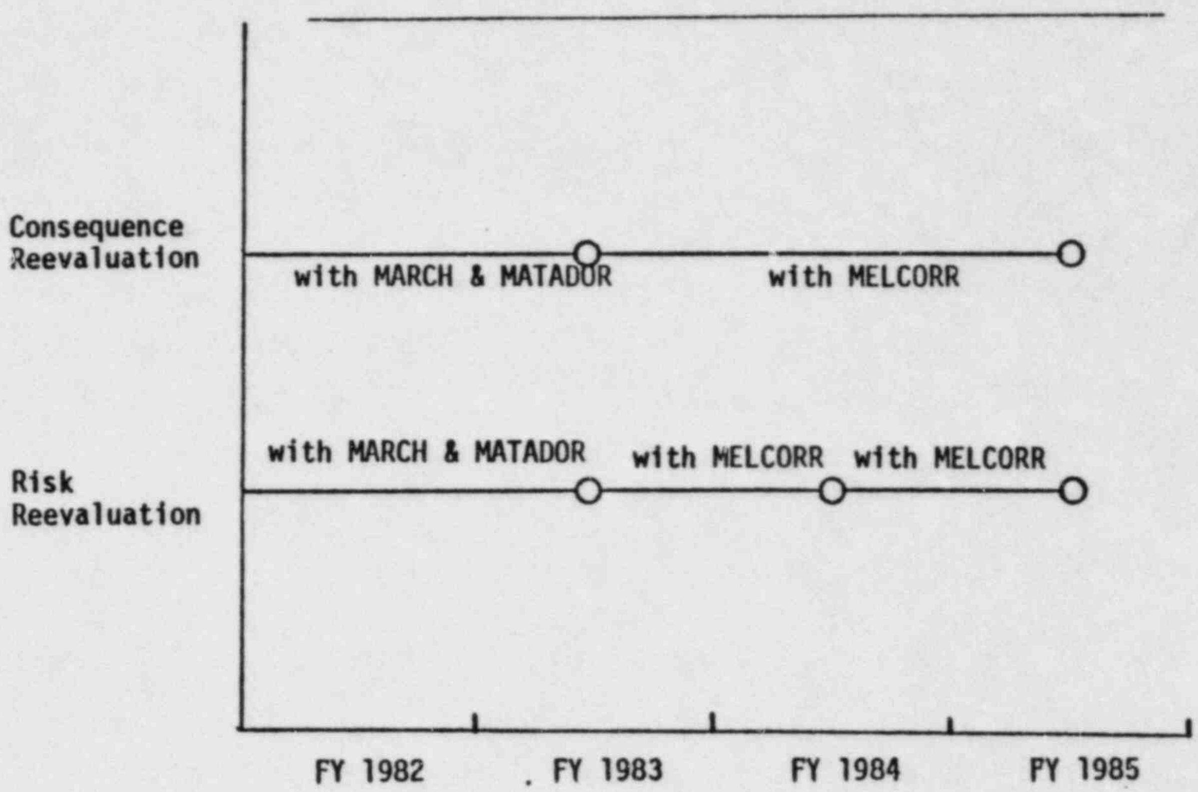

Figure 5.11 Accident Consequence and Risk Evaluation

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5.12 Risk Reduction And Cost Analysis

5.12.1 Element Description

In this element, analyses are to be performed of the risk reduction potential and costs associated with a spectrum of possible plant modifications. Included in these possible modifications are, for example, filtered-vent containment systems, alternate shutdown heat removal systems, and stronger containments. The objective of such analyses is to identify those modifications that appear to present the most cost-effective risk reduction. Since such results will vary with the specific plant design being considered, analyses are to be performed for all major design types (PWR large dry and ice condenser containments and BWR Mark I, II, and III designs).

5.12.2 Technical Issues Being Resolved by This Element

The principal technical issue being resolved in this element is the identification of those possible plant modifications that offer the most cost-effective means of reducing risk for the set of major LWR design types.

5.12.3 Key Interfaces with Other Elements

Important information for this element is to be obtained from the "Risk Code Development," and "Accident Consequence and Risk Reevaluation" elements "Risk Reduction of Add-Ons." The first element will provide the analytical models for use in the risk reduction studies, while the second element will provide updated levels of plant risk to be used as benchmarks for the risk reduction studies. The third element will provide reasonable information on cost for potential plant modification.

5.12.4 Background and Status

In 1978, the NRC performed studies at the request of Congress to identify those areas of LWR design that appeared to offer the greatest potential for improving the safety of these plants. The results of these studies are reported in NUREG-0438, "Plan for Research to Improve the Safety of Light-Water Nuclear Power Plants" In this report, two possible plant modifications were identified as having the most promise for improving safety, these being filtered-vent containment systems and alternate shutdown heat removal systems. As a result, programs to investigate this potential in more detail were initiated in 1979 and have continued up to the present time. In 1981, it was decided to link these programs with a new program that would study these modifications in concert with studies of a somewhat broader spectrum of possible changes. This program is discussed below.

5.12.5 Plan of Work as a Function of Time

This program has two general objectives (1) to develop conceptual designs of a set of possible plant modifications for preventing or mitigating severe accidents, and (2) to perform analysis of the risk reduction value and cost of these possible additional plant features, for comparison with developed reevaluations of LWR risk.

This program will be performed in an iterative manner. In the program's first phase, a broad spectrum of possible severe accident prevention and mitigation features will be screened using semi-quantitative techniques to develop a small set of the apparently most promising features. Following this, conceptual designs will be developed for these possible plant modifications. These designs will be developed only to the degree necessary to permit rough evaluations of their reliability and costs. To fulfill the second objective, the risk reduction value of these designs will then be evaluated, using MARCH-2 initially, to be followed later by the use of MELCOR. For each design, this evaluation will include estimates of its capability to reduce either the likelihood or consequences of a severe accident and the potential risk increase associated with such matters as spurious operation. This assessment of the net risk reduction value will then be combined with cost estimates to determine the values and impacts of the various features.

5.12.5 Plan of Work as a Function of Time

The details of the schedules of these programs and their relationships to other elements are shown in Figure 5-12.

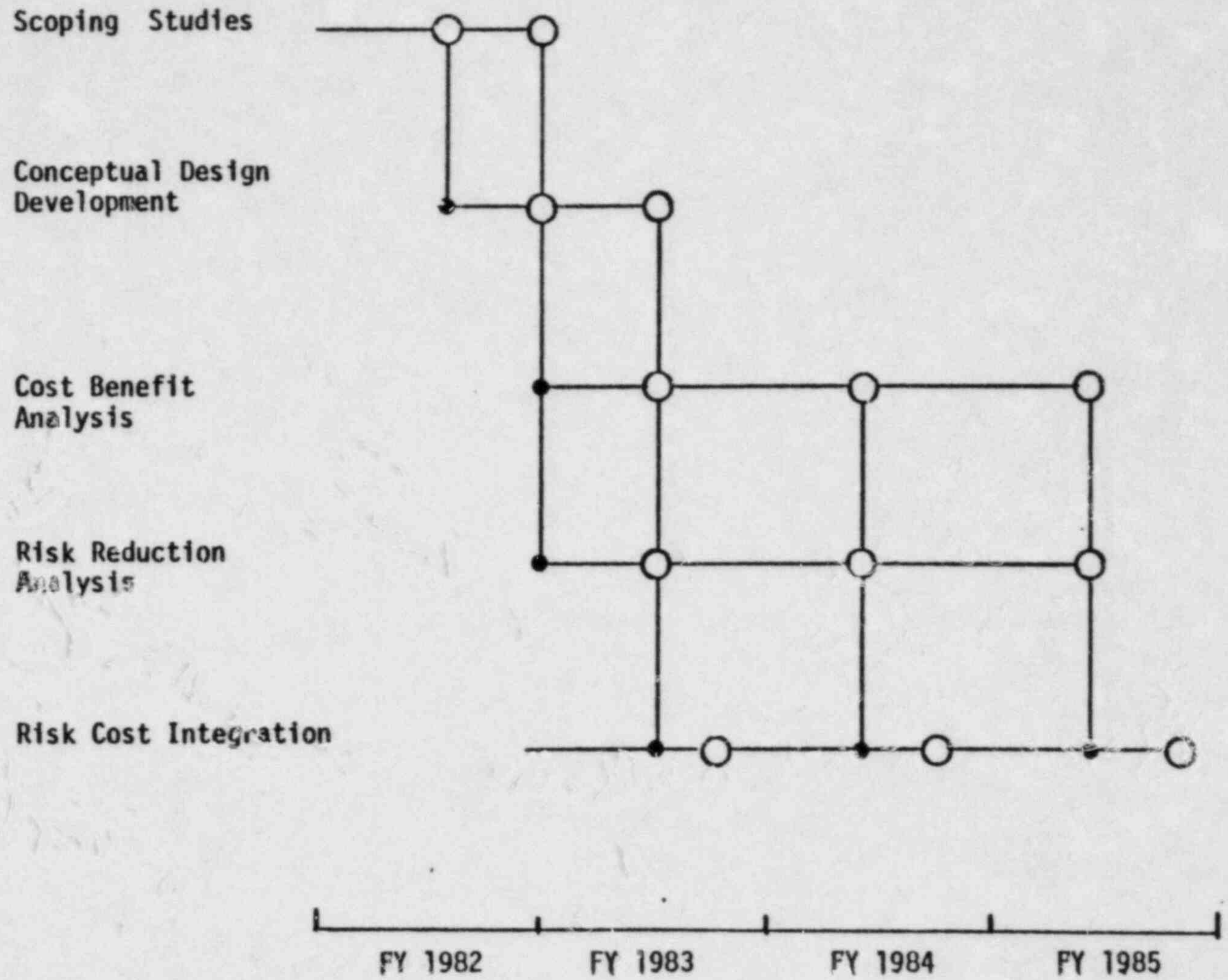

Figure 5.12 Risk Reduction and Add-On Cost Benefit

5.13 Evaluation of Accident Mitigation Systems

5.13.1 Element Description

The purpose of this element is to make a value-impact study of additional accident mitigation systems that are identified as having potential for significant risk reduction by the risk evaluation and reduction studies made in other elements of the program. Sufficient engineering design studies will be completed on mitigation systems to make cost estimates for use in cost-benefit analysis.

5.13.2 Technical Issues Resolved by This Element

Probabilistic risk evaluation and risk reduction elements of the program have identified or may identify mitigation systems or devices that have potential for substantial risk reduction such as filtered vented containment systems (FVCS) and core-melt mitigation systems. This element will do sufficient engineering design on these concepts to establish technical feasibility (i.e., can it really be added and will it work?) and make a firm cost estimate so that a value-impact evaluation will be credible.

5.13.3 Key Interfaces with Other Elements

The key interfaces with these elements are the Accident Consequence and Risk Evaluation and Risk Reduction add-on elements. These elements will identify conceptual designs to be evaluated for feasibility and cost effectiveness.

5.13.4 Background and Status

A small study on mitigation systems for the Sequoyah Unit 1 ice condenser PWR has been made at INEL to evaluate hydrogen control concepts. The final report is under preparation. Phase 1 of a similar study has been completed at INEL.

5.13.5 Plan of Work as a Function of Time

The feasibility study and cost estimate of adding a FVCS to an ice condenser PWR and to Mode I, II, and III type BWR containments will be completed at the end of FY 1984. Alternative systems for containment cooling will be evaluated. This study will be completed at the end of FY 1984.

Backfit studies to evaluate the feasibility and cost of increasing containment size by interconnection to external volumes will be completed in FY 1985.

The risk reduction add-on element of program is expected to develop a standardized add-on concept for improving safety. The feasibility and costs for these systems will be evaluated by the end of FY 1985. If these concepts prove to be feasible and cost effective, a set of design criteria for these standardized add-ons will be developed by the end of FY 1987.

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PROGRAM ELEMENT 13 - EVALUATION OF ACCIDENT MITIGATION SYSTEMS

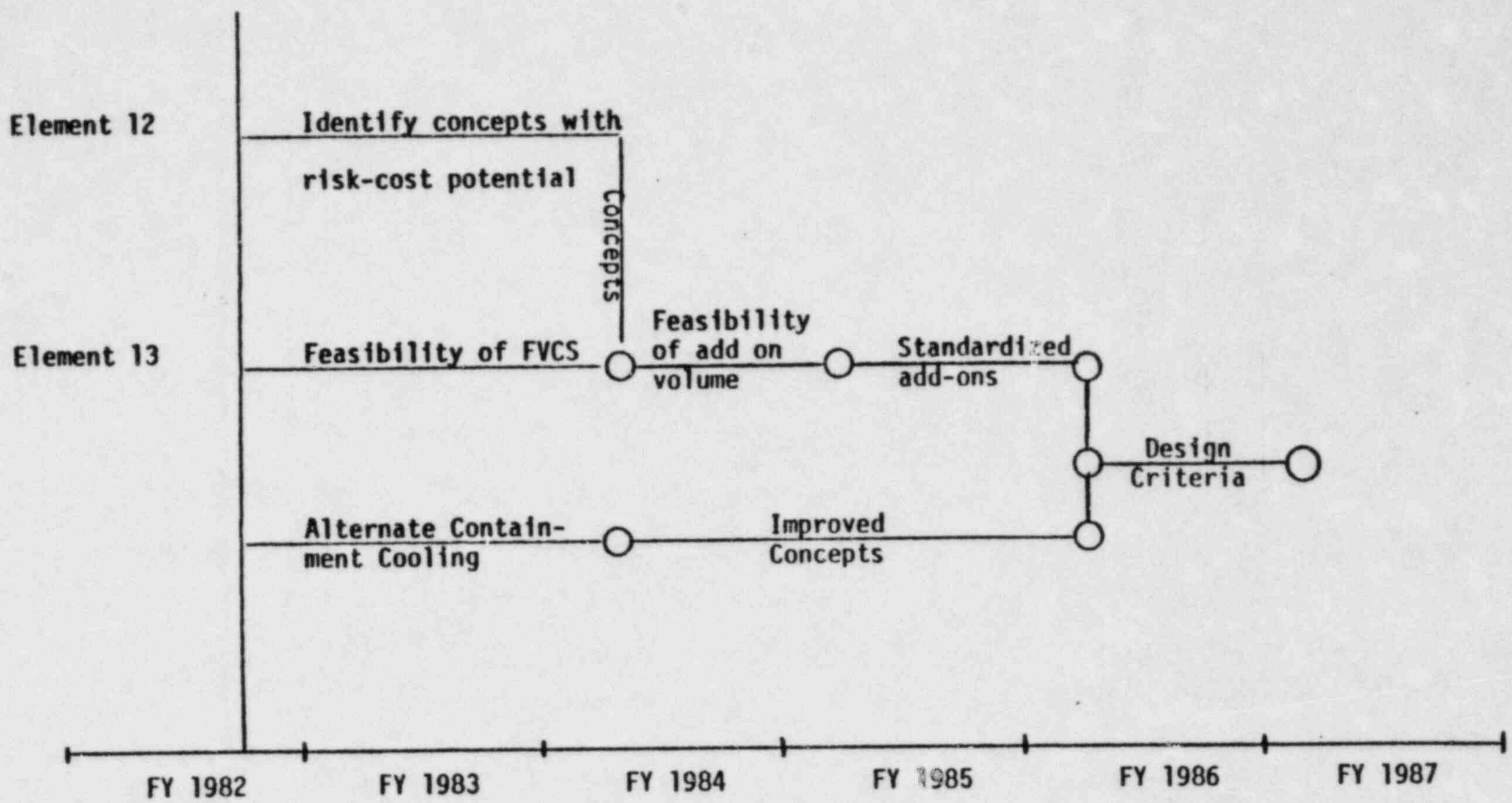

Figure 5.13 Add-On System Evaluation

5.14 Regulatory Analysis and Standards Development

5.14.1 Element Description

A variety of research and standards development projects are under way or planned that support reactor safety standards development and the development of staff aids and guides for use in the arena of severe accident risks. These include:

1. Development of value-impact or cost benefit guides and improved techniques for use by the staff in evaluating new requirements,

2. Risk assessment sensitivity analyses to catalog the ways in which nuclear power plants might pose severe accident risks well in excess of those in the Commission's safety goals or suggested by published reactor PRAs,

3. Studies of the risk-limitation effectiveness of the General Design Criteria, the Regulatory Guides, and the Standard Review Plan,

4. Studies of the reliability with which high risk designs or procedures are identified and corrected by safety evaluation, QA, inspection and enforcement, industry safety practices, and experience feedback systems,

5. Studies of the limitation, implications, and ways of implementing the Commission safety or safety goals policy,

6. Analysis of reliability assurance practices in non nuclear industries and adaptation of the more promising methods to the nuclear regulatory arena,

7. Systematic review of the technical content, interdependence, and institutional forces acting on the current body of reactor safety standards, and

8. Development of draft rules, regulatory guides, review plans, and aids to regulatory decisionmaking.

5.14.2 Technical Issues Resolved by This Element

The current body of reactor safety standards limit the risk posed by severe accidents through the mechanism of conservative reactor design, the postulation of design basis accidents (from which most safety systems derive their principal safety design bases), and through a variety of regulations intended to ensure safety system reliability (the single failure criterion, QA, technical specifications, conservative deterministic codes and standards, etc.). It is now widely recognized that accidents different in character or far more severe than the design basis accidents are credible. The more probable causes of such accidents are believed to originate in multiple failures or human errors outside the domain of the single failure criterion or of those common-cause failure mechanisms currently addressed in the regulations (seismic qualification, safeguards, fire protection, etc.).

