INDIAN POINT AND ZION

NEAR SITE STUDY

Report to the

Nuclear Regulatory Commission

February 20, 1980

Commonwealth Edison Company

Consolidated Edison Company

Power Authority of the State of New York

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1.0 INTRODUCTORY REMARKS

On December 5, 1979, representatives of Commonwealth Edison, Consolidated Edison and Power Authority of the State of New York met with you here in Bethesda at your request. During that meeting you expressed your intention to review all operating plants and plants in the operating license stage to establish their capability to effectively reduce the risk resulting from postulated Class 9 accidents. You placed particular emphasis on core melt accidents. You further indicated that, while the Zion and Indian Point plants were no less safe than any other plant, your first priority in this effort was operating plants in areas of relatively high population density.

At the conclusion of the meeting, we agreed to respond to you, within 60 days, with the results of our studies in three areas: These are:

1. Ways to mitigate the effects of severe accidents
2. Ways to reduce the probability of severe accidents
3. Considerations regarding potential interim actions.

We notified you in late January 1980 that we would be prepared to furnish our response on or after February 4, 1980.

It might be worthwhile to summarize the program which we instituted in December 1979 to develop this response. The basic plan was presented to you on December 20, 1979 in some detail. We have not deviated substantially from the plan. This summary will serve to refresh everyone's memory and will identify some important study inputs which we alluded to briefly in our January 11, 1980 meeting.

(Slide 1)

This slide summarizes, briefly, the programs for studying ways to mitigate the effects of severe accidents and ways to reduce the probability of severe accidents. You can see that inputs from Argonne on steam explosions and containment base mat penetration have been factored into Item 4 on the left. You can also see from the last item on the left, that we have not made a final, restrictive, selection of any one design. This is a logical result of the short time allowed for our work.

The right hand column on this slide outlines our efforts on probability reduction. Steps 1 and 2 have been completed. Interim measures to reduce the probability of occurrence of a severe accident are being or will be taken following NRC orders. Steps 4 and 5 relate to the longer term program which will be described in more detail later in the presentation.

(Slide 2)

We propose to cover several areas relevant to our study today. In order these are:

1. The analytical approach used in our study
2. Our conclusions regarding certain technological aspects of the study
3. An overview of the features and actions in existence, proposed and under study to mitigate public risk
4. An outline of our proposed future actions
5. An outline of input needed from the NRC to complete this effort.

Our work allows us to state with a great deal of confidence that the features built into these plants have, in conjunction with recent interim actions, led to plants with risks no greater than those associated with plants having a lower level of demographic interest. We can also state that a great deal of conservatism is carried in our results. This preliminary effort needs a great deal of refinement and clarification before major decisions are reasonable or possible. However, the work we have done to-date suggests that some straightforward test procedures and simple hardware changes may be appropriate means to reduce the relative public risk.

I would now like to introduce Dr. D. Walker of Westinghouse Offshore Power Systems who will start our discussion of the analytical approach.

2.0 ANALYTICAL APPROACH

2.1 WASH-1400 Mini Study

There exists a broad spectrum of possible accidents of low probability that could lead to core melt should they occur. The WASH-1400 study developed a systematic methodology for estimating probability for these sequences in a consistent way so that the dominant sequences with respect to the contribution to risk could be ascertained. In our WASH-1400 Short Term Study, the accident sequence and component failure data base developed in WASH-1400 and in follow-on studies on a 4-loop PWR were utilized to estimate failure probabilites for those accident sequences which are the dominant risk contributors for core melt accidents.

The objective of the short term study was to establish, within a limited period of time, a reasonable estimate for the residual risk associated with core melt accidents for the Zion Nuclear Station, Units 1 and 2 and the Indian Point Nuclear Station, Units 2 and 3. In order to reduce the time required and in order to help preserve a frame of reference for comparative purposes, this study was based to the extent possible on the Reactor Safety Study, WASH-1400.

To calculate residual risk, one first identifies accidents beyond the regulatory design basis and establishes their probabilities. Knowing the characteristics of the accident, one proceeds to determine accident consequences. Finally probability and consequence are combined to arrive at risk.

The starting point for the present 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 likewise dominate risk for Zion and Indian Point. During examination of the Zion and Indian Point systems, cause was found to both add to and delete from this set of accident sequences in a few instances as will be discussed later.

In estimating accident sequence probabilities for Indian Point and Zion, the systems and components in these plants were compared with systems and components for the reference plants already evaluated in detail. This comparison was performed for all of the dominant accident sequences. Engineering evaluations have been used when necessary to modify the models employed in the reference study so that the models can be applied to the actual systems design of Zion and Indian Point.

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 short term study. It is expected that risk values derived in the preliminary assessment will be reasonably close to 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 short term study is regarded as an adequate basis for study of major alternatives.

