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MINUTES OF THE ACRS AD HOC TMI-2 ACCIDENT IMPLICATIONS SUBCOMMITTEE MEETING REGARDING NUCLEAR POWER PLANT DESIGN
WASHINGTON, DC
MARCH 5, 1980

The ACRS Ad Hoc Subcommittee on the Three Mile Island 2 Accident Implications Regarding Nuclear Power Plant Design held an open meeting on March 5, 1980 in Room 1046, 1717 H St., NW, Washington, D.C. The purpose of this meeting was to discuss the implications of the March 28, 1979 accident at the Three Mile Island, Unit 2 station and to discuss recent studies on additional engineered safety features at Indian Point, Units 2 and 3 and Zion, Units 1 and 2. Notice of this meeting was published in the Federal Register on February 19, 1980. A copy of this notice is included as Attachment A. A list of attendees for this meeting is included as Attachment B, and a schedule for this meeting is included as Attachment C. Selected portions of the meeting handouts are included as Attachment D. A complete set of handouts has been included in the ACRS Files. There were no written statements or requests for time to make oral statements received from members of the public. The Designated Federal Employee for this meeting was Mr. R. Major.

EXECUTIVE SESSION

Dr. Okrent, Subcommittee Chairman, opened the meeting by stating the purpose of the meeting, which was to discuss recent studies on additional engineered safety features at Indian Point, Units 2 and 3, and Zion, Units 1 and 2. The Subcommittee, in joint session with the ACRS Ad Hoc TMI-2 Accident Action Plans Subcommittee, heard a briefing on the February 26, 1980 transient that took place at the Crystal River-3 Nuclear Station.

INTRODUCTION - ZION/INDIAN POINT TASK FORCE - J. Olshinski, NRC Staff

Mr. Olshinski said that as a result of TMI-2 follow-up actions, the Staff has looked in more detail at emergency planning and evacuation in general. Because of the high population densities surrounding the Zion and Indian Point Plants, the Staff is undertaking a special review of these plants. The Task Force is attempting to address the question of whether or not these plants, because of the high population densities, should add additional accident mitigation features not required at other plants. Severe

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accident mitigation features being considered include filtered-vented containments, core retention devices, and hydrogen control methods.

Mr. Olshinski mentioned a Union of Concerned Scientist petition regarding Indian Point (IP) that requested decommissioning IP 1 and shutting down of IP 2&3. The Staff held meetings in early February with the Commissioners. A showcause order was issued in relation to decommissioning IP-1; the petition was denied regarding IP 2&3. However, confirmatory orders were issued to IP 2&3 and then on Zion 1&2 regarding a number of interim operational actions to be taken at these plants because of the high population densities involved. The Commissioners have issued a solicitation for public comment on the NRR Director's decision concerning the petition regarding the IP plants and concerning the orders issued to these plants.

The Staff concern centers on assuming the PWR plant design of WASH-1400 was moved to the Zion or IP sites. If the design is assumed essentially the same as in WASH-1400 together with the higher population density, there would be a significant increase in risk.

Certain interim operational actions, for the Zion and IP plants, will be required by the Staff while reviewing and evaluating severe accident mitigation features. In addition, the Staff is pursuing, under an accelerated schedule, any outstanding plant specific or generic actions at these plants. The interim operational actions include certain staffing and training requirements, improving testing and maintenance, augmentation of the onsite technical staff, certain operational requirements, certain analyses that are being conducted, and certain actions and reviews by the NRC Staff. Slides 1-5 list these items and indicate the number of days after the order is issued for implementation of each item. Some example items included: additional SRO manning, containment leak test, vendor representation onsite, restrict plant to base load operation, and analyze control room habitability.

The Staff noted that the long-term design changes for severe accident mitigation features will require evaluation, development, and consideration as to whether or not they are feasible and what benefits they will provide. In the meantime, a number of interim operational actions (dealing with staffing and training, test and maintenance, augmented onsite technical staff, etc.) will be required. The Staff feels these items taken collectively are of value and should be done as the long-term effort proceeds.

OPERATING REACTORS AT HIGH POPULATION DENSITY SITES - J. Meyer, NRC Staff

Mr. Meyer noted that the IP and Zion sites are believed to present a disproportionately high contribution to the total societal risk from reactor accidents. The cumulative population around these sites is greater than that suggested in Regulatory guide 4.7 and the average site. The Staff has asked the IP and Zion licensees to determine what additional measures and/or design changes can and should be implemented that will further reduce the probability of a severe reactor accident and will reduce the consequences of such an accident by either reducing the amount of radioactive releases and/or by delaying any radioactive releases which would provide additional time for evacuation near the sites.

Three key points Mr. Meyer made were:

- Based on population density and assuming similar accident probabilities, reduction of accident consequences by a factor of 10 would bring IP and Zion, down to the same consequences as those of an average site (assuming similar meteorology and evacuation routes).

- Based on a number of studies (Sandia, BCL, UCLA) a number of mitigation features (such as a filtered-vented containment system for example) have been identified that have the potential to reduce the risks from severe accidents by at least and order of magnitude.

- NRC has a program underway, in parallel to the utilities program, to identify that package of viable systems that is sufficient to produce at least an order of magnitude in risk reduction and to require that these systems be installed in a timely manner (~2 yrs.) in these plants.

The purpose of NRC's severe accident mitigation features study is to determine how immediate and practical technical fixes can be implemented in the Zion and IP units that assure a real and significant reduction in societal and individual risk due to severe accidents, including core melt. The general approach is to pursue actively those design features that contribute favorably toward the mitigation of the consequences of a severe accident. Mr. Meyer noted that it is not the Staff's intention to design mitigating features,as this would be the responsibility of the utilities. The NRC does plan to work through conceptual designs considering all practical alternatives.

