# Indian Point/Zion Near Site Studies

## Presentation to NRC

December 20, 1979

Commonwealth Edison Company

Consolidated Edison Company

Power Authority of the State of New York

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On December 5, 1979, the NRC identified a potential concern for the effects of a very low probability core melt event and possible subsequent radiation release to populations near nuclear sites. Zion and Indian Point were selected as the two sites for immediate study because of their proximity to areas of higher population density.

Commonwealth Edison, Consolidated Edison, and The Power Authority of the State of New York have joined forces to review this potential concern. This presentation will address the evaluations conducted to-date and plans for future work in two areas:

1) Mitigation of the effects of core melt
2) Reduction of the probability of a severe accident.

In addition, we will define a plan of action and a schedule which will lead to a thorough evaluation of the risk of core melt and methods to reduce this risk. Finally, we will address considerations involved in plant design and operation which continue to assure public health and safety while detailed studies are in progress and the evaluated results are being implemented.

It is, in our opinion, extremely important to recognize that this safety review is consistent with the ACRS, NRC, and licensees' long standing philosophy of providing added safety margin for the Zion and Indian Point sites. Such philosophy evolved during the licensing process and resulted in features such as: containment liner weld channel pressurization systems; containment penetration pressurization systems; and the isolation valve seal water systems.

We have in place and underway a program which we feel is fully responsive to the points you raised in our last meeting. (Screen 1 & 2, Slide 1). This program contains studies and evaluations of the means to mitigate the effects of an unlikely severe accident, by reducing radiation release and increasing the available evacuation time. It also includes evaluations of potential means to further reduce the already low probability of a core melt. A preliminary task sequence and schedule has been developed which is responsive to the 60 day timetable established by the NRC in our last meeting.

Utilizing the resources of the three utilities and Westinghouse, we have already performed scoping studies to better define our current, substantial capabilities in the areas of core melt prevention and mitigation. This work includes a review of existing plant design, a review of our status relative to interim concerns, and preliminary probabilistic risk assessments.

## Mitigation of the Effects of Core Melt

The first area, mitigation of the effects of the very unlikely core melt, is centered around the development of means to increase the plant in-depth defenses so as to reduce or delay releases of radioactivity from such a core melt.

The task of developing alternative ways to mitigate the effects of a severe accident and to properly evaluate these means would normally be a lengthy and very complex engineering effort. However, we appreciate the need to be responsive to the timetable set forth in our last meeting. We are very fortunate in that a great deal of work in this area has already been performed. We have found that we can draw on work performed by Westinghouse, Off-Shore Power Systems, Clinch River, FFTF, Sandia, and others and tailor much of it to our plant specific considerations.

The overall program for developing mitigating features (Slide 2, Screen 1), involves four (4) basic steps:

1) defining the engineering problem
2) developing alternatives
3) evaluation
4) decision

Defining the Task

The first step is to define the task in realistic engineering terms, i.e., time to core melt, mechanism for radiation release, amount of radiation released, etc. To do this, we need to develop design conditions based on selected severe transient sequences from a plant specific preliminary probabilistic risk assessment utilizing WASH-1400 methodology (a mini-WASH-1400 review).

In this task, the accident sequence and component failure data base developed in WASH-1400 and in follow-on studies on a 4-loop PWR will be utilized to estimate failure probabilities for those accident sequences which are the dominant risk contributors for core melt accidents. Estimates have been derived for the plants under consideration by comparing the Zion and Indian Point systems and components to the systems and components in the reference plants already evaluated in detail. This comparison has been performed for all of the dominant accident sequences. Engineering judgment has been used to modify the reference study so that it reasonably models the actual systems design of Zion and Indian Point.

As part of this task, estimates of fission product release have been made for each of the dominant accident sequences and associated containment failure modes. For the mini-assessment, differences between fission product removal processes for the reference plants and Zion or Indian Point have been examined. Where differences are not significant, releases for a sequence have been assigned to the same release category employed for the reference plants. Where differences in fission product removal capabilities are significant for a particular sequence and associated failure mode, engineering judgment has been used to assign the sequence to the appropriate release category. For a PWR, seven (7) release categories were utilized in WASH-1400. They ranged in severity from the large atmospheric release associated with an early containment failure to a small atmospheric release associated with melt through of the containment base mat.

