MAR 1 1 1994

Dr. G. M. Frescura
OECD Nuclear Energy Agency
Le Seine St. Germain
12, Boulevard des Iles
F-92139 Issy les Moulineaux France

Dear Dr. Frescura:

SUBJECT: CNRA QUESTIONNAIRE ON SEVERE ACCIDENT ISSUES

Enclosed please find the U.S. Nuclear Regulatory Commission's (NRC) response to the CNRA questionnaire.

Also, as we agreed, NRC has the coordinating responsibility for preparing the draft technical report on the status of the Severe Accident Issues in the OECD countries. In addition to the committed NRC staff, we have contracted Dr. Trevor Pratt of the Brookhaven National Laboratory, to help us in compiling the responses. The schedule calls for issuance of the draft by April 1, 1994. We will do our best to maintain the schedule.

If you have any questions or comments, please contact either myself at (301) 504-3226 or Mr. Jack Kudrick, at (301) 504-2871.

Original signed by

Martin J. Virgilio, Acting Director
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

Enclosure:
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ENCLOSURE

THE U.S. NRC RESPONSE TO THE
COMMITTEE ON NUCLEAR REGULATORY ACTIVITIES (CNRA)
SPECIAL ISSUE MEETING ON STATUS OF SEVERE ACCIDENT ISSUES
QUESTIONNAIRE

The questionnaire has been structured to gather information on how nuclear safety authorities of different countries address the severe accident issues, for both operating and future reactors. It is important to both describe the positions that have been taken and to provide the rationale upon which they have been based.

In this regard, it is important to describe; the level of prevention, mitigation and accident management, the respective use of deterministic and probabilistic analyses, and the treatment of uncertainties. These considerations should be addressed for both the generic policy as well as for each technical issue.

QUESTION 1

The introductory question is intended to allow each country to provide its regulatory goals relative to defining acceptable risks levels associated with severe accidents.

> What are the regulatory requirements, thinking, or plans relative to an acceptable position on severe accident risks?

Staff Response

Prior to commenting on the regulatory requirements, a few words are needed in order to understand the relationships between requirements and guidance provided to U.S. industry. The requirements are specified in the Title 10 of the Federal Code of Regulations (10 CFR), which is a series of laws enacted through a formal rulemaking process. In addition to these requirements, the NRC has published both clarifications and supplementary guidance in the form of Regulatory Guidelines (RGs) and the Standard Review Plan (SRP). These latter publications provide approaches which the NRC believes will satisfy the regulatory requirements. But, the identified approaches may not be the only acceptable path. The licensee may elect to follow a unique path to resolution which the NRC would review on a case by case basis. In addition, the NRC issues various generic communications, like policy statements (Commission Papers, or SECYs) and generic letters (GLs), that address safety related concerns. While not requirements, they are intended to alert industry of specific issues which the NRC believes should be evaluated to assure continued safe operation of nuclear power plants.

The NRC's current regulatory requirements are largely based upon deterministic engineering criteria, where the intent is to ensure safe nuclear power activities with multiple layers of defense in depth. This approach defines design basis standards and criteria for a predefined set of plant performance scenarios, and mandates a specific capability and quality of safety systems needed to respond. Beyond our deterministic criteria, the NRC has additionally formulated guidance, as in the Safety Goal Policy Statement, which utilizes quantitative risk measures. This policy statement proposes high level objectives of limiting societal risk from nuclear power and includes guidelines on the levels of core damage frequency and containment performance. While deterministic criteria have remained the focus of the NRC's regulatory base, the numerical risk goals of the Safety Goal Policy have been used to some degree for reassessing regulatory requirements and regulations, where appropriate, using the quantitative subsidiary goals.

For many years the Agency has been active in the development of quantitative risk assessment methods and data bases which provided tools to allow assessment of a broad scope of beyond design basis conditions, involving multiple failures or complex interdependencies. Assessment of these conditions allows for a more comprehensive understanding of potential design strengths and weaknesses, and has been utilized to varying degrees in the NRC's regulatory activities.

It is acknowledged that the use of probabilistic risk assessment (PRA) technology is not a substitute for the defense-in-depth philosophy which is the cornerstone of the NRC's regulatory activities. It is also recognized that uncertainties in calculated probabilities arising from limitations in data and modeling as well as limitations of the PRA technology must be considered when implementing the results of risk assessment. Technology limitations include its limited usefulness for discovering design and construction errors and for modeling human performance considerations, especially errors of commission and organizational or safety culture issues. Cognizant of these important limitations of current PRA techniques, PRA is nevertheless viewed as an important adjunct to the NRC's regulatory resources, and it is intended to be utilized more extensively in the future.

After Three Mile Island (TMI), severe accidents were recognized as a reality. While numerous modifications to plant design and procedures were enforced as a result of the accident, there were many questions as to what would be the best approach to address the severe accident issues. After a great deal of industry/staff interaction on the technical aspects of severe accidents, the Commission decided to publish these issues through policy statements rather than rule changes. Four such statements were issued in 1985 through 1987.

The initial statement in 1985, was intended to put into proper perspective, the threat of severe accidents to the operating plants and to identify longer term efforts to fully evaluate the issue. Based on a thorough review of existing information, the Commission concluded that existing plants pose no undue risk to public health and safety and that there was no need for generic rulemaking and other regulatory changes with respect to the severe accident risk. However, there was a need to perform a systematic examination of individual plants to identify any plant-specific vulnerabilities to severe accidents.

Additionally, it was felt that all reasonable steps should be taken to reduce the probability of a severe accident and to mitigate the consequences of such an accident should one occur. In other words, there should be a balance between accident prevention and mitigation.

Following policy statements focused on discussions of the risk from nuclear plant operation. The Safety Goal Policy Statement expressed an opinion that individual and societal risks to life and health should not increase significantly from the use of nuclear energy to produce electric power, and should be comparable, or be less than the risk represented by other viable technologies of generating electricity.

The last policy statement which has been issued on the subject addressed the future plants and the use of standardization in this process. Beyond the stated belief that standardization would enhance overall plant safety, it also indicated that future designs should follow the guidance provided in 10 CFR 50.34(f). This requirement was originally intended for plants near licensing after TMI, but the contents seemed to be consistent with the guidance provided within the policy statements. Therefore, this guidance was extended to include advanced designs.

Based on the above guidance, separate approaches evolved for the operating plants (OR) and for future LWR designs (ALWR). These are:

- Individual Plant Evaluation (IPE) for ORs (GL 88-22), and
- SA requirements for ALWRs (SECY 90-016, and 93-087.)