There are three severe accident mitigation features that are being addressed in the Staff's program. These are filtered-vented containment systems, core retention devices, and hydrogen control methods. In addition, the Staff is proceeding with steam explosion studies and accident risk evaluations.

Mr. Meyer continued with an overview of the basic components of the NRC Zion/IP program. A key input into the Staff's evaluation will be the Zion/IP specific integrated reliability evaluation program reviews. This program will be used as input to determine significant risk contributors and accident sequences for Zion and IP. Part two of the program will be to determine the evolution of accident scenarios. The MARCH/CORRAL system of codes will be used to develop a pressure, temperature and radiological source term history in the containment. This in turn will be input to a consequence evaluation for Zion and IP (CRAC analysis). In addition to determining the consequences specifically for the Zion and IP plants as they are currently built; this program can also consider the change in consequences when mitigating features such as filtered-vented containments are added to the system

Mr. Meyer noted that in terms of mitigating features, presently the most important in terms of effort is the filtered-vented containment system. The goal is to determine the feasibility of a filtered-vented containment system, assuming the accident history determined by the MARCH/CORRAL system of codes. Effectiveness and reliability of conceptual designs will be studied.

The Staff has a program in place to address the question of hydrogen control. The goal is to determine the behavior of hydrogen in containment and practical methods of controling it through burning or burn suppression. A third mitigation feature being studied is a core retention system. The goal of this study is to determine the utility of these devices vis-a-vis their contribution to lowering risk from both atmospheric and liquid pathways.

Dr. Okrent asked the Staff if there were recent studies on liquid pathways which were site specific. Dr. T. Speis of the NRC Staff believed a recent Sandia study on the liquid pathway had site specific aspects and agreed to send Dr. Okrent a copy of the draft report.

Mr. Meyer returned to component five of the Staff's program, which deals with structural response, and structural failure mode analysis. The goal of this effort is to determine realistic failure pressures and failure modes (Dynamic/Static)from Zion/IP containments and reactor vessels.

(NOTE: At this point, the TMI-2 Accident Action Plan Subcommittee joined the TMI-2 Accident Implications Subcommittee and together they heard a presentation on the February 26, 1980 Crystal River-3 Transient. This portion of the meeting is recorded in the minutes of the March 5, 1980 TMI-2 Accident Action Plan Meeting.)

The sixth component of the Zion/IP program is a study of steam explosions. The purpose of this effort is to assess the potential and magnitude of a steam explosion (based on "BEST" current analytical and experimental information) and the impact of realistic "steam explosion" events on vessel and containment failure. The final element in the program is to establish current thinking for comment and review on mitigative system design criteria.

Mr. Meyer explained that the containment failure modes under consideration are essentially the same as the modes considered in the WASH-1400 study. The six failure modes, using the WASH-1400 nomenclature are:

- α mode - containment rupture due to steam explosion
- β mode - containment failure resulting from inadequate isolation
- γ mode - containment failure due to overpressure (hydrogen burning)
- δ mode - containment failure due to overpressure (non-condensibles and steam)
- ϵ mode - containment failure due to melt-through

V vent - low pressure injection system check valve fails resulting in direct path from primary system to atmosphere

The filtered-vented containment system (FVCS) program is divided into three components. The input data that is necessary in order to do conceptual designs is the first component. The second component is the exploring of the conceptual designs themselves and the third component is the performing of consequence analysis using the CRAC Code.

Some of the key input parameters taken into account in considering filtered-vented containment systems included pressure, temperature, aerosol, and radiological source term histories from the MARCH/CORRAL and independent analyses. Variation of histories due to the presence of core retention devices, hydrogen control, accumulator water control, and restoration of AC power are also considered.

Various histories are calculated assuming certain prominent accident sequences. There are three basic categories taken from WASH-1400. The first is TMLB' which assumes a loss of offsite and onsite AC power for an indefinite period. In this sequence of events the primary system maintains its integrity until the core melt fails the lower vessel head. A second accident scenario that is considered in generating the input for the conceptual designs is a large LOCA with hydrogen burn. A third scenario is a small LOCA with failure of ECC inspection, again, anticipating a hydrogen burn.

The Staff is looking at conceptual designs for filtered-vented containment systems (FVCS). They are taking into consideration practical layouts, presence or lack of AC power; decontamination factors achievable; practical design flows; activation levels; operator/automatic controls; venting to atmosphere vs to special building; and environmental requirements (seismic, tornado, etc.).

In response to questions, Mr. Meyer explained that based on preliminary Sandia work a four foot diameter penetration is appropriate and can accommodate a certain spectrum of the pressure histories studied. However, in certain transients depending on how fast the pressure spike rises (for example the accumulator dumping its inventory on a molten core in the reactor cavity) a 20 foot diameter vent would be required. The feasibility of such a vent is questionable from a design standpoint. Also being considered is the practicality of early venting to accommodate pressure spikes.

In response to a question from Dr. Okrent, Dr. T. Speis (NRC Staff) noted that in some recent studies, involving a core melt scenario, 70% of the cases involving containment failure occurred before molten core-concrete basemat interaction. In 30% of the cases, interaction of the core melt and concrete basemat, leads to the generation of additional noncendensibles which contribute to containment overpressure. Dr. Okrent observed that this might be a departure from the analysis contained in WASH-1400.

One concept being explored is venting to another building. This building would act as a heat sink enabling cool/dry air to be returned to the containment.

Consequence analysis will be done with the CRAC code. It will analyze the impact of mitigation features with and without: noble-gas attenuation, and AC power to FVCS.

Another mitigating feature under consideration is a hydrogen control system. Practical methods of hydrogen control include controlled burning and/or providing an environment which suppresses ignition. Also under study are hydrogen gas dynamics; its diffusion and mixing characteristics. A third area of study is the actual hydrogen burning/explosion dynamics and the pressure time space evolution of a hydrogen burn or a hydrogen explosion. A fourth important area of study is hydrogen detection and operator intervention capabilities. Another area of study involves the potential for control in and damage to a FVCS from hydrogen.