This mini-assessment task has been performed to provide early indication of which accident sequence contributes most to core melt accident risk. The preliminary severe accident risk spectrum for each plant will be utilized to assess the relative potential for risk reduction offered by various system modifications or by new containment features or systems.

A detailed quantitative WASH-1400 type evaluation of the Zion and Indian Point plants is planned as part of the longer term follow-on studies. This detailed study will define risk estimates for the dominant accident sequences and indicate whether sequences with substantial contributions to risk have been omitted in the mini-study. It is expected that risk values derived in the preliminary assessment will be within a factor of 2 of those derived from the quantitative study. In view of the detailed studies conducted by NRC for two other Westinghouse PWR's (a 3-loop PWR with a high pressure containment and a 4-loop PWR with an ice containment), the likelihood of missing an accident sequence which is a significant contributor to risk is regarded as small. Thus the mini-study is regarded as an adequate basis for early decision processes.

Development of Alternatives

The sequences which are major contributors to risk of a core melt accident will be utilized to identify one or more scenarios as design bases for the new features. Included in this process will be a definition of the accident transient associated with the design bases. The transient evaluation will lead directly to the definition of one or more sets of design parameters to be used in the development and evaluation of new mitigating features.

Our development and evaluation process will include those features commonly associated with the mitigation of the effects of a severe accident leading to core melt. In addition, we will have the benefit of plant specific engineering input and of our preliminary probabilistic risk analysis to ensure that other features of possible merit are considered. At this time, it is appropriate to consider the engineering factors involved in this effort (Slide 2, Screen 2). Our review will include preliminary or scoping design layouts for features or, at least, a close enough examination to determine that a feature should or should not be given further consideration. A number of factors will be considered in this work. These include determinations that the features:

- address the design goal (i.e., the reduction of risk to the population)
- can be added to the plant
- employ a sound technological base
- do not degrade other safety aspects of the plant
- can be implemented on a reasonable timetable, and
- are most effective compared to alternatives being considered.

Some of the features that will be considered are:

- Core ladles
- Controlled, filtered containment venting
- Hydrogen control measures, and
- Augmented containment cooling measures.

Core Ladle

The core ladle has historically been associated with core melt postulations and has a substantial background of development. This device is constructed by lining the bottom and side walls of the cavity beneath the reactor vessel with a layer of refractory material such as magnesium oxide. The purpose of the ladle is to retain the debris resulting from core melt for a period of time (a few days) while appropriate actions are taken to reduce the potential consequences of such a severe accident. In concept, the refractory is a sacrificial material; that is, the refractory material is not cooled.

Incorporation of a core ladle was required by NRC for the Floating Nuclear Plant for the purpose of delaying releases to liquid pathways so that liquid pathways interdiction could be initiated. It is not at all clear that a ladle provides similar benefits with respect to reduction of radioactivity release and dose consequences via air pathways following a core melt accident. This aspect of incorporating a core ladle into the Indian Point or Zion design will be evaluated as part of the task.

One related concept that will also be evaluated is cooling either the existing base mat material or a new refractory material following core melt. The purpose of cooling would be to increase the time which the core could be retained in the region below the vessel without melt through. Permanent retention while difficult to ensure, might be possible as part of this evaluation, both active and passive cooling schemes will be investigated.

Containment Venting

Controlled, filtered venting of the containment is a promising feature that has received attention recently. Such venting serves the purpose of relieving containment pressure through radioactivity removal systems at a controlled rate. The venting thereby further reduces the already small potential for overpressure failure of the containment and subsequent uncontrolled release of radioactive material. We intend to look very closely at such systems with a variety of treatment systems including sand filters and scrubbers. System capacities will be conservatively established and the effectiveness of the systems in addressing the dominant containment failure modes will be given careful consideration. Consideration may be given to elevated stack releases to augment these systems should the results of evaluations show that worthwhile improvements can be obtained.