Most of the purely technical issues involved in resolving these regulatory issues are dealt with in the many other research elements described in this plan. However, the regulatory issues and the interface with the technical issues embracing the adequacy of the design basis accidents and the adequacy of the several ways of assuring safety system reliability, are the subject of this element.

5.14.3 Key Interfaces with Other Elements

The work of this element is closely coupled (both input and output) with policy standards development initiatives at the Commission and NRR, including: (1) safety goals and NRR implementation strategies; (2) plans for the resolution of Unresolved Safety Issues; (3) requirements for plants at high-population density sites such as Indian Point and Zion; (4) requirements for near-term CP's; and (5) requirements for standard plants or licenses to manufacture.

The element will draw from three other elements of the plan: (1) Accident Likelihood Analysis; (2) Accident Consequence and Risk Reevaluation; and (3) Risk Reduction and Cost Analysis. The draft standards emerging from the element will be the principal product of the accident research program apart from the codes, data, insights, and reports issued by the many other elements.

5.14.4 Background and Status

A number of severe accident related rule initiatives have already been undertaken. These include the final and proposed rules for the control of hydrogen during severe accidents and the proposed rules for dealing with Anticipated Transients

Without Scram (ATWS). A notable change in regulatory approach is the recently issued final rule for near term construction permits and manufacturing licenses, which explicitly requires containment modifications for dealing with severe accident forces and requires the use of PRA as a design evaluation tool.

Research to address the DC power Unresolved Safety Issue has been completed and has resulted in the formulation of new proposed regulations. Comparable research on the Station Blackout issue, Alternate Decay Heat Removal Systems, Alternate Containment Concepts, and Core Catchers are also dealing with issues involving design criteria and regulatory formulation.

Research into the consequences of severe reactor accidents is currently being formulated into a guide and a handbook for value-impact assessment to enable a value to be placed on regulatory initiatives that reduce the frequency or severity of severe accidents. Tables of the present worth of projected losses for accidents characterized by user-input frequency and severity classification will be published in the late spring of 1982. A first, partial edition of the value-impact handbook will appear near the end of FY 1982. More comprehensive editions will follow at intervals thereafter.

A "reverse" risk assessment a sensitivity analysis for light water reactors will be published in FY 1983. It will employ current PRA results, together with state-of-the-art analysis of the limitations of PRA, to give a systematic answer to the question, "If a nuclear plant were to pose severe accident risks greatly in excess of the Commission's safety goals, how might this come about?"

It will attempt a comprehensive catalog of the uncertainties in accident likelihood and consequence estimation through which very high risks might have escaped discovery.

A study of the risk-relevance and effort of implementation of the pre-TMI Standard Review Plan has been completed in draft form. It demonstrates that studies of the risk-limitation effectiveness of regulations are feasible. A new effort to examine the risk-limitation effectiveness of the General Design Criteria is scheduled for FY 1982. It will be followed by comparable studies of other parts of 10 CFR 50, the Regulatory Guides, and the current Standard Review Plan at six month intervals thereafter. The study will be similar to that done previously to rank the 133 generic safety issues in 1978. The basis will be the many reactor PRAs and safety system reliability studies that have been and are currently being published. These contain many findings about the comparative importance to safety of system design features and operations practices. These studies are expected to identify and document the evidence supporting inferences of unnecessary over-regulation as well as cases of safety-significant loopholes in the regulations.

The studies of the risk-limitation effectiveness of the regulations will be followed by a study of the ways in which industry compliance practices and NRC inspection and enforcement, and the experience feedback mechanisms of the industry and of the NRC detect and correct safety-significant defects in design and plant operations. The risk-based sutides of nuclear plants, particularly the reverse risk assessment, will provide many clues about the ways safety might be compromised by poor design choices, provisions for test and maintenance, and the conduct of operations. These will be assembled into an analysis of the reliability with which such flaws are detected and corrected. The resulting findings will be classified and proposals made for regulatory initiatives to deal with any prominant weakspots in the fabric of safety assurance practices. This work is scheduled for FY 1984.

There has been research into the technical bases of quantitative safety goals for some time. At this writing, the Commission is nearing the publication of a safety policy statement including qualitative and quantitative safety goals. A technical analysis of the implications, strengths and weaknesses of the Commission goals will be prepared in the spring of 1982. Additional studies of implementation strategies willl be performed in late FY 1982. The impact of the safety policy on severe accident regulatory policy will be studied in detail.

A program commenced in FY 1982 to explore the reliability assurance management practices and reliability engineering techniques developed in other industries for possible application to nuclear safety assurance. The aerospace, weapons, and electronics industries have pioneered management and technical analysis techniques to assure the reliability of complex systems. Probabilistic system reliability analysis was originally invented in these industries. Many of their techniques have never been tried in the arena of reactor safety, however. The FAA has adapted many of these techniques in a regulatory arena. These approaches to safety assurance or reliability assurance will be studied to identify, if they can, help to sharpen the focus of reactor safety assurance requirements to establish a higher level of risk-limitation effectiveness while avoiding unnecessary overregulation.

As the foregoing research programs develop a picture of the needed or desirable changes in the reactor safety regulations, an effort will be made to propose optional rules, regulatory guides, and review practices. The advantages and disadvantages, values and impacts, cost and benefit will be documented for each formulation. Supportive analyses of the interdependence of the regulations and of the impact of implementing changes will be performed to provide a comprehensive basis for agency and public consideration of the pros and cons of each approach.

5.15.5 Plan of Work as a Function of Time

The schedule for this element can been seen in the attached figure.

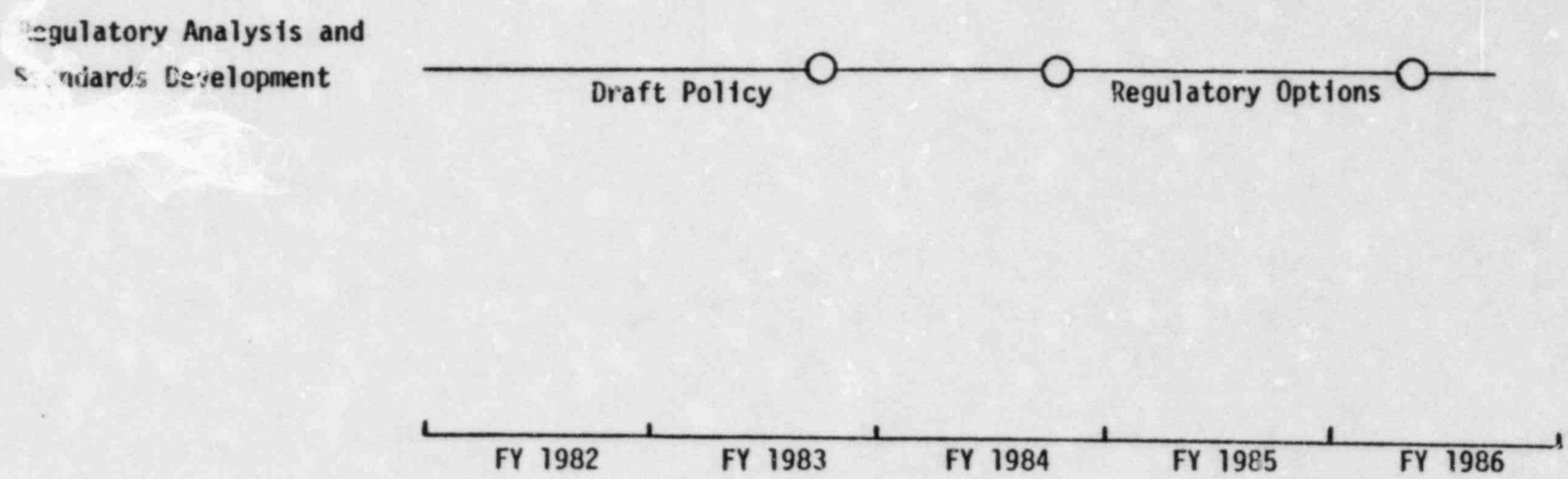

Figure 5.14 Regulatory Analysis and Standards Development

6.0 CONCLUSIONS

The plan presented in this report is of an integrated program of research to establish a data base for policy and regulatory decisions regarding the treatment of accidents in nuclear power plants. The plan treats the aspects of Accident Prevention, Accident Management, and Accident Mitigation. In covering these aspects, the plan has sought to be consistent with the Commission's long-established policy of defense-in-depth. The need for a tightly integrated research program grows out of the wide range of uncertainties in the phenomena and methodology, as revealed by casework, most specifically casework on the Zion and Indian Point plants That is, the integrated nature of the plan is as much a consequence of logical necessity as it is of policy. This integrated nature poses some pitfalls as well as advantages.

The advantages stem from the finding that, if an even pace is maintained among the various program elements, then interim results can be used to guide the development of regulatory decisions and to focus the program more closely on the needed technical information. Thus,some of the fruits of the program will be available in about 18 months so that the program will have the feature of being self-correcting, contracting in scope, rather than expanding. Significant results are forecast by the end of 1983 with completion of most major tasks by 1985.

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

OFFICE OF THE SECRETARY

January 7, 1982

APPENDIX XI
FY 1983-87 POLICY AND PLANNING GUIDANCE

MEMORANDUM FOR: William J. Dircks, Executive Director for Operations

FROM: Samuel J. Chilk, Secretary

SUBJECT: FY 1983-87 POLICY AND PLANNING GUIDANCE

The Commission has approved the attached Policy and Planning Guidance document. The purpose of the document is to provide guidance to the staff for establishing priorities and for improving the regulatory process. Guidance with respect to each and every activity within NRC is not furnished, since it is not intended that the document be all inclusive. Programming and budgeting within the agency should be consistent with this guidance unless it is specifically superseded or amended.

Attachment:
As Stated

cc: Chairman Palladino
Commissioner Gilinsky
Commissioner Bradford
Commissioner Ahearne
Commissioner Roberts
OPE
OGC

Policy and Planning Guidance

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POLICY AND PLANNING GUIDANCE

INTRODUCTION

The purpose of this document is to provide guidance to the staff for establishing priorities and for improving the regulatory process -- starting immediately. It is therefore more than just a document to be used for preparing the FY 84-86 budget request. It is management guidance intended to focus on specific areas where the Commission believes additional emphasis is required.

Guidance with respect to each and every activity within NRC is not furnished, since it is not intended that the document be all inclusive. However, this should not be perceived as a Commission belief that other areas are not equally as important to protecting the public health and safety. Many of these other areas have effective ongoing programs where major problems do not exist but adequate management attention and initiative are still important. Although specific policy and planning guidance is not provided in this document, top management attention is still required for radiation health effects protection, fuel cycle licensing and inspection, materials licensing and inspection, and facets of emergency planning and preparedness not addressed here. These and other support functions are vital to accomplishment of the overall agency mission and objectives.

The document is organized in terms of seven major themes: Safe Operation of Licensed Plants; Near-Term Licensing Problems and Responses; Coordinating Regulatory Requirements; Improving the Licensing Process; Supporting Initiatives in Nuclear Waste and the Cleanup of Three Mile Island; Improving Related Regulatory Tools; and Other Policies. The policy section is intended to establish a general framework for NRC managers to shape their own particular programs. Planning guidance is furnished in those areas where the Commission believes more detail is warranted to meet specific concerns about schedules and priorities or where major assumptions are needed for program development.

Specific guidance involving programs will be provided by the Executive Director for Operations. The EDO will also develop and provide a management system for the Commission to keep track of the major 1982 program accomplishments and resource expenditures that support this policy and planning guidance.

It is the Commission's intention that nuclear regulation reflect a continuing commitment to come to grips with the reality of nuclear technology and of its relationship to those who control it, to those who work with it; to those who live near it, and to the general public. This commitment requires not only an open and effective approach within the agency, but an approach to the public (including the regulated public) that permits more efficient decisionmaking. As part of this process, the Commission must state its basic assumptions and criteria clearly, amend them when the facts so require, and live by them consistently and forthrightly in all activities.

To carry out the policy of the Commission will require the dedicated effort of all employees as well as the effective and efficient use of all NRC resources. Innovative, attentive and responsive management effort will be required to accomplish the Commission's goals.

NRC's greatest resource is its employees. Retention of our most creative and productive talent and the recruitment of new personnel with fresh insights and perspectives should be a management priority. To maintain a highly qualified and informed staff, the Commission's most creative and productive employees should be recognized and provided further opportunity for development. Increased effort should be expended, in the face of highly competitive conditions, to hire the best qualified individuals essential to the future ability of the NRC to carry out its regulatory responsibilities.

SAFE OPERATION OF LICENSED PLANTS

Policy

A. NRC's fundamental task is to make sure that existing plants and those coming on-line operate safely. To this end, the highest priority will be given to assuring that operating facilities achieve and maintain adequate levels of protection of public health and safety.

Planning Guidance

1. NRC on-site inspection of operating reactors will focus directly on the activities and operations of licensees, licensee contractors, and vendors. NRC will improve its own capabilities for independent and confirmatory measurements. The analysis of operational data and systematic assessment of licensee performance will be used to help focus inspections and to allocate inspection resources.

2. The NRC and the industry must continue to learn the lessons that only experience can teach. Efforts to collect, analyze, disseminate, and act upon operational data relevant to the safe operation of major licensed facilities must continue to receive priority attention. The framing of effective regulations must be based on a close study of operating experience.

3. NRC will continue to operate and improve, as needed, a Licensee Event Reporting (LER) system. NRC should continue to work with the Institute of Nuclear Power Operations (INPO) in its operation and development of the Nuclear Plant Reliability Data System (NPRDS). NRC should continue to support INPO in the operation of an industry-wide screening service to identify LERs and other operating experiences of significance to nuclear power plant licensees.

The NRC must develop a long range human factors program plan by mid 1982. INPO and the NRC both have programs for developing standards and requirements in the human factors area. These programs should be coordinated. In some areas it may be sensible to conduct activities in parallel and in others it may be appropriate to drop the NRC program should INPO's efforts be acceptable.

(a) The NRC should make maximum use of available human factors data.

(b) Alternative approaches exist for resolving certain human factors concerns, e.g., in the operator licensing area either the NRC or its contractors could administer examinations or individuals in the industry approved by NRC could undertake this activity. Where fundamentally different approaches are possible, the staff should prepare policy papers as soon as practicable for Commission consideration which recommend a course of action.

(c) The staff should continue to evaluate and improve the licensing and training requirements for reactor operators. Resource efficient methods should be pursued which will provide improved initial and requalification testing of operators.

(d) NRC should require key licensee employees including certain management and maintenance personnel to be adequately qualified. NRC will examine the possible need for qualifying these key employees and working with industry (e.g., INPO) to develop some form of licensing or certification.

5. Utility management performance needs to be evaluated to assure the quality of operation of nuclear facilities. Utility managers and supervisors as well as operators should be encouraged to improve their ability to promptly diagnose and deal with off-normal conditions.