An investigation of the proposed benefits of a core retention device is a third area of study into accident mitigating features. The benefit of a core retention device in regards to the liquid pathway is being explored as a delaying or complete stopping device of a molten core penetrating the basemat. A core retention device also effects the atmospheric pathway by providing a reduction in: containment pressure, aerosol generation from molten core concrete interactions, and hydrogen gas generation (and other combustibles) from molten metal constituents (steel) in contact with concrete. For Zion and Indian Point in particular an assessment must be made of the practicality/feasibility of installing a retention device, including the negative aspects of installing such a device in an already built and operated reactor.

Considering the various mitigating features, the Staff will perform reactor consequence evaluations. The purpose will be to evaluate and assess the consequence reduction achievable by various mitigation features.

The Staff is currently performing case calculations for Zion and IP based on the Surry design, but taking into account site specific meteorology, and population distribution, plant specific power levels, release fractions of WASH-1400 PWR release category probabilities. The calculations will then be repeated postulating the addition of a FVCS. Several options for the treatment of noble gases, plus other potential mitigation features (e.g., hydrogen control) will be considered.

A third consequence evaluation will be made assuming containment basemat melt-through, and using migration times, estimate consequences via liquid or groundwater pathways with and without mitigation features (e.g. core ladle).

Mr. Meyer discussed design and quality requirements for Class 9 accident mitigation systems. This effort involves the setting of design and quality requirements for the mitigative system(s) whether they involve a filtered-vented system, a core retention system or some combination of these other systems. Staff requirements are preliminary and subject to change pending further licensee/Staff interactions and evolution of system designs. Conservative design criteria applied to design of ESFs will be avoided, considering the low probability of the events considered. In general, the design approach should be reasonable and evaluated on a realistic basis where possible. Group C quality standards as defined in Reg. Guide 1.26 should be applied to mechanical and fluid systems where appropriate. The systems shall be designed and analyzed to remain functional for all of the operating basis environmental conditions, including the loads imposed by an operating basis earthquake. For more severe conditions the plant should be shut down for inspection of the mitigative systems. In addition, the mitigative systems should be evaluated for design basis loads, including the safe shutdown earthquake, in order to insure that there will be no gross failure which might impart or impair the functioning of safety class components/systems needed for design basis events.

Dr. Okrent observed that he was at loss to judge the adequacy of the preliminary requirements given without more explaination of the rationale used in deciding on the criteria. Mr. Meyer noted that these criteria were presented only as a starting point for discussion and to generate comments. Mr. Marchese of the Staff felt before final criteria were set, cost-benefit studies should be done.

Mr. Meyer presented a tentative schedule for the Staff's task force work. By the first week of April, a preliminary NRC FVCS design study to set preliminary design criteria should be complete. In mid-July the NRC will update FVCS design criteria. By the first week in December, the Staff intends to complete its FVCS design study. In parallel with these efforts, the Staff, together with the utilities, is planning technical meetings on "key" issues/areas relating to a FVCS and other technical areas related to severe accident mitigation.

Currently, there are five meetings planned to address key issues. The objective of these meetings will be to obtain relevant information and expert opinion on a number of technical areas pertaining to designing, selecting, and evaluating the effectiveness of severe-accident mitigating features. The subject areas of the five meetings planned are:

1. Accident Scenarios And Related Phenomenology-Evolving to Core Meltdown.
2. Material Interactions.
3. Hydrogen Dynamics and Hydrogen Control Measures.
4. Mitigating Features Filtered-Vented Containment and Core Retention Systems.
5. Structural Response to Dynamic/Static Loading.

A number of questions which highlight the program were given. Answers are sought for the following questions:

- Are the present analysis methods and their experimental/theoretical basis sufficiently adequate to be used as a basis for the design of severe accident mitigation features?
- What will be the role of probabilistic evaluations in choosing the set of accident scenarios that will form the design bases for the severe accident mitigation features?
- For that subset of accident scenarios that are amenable to immediate technical fixes, is there reasonable assurance that a significant reduction in risk can be achieved?
- Finally, practical problems/considerations in designing and implementing a severe accident mitigation feature; design criteria, potential system interactions (i. e., creation of new accident paths) must be addressed.

In conclusion, Dr. Speis offered the philosophy that the Staff believes they have reached a point of diminishing returns in significantly reducing the probability of events outside of the current design basis. If a general improvement in safety beyond that level is required, then the area of accident consequence mitigation seems to have the potential for the largest risk reduction.

PRESENTATION BY LICENSEES - L. Peoples, Director of Nuclear Licensing for Commonwealth Edison

Mr. Peoples presented the results of the Indian Point/Zion Near Site Study along with the program results to date. He noted that the owners of IP and Zion feel very strongly that these plants do not pose any greater risk to the public than the average plant in this country. This conclusion takes into account the specific design features that were built into these plants when they were first licensed, as well as their site location and meteorology.

Mr. Peoples noted that the Staff requested the owners of IP 2&3 and Zion 1&2 (Consolidated Edison, the Power Authority of the State of New York, and

Commonwealth Edison) initiate studies in three areas:

1. Means to mitigate the effects of a core melt.
2. Means to reduce the probability of a core melt.
3. Potential interim actions.

This meeting took place on December 5, 1979. The owners were given 60 days to complete the study program. The owners were told the general objective was to improve the time available for public evacuation given a core melt. The results of the 60-day program were presented to the NRC Staff on February 20, 1980.

The interim actions have been addressed via NRC confirmatory orders to Indian Point and Zion Stations. Mitigation of core melt consequences, received the bulk of the attention from the NRC Staff during the early meetings. Substantial emphasis was placed on the filtered vented containment concept by the Staff. The owners were directed to consider this concept and others regardless of the probability of a core melt. The owners were also directed to investigate means to reduce the probability of such a core melt.