Engineering review of Indian Point and Zion identified system differences from WASH-1400 that were particularly significant with respect to the probability of occurrence for some of the dominant accident sequences. These were factored into our work. Specifically, it was found that failure of containment spray recirculation could result directly from a loss of emergency coolant recirculation to the core. Sequences involving coincident loss of containment spray recirculation and core recirculation were not dominant in WASH-1400 because the systems, and hence these failures, were not coupled. Also in WASH-1400 a major contributor to risk is a sequence which proceeds from loss of containment spray injection following a small pipe break. In this sequence, containment failure occurs (from steam overpressure) before sufficient water collects in the containment sump to support containment spray recirculation. The capability to recirculate cooled sump water to the core is lost and core melt results. Both the Zion and Indian Point plants have fan coolers which are redundant to containment spray injection and whose failure is independent of the spray injection system. Thus the result obtained in WASH-1400 required an additional independent failure and the sequence (S_2C) is deleted from the dominant list.

In the aftermath of Three Mile Island, it is widely believed that for at least some sequences, core melt requires failures beyond those considered sufficient for core melt in WASH-1400. One example is the loss of auxiliary feedwater (heat sink) following shutdown. Studies more recent than WASH-1400 indicate that emergency cooling injection systems can provide the cooling necessary to avoid core melt provided the pressurizer relief valves are open. For this reason sequences were deleted which involve loss of secondary side heat sink, i.e., steam generator feedwater, except for the special case where there coexists a complete loss of AC power.

Finally, the two accident sequences involving transients followed by failure of the Reactor Trip System were deleted. These transients have been analyzed by Westinghouse, the NSSS vendor, and found not to result in core melt.

Having arrived at a set of accident sequences, each individual sequence probability was calculated by combining the probability of the initiating event with the probability that the required safety systems fail upon demand.

Except for the probablity of interfacing check valve failure and the frequency of offsite power interruption (at Zion only), initiating event probabilities were adopted without change from WASH-1400. Check valve failure probability was calculated for each plant using the same model as WASH-1400 but taking into account plant-specific differences. The frequency of offsite power transients for the Zion analysis was based on data applicable specifically to the Zion Station. For Indian Point, the values in WASH-1400 were employed.

The probability of safety systems failure was calculated in the form of point estimates based on component failure rates. In general, component failure rates used in this study are obtained from data in WASH-1400.

The second phase of this study involved the calculation of accident consequences and the transition from accident probability to risk. Again, the methodology is basically that of WASH-1400.

For each accident sequence leading to core melt, there are a number of ways in which containment failure might potentially occur, each with its own probability of occurrence. For each combination of accident sequence and containment failure there will be a unique source term upon which to base radiological consequence calculations. In WASH-1400, the spectrum of source terms was divided into seven discrete categories; each combination of accident sequence and containment failure mode was then placed in the most appropriate release category. This approach was used here as well to retain a frame of reference for the study results.

The conditional probability that a particular containment failure mode will occur (given that a specific accident sequence occurs) was estimated from several sources, primarily from WASH-1400. The combination of accident sequence leading to core melt and containment failure mode is the definition of an overall scenario which enables one to define a source term for consequence calculations.

Accident scenarios were assigned to a release category on the basis of (1) the amount of radioactivity released to containment, (2) fission product removal from the containment atmosphere and (3) the magnitude of the release following containment failure. Placement of the dominant accident sequences in the appropriate release category leads to the determination of the total probability of occurrence for each release category by summing.

(Slide 3)

This viewgraph indicates the end point of this process for the Zion & Indian Point plants. Accident sequence & containment failure mode combinations were assigned to appropriate release categories and total probabilities for each release category are obtained by summing.

With this introduction, we shall treat the way in which accident sequence probabilities, containment failure mode probabilities and assignment to release categories are developed and combined in enough detail to permit you to understand the method we employed. We shall also describe how these results are combined with consequence values to arrive at relative risks.

First of all, we shall look at the way an accident sequence probability was derived. We will use the AH & AHF sequences as examples. The AH sequence involves a large coolant pipe break and a loss of recirculation phase emergency core cooling. The AHF sequence involves the AH sequence combined with a loss of recirculation phase containment spray capability.

In sequence AH, core melt occurs because Emergency Coolant Recirculation(ECR) fails to operate properly. The next viewgraph is a diagram of the Low Pressure Recirculation System.

(Slide 4)

Successful ECR requires flow from at least one of the RHR pumps taking suction on the containment sump. Successful ECR also requires that the operator make no errors during the shift from injection to recirculation which cannot be corrected.

The Low Pressure Recirculation System (LPRS) provides containment spray recirculation flow to the spray header in addition to core recirculation.

For this reason certain LPRS failures will also result in failure of containment spray recirculation, which is designated Event F.

Looking specifically at the system diagram, two redundant trains provide suction from the containment sump and discharge to both the RCS and CSRS. The first step in calculating system unavailability is to group system components into blocks. The flow diagram is then converted into a much simpler block diagram which is shown in the next viewgraph.

(Slide 5)

At the top of this viewgraph is the block diagram of the LPRS. Directly below are schematics which show the components which comprise each block. The probability of component failure within each block is calculated using component failure rate data extraced from WASH-1400. For example, consider block (1) which happens to be the same as block (3) - component failure rates are shown beneath each component. The total failure probability for block (1) is the sum of its individual component failure probabilities, or 3.6×10^{-2}.