6. As applied to operating reactors, the goal of NRC's enforcement program will be to assure compliance with NRC regulations and license conditions and to use experience gained from application of the enforcement policy to evaluate and revise the policy and make it more effective. For licensees who do not comply with NRC requirements, prompt and vigorous action will be taken; a licensee must not benefit by violating NRC requirements. Licensees who cannot achieve and maintain an adequate level of protection of public health and safety will not be permitted to operate.

7. The Commission supports the systematic evaluation program for operating reactors. The program should continue at its current pace. The goals and objectives of the program should be met expeditiously.

8. The staff should expedite the assessment of pressurized thermal shock, so that the Commission can consider if actions need to be taken to protect the public health and safety.

NEAR-TERM LICENSING PROBLEMS AND RESPONSES

Policy

A. The NRC intends that its regulatory processes be efficient and cost effective. Unnecessary regulatory burdens are to be avcided, and NRC regulations should allow licensees to select the most cost effective ways to satisfy NRC safety objectives. At the same time, pressure to issue new licenses will not be allowed to compromise safety.

Planning Guidance

1. Actions should be taken to eliminate all unwarranted delay in reaching regulatory decisions.

2. Consistent with maintaining safety of oprating plants, staff reviews and public hearings should be completed on a schedule that assures the licensing process will not unnecessarily be a critical path item which would delay reactor startup. Recognizing that the length of hearings may depend on the number of contested issues, normally it should take not more than 11 months from issuance of the final supplemental safety evaluation report to an operating license decision by the Commission in contested cases. The staff should make independent estimates of construction completion dates.

3. Licensing boards should adhere rigorously to established schedules in order to reach timely decisions, while preserving individual rights of the public to pursue valid safety issues. The Commission reaffirms its Statement of Policy on the Conduct of Licensing Proceedings of May, 1981, which urged Boards to take firm hold of hearings and keep them moving.

4. The Congress has directed that the Clinch River Breeder Reactor be built in a timely and expeditious manner as long as the public health and safety is adequately protected. The NRC will conduct the licensing review consistent with its statutory regulatory responsibilities and without delay.

5. NRC will maintain an internal project management structure to oversee project reviews on an integrated plant basis, and ensure that decisions and commitments made early in the project are not abrogated or forgotten, thereby requiring the same issue to be resolved more than once during a project.

6. NRC must continue to work with FEMA to resolve the difficulties in securing the findings for off-site emergency plans for a proposed nuclear plant site in a timely fashion.

7. The staff is encouraged to urge applicants to conduct independent design reviews prior to selection of major systems. The purpose should be to get applicants to understand more fully the equipment and systems which are offered by the vendors and architect-engineers.

8. For the FY 84-86 time period, staffing proposals should decline consistent with the completion of existing reactor casework.

COORDINATING REGULATORY REQUIREMENTS

Policy

A. NRC must be sensitive to criticism that there is a large volume of requirements imposed on licensees, that frequently these requirements are not coordinated, and that sufficient time is not allowed for proper implementation of issued requirements. Strong measures need to be taken to control the issuance of new requirements.

B. In cases where there are conflicting priorities in establishing and implementing new requirements, priorities will be based on the expected risk reduction potential associated with the new requirement.

C. Requirements imposed on the regulated industry by NRC are to have a positive contribution to safety, not only individually, but also when the requirements are taken as a whole. Requirements proposed to achieve incremental reductions in residual risk should be evaluated on a cost-benefit basis.

D. Unresolved Safety Issues should be promptly pursued, and the solutions implemented based on a careful analysis of the costs and benefits of implementation. Priorities for implementation should be established in light of all other requirements imposed on licensees.

E. Issues which affect numerous licensees should be addressed in the context of rule-making as opposed to case-by-case review.

Planning Guidance

1. In order to control and coordinate requirements imposed on licensees, a Committee for Review of Generic Requirements (CRGR) has been established to review proposed requirements and recommend action by the EDO. The CRGR, chaired by the Deputy Executive Director for Regional Operations and Generic Requirements (DEDROGR), is expected to assume a central role in reviewing and recommending action in the full range of generic requirements considered by the agency, including backfitting. The EDO, assisted by the DEDROGR, will exert strong management control over operating reactor licensing actions in order to reduce the existing backlog. Priorities and procedures must be developed for eliminating the backlog expeditiously (i.e., by FY 84). Since the scope of the regional offices has been expanded to create agency wide regional operations that include licensing as well as inspection and enforcement functions, a significant portion of the reactor license amendment reviews should be transferred to the regional offices to assist in cleaning up the large backlog.

2. All generic issues will be integrated in an agency-wide program. Emphasis will be placed on implementing approved solutions to generic safety issues which have been resolved. As a first step in resolving existing generic issues, the staff will examine these issues and recommend to the Commission a priority list based on the

potential significance and cost of implementation of each issue. Those issues which are of marginal importance to the regulatory program will be dropped. The criteria for setting priorities should be reviewed and possibly revised. Issues will be added to the program only after careful evaluation to assure that they warrant resource expenditures.

3. A management system will be developed to account for all new requirements issued, their relationship to the revised Standard Review Plan, and the status of their implementation. The management system to be developed should be coordinated with the systems currently in use and should be capable of incorporating existing, ongoing regulatory requirements.

4. Implementation schedules for new requirements, will be established consistent with the safety importance of the requirement. Licensees should be allowed sufficient time for in-depth engineering, evaluation and design, procurement of high quality equipment, and its proper installation, to the extent compatible with public health and safety. In setting schedules, industry capability (e.g., engineering resources and manufacturing capacity) to implement the new requirement will be considered. NRC's ability to review licensees proposals and to inspect implementation will also be considered. To the extent consistent with safety implications, schedules for

requirements will be set so as to avoid downtime on operating plants or delay in startup of new plants. The staff should work through owners groups to establish realistic schedules. The nuclear industry must be responsible for providing realistic estimates of time needed to achieve compliance. Once compliance dates have been established, the Commission will vigorously enforce such dates.

IMPROVING RELATED REGULATORY TOOLS

Policy

A. The Commission intends to make the present licensing process for power plants more efficient. Consideration is being given to various changes which could affect both new power plant license applications and those already under review. For new construction permits the main elements of the improved process will be based on concepts already studied such as one-step licensing, standardization, and early site reviews. For plants currently under review, improvements such as changes in hearing formats and the nature of technical reviews must be studied. The Commission intends to consider a legislative package for submittal to the Congress and also a set of reforms which can be implemented by the Commission without the need for legislation.

Planning Guidance

1. A special task force will identify the issues which should be addressed in a legislative proposal as well as the specific changes that should be made internally to facilitate streamlining. A senior Advisory Group will assist the Chairman in making specific recommendations to the Commission as a result of the task force's work. By January, 1982 legislative proposals will be prepared. By April, 1982 recommendations for administrative remedies, together with the necessary paper work to implement them, will be ready.

2. In anticipation that legislation will be enacted which provides for increased use of standardization and early siting in connection with one-step licensing starting in FY-84, the staff should examine its existing authority and regulations, identify resource requirements, and determine what changes are needed to undertake such reviews.

SUPPORTING NEW INITIATIVES

Waste Management

Policy

A. The NRC waste management program is critical to the success of an urgent national task. NRC will organize and plan its waste management program to be consistent with the Executive Branch's program as approved by Congress. NRC's waste management program will be based ...e premise that, in the absence of unresolved safety concerns, the NRC regulatory program will not delay implementation of the Executive Branch's program. NRC high-level waste management efforts will focus on the review of DOE site characterization activities and the development of methods to implement licensing criteria for high-level waste repositories. These criteria will be based on a defense-in-depth strategy that requires thorough consideration of various types of sites, demonstrated capabilities of the waste form selected, and the interaction of the waste form and packaging with the geological, hydrological, and engineered systems involved.

Planning Guidance

1. The Commission will expeditiously complete the rulemaking on the storage and disposal of nuclear waste (Waste Confidence Proceeding).

2. The planning basis for waste management activities will be that during FY 82-85, three site characterization reports for a high level waste repository will be submitted to NRC for review. NRC should publish a final rule before January, 1983 covering the

technical criteria for high-level waste repositories. The staff should work with the technical community and the public to develop methods and tests needed to demonstrate compliance with the regulations. After site characterization, the staff will be prepared to review a license application to obtain construction authorization for a high-level waste storage facility. The NRC review and hearing process should permit a decision on issuing a construction authorization within three and one half years of receiving the license application from DOE. Should the Congress pass legislation requiring an earlier decision than presently planned for, the staff should inform the Commission of any obstacles which may exist preventing compliance with the law.

3. Published projections of spent fuel storage requirements indicate that, using currently approved technology, existing reactor storage basins can be modified to accommodate discharged fuel until about the mid 1980s. Longer-term storage will involve proposed new storage pools and the development of dry storage technologies. A licensing capability for independent spent fuel storage facilities exists which will permit the NRC to act promptly on applications for new storage facilities. The NRC must be prepared to review industry or government proposals for away-from-reactor or at-reactor independent spent fuel storage facilities. Because of the lead time for design, licensing and construction, at least one application for a new spent fuel storage facility is expected by 1983. Licensing review will also be required for developmental facilities and work involving the storage of spent fuel in dry storage casks.

TMI-2 Cleanup

Policy

A. The content of the containment at TMI-2 is a potential safety and health hazard to the public. Expeditious cleanup of the TMI-2 reactor is one of NRC's highest safety priorities. While direct responsibility for cleanup rests with the licensee, NRC will provide oversight and support to ensure decontamination of the facility as well as safe and timely removal of radioactive products from the site.

B. NRC should work closely with DOE to reach timely decisions on the disposition of reactor fuel.

Planning Guidance

1. NRC will continue monitoring site cleanup activities through a dedicated TMI program office. The staff should encourage timely completion of reactor building water processing and timely start of containment decontamination (both by mid FY 82). NRC should urge the licensee to submit plans and schedules in mid-FY 83 for reactor head removal. The NRC staff will review these plans and make recommendations to the Commission within three months. Planning for upper internals removal should begin by the end of FY 83, with a goal of having the upper internals removed during FY 84. Since the pace of cleanup is dependent upon licensee's funding ability, the licensee's financial condition will be monitored closely by NRC.

2. NRC will closely monitor the agreement with DOE which calls for removal of high specific activity wastes for research and development, including complete removal of the Epicor liners remaining from the processing of auxiliary building water and the submerged demineralizer system liners after completion of water processing. The objective of NRC's monitoring is to help assure that the wastes are expeditiously removed from the site. NRC should work toward the goal of assuring that DOE will assume responsibility for offsite disposition of the damaged core.

IMPROVING RELATED REGULATORY TOOLS

Safety Goal

Policy

A. The Commission has decided to develop a safety goal and related safety guidance with initial emphasis on individual and societal risks which might arise from reactor accidents. The purpose of this project is to develop a general approach to risk acceptability and safety-cost tradeoffs, and, to the extent possible, to specify qualitative safety goals and quantitative safety guidance and standards for review of rules and practices.

Planning Guidance

1. Simultaneously with obtaining public comment on safety goals the staff should prepare for Commission review a step-by-step action plan describing how it intends to use the goals and numerical guidance within the regulatory process.

2. Qualitative safety goals and associated quantitative numerical guidance, when approved by the Commission, should be used in the evaluation of proposed and existing NRC reactor safety requirements.

Risk Assessment

Policy

A. Probabilistic risk assessment is an important tool for weighing risks against one another and for defining achieved safety levels. Quantitative risk assessment techniques will be used to estimate the relative importance of potential nuclear power plant accident sequences.

Planning Guidance

1. Special attention should be given to using probabilistic assessment techniques where the data warrants such use and in areas especially amenable to risk assessment, e.g., in licensing reviews as appropriate, dealing with generic safety issues, formulating new regulatory requirements, assessing and revalidating or eliminating existing regulatory requirements, evaluating new designs, and formulating reactor safety research and inspection priorities.

Siting Policy

Policy

A. Siting criteria for nuclear power plants and other major nuclear facilities need improvement. The staff has been working to prepare in the very near term modified regulations concerning the siting of nuclear power plants. The Commission now believes that preparation of a safety goal and a better characterization of the radioactive source term must precede new siting regulations.

Planning Guidance

1. The radioactive source term should be reassessed by early 1983.

2. Based on the safety goal and the formulation of a new radioactive source term, a proposed siting rule should be published by late 1983.

Quality Assurance

Policy

The NRC and the industry must strengthen their Quality Assurance programs with specific attention to their implementation. The NRC must encourage the industry to be more aggressive in assuring the adequacy of design, construction, and operation. Quality Assurance programs for plants under construction and awaiting licensing review must receive priority attention to ensure that the plants can be operated with minimum risk to the public health and safety and that costly licensing delays are avoided.

Planning Guidance

1. The NRC staff will review its Quality Assurance efforts and propose an agency-wide plan by early 1982.

2. NRC will develop a program for the systematic review of its Quality Assurance requirements and licensing guidelines.

3. NRC will coordinate with industry to the maximum extent possible in seeking solutions to the Quality Assurance problems currently being experienced by plants under design and construction. In the event that these activities prove to be unsatisfactory, consideration should be given to requiring that industry have independent performance audits of their QA activities.

4. The NRC staff will develop improved inspection and licensing initiatives to ensure the increased effectiveness of utility management control systems. This may include obtaining contracted assistance to evaluate the effectiveness of utility management control systems at selected plants presently under construction.

Research

Policy

A. The research program will continue to emphasize support of the safety of operating reactors and other operating facilities. The purpose of the research program is to assist in establishing regulations for existing and future facilities.

Planning Guidance

1. In view of general budgetary considerations, the agency must be prepared to carry out its research mission with fewer resources. This can be accomplished through more business-like methods, consolidation and coordination of programs with industry and other agencies, and the elimination of marginal programs.

2. The first priority for NRC research efforts will be light water reactor safety.

3. Resource requests to support fast breeder reactor application in the FY 84-85 budget should be consistent with Administration plans. The staff should identify its research and information needs related to: the licensing of breeder reactors, waste, and reprocessing facilities; notify the Department of Energy of these requirements; and to the maximum extent possible, have DOE provide the needed research and information.

4. NRC will develop and maintain a long-range research plan to assure that agency resources are being properly directed toward areas of importance to the licensing and inspection processes. The research plan will be revised and updated annually and subjected to agency-wide review and be approved by the Commission. Research undertaken by the staff will be consistent with the approved long-range research plan.

SAFEGUARDS

International

Policy

A. With respect to its international responsibilities, the NRC recognizes that the proliferation of nuclear explosive devices poses a threat to the security interests of the United States. Hence, the NRC will continue to discharge its statutory licensing responsibilities to ensure that effective controls are applied to the import and export of nuclear materials, equipment, and facilities; the NRC will also seek to support the reliability of the U.S. in meeting its supply commitments to nations which adhere to effective non-proliferation policies by implementing procedures that facilitate the timely processing of export licenses. NRC supports the President's commitment to work with other nations to help the IAEA improve the international safeguards regime.

Planning Guidance

1. Reviews and assessments relating to the applicable criteria mandated in the Nuclear Non-Proliferation Act of 1978 will be conducted.

2. Staff, in consultation with appropriate Executive Branch agencies, will work to develop NRC recommendations for strengthening IAEA safeguards.

Domestic

Policy

A. The Commission considers safeguards an integral and ongoing element of its responsibility for protection of the public. Safeguards regulation should be conducted with the same philosophy as safety regulation.

Planning Guidance

1. Emphasis should be given to performance requirements rather than prescriptive requirements to place the responsibility on licensees to select the most cost effective ways to satisfy NRC requirements.

2. The completion of the remaining elements of the basic safeguards-related regulations -- control of the "insider" and the material control and accounting reform amendments -- should be expedited. The "insider" rule will be submitted to the Commission by June, 1982. The material control and accounting reform amendments should be submitted to the Commission by March, 1983.

3. Recognizing the number of staff organizational elements, federal agencies, state agencies, and local agencies which may become involved in a safeguards event, staff should examine its safeguards emergency planning to assure the establishment of clear lines of responsibility, authority, and communications.

4. Evaluation of safeguards events will serve as a basis for regulatory change and response. This evaluation should include domestic events -- within both the defense and the regulated community -- and foreign events. The staff should not engage in any intelligence activities but rely on the intelligence community for appropriate information.

