29 FEB 1988

Councilwoman Barbara A. Risacher
County Council of Harford County, Maryland
20 West Courtland Street
Bel Air, Maryland 21014

Dear Ms. Risacher:

This letter is in response to your January 22, 1988 letter in which you expressed concerns regarding the Peach Bottom Atomic Power Station. In that letter you requested our opinion regarding the potential for failure of the Peach Bottom containment in the event of a severe accident and deficiencies in management control at the plant.

With regard to the first item, results of NRC sponsored research indicate that there are some low probability severe accident sequences for which the integrity of the Peach Bottom containment could be seriously challenged. The studies indicate that the severe accident conditions for which containment failure is predicted have a very low probability of occurrence and the overall risk of plant operation satisfies the safety goals established by the Commission. Nonetheless, the containment is a principal element in terms of the defense-in-depth philosophy applied to nuclear power plant design and operation, and the Commission has assigned high priority to the assessment of methods for improving containment reliability under postulated severe accident conditions. The NRC Offices of Research and Nuclear Reactor Regulation are currently performing studies in this area.

The Commission's goal is to minimize the overall risk of plant operation and a careful, methodical approach must be used in addressing the containment issue. Care must be taken not to require changes that might reduce risk in one area while causing risk to increase disproportionately in another area. Also, in terms of reducing overall plant risk, the most effective use of resources may be in reducing the probability of initiating events. Attachment I describes the NRC Mark I Containment Performance Program Plan. This program is being implemented on a high priority basis and I believe will provide a firm and timely basis for deciding an appropriate course of action. An interim report on these activities is due to the Commission in April 1988. This report will address differences in existing risk studies and indicate whether existing analyses justify changes in the Mark I containment systems or operating procedures in the near term. A final report to the Commission is scheduled for August 1988. In summary, based on existing studies, the calculated failure probability for the Peach Bottom containment in the event of certain severe accident scenarios does not in itself constitute an unacceptable risk to the public health and safety. Nonetheless, the Commission, consistent with its defense in depth philosophy, is pursuing methods for improving containment reliability and reducing overall risk.

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We too are concerned about the management controls at Peach Bottom. This was an important factor in shutting the plant down in 1987. Your second concern gave some examples of the types of problems of a recurring nature which we have identified. We met with PECo to discuss some of these problems in the Health Physics and Security areas on February 26, 1988. Our review of the PECo response to the shutdown order continues following PECo's February 12, 1988 submittal of section 2 of the restart plan. We will be holding additional public meetings to receive your specific comments before we conclude that review.

An identical letter is being sent to Mr. Habern Freeman.

Sincerely,

Original Signed By
WILLIAM T. RUSSELL

William T. Russell
Regional Administrator

Attachment:
As Stated

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MARK I file

# POLICY ISSUE
## (Information)

December 8, 1987

SECY-87-297

For: The Commissioners

From: Victor Stello, Jr.
Executive Director for Operations

Subject: MARK I CONTAINMENT PERFORMANCE PROGRAM PLAN

Purpose: To present staff plans to resolve issues relating to the performance of MARK I containments during severe accidents.


Summary: In this paper the staff proposes a plan to effect closure of generic MARK I containment performance issues. The plan stems from a staff judgment that MARK I containments can have an improved level of mitigation under severe accident conditions. For those issues for which sufficient information exists, closure is to be effected by interim recommendations to the Commission in April 1988, and final recommendations in August 1988. For those issues for which sufficient information does not exist to effect closure, the severe accident research program will be used to provide bases for potential future recommendations.

Assessments of accident sequences indicate that there is substantial safety margin in the ability of MARK I containments to attenuate accidentally released fission products. There are, nevertheless, some low probability severe accident sequences for which the integrity of the containment function can be seriously challenged.

The key issue is reasonable assurance of the capability of containment systems to mitigate the consequences of core melt accidents for moderate to low probability sequences. This issue should be viewed in terms of defense-in-depth in that it involves striking a balance between accident prevention and mitigation. The plan is intended to achieve



Contact: J. Hulman, RES
492-8016, 443-7622


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4 pp.

regulatory resolution of containment performance issues starting with Boiling Water Reactor (BWR) MARK I containments in 1988. The plan includes development and application of criteria for judging of containment performance.

<u>Discussion:</u> The staff briefed the Commission on a plan for closure of severe accident issues, including matters relating to BWR MARK I containments, on July 15, 1987. At this briefing, the staff indicated its intent to pursue an integrated approach to the resolution of severe accident issues. Included were the Individual Plant Examination (IPE) program, a containment performance program for each of the various containment types, a program to improve plant operations, and a program to provide guidance on severe accident management strategies. In addition, severe accident/source term and risk reassessment research programs support the integrated program.

The closure plan for MARK I containments calls for a two step process; 1) an NRC staff, researcher and industry identification and narrowing of technical issues, and 2) a staff evaluation process. The issues would include those associated with core melt phenomena, containment failure modes, and those associated with the efficacy of potential improvements. Many analyses of MARK I containment performance have been done by the staff, staff contractors and industry analysts. This work will form the primary bases for issue identification. To aid in narrowing and focusing issues, additional in-vessel and ex-vessel core melt progression calculations are to be made and related experimental data are to be assessed. The staff evaluation would serve to eliminate some generic issues as not sufficiently important to consider further, to undertake research to provide sufficient information to resolve other issues, or to recommend regulatory initiatives. It is the identification and narrowing of issues, focusing related research, and assessing whether improvements are justified that the staff will pursue in this program.