Now the system failure probability is calculated using the block diagram. Individual blocks are combined into single, double, or higher level failures which can result in loss of one or both the system function. Note that single, double, etc., refer to the number of blocks which fail, not the number of individual component failures. For this particular system there is no single failure which results in loss of system function. After the combinations of blocks are defined, the probability of each combination is formulated. The sum of the probability for each block is the system failure probability for the particular function being considered.

Specifically for the Zion plant, there are two distinct ways in which the system can fail to perform its function: First, loss of core recirculation only. That is, containment spray recirculation is not lost. The combinations leading to this mode of failure are:

(1) and (4) with a probability of 1×10^{-4}
(2) and (3) with a probability of 1×10^{-4}
and (2) and (4) with a probability of 7.8×10^{-6}

The total probability of LPRS failure with no failure of CSRS is 2×10^{-4}. This is designated failure H.

The second way in which the LPRS can fail is that in which CSRS is also lost. Only one block combination, (1) and (3), leads to this failure mode and the probability is seen to be 1.3×10^{-3}. This is designated failure HF. In order to calculate overall sequence probability, the initiating event probability and mitigating systems failure probability are combined. Other factors, such as operator error, are also considered.

First, consider sequence AH which is a large break LOCA (Event A) followed by failure of ECR to the core. In WASH-1400 two types of operator error were considered for LPR systems. The first of these was failure of the operator to shift to hot leg recirculation after 24 hours. The probability of the failure to perform this action is significantly lower than previously thought because the time of switch-over is not critical (can be accomplished over a 12-hour period) and because of the probability of increased operator attention. Therefore, since this action is not in fact required to prevent core melt, it has not been included in this analysis. The second type of operator error is failure to properly perform the shift from the recirculation mode such that recirculation capability is lost. This type of error appears to apply to the HF sequence as will be discussed subsequently. Calculation of overall probability for event AH is indicated on the next viewgraph. You see the probability was calculated as the product of the pipe break probability (A) and the probability of loss of recirculation function (H).

(Slide 6)

Next consider event HF which consists of both a hardware contribution and an operator error contribution. We have just shown how the hardware contribution of 1.3×10^{-3} was arrived at. The operator error contribution of 3×10^{-3} is taken directly from WASH-1400. The error postulated in WASH-1400 is not defined but leads to loss of recirculation. Since most of the operator manipulations involve the pumps and suction piping (which are common to both core recirculation and containment spray), it is likely that operator error would contribute to event HF, rather than similar Event H. The probability of Event HF is then the sum of hardware and operator contributions. The probability of sequence AHF is calculated by multiplying p(HF), and p(A).

This discussion illustrates the method generally employed throughout the study to develop sequence probabilities.

Our next topic is containment failure mechanisms and probabilities.

Containment Failure Mode Probabilities

For there to be significant release of radioactivity of levels beyond the design basis, some type of failure of containment function must occur beyond the core melt accident sequence. For each core melt accident sequence, there are a number of possible containment failure modes. In WASH-1400, five containment failure modes were considered and they are indicated on the next viewgraph.

(Slide 7)

They are: 1) in vessel steam explosion generating a missile and failing containment (Designated α), 2) failure of containment isolation (Designated β), 3) overpressure failure resulting from relatively rapid hydrogen combustion (Designated γ), 4) overpressure failure resulting from accumulation of steam and/or non-condensible gases (Designated δ), and 5) failure resulting from containment melt-through (Designated ϵ).

Steam Explosion

For the in-vessel steam explosion containment failure mode, the probability of 0.01 suggested by WASH-1400 is modified for the sequences initiated by large breaks (A) to a magnitude of 10^{-3}. For sequences where core melt will likely occur while the reactor coolant system is still pressurized, recent data suggest in-vessel steam explosion is much less likely. For these sequences (S_1, S_2 and TMLB'), the probability of an in-vessel steam explosion has been reduced by two orders of magnitude to 10^{-4}. Details supporting these values will be presented later in this discussion.

Failure of Containment Isolation

For failure of containment isolation, values from WASH-1400 have been employed in this study.

Over Pressure

For containment overpressure failures, values were not specifically identified in WASH-1400 although they can be derived in some cases from the data tables. As part of the study for Indian Point and Zion, containment temperature and pressure calculations following various accidents are being performed and will provide some basis for overpressure failure estimates. Values in the range 0.1 to 0.2 appear to have been used in WASH-1400 except for TMLB' sequence where a value of 0.8 was used. For the purpose of this study a value of 0.1 was used for containment overpressure failure probability except for TMLB' sequence. For TMLB' the value of 0.8 from WASH-1400 was used. For the purpose of this study, the hydrogen combustion and steam or non-condensible overpressure failure modes were considered together because release category assignment appears to be the same for both failure modes.