5. By June 1982, staff should recommend to the Commission its safeguards information needs. Staff should assure that information needs are coordinated with other responsible agencies and provide a basis for agency decisions during safeguards events.

6. The fundamentals of quality assurance will be used in licensee safeguards programs. Principal among these fundamentals are emphasis on appropriate licensee management commitment to safeguards and independent safeguard audits by licensees.

7. Staff, in addition to assuring that safeguards plans are in place at operating facilities and for transportation, will accelerate its independent assessment that these implemented plans meet safeguards objectives and that safeguards regulations adequately support those objectives.

8. The effectiveness of the safeguards system, just as the safety system, is highly dependent on the performance of the human factor. Staff should assure that lessons learned from the reactor human factors program are incorporated as appropriate, in the safeguards program.

*Fire Control

In order to write guides and standards to minimize fires in the LMFBR, all information (worldwide) on sodium fire in tests and all information regarding fires in sodium-cooled reactors should be gathered. We note that an extensive program has been underway in France in their "Esmeralda" facility for a number of years. At least one small sodium fire occurred in the Phenix secondary sodium system while a major fire occurred in this circuit of the Soviet BN-350 reactor.

*Debris Coolability Tests for Partially and Completely Damaged Cores

The two accidents postulated for consideration for the LMFBR are the loss-of-flow and the transient overpower conditions. Both of these abnormal situations, unless corrected, will be plugged at least in part, and fuel "debris" may accumulate within the reactor vessel.

We support the continuation of the international program on core debris coolability tests. We further comment that the probability of an accident creating some debris, by small per se, must be very much larger than the accident which could lead to a potential for recriticality.

*Containment Design Concepts and Criteria

In regard to a containment design, the Review Group notes the following:

(a) Neither the French Phenix nor the British PFR have a tight, high pressure containment. There must be reasons for these decisions. This information should be available to the NRC and the DOE.

(b) The original design of the contaiment for the FFTF was larger than that which was built. Space within the existing shell is limited, and operations and maintenance are somewhat constrained. A larger building can allow for easier maintenance, clearly a safety matter.

We recommend that the principle of a high integrity containment system as a basic design requirement for a LMFBR be reexamined in the context of its influence on maintenance system designs, on the behavior of accident-generated aerosols, and on the possible escape of fission products to the environment.

2.4 CDA Energetics Evaluation

Identifying the mechanism for reactivity addition and bounding their magnitude within a general mechanistic framework is basic to CDA analysis. This scope begins with corewide coolant boiling and fuel melting and terminates with fuel relocation into a permanently subcritical and coolable configuration. The mechanistic framework is required to rule out physically unrealistic situations rather than to claim code-predictive capability of the complete accident evolution in detail. Fuel escape paths and driving forces for dispersal must be evaluated. Detailed evaluations of CDAs for the FFTF and CRBR, together with supporting experimental work, have yielded a mechanistic perspective quite adequate in the above sense. On the basis of this understanding the following observations may be made:

(a) Material motions and possible reactivity additions are in general gradual and to the extent that they yield power increases and development of dispersive forces are self-terminating. In a crude sense, the violence of the resulting dispersal increases with the value of the peak reactivity obtained. Reactivity addition rate (i.e., coherence and speed of material relocation) and the amount of energy required to develop dispersive forces (i.e., fuel temperature at the inception of reactivity ramp) are, therefore, the crucial parameters. Although not all problems have been resolved, it appears that reasonable arguments excluding high levels of energetics may be made on the above bases. The opposite of the self-regulating features mentioned above is termed autocatalysis. Autocatalytic behavior is that condition obtained when power increases cause reactivity increasing motions that overshadow the contribution of the negative ones. Even though this condition is only a temporary one, it has the potential to extend the time duration and perhaps the magnitude of the power burst.

(b) The essential character of the CDA is strongly affected by the sensitivity of reactivity to material motions and it increases with core size. Even undercooling-initiated accidents in homogeneous cores develop into highly overpower conditions, thus attaining an accelerating character such that portion of the core may experience fuel failure prior to coolant boiling. This situation is termed Loss-of-Flow driven Transient Over Power (LOF-d-TOP) and its reactivity consequences are highly uncertain today. Undercooling accidents with heterogeneous cores, on the other hand, develop comparatively in a slow fashion with a gradual evolution into sodium boil-off, clad melting, and fuel disruption. From an energetics standpoint, substantial reactivity insertion mechanisms do not exist in a heterogeneous core until fuel disruption, and even then, coherent motions develop slowly with the gradual spread of fuel melting.

(c) The presence of relatively cold structures and narrow passages in both extremities of the active fuel region inhibit otherwise convenient fuel escape paths. The interassembly space and control rod for internal blankets (heterogeneous cores) offer possibilities of clad-fuel escape paths. Available evidence indicates that, at least initially (especially under the extended clad motion period and low power conditions of heterogeneous cores), a bottled-up configuration caused by plugging of fuel escape paths is considered likely. From this state, establishing reasonable mechanisms for termination depends on the thermal and mechanical integrity of the confining structures under the influence of continuing heating (including power transients) and the associated internal pressures.

Based on this understanding the following short-term specific recommendations are made:

***<u>SIMMER Recriticality Analysis of the Heterogeneous CRBR Core</u>

The heterogeneous core design minimizes the likelihood of energetic behavior in the early stages of core disruption. Instead, the initial development of a bottled-up core configuration is projected. In this configuration, spatial fuel distribution dominates system reactivity. Following the initial disruption, fuel/steel vapor flows will tend to counter gravity forces and maintain a dispersed state. However, these flows are determined by high transient (vapor) sources (power) and sinks (cold steel meltthrough and entrainment). Highly dynamic states and associated recriticalities can be envisioned. The initial stages are dominated by local effects; however, power transients promote coherence and a gradual transition to corewide response. The implication is that recriticality severity may increase. The object is to establish reasonable bounds for it.

In our opinion, SIMMER computations can provide a useful perspective in addressing such systems effects. CRBR-specific computations are therefore recommended. We further believe that such studies should take the form of learning exercises with results carefully scrutinized, discussed, and documented such that the predicted local (separate effects) phenomenology become open to independent investigations.

**Assessment of the Role of Plenum Fission Gases in Aggravating CDAs

Fission gases accumulate at large quantities and high pressures (up to 50 bar) in a plenum provided in the upper two feet of the fuel pin. The resulting axial force acting upon the fuel column may provide motion following fuel overheating and loss-of-pin structural integrity. Though the initial tendency will be to produce buckling, it may also develop into forceful inward injection of the blanket and the upper portion of fuel. The associated reactivity increases are determined by fuel relocation displacements. The issue at hand is whether this effect, acting initially in local subassemblies, has the potential for autocatalytic spread to many assemblies.

Apparently none of the in-pile fuel pin disruption experiments were carried out with prepressurized pins, and we have been unable to locate any other evidence of substance addressing this concern. Scoping modeling studies of plenum fission gas behavior are, therefore, recommended initially to assess the range of expected behavior, and if necessary, guide appropriate in-pile experimentation.

The long-term issues are as follows:

- Study of LOF-d-TOP Reactivity Feedback and Energetics

As already mentioned, LOF-d-TOP phenomena involve fuel pin disruption (at highly overpower conditions) in the presence of coolant and are pertinent to homogeneous rather than heterogeneous core designs. The outcome depends on a complex set of pin-internal and pin-external material motions. Depending on the location of pin failure, pin-internal fuel motion (towards the failure location) may be strongly positive or strongly negative. Depending upon the hydrodynamic and thermal interactions that accompany fuel ejection into the ambient liquid sodium, pin-external motions may be positive if they are coolant-dominated or negative if they are fuel-dominated. Again, the issue of potentially autocatalytic behavior needs to be addressed.

This issue was of central importance during the initial licensing proceedings of the CRBR (homogeneous core). Extensive analytical capability was developed for the occasion; however, experimental information is limited, and important uncertainties remain. With the adoption of a heterogeneous CRBR core, the issue is bypassed for the time being. However, due to its generic nature and its potential relevance to future commercial core designs, it is incorporated in the long-term recommendations.

In light of the vigorous (DOE) research program of fuel-pin failure dynamics (primarily oriented to straight TOP conditions) already in place, a modest evaluation/assessment-oriented effort is recommended. Eventually, inpile testing at the high power levels characteristic of the LOF-d-TOP condition might become necessary.

- <u>Study of Fuel Escape Paths and Dispersal Mechanisms</u>

The major portion of the work in this area to date has concentrated upon the axial paths through the blanket regions and the fission gas plenum. In view of the importance of establishing a reasonable mechanistic path to termination, a more comprehensive effort in this area is recommended. Specifically inter-subassembly spacing, control rod guide tubes, and internal blankets need to be carefully evaluated as possible fuel escape paths. Further, the circumstances yielding relief paths from a bottled-up configuration under the influence of continuing heating (melting attack at the boundaries) and internal pressures need to be established.

- <u>Improvement of Understanding of Recriticality Potential and Severity</u> (Separate effects experiments, analysis, and system effect computations)

This recriticality oriented task is the long-term counterpart (or continuation) of the short-term CRBR-related recommendation. A generic approach with successive interactions between estimating (computationally) system response and refining the fundamental understanding and mathematical representation of separate effects (with the help of experiments and analyses) is recommended.

- <u>Investigation of Fuel-Coolant Interactions</u>

In general, CDA conditions are not conducive to large-scale, efficient, thermal interactions. Energetically, therefore, only the indirect effects of local interactions, associated pressure pulses, and the potential for recriticalities need to be examined. Since the issue is of fundamental significance, an improved understanding is warranted.

3. SUMMARY

The summary of the recommended short-term and long-term research subjects is listed in Table 1 in the order of their relative priorities. This summary is a consensus of our opinion in response to the charter of the Review Group.

H. S. Isbin

H. S. Isbin

R. L. Seale

R. L. Seale

W. R. Stratton

W. P. Stratton

T. Theofanous

T. G. Theofanous

L. S. Tong

L. S. Tong (Chairman)

Los Alamos
Los Alamos National Laboratory
Los Alamos,New Mexico 87545

Professor Max Carbon
Department of Nuclear Engineering
University of Wisconsin
Madison, Wisconsin 53141

Dear Max,

My recent conversations with you and Carson Mark were stimulated by the forthcoming safety review of the Clinch River breeder reactor and by my belief that the problem of the "Hypothetical Core Disruption Accident" (HCDA), the "Maximum Credible Accident," the "Explosion Accident," the "Bethe-Tait Accident," or however designated, can now be solved or at least settled in the sense of not offering an undue risk to the health and safety of the public. I believe that analytical tools now exist that are much superior to those used for the FFTF (and for earlier reactor designs) and that our understanding of the necessary phenomenology also is greatly improved. Certainly, knowledge is not complete nor are computer programs perfect, but both are adequate for the purpose. The present political climate also seems favorable, and the appropriate reactor to examine first is the CRBR.

Thus, I believe that the time has arrived, both technically and politically, for a comprehensive review of the matter and that the proper (and best) forum for this review is the Advisory Committee on Reactor Safeguards with the assistance of its appropriate subcommittees. I suggest that an ad hoc and especial task force of, say, 8 to 12 nationally recognized experts in the appropriate specialties (physics, nuclear engineering, metallury, neutronics, explosives, hydrodynamics, chemistry, or whatever may be necessary) who are independent of the developmental and regulatory agencies be chosen. They should work closely and actively with the ACRS, the NRC, the DOE, and the several investigative groups (e.g. national laboratories, reactor vendors and safety specialists in the U.K., France, and Germany) that have been studying this and related problems for many years.

A fresh look and an intensive effort of some months would be needed probably beginning when the several computational centers (ANL, Los Alamos, GE) have completed or are well along in their analysis of the present core proposed for the CRBR. Precedents exist in the history of the ACRS for such a specialized and intensive effort, for example the pressure vessel study completed in 1974.

To put my proposal in perspective, allow me to set down my perception of the political situation and some general thoughts on the

ATTACHMENT 5

LMFBR and comment on what I see as its technical advantages and disadvantages. The reasons for this special study at this time are part and parcel of these several factors and discussion.

Political Situation

Both the Executive Branch of the government and the Congress are now in favor of constructing the Clinch River Plant and proceeding with additional developmental plants. I have been told that the Department of Energy has formally requested permission from the NRC to commence construction, but I do not expect that the NRC has yet responded nor should it do so without adequate consideration. However, the NRC has reacted to the President's statement of a few weeks ago by reactivating a review organization within its licensing division and by reviewing its own fast reactor research efforts (a report by L. S. Tong's Special Review Group was posted to you under separate cover). Thus, I believe that this administration is determined to begin construction of the CRBR project; support for efforts to solve its licensing problems, therefore, should be forthcoming.

Advantages of the LMFBR

The sodium-cooled reactor is an interesting creation that has a number of safety advantages. A few of the obvious advantages are as follows:

1. The primary system operates at low pressure--only high enough to move the sodium through the system.

2. The coolant is noncorrosive to materials and components designed for its environment. EBR-II experience is showing some remarkable results from components that have been in sodium for nearly a generation.

3. The coolant operates far below its boiling temperature.

4. The large volume of sodium provides an enormous heat sink.

5. The coolant's heat transfer characteristics are excellent. These characteristics--the operating temperature, the heat sink, and the high heat transfer rate--have not been investigated systematically or exploited fully, but it is clear that a significant power transient involving a large temperature rise could be accommodated without damage to the core.

6. The coefficient of expansion of the coolant is large enough that convective cooling can be designed into the system. The advantages of this property of sodium have not been fully exploited in existing designs.

7. The coolant, sodium, is a marvelous getter for iodine, which is, by far, the most dangerous of the fission products. This property is a safety factor of great significance.

Prima facie, it appears that if the reactivity control and decay heat removal systems operate reasonably effectively, nothing much can go wrong. Indeed, on a second look, this still seems to be the case; even if the latter system works only poorly or not at all, a long time, depending upon design, should be available to take action before the health and safety of the public is threatened.

Some persons regard other factors as good reasons for continuing the development of this reactor concept. These factors include such matters as the following.

1. The fuel is U-238, which is in abundant supply and inexpensive per se. Given a successful design or designs, the price of energy (electricity, hydrogen) from this source should be constant except for inflation of the economy for other reasons. For the first time, one can truly speak of a "lid" on the price of power and energy.

2. The limited amount of U-235 in the world would not be consumed in a few generations as will be the case if only U-235 burners are used. U-235 is, after all, the only naturally occurring fissionable isotope, and, like seed corn, we should be niggardly about using it.

3. The supply of uranium already above ground is sufficiently large that mining operations would not be needed for this reactor concept for generations, perhaps a century. This reduction of mining requirements is, of course, a safety matter of significance.

4. Most of the excess plutonium would be in use in reactors and hence inaccessible to terrorists. A chemical processing plant and fuel production plant may well be easier to safeguard than a large number of spent fuel storage pools (sometimes referred to as latent plutonium mines). Additionally, the sodium-cooled reactor can be regarded as a plutonium burner, and hence actually reduces the amount of plutonium in the world and truly lessens the terrorist threat.

5. The breeder reactor is the only sure thing we have for future generations. If we (our generation) are so selfish as to burn all the fossil fuels and U-235, the least we can do for our children and grandchildren is to provide them with the technology to produce an abundant and assured supply of energy. Whether or not they use it is their choice; our task is to create the capability.

Disadvantages of the LMFBR

Disadvantages exist, and the last of these get to the crux of my proposal to you and Carson. Some of the difficulties or problems include the following.

1. The optimum design of the sodium-cooled, plutonium-fueled power plant must be at least a couple of decades away, commercialization and accumulation of experience certainly is a matter of decades. Things take longer now than they did 30 years ago, and introduction of this concept probably will be more time consuming than was the case with the LWR. Its expense is greater than can be afforded by a single corporation, and the first few plants must be funded by the federal government or, perhaps, by a small tax on the entire electric utility industry.

2. The coolant is liquid sodium; large amounts have been handled successfully, but generally, the utility industry is unfamiliar with handling the necessary very large amounts.

3. The coolant is flammable in air and reacts violently with water.

4. The sodium captures neutrons and becomes radioactive with a half-life of about 15 hours. Given the fallibility of mankind, one must assume that sooner or later serious sodium fires will occur and that some of these will be fires with radioactive sodium. Fortunately, because of containment or confinement, such a fire need not pose a threat to the health and safety of the public.