Mr. Peoples explained that Zion and IP were viewed by Mr. Denton as presenting substantially more risk than the WASH-1400 plant at its composite site. Mr. Denton informed the Commissioners that it was his goal to reduce the risk posed by IP and Zion by a factor of about ten so as to bring these plants in line with an average plant. The owners contend, based on more detailed evaluations, that Zion and IP, as built, are lesser contributors to risk than the WASH-1400 average plant.

As part of the 60-day Near Site Study, the owners conducted a mini-WASH-1400 study of IP and Zion taking into account major design differences between these plants and the reference plant. Actual site data for Zion and IP were used. Consequence models were used to conduct a very preliminary evaluation of ideas for the mitigation of risk (filtered vented containment) and reduction in probability of a severe accident (training, testing of components, etc.)

The owners of IP and Zion are actively engaged in a more detailed, longer term, probabilistic risk assessment of these plants. This work will confirm that the accident sequences selected represent the dominant contributors and may identify additional areas where the probability of severe accidents could be reduced. The firm of Pichard, Lowe, and Garrick has been hired as consultants to aid in this effort.

The starting point for the present Zion/IP work was the table of dominant PWR accident sequences from WASH-1400, the initial assumption being that the accident sequences which dominated risk in WASH-1400 would also dominate risk for IP and Zion. During examination of the Zion and IP systems cause was found to both add and delete from this set of accident sequences.

Risk characteristics for the reactors at the Zion and IP sites were calculated in a preliminary fashion using the probabilities of each release category and the CRAC code developed for the Reactor Safety Study (RSS). The core inventories of fission products used in the RSS were adjusted proportionately for the respective power outputs of the Zion and IP reactors. Actual demographic data and site meteorology for the Zion and IP sites were used. The evacuation model employed in the RSS was used.

A more detailed quantitative WASH-1400 type evaluation of the Zion and IP plan is part of the longer term follow-on studies. This detailed study will develop refined risk estimates for dominant accident sequences and indicate whether sequences with substantial contributions to risk have been omitted in the short-term study. However, the short-term study is regarded as an adequate basis for study of major alternatives.

Program Results To Date

Base Plant Probabilities and Risks

One of the conclusions reached in this study was for a core melt accident. Containment failure by melt through may be postulated to occur if the containment has not failed earlier by some other mode. Given that water is likely to be present underneath the reactor vessel at melt through and that reflux of water to this region after vessel melt-through will continue, basemat melt-thrbugh is highly unlikely. Basemat melt-through is assigned the residual probability remaining after other failure mode probabilities are subtracted from 1 for TMLB'.

Assignment to Release Categories

For each combination of accident sequence and containment failure mode, there will result a particular quantity of fission products released to the environment. The total spectrum of containment releases was divided into seven discrete categories for core melt events, this followed the pattern from WASH-1400. Types of sequences assigned each category can be summarized as follows: steam explosion failures without containment safety features functioning were assigned to release Category 1 and steam explosion failures with safety features functioning were assigned to Category 3. Containment overpressure failures without safety features functioning were assigned to Category 2 and overpressure failures with safety features functioning were assigned to Category 5. Containment failure via basemat melt-through were assigned to Category 6 if containment safety features for fission product removal were not functioning and to Category 7 if they were functioning. Release Category 4 was not utilized in this study because none of the sequences evaluate for Zion and IP resulted in release within Category 4.

Accident sequence and containment failure mode combinations were assigned to appropriate release categories and total probabilities for each release category were obtained by summing. The result was that accident sequences in fission product release Categories 2 and 5 exhibited the highest probabilities of occurrence.

Mr. Peoples argued that Zion and IP represent risks consistent with the industry. This is in part he said the result of care taken in the initial design of these plants. A few of the specific differences noted from the reference plant include: the inclusion of containment fan coolers at Zion and IP; the third diesel-driven containment spray pump and Zion; additional diesel generators at Zion and IP; more extensive use of power-operated valves rather than manual valves on Zion and IP safety systems. This also includes the use of confirmatory ESF signals to appropriate valves. The use of a third series check valve at Zion on the RHR cold leg injection lines; and the use, at Zion and IP, of three types of ECCS pumps in each train as opposed to two types in the reference plant. These examples were given as the kind of differences which affect the risk curves.

Mr. Peoples again stressed the fact that the important conclusion from the risk calculations is that major contributors to risk are Category 2 and Category 5 releases. The principal failure modes for these release categories are the containment overpressure failures. Risk through these failure modes are potentially amenable to reduction by containment failures which reduce the probability of or eliminate these failure modes.

For overpressure due to noncondensible gases or excessive steam generation, containment vent systems with filter capabilities for fission product removal or retention may be one of several potentially useful concepts.

Steam Explosion

Mr. Peoples described steam explosions within the vessel. A mechanism for vessel failure was described as a steam explosion in the lower plenum of the reactor vessel which accelerated a continuous overlying liquid layer in a piston-like manner until it impacted on the vessel head. It is then postulated that the vessel head fails and is propelled against the containment wall with enough energy to cause failure of the containment. He then noted that available experiments show that a pressure of 150 psi is sufficient to eliminate explosions. Consequently, when the primary system pressure is above this value, steam explosions will not occur.

Mr. Peoples noted that for some postulated accident sequences, such as a large break LOCA, the system pressure can be less than 150 psi at the time core melt conditions are hypothesized. In this case, the molten core material could fall into the water in the lower vessel plenum. However, the fuel melt and fragmentation process itself is responsible for ensuring that a continuous overlying liquid layer, which is required to fail the containment via missile generation does not exist.