Containment Base Mat Penetrations

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 a reflux of water underneath the reactor vessel, this is highly unlikely. Melt through is therefore conservatively assigned a probability of 10^{-2} for LOCA sequences and is assigned the residual probability remaining after other failure mode probabilities are subtracted from 1.0 for TMLB'. These values are shown to be conservative by information to be presented later.

Assignment to Release Categories

In this study, each accident sequence - containment failure mode combination is assigned to a release category using the same general approach as used in WASH-1400.

For each combination of an 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 7 discrete release categories (for core melt events) in WASH-1400 for the purpose of evaluating consequences via air pathways. Each combination of an accident sequence and containment failure mode was assigned to the most appropriate release category along with the estimated probability of occurrence. The probabilities of all entries in a category were then summed to estimate the total probability of occurrence of a release to the environment for a particular release category.

Release Category 1

Sequences in which containment failure results from steam explosion and in which containment fission product removal systems are not functioning are placed in Category 1. The CRAC Code treats two types of Category 1 releases. Subcategory 1a is characteristic of events which occur with reduced containment pressure. Subcategory 1b is characteristic of events which occur with the containment at high pressure. In this study, for example, sequences S_1HF, S_2HF and TMLB' for Indian Point have been assigned to Category 1b. Sequence AHF was assigned to Category 1a on the basis that containment pressure will have been reduced by injection phase spray operation.

Release Category 2

Sequences in which containment failure results from overpressure (either γ or δ) and in which containment fission product removal systems are not functioning are placed in Category 2 in WASH-1400. Also placed in Category 2 was the V failure sequence (failure of interfacing check valves). The same categorization has been used in this study.

Release Category 3

Sequences in which containment failure results from steam explosion and in which containment fission product removal systems are functioning are placed in Category 3, as they were in WASH-1400. For Zion TMLB' is assigned to Category 3 rather than Category 1 because of the dedicated diesel-driven spray pump.

Release Category 4

Release Category 4 has not been utilized in this study. In WASH-1400, principal sequences in this category involved sequences with failure of containment isolation and fission product removal systems not functioning. Such sequences do not appear to be important contributors for this study.

Release Category 5

Sequences in which containment failure results from overpressure and in which containment fission product removal systems are functioning are placed in Category 5. This category assignment is similar to that employed for this same type of sequence in a recent study of an ice condenser PWR.

Release Category 6

Sequences in which containment failure results from melt-through and in which containment fission product removal systems are not functioning are placed in this category. This category assignment is the same as employed in WASH-1400.

Release Category 7

Combinations in which containment failure results from melt-through and in which containment fission product removal systems are functioning are placed in this category. This category assignment is the same as employed in WASH-1400.

Consequence Calculation

The risk characteristics (complementary cumulative distribution function) for the reactors at the Zion and Indian Point sites were preliminarily calculated from the probabilities of each Release Category by using 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 Indian Point reactors. Actual demographic data and site meteorology for the Zion and Indian Point sites were used. The calculations followed the Reactor Safety Study methodology very closely, including the evacuation model. A spectrum of evacuation speeds was incorporated; 30% chance of 0 mph, 40% chance of 1.2 mph, and 30% chance of 7 mph. A more sophisticated representation for the meteorology and evacuation was not feasible within the time available but will be part of the detailed study.

Risk Calculations

The following viewgraphs show the risk characteristics for the Zion and Indian Point plants for the following cases

(Slide 8)

- WASH-1400 curve as a reference point which is based on an average risk of the Surry reactor at a composite of 68 sites and which appears to constitute an acceptable level of risk.

(Slide 9)

- Assumption of Surry reactor at the Zion and Indian Point sites which appears to be the basis for NRC concern about these sites.

(Slide 10)

- Current Zion and Indian Point plants at their respective sites. Basically, a more realistic model for steam explosions has been incorporated and plant specific probabilistic analyses have been performed.

(Slide 11)

- Zion and Indian Point plants at their respective sites including the modifications for probability reduction such as check valve testing.

- Zion and Indian Point plants at their respective sites including a reference containment vent and filter design as well as probability reduction modifications

The last two characteristics are totally below the 10^{-9} per reactor-year probability and are, therefore, omitted. As you can see, earlier estimates of the risks from these plants are substantially higher than they should be. The risk scale used here is the same as that used by Dr. Denton in his recent presentation to the Commissioners. These are smoothed probability curves as used in

WASH-1400 and represent a conservative assessment. The important message presented here is that the Zion and Indian Point plants do not represent unusual risk. This is, in part, the result of the care taken in the initial design of these plants.

The important conclusion from the risk calculations, is that major contributors to risk for types of risk tabulated are Category 2 and Category 5 releases. The principal failure modes for these release categories are the containment overpressure failures. Risk via these failure modes are potentially amenable to reduction by containment features which reduce the probability of or eliminate these failure modes.

For overpressure due to non-condensible 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. Such system concepts will be discussed later in this presentation.

A lower level contribution to risk results from containment failures induced by steam explosions. As will be discussed subsequently, steam explosion probabilities employed in WASH-1400 are too high and have been reduced, thereby reducing risk associated with Category 1 and Category 3 releases.