The venting systems themselves will be evaluated with particular attention to ensuring that no degradation of other plant safety aspects occurs. Interactions with other systems will be very carefully considered. The venting systems will be considered with respect to those mechanisms leading to containment overpressure which are determined to be dominant risk contributors for these plants.

Hydrogen Control

One possible contributor in this regard is hydrogen evolution and combustion. A number of activities and alternatives for consideration have been planned to complete our evaluation of hydrogen control. (Slide 3, Screen 2). These are all considerations well beyond the hydrogen control required by current design basis accidents.

The first action will be to establish the need, if any, for specific hydrogen control features. Loads from postulated severe accident consequences will be applied to the containment to determine its response. Computer codes will be used to evaluate the limiting conditions the containment can withstand. These loads will be compared with those predicted in other parts of the program and an overall picture of capability will be developed. Should the load picture and hydrogen failure mode warrant, added features will be considered. These include:

1. Containment Venting
2. Multiple Ignitions Sources
3. Recombination
4. Containment Inerting
5. Removal of Containment $O_2$ on Demand (Aqueous Chemical Spray)

Augmented Containment Cooling

The possibility of augmenting existing containment cooling capability with systems of a diverse nature and/or power source will also be given some consideration. At first glance, this alternative appears least likely to be successful given space limitations, known technology, and other considerations. We will, however, give serious consideration to any concepts which appear to have merit in this regard.

Evaluation of Relative Effectiveness of Feasible Alternatives

The next step in our program, given a reasonable selection of competing alternatives, is to evaluate the effectiveness of these alternatives. To accomplish this, we plan to evaluate the relative reduction in dose to the population for each alternative and to compare these values to the values for the plants without any mitigating features.

This work will be done using the CRAC code. As part of the WASH-1400 evaluation, the CRAC code was developed for estimating both dose and risks (the product of probability and consequences) for releases of radioactivity to air pathways following core melt accidents. The form of the code utilized for WASH-1400 performs these estimates for generic sites. As part of this task, specific demographic and meteorological data for the Indian Point and Zion sites will be incorporated into the code.

Population distribution information for Zion based on 1971 statistics, including projections out to 1985, have been collected for use with evacuation plan and meteorological data in establishing baseline risk values. Meteorological data for Zion through 1975 has also been collected. Reviews are underway to establish the need for and feasibility of updating this information.

The most recent docketed meteorological information for the Indian Point site is found in the Site Appendix I Evaluation Report submitted to NRC on March 14, 1977. This report utilizes "average annual meteorological data" which was developed from actual meteorological data collected during 1974 and 1975.

The most recent demographic information is still that information included in the Indian Point 3 FSAR. It should be noted that even the most recently submitted Con Edison and PASNY revised Emergency Plans, which were submitted to NRC in November, 1979, utilized the demographic data in the Indian Point 3 FSAR. This demographic data is based on the 1970 population census and was projected to the year 2010.

The modified version of CRAC will then be employed for core melt accident consequence estimates for these specific sites.

The code will first be utilized for baseline risk estimates for the plants as now designed. Risk calculations will also be performed for the plants, with the proposed plant and system modifications, to estimate the magnitude of risk reduction available from the proposed modifications.

Parameters reflecting current evacuation plans for each of the two sites will also be incorporated into the CRAC code and calculations will be performed to estimate dose reduction from existing evacuation procedures to add to the perspective of the study.

It might be in order to touch briefly on the current status of the individual evacuation plans.

Commonwealth Edison has and will continue to work with the State of Illinois to develop evacuation plans which will receive NRC concurrence. An outside consulting firm has also been commissioned to independently develop a detailed study of times required to evacuate, meeting NRC requirements, for all of our nuclear plants. The first plant to be studied will be Zion. Such information as is available in the 60 day period will be factored into our study.

The most recent Indian Point 2 Emergency Plans were submitted to NRC in November, 1979, and were based on the new guidelines and criteria issued by NRC. These emergency plans are revised periodically to adhere to new criteria and regulations.

At the present time, Con Edison representatives are working with state and county officials to assist them in fulfilling their responsibilities to formulate local emergency plans in accordance with NRC guidelines. The county emergency plans will include planning for evacuation, should such a step be determined advisable by local or state governmental agencies.