An interim report will be provided to the Commission in April 1988. The report will discuss the major areas of agreement and disagreement between analysts and researchers on the important issues, and will indicate whether analyses at that time justify recommendations for near-term improvements to MARK I containments. A final report for MARK I containments is scheduled for August 1988.

The containment performance effort is being carried forward by RES in coordination with NRR. The philosophy for an approach to the evaluation of MARK I issues is described in a memo from T. E. Murley to V. Stello, Jr., dated June 29, 1987 (Enclosure 1). This philosophy was used as the basis for the staff severe accident discussion with the Commission on July 15, 1987. Enclosure 2 is the staff's plan for resolution of MARK I issues. A description of staff plans for integration and closure of all severe accident issues is scheduled to reach the Commission in April. Enclosure 3 illustrates the relationships between the primary MARK I tasks, and indicates several important milestones. This effort will be coordinated with the anticipated utility and staff IPE efforts, and will use such information as can be provided from utilities.

While recognizing the importance of individual plant variations, the staff has also recognized that there are potential severe accident vulnerabilities that have a common character within a class of plant containment systems. This recognition has evolved from severe accident research and plant evaluation programs both here and abroad, including findings from numerous probabilistic risk analyses starting with the Reactor Safety Study and including the recent draft NUREG-1150. For BWR MARK I containments in particular, these vulnerabilities are reflected in relatively high estimates of the probability of containment failure, given a core melt (also referred to as the conditional containment failure probability, CCFP). Staff sponsored research presented in draft NUREG-1150 indicates that this conditional probability is highly uncertain, but could be quite high for MARK I plants. Industry sponsored research, on the other hand, has provided estimates of CCFP for two reactors (Peach Bottom and Vermont Yankee) at less than 10 percent. The staff's judgment is that these discrepant views are unlikely to be fully reconciled soon. In view of the Commission's defense-in-depth philosophy, therefore, the staff believes it is prudent to examine ways to improve the capability of MARK I containments to mitigate the potentially large fission product releases that could result from outlier accident sequences.

Our examination of these differences in expected MARK I containment performance has resulted in two conclusions. First, many technical differences may be narrowed by further discussions among staff, researchers, industry representatives and interested members of the public. The discussions and any regulatory decisionmaking can be facilitated significantly by short term analyses and assessments of existing experimental activities related to BWR MARK I core melt phenomena and containment response. Second, some residual differences are likely to remain that only answers from a relatively long term research program can provide.

In an August 11, 1987, memo from S. J. Chilk to V. Stello, Jr., the Commission requested (M870715A) an assessment of whether or not additional resources for this activity could be used effectively. Resources were budgeted for FY 88 and subsequent years for related activities in the recent RES budget submittal. No additional resources are considered necessary for FY 88.

In summary, the approach identified in this paper is expected to result in both improvements in our understanding of the performance of MARK I containments during severe accidents, and in the identification of potential design and operational improvements. BWR MARK I containments are to be assessed by the end of FY 88, assessments of the other containment types are to be completed by the end of FY 89. A report to the Commission with interim MARK I recommendations is scheduled for April 1988. An integrated plan for effecting closure of severe accident issues for all plants is also scheduled for submission to the Commission in April 1988. A final report on MARK I containments is scheduled for August 1988.

James M. Taylor for
Victor Stello, Jr.
Executive Director for Operations

Enclosures

1. June 29, 1987 memo from T. E. Murley to V. Stello, Jr.
2. Program Plan
3. MARK I Key Activities & Milestones

Contact: J. Hulman, RES
492-8016, 443-7622

ENCLOSURE 1

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

JUN 29 1987

MEMORANDUM FOR: Victor Stello, Jr.
Executive Director for Operations

FROM: Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

SUBJECT: PROPOSED COURSE OF ACTION ON BWR MARK I CONTAINMENT

Your memorandum of April 20, 1987 directed NRR to determine a recommended course of action with regard to earlier proposals for an initiative to enhance BWR containment performance in the event of a severe core damage accident.

The staff has for some time recognized the potential vulnerability of BWR Mark I containments under certain severe accident conditions (see e.g., NUREGs 1079 and 1150) and, as a result, has studied means for reducing the Mark I containment failure probability. Last year the staff developed a set of proposed generic improvements with the general intention of reducing the conditional probability of Mark I containment failure during severe accidents. It was thought that, if these improvements were implemented, it would be unnecessary for these BWR plants to have containment performance evaluated as part of the Individual Plant Examinations (IPE).

In the intervening time since the generic improvements were put forward there have been several discussions among the staff, industry groups and the research community. The Reactor Risk Reference Document (NUREG 1150) was completed in February 1987 as well. The conclusion that seems to have emerged from these activities is that there is no clear consensus on whether the Mark I generic improvements are needed, whether the cost estimates are realistic, and whether the proposed improvements would be effective in significantly reducing risks. After reviewing these matters, I have concluded that a more comprehensive approach to this issue should be taken. The approach outlined below is not intended to delay clear safety improvements but rather to ensure we look at all reactor types and understand those areas where we are most likely to attain safety improvements.