5. The sodium-water steam generator is a difficult device to design and construct so that no leaks, even pinhole size, exist. Success has been achieved (e.g., EBR-II), but the task is not easy or inexpensive, and difficulties have been encountered (e.g., PFR in the UK, BN-350 in the Soviet Union). Fortunately, this area of the plant is not radioactive so that additional hazard is not present.

6. The neutronic and reactivity characteristics of the fast neutron core are such that the voiding of sodium coolant from some parts of the core will increase reactivity and thus reactor power. A reactivity control or scram system must work, should a situation develop that involves boiling of sodium in a significant fraction of the core.

7. The core of the fast neutron reactor is not in its most reactive configuration. Should some accident or incident cause the core or some fraction of the core to be driven into a smaller volume by even a small amount, its reactivity and the reactor power level would increase. Again, the reactivity or shut-down controls must work properly, to avoid damage to the core.

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Item 7 is the origin of the reactivity accident that is associated with the fast neutron reactor. Indeed, in the early 1950s, during the design of the Dounreay Fast Reactor, some people were willing to postulate spherical implosions of that little core. The resulting calculations naturally predicted explosive energy releases of the order of tons of high explosive equivalent, given such an unrealistic and imaginary situation. In order to resolve this apparent dilemma, in 1957 Bethe and Tait assumed a gravity induced collapse of a voided and molten (but in-place) core and showed that even with these assumptions, the explosive energy could not be more than the equivalent of 160 kg of HE.* The Bethe-Tait result was accepted, even though it was unrealistic, as an upper limit for the Dounreay Reactor and was satisfactory then because it showed that containment of an explosion of this magnitude was quite feasible. Unfortunately, the precedent of assuming a very unlikely or even a near-impossible situation for a worst-case analysis was set and has plagued all subsequent LMFBR proposals and designs and discussions, both technical and popular. Indeed, the fuel-melting accident in the Fermi Plant was caused by a hastily installed safeguard to protect against the threat of accumulation of molten fuel and a possible "Bethe-Tait" accident.

Since the time of the Dounreay calculation, the history of analysis of this and related hypothetical accidents (for various reactors) has been to insert more realism and less arbitrariness into the initial assumptions and calculational technique. The result has been a fairly steady reduction of the estimate of the possible magnitude of the "explosion" or "energetics" as it is sometimes called.

A good many fast reactor designers, analysts, and technical specialists believe that the day of the "explosion" accident concept has come and gone; however, this belief is sometimes based on physical intuition and engineering experience rather than a rigorous investigation and analysis. I place myself in the group of those who think about the problem and have this opinion, but I have worked in this field; hence my proposal in the beginning of this letter is founded on a background of experience and quantitative studies. I believe that a rigorous examination of the facts of the case will show no "energetics" for the Clinch River Plant and, further, will at least be strongly indicative for future, larger LMFBRs.

<u>Conclusions</u>

The possible reward is potentially very great as I discussed above; the reactivity accident is about the only conceptual accident characteristic of the LMFBR that would be of significance to the health

*Modern, but still conservative, calculations of the Bethe-Tait model show about the same number of fissions but <u>no</u> explosive energy. Note that the energy equivalent of 1 kg HE is 4.2 megajoules.

and safety of the public. If this proposed study leads to a positive result, it will certainly suggest that the LMFBR may be unique in regard to its public health and safety characteristics.

In conclusion, I repeat my recommendation for an ad hoc, pre-eminent advisory review panel to assist the ACRS in this part of its consideration of the Clinch River Plant. The tools and knowledge are available, and the political climate (and hence funding) is favorable to such a special effort. The ACRS provides the proper forum and commands sufficient respect worldwide to collect the best talent available in the United States and abroad. The task is worthy of our best efforts. Please be assured that I am available to cooperate with you and the Committee on this matter at any time.

Sincerely,

Bill Stratton

William R. Stratton

WRS:hmb

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Carson Mark, T-DO, MS 210
ITO Files
CRMO (2)

United States Department of the Interior

GEOLOGICAL SURVEY
RESTON, VA. 22092

In Reply Refer To:
Mail Stop 905

January 25, 1982

APPENDIX XII
EARTHQUAKE HAZARDS REDUCTION PROGRAM

Dear Colleague:

It has now been five years since the passage of the Earthquake Hazards Reduction Act of 1977 and the beginning of an enhanced earthquake program within the Geological Survey. During the next five months a review will be conducted of the various elements of the National Earthquake Hazards Reduction Program for which the Geological Survey has responsibility, viz: earthquake prediction, earthquake hazards and risk assessment, earthquake data and information services (global seismology), and induced seismicity.

We are asking you, because of your participation in or observation of the earthquake program, to give us your views on:

(1) The progress the program has made,

(2) The problems that the program has encountered, and

(3) Changes that you recommend be made in program structure or focus.

In addition to your views from a scientific perspective, we seek advice on general programmatic matters such as division of effort between program elements, the relationship between operational and research responsibilities, and the geographic emphasis of the program. We ask that your recommendations take into account the reality of the current and projected funding trends, which are anticipated to be level, at best.

We have enclosed a list of the goals and objectives of the various elements of the earthquake program as it is now cast. We also indicate the approximate levels of resources that are currently being applied to each program element and objective. We point out that due to the nature of the legislation cited in the first paragraph and subsequent directives and agreements within the executive branch, we do not have complete freedom to redefine the responsibilities or goals of the Geological Survey within the national program. We can, however, review and alter our approach to meeting these responsibilities and attaining these goals. We appreciate any advice and guidance you can give us in this process, and hope to receive any written response to this request within a month.

Thank you for your help.

Sincerely yours,

John R. Filson

John R. Filson
Chief, Office of Earthquake Studies

Enclosure

EARTHQUAKE HAZARDS REDUCTION PROGRAM

Element I. Earthquake Hazards and Risk Assessment $11.1M

<u>Goals:</u>

- Delineate and evaluate earthquake hazards and risk on a national scale.
- Delineate and evaluate earthquake hazards and risk on a regional scale in urbanized locals of high seismic risk.
- Develop improved methods for evaluating earthquake potential, for predicting the character of damaging ground motion and the incident, nature, and extent of earthquake-induced ground failure, and for estimating potential earthquake losses.
- Provide advance "warnings" of earthquake and related geological hazards to allow for the mitigation of their effects.

<u>Objectives</u>

1. Delineate and evaluate earthquake hazards and risk in the United States on a national scale. $1.1M
2. Delineate and evaluate earthquake hazards and risk in earthquake-prone urbanized regions in the western United States. $4.5M
3. Delineate and evaluate earthquake hazards and risk in earthquake-prone regions in the eastern United States. $1.8M
4. Establish an accurate and reliable national earthquake data base. $.5M
5. Improve capability to evaluate earthquake potential and predict character of surface faulting. $.6M
6. Improve capability to predict character of damaging ground shaking. $1.8M
7. Improved capability to predict incidence, nature and extent of earthquake-induced ground failures, particularly landsliding and liquefaction. $.7M
8. Improve capability to predict earthquake losses. $.1M

Element II. Earthquake Prediction $15.4M

<u>Goals:</u>

- Obtain pertinent geophysical observations and attempt to predict great or very damaging earthquakes.

- Obtain definitive data that may reflect precursory changes near the source of moderately large earthquakes.

- Provide a physical basis for short-term earthquake predictions through understanding the mechanics of faulting.

- Determine the geometry, boundary conditions and constitutive relations in seismically active regions to characterize the physical conditions accompanying earthquakes.

Objectives:

1. Operate seismic networks and analyze data to determine character of seismicity preceding major earthquakes. $6.0M

2. Measure and interpret geodetic strain and elevation changes in regions of high seismic potential, especially in seismic gaps. $2.4M

3. Measure strain and tilt near-continuously to search for short-term variations preceding large earthquakes. Complete development systems (2) for stable, continuous monitoring of strain. $2.0M

4. a. Monitor subsurface water levels, water chemistry, emanation of radon and other gases in close association with other monitoring systems to measure and understand precursory changes in these phenomena. $.9M

   b. Monitor apparent resistivity, magnetic field, seismic wave velocity, and attenuation in and near the San Andreas fault zone. $.8M

5. Develop theoretical and experimental models to guide and be tested against observations of strain, seismicity, variations in properties of the seismic source, etc., prior to large earthquakes. $1.5M

6. Measure physical properties including stress, temperature, elastic and anelastic properties pore pressure, and material properties of the seismogenic zone and the surrounding region. $1.8M

Element III. Earthquake Data and Information Services $3.6M

Goals:

- Install, operate, and maintain, and improve by system design studies global and national networks of seismograph stations to provide a sound and dependable data resource for fundamental studies in observational seismology.

- Collect, analyze, and disseminate information on the occurrence and effects of earthquakes to the public, to the research community, and to those involved in earthquake prediction and hazards assessments.

- Provide basic and applied research in support of these activities to insure that further refinements keep pace with technological advances in instrumentation and new developments in theoretical seismology.

<u>Objectives:</u>

1. Install, operate, maintain and improve standardized networks of seismograph stations. Process and provide digital seismic data on magnetic tape in network-day tape format. $2.1M

2. Provide seismological data and information services to the public and to the research community. $1.2M

3. Improve seismological data services through basic and applied research and through application of advances in earthquake source specification and data analysis and management. $.3M

Element IV. Induced Seismicity $1.2M

<u>Goals:</u>

- Devise techniques for diagnosing in advance whether reservoir impoundment or fluid injection or withdrawal in wells at a particular site holds the potential for triggering damaging earthquakes.

- Determine the physical mechanism responsible for reservoir-induced seismicity in cases where it is known to occur.

- Devise techniques permitting the design for each site of a strategy for management of water levels or injection pressures so as to minimize, if not eliminate, the possibility of inducing a damaging earthquake.

<u>Objectives:</u>

1. Understanding the physical mechanism of induced seismicity through intensive geological and geophysical investigation. $.7M

2. Understanding the geologic conditions under which induced seismicity is probable and establish criteria for diagnosing cases of reservoir-induced activity from naturally occurring events. $.2M

3. Devise actions for hazard assessment and mitigation at sites of reservoir-induced activity. $.3M

Seismicity is a significant contributor to risk. Selecting those areas where there is a greater concentration of plants, what are the questions we need to study in order to address the question of uncertainty and assessment of risk to reduce uncertainty?

In the Eastern United States, the seismic activity occurs at depths of at least several kilometers, in general (there are some exceptions) and in many places the structures in which the activity is occurring are also covered by some type of overburden. Thus, it is difficult to relate the underlying causative structures to the seismicity. Achieving an understanding of this problem is one of the objectives of a balanced research program in geology and seismology.

The decision to continue the research in seismology and reduce the research effort in geology in FY 82 was because we did not want to interrupt the seismic record.

- All regional geology programs are fully funded through July 1982.
- On the basis of work completed over the past 5 years, it is clear the program in geology needs to be redirected to deal with issues important to licensing case review. Particularly as related to operating plants and plants about to come on line.
- After discussion with the staff of NRR the decision was made to maintain the operation of all seismic networks with the exception of one in Minnesota and one proposed for Oregon.

- The increased use of probabilistic techniques used in licensing actions requires a seismic data base which this program provides.

- Continued monitoring of the seismic networks is necessary to support NRR licensing actions to protect the public health and safety. NRR has been in the position that it needed data but ~~could~~ DID not require the utility to provide the data. (Indian Point) Data was obtained from networks.

- Seismic data and statistical input can be used in the systematic evaluation program. Most of the older plants in the Eastern United States are in areas covered by the seismic networks.

- Data generated by the seismic networks will be essential to the rewrite of Appendix A to 10 CFR 100 and in generating appropriate guidance.

- There is broad general application being made of the data obtained from the networks and we are seeking a broader base of funding.

Redirection of the Geologic Program

Aim of the program is to identify areas of greatest risk and tailor the program so that risk can be reduced.

High Risk Areas with significant numbers of NPP's.

New England

New Madrid - Anna, Ohio

Southeastern United States

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Types of Investigations

Crustal strain measurements

Crustal stress - insitu measurements

Geologic studies & mapping in specific problematic areas as indicated by other studies.

Geophysical studies of structure of crust at earthquake hypocenters and key tectonic structures.

New England

- Examine recent vertical crustal movements as the basis for examining recent geology, such as geomorphology, geodetic surveys, historical archeology and marine geology.
- Examine the recent stress field in New England
- Critical areas should be examined for geologic anomalies (faults, fractures, etc.).

South Eastern U.S.

- Continue work with the USGS in study of the Charleston earthquake.
- Drilling in the coastal plain along suspected faults and additional vibroseismic studies.

New Madrid - Anna, Ohio

- Determine the nature and limit of the Northern extension of the New Madrid fault zone.
- Continue specific and geophysical mapping of the Reelfoot Rift.
- Determine recent crustal strain by means of geologic and geodetic measurements.

- By means of geophysical techniques study the detailed structure of the Anna Ohio earthquake area.

Geology/Seismology Funding Levels

	FY 81	FY 82
Geology	870	*44
Seismology	2033	1500
USGS	1270	875
	4173	2419
Topical Studies	827	336
	5000	2755

*Geology fully funded through July 1982

Seismology	FY 82	Geology	FY 81
Northeast US	660		320
Southeast US	440		80
New Madrid	320		385
Nemaha Ridge	110		160
	1530		945

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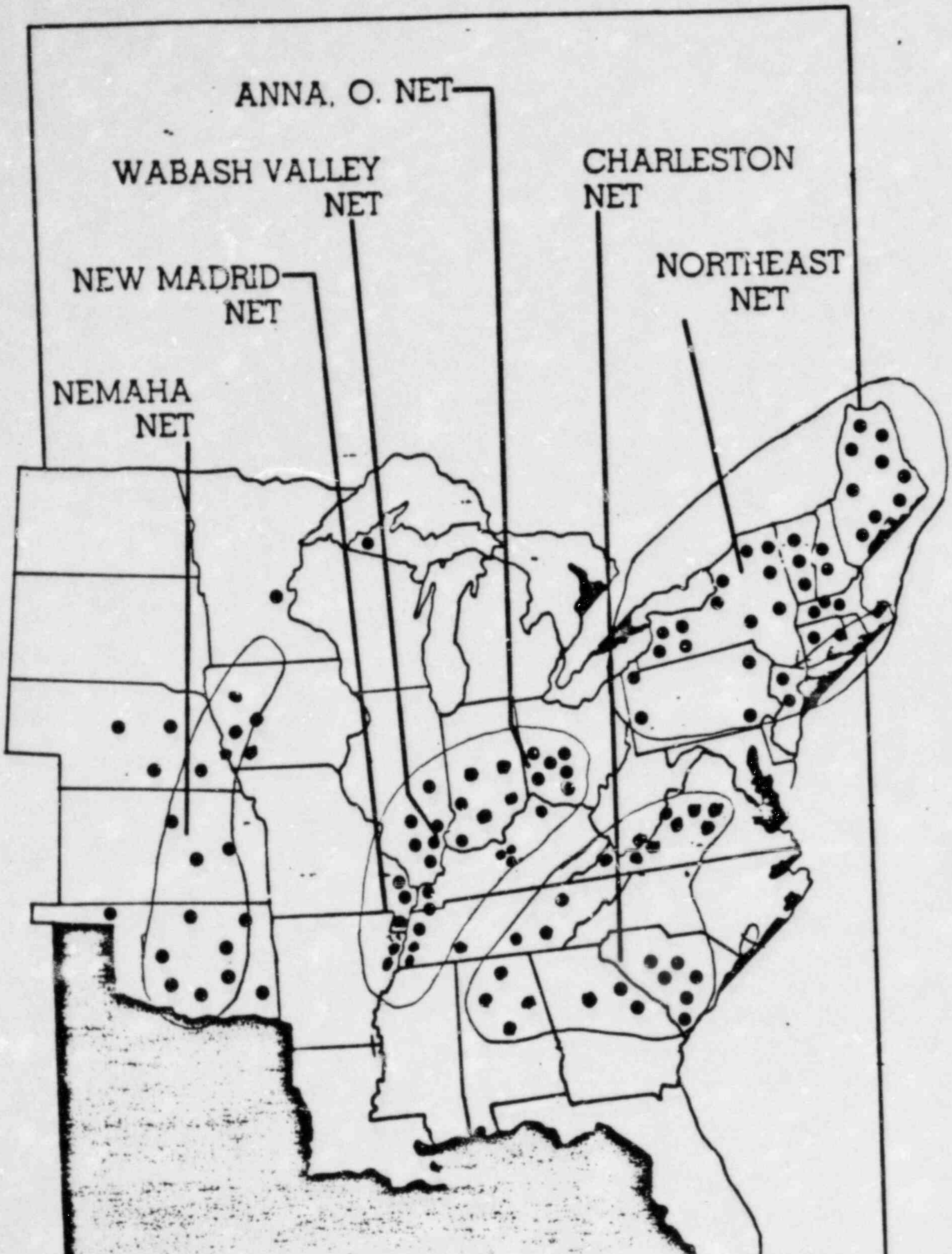
ANNA, O. NET
WABASH VALLEY NET
CHARLESTON NET
NORTHEAST NET
NEW MADRID NET
NEMAHA NET

A-339

METEOROLOGICAL RESEARCH PROGRAM

DISPERSION PROGRAM:

THIS IS A MODEL EVALUATION PROGRAM FOR USE IN ACCIDENT ANALYSIS AND EMERGENCY PREPAREDNESS.