It was also noted that the behavior of a steam explosion within the vessel at low system pressures would resemble that of a shallow underwater explosion. If it is postulated that a continuous overlying layer exists, even though it cannot, the radius of a subsurface explosion would quickly approach the depth of the mixture and the bubble would break through in its first expansion. The phenomena to be observed would be a hollow splash. This would definitely not lead to the slug type of impact used for containment failure analysis.

For conditions in which molten fuel is assumed to melt through and be discharged from the reactor vessel, the core material would come into contact with water in the reactor cavity at pressures where explosions could occur. The major issue to address in the case of an ex-vessel steam explosion is the shock from the explosion itself.

Typical maximum interaction pressures from steam explosions were given to several hundred psi, but even using a conservative value of half the critical pressure (1600 psi) at the cavity radius, the expansion from the reactor cavity to the containment walls reduces the shock wave overpressure to about 1 atm. Such a shock wave does not pose a threat to the containment integrity.

Considerations of the spectrum of conditions representing possible core meltdown scenarios show that: (1) For the in-vessel case, vapor explosions are eliminated when the system pressure exceeds 150 psia. (2) For the low-pressure in-vessel case, the continuous overlying liquid layer required to fail the containment via missile generation is precluded for all reasonable fragmentation levels. (Fragmentation levels would have to be more than 2 meters in diameter in order to nullify this conclusion.) (3) For the ex-vessel case, the shock waves resulting from the steam explosion are below the containment design pressures when they reach the containment walls.

Core Coolability

The coolability of the core in an unconfigured geometry is investigated by finding the minimum amounts of water needed to maintain coolability.

In order for a core to become badly damaged, water must not only be lost from the primary system but it must also be kept out of the core for an extended period of time. In this accident scenario, fuel heatup and melting would begin near the top of the core, eventually leading to the collapse of core material upon melting and refreezing at lower core elevations. A system such as this could become completely blocked at the bottom but would still have water accessible from the top. Thermal conduction analysis of a sphere with uniform internal heat generation at decay heat equal to 1% full power and the outer surface in contact with water show that particles of up to 10 inches in diameter are permanently coolable. Such characteristic sizes would present a coolable configuration if water could be supplied to the core region.

For in-vessel coolability, Mr. Peoples discussed a damaged core configuration which is completely blocked at the bottom. This is the most conservative case since any leakage path through the bottom would greatly enhance the coolability of the core and would permit water from a cold leg injection to be available for heat removal. In the event of a complete blockage, the damaged core must be cooled by water supplied from above the core such as hot leg injection or leakage around the outlet nozzles from the downcomer. Two phenomena must be considered in evaluating the coolability of a damaged core; first, the ability of the water to contact the top of the core and second, the penetration of the water down through the distorted core. The particular plants in question have hot leg injection, which allows water to be added directly to the upper internal region. In addition, a reflux heat removal path can be established and can be accomplished with a small fraction of the heat tranfer area in one steam generator which is operable in the presence of noncondensible gas such as hydrogen. In summary, considerations of a badly damaged core, which is assumed to be completely blocked at the inlet, show that the core is coolable if water can be supplied to the upper surface.

To begin the ex-vessel cooling scenario, the complete absence of primary cooling water must be assumed. The core material would averheat due to the decay power and failure of the reactor vessel could not be ruled out. If such a failure occurred, and its coolability in this region is determined by the availability of water to the reactor cavity. Conclusions reached were that the cooling water that must be lost from the primary system to result in core damage is sufficient to establish a heat transport path th remove the decay power. Considerations of the ex-vessel debris bed show the bed will indeed be coolable. The onwers of IP and Zion have, as a result, concluded that a core ladle is not required and have dropped this concept from further consideration. The Staff noted that they had heard the same presentation from the owners of IP and Zion. However, the Staff has not reached any conclusions regarding their study.

Dr. Speis of the Staff noted that he though it was premature to reach a conclusion that a core ladle was not required. Dr. Speis noted that there were questions remaining in the field of coolability of debris beds such as how far the core material spreads and what size the fragments will ultimately take. (The owners claim that a 10-inch size or smaller fragment represents a coolable configuration for a debris bed.

Dr. Okrent suggested the seismic design basis for these plants might be reviewed to determined the probability per year with some uncertainty that is defensable, that a larger earthquake might occur which could change preliminary conclusions. A more quantitative evaluation of the seismic contribution to risk for IP and Zion might prove such considerations to be a significant contributor to risk. Such a conclusions might affect decisions that have been drawn regarding the possible usefulness of some kind of core retention device. Dr. Okrent noted a concern about abandoning additional engineered safety features at this stage. He though such a conclusion before more information is developed to be premature.

Transient Containment Analyses

Containment transients were evaluated for core melt sequences similar to those found limiting in the short term WASH-1400 study. Those sequences were a large break with no ECCS, a small break LOCA with no ECCS, and a loss of all AC power with loss of heat sink. The mass and energy releases to the containment were obtained for each of these sequences. This data was used in scoping calculations for the containment response.

Each plant with its individual heat sinks, containment volume, and containment safeguards systems were modelled. Among the phenomena studied were: hydrogen burning, containment venting, and continuous core melt cooling. These parametric studies indicated that the possibility of combusti[illegible] gas burning was one of the key parameters which affected containment response.

Preliminary results from the studies showed: (1) LOCA melt sequences with minimum containment safeguards and no hydrogen burn are acceptable in terms of containment capability. In fact they remain within containment design pressure. (2) LOCA melt sequences with minimum containment safeguards and continuous H_2 burn go slightly above containment design pressure but do not exceed containment capability. (3) LOCA melt sequences with containment safeguards and H_2 burn when satisfying the "flame temperature criteria" (if the calculated flame temperature of the containment mixture exceeds this criteria, then the containment atmosphere was flammable and could burn if a spark or ignition source were present. The 710°C criteria corresponds to 8 1/2% hydrogen in room temperature dry air at which spherical flame propagation commences), can exceed containment capability in some cases. (4) TMLB' sequences (loss of all AC power with loss of heat sink) and either continuous H_2 burn or H_2 burn when the "flame temperature criteria" is exceeded will exceed containment capability. Calculations with a continuous steam generation in containment indicate that the H_2 may not burn. However, the containment design pressure can be exceeded from the steam generation alone.