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14 pp.

In examining the broad question of how to reduce the risks of severe accidents, the following three areas must be considered.

1. IMPROVED PLANT OPERATIONS

Every safety study since WASH 1400 has shown the sensitivity of risk to human errors. Our own analysis of operating experience confirms the importance of reducing maintenance, surveillance, testing and control room errors. Thus, an overall approach to this issue must include a program to improve plant operations and should consider at least the tasks below:

(a) Continued improvement of the SALP program;

(b) Regular reviews by senior NRC managers to evaluate those plants that may not be meeting NRC and industry standards of operational performance;

(c) Diagnostic Team Inspections to probe further the performance of those plants above;

(d) Regulatory actions to improve operational performance where it has fallen below expected standards;

(e) Improved Technical Specifications;

(f) Continued improvement of Emergency Operating Procedures (EOPs); and

(g) Expanding EOPs to include Severe Accident Procedures.

2. COMPREHENSIVE SEARCH FOR SEVERE ACCIDENT VULNERABILITIES

The Severe Accident Policy Statement contemplated a program of Individual Plant Examinations (IPEs) that would be a systematic approach to examine all plants for possible significant risk contributors. The staff has been working with the IDCOR industry group to develop the IPE methodology and has reached conclusion on a proposed program. The IPE program will have to be integrated with the improved operations program and with the containment performance research program below.

3. CONTAINMENT PERFORMANCE RESEARCH

The assessment of containment performance during severe accidents is a very difficult problem, and years of research have not yielded a consensus on what improvements are needed, if any. We should anticipate there will be the need for a long-range, continuing research program to assess the challenges to containments, to evaluate potential improvements, and to continue improving our understanding of source terms. Within this long range program there should be near-term results where the weight of technological evidence supports recommendations for containment improvements. The BWR Mark I would be one area targeted for near-term results. Clearly, this research effort must be integrated closely with the IPE program which will be examining accident vulnerabilities that could threaten containment integrity at specific plants.

The comprehensive program for reducing severe accident risks outlined above has not been fully developed. A schematic portrayal is shown in the attached figure. When developed and implemented I believe the program should lead to closure of the severe accident issue. Nonetheless, elements of the containment performance research program and the improved plant operations program will no doubt extend well into the future as we gain more research knowledge and more operating experience.

In keeping with the intent of the reorganization that RES develop resolutions for generic safety matters, I suggest that RES develop, with NRR guidance and support, the overall program outlined above. I further suggest that RES develop an interim response to the Commission's request for an options paper (February 9, 1987, memo from Chilk to Stello). This interim response would provide an outline of the program discussed above and would provide schedules for implementing key parts of the program such as IPE and containment performance evaluations. Finally, because of the importance of this issue, I will continue to work closely with the Director, RES, to coordinate the overall guidance for these activities. Similarly, the NRR and RES staffs will work closely on this program.

Original signed by
Thomas E. Murley

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc w/encl.
E. Beckjord, RES

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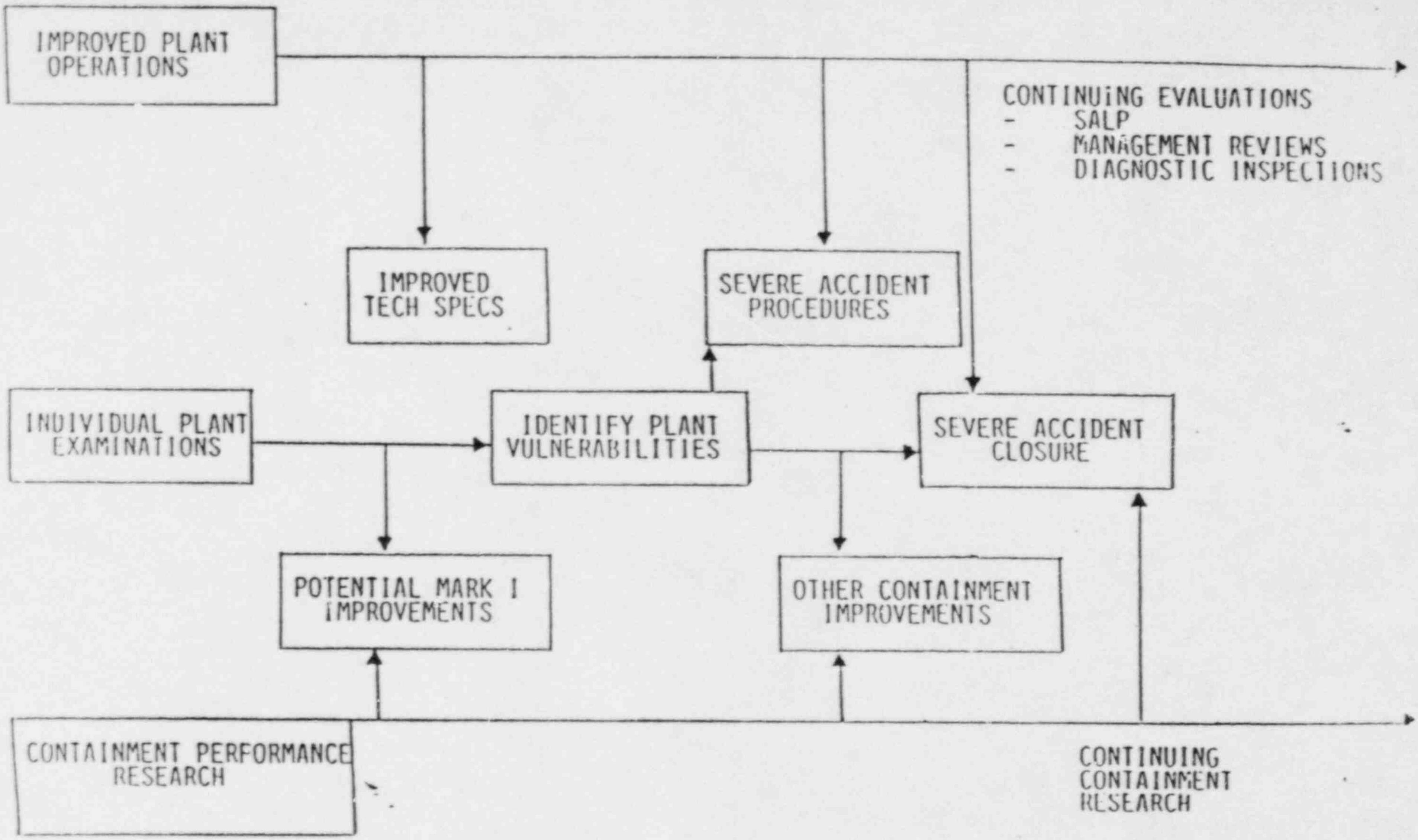
# SCHEMATIC SEVERE ACCIDENT PROGRAM