The two approaches, although different, are not totally independent, as the insights from IPEs and SA Research were continuously incorporated into both programs.

For the operating plants, a Severe Accident Closure Program was established. As depicted in Figure 1, the program consisted of eight elements leading to severe accident closure. There were three major elements with the remaining elements having a supportive role. The three elements are: the Individual Plant Evaluation (IPE), the Containment Performance Improvement (CPI) program (combined Mark I and other containment improvements), and the Accident Management Program (AM). Each element addressed a specific need. The IPE calls for a systematic examination of each plant, both operating and holding construction permits. The main objective of the program was to uncover any plant unique vulnerabilities to severe accidents that potentially may exist, and were not previously identified by traditional methods. Although PRA was not the required approach for an IPE, almost all licensees have performed at least a Level 1 PRA, with many performing a full Level 2 PRA. A few licensees have performed a Level 3 analysis. GL 88-20 (December 1, 1988) initiated the IPE process, but deferred resolution of two technical issues which were not well understood at the time; High Pressure Melt Ejection and BWR liner melt-through. Research in these areas has produced two resolution documents which are currently being reviewed within the staff.

CPI was established as a short term effort which was intended to compliment the IPE program. The program objective was to identify risk significant vulnerabilities to the various containment designs earlier than could be expected from the larger and more protracted IPE program. If major threats were identified, immediate design fixes would be considered. As a result, the plants with Mark I containment were required to reduce vulnerabilities by installing hardened wetwell vents.

The last and possibly the most important phase of the program is the AM program. The AM program was implemented as a cooperative NRC/industry effort to provide guidance to plants in five areas: procedures, calculational aids, training, instrumentation, and decisionmaking. The accident management guidance will address both prevention and mitigation of accidents, and its issuance is in progress.

In parallel with the severe accident effort, the agency pursued rulemaking on three issues to address important contributions to core damage frequency; Anticipated Transient Without Scram (ATWS), Pressurized Thermal Shock (PTS), and Station Black-out (SBO). Also, the agency will continue to issue generic letters to address safety significant issues as they arise in the future. In all such decisions the magnitude and urgency of agency action will account for the level of risk associated with the issue. the NRC Safety Goal Policy as well as existing industry criteria provide guidance in making these decisions. These efforts, in combination with the IPEs, give confidence that core damage prevention for the operating plants is being adequately addressed.

Future reactors are expected to achieve a higher standard of SA safety performance than prior designs. It is believed that this additional level of safety can be achieved by following generic guidelines, as formulated in the Policy Statements, as well as by satisfying specific criteria, as described in SECY 90-016 and SECY 93-087. The thrust of the guidance is to produce a balanced design. Adequate level of preventive features is accomplished by complying with the guidance in SECY 90-016 and SECY 93-087. Specifically, the vendor for an advanced design should address how the design satisfies SBO, ATWS, ISLOCA, and fire protection criteria. The intent is for the designer to provide preventive features to avoid challenges to safety systems.

Mitigation has also been addressed within the above references. Generally, the approach has been to prevent the early challenges to the containment and to mitigate longer term challenges. The design should include mitigative features related to hydrogen generation and control, high pressure melt ejection, core debris coolability, enhanced containment performance, and equipment survivability.

Details of specific requirements are addressed in the follow-up questions, i.e., radiological consequences and emergency planning in Q2, the role of PRA in Q3, the SA phenomenology in Q4, and the AM approach in Q5.

In summary, numerous activities have contributed to the development of NRC SA requirements. In general, the emphasis is on; (i) achieving a balance between prevention and mitigation, (ii) developing accident management and training programs, and (iii) performing research to develop a better understanding of severe accident phenomena.

QUESTION 2

Regulatory positions relative to the radiological consequences of a Severe Accident are an important aspect in understanding the overall philosophy of a country's position. To better understand how radiation consequences have been incorporated into the overall strategy for developing severe accident requirements, the following questions have been provided. For each subquestion, provide the status of the requirement (i.e., in place, planned, or under discussion)

a) What limits or objectives are placed on short and long term release magnitudes?

b) How are the health and environmental (land and water) effects (short and long term) addressed?

c) What is the role and the scope of emergency planning?

d) Do the analyses performed to calculate the releases and their consequences use conservative or realistic assumptions and models?

QUESTION (2a)

What limits or objectives are placed on short and long term release magnitudes?

Staff Response

NRC rules governing releases from design basis accidents (DBA) are provided in 10 CFR 100. Specifically, the rule prescribes a maximum dose of 25 rem to the whole body and 300 rem to the thyroid from iodine at the exclusion area boundary for two hours immediately following the onset of the release, and the same dose limits for the entire period of the passage of the release at the boundary of the low population zone. The source terms used in these dose assessments are specified in Regulatory Guides 1.3 and 1.4. These requirements apply to OR as well as to the futures plants

There are no corresponding specific release limits for severe accidents. The Commission has, however, defined an acceptable level of "risk" in its policy statement on safety goals for the operation of nuclear power plants. In it, the Commission established two qualitative safety goals which are supported by two quantitative objectives. The Commission's qualitative safety goals are; (1) the risk due to a severe reactor accident including normal plant operation should not be a significant contributor to a person's risk of accidental death or injury, and (2) societal risks to life and health due to a severe reactor accident including normal plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

The Commission's quantitative objectives are; (1) the risk to an individual in the vicinity of a nuclear power plant 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, and (2) the risk to the population in the area near a nuclear power plant operation should not exceed one-tenth of one percent of the sum of cancer fatality risks resulting from all other causes.

The quantitative objectives of the Safety Goal Policy are not intended to be applied to each operating reactor. Rather, they are to be used to evaluate the effectiveness of the NRC's regulatory programs and the overall safety of nuclear reactor operation in the United States.

The Commission further provided the staff with guidance for implementing the quantitative health effect objectives. The guidance states that the overall mean frequency of "a large release" of radioactive materials to the environment from a reactor accident should be less that 1 in 1,000,000 per year of reactor operation. For ALWR designs, the staff has achieved this objective by requiring a conditional containment failure probability of 0.1.

There are no specific limits established for the future plants. However, in applying the safety goals to advanced light-water reactors, the staff currently proposes to take a deterministic approach by establishing a containment integrity criterion of 24 hours and defining timed fission product release rates in draft NUREG-1465. These release rates are based on substantial research and experience gained over two decades. They are physically based and are considered more realistic than the conservative source terms defined in Regulatory Guides 1.3 and 1.4 referred to above for DBA assessments. By combining these requirements with specific design features to prevent the onset of a severe accident, the staff has adopted a balance between prevention and mitigation in assuring adequate protection against severe accidents.