STRONG USER NEED FROM I&E AND NRR REQUESTING THIS PROGRAM BE CARRIED ON.

THE EXPERIMENTAL FIELD TEST PROGRAM IS DESIGNED TO OBTAIN CONCURRENT METEOROLOGICAL DATA AND TRACER CONCENTRATION DATA TO VALIDATE MODELS.

THE MODELS ARE USED IN ASSESSING ACTUAL OR POTENTIAL CONSEQUENCES DURING RADIOLOGICAL EMERGENCY CONDITIONS TO DISTANCES OF UP TO 30 MILES. THE MODELS MUST BE APPLICABLE OVER A VARIETY OF TERRAIN CONDITIONS.

THIS IS NOT A MODEL DEVELOPMENT EXERCISE.

FY 82

880^K FIELD TEST-OIL FOG & SF_6 RELEASED - AIR SAMPLER COLLECT SAMPLES ON A GRID SPACING OUT TO 30 MILES. NOAA

200^K MODEL EVALUATION BASED ON FIELD TEST DATA OAK RIDGE

XOQ DOQ NRC GAUSSIAN PLUME
EPA-ISC/CRSTER

MESO PUFF PUFF MODELS ERT-NEW ENGLAND CONSULTING FIRM
ARL PUFF NOAA PUFF MODEL

MESODIF-II USED AT HANFORD AND I&E FOR EMERG. PREP.

PATRIC (PARTICLE IN CELL) DOE
MATHEW/ADPIC PIC MODELS

IMPACT F&S-DEVELOPED BY FORM & SUBSTANCE (F&S)
NOABC SAI-SCIENCE APPLICATIONS INC. (SAI)

PIELICE COMPLEX SEA BREEZE MODELS
YAMADA " " " "

CRAC-METEOROLOGY - DEVELOPED BY STAFF NEEDS TO BE EVALUATED.

315K	Lidar measurements of plume created by the release of the oil fog. These are airborne measurements to determine the dimension and concentration of the elevated plume.
195K	Reduction of Richland, Washington Hanford dispersion data from U.S. Army Chemical Warfare Department tests.
75K	Estimation of building surface concentrations and residents times of effluents in building wakes (control room close problem).
	Estimate turbulent diffusion by analyzing anemometer records from an upwind and downwind met tower.
85K	Generic study of various building configurations to study the dispersion near reactor complex to assess control room habitability problems. User need for this study from AEB, NRR.
1750	

SEVERE STORMS PROGRAM

THERE ARE NO NEW STARTS IN THIS PROGRAM IN FY 83.

THE PROGRAM WILL CULMINATE THE MANY YEARS OF RESEARCH IN THIS AREA. THERE ARE USER NEED ENDORSEMENTS FOR ALL PROJECTS.

<u>NEAR GROUND TORNADO WINDFIELDS</u> 85^K

<u>TORNADO WINDSPEED ASSESSMENT</u> 145^K

RECONCILIATION OF TORNADO DATA SETS.
REGIONALIZATION OF DESIGN BASIS TORNADO CHARACTERISTICS.
TORNADO DATA/WINDSPEED POSSIBLE METHODOLOGY.
TORNADO FLOW FIELD MODEL RECOMMENDATION.
TORNADO DAMAGE DOCUMENTATION.

<u>TORNADO DATA ASSESSMENT</u> 55^K

RECONCILIATION OF KANSAS CITY AND CHICAGO DATA SETS.

<u>U.S. TORNADO STATISTICS</u> 40^K

UPDATE "KANSAS CITY" TORNADO DATA AND PROVIDE METEOROLOGICAL JUSTIFICATION FOR TORNADO SPARSE REGIONS.

<u>WIND LOAD COMPARISONS</u> 30^K

COMPARE PRESSURE MEASURMENTS IN SIMULATED LABORATORY TORNADO FLOW FIELD WITH MEASUREMENTS ON SOME MODEL STRUCTURES IN BOUNDARY LAYER WIND TUNNEL WITH FULL SCALE FIELD MEASUREMENTS.
EVALUATE CODES PRESENTLY USED IN DESIGN OF NUCLEAR STRUCTURES. (NMSS TYPE & NPP's)

APPENDIX XIV
ADVANCED REACTOR SUBCOMMITTEE: REQUEST FOR GUIDANCE IN IDENTIFYING SAFETY ISSUES & SAFETY RESEARCH NEEDS FOR COMMERCIAL-SIZED LMFBRS

The following pages A-344 thru A-350 has been deleted as I.

DELETION

COMMITTEE ON INTERIOR AND INSULAR AFFAIRS
U.S. HOUSE OF REPRESENTATIVES
WASHINGTON, D.C. 20515

APPENDIX XV
APPEARANCE BEFORE THE SUBCOMMITTEE ON ENERGY AND THE ENVIRONMENT OVERSIGHT HEARING

February 2, 1982

Dr. Paul G. Shewmon, Chairman
Advisory Committee on Reactor Safeguards
Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Shewmon:

This is to request that you and other appropriate members of the Advisory Committee on Reactor Safeguards (ACRS) appear before the Subcommittee on Energy and the Environment oversight hearing on Tuesday, March 2, 1982. The subject of the hearing will be the Nuclear Regulatory Commission budget request for fiscal year 1983.

As you know, the House passed H.R. 2330 to authorize appropriations for NRC for both FY 1982 and FY 1983. In the next several weeks the Senate is scheduled to consider a similar bill, thereby setting the stage for what I expect will be an expeditious and successful conference between the two Houses. When the Congress completes action on this measure, it will mark the first time that a two-year authorization has been approved for NRC. We undertook the development of a two-year authorization bill as somewhat of an experiment, and it appears to me that the results to date are generally positive. In this regard, please be prepared to discuss at the hearing the pros and cons of transforming the ACRS annual report to Congress on NRC's safety research program into a biennial report.

The ACRS testimony of March 2, 1982 should provide the Subcommittee with an overview and summary of your review and evaluation of the NRC safety research program for fiscal year 1983. Special attention should be paid to your assessment of significant changes, if any, initiated by NRC subsequent to the agency's first submittal of an FY 1983 research budget request in February of 1981.

With regard to the Loss-of-Fluid-Test (LOFT) program, the ACRS testimony should include a discussion and critique of NRC's current plans for bringing the test program to an early and orderly conclusion. In addition, please provide a statement of your assessment of the minimum level of

FY 1982 (and if necessary, FY 1983) funding required to complete the Commission-endorsed LOFT test program.

As part of your discussion, I would appreciate your providing a statement of your views on the most significant findings to date from the NRC's research program and the effect of these findings upon the Commission's standards, regulations and regulatory guides.

The Subcommittee also would be interested in your current thinking on problems associated with steam generators at nuclear plants. In this regard, we would appreciate your comments on whether the sudden tube rupture, the PORV failure, development of steam bubbles in the pressure vessel and steam generators, or other aspects of the Ginna incident cause particular concern. Does the existence of a pathway out of the reactor bypassing the containment raise questions as to whether the barriers to radiological releases are as strong as has been commonly believed? In addition, please include a discussion of the impact of the recent discovery of steam generator tube degradation at Three Mile Island Unit I or other aspects of TMI-I on the ACRS position on the restart of that plant.

Finally, your testimony should address the following two aspects of the NRC's FY 1983 budget request: ACRS views on the Nuclear Data Link; and, the adequacy of funds requested for your Committee to carry out its various responsibilities.

In accordance with Committeerules, I ask that you make available to the Subcommittee fifty copies of a written statement of your prepared testimony on Friday, February 26, 1982. The Subcommittee further requests that you limit your oral presentation to a brief summary of the prepared statement. As always, the Subcommittee welcomes the separate view of individual ACRS members.

I appreciate your cooperation in this matter, and look forward to the ACRS appearance before the Subcommittee on March 2.

Sincerely,

Mo Udall

MORRIS K. UDALL
Chairman

A-352

The following pages A-353 thru A-355 has been deleted as 1.

DELETION

EG&G Idaho, Inc.
P.O. BOX 1625, IDAHO FALLS, IDAHO 83415

January 28, 1982

Mr. R. E. Tiller
Reactor Operations and Programs Division
Idaho Operations Office - DOE
Idaho Falls, Idaho 83401

TRANSMITTAL OF OBJECTIVES FOR REMAINING LOFT TESTS - LPL-18-82

Dear Mr. Tiller:

This letter defines "programmatic objectives" for the remaining tests in the current LOFT test schedule.

The "programmatic objectives," which have been written to provide a clear and concise statement of the issues being addressed in each test, are:

<u>L9-3 and L9-4</u>:

1. Provide experimental data for benchmarking pressurized water reactor (PWR) vendors' Anticipated Transient Without Scram (ATWS) computer codes as required by the Nuclear Regulatory Commission (NRC) proposed ATWS rule (USNRC SECY-80-409).

2. Evaluate alternate methods of achieving long term shutdown (without the insertion of control rods) following an ATWS event, to address concerns defined in the proposed NRC staff rule (Federal Register Vol. 46, No. 226).

<u>L6-6</u>:

1. Provide NRC with data to assess the conservatism of analytical methods used by PWR vendors to demonstrate compliance with Standard Review Plan requirements regarding the minimum time to reach criticality in an inadvertent boron dilution accident.

<u>L2-5</u>:

1. Provide experimental data to demonstrate that Appendix K assumptions result in a conservative prediction of peak clad temperature, even if core hydraulic conditions were to occur in a commercial reactor which precluded the early return to nucleate boiling (rewet).

2. Provide data to confirm that results from early LOFT large break experiments were not significantly affected by external cladding thermocouples.

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L6-8:

1. Assist NRC in evaluating reactor transient analysis techniques used in reactor licensing by applying the same techniques to transients performed in the LOFT facility.

2. Demonstrate that LOFT results can be related to larger PWRs by providing data that can be compared to data obtained from commercial plants (traceability).

3. Provide data for evaluating commercial plant instrumentation and control system response characteristics over a range of transients which could occur in a commercial plant.

L2-6:

1. Demonstrate that for an end-of-life fuel pressure of 600 psi and core hydraulic conditions which limit the potential for early rewet, acceptance criteria defined in 10 CFR50.46 will be met.

2. Contribute to severe core damage rulemaking by experimentally validating fuel rod balloon and burst models and the separate effects data base in a large integral nuclear facility.

3. Determine primary system fission product transport characteristics in a realistic PWR environment during an accident scenario involving fuel cladding damage.

To aid in the interpretation of the issues addressed in the LOFT tests, the attachment provides a brief description of each test, the issues being addressed, and the test specific objectives that will contribute to the resolution of these major programmatic objectives.

Very truly yours,

L P Leach

L. P. Leach, Manager
LOFT Department

EAH:seb

Attachment:
As Stated

cc: R. W. Barber, OE, DOE-HQ
G. D. McPherson, RSR, NRC
R. W. Kiehn, EG&G Idaho, w/o attach

ISSUES AND OBJECTIVES

FOR REMAINING LOFT

TESTS

A-358

L9-3 LOSS-OF-FEEDWATER ATWS

Anticipated Transients without Scram (ATWS) for light water reactors have been a long-standing unresolved safety issue of the Nuclear Regulatory Commission (NRC). The significance of ATWS in reactor safety is that some ATWS events can result in high system pressures, which can ultimately lead to fuel damage and the potential for release of a large amount of fission products. Despite differences of opinion within the nuclear industry, the regulatory staff of the NRC has consistently held that "... the likelihood of severe consequences arising from an ATWS event is acceptably small, but that the future likelihood of severe ATWS consequences could become unacceptably large and measures should be taken to diminish such consequences."(1)

To address the ATWS issue, the NRC in September 1980 published a proposed ATWS rule(2) to amend the Code of Federal Regulations (10CFR50). Now, after more than a year of public meetings and comments, the NRC is about to issue a revised ATWS rule. It is expected the revised rule will include measures both to reduce the likelihood of ATWS events and to mitigate the consequences of an ATWS once it has occurred. The L9-3 experiment has been developed to address issues identified in SECY-80-409 as well as those expected to be addressed in the forthcoming revision to the ATWS rule.

In evaluating ATWS accidents, the NRC lists ten initiating events for pressurized water reactors (PWRs), which are expected to occur one or more times during the life of the nuclear power unit.(3) These events can be classified into four categories: (1) reactivity related accidents (rod withdrawal, boron dilution, inactive primary loop startup, load increase, excessive cooldown), (2) degradation of reactor heat transfer (loss of primary flow, loss of electrical load, loss of normal electrical power), (3) degradation of reactor heat sink (loss of normal feedwater), and (4) primary system depressurization caused by accidental opening of a pressurizer safety valve. The L9-3 experiment is intended to simulate the important physical conditions following a loss-of-feedwater without scram transient hypothesized for future commercial PWRs conforming to the acceptance criteria proposed by the NRC.

Upon loss of feedwater to the steam generators in a PWR power plant, the heat transfer from the primary to the secondary system is degraded as the water inventory in the steam generators decreases. Normally the reactor will trip (insert control rods to shut down the reactor or scram on a signal of low feedwater flow or low steam generator level). In the absence of a scram, the steam generators will soon boil dry and most of the heat produced by the reactor core will be dissipated in the primary fluid, raising its temperature. The expansion of the primary fluid associated with its temperature rise at first compresses the vapor space of the pressurizer, forcing the relief and safety valves to open. Subsequently, the pressurizer will be filled with liquid water and the system pressure will continue to rise to a maximum when the volumetric relief flow rate equals the volumetric expansion rate of the primary fluid. It is this maximum pressure that constitutes one of the primary safety concerns of ATWS events.

According to PWR vendors' calculations,[4] a loss of feedwater ATWS yields one of the highest primary pressures among the initiating events mentioned earlier (the others are loss of load or rod withdrawal at zero power). With the exception of Westinghouse plants, such an accident in all presently operating or designed large commercial PWRs early in their life will result in maximum primary stresses exceeding the "Level C Service Limit as defined in Article NCA-2000 of Section III of the ASME Boiler and Pressure Vessel Code. Such stress levels can cause large deformations in areas of structural discontinuity or damages to components that will breach the integrity of the primary coolant pressure boundary, leading to a loss of coolant accident. The NRC's proposed regulation will limit the maximum primary stress to less than the "Level C Service Limit" in all components except in the steam generator tubes whose integrity may be evaluated based on a conservative assessment of the likely condition of the tubes over their design life. The LOFT L9-3 experiment is designed to achieve a peak primary pressure that will result in stress levels slightly below the "Level C Service Limit" in commercial PWR's. The L9-3 peak pressure will, therefore, be representative of the maximum expected pressure in commercial reactors that will be allowed under future rulemaking.