The vented containment study performed by Battelle (NUREG-CR-0138) has been reviewed and used by the owners as a guide to examine containment venting. Calculations performed by the owners used as a reference, two 12-inch vents with a realistic back pressure and line resistance. The calculations indicate the vent would help some sequences, particularly TMLB' and the LOCA sequences with minimum safeguards which had continuous steaming. The scoping calculations indicated that if a very rapid and large energy additions occurred in the containment, this size vent could not keep the pressure below the containment capability. Mr. Peoples stressed, however, that these were preliminary results and are subject to confirmation or changes as investigations continue.

Directions of Conceptual Design

The effects associated with steam explosion phenomena as analyzed earlier, were of minimal significance and readily accommodated by presently available structures and components in the plant.

The investigation of combustible gases and their potential effects on containments suggested consideration of measures which include the containment vent concept. Several other approaches have also been studied, each of which deals with either consuming the gases in a controlled fashion of adjust the vapor mix constituents in containment such that combustion is prohibited. Continued work in this area will be conducted to insure that a complete program evaluation is performed.

The potential for core debris melt-through and resulting containment failure was dismissed based on the technology review and analysis discussed previously.

Two methods for a filtered containment vent were discussed. The first candidate uses a sparger condenser and water scrubber. This system provides the means to vent the containment in a controlled manner and prevent containment failure by overpressurization. The system consists of two pressure operated isolation valves, a vent line with an orifice, and a sparger tank. Gases enter the vent system when the containment pressure reaches 60 psia and the containment isolation valves open. An orifice limits initial flow through the valves and the vent line. When containment pressure drops, the containment isolation valves close.

The vent line discharges into a manifold which is located in the bottom of the sparger tank. Gases exit the manifold and pass up through the water where heat is removed. The sparger tank has the capacity to absorb a total of 2 billion BTU. Water saturated gas is vented from the sparger tank to the atmosphere at temperatures ranging from 50°F to 180°F. The process is estimated, on a preliminary basis, to provide a decontamination factor of 100 for particulates and a decontamination factor of 10 for molecular iodine.

A tank with a usable volume of 1,875,000 gallons is required. Such a tank would be approximately 80 feet in diameter by 50 feet high.

The second candidate is a filtered containment vent with a submerged gravel filter. It is similar in concept to the first candidate. The size of the submerged gravel scrubber is 130 feet square by 25 feet deep. To fill the gravel bed, almost 25,000 tons of gravel is required.

Mr. Peoples noted that the vent system examples described, do not offer a clear, well-defined reduction in risk at this time. It is important to take into account the as yet unquantified reduction in overall safety which may be caused by such systems. Such reductions in safety could result from the possibility of failures, inadvertent activation during lesser transients or accidents and possible system interactions. The work of the owners in evaluating these concerns and balancing them against possible gains in safety is far from complete.

<u>Future Action</u>

Mr. Peoples noted that continued action in this program consists of maintaining two parallel, closely coupled efforts. The owners plan to complete a long term probabilistic risk assessment program currently underway with consultants, Pickard, Lowe, and Garrick. Secondly, it is planned to proceed with further investigations of the core melt related technology and of potential mitigation concepts.

The owners program with Pickard, Lowe, and Garrick should be complete during late summer. All event and fault trees will be initially completed by spring. This program uses plant specific design information to develop these tools. The study will also use plant specific reliability data for sequence evaluation where such data is available and is meaningful. Site specific data on demography and meteorology is also used for both sites along with an improved CRAC program for modeling consequences.

A second phase of the continuing action concerns mitigation concepts. The owners plan limited further work in the areas of containment transient analyses and containment ultimate strength, and in the area of combustible gas evolution, behavior and combustion. This work combined with the results of the probabilistic assessment program serves as the basis for further discussions with NRC Staff.

The second portion of the continuing action on the part of the owners concerns concepts. Limited further work in the areas of containment transient analyses and containment ultimate strength, and in the area of combustible gas evolution and behavior is planned. That work, combined with the results of the probabilistic risk assessment program will serve as the basis for further discussions with the NRC Staff. Following these studies, if necessary, the owners will begin to prioritize alternative design concepts for mitigation features. Mr. Peoples stressed, however, that the owners still need a clear unambiguous safety goal for these plants. Such a goal would have to come from the Staff.

The owners believe that the definition of uniform, quantitative criteria and methodology for evaluating Class 9 events and evaluating potential features to reduce the probability of occurrence or mitigate the consequences of such events is necessary for further progress. The owners believe that the risk assessment methodology, applied reasonably, is the proper tool. They believe that a criteria based on WASH-1400 average risk is appropriate to use for decision making at this time. NRC concurrence with both of these approaches has been requested.

In concluding, Mr. Peoples noted that the results of their current assessment of the relative risks from Zion and IP show that these plants do not contribute excessively to public risk. These results coupled with the Interim Actions ordered by the NRC and coupled with the actions taken in response to NUREG-0578 give the owners a great deal of assurance that the mutual interest in excellence has been and is still being satisfied at these plants. The owners stressed there are still areas where substantial work remains to be done before any final selections of mitigating features are possible.

Mr. Peoples suggested that it was possible that studies would be completed in June of 1981. This is the present target date of the owners of IP and Zion.