QUESTION (2b)

> How are the health and environmental (land and water) effects (short and long term) addressed?

Staff Response

Final Environmental Statement (FES) reports have been issued for approximately 28 plant sites since 1980. These reports included a discussion of the health and environmental effects due to a severe reactor accident. The staff typically addressed three pathways for release of radioactive material to the environment from severe reactor accidents. These pathways were; (1) air, (2) air to surface water, and (3) groundwater to surface water. For most plants, the air pathway represented the most likely pathway for significant dose to the public. The second pathway was significant for only a few sites that are close to large but confined bodies of water. The third pathway represented a less significant potential for dose because of reduction in radioactivity due to retention in the ground and greater flexibility and time to implement interdiction measures.

In the FES analyses, rebaselined Reactor Safety Study (NUREG-75/014, formerly WASH-1400) source terms from NUREG-0773 were used with site-specific meteorology and demographic data to calculate offsite risk. All plants used the Calculation of Reactor Accident Consequences (CRAC) computer code to determine environmental consequences. Recently, uncertainties were explored (NUREG-1150) in accident frequency, containment behavior, and radioactive material release and transport, so that mean values of risk could be determined. Source terms and accident frequencies specific to the plants considered in NUREG-1150 were determined using advanced computer codes. For example, the MACCS (MELCOR Accident Consequence Code System) computer code was used instead of CRAC for consequence evaluation.

In these FES probabilistic assessments of severe accidents, the staff concluded that; (1) the risk of early and latent fatalities from the air pathway is small as it represented only a small fraction of the risk to which the public is exposed from other sources, (2) population dose for the drinking water pathway is found to be small relative to the atmospheric pathway, (3) risk associated with the aquatic food pathway is also found to be small relative to the atmospheric pathway for most sites and essentially the same as the atmospheric pathway for the few sites with large annual aquatic food harvest, and (4) the groundwater pathway generally contributes only a small fraction of that risk attributable to the atmospheric pathway.

In conclusion, the staff considers the radioactive releases through various pathways as relatively well understood. However, the staff recognizes the uncertainties associated with health environmental effects evaluation and continues to investigate the transport of radioactive materials, as well as the radiological consequences.

QUESTION (2c)

What is the role and the scope of emergency planning?

Staff Response

Emergency planning is one of the features of the NRC's defense-in-depth safety philosophy. This philosophy; (1) requires high quality in the design, construction, and operation of nuclear plants to reduce the likelihood of malfunctions in the first instance, (2) recognizes that equipment can fail and operators can make mistakes, therefore requiring safety systems to reduce the chances that malfunctions will lead to accidents that release fission products, and (3) recognizes that, in spite of these precautions, severe fuel damage accidents can happen, therefore requiring containment structures and other safety features to prevent the release of fission products offsite. The added feature of emergency planning to the defense-in-depth philosophy provides that even in the unlikely event of an offsite fission product release, there is reasonable assurance that emergency protective actions can be taken to protect the population around nuclear power plants.

The scope of emergency planning includes plans, procedures, equipment, and personnel needed to enable measures to be taken to protect the public in the event of a radiological accident. Both the nuclear power plant licensee and State and local governments have a role in emergency planning. Detailed plans are developed for protective actions, including evacuation of the public, in the plume exposure pathway emergency planning zone (an area of about a 10-mile radius around the plant). In addition, plans are developed for protective actions, including the interdiction of contaminated food, for the ingestion pathway emergency planning zone (an area of about a 50-mile radius of the plant). In determining the appropriate size of these two emergency planning zones, the NRC considered the potential consequences, timing, and release characteristics of a spectrum of accidents including severe accidents.

Requirements relative to emergency planning are in place and are primarily contained in section 50.47 and Appendix E to 10 CFR Part 50.

QUESTION (2d)

> Do the analyses performed to calculate the releases and their consequences use conservative or realistic assumptions and models?

Staff Response

The use of conservative vs. realistic calculations depends on the regulatory issue under consideration. For example, the calculations performed to show compliance with NRC rules governing releases from DBA events (described in the response to Question 2a) are conservative. The DBA fission product source term* is large and assumed to be instantaneously released into the containment atmosphere. The dose estimates resulting from containment leakage are also based on conservative assumptions (centerline dose, extreme weather conditions, etc.). This approach was adopted because the rules determine the allowable containment design leakage. Therefore, by using conservative assumptions and models a margin is established between the calculated and actual dose levels that compensates to some extent for uncertainties in the analysis. Similar approaches were adopted for the FES reports (response to Question 2b) and in the development of the emergency planning requirements (response to Question 2c) discussed above. For example, the source terms used in these regulatory activities were based on WASH-1400, which was published in 1975, and they are in some respects conservative. Research over the last decade has in general found lower fission product release fractions for most accident sequences and lower frequencies for those accident sequences that result in relatively large source terms.

However when releases and their consequences are determined for a PRA of a particular plant, best estimate calculations are performed supplemented by sensitivity or uncertainty analyses. For example, comparisons with the Commission's quantitative objectives (refer to the response to Question 2c) are done with mean risk estimates generated from a PRA. Also, if a cost

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* In response to Question (2a) above it was noted that attempts are being made to develop more realistic source terms for application to future plants.

benefit analysis is being performed based on a PRA to determine if a modification to a plant is warranted under the backfit rule, best estimate or mean values of the source terms, their frequencies and consequences are used to determine the dose averted component of the cost benefit equation. Best estimate or mean values are used for these types of regulatory applications because they are not part of the design basis of the plant and therefore not subject to the conservatisms or margins that are built into the DBA approach. However, regulatory decisions of this type are also made with consideration given to the uncertainty ranges.

QUESTION 3

As part of the responses to question 1 and 2, the basis for developing positions on severe accidents will have been discussed. The reliance on probabilistic and/or deterministic analyses should also have been indicated. The following questions have been included to determine to what extent these approaches were used in the development of positions related to severe accident issues:

QUESTION (3a)

What is the role of probabilistic and deterministic analyses in the development of regulatory positions relative to severe accidents?

Staff Response

Current regulatory requirements are largely based upon deterministic engineering criteria. However, the expected frequency of events influences the level of regulatory oversight. For example reactors are designed to handle departures from normal operation and transient events that occur quite frequently. Accidents that occur less frequently (10^{-2} to 10^{-4} per reactor year) form the design basis of the plant. Systems and barriers are incorporated into the plant to mitigate design basis events. Once these systems and barriers are identified then they are designed using a conservative deterministic approach, which provides a margin that makes some allowance for uncertainties.

Severe accidents are relatively unlikely events that are outside the design basis of the plant. These events therefore fall in the domain of PRA and probabilistic considerations become more important. When making regulatory decisions, for example, comparisons with the commission's quantitative objectives and cost-benefit analyses require a probabilistic framework. However, probabilistic and deterministic analyses are treated as complementary when dealing with severe accident issues. Probabilistic methods are used to identify risk-dominant phenomena but deterministic methods are used to analyze the phenomena in detail. Deterministic methods coupled with appropriate experimental data can be used to reduce uncertainties or eliminate severe accident phenomena by hardware modifications. For example, the interim H_2 rule eliminated H_2 combustion as a threat to containment integrity in some reactor designs by requiring the containment atmosphere be inerted during operation (refer to the response to Question 4ii).

For advanced reactor designs, the NRC approach explicitly requires a combination of probabilistic and deterministic analyses. The standardization rule (10CFR52) requires a PRA as part of the design process for advanced plants. In addition, Commission policy, as embodied in SECY 90-016 and SECY 93-087, calls for specific design features to deterministically deal with severe accident phenomena (refer to Question 4).

In summary, the design basis of ORs and ALWRs are largely based on conservative deterministic analyses. PRAs are used to address regulatory actions for lower probability severe accidents. PRAs are being used for OR as part of the IPE program to search for plant-specific vulnerabilities. In addition, PRAs are used in cost-benefit calculations. PRAs are an integral part of the design certification process for ALWRs. Probabilistic and deterministic analyses compliment each other when dealing with severe accidents.

QUESTION (3b)

> To what extent should the above analyses be based on conservative or realistic assumptions? What is the scope of sensitivity and uncertainty analysis?

Staff Response

It is the NRC's position that severe accident analyses be based on best estimate assumptions. However, given the uncertainties associated with predicting severe accident phenomena, regulatory actions should be made with an appreciation of the associated uncertainty ranges. There are several ways of characterizing the uncertainties associated with severe accident phenomena. By varying key model parameters and data over credible ranges the sensitivity of the predictions to the changes can be determined. Sensitivity studies of this type can be useful, for example, if a particular severe accident phenomenon has been found unimportant from a regulatory point of view based on best estimate assumptions. The robustness of this conclusion can be demonstrated by sensitivity studies. Examples of severe accident phenomena that have been addressed via this approach are given in the response to Question 4.

To characterize uncertainty ranges that reflect our current understanding of severe accident phenomena is a difficult task. Examples of approaches used to determine uncertainty ranges are given in the responses to Question 4 for different severe accident phenomena. However, given the lack of experimental data and the level of detail in the available analytical models, any approach to determining uncertainty ranges will have to rely, to some extent, on expert judgement. The objective is to determine an expected range of values and the shape of a probability curve that appropriately reflects the available experimental data, analyses, and expert judgement. Once such a curve is established judgements can be made as to the importance of the phenomena from a regulatory point of view.

QUESTION (3c)

> If PRA is needed, at what level and at what stage should they be performed? Indicate if the analysis should be level 1, 2, or 3 and if it should consider external events and all possible operating modes?

Staff Response

PRAs are not required for operating plants in the U.S. However, Generic

Letter 88-20 requested all utilities to perform an Individual Plant Examination (IPE) for accidents initiated by internal events. In responding to GI 88-20 most utilities performed a level 1, 2 PRA. In addition, a few utilities also performed a level 3 PRA (refer to the response to Question 1). In Supplement 4 to Generic Letter 88-20 each utility was asked to perform an IPE for accidents initiated by external events (IPEEE). Therefore, IPEs for internal and external events will eventually be performed for all operating reactors in the U.S.

Currently the IPE program addresses accidents that might occur when the plant is at full power operation. However, recent PRAs for accidents when the plant is at low power and shutdown indicate that the risk may be similar to the risk at full power. Currently utilities have not been requested to extend the IPE program to include accidents that might be initiated at low power and shutdown. The NRC is currently pursuing generic rulemaking to address the shutdown risk vulnerabilities identified by existing risk analyses and operational experience.

PRAs are required for all plants seeking design approval. Therefore, as part of NRC's design certification of the evolutionary and advanced reactors, a PRA is included as a chapter in the Safety Analysis Reports (SAR) for the various reactor designs. PRAs are therefore being prepared in the U.S. during the design stage and completed prior to final design approval. The regulations currently do not specify the form, content, and scope of the PRA supporting design certification. However, the staff expects a Level 3, internal events, PRA for the evolutionary LWRs (namely, GESSAR, SP-90, ABWR, and CE System 80+) and the advanced passive LWRs (namely, AP-600 and SBWR.) Consideration should also be given to external events and accidents during shutdown, but not with the same level of detail as with the internal event PRA. As the PRAs are being proposed at the design stage, a great deal of plant specific information on systems and components is not available. Thus the PRA model reflects a level of safety that the reactor design should be able to achieve if built. Several programs and requirements are derived from the PRA results to ensure that the as-built plant achieves the level of safety estimated in the PRA.

In summary PRAs should be performed during the initial design stage. However, PRAs were not performed for most operating reactors at the design stage in the U.S. The IPE program is resulting in a level 2 PRA being performed for most ORs in the U.S. This program will help identify any remaining plant-specific vulnerabilities. For all future designs the staff expects a full scope level 3 PRAs.

QUESTION 4

The following questions address specific issues. A list of phenomena concerning LWRs is provided. Note that the list is not intended to be exhaustive. Any other relevant phenomenon for any reactor type should also be addressed.

a) For each topic, discuss how prevention, mitigation, and accident management are considered in dealing with these issues for both operating and future reactors;

b) Discuss how uncertainties are considered in the regulatory decisions. Where available, provide case studies to show how these issues are resolved.

Staff Response

As a matter of policy, the SA phenomena are not addressed in the same way as existing DBA regulations. Given that large uncertainties still exist in understanding the phenomena, the emphasis is on how probable a specific event is, and how a plant would respond, if an event would have occurred. In addition, the wide variety of US plant designs complicates a potential generic approach even further. Therefore, the approach is to learn as much as possible about a phenomenon, and then evaluate its potential impact on plant safety. Best estimate methodology is acceptable in performing an evaluation, however, applied safety margins are accepted on a case-by-case basis. For any future designs, an ultimate test of the safety margins is compliance with the containment performance requirements, i.e., conditional containment failure probability (CCFP) less than 0.1, and/or containment integrity for 24 hrs. after the onset of core degradation.

For operating reactors, combustible gas control is the only severe accident phenomenon specifically addressed through design features. The Hydrogen Rule (10 CFR 50.44) specifies requirements for the design of combustible gas control systems for all currently operating reactors and certain reactors which have a Construction Permit. Systems used in US facilities include nitrogen inerting systems, electric thermal recombiners, glow plug igniters, post-accident containment nitrogen injection (containment atmosphere dilution) systems and vent/purge systems in various combinations. The specific systems and equipment provided vary according to containment type and vintage. Capability must be provided to cope with the short-term hydrogen resulting from metal water reaction and the additional long-term hydrogen resulting from corrosion of metals and radiolysis of water. Regulatory Guide 1.7 prescribes a minimum amount of metal-water reaction to be assumed in the design of hydrogen control systems based on the ECCS analysis. However, the Hydrogen Rule specifies consideration of larger amounts (i.e., 75%) for certain facilities.

Emergency Operating Procedures developed as a result of the TMI Action Plan have been implemented to assure that operators have guidance in the use of hydrogen control systems. The guidance in the EOPs encompasses severe accidents.

Severe accident phenomena are considered under the Individual Plant Examination (IPE) program and under accident management programs. In addressing the phenomena, licensees were allowed flexibility to perform plant-specific analyses using recognized computer codes or to draw upon previous studies of similar plants. Licensees are not required to deal specifically with any severe accident phenomena by prevention, mitigation or accident management. However, they are to consider the uncertainties surrounding these issues, drawing upon available studies, and, if unique severe accident vulnerabilities are identified during their individual plant examinations, they are expected to take action to reduce the likelihood of the vulnerabilities. These actions may involve prevention, mitigation or accident management, or combinations of these. In addition, insights from the IPEs will be used to consider whether additional actions are required of particular plants or for classes of plants. Any changes proposed by the NRC will be done under the backfit rule.

The new designs are expected to be "safer," and thus they have to be evaluated for potential SA events. Although no specific design features are currently required, new designs must reflect the state-of-the-art understanding of the physics of involved phenomena and their consequences. The requirement of preventive features is implied in the Policy Statement regarding maximum allowable core damage frequency. Also, the development of accident management guidelines and training programs are an important part of preventive actions.

Future plants must reflect a thorough understanding of the response to SA, and every effort must be made to mitigate potential consequences. An important part of the SA evaluation is the equipment qualification requirement for any hardware designed to mitigate the consequences of SA. Case-by-case decisions are being made during licensing process of the evolutionary designs (ABWR and CE-80+). These decisions may be reflected in the licensing process of passive designs (AP-600 and SBWR).

An important part of an enhanced safety of any future reactors would be existence of severe accident management guidelines (SAMGs) and procedures. Although there is no specific requirement to develop such guidelines, the NRC position is that the industry should do so. Currently, vendor specific SAMGs are being developed and the NRC is continuously reviewing the process. Special attention is paid to the clarity of the overall structure of the SAMGs, element of declaration of the SA status, and supporting technical computational aids. In general, the SAMGs should address all of the SA issues.

For future reactors, selective preventive hardware requirements are based on several elements, i.e., insights from the Cooperative Severe Accident Research Program (CSARP), PRA, and IPEs; case specific analyses; and interaction with the industry. The following describes how specific issues are addressed for future reactors:

(i) Reactivity Accidents (e.g., rapid dilution, recriticality during or after severe accidents);

Reactivity accidents are currently considered within the realm of DBA for both operating as well as advanced designs. In addition, the potential for recriticality after the core has degraded has been recognized as a possibility. As a result, studies have been performed to assess the issue. The studies involved a mixture of probabilistic and deterministic analyses. The preliminary results indicate that the recriticality issue during and after severe accident is not a significant risk contributor to the public health and safety.

The beyond design bases reactivity events were examined in some detail following the Chernobyl accident. This was discussed in the Chernobyl implications report (NUREG-1251). Standard design basis accidents were reexamined (without specific new analyses) with parameters of the events expanded to find indications of events which might need further examination.

The events of most safety concern were studied in detail by NRC consultants at BNL (discussed in NUREG-5368). The study set up a wide range of scenarios and a probability/consequence matrix to select those events which should be considered for some type of regulatory action. This study concluded that only two event sequences had the potential for significant fuel damage with fragmentation and dispersion of fuel at a sufficiently high probability level to warrant further consideration. The BWR fuel misloading event analysis resulted in a technical specification (TS) change in the new generic TS in order to decrease the event probability. The transient dynamics of the BWR ATWS boron washout event was examined in detail and it was decided that rapid fuel failure would not occur.

There have been a series of concerns about rapid boron dilution events in PWRs. These are characterized by postulated accumulations of unborated and/or cold water which are then forced through the core by startup of a pumping system. These have included (1) the startup problem (losing pump power during startup deboration), (2) condensation problem (condensation in the steam generator during small break LOCA), (3) large break LOCA (cold water remaining from depressurization). All of these have the potential for significant, rapid reactivity insertion. They have been or are being studied in detail. The startup problem has been discussed in NUREG/CR-5819. It was concluded that damage may occur for extreme event parameters, but with relatively low probability, and the event can be prevented with appropriate procedures, such as preventing inappropriate pump restarts.

(ii) Combustible Gases (hydrogen, CO)

For evolutionary and passive designs, 10 CFR 52.47(a)(1)(ii) requires applicants for a standard design certification to provide demonstration of compliance with any technically relevant portions of the TMI Requirements set forth in 10 CFR 50.34(f). 10 CFR 50.34(f)(2)(ix) requires a system for hydrogen control that can provide with reasonable assurance that uniformly distributed hydrogen concentrations in the containment do not exceed 10% during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100% fuel-clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion.

In SECY-93-087, the staff recommended that the Commission approve the staff's position that the requirements of 10 CFR 50.34(f)(2)(ix) remain unchanged for evolutionary and passive plants. In its July 21, 1993 SRM, the Commission approved the staff's position along with the staff's clarification that the possible use of passive autocatalytic hydrogen recombiners should not be precluded from consideration. The staff was cautioned to carefully consider the relatively slow time response of autocatalytic recombiners as a possible impediment to their efficiency.

Thus far the staff has evaluated two evolutionary designs, the GE ABWR and the ABB System 80+. The ABWR has an inert containment and the System 80+ has been equipped with 80 igniters to meet the requirements of 10 CFR 50.34(f)(2)(ix). Other approaches could be used to meet these requirements. For example sizing the containment such that uniformly distributed hydrogen concentrations do not exceed 10% for an amount of hydrogen associated with a 100% metal-water reaction. As directed by the Commission the staff would also consider the use of passive autocatalytic recombiners either by themselves or in combination with igniters.

Uncertainties have been addressed both in the development of the regulations and in their implementation during the review of proposed industry solutions. For the advanced reactors the requirement that uniform hydrogen concentration not exceed 10% was, in part, designed to assure that detonable concentrations would not exist. The choice of the 10% limit provided some margin to address uncertainties in establishing conditions necessary to produce detonations. Additionally, irrespective of the requirement to limit uniform hydrogen concentration, the NRC has indicated that advanced reactor designs consider the potential for hydrogen accumulations within subvolumes of the containment design. The goal is to minimize the potential for local pocketing of hydrogen.

(iii) High Pressure Core Melt

(iii-1) High Pressure Melt/Direct Containment Heating

It has been postulated that if reactor vessel failure occurs in an unrecovered core melt accident while the reactor coolant system is at high pressure, the subsequent expulsion of core debris (i.e., high pressure melt ejection [HPME]) could pressurize the reactor containment beyond its ultimate capability by direct containment heating (DCH). The risk significance of direct containment heating was identified as one of the major areas of technical uncertainties in NUREG-0956 and in NUREG-1150.

The advanced designs have incorporated a reliable depressurization system (ADS) for the RCS. The ADS would eliminate the high pressure driving force behind an HPME.

The probability of DCH cannot be totally eliminated (i.e., if the RPV fails at high pressure due to failure of the ADS). However, the cavity designs of the advanced designs are such that very little of the ejected molten debris is expected to reach the open volume of the containment.

Case studies

In order to address the DCH issue for PWRs, NRC has funded research in; (a) DCH integral effects and separate effects testing, (b) analytical model development and assessment, and (c) evaluation of the potential for depressurization of the reactor coolant system (RCS) under severe accident conditions.

With this framework, the NRC DCH experimental testing program as well as the DCH issue resolution plan has focused on PWRs and especially on two representative plant geometries, Zion and Surry. The case studies are:

- Analyses for natural circulation in the RCS under severe accident conditions using the SCDAP/RELAP5 code to evaluate the depressurization of the RCS (the Surry plant: NUREG/CR-5447, and NUREG/CR-5937)

- Estimated conditional probability of HPME for the Surry and Zion plants during station blackout transient without operator action and without recovery (NUREG/CR-5949 and letter report, Knudson to Odar, June 17, 1993).

- Severe accident scaling methodology (SASM) to guide the formulation of experimental programs and analytical methods was developed (NUREG/CR-5809).

- Development of stand-alone models for predicting DCH loads (Two Cell Equilibrium Model, (TCEM) and Convective Limited Containment Heating Model (CLCHM)), and also an assessment of the DCH phenomenological models in the CONTAIN code. These stand-alone models and CONTAIN analyses were used to predict DCH loads in full size Surry and Zion plants.

- Risk Oriented Accident Analysis Methodology (ROAAM) and the Accident Progression Event Trees (APET) to evaluate the uncertainties related to the initial conditions (e.g., UO_2 mass, zirconium oxidation fraction) on the Zion and Surry containment loads respectively (NUREG/CR-6075, NUREG/CR-6109).

(iii-2) Induced Steam Generator Tube Rupture and other induced containment bypasses

This issue is currently being studied as part of the overall rulemaking process associated with the integrity of steam generator tubes. Preliminary analyses indicate that induced SGTR is less likely to occur than a failure of other pipes in the system, specifically the surge line. The staff is in the process of assessing the probability of occurrence of the induced SGTR given various tube plugging criteria. Also, the staff is evaluating various design features to mitigate the consequences if a tube rupture were to occur.

(iv) Impact on Vessel Support Structures

This issue applies to those reactor designs in which the reactor vessel is supported by a cylindrical pedestal structure. During a severe accident in which the core debris penetrates the vessel and falls into pedestal region the pedestal walls may be attacked by the core debris. If sufficient degradation of the wall occurs its structural integrity may decrease to the point that the vessel can no longer be supported. For advanced reactor designs SECY 93-87 includes a number of criteria that are designated to reduce the potential for the core debris to attack vessel support structures:

- Provide sufficient pedestal floor space to enhance debris spreading;

- provide a means to flood the pedestal region to assist in the cooling process; and

- protect the vessel structural support members with concrete.

The first two criteria are designed to promote formation of a coolable debris bed, which in turn eliminates the potential for the core debris to attack support structures. The third criteria is included to protect the vessel structural support members from attack by the core debris if a coolable debris bed is not immediately formed. Enough concrete should be provided to ensure that the structural support members are protected for a period of time, that is sufficiently long enough to allow for a coolable debris bed to form.

(v) Fuel Coolant Interaction (In-vessel and Ex-vessel);

Within the section on containment performance in SECY 93-087, the need to evaluate the impact of interaction between molten fuel and coolant, and the resulting steam and hydrogen generation on the integrity of the containment is identified. In accordance with the SECY paper, applicants must ensure that the containment can accommodate the pressure increases resulting from FCIs for both internal and external events. For in-vessel events, the staff has concluded that the resulting steam explosions can be accommodated by the four advanced designs under review. With respect to ex-vessel FCI, the determination of whether or not the event is credible depends upon whether the reactor cavity is wet or dry at the time of reactor vessel breech. If the design is based on a dry cavity with water entering after vessel breech, the staff has concluded that FCI is an incredible event. For a wet cavity, FCI must be considered when one determines the overall containment performance.

(vi) Direct Contact with the Containment Boundary;

This issue was found to be important for several types of ORs, namely BWRs with Mark I containments, some PWRs with ice condenser containment and for at least one PWR with large volume containment.

For advanced and evolutionary LWRs, the issue of direct contact with the containment boundary is not addressed separately, in part, because it is recognized that the new designs will provide sufficiently large floor space for debris coolability so that the direct contact becomes a less critical matter. SECY-93-087 specifies, however; that as part of the ex-vessel debris coolability criteria, the new designs must meet the requirement of protecting containment liners and other structural members against direct thermal attack, if necessary.

Case studies

- Studies of the potential of early failure of the steel liner (shell) in BWR Mark I containments (NUREG/CR-5423)

- Follow-up studies of the flooded Mark I containment (NUREG/CR-6025)

(vii) Slow Containment Overpressure

For advanced and evolutionary LWRs, the issue of slow containment overpressurization has been addressed in SECY-93-087 in the context of ex-vessel debris coolability (refer to Question 4-x). Specifically, the NRC established that all future LWR designs (advanced and evolutionary) must ensure that the best estimate environmental conditions (pressure and temperature) resulting from core-concrete interactions do not exceed Service Level C for steel containments or Factored Load Category for concrete containments, for approximately 24 hours. The designs must also ensure that the containment capability has margin to accommodate uncertainties in the environmental conditions resulting from core-concrete interactions.

(viii) Basemat Melt-through;

For advanced and evolutionary LWRs, the issue of basemat meltthrough has been addressed in SECY-93-087 as part of the containment performance goal in the context of core debris coolability. Specifically, the NRC established that all future LWR designs (advanced and evolutionary) must ensure a leak-tight containment (no liner or shell meltthrough) for approximately 24 hours, and for the range of severe accidents of concern, the designs must incorporate sufficient margin to accommodate uncertainties in the calculated extent of meltthrough resulting from core-concrete interactions.

(ix) Containment Bypass;

With respect to ISLOCA, the staff position is presented in SECY-90-016 (evolutionary) and SECY-93-087 (passive). The major design difference between ALWRs and operating reactors in regard to ISLOCA is preventive in nature. In particular, the EPRI standards require that all interfacing systems extending outside containment "...shall be designed to the extent practicable to an ultimate rupture strength (URS) at least equal to full RCS pressure. For those interfacing systems which do not meet the full reactor coolant (RCS) ultimate rupture strength (URS) requirement, the Plant Designer shall determine by evaluation that the degree and quality of isolation or reduced severity of the potential pressure challenges are low enough to preclude an intersystem LOCA." In this case, additional preventive measures include:

- Capability for leak testing pressure isolation valves;
- Isolation valve position indication;
- High-pressure alarms to warn control room operators when rising RCS pressure approaches the design pressure of the attached low-pressure systems and both isolation valves are not closed.

An additional requirement for the passive BWRs is:

- the design pressure of the Reactor Water Cleanup (RWCU) system shall be at least as high as that of the RPV.

An additional requirement for the passive PWRs is:

- the Passive Residual Heat Removal (PRHR) system shall be designed for full RCS design pressure and temperature.

(x) Others.

Ex-vessel core debris coolability

In SECY-90-016, the staff proposed the following criteria for advanced reactors:

- Provide sufficient reactor cavity floor space to enhance debris spreading; and
- provide for quenching debris in the reactor cavity.

The Electric Power Research Institute (EPRI) proposed a minimum floor area requirement of 0.02 m^2/MWt and provisions to flood the lower drywell or reactor cavity to meet the criteria above. Several experiments (SWISS, FRAG, and MACE series) were conducted in the past under NRC and NRC/EPRI joint sponsorship to understand the phenomenon and to determine the limit of debris coolability. The results of these experiments have been inconclusive. As a result, the staff developed revised criteria to assure containment integrity in the face of this uncertainty. The new criteria, published in SECY-93-087, are:

- provide reactor cavity floor space to enhance debris spreading;
- provide a means to flood the reactor cavity to assist in the cooling process;
- protect the containment liner and other structural members with concrete, if necessary; and
- ensure that the best estimate environmental conditions (pressure and

temperature) resulting from core-concrete interactions do not exceed Service Level C for steel containments or Factored Load Category for concrete containments, for approximately 24 hours. Ensure that containment capability has margin to accommodate uncertainties in the environmental conditions from core-concrete interactions.

QUESTION 5

The following addresses severe accident management issues. A list of items is provided. For each item describe the requirements, the thinking, and the plans of the regulatory body:

- criteria to declare SA status;
- strategies and procedures;
- instrumentation and equipment needed;
- qualification needed;
- training.

Staff Response

Criteria to Declare Severe Accident Status

The US approach to accident response does not rely on precise knowledge of the severity of core damage, or the ability to define or declare the onset of a severe accident. Rather, accident response is based on a more general assessment of the potential state of the core given the previous, current, and projected status of major plant systems and safety functions. Knowledge of the state of the core is important for two purposes; (1) determining the appropriate Emergency Action Level (EAL) and initiating the associated onsite and offsite protective measures, and (2) determining when entry into Severe Accident Management Guidelines (SAMG) is appropriate. These determinations would generally be made by the same personnel in the same time period, however, the criteria used for each of these purposes is different, and have not been integrated. The two types of criteria are discussed below.

- Criteria for EAL classification are plant-specific, and is part of each licensee's Emergency Plan Implementing Procedures. These procedures are required by 10 CFR 50.47 and 10 CFR Appendix E.

  For purposes of initiating protective measures, events are classified into one of four EALs based on the extent to which the barriers to fission product release (fuel elements, fuel cladding, reactor coolant system, and containment system) have been or are anticipated to be compromised. The four classes are: Unusual Event, Alert, Site Area Emergency, and General Emergency. A general description of each class is provided in NUREG-0654, Rev. 1, along with expected licensee and state/local authority response actions for each class. An important licensee response action is the activation of the Technical Support Center (TSC) for any accident classification more severe than an Unusual Event. Example initiating conditions for each class are also provided in the NUREG to form the basis for establishment of more specific decision criteria by each licensee. These initiating conditions tend to be event-based rather than symptom-based, e.g., emergency core cooling system initiated and discharged to the vessel, or loss of offsite power and loss of onsite AC power for more than 15 minutes. Accordingly, classification of events into EALs is generally based on a diagnosis of plant conditions and trends, rather than on plant parameters (such as core exit thermocouples) exceeding specified values.

- Criteria for entering SAMG will be part of the plant-specific accident management guidance that will eventually be implemented by each licensee. The plant-specific guidance will be developed by the owners group for each NSSS design. Plant-specific implementation will be on a voluntary basis. Although there are no specific regulatory requirements in this area, the regulatory view is that transitions and interfaces between Emergency Operating Procedures (EOPs) and accident management procedures/guidelines should be well-defined and unambiguous.

Strategies and Procedures

As stated in SECY-89-012, one of the fundamental objectives of the US accident management program is that each licensee implement an accident management plan which provides a framework for preparing and implementing severe accident operating procedures. Although there are no specific regulatory requirements in this area, the objectives of this effort should be to:

- identify strategies to address each of the accident management goals, that is; (1) prevent core damage, (2) terminate core damage and retain the core within the reactor vessel, (3) maintain containment integrity, and (4) minimize off-site releases.

- use strategies that maximize use of existing plant equipment and capabilities, including non-safety grade equipment,

- take preparatory measures to minimize the adverse effects of implementing strategies and uncertainties, for example, construct and pre-stage cables, adapters, jumpers, and spool pieces,

- identify and assign responsibility for the implementation of each accident management strategy to specific personnel or positions in the emergency response organization. Also identify alternates for key individuals,

- provide accident management procedures (AMPs) for the operating crew to implement strategies, for example, provide procedures with well defined conditions for strategy initiation and steps for implementation, and for manning the technical support center,

- provide accident management guidance (AMG) for the technical support staff and managers for strategies with negative effects (for example, radiological releases or irreversible equipment damage),

- structure AMPs and AMG to: (1) use the most reliable plant instrumentation, (2) provide guidance on instrumentation output interpretation during severe accident conditions, and (3) identify alternate means of obtaining information,

- include potential system responses to strategies and appropriate actions for these responses in the AMPs and AMG,

- provide well-defined and unambiguous transitions and interfaces between EOPs, AMPs, and AMG, and

- coordinate AMP and AMG development with EOP development, following rules specified for EOP development, to ensure that the resulting guidance and procedures are compatible, clear, and useable.

Accident management strategies and procedures have been the focus of three major activities in the US, as discussed below. These are; (1) Supplement 2 to the Generic Letter regarding the Individual Plant Examination (IPE), (2) the NRC research program on accident management strategies, and (3) the severe accident management guidelines being developed by the owners group for each NSSS design.

- Supplement 2 to the IPE Generic Letter. On the basis of existing PRAs, the NRC identified several generic accident management strategies that could enhance a licensee's ability to cope with the accident scenarios that tend to dominate risk in PRAs. These strategies were oriented towards prevention rather than mitigation of core damage.

- NRC Research Program on Accident Management Strategies. Following the publication of Supplement 2 to Generic Letter 88-20, the USNRC Office of Research redirected its efforts to identification and assessment of mitigative strategies, i.e., strategies which would be pre-planned to aid the operating crew in mitigating the consequences of core damage, vessel, or containment breach. This effort was undertaken to provide a technical basis for the staff's evaluation of the industry guidance on accident management, which was expected to include significant information on accident management strategies. Strategies considered as part of this effort include; (1) the need for and efficacy of negative reactivity insertion during reflood of a damaged core (NUREG/CR-5653), (2) the effectiveness of external reactor vessel flooding as a strategy for preventing or delaying vessel failure during a severe core damage accident (NUREG/CR-6056), and (3) PWR primary system depressurization (NUREG/CR-5937).

- Severe Accident Management Guidelines (SAMG). The SAMG being developed by the owners group for each reactor vendor provide specific guidance and strategies for severe accidents. In developing these guidelines, the owners groups have considered the accident management strategies identified in; (1) the EPRI "Severe Accident Management Guidance Technical Basis Report", dated

December 1992, (2) the IPEs performed by member utilities, and (3) NUREG/CR-5474 and other contractor reports developed as part of the NRC research program on accident management.

The staff intends to perform a limited review of the guidance documents in view of the potential impacts on execution of emergency operating procedures and decisionmaking within the utility emergency response organization.

The review will focus on assuring that; (1) the accident management guidance is appropriately interfaced with EOPs, and does not conflict with the strategies embodied within the EOPS, (2) the accident management strategies as implemented in the accident management guidance (AMG) are generally consistent with the current understanding of severe accident progression and phenomenology, and (3) the AMG can be used by utility technical support staff and guideline without consuming undue emergency response organization resources, thereby detracting from the effectiveness of the technical support function.

Instrumentation

As stated in SECY-89-012, as part of accident management implementation licensees would be expected to review the information retrieval capability during a severe accident to confirm (1) reliable indication of parameters needed to assess plant status and implement and monitor AM strategies exists, and (2) that Accident Management Procedures (AMPs) and Accident Management Guidelines (AMG) rely on appropriate plant instrumentation. Licensees should assess instrumentation survivability in a severe accident environment, instrument and support system availability for each phase of the accident, instrument response to severe accidents, parameter interpretation outside its normal range, and use of alternative existing instruments to provide required information. the focus of this effort is to:

- identify and document information needed to select, implement, and monitor each AM strategy and to assess plant status as part of the AM planning process,

- identify the availability, reliability, and survivability of instruments and their support systems during the sequences in which they would be needed, and

- structure AMPs and AMG to rely on available plant information systems, to provide guidance on interpreting instrumentation outputs during severe accidents, and to use alternate means of providing information.

The US accident management program is aimed at promoting the most effective use of available utility resources (hardware and personnel). The scope of the program is limited to use of existing plant equipment, and does not extend to identifying and implementing major hardware changes to reduce the frequency of core damage or risk. Although not the aim of this program, limited, minor hardware changes and equipment modifications may be identified during program implementation.

Training

A key objective of accident management is for licensees to provide a broader base of severe accident training for operations staff, technical support staff, and managers, commensurate with the responsibilities of these personnel under accident conditions. In this regard, it is expected that as part of accident management implementation licensees would evaluate and, as appropriate, upgrade their training programs to assure that:

- severe accident training is provided and maintained for all personnel with accident assessment and mitigation responsibilities,

- training programs are developed using a systems approach to training,

- personnel responsible for authorizing implementation of accident management strategies receive additional training specific to these strategies.

Severe accident training materials are being developed as part of the U.S. industry accident management program, for use by licensees. The Institute of Nuclear Power Operations (INPO) is developing generic training materials and guidance for both PWRs and BWRs. An initial set of tasks important to severe accident management has been developed based on the accident management strategies (candidate high level actions) treated in the EPRI Severe Accident Management Technical Basis Report. Three different types of individual accident response roles have been defined in conjunction with the task list, namely, "evaluator," "decision maker," and "implementor." Defined tasks and knowledge items have also been identified for the evaluator and decision maker positions in the form of a task training matrix. Existing operator training programs for the most part already address the role of the implementor.

FIGURE 1
SEVERE ACCIDENT PROGRAM - SCHEMATIC