# MARK I CONTAINMENT PERFORMANCE PROGRAM PLAN

INTRODUCTION - The ability to mitigate the consequences of accidents is a function of the containment systems that are provided at all U. S. light water reactors. One class of containments is referred to as MARK Is, which have been used with 24 licensed BWR reactors. Although all U. S. light water reactors have containments designed to safely attenuate the energy that would be released in a loss-of-coolant accident in which safety systems would function to supply cooling water, MARK I containments have among the smallest internal volumes. This relatively small volume is offset, for some accidents, by a pressure suppression water pool. Such volumes and suppression systems are important if safety systems do not function properly and a large pressure rise ensues from releases of gases such as hydrogen and concrete ablation products. In addition, MARK I containments have no deep concrete slabs or water pools directly beneath their reactors. As a result, for many severe accidents MARK I containments may be viewed as potentially more susceptible to containment failure than other containment types.

The designs of these containments consider external events (such as earthquakes and tornadoes), while the containment temperature and pressure design bases are determined by a postulated design basis loss of coolant accident (LOCA) in which operation of the emergency core cooling system (ECCS) would prevent a core melt from occurring. The peak MARK I containment pressure associated with such a postulated LOCA has been estimated as high as about 57 psig.

Despite this containment design basis which would not result in core melting, the radiological consequences of a substantial core melt are nevertheless postulated in accordance with the provisions of 10 CFR 100.11. This postulation is used to assure the adequacy of certain plant features such as containment leak tightness and fission product filter systems, as well as the adequacy of the reactor site. The temperature and pressure conditions associated with a core melt accident are not part of the containment design bases. There is some assurance, however, that existing containments are capable of surviving the temperature and pressure conditions associated with some severe accidents, as well as for arrested core melt accidents. The TMI accident is an example which represents a partial core melt accident that was arrested prior to reactor pressure vessel failure, and was one in which the containment was not failed. Furthermore, there is an expectation that some containment failure events may result in significant fission product attenuation in adjacent plant buildings.

Studies of examples of various containment types under beyond design basis loading conditions (NUREG-1079) indicate survival at load levels of 2 to 3 times design basis LOCA loads and at elevated temperature conditions. Although only a few detailed structural analyses of MARK I containments have been attempted, inferences and extrapolations from design assessments and testing on scale models of containments and penetrations at Sandia National Laboratory confirm these higher failure pressure conclusions. Such confirmation, however, assumes containment isolation devices (including seals) isolate and do not fail.

PRA assessments to date of some MARK I plants indicate the initiator for the most risk significant accident may be a station blackout (SBO) event. One contemporary MARK I risk assessment (draft NUREG-1150) indicates an SBO is dominant, and the probability of core melt accidents may be very low. Other assessments for a limited number of MARK I plants do not confirm the low probability conclusion. It is important to note, therefore, that risk assessments of other MARK Is could identify other risk significant initiators and substantially different risk levels.

A generic issue, A-44, dealing with SBO events is close to completion. The objective of a proposed rule change (51FR9829) is to provide assurance that the probability of core melt arising from station blackout will be at or about $10^{-5}$ per reactor year or less. However, the rule change may not eliminate the event as a potentially dominant one at some MARK I plants, nor eliminate concern over the ability of such a containment design to mitigate such accidents. Furthermore, analyses of the characteristics of other severe accidents indicate the potential for generic concerns over the ability of the BWR MARK I containment type to mitigate the consequences of such events. Stated another way, it is not clear that the balance between accident prevention and mitigation called for in the Severe Accident Policy Statement for the various combinations of reactor types and containments has been achieved.