The Power Authority of the State of New York has a similar program underway. In December, 1979, a joint effort was established to complete the evacuation analysis. This joint effort consisted of the Consoldiated Edison Company, The Power Authority of the State of New York, the New York State Nuclear Civil Protection Planning Section and the consulting firm of Parsons, Brinckerhoff, Quade & Douglas, Inc. The objective of this joint effort was to concentrate resources to ensure the completion of the evacuation analysis to the requirements of the NRC in their November 29, 1979, letter by January 31, 1980.

Decisions

Once the relative effectiveness of feasible alternatives has been determined, decisions can be made to implement those features which would significantly reduce risk for the two sites. Given the constraints of time, we feel this program is as thorough and responsive as possible. (Slide 1, Screen 1).

## Reduction of Probability of Core Melt

The next major portion of the program is to evaluate methods which may reduce the probability of severe accidents including core melt. Probabilistic risk assessments will be accomplished in two phases. In Phase A, WASH-1400 event sequences will be investigated in detail for Indian Point and Zion. The goal will be to identify the more likely sequences and those sequences leading to a rapid core melt. Phase B will expand the Phase A effort and result in comprehensive, plant specific, event trees and fault trees, which will be quantitatively evaluated.

The Phase A effort will be accomplished within the near term by two teams; one team for Zion and one team for Indian Point. Each team will consist of experts in probabilistic risk assessment and experts on plant systems and operations.

These teams will construct event trees specific to each plant. They will utilize relevant, previous work. This will include WASH-1400, work at the Electric Power Research Institute (EPRI), event trees for Diablo Canyon seismic analysis, and other recent work.

These models will utilize the best available data to determine for each plant the "important" sequences. Reliability data sources will include WASH-1400, EPRI, and recent NRC assessments. Plant specific experience will be used where appropriate.

This effort will allow us to identify major contributors to risk and to investigate the feasibility of plant modifications that might yield major reductions in risk. We will also employ this work as a check on the results of the mini-WASH-1400 review conducted as a part of our evaluation of alternatives which would mitigate a core melt.

Phase B of this work will be an effort to make the Phase A models more complete and to allow further evaluation of any design changes identified in Phase A. Phase A work will be modified with additional fault trees and, where necessary, more complete versions of Phase A event/fault trees. The data will be structured for evaluation by appropriate computer codes. The data utilized during this phase will be of the same sources as Phase A, but more time will be available to further consider plant experiences. During this phase, the baseline models (plant "as is" models) will be modified for any design changes which result from the design efforts paralleling Phase A. This revised model will be used to evaluate the improvement (i.e., reduction) in risk due to the design changes.

An additional benefit to be gained from Phase B will be utility versions of the Integrated Reliability Evaluation Program (IREP) models for each specific plant. Since the IREP for these plants will most likely be in progress during the same time frame, the utility effort and the NRC effort can be used as a basis for interchange which would assure comprehensive models of the specific plants. (Slide 4, Screen 2).

## Plan of Action and Task Sequence

A plan of action and first cut at a task sequence are shown on this viewgraph. We are serious about meeting the program timing you requested. We are already hard at work on these tasks. In addition to resources at The Power Authority of the State of New York, Consolidated Edison, and Commonwealth Edison, we have retained

Westinghouse and Argonne National Laboratory. We are considering other consultants in specialty areas. In the capacity of architect-engineers, United Engineers & Constructors and Bechtel are both assisting Consolidated Edison and The Power Authority of the State of New York, and Sargent & Lundy is assisting Commonwealth Edison.

Our plan of action for mitigation of the effects of core melt includes a detailed review of core ladles, filtered containment vents, hydrogen control, and augmented containment cooling. The analytical process for conducting this review has been described in detail earlier in this presentation.

Our plan of action for reducing the probability of a significant accident includes a determination of major contributors to severe transient probability. This task, described earlier as Phase A of our program, is underway. Once major contributors are defined, feasible modifications will be evaluated for their effectiveness in reducing severe transient probability.