Another major concern of a loss of feedwater without scram accident is the long-term shutdown capabilities of PWR systems after the initial peak pressure has passed. According to the NRC staff, after evaluating the PWR vendors' submittals in response to an NRC request on ATWS analysis, long-term shutdown has not been adequately addressed by the vendors.[2] One reason is that the transient computer codes[5],[6],[7] used by the PWR vendors are no longer applicable when significant voids appear in the primary system after coolant loss from the relief and code safety valves; another is that the vendors have not clearly delineated the recovery procedures and the corresponding mitigating systems. The L9-3 experiment will explore one procedure for depressurizing the primary system by latching open the PORV and using the auxiliary feedwater system to reduce the temperature and pressure of the primary system to the point where high concentration boron solution can be injected into the system to permanently shut down the reactor. This procedure will be at least a first step in bringing the reactor to a stable shutdown condition after a loss-of-feedwater ATWS.

To address issues relating to system response and plant recovery procedures following a loss-of-feedwater ATWS in a commercial nuclear power plant, the following programmatic objectives have been established for the L9-3 experiment:

1. Provide experimental data for benchmarking PWR vendors' ATWS computer codes as required by the NRC proposed ATWS rule (USNRC SECY-80-409).

2. Evaluate alternate methods of achieving long term shutdown (without the insertion of control rods) following an ATWS event, to address concerns defined in the proposed NRC staff rule (Federal Register Vol. 46, No. 226).

TEST OBJECTIVES

To support the above programmatic objectives, several specific test objectives for the L9-3 experiment have been defined. In establishing these test objectives, it is realized that results from the LOFT experiment will not be directly applicable to the larger commercial plants. However, by application of the codes to LOFT results, it is expected that an assessment of the capabilities of the codes to predict important system response characteristics during an ATWS event in a commercial pressurized water reactor can be obtained. Therefore, the test specific objectives for L9-3 are:

1. To achieve a maximum primary system pressure that is several measuring standard errors above the code safety valve opening pressure setpoint but below 110% of the setpoint pressure.

2. To determine the transient reactor power by using available neutron flux instrumentation and measured core thermal-hydraulic parameters to assess the applicability of the point kinetics model used in predicting transient reactor power.

3. To determine the steam generator secondary dryout behavior and its effect on the primary system response characteristics.

4. To determine the two-phase and subcooled flow characteristics of the experimental pressurizer PORV and safety valve at high pressures ($\geq$ 17 MPa (2500 psia)).

REFERENCES

1. Anticipated Transients Without Scram for Light Water Reactors, NUREG-0460, Vol. 4, March 1980.

2. USNRC SECY-80-409, Proposed Rulemaking to Amend 10 CFR Part 50 Concerning Anticipated Transients Without Scram (ATWS) Events, September 4, 1980.

3. Anticipated Transients Without Scram for Light Water Reactors, NUREG-0460, Vol. 2, Appendix IV, April 1978.

4. Anticipated Transients Without Scram for Light Water Reactors, NUREG-0460, Vol. 2, Appendices XIV, XV, and XVII, April 1978.

5. Babcock & Wilcox Company, CADDS-Computer Applications to Direct Digital Simulation of Transients in PWRs with or without Scram, Report BAW-10098, Rev. 1, February 1978.

6. CENPD-107P, CESEC-Digital Simulation of a Combustion Engineering Nuclear Supply System, April 1974 and Suppl. 1 P, April 1974, Suppl. 1, Amendment 1 P, November 1975, Suppl. e, August 1975, Suppl. 4 P, December 1975, Suppl. 5 P, June 1976 on ATWS Model Modifications to CESEC.

7. Westinghouse Electric Corporation Report, LOFTRAN Code Description, WSAP-7878, Rev. 1, January 1977

L6-6 BORON DILUTION EXPERIMENT

The NRC has expressed concern regarding the conservatism of the calculational methods used by pressurized water reactor vendors in analyzing a boron dilution accident.

Dissolved boric acid (H_3BO_3) is used in pressurized water reactors to provide chemical shim. Since boron is an effective neutron absorber, its presence reduces core reactivity. However, when the boron concentration is reduced by the addition of water, positive reactivity is added. If the reactor is subcritical, this dilution brings the reactor closer to a critical condition and if continued long enough will produce a return to criticality.

The pressurized water reactor vendors do not address what happens if the reactor returns critical. Instead, they attempt to demonstrate that the operator has adequate time to act to prevent a recriticality.

To calculate the time required to reach criticality, the vendors assume uniform boron concentration throughout the system volume (perfect mixing). To provide some degree of conservatism, they only take credit for portions of the system volume in which there is a relatively high flow rate. The methods used by the vendors are discussed in more detail in References 1 and 2.

In Standard Review Plan 15.4.6,(3) NRC has taken the position that the operator should have 30 minutes in which to respond to a boron dilution from a refueling condition and 15 minutes from other conditions. This is the minimum time between an alarm, making the operator aware of the dilution and criticality.

In most vendors' analysis, a malfunction of the chemical and volume control system is assumed to cause the dilution. The operator is made aware of the situation by an alarm indicating this malfunction and in general, the operator does have adequate time to react.

In reality though, dilution events which have occurred to date have been the result of maintenance activities and not of operation or malfunction of the chemical and volume control system. The operator would be made aware of the situation by a high flux at shutdown alarm which would occur after a significant amount of reactivity had been added by the dilution. The time available for the operator to act to prevent criticality in this case would be much shorter.

Experiment L6-6 will simulate a boron dilution event from cold shutdown conditions. Results from this experiment will then be compared with the predicted system behavior using the uniform mixing assumption. This comparison will allow an evaluation of the conservatisms in the vendors' methods.

TEST OBJECTIVE

The programmatic and test specific objective of Experiment L6-6 is to:

> Provide NRC with data to assess the conservatism of analytical methods used by PWR vendors to demonstrate compliance with Standard Review Plan requirements regarding the minimum time to reach criticality in an inadvertent boron dilution accident.

REFERENCES

1. "System 80 Standard Safety Analysis Report (CESSAR)," Combustion Engineering, Inc., August 1973 (under review).

2. Trojan Final Safety Analysis Report, Section 15.2.4, "Uncontrolled Boron Dilution," Portland General Electric.

3. Standard Review Plan, Section 15.4.6, "Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant," Office of Nuclear Reactor Regulation.

L2-5 LARGE BREAK LOSS-OF-COOLANT EXPERIMENT

Test L2-5 is being conducted to address conservatisms in licensing criteria defined in the code of Federal Regulations (10CFR50 Appendix K).[1] Current licensing criteria limit fuel rod cladding temperatures to 2200°F. For most plants limited by this criteria, the peak temperature occurs during the reflood portion of the licensing calculation. While there are other limits that determine the maximum power a plant can operate at (i.e.. departure from nucleate boiling or hardware limits), the majority of plants (essentially all Westinghouse plants) are limited by the LOCA analysis. Initial estimates indicate that an increase of approximately 5 percent in power can be realized in these plants if the LOCA limit were eliminated.[2]

Previous large break experiments in LOFT have identified two areas where Appendix K assumptions may be overly conservative. The first conservatism is in the definition of "end of bypass" (para. I.C.1.c), where Appendix K assumes reflood cannot occur until countercurrent flow in the downcomer is predicted to occur. Previous LOFT tests have demonstrated that "flow channeling" in the downcomer will allow reflood to occur much earlier than assumed in Appendix K.

The second area of conservatism, identified by previous LOFT large break experiments, is the Appendix K criterion precluding the return to nucleate boiling (rewetting) prior to the end of blowdown (para.I.C.4.e). LOFT large break experiments L2-2 and L2-3 have demonstrated that system hydraulic behavior can lead to an early rewet of the fuel cladding (5-10 seconds after break initiation). This early rewet is significant in that not only does the rewet limit the peak cladding temperature during blowdown, but it also removes a significant amount of stored energy in the fuel rod. As a result, even if reflood is delayed due to the assumed ECC bypass, cladding temperatures during reflood will be much lower than currently predicted without rewet.

Although LOFT Test L2-5 will provide information relating to conservatisms in Appendix K reflood assumptions, the principal purpose of the test is to address the question of early rewet. Computer code calculations have indicated

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that rewet may not occur if fluid flow through the core during the initial 20 seconds of the transient is reduced sufficiently to cause a nearly stagnate condition in the core. This flow stagnation causes a reduction in heat transfer which has been calculated to prevent rewet. If this core thermal-hydraulic behavior can be confirmed in LOFT Test L2-5, then it will provide information needed to assess correlations used in the code predictions. The rewet issue can then be addressed by calculating LPWR response characteristics with the verified code correlations.

To address issues relating to early rewet, the following programmatic objectives have been established for the L2-5 experiment:

1. Provide experimental data to demonstrate that Appendix K assumptions result in a conservative prediction of peak clad temperature, even if core hydraulic conditions were to occur in a commercial reactor which precluded the early return to nucleate boiling (rewet).

2. Provide data to confirm that results from early LOFT large break experiments were not being significantly affected by external cladding thermocouples.

TEST OBJECTIVES

To support the above programmatic objectives several test specific objectives have been identified. These objectives are:

1. To determine if early core rewet occurs following a scaled LOFT 200% double-ended cold leg break with immediate primary coolant pump trip.

2. To provide data on core thermal response which can be used to evaluate computer code predictions and to compare with acceptance criteria in 10CFR50.46.

3. To determine system behavior and core thermal response during the reflood portion of a double-ended cold leg break experiment.

4. To evaluate cladding surface thermocouple effects during blowdown and reflood by comparing the responses of LOFT core centerline fuel thermocouples, external pellet thermocouples, embedded cladding thermocouples and cladding surface thermocouples.

REFERENCES

1. Code of Federal Regulations, Part 50, Appendix K, ECCS Evaluation Models, October 3, 1980.

2. J. E. Koske, LOFT Cost/Benefit Study, to be published.

L6-8 ANTICIPATED TRANSIENT EXPERIMENTS

The L6 test series[1] was developed to study transients in which a disturbance to plant equilibrium occurs, resulting in a reactor scram when the first safety system setpoint is reached Data from the tests will be utilized in evaluating the computer codes and analytical techniques used to predict anticipated transients, including anticipated transients without scram (ATWS).[2]

The intent of the Nuclear Regulatory Commission in considering anticipated transients is to ensure sufficient plant monitoring and control system capability to allow identification and recovery from conditions which could lead to an inadequate core cooling situation.[3] The need to consider anticipated transients in reactor safety evaluation is not that reactor protection and reactivity shutdown systems are unreliable, but that considering the relatively high rate they are challenged and the number of nuclear power plants in operation, an extraordinary high reliability is required.

The intent of the LOFT tests is to provide data which, through the verification of computer codes and analytical models, will contribute to an understanding of symptoms, events, and plant conditions leading to emergency or off-normal situations. In addition, the LOFT tests will provide information on the ability of reactor trip systems, engineered safety features, and manually initiated systems to perform their intended functions.

Three major programmatic objectives have been defined for the L6 test series which should contribute to a better understanding of commercial plant response characteristics over a range of anticipated transient events. These objectives are to:

1. Assist NRC in evaluating reactor transient analysis techniques used in reactor licensing by applying the same techniques to transients performed in the LOFT facility.

2. Demonstrate that LOFT results can be related to larger PWRs by providing data that can be compared to data obtained from commercial plants (Traceability).

3. Provide data for evaluating commercial plant instrumentation and control system response characteristics over a range of transients which could occur in a commercial plant.

TEST OBJECTIVES

The selection of the L6-8 transients was based, in part, on a review of Chapter 15 of Regulatory Guide 1.70 which defines eight categories of plant accidents and transients which must be analyzed by each nuclear power plant licensing applicant in its Safety Analysis Report. Typical initiating events for each of these categories of plant transients are also identified. To date, LOFT has conducted a series of LOCA and non-LOCA experiments that have included one or more transients in five of the eight general categories. In addition, several of the LOFT experiments have simulated one or more of the actual initiating events identified under each of the general transient categories in the regulatory guide. The L6-8 experiments have been developed to study those areas of transient reactor behavior which have not been previously considered in LOFT or in other experimental research facilities.

The specific objectives for each of the tests have been developed to support the general programmatic objectives discussed previously. It is expected that analysis, other than that required for data qualification, will not be needed to assess the degree of completion of these objectives.

Test L6-8B:

This test will investigate two different control rod assembly withdrawal rates in the LOFT facility. These transients fall into the general category of "Reactivity and Power Distribution Anomalies" defined in Chapter 15 of Regulatory Guide 1.70. This is the first time transients of this nature have been performed in any experimental research facility. The selection of these transients for performance in LOFT was based on (1) the need for an experimental data base for evaluation of reactivity initiated events in commercial plants, and (2) the recognition that LOFT is the only existing experimental facility capable of simulating both the nuclear aspects (reactivity feedback) and integral system response characteristics of a commercial nuclear power plant.

The following specific test objectives have been defined for the L6-8B transients:

1. Investigate plant response to a reactivity insertion event caused by the withdrawal of all four LOFT control rod assemblies.

2. Provide data which can be used to assess the applicability of kinetics models used to predict transient reactor power.

3. Evaluate the effectiveness of the plant protection systems (power and pressure trips) during a reactivity insertion event.

Test L6-8C:

The L6-8C test will evaluate operating procedures designed to minimize radioactive release to the secondary system following a steam generator tube rupture event. Two transients are currently planned. In the first transient (L6-8C-1), the LOFT operators will follow commercial plant operating procedures to bring the plant temperature and pressure down below the initial steam generator secondary pressure as quickly as possible (to stop the radioactive release to the secondary). This procedure will be accomplished using the pressurizer sprays while cooling down with the steam generator. The intent of this procedure is to depressurize the primary system as rapidly as possible while maintaining sufficient primary system subcooling to prevent the formation of steam voids. This procedure has been generally adopted by the industry because of the concern that void formation in the primary system can adversely affect operator control of the plant.

The second transient (L6-8C-2) will involve the depressurization of the primary system using the pressurizer sprays only. In this case, the primary system subcooling will not be maintained by cooling with the steam generator. As a result, voids are expected to form in the primary, and an assessment of the potential effects of the steam voiding on plant response and controllability will be obtained.

The following specific test objectives have been defined for the L6-8C transients:

1. Evaluate the effectiveness of currently defined operating procedures in mitigating the consequences of a simulated steam generator tube rupture event in LOFT (L6-8C-1).

2. Investigate the feasibility of an improved procedure to further reduce the potential for a radioactive release to the secondary following a steam generator tube rupture event (L6-8C-2).

3. Determine the effect on plant response of voiding in the primary system when the LOFT facility is subjected to a rapid cooldown without maintaining the primary system fluid subcooling (L6-8C-2).

L6-8D:

The L6-8D transient is characteristic of transients defined under "Increase in Heat Removal by the Secondary System" in Chapter 15 of Regulatory Guide 1.70. While several cooldown transients have been run in LOFT in the past, this transient is unique in that it will simulate a very slow cooldown similar to the cooldown transient that occurred at the St. Lucie plant in June 1980 (about 60°F/hr).[5]

This transient is designed principally to investigate the effect of steam voiding in the primary on pressurizer level response and plant controllability. (In the St. Lucie transient, the plant operators experienced difficulty controlling pressurizer liquid level when voiding in the vessel upper head occurred.)

The following objectives have been defined for this test:

1. Assess the effect of the formation of a steam bubble in the upper head of the LOFT facility on natural circulation and pressurizer level response characteristics.

2. Using available LOFT instrumentation, assess code capabilities to predict the plant primary and secondary response during a slow controlled coo'down (60°F/hr).

3. Evaluate the effecciveness of natural circulation as a core cooling mechanism at low plant decay heat levels.

REFERENCES

1. R. P. Jordan, LOFT Experiment Operating Specification Non-LOCE Baseline Test Series L6, NE L6 Series EOS, Rev. 1, September 5, 1980.

2. Technical Report on Anticipated Transients without Scram for Water-Cooled Power Reactors, WASH-1270, September 1973.

3. NRC Action Plan Developed as a Result of the TMI-2 Accident, Task II-F, May 1980.

4. Regulatory Guide 1.70, Revision 3, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, November 1978.

5. Power Reactor Events, Vol. 2, No. 4, July 1980.

A-376

L9-4 LOSS OF OFFSITE POWER ATWS

Anticipated transients without scram (ATWS) have been the subject of discussions and analyses within the nuclear industry since early 1969, and have been designated an unresolved safety issue by the Nuclear Regulatory Commission (TAP A-9).[1] The significance of ATWS in the evaluation of reactor safety is that some ATWS events could result in melting of the reactor fuel and the release of a large amount of radioactive fission products. The questions in contention concern whether the probability of such events is great enough to justify their consideration and if so, what are the consequences of various postulated events?