The least contributor to risk to be discussed is the containment failure resulting from penetration or melt through of the containment base mat. As will be discussed in more detail later in this presentation, the assumption that the base mat fails is unrealistically conservative. The probabilities for this failure mode have been reduced for all but TMLB'.

It is worth noting that substantial conservatism exists in the work covered by the preceeding discussions. A few examples of this might help to put the results in perspective.

1) The sequence probabilities are, in many cases, higher than they should be because some double accounting of failure probabilities is certain to have occurred in our analyses.

2) In general, the sequence probabilities do not reflect any probability of successful corrective action given a failure. Industry experience shows that many failures are either momentary or of short duration and are amenable to very quick correction.

3) The probability of complete diesel-generator failure is approximately 2 orders of magnitude higher than Zion data and studies indicate it should be. The value employed was chosen to reflect conservatism in the face of undefined, postulated common mode failures.

4) The values employed for the steam explosion containment failure mode are judged to be very conservative, even though reduced from WASH-1400 values, since the steam explosion mode is expected to have a vanishingly small probability.

5) The criteria for core melt in the event of a LOCA were the same as those used in WASH-1400 in terms of fuel temperature and the finality of failure. This is extremely conservative.

In summary, while we have attempted to perform plant specific, realistic analyses, a substantial margin of conservatism exists in this work.

Our next topic of discussion is the long term reliability evaluation for these sites which will be presented by Mr. Jim Davis of the Power Authority of the State of New York.

2.2 DETAILED RELIABILITY EVALUATION

A detailed reliability evaluation program for both sites has been underway since December 20, 1979. This program consists of two parts: (a) the systematic development of event and fault trees of accident scenarios leading to core melt from which reliability improvements can be evaluated; and (b) Site specific consequence evaluations using an improved version of the CRAC program that takes into account population movement under general emergency conditions and changes in meteorology. The overall effort will allow us to confirm our identification of major contributions to risk.

The event trees will include the following:

1. Functional dependencies among systems and the initiating event.

2. Identification of components shared by several systems.

The fault trees will include the following:

1. Active component faults.

2. Unavailability of redundant loops due to test and maintenance.

3. Human errors during test and maintenance.

4. Other potential common cause failures as per OPSA and WASH-1400. These will receive attention only if a clear threat is identified.

Pickard, Lowe and Garrick are the consultants to the utilities for this program. Meetings were held in January to review site meteorological data and terrain features. Modeling has commenced for both sites for MET data in the improved CRAC program. With regard to systems reliabilities, initial fault trees have been developed for all four Units. Meetings at the site were held at Zion on January 30 and 31, and Indian Point on February 4 and 5, 1980 to review these fault logic diagrams with plant operations personnel. The system fault trees to be developed will include:

A. Electrical Systems
B. Condensate and Feedwater System
C. Auxiliary Feedwater System
D. Main Steam
E. Reactor Protection System
F. Reactor Coolant System
G. Emergency Core Cooling System
H. Containment Spray System
I. Containment Air Recirculation Cooling and Filtration System
J. Chemical and Volume Control System
K. Auxiliary Coolant System

In addition, plant specific failure data is being collected and evaluated to help quantify the fault trees. The program is expected to be complete in the summer of 1980.

We would now like to discuss some of the major technological considerations encountered in the course of the study. Our next speaker will be Dr. Robert Henry of Argonne National Laboratory.

3.0 TECHNOLOGY

3.1 STEAM EXPLOSIONS

(Slide 12a)

Steam explosions have occurred in various industrial accidents such as molten metal spills in foundaries. These explosions, which occur when high temperature, molten material comes in direct contact with water, have resulted in damage to the furnaces and the enclosure buildings. In hypothetical core meltdown accidents, intimate contact between molten core materials and water could occur (a) within the vessel and (b) in the reactor cavity if vessel failure is postulated.

(Slide 12b)

In such cases, the issues to be addressed are (1) under what conditions can steam explosions occur, and (2) can such events result in the failure of the containment building. These questions will be considered first for in-vessel interactions and then for the ex-vessel case.

A. In-Vessel

(Slides 12c & 12d)

In WASH-1400, steam explosions within the vessel were assigned a probability of 10^{-2} of resulting in containment failure. The mechanism for this failure was 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 upon the vessel head. This calculated impact resulted in failure of the vessel head and propelled it against the containment wall with sufficient energy to cause failure to this boundary. To achieve this, the calculations required intimate intermixing of essentially all the core material with water in a time interval of about 0.5 seconds while fragmenting to diameters less than 4 mm.

(Slide 12e)

In assessing the likelihood of such occurrences, we must first address the conditions under which steam explosions can occur and whether such a continuous overlying liquid layer can be formed and impact on the vessel head.