The key point, for both the mitigation and probability reduction task sequences, is that we are scheduling completion for the end of the 60 day period requested by the NRC.

Response to Interim Concerns

The last major area we wish to address is the plant status relative to interim concerns. (Slide 3, Screen 1).

As Mr. Eisenhut pointed out in our meeting on December 5, 1979, both we and the NRC recognized the existence of above average population densities near the Zion and Indian Point sites during the early design and licensing stages of the jobs. Over 11 years ago, during the construction permit stages, we, the NRC and the ACRS had extensive discussions regarding this matter.

Out of these discussions and our own internal assessments, a common philosophy emerged relative to both sites. That philosophy called for extraordinary design measures to reduce the risk to the public and extraordinary measures during all phases of plant operation to that same end. Hardware features are built into these plants to meet this philosophy that are not found on contemporary plants or even newer plants. Provisions were made for the incorporation of even more features should extensive, on-going research programs show a need for such features. Plant operating and training measures were instituted which went beyond then current practice and the continuing evolution of our practices in this area have kept pace with or gone beyond current regulatory standards and industry norms. It is worth noting that many of the practices now considered routine by the industry and NRC trace their origins to work done at the Zion and Indian Point sites. These include such areas as quality assurance and single failure criteria.

We would like to present some specifics regarding these plant features which are still, after all these years, as a total package, unique to these plants. (Slide 5, Screen 2).

A number of extra features are found on all four units. These are:

1. Containment weld channel and weld channel pressurization system: All containment liner welds are enclosed by continuous linear channels welded to the liner to form a redundant seal at the joints of liner plates. Those channels which cover joints not buried in concrete are pressurized with air to a pressure exceeding calculated containment peak pressure. This eliminates leakage at liner plate joints.
2. Penetration pressurization system: In addition to the normal pressurization of electrical penetrations (with dry nitrogen), mechanical penetrations are pressurized with air to a pressure above calculated containment peak pressure. This eliminates leakage through penetration assemblies.
3. Isolation valve seal water system: Those double isolation valves, normally closed on a containment isolation signal, in water and small air systems have the area between valves filled (if needed) and maintained in a filled condition at a pressure exceeding calculated containment design pressure by this system. This eliminates any leakage of containment atmosphere via an open (or ruptured) line through the redundant isolation valves.

4. Extra containment fan cooler capacity: Each containment has 5 fan cooler units, 3 of which are required for post accident containment cooling. The added capacity provides assurance of system availability.

5. Post LOCA hydrogen control: Each unit has both recombiner and post-LOCA containment purge capability. The recombiner capability was added to provide added conservatism.

6. Third auxiliary feedwater pump: Each unit has 3 auxiliary feedwater pumps per unit. Two of these are 100% capacity motor driven pumps and the third is a 200% capacity steam turbine driven pump. All three pumps are intertied through lines and valves designed for an active or passive failure. This extra capacity over a 2-100% capacity pump configuration provides added assurance of system availability.

7. Added containment radioactivity removal has been provided. On Zion a third, 100% capacity, diesel driven, containment spray pump is installed for each unit. This added conservatism over a conventional, 2 pump per unit, configuration gives added assurance of system availability. On Indian Point, each fan cooler unit is equipped with HEPA and charcoal filters for post-accident particulate and iodine removal.

8. Confirmatory "S" signals: Confirmatory Emergency Safeguards Features (ESF) actuation signals are sent to power operated valves which are not required to change position. This ensures that, if a valve had inadvertently been placed in an incorrect position, it would move to the correct position upon ESF actuation. This has been applied to critical safety system valves.

In addition, each unit has extra margin in service water and component cooling water capacity and availability. They have augmented auxiliary building air filtration systems and closed valve leak off systems to reduce offsite exposure due to leakage. They have redundant electrical heat tracing on vital borated systems.

## Conclusion

In summary, a great deal of margin already exists for these units and has existed from the day they started operating.

As noted earlier, the original design and licensing philosophy of these two sites called for extra measures which would assure the safety of the nearby populace. In addition to hardware features built into these plants and plant operating and training, measures have been instituted which go beyond minimum requirements.