<u>CHALLENGES</u> - There are a number of potentially important challenges to MARK I containments. These are:

1) Containment bypass (including failure to isolate containment on demand, suppression pool bypass, and interfacing system LOCAs);
2) Early overpressure or overtemperature failures (including sequences involving melt quenching in-vessel, direct containment heating, and noncondensable gas generation and potential ignition);
3) Missiles from rapid steam pressures;
4) Core debris attack on the steel containment liner resulting in liner melt through;
5) Later overtemperature or overpressure failure; and
6) Basemat penetration.

Recent MARK I assessments have identified early overpressure (2), core debris attack on the steel liner resulting in liner melt through (4), and later overtemperature and overpressure (5) failures as the primary challenges. The likelihoods of early failure and liner melt through are areas of controversy among some analysts. The core melt progression phenomena associated with accident sequences which could lead to such challenges, therefore, are also important issues requiring better understanding.

<u>CONTAINMENT FAILURE MODES</u> - Containment response to challenges can have several outcomes. These can range from relatively little leakage to large scale failures. Large scale failure modes can generally be of two types. One type is a slowly developing breach of containment; e.g., a progressive failure of gasket material around a containment penetration such as an equipment hatch, a small structural failure of a suppression pool vent pipe expansion bellows, or a structural tear in the steel liner of a concrete containment. The other type of failure involves a very rapid depressurization such as would be displayed by the catastrophic rupture of steel containment.* The locations of predicted pressure induced

*Only two MARK I containments are of reinforced concrete construction; Brunswick Unit 1 and Unit 2.

structural failures for the MARK I containment NUREG/CR-3653) are in the drywell at either the knuckle between the upper cylinder and lower spherical section, or at the drywell head. Failure in the wetwell air space or suppression pool have also been postulated. (Leakage at the drywell head prior to failure in the wetwell airspace has been identified.)

POTENTIAL IMPROVEMENTS - A small number of relatively low-cost improvements have been assessed by the staff, its contractors, two utilities (Vermont Yankee and Boston Edison), and IDCOR. These improvements may substantially mitigate potential offsite releases. Some of these potential improvements are;

a) Hydrogen control - In normal operation MARK Is are inerted by replacing much of the oxygen in the containment atmosphere with nitrogen. Both the time MARK I containments are allowed not to be inerted, and the ability to keep such containments inerted during long duration station blackout sequences, have been questioned. Information provided by the Vermont Yankee licensee indicates that the relatively brief periods of time required for inerting and deinerting during startup and shutdown periods may be acceptable. An analysis of improvements proposed for Pilgrim by Boston Edison indicates improved nitrogen supplies are potentially warranted for long term accident sequences.

b) Containment spray - During a station blackout, power for pumping water to containment sprays or the vessel would not be available. One proposal has been to cross-tie an existing diesel powered fire pump to the water system for use in the vessel to prevent core melt, or for containment spraying if vessel failure has occurred. This proposal has been found technically feasible by IDCOR and the Vermont Yankee and Pilgrim licensees. An alternate (Hope Creek) is to provide an external valve on the Reactor Building which would allow water to be supplied via a fire truck. Providing spray water during accidents such as a station blackout would serve several functions. Such water can help dissipate heat, cool core debris, and scrub fission products from the containment atmosphere. Because the pumping capability of the fire water system is a fraction of that of the containment spray system, the use of the fire pump without other modifications would not produce an adequate containment spray pattern. By relatively simple modifications, the existing containment spray heads may be modified to ensure an adequate spray coverage for fission product scrubbing and heat removal for some scenarios. Indeed, Boston Edison has proposed blocking 6 of 7 nozzles in each spray head. The impact of such modifications on other accidents still requires assessment.

c) Venting - Venting the containment can reduce core melt probabilities for some accidents, and prevent overpressure or overtemperature failures for others. It may also be viewed as a last ditch effort to prevent the containment from bursting (overpressure). If done before the core melts, little in the way of fission products would be released. If done after core melting, substantial fission products could be released. These fission products are of two types; noble gases and other fission products. Filtering or scrubbing can be effective in reducing non-noble gas fission products. However, only relatively long period hold-up of noble gases can be effective in reducing their potential biological impacts. The MARK I suppression pool is an excellent potential post-accident scrubber for the other fission products. Therefore, any venting should be of the wetwell airspace to at least take advantage of suppression pool scrubbing of non-noble gas fission products. To the staff's knowledge, separate filtered vents such as have been or are being installed in Europe (i.e., Sweden, France and Germany) have not been considered for a BWR MARK I* in the U. S.

*A filtered vented containment has been proposed at one U. S. MARK II plant.

It is noted that venting procedures using existing equipment have been incorporated in the emergency procedures for some U. S. reactors. To the staff's knowledge, all MARK I plants have such procedures for utilizing various sized penetrations.

Most existing BWR MARK I wetwell vent paths outside primary containment are incapable of operating in, or withstanding the pressures and temperatures associated with, severe accidents. If such vents were used without modification, their failure could result in contamination and hydrogen ignition in vital spaces outside containment. By connecting wetwell airspaces to the existing Standby Gas Treatment System (SGTS) stack with valves and piping capable of withstanding severe accident temperatures and pressure, and providing for remote manual operation *, fission products would be discharged without contaminating vital areas, and would gain benefit from dispersion at a high (up to about 100 meters) level.