A1. Elevated System Pressures

(Slide 12f)

Experiments conducted with potentially explosive systems such as Freon-22 and mineral oil, Freon-22 and water, Freon-11 and mineral oil, water and sodium, water and corium, and water and molten sodium chloride, have shown that explosive vapor formation can be eliminated by elevated system pressures. This is the case for both the "free contacting" mode and externally triggered system. Analyses indicate that the pressure at which vapor explosions are precluded for a given system can be correlated on the basis of reduced pressure, i.e., the ratio of the system pressure to the thermodynamic critical pressure of the exploding liquid. Available experiments show that a reduced pressure of 0.05 is sufficient to eliminate explosions, which translates to a pressure of 150 psia for water. Consequently, when the primary system pressure is above this value, steam explosions will not occur when molten core material comes into intimate contact with water.

A2. Low System Pressures

(Slide 12g)

For some postulated accident sequences, such as a large break LOCA, the system pressures 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 plenum. In WASH-1400, it was envisioned that the core would enter into film boiling, which is indeed the case, fragment while in film boiling down to small particles, and then the explosion ensue. An important consideration is that the water in the lower plenum would be saturated and that the film boiling process would result in net vaporization, i.e., the vapor produced must escape upward through the pool.

(Slide 12h)

The velocity at which the vapor must flow through the pool determines the pool stability and this can be related to the fragmented particle size and the film boiling process. Such calculations show that when the molten core material reaches sizes 2 m in diameter, the resulting vapor flux from the film boiling alone exceeds the velocity at which bubbly flow can be sustained within the pool. This is the only flow regime in which a continuous overlying layer would have a chance of forming. In reality, as fragmentation progresses to diameters less than 2 m, the increasing rate of vapor removal would cause the pool to enter the "churn-turbulent" flow regime. In this flow regime significant "vapor channeling" would occur to accommodate the increased vapor production rate. Such a flow pattern does not allow the formation of a continuous overlying liquid layer and if a steam explosion did occur, it would quickly vent through the vapor channels as opposed to uniformly accelerating a liquid slug toward the head. Therefore, the fragmentation process itself is responsible for ensuring that a continuous overlying liquid layer, which is required to fail the containment via missile generation (as modeled in WASH-1400), does not exist.

(Slide 12i)

It is also worth noting 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 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 phenomenon to be observed would be a hollow splash up the center with a tall narrow water column rising later. This would definitely not lead to the slug type of impact used for the analyses in WASH-1400.

B. Ex-Vessel

(Slide 12j)

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 indeed occur. The water in this case could be subcooled, but there are no long continuous paths over which liquid slugs can be accelerated. The in-core instrument shaft provides a vent for the explosion which is directed toward the missile barrier. In addition, the shallow underwater explosion analogy discussed above is particularly relevant here. Therefore, the major issue to address in the case of ex-vessel steam explosions is the shock from the explosion itself.

Chemical explosives generate a shock wave that decays approximately as the square of the radius for strong finite waves that are well removed from the source. Typical maximum interaction pressures from steam explosions are 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. Even if it is doubled upon reflection, the pressure is well below the design level. Such considerations do not take credit for the building geometry (area expansion) which would also decrease the wave magnitude considerably. The end result of these considertions is that the shock waves themselves do not pose a threat to the containment integrity, and this is also the same conclusion arrived at in WASH-1400.

C. Conclusions

(Slide 12k)

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 (WASH-1400) is precluded for all reasonable fragmentation levels.

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.

(Slide 13a)

3.2 CORE COOLABILITY

(Slide 13b)

In the event that a core is badly damaged, the principal issue to be assessed is the permanent coolability of the damaged core either within the reactor vessel or in the reactor cavity if vessel failure were to occur. Making this assessment requires an evaluation of (1) the minimum water inventory that must be lost from the primary system before damage can be initiated, (2) the types, sizes, and distributions of core debris which would be permanently coolable, (3) the governing mechanisms for debris cooling both in the vessel, and (4) within the reactor cavity. These issues will be addressed sequentially.

A. Permanent Coolable Size

(Slide 13c)

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. Such a system could be nearly completely blocked at the bottom but would still have water accessible from the top. Permanent cooling of such a badly damaged core would be achieved when the characteristic conduction dimension of the debris is sufficiently small to keep the maximum fuel temperature below melting.

(Slide 13d)

Thermal conduction analysis of a sphere with uniform internal heat generation at 1% of full power and the outer surface in contact with water shows that particles of up to 10 in. diameter are permanently coolable. Such characteristic sizes would present a coolable configuration if water could be supplied to the core region.

B. In-Vessel Coolability

A damage configuration which is completely blocked at the bottom 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.

(Slide 13e)

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. Water available from these sources would pour into the upper core region.

(Slide 13f)

Two phenomena must be considered in evaluating the coolability of the 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.

Sustained contact of water with the top of the bed is determined by the ability of the core material to sustain film boiling, which calculations indicate could be the case only for a very short time, and insuring that water can be continually supplied to the top of the core. This latter condition is represented by a critical heat flux criterion on the upper surface, which describes the upward vapor flux necessary to prohibit the liquid from returning to the surface.