As valuable as this margin is, we have not ignored the constant evolution of the industry at large nor have we ignored recent history. (Slide 6, Screen 2).

Both the Zion and Indian Point Plants have, within the last two years, made substantial improvements in their safety margins. A number of these improvements are common to all three utilities. These are presented today. Other major improvements and factors contributing to safety margins exist. Each utility will be pleased to discuss these individually with you either after this presentation or at any other time convenient to you.

The first of the common actions is the implementation of the NUREG-0578 recommendations including, specifically, the shift technical advisor.

The second action includes the existence and use of plant specific simulators for training. These simulators are at or very near each site.

The third action consists of the use of either 5 or 6 shift rotation to reduce operator fatigue and promote training.

The fourth major action is the early incorporation of the new NRC operator qualification requirements into the training programs. This has been accomplished at Zion and is being implemented on the Indian Point units.

Other factors and actions also point to added margin. These include actions taken some time ago to enhance control room man-machine interfaces, constantly improving operating records, and high levels of management commitment and involvement in plant operations.

In conclusion, (Slides off) we feel we have presented you with an aggressive and responsive program to address the areas you presented in our December 5th meeting. We are already actively into this work. We are drawing heavily on expertise from all quarters of the industry. The full resources of each utility, Westinghouse Electric Corporation including Off Shore Power Systems are active in this effort. In addition, we expect to employ the expertise of national laboratories, outside consultants, and our architect-engineers in this work. We invite close coordination with your staff.

We have also presented some of the factors which contribute to the very substantial and very real extra safety margins enjoyed by these plants even without the further actions planned. We are convinced that our current and continued operations embody more than adequate extra margin at these sites.

We will be happy to expand on any of these remarks and to answer any questions you may have.

CECO, PASNY, CON ED HAVE INITIATED RESPONSIVE PROGRAM TO EXAMINE AND IMPLEMENT FEATURES WHICH ARE EFFECTIVE IN MITIGATING THE CONSEQUENCES OF SEVERE ACCIDENTS

STUDIES AND EVALUATION

- MITIGATION
- REDUCTION OF PROBABILITY
- PLAN/SCHEDULE

SCOPING EVALUATIONS INDICATE SUBSTANTIAL CURRENT CAPABILITIES TO PRECLUDE AND MITIGATE SEVERE ACCIDENTS

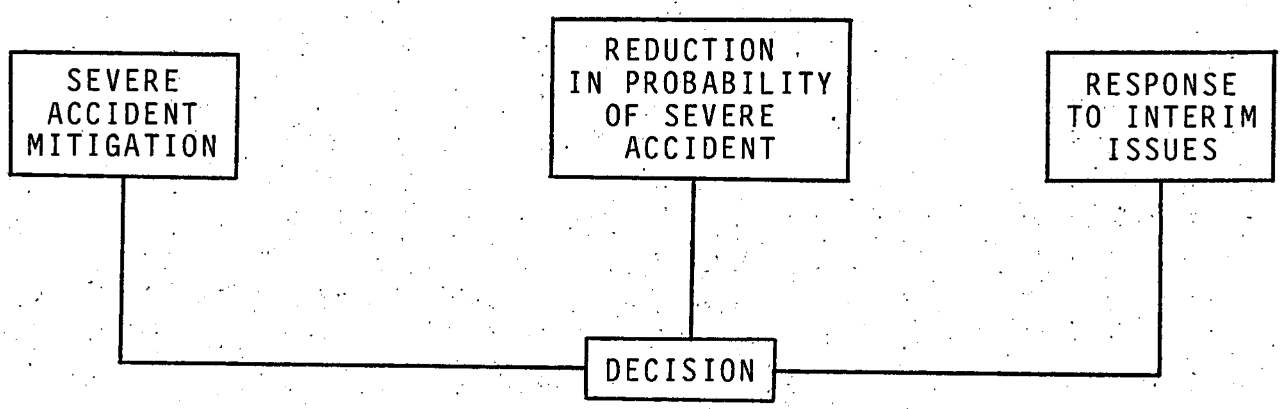
PLANT STATUS RELATED TO INTERIM CONCERNS