CYRSTAL RIVER IREP PRELIMINARY INSITES ON CRYSTAL RIVER 3 - J. Murphy, PAS Staff

Mr. Murphy reported on the present progress on the Crystal River IREP study. Currently, the Staff has gone through the event trees, fault trees on all the systems the Staff deemed to be important. The Staff is trying to conduct the IREP study in the same way WASH-1400 was done -- with the minimal use of computers. The amount of hand calculations are increasing the amount of time necessary for this project. The Staff noted that they were finding quite a bit of coupling between supporting systems for this particular plant.

In performing the IREP study on Crystal River, the PAS Staff assumed that the things Crystal River has committed to have been performed and have been performed correctly. The Staff took the plant as they found it during the first week of December plus what the plant and committed to change at that time. That was the basis of the PAS analysis.

Mr. Murphy stressed that the IREP study was never intended to be complete and the PAS Staff is making no completeness argument. What was hoped to be accomplished in the IREP study was to find obvious problems and go through Crystal River and other plants in a rapid fashion. This means a thorough job would be impossible. To put it in perspective, Mr. Murphy noted that approximately 5 man-years of effort was focussed on Crystal River. Probably less than that amount will be expended on the rest of the IREP plants, maybe by a factor of 5. By contrast, he noted the Reactor Safety Study consumed 60 man-years of effort.

Mr. Murphy noted the preliminary nature of the IREP results. There are a whole class of accidents which have not been considered yet, the best example of those are the ICS related initiators like the February 26th event at Crystal River. This transient itself is on the event trees (the loss of feedwater with an opened PORV). However, the Staff has not gone into the depth of detail necessary to analyze this event.

Mr. Murphy indicated that he recognize that there are a whole family of accidents that are not included in this study. These other accidents include those related to non-nuclear instrumentation and the ICS. He did note that the PAS would take a quick look at those. Mr. Murphy noted that the IREP is still being organized with regard to what plant would be studied first. He noted that the Staff thinks it will take about 26 weeks to study a particular plant. Currently, the Staff believes that three plants will be done locally in Bethesda and three plants will be contracted out. At this stage, PAS is still trying to develop standardized techniques for the program. It is hoped to be able to put together relatively short manuals that describe the type of analysis PAS is looking for. In this manner, all six or seven groups working on the various plants will have generally the same basic format. They would draw their fault trees in the same basic style. Mr. Murphy concluded by saying that the main value he sees coming out of the IREP is a qualitative one rather than a quantitative one. The final output should be a listing of what effectively says "if you want to buy the greatest risk reduction for your buck, this is what you should attack."

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NOTE: For additional details, a complete transcript of the meeting is available in the NRC Public Document Room, 1717 H St., NW, Washington, DC 20555 or from International Verbatim Reporters, Inc., 422 South Capitol Street, SW, Suite 107, Washington, DC 20002, (202) 484-3550.

(a) Lighting fixtures on the side of the equipment would "blind" the operator and nearby miners or require constant adjustment to changes in illumination; fixtures would be sheared off or broken increasing the likelihood of more serious equipment failure, wedging, jamming or upset. Also, as lighting fixtures on the side or top are sheared off, roof bolts, cross beams and straps will be sheared off, thereby damaging or destroying roof support.

(b) Installation of stationary lighting equipment would similarly impair the operators' and nearby miners' vision. It would also create additional heat in the confiningly small areas in which the miners must work.

3. For these reasons, the petitioner requests a modification of the application of the standard to its mine.

Request for Comments

Persons interested in this petition may furnish written comments on or before March 20, 1980. Comments must be filed with the Office of Standards, Regulations and Variances, Mine Safety and Health Administration, Room 627, 4015 Wilson Boulevard, Arlington, Virginia 22203. Copies of the petition are available for inspection at that address.

Dated: February 11, 1980.

Frank A. White,

Director, Office of Standards, Regulations and Variances.

[FR Doc. 80-[illegible] Filed 2-15-80; 8:45 am]

BILLING CODE [illegible]

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards, Subcommittee on Three Mile Island, Unit 2 Accident Implications; Meeting

The ACRS Subcommittee on the Three Mile Island, Unit 2 Accident Implications will hold a meeting on March 5, 1980 in Room 1046, 1717 H St., NW., Washington, DC 20555 to consider the potential installation of molten core crucibles under the Indian Point 2 & 3 and the Zion 1 & 2 Reactors. Notice of this meeting was published January 22, 1980.

In accordance with the procedures outlined in the Federal Register on October 1, 1979, (44 FR 56408), oral or written statements may be presented by members of the public, recordings will be permitted only during those portions of the meeting when a transcript is being kept, and questions may be asked only by members of the Subcommittee, its consultants, and Staff. Persons desiring to make oral statements should notify the Designated Federal Employee as far in advance as practicable so that appropriate arrangements can be made to allow the necessary time during the meeting for such statements.

The agenda for subject meeting shall be as follows: Wednesday, March 5, 1980, 8:30 a.m. until the conclusion of business.

The Subcommittee may meet in Executive Session, with any of its consultants who may be present, to explore and exchange their preliminary opinions regarding matters which should be considered during the meeting.

At the conclusion of the Executive Session, the Subcommittee will hear presentations by and hold discussions with representatives of the NRC Staff, the Power Authority of the State of New York, the Consolidated Edison Co. of New York, Inc., the Commonwealth Edison Co., their consultants, and other interested persons.

In addition, it may be necessary for the Subcommittee to hold one or more closed sessions for the purpose of exploring matters involving proprietary information. I have determined, in accordance with Subsection 10(d) of the Federal Advisory Committee Act (Pub. L. 92-463), that, should such sessions be required, it is necessary to close these sessions to protect proprietary information. See 5 U.S.C. 552b(c)(4).

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by a prepaid telephone call to the cognizant Designated Federal Employee, Mr. Richard K. Major (telephone 202/634-1414) between 8:15 a.m. and 5:00 p.m., EST.