(Slide 13g)

At a pressure of 1000 psia, the critical heat flux criterion predicts that 50 MW can be removed from the upper surface of the pool before a limitation would occur. This is considerably greater than the 30 MW power level typical of the time at which major damage could begin for the most probable class of accidents. Consequently, upward vapor flux does not provide a limitation to core cooling.

(Slide 13h)

Once the liquid comes in contact with the top of the bed, it must also be able to penetrate down through to the bottom of the core. In order to ensure downward penetration of the liquid in the presence of upward flowing vapor, the pressure gradient imposed by the vapor moving through the damaged region cannot exceed the static head of the liquid.

(Slide 13i)

Maximum vapor velocities occur at the top of the core since all the energy is removed upward. Thus, the maximum pressure gradient induced by the vapor flow is also at the top. Equating the vapor flow rate corresponding to 1% of full power to the static head shows that the entire core mass would have to be fragmented to millimeter size particles before water transport within the bed would become limited. Such sizes are orders of magnitude below the permanently coolable size and far smaller than the particle size measured in large scale (kilogram quantity) tests for UO_2 and water. It should also be noted that the maximum vapor pressure gradient is at the top of the pool, so that once the liquid penetrates this level, the rate of penetration increases, i.e., the liquid can move more easily at lower elevations because the vapor velocity is less. Thus, the available experimental and analytical evidence indicates that a badly damaged core can be cooled within the vessel if water can be supplied from the top.

(Slide 13j)

If water is available to the core, a heat removal path must also be established and this can indeed be accomplished with a small fraction of the heat transfer area in one steam generator. With the bottom primary coolant entry point in the Zion and Indian Point steam generators, steam can flow up into the unit, condense, and the liquid can drain out and return to the core. This reflux heat removal path has a key safety aspect in that this energy transport cycle can be effected in the presence of non-condensible gases.

(Slide 13k)

In summary, considerations of a badly damaged core, which is assumed to be completely blocked at the inlet, show that the core is indeed coolable if water can be supplied to the upper surface. The particular plants in question have hot leg injection, which allows water to be added directly to the upper internals region. In addition, a reflux heat removal path can be established which is operable in the presence of non-condensible gases such as hydrogen.

C. Ex-Vessel Cooling

In the complete absence of primary cooling water, the core material would overheat due to decay power and failure of the reactor vessel could not be ruled out. If such a failure occurred, the core material would fall into the reactor cavity below the vessel, and its coolability in this region is determined by the availability of water to the reactor cavity.

(Slide 13b)

Analyses of the recent accident in TMI-2 show that approximately one-half of the primary cooling inventory was lost before core damage was initiated. This value is design specific because of the particular location of the steam generators, and applying similar analyses to the Zion and Indian Point designs shows that 65% of the inventory must be lost before damage would occur. The additional inventory decrement is required for these plants because of the elevated steam generators. Water lost from the primary system would be available within the containment building in the event that the core material would be released from the vessel. The specific designs of Zion and Indian Point are somewhat different, but each system could be arranged to ensure that tens of thousands of gallons would be in the reactor cavity before core damage could occur. Such considerations are important in establishing the heat transport path since the fan coolers, which condense the steam from the environment and extract the heat from the building, return their condensate to the containment floor. Consequently, if the decay heat can be removed from the core debris by vaporization of water, an energy transport path can be established with the water that had to be lost from the primary system in order to cause the core damage.

The phenomena controlling debris bed coolability in the reactor cavity are the same as those considered for in-vessel cooling with two exceptions. First is the possibility of a steam explosion, which will not pose a threat to the containment, but will provide rapid cooling and dispersal of the core material, i.e., such an event would only act to improve coolability by increasing the surface to volume ratio of the fuel. The second feature is attack of the reactor cavity concrete by the core material if complete quenching and cooling is not established immediately. The fact that the cavity would have contained tens of thousands of gallons of primary coolant ensures that the concrete is saturated with water. Attack of the concrete implies that this water will be vaporized and experiments have shown that this results in very high superficial velocities, i.e., velocities sufficiently high to boil-up the molten debris pool and ensure a "churn-turbulent" flow regime. This increases the energy transfer with the overlying layer of water and promotes freezing of the core material, recalling that particles of about 10 in. in diameter represent a permanently coolable state.

(Slide 131)

Once the solid bed is achieved, the bed coolability is determined by critical heat flux and water penetration considerations. These analyses show that liquid penetration is not limiting and neither is the critical heat flux criterion, i.e., the bed is permanently coolable in the reactor cavity.

(Slide 13m)

In summary, the cooling water that must be lost from the primary system to result in core damage is sufficient to establish a heat transport path to remove the decay power. Considerations of the ex-vessel debris bed show the bed will indeed be coolable. Therefore, a core ladle will not be needed.

Our next topic of discussion, containment transient analysis, will be presented by Dr. Richard Slember.