THIS ACTIVITY IS CONSISTENT WITH LONG STANDING UTILITY PHILOSOPHY TO PROVIDE AN EXTRA MEASURE OF SAFETY MARGIN FOR THESE SITES

Slide 1
Screen 2

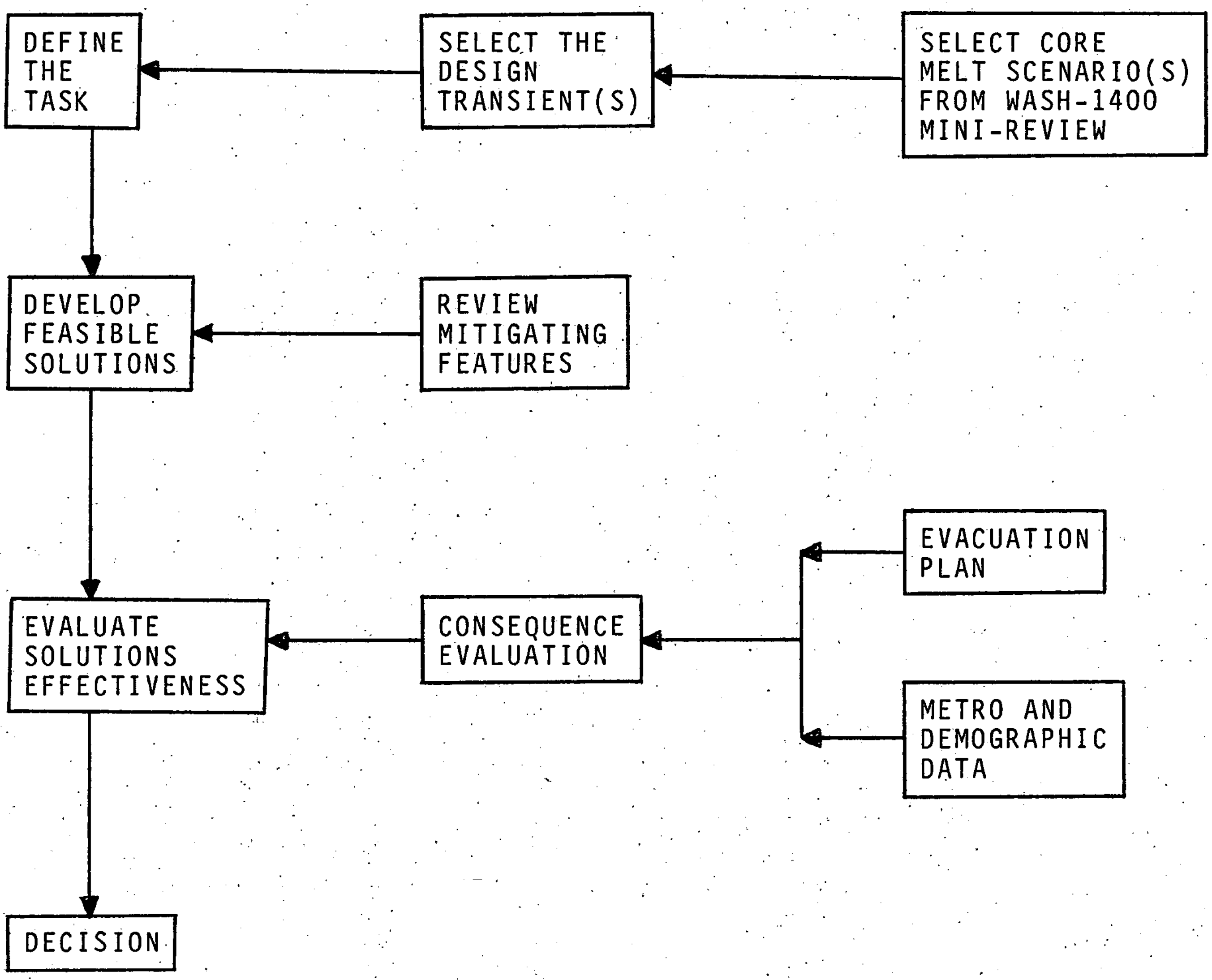
PROGRAM IS COMPREHENSIVE AND RESPONSIVE TO SEVERE ACCIDENT ISSUES

TOTAL PROGRAM




Slide 1
Screen 1

MITIGATING STUDIES PROVIDE A BASIS FOR DESIGN, EVALUATION, AND EFFECTIVENESS ASSESSMENT OF FEATURES TO REDUCE THE EFFECTS OF SEVERE ACCIDENTS