Criteria for emergency venting through relatively small containment penetrations has been approved for licensed BWRs as part of the implementation of post TMI improvements. However, the smaller vents are generally not capable of sufficient pressure relief during severe accidents to ensure containment integrity. Unnecessary and untimely venting could put the public near a reactor at some risk. The procedures for venting, and the control of decisionmaking, have been raised as issues that require further assessment. Therefore, a systematic assessment of the negative safety impacts of containment vents will be made.

d) Core debris control - Proposals by the staff have been made to provide for core debris control on the drywell floor of the containment in the form of guide walls, and in the torus room under the steel suppression pool liner in the form of an additional water/debris catcher. Preliminary assessments of containment guide walls indicate they are unlikely to be effective in directing core debris away from downcomers or the containment wall. Curbs in the torus room in the reactor building would be expected to form a dam if core debris penetrated the steel suppression pool liner, and would retain suppression pool water. The water would help quench core debris and scrub fission products.

e) Enhanced Reactor Building Fission Product Attenuation - If core debris fails or bypasses the steel suppression pool liner in the torus room, a direct path for fission products through ventilated spaces in the reactor building would exist. Attenuation of fission products (see draft NUREG-1150) could be enhanced significantly by the use of sprays from the plant fire system. The enhancement could be accomplished by either a significant design change with little procedural impact, or a small design change with a significant procedural impact. The former has been studied and found to be relatively costly. What has not been fully considered is an improvement such as use of fire hose nozzles to provide a low flow rate "fog" in important airspaces. The nozzles would be clamped to hand rails, and initiated prior to major reactor building contamination.

f) Basemat isolation - The possibility exists that the basemat may be penetrated by core debris, and contaminate water supplies. Methods for isolating such core debris have been evaluated in the U. S. (Research Letter 150), and were used at Chernobyl. Because of the torus design of MARK Is, this type of event is considered relatively unlikely. Because of this, such an undertaking could be done on an ad hoc basis with only references provided in plant emergency procedures.

*One possible means of powering such valves in station blackout events may be by the use of small portable DC generators. Such generators could also be used to power ADS valves and reduce accident likelihoods and consequences.

g) Automatic Depressurization - The automatic depressurization system (ADS) may not be available during SBO scenarios. System availability has been generally recognized as important in preventing core damage. However, the use of the system to mitigate the high pressures, temperatures and fission products in the vessel after core melting, but before vessel failure, may be useful. The advantages of improvements to the system for mitigation purposes have not been fully examined.

h) Procedures and Training - Existing emergency procedures and training at BWRs with MARK I containments have not been fully developed with respect to risk significant severe accident challenges. Improvements in existing procedures and operator training should substantially improve the capability of operators to cope with severe accidents. (It is noted that procedures and training are also related to other severe accident programs such as IPEs and improved licensee performance.)

CONTAINMENT PERFORMANCE RESOLUTION PROCESS - Resolution of issues is to be achieved by a two stage process. The first stage will consist of issue characterization, parametric studies, experiment assessments and a critical focusing on each of the relevant technical issues. Both phenomenological issues and potential improvement issues potentially important to the mitigation of MARK I severe accidents are to be identified. Examples of phenomenological issues are the manner in which a core disassembles in-vessel, how debris in the bottom of the vessel attacks the lower head and may induce failure, ejection of debris, and core debris attack on the containment liner. Examples of potential improvement issues include the usefulness of venting, containment sprays, ADS enhancements, hydrogen control improvements, core debris control, reactor building fission product attenuation, and basemat isolation. Parametric studies will include assessments of related experiments, and analytical evaluations of the impacts of a range of core melt progression assumptions and potential improvements on containment performance. After initial issue characterization, a meeting will be held with representatives from RES contractors, industry, other experts and interested members of the public on each issue. The second stage will be a sorting and evaluation process performed by the staff where each issue will be categorized as being either a) resolved or unimportant, b) potentially resolvable by future research, or c) candidates for regulatory initiatives. The criteria to be used for judging if a regulatory initiative will be recommended (to effect closure) include the backfit rule (needed for safety, or a justifiable safety enhancement), and the Safety Goal Policy and implementation plan.

The process and related target dates are summarized below:

MARK I CONTAINMENT PERFORMANCE RESOLUTION PROCESS*

1. Prepare Program Plan

   a. Prepare Commission Information Paper responding to SRMs. Include early identification of challenges, failure modes, potential improvements and primary generic issues related to the containment type being considered. Identify details of the process for narrowing and resolving issues. Coordinate with NRR. — Nov. 1987

*See Enclosure 3 for relationship of activities and schedule dates.

b. Seek ACRS comment on plan and closure criteria. Dec., 1987

c. Revise plan based on ACRS and Commission comment. Dec., 1987

2. Prepare for and hold a meeting with representatives from National Labs, industry, other experts and interested members of the public to narrow and, to the extent possible, resolve phenomenological and improvement issues. Identify details of the criteria the staff will use to judge whether or not an initiative is warranted. Consider the use of "success states" defined in terms of the magnitude and timing of releases from outlier accident sequences based on a definition of a large release. Use available resources (NUREGS, etc.) to formulate initial issue characterization, and request meeting invitees to comment on characterizations prior to the meeting. Issues are to be related to containment challenges, failure modes and potential improvements. Use contractors to help with issue characterizations, to facilitate a meeting, resolve issues, and to prepare summaries. Use reviews of PRAs (i.e., Peach Bottom, Cooper, etc.), utility containment safety studies (Vermont Yankee and Pilgrim), IDCOR evaluations, and the anticipated NUMARC evaluation to characterize issues. Undertake parametric core melt and containment challenge calculations, where practicable, and review experiments to aid in focusing issues.