3.3 TRANSIENT ANALYSIS

Transients selected for evaluation in this study were LOCA's followed by ECCS failure, and the station blackout followed by failure of the turbine driven auxiliary feedwater train. The Battelle Columbus Laboratories were contacted to obtain core melt calculations typical of those performed for WASH-1400. The computer code used to generate these calculations was an early release version of an NRC development code called the MARCH code, which models core melt and slump, reactor vessel heatup and failure, and concrete melting. MARCH code calculations were obtained for a four loop Westinghouse PWR for the AD, S_2D, and TMLB' sequences. The mass and energy releases to the containment were obtained for each of these sequences and used as input into the Westinghouse containment code called COCO for scoping calculations of the containment response.

Each plant with its individual heat sinks, containment volume, and containment safeguards systems were modeled in the COCO code. Using the containment mass and energy input information from MARCH code calculations, several containment parameter studies were performed. Additional models including containment vent models and hydrogen burn models were also added to the COCO code such that possible hydrogen and carbon monoxide burn and containment venting could be modeled and studied in a parametric manner. These parametric studies indicated that the possibility of combustible gas burning was one of the key parameters which affected containment response.

The amounts and concentrations of H_2, CO, CO_2, N_2, O_2 and steam evolved or present during the different transients were examined. Upon examination, two models for burning the hydrogen and CO were formulated, (a) a continuous burn of the gases as they were released, and (b) a build up of the gases to a concentration level at which a "flame temperature criteria" is exceeded which indicated when burning could occur in the containment.

In the continuous burn case, the hydrogen was burned as it was released to the containment. The resultant burn energy was factored into the containment mass and energy analysis to evaluate the containment atmosphere pressure and temperature.

In order to have a preliminary evaluation of the effect of hydrogen burning at different times in the containment transient a "flame temperature criteria" was developed by Dr.'s Bernard Lewis and Bela Kavlowitz of Combustion and Explosives Research, Inc. In using this criteria the containment atmospheric temperature and composition are considered at each time step and the flame temperature for burning this mixture at constant pressure is calculated. This calculated mixture flame temperature is compared with a "flame temperature criteria" of 710°C. 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 propogation commences. The 710°C criteria accounts for a lower hydrogen concentration being flammable at elevated mixture temperatures, and conversely requires a higher concentration for flammability when diluents such as steam are present in the mixture. As before, the pressure and temperature from the burn are incorporated in the computer code to obtain the pressure transient for the different scenarios under the criteria.

It must be emphasized that these transients are conservative in that 100% clad-water reaction is assumed along with a 100% H_2 burn. A great deal of additional work is necessary in the area of hydrogen evolution and combustion before any firm conclusions are reached. While the MARCH code insures that proper integrated mass and energy is added to the containment consistent with its model assumptions, the rates of addition also depend on the assumptions used for core melt, core slump and failure, and the reactor vessel head failure. In calculating the containment response, the rates of the associated mass and energy additions are most sensitive parameters. If these mass and energy additions are over a sufficiently long time, the containment heat sinks can be quite effective. If, however, the mass and energy additions are quite abrupt, as in the case of some of the non-mechanistic MARCH models, the containment heat sinks are ineffective. Hence, improved models for the MARCH code on rates of mass and energy additions into the containment are needed if an accurate containment response is desired.

Our Preliminary results show that:

1. LOCA melt sequences with minimum containment safeguards and no H_2 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 on satisfying the "flame temperature criteria" can exceed containment capability in some cases.

4. For TMLB' sequences and either continuous H_2 burn or H_2 burn when the "flame temperature criteria" is exceeded, the potential exists to 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 as a guide to examine containment venting. While the Battelle study uses one 8 to 10-inch vent, the calculations in our study have 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 did indicate that if very rapid and large energy addition occurred in the containment, this size vent could not keep the pressure below the containment capability. However, in other cases, the addition of a vent was seen to be effective.

In summary, it appears that a vent system based for reference on two 12-inch vents may be useful in mitigating the consequence of containment transients. It is also important to recognize that these are preliminary results and are subject to confirmation as investigations continue. Lastly, it is very important to note that we are looking at very substantial systems. The two 12-inch vent lines used as reference for this study are considerably larger than the 4" vent mentioned by the NRC in the initial meeting on the subject and the downstream portions of any such system are correspondingly bigger as well.

4.0 RISK REDUCTION

Our next topic of discussion is those features, installed, proposed, and under study, which help to mitigate public risk. The first area of interest in this regard is the collection of extra plant features installed in these units during their initial construction.

4.1 ORIGINAL PLANT DESIGN FEATURES

4.1.1 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.

4.1.2 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.

4.1.3 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.

4.1.4 Extra 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.

4.1.5 Post LOCA Hydrogen Control

Each unit has both recombiner and post-LOCA containment purge capability. The recombiner capability was added to provide added conservatism.

4.1.6 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.

4.1.7 Added Containment Radioactivity Removal

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.

4.1.8 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 restore to its proper position upon ESF actuation. This has been applied to critical safety systems valves.

4.1.9 Internal Core Cooling Recirculating System on I.P.

Two recirculating pumps located inside the containment provide for sump recirculation into the Hot leg. These are in addition