Despite differences in opinion within the industry, the Nuclear Regulatory Commission (NRC) considers the risk associated with ATWS events sufficient to justify their consideration.[2,3] Therefore, LOFT Test L9-4 has been developed to gain a better understanding of system response characteristics for a postulated ATWS and to determine the ability of existing analytical techniques to predict those response characteristics.

The selection of L9-4 as an ATWS initiated by a loss of offsite power was based on a review of transients discussed in WASH-1270[2] and in Chapter 15 of Regulatory Guide 1.70. Loss of offsite power was selected as the initiating event for L9-4 because it represents the most demanding test for validating point kinetics approximations used in the calculation of transient reactor power.[4]

To address issues relating to the consequences of a postulated ATWS, two major programmatic objectives have been defined for the LOFT ATWS experiments. These objectives are:

1. Provide experimental data for benchmarking PWR vendors' ATWS computer codes as required by the NRC proposed ATWS rule (USNRC SECY-80-409).

2. Evaluate alternate methods of achieving long term shutdown (without the insertion of control rods) following an ATWS event, to address concerns defined in the proposed NRC staff rule (Federal Register Vol. 46, No. 226).

TEST OBJECTIVES

To support the above programmatic objectives, several test specific objectives have been identified.[5] The test specific objectives for this test are defined as those which can be evaluated shortly after the conduct of the test. Analysis, other than that required for data qualification, will not be needed to assess the degree of completion of these objectives. The specific objectives for this test are:

1. To determine the effect of primary coolant pump operation on initial system response and peak pressure by comparing results from L9-4 (pumps tripped) with results from L9-3 (pumps running).

2. To evaluate the effectiveness of natural circulation as a cooling mechanism during an ATWS event involving trip of the primary coolant pumps.

3. To provide data of sufficient quality to evaluate the capabilities of the computer codes to predict the fluid conditions (temperature, pressure, and quality) in both the primary and the secondary systems and to evaluate the adequacy of point kinetics assumptions used in the prediction of reactor power levels.

4. To evaluate the adequacy of the proposed recovery procedure defined for this experiment.

REFERENCES

1. Unresolved Safety Issues Summary (Aqua Book), NUREG-0606, Volume 3, No. 2, May 15, 1981.

2. Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors, WASH-1270, September 1973.

3. Anticipated Transients Without Scram for Light Water Reactors, NUREG-0460, Volume 1, April 1978.

4. P. Kuan, Candidates for the Contingency ATWS Test in A Recent NRC Research Proposal for the LOFT Test Program, PK-8-81, March 4, 1981.

5. P. K[illegible]n, Loss-of-Fluid Test (LOFT) L9-4 Anticipated Transient Without Scram (ATWS) Loss of Off-Site Power Experiment, PK-25-81, August 21, 1981.

A-379

L2-6 LARGE BREAK LOSS-OF-COOLANT EXPERIMENT

LOFT Test L2-6 will be a double-ended cold leg break with an initial power level corresponding to a maximum linear heat generation rate of 12 kw/ft. The test has been separated into two parts designed to address different issues relating to the safety of nuclear power plants. The first part of the test, designated L2-6A, lasts from rupture initiation until a system pressure of 600 psi (corresponding to the normal accumulator injection pressure) is reached. The second part of the test, designated L2-6B, begins at a system pressure of 600 psig. In this part of the experiment, however, normal accumulator injection will be delayed. The delayed accumulator injection will allow core temperatures to heat up to the point where fuel ballooning and rupture in the center fuel module are expected to occur.

The objectives and issues being addressed in each portion of the test are discussed in the following sections. While specific objectives for each part of the L2-6 experiment have been defined, it is not clear at this time that both sets of objectives can be met. Should future planning analysis show a conflict in the objectives, the objectives for the L2-6B portion of the transient will be considered the primary objectives of the test. The L2-6A objectives will then be considered secondary objectives, to be met only if they do not interfere with the successful completion of the L2-6B portion of the experiment.

Experiment L2-6A

Test L2-6A is being conducted to address conservatisms in licensing criteria defined in the Code of Federal Regulations (10CFR50, Appendix K).[1] Specifically, LOFT Test L2-6A is designed to determine whether critera precluding the return to nucleate boiling (rewet) prior to the end of blowdown (para. I.C.4.e.) are justified.

Previous large break experiments in LOFT (Tests L2-2 and L2-3) have demonstrated that system hydraulic behavior can lead to an early rewet of the fuel cladding (5-10 seconds after rupture). Test L2-6A is being conducted to determine if hydraulic condition can be established in the core which will present the early rewet from occurring. In addition, L2-6A will be conducted with the center fuel module initially pressurized to 600 psi. This pressure corresponds to end-of-life fuel conditions and represents the maximum internal (cold) fuel pressure expected in a commercial plant. This combination of fuel pressure and fluid hydraulic conditions is therefore expected to produce the most severe combination of conditions which could lead to fuel damage in a commercial pressurized water reactor.

The major programmatic objective for this test is, therefore, to demonstrate that for an end-of-life fuel pressure of 600 psi and core hydraulic conditions which limit the potential for early rewet, acceptance criteria defined in 10CFR50.46 will be met.

Test L2-6A Objectives

To support the above programmatic objective, the several specific test objectives have been defined. These test specific objectives are:

1. To determine if early core rewet occurs during a scaled LOFT 200% DECL break with primary coolant pump trip.

2. To determine whether fuel damage occurs during the initial rapid fuel cladding temperature rise (during the first 10 to 15 s of the transient) by observing available LOFT core instrument response.

3. To evaluate any cladding surface thermocouple effects during blowdown by comparing the responses of LOFT core internal fuel thermocouples, embedded cladding thermocouples, and cladding surface thermocouples.

Experiment L2-6B

LOFT Experiment L2-6B is being conducted to address specific issues that will be considered in future rulemaking proceedings dealing with degraded core cooling. These proceedings will determine to what extent, if any, reactor plant designs and safety analyses should consider reactor accidents beyond those addressed in the current design basis accident approach, including a range of loss-of-cooling, core damage, and core-melt events.

In a preliminary copy of the notice of proposed rulemaking,[2] several specific areas were identified as needing further consideration. Among these was the need to identify those aspects of accidents resulting in fuel damage that are"...sufficiently unknown or uncertain as to impede design and analysis of mitigating systems, thus requiring additional research or experimentation." To address this need, two primary programmatic objectives have been defined for Experiment L2-6. These objectives are to:

1. Contribute to severe core damage rulemaking by experimentally validating fuel rod balloon and burst models and the separate effects data base in a large integral nuclear facility.

2. Determine primary system fission product transport characteristics in a realistic PWR environment during an accident scenario involving fuel cladding damage.

Test L2-6B Objectives

To support the above programmatic objectives, several test specific objectives have been identified. The test specific objectives for this experiment are defined as those that can be evaluated by (1) interpretation of experimental measurements immediately after the test, or (2) posttest destructive examination of the center fuel module after removal from the vessel. Successful completion of these test specific objectives will provide information needed to address the overall programmatic objectives. The specific objectives of L2-6B are:

1. To conduct a large break experiment that is predicted to maximize ballooning and core flow blockage in the LOFT

central bundle.

2. To determine fuel rod burst patterns and the temperature and pressure at which burst occurs when cladding ruptures in the high-alpha temperature zone (1500-1600°F).

3. To investigate the system thermal-hydraulic response and central bundle coolability when delayed ECC injection is initiated following the severe ballooning of the central bundle.

4. To assess the effectiveness of normal ECC injection by comparing the core thermal response of L2-5 (normal ECC injection) and L2-6 (delayed ECC injection).

5. To investigate fission product transport in the primary system following fuel rod burst in the central bundle.

REFERENCES

1. Code of Federal Regulations, Part 50, Appendix K, ECCS Evaluation Models, October 3, 1980.

2. Degraded Cooling Rule Making, SECY-80-357, September 9, 1980.

3. M. L. Picklesimer, et. al., F2 Fuel Bundle Instrumentation Arrangement, Memo of Conversation, September 15, 1981.

4. Regulatory Impact of Nuclear Reactor Accident Source Term Assumptions, NUREG-0771, June 1981.

5. M. L. Pickesimer, Chemical Form of Iodine Release Memorandum, September 5, 1980.

6. Technical Bases for Estimating Fission Product Behavior During LWR Accidents, NUREG-0772, June 1981.

APPENDIX XVII
NUCLEAR DATA LINK

The following pages A-385 thru ________ has been deleted as 1.

DELETION

APPENDIX XVIII

Additional Documents Provided for ACRS' Use

1. Memorandum, E. F. Goodwin to R. F. Fraley, Revised Proposed NRR Agenda Items for the March, April, and May 1982 ACRS Meetings, February 2, 1982

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

JAN 29 1982

APPENDIX XIX
INSTRUMENTATION FOR DETECTION OF
INADEQUATE CORE COOLING IN
PRESSURIZED WATER REACTORS

MEMORANDUM FOR: Chairman Palladino

FROM: William J. Dircks, Executive Director
for Operations

SUBJECT: INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING IN PRESSURIZED WATER REACTORS

This is in response to your memorandum of January 19 concerning our plans to address the issues considered at the Commission's meeting of January 8, 1982 on the subject instrumentation.

We have scheduled a two-day NRC/Industry meeting for mid-February. The level measurement suppliers are being asked to give presentations assessing the performance of their proposed instrumentation systems for a broad spectrum of accident scenarios. These presentations are being specifically designed for response to the issues discussed at the January 8, 1982 meeting. The vendors have been requested to address the points raised in Enclosure 1 to this memorandum. We will also invite representatives of licensed plants to participate in the exchange with the suppliers to assure adequate attention to the operational aspects of the issues that have been raised.

Subsequent to our meeting with licensees and vendors, the staff will discuss.. the requirements and proposed designs with the Committee to Review Generic Requirements (CRGR) of the NRC and seek that committee's guidance.

We expect that an agenda will then be established for detailed industry and staff presentations to the ACRS subcommittee and full committee in March. These presentations will reflect guidance received from the CRGR. By that time the staff's technical assistance contractor, Oak Ridge National

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UNITED STATES
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A-387

Laboratory, will also have completed its review of the suppliers proposed systems. Following our discussions with the ACRS, we expect to have a recommendation for the Commission's consideration by the end of March, taking into account the reviews of the CRGR, the ACRS, our contractors and the staff.

William J. Dircks, Executive Director
for Operations

Enclosure:
Additional Instrumentation for
ICC in PWRs

cc: Commissioner Gilinsky
Commissioner Bradford
Commissioner Ahearne
Commissioner Roberts
OGC
OPE
SECY

ENCLOSURE

ADDITIONAL INSTRUMENTATION FOR INADEQUATE CORE COOLING IN PWRs

Please evaluate the capability of your existing and your proposed additional instrumentation for detection of inadequate core cooling in light of various types or classes of accident sequences and justify the sufficiency of the spectrum of accidents considered. Identify what information would aid the operators for the various accident sequences, and show how the various elements of your proposed complement of instruments contributes to supplying that information. Summarize the accident scenario presentation by describing the accident scenarios for which your proposed water level indicators will provide reliable information and those for which your water level indicators would not be useful or would give misleading or ambiguous readings. For these latter cases, what specific instructions, training or procedures would be provided to operators to prevent them from misinterpreting ambiguous indications and being misled. Include discussions of the integration with control room display of the measurements. Then explain how the symptom-oriented operating procedure guidelines will be integrated with the measurements and displays for the identified scenarios. What steps have you taken in system design and in procedure development to assure that the instruments provide complementary information and unambiguous guidance to the operators.

Discuss the design objectives for your proposed water level measurement system, and the bases of your selection and evaluation of specific instrumentation to measure water level. Summarize the other types of instrumentation you considered and the reasons they were rejected.

It has been suggested that water level indicators are superfluous to other inadequate core cooling indicators in PWRs. Please identify those parts of the accident scenarios where the information from water level indicators would be unique. What specific actions might be taken by operators because of the level measurement that would not otherwise be taken. Describe where

the water level information merely verifies other signals which are the basis for operator actions and those instances where it provides unique diagnostic information which is significant input to operating decisions. Contrast these various contributions to the potential ambiguities that the proposed level measurement systems may create. On balance, does your system help or hurt safety? Would you rely on it if you were an operator?

Discuss the quality of the information to be provided by the level monitoring instrumentation; i.e., what are the error bands under various circumstances when following the course of an accident. In particular, identify possible ranges of uncertainty when approaching core uncovery in times of rapid depressurization, rapid flow changes, reactor coolant pumps operating, ECCS pumps operating or severe core damage (flow blockage) and relate the significance of the uncertainty to interpretation and response to the event. Address the possibility and significance of circumstances where there could be an indication of water above the core while the core is actually partially uncovered or while local or global conditions of inadequate core cooling (temperature rise or sustained high temperature of the fuel) exist within the core.

Describe the procedures you recommend for implementation of an installed system, such as calibration and testing requirements, debugging, verification of displays, and operator training. When do you propose that the plant specific NRC implementation review be conducted.

Describe the development and verification testing programs for the proposed instrumentation. Discuss the results and how they have been used in the design evolution of the proposed instrumentation. Discuss conclusions from any test programs and show how the results demonstrate the capabilities and limitations of the proposed water level measurement systems.

Discuss qualification requirements and status of the final ICC monitoring instrumentation systems.

r. Ed Scherer
mbustion Engineering, Inc.
00 Prospect Hill Road
ndsor, Connecticut 06095

JAN 28 1982

ar Mr. Scherer:

e NRC staff has been reviewing the schedules and the status of your program r meeting TMI Action Item II.F.2, the requirements for inadequate core oling measurement systems in light water reactors. Our review has involved series of discussions with the industry, the ACRS and the Commission. The RS and representatives of level measurement suppliers made presentations to e Commissioners on January 8, 1982.

ring the course of these discussions, a number of important questions have en raised. We have decided it is necessary to meet again to better articulate e purposes of inadequate core cooling measurements, to obtain a better derstanding of the industry's general approach to these measurements, actor water level indicators in particular, and to provide additional insight to the basis for your design selections. We invite you to meet with the aff on February 16 and 17 starting at 9:00 a.m. We are also inviting presentatives of the applicable owners groups to participate in the meeting.

a later date, you may also be asked to meet with the NRC Committee to vi Generic Requirements. It is our expectation that an agenda would then established for detailed industry and staff presentations to the ACRS and s combined Electrical Systems and ECCS subcommittees in March.

the meeting with the staff and other industry representatives on bruary 16 and 17, we request that you structure a formal presentation to dress the points raised in Enclosure 1. The agenda for your presentation d others is provided in Enclosure 2. Please call L. S. Rubenstein of my aff if you have any questions regarding this subject.

Sincerely,

Original Signed by:
Roger J. Mattson

Roger J. Mattson, Director
Division of Systems Integration
Office of Nuclear Reactor Regulation

closures:
Additional Information for
ICC in PWRs.
Preliminary Agenda

IDENTICAL LETTERS TO:
Westinghouse - PRahe
B&W - JTaylor
NNC- LKornblith

A-391

ENCLOSURE 1

ADDITIONAL INSTRUMENTATION FOR INADEQUATE CORE COOLING IN PWRs

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Discuss qualification requirements and status of the final ICC monitoring instrumentation systems.

ENCLOSURE 2

PRELIMINARY AGENDA

NRC/INDUSTRY MEETING

SUBJECT: Design, Review, and Implementation Status of Reactor Vessel Level Measurement Systems - Instrumentation for Detection of Inadequate Core Cooling

Date: February 16 and 17, 1982

		Time
1.	Opening Remarks - R. Mattson, L. Rubenstein, NRC	30 min.
2.	Vendor Presentation - General Content per enclosure	2 hrs. each
3.	Review Status of Proposed Systems - NRC staff	1½ hrs.
4.	Research Status - ORNL	30 min.
5.	Additional Remarks - Vendors and Licensees	1 hrs.
6.	Closing Comments - NRC staff	

Note: Roger Mattson and Lester Rubenstein will co-chair the meeting. Requests to ask questions of any of the vendor, licensee or contractor attendees will be entertained at any time and ruled on by the Chairmen.