| | | |
|---|---|---|
| a. | formulate initial issue characterization | Dec., 1987 |
| b. | issue meeting invitation | Jan., 1988 |
| c. | revise issue characterizations based on comments, develop preliminary bases for staff evaluation of issues in terms of the magnitude and timing of fission product releases for outlier sequences, and hold meeting. | Feb., 1988 |
| d. | issue draft summary for comment | Mar., 1988 |
| e. | issue final summary | Apr., 1988 |

3. Prepare interim report to Commission with possible recommendations for improvements. Identify primary areas of agreement and disagreement among parties. For those issues for which analyses indicate that safety improvements could be effective (e. g. a form of venting), recommend a regulatory initiative. Apr., 1988

4. Identify a) issues that are resolved or are unimportant, b) issues that may be best resolved by future analytical and experimental research, and c) candidate issues for regulatory initiatives using the criteria based on the Backfit Rule and Safety Goals. Complete Backfit/Safety Goal assessment. June, 1988

5. Prepare Commission paper and/or NUREG with staff recommendations.

| | | |
|---|---|---|
| a. | complete draft Commission Paper | June, 1988 |
| b. | ACRS discussion | July, 1988 |
| c. | CRGR meeting | July, 1988 |
| d. | complete final Commission Paper | Aug., 1988 |
| e. | undertake implementation of Commission approved intiatives (e. g., rulemaking, Generic Letters and licensee implementation) | To be determined |

6. Continue long term MARK I research. — To be determined

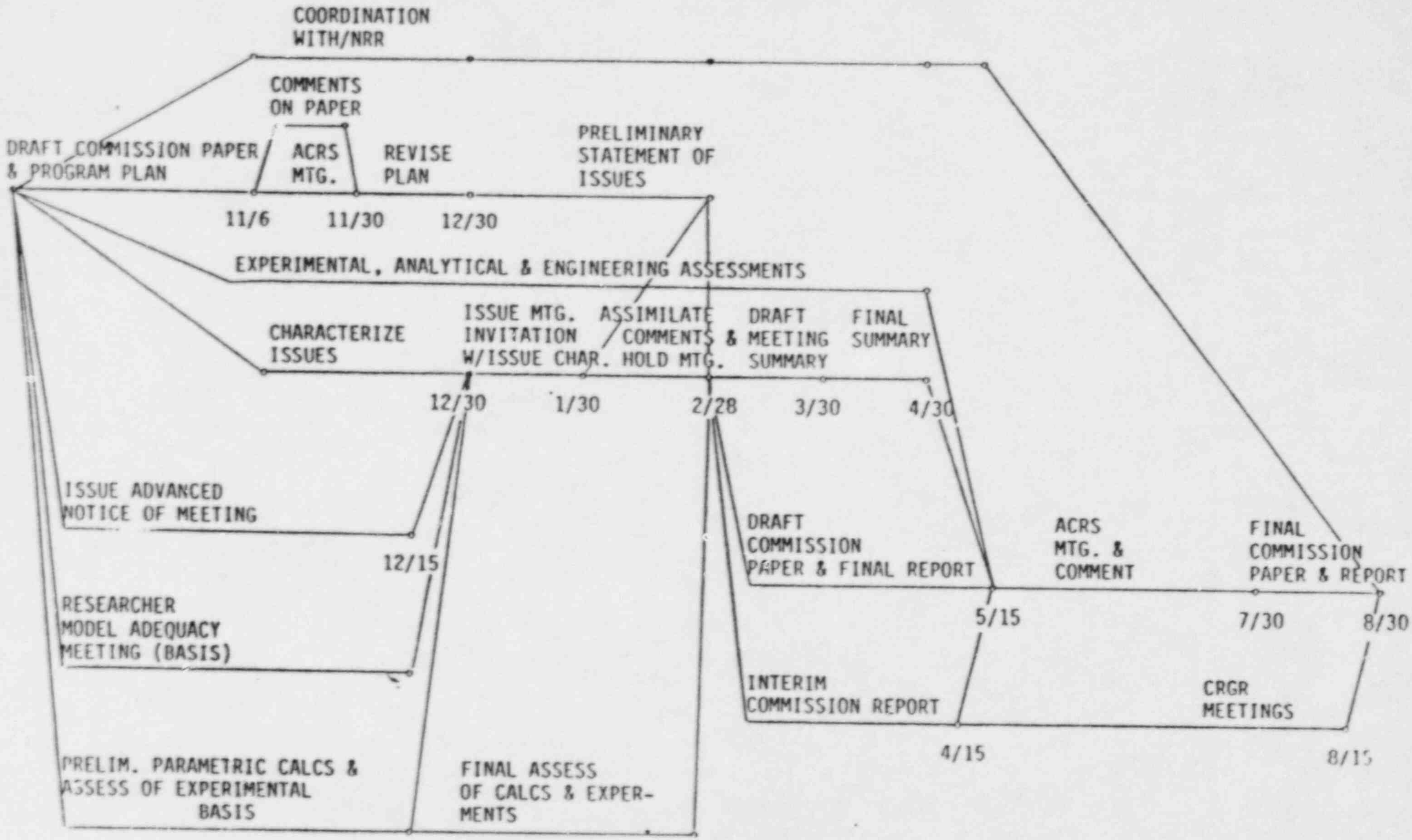
7. If warranted, assess other containment types. — To be determined