Slide 2
Screen 1

ENGINEERING FACTORS INVOLVED IN DEVELOPMENT OF MITIGATING FEATURES:

- FEATURE ADDRESSES DESIGN GOAL
- FEATURE CAN BE ADDED TO PLANT (Physical constraints)
- FEATURE EMPLOYS SOUND TECHNOLOGICAL BASE
- FEATURE DOES NOT DEGRADE OTHER SAFETY ASPECTS OF PLANT
- FEATURE CAN BE IMPLEMENTED ON REASONABLE TIMETABLE
- FEATURE IS MOST EFFECTIVE COMPARED TO ALTERNATIVES BEING CONSIDERED

## HYDROGEN CONTROL CONSIDERATIONS

- CONTAINMENT STRUCTURAL CAPABILITY
- CONTAINMENT VENTING
- MULTIPLE IGNITION SOURCES
- RECOMBINERS
- CONTAINMENT INERTING
- OXYGEN REMOVAL

## Plan of Action and Task Sequence
## Developed to Meet 60 Day NRC Requirement

| Mitigation Action Points | Probability Reduction Action Points |
|---|---|
| 1. *NRC Meeting/Initiate Study (12/5/79) | 1. *NRC Meeting/Initiate Study (12/5/79) |
| 2. *WASH-1400 mini-review and Design Sequence Selection | 2. *Investigate Consultants |
| 3. *Collect Metro and Demographic Data | |
| 4. *NRC Meeting | 3. *NRC Meeting |
| 5. Containment Transient Development | 4. Select Consultant |
| 6. Develop Feasible Mitigating Features | 5. Perform Phase A Program |
| 7. Evaluate Feasible Mitigating Features | 6. Evaluate Feasible Modifications |
| 8. **NRC Meeting | 7. **NRC Meeting |

* Item complete
** To be completed by 2/4/80

Slide 4
Screen 2

INDIAN POINT AND ZION PLANTS

INCLUDE ADDITIONAL SAFETY MARGINS

- ADDITIONAL FEATURES IN ORIGINAL PLANT DESIGN
- RECENT ACTIONS

Slide 3
Screen 1

ZION AND INDIAN POINT PLANTS INCLUDE

ADDITIONAL FEATURES TO PROVIDE SAFETY MARGIN

EXAMPLES:

CONTAINMENT INTEGRITY

- WELD CHANNEL & PRESSURIZATION
- ISOLATION VALVE SEAL WATER
- EXTRA FAN COOLER CAPACITY
- HYDROGEN RECOMBINERS

HEAT REMOVAL FEATURES

- AN EXTRA CAPACITY FEEDWATER PUMP
- 3RD DIVERSE PUMP

RADIOACTIVITY REMOVAL

- ADDITIONAL DIVERSE SPRAY PUMP OR CHARCOAL FILTERS

CONFIRMATORY SAFETY SYSTEM VALVE ACTUATION SIGNALS

## RECENT ACTIONS AT INDIAN POINT AND ZION PLANTS PROVIDE ADDITIONAL SAFETY MARGIN

SHIFT TECHNICAL ADVISOR

TRAINING / QUALIFICATIONS

- PLANT SPECIFIC SIMULATORS
- 5 OR 6 SHIFT ROTATION
- NEW NRC REQUALIFICATION STANDARDS MET OR BEING IMPLEMENTED

OTHER

- CONSTANTLY IMPROVING OPERATING RECORD
- ENHANCED MAN-MACHINE INTERFACE CONTROL ROOM
- HIGH LEVEL OF MANAGEMENT COMMITMENT AND INVOLVEMENT