Background information concerning items to be discussed at this meeting can be found in documents on file and available for public inspection at the NRC Public Document Room, 1717 H Street, N.W., Washington, DC 20555 and at the Government Publications Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Street, Harrisburg, PA 17126 (regarding Three Mile Island), the White Plains Public Library, 100 Maritime Avenue, White Plains, New York 10601, (regarding Indian Point), and the Zion-Benton Public Library, 2600 Emmaus Avenue, Zion, IL 60099 (regarding Zion).

Dated: February 13, 1980.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 80-5096 Filed 2-15-80; 8:45 am]

BILLING CODE 7590-01-M

Advisory Committee on Reactor Safeguards, Ad Hoc Subcommittee on Three Mile Island, Unit 2 Accident Action Plan; Meeting

The ACRS Ad Hoc Subcommittee on the Three Mile Island, Unit 2 Accident Action Plan will hold a meeting on March 5, 1980 in Room 1167, 1717 H St., NW., Washington, DC 20555 to consider Draft 3 of the NRC "Action Plans for Implementing Recommendations of the President's Commission and Other Studies of the Three Mile Island, Unit 2 Accident."

In accordance with the procedures outlined in the Federal Register on October 1, 1979 (44 FR 56408), oral or written statements may be presented by members of the public, recordings will be permitted only during those portions of the meeting when a transcript is being kept, and questions may be asked only by members of the Subcommittee, its consultants, and Staff. Persons desiring to make oral statements should notify the Designated Federal Employee as far in advance as practicable so that appropriate arrangements can be made to allow the necessary time during the meeting for such statements.

The agenda for subject meeting shall be as follows: Wednesday, March 5, 1980, 8:30 a.m. until the conclusion of business.

The Subcommittee may meet in Executive Session, with any of its consultants who may be present, to explore and exchange their preliminary opinions regarding matters which should be considered during the meeting.

At the conclusion of the Executive Session, the Subcommittee will hear presentations by and hold discussions with representatives of the NRC Staff, the nuclear industry, various utilities, and their consultants, and other interested persons.

In addition, it may be necessary for the Subcommittee to hold one or more closed sessions for the purpose of exploring matters involving proprietary information. I have determined, in accordance with Subsection 10(d) of the Federal Advisory Committee Act (Pub. L. 92-463), that, should such sessions be required, it is necessary to close these sessions to protect proprietary information. See 5 U.S.C. 552b(c)(4).

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by a prepaid telephone call to the cognizant Designated Federal Employee, Mr. John C. McKinley

Attachment A

POOR ORIGINAL

ACRS AD HOC SUBCOMMITTEE MEETING ON TMI-2 ACCIDENT IMPLICATIONS
WASHINGTON, DC
MARCH 5, 1980

ATTENDEE LIST

ACRS

D. Okrent, Chairman
M. Carbon
S. Lawroski
W. Mathis
J. Ray
R. Major, Designated Federal Employee

CONSOLIDATED EDISON

W. Bennett
M. Scott

POWER AUTHORITY OF THE STATE OF NEW YORK

J. Davis
J. Schmieder
W. Sayed
C. Pratt
J. Bayne
R. Goyette

WESTINGHOUSE ELECTRIC CORP.

R. Slember
D. Goeser
L. Hochreiter
N. Liparulo
D. Paddleford
S. Jacobs
J. O'Cilka
R. Marchese
H. Keller
W. Kortier

BURNS & ROE

J. Rubin

NRC STAFF

R. DiSalvo
M. Medeiros, Jr.
J. Long
S. Acharya
L. Olshan
E. Reeves
A. Schwencer
A. Marchese
L. Soffer
T. Speis
J. Meyer
J. Olshinski
G. Zech

COMMONWEALTH EDISON CO.

G. Klopp
W. Naughton
J. Deress
D. Peoples
J. Marianyi

EPRI - NSAC

R. Leyse
E. Zebroski

MITSUBISHI

K. Okabe
A. Hoizumi

NEW YORK STATE ENERGY OFFICE

J. Dunkleberger

Attachment B

SARGENT & LUNDY

N. Weber
J. LaVallee

TAEC

O. Akalin

KMC

R. Boyd

ARGONNE NATIONAL LAB.

B. Spencer
R. Henry
D. Cho

ATOMIC INDUSTRIAL FORUM

R. Szalay

IVP

A. Young
S Glocker

DEPARTMENT OF ENERGY

J. Yerick

TENNESSEE VALLEY AUTHORITY

T. Price

SNUPPS

F. Schwoerer

OFFSHORE POWER SYSTEMS

R. Walker

BECHTEL CORP.

R. McDermott
N. Willoughby

ISHAM LINCOLN & BEALE

P. Steptoe

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

February 29, 1980

TENTATIVE DETAILED SCHEDULE - SUBCOMMITTEE ON THREE MILE ISLAND, UNIT 2 ACCIDENT IMPLICATIONS, ROOM 1046, 1717 H STREET, N.W., WASHINGTON, D.C., MARCH 5, 1980

Times are
approximate

8:30 a.m. Executive Session (Open)
Review schedule and add or delete topics

8:45 a.m. Meeting With NRC Staff (D.Eisenhut et al.)

Summary and discussion of additional engineered safety features for nuclear power plant in close proximity to large populations

10:00 a.m. Meeting With Utility Representatives (L.Peoples et al.)

1. Introduction
History
Summary

2. Discussion
Study Objectives
Utility Study Program
Utility Methodology
Utility Program Results
Future Plans

3. Conclusion

12:30 p.m. LUNCH

1:30 p.m. Meeting With NRC Staff - Discussion of other possible accident contributors (control system caused, seismically caused, etc.) such as the Rancho Seco occurrence of January 5, 1979.

5:00 p.m. Adjourn

J C McKinley
J. C. McKinley, Chief
Project Review Branch No. 1

Attachment C