RESOURCES - Existing RES resources related to severe accidents and MARK I containments of approximately $1M and 4FTEs are considered adequate for FY 88. These resources include those earmarked for resolution of MARK I issues, and those allocated for longer term BWR severe accident/source term research. Additional resources during this period would not be expected to accelerate completion because of the time required for assessing, narrowing and resolving issues.

MANAGEMENT - The organizational unit responsible is the Severe Accident Issues Branch in RES. Project management for MARK I resolution is to be provided by the branch chief. Inputs from other branches in RES and NRR are to be solicited. Furthermore, a senior level management steering group composed of representatives from RES, NRR and AEOD is to provide oversight.

ENCLOSURE 3

MARK I CONTAINMENT PERFORMANCE KEY ACTIVITIES & MILESTONES



## SUPPORTING RESEARCH

1. EXPERIMENTAL

   a. ACRR FACILITY CORE MELT DF-4 EXPERIMENT (SNL) - This DF-4 BWR early melt progression experiment with control blade, canister wall and fuel pins has been completed. It represents an initial data base for modeling BWR early melting in the ORNL assessment of Peach Bottom. The documentation of the experiment is to be completed by about April, 1988.

   b. $B_4C$/INTERACTIONS (SNL) - One intermediate mix test has been completed and analyzed. Other tests are planned and documentation is to be completed. Information from the one completed test is expected in December, 1987.

   c. EUTECTICS IMPACTS ON CORE MELT PROGRESSION (ORNL) - Some small scale tests (a few kg.) are expected to be completed by January, 1988 with documentation to follow. The results are to be used in the ORNL assessment of Peach Bottom.

   d. SIMULANT MELT SPREADING & CONTACT (BNL) - Benchtop tests have been completed and applied in analyse of melt spreading, including the presence of an overlying water pool. Documentation is to be completed and the results used by ORNL.

   e. MOLTEN CORE CONCRETE INTERACTION TEST - Information on molten material spreading was obtained from an intermediate scale test (187 kg.). This test is used to support analyses of melt spreading within containments by ORNL.

   f. HIGH TEMPERATURE HYDROGEN COMBUSTION - The combustion behavior of hydrogen, oxygen, steam, carbon monoxide and carbon dioxide mixtures in the reactor building is being investigated using recently developed models. These models cannot be experimentally verified at temperatures above 150 degrees C, but an initial peer review will be completed in November. The results are to be used by ORNL.

2. ANALYTICAL

   a. MARK I MELT SPREADING (ORNL) - This is a "first effort", parametric analysis using results from the experiments identified above to develop code models for the eutectics formed by zirconium and uranium oxides. The liquid/solid temperatures of constituents affect predictions of concrete ablation, outgassing and aerosol containment emissions. Melt spreading velocity and liner erosion rate predictions are also affected.

   b. LINER MELT (BNL) - Liner failure predictions are to be made as functions of time based on core melt spreading, liner contact erosion and melting.

   c. BWR CORE MELT PHENOMENOLOGY (ORNL) This represents an initial parametric modeling effort. The DF-4 tests, the TMI-2 examination, and other severe fuel damage experiments are consistant with the BWR models to be used for MARK I analyses. Further review of the models within the context of emerging research on overall BWR core melt progression is to be conducted.

d. PARAMETRIC ASSESSMENTS (ORNL)

1) REACTOR BLDG FISSION PRODUCT ATTENUATION FOR PEACH BOTTOM & BROWNS FERRY (ORNL)
2) ADVANTAGES OF ADS IMPROVEMENTS (BNL/ORNL)
3) FIRE WATER SPRAY ADVANTAGES (ORNL)
4) ADVANTAGES OF CURBS & WATER IN TORUS ROOM (ORNL)
5) ADVANTAGES OF VENTING IMPROVEMENTS (INEL/ORNL)
6) UNCERTAINTY ASSESSMENT(ORNL)

3. ENGINEERING ASSESSMENTS

a) ISSUE CHARACTERIZATION (BNL/STAFF)
b) USE OF CURBS IN THE DRY WELL & TORUS ROOM (ORNL/STAFF)
c) USE OF FIRE WATER/SPRAYS AND SAFETY IMPACTS (ORNL/STAFF)
d) ADS IMPROVEMENT (ORNL/BNL/STAFF)
e) VENTING IMPROVEMENTS & SAFETY IMPACTS (INEL/ORNL/STAFF)
f) $H_2$ CONTROL IMPROVEMENTS (STAFF)
g) IMPROVEMENT COSTS AND BENEFITS (STAFF/INEL)
h) REGULATORY ISSUE EVALUATION (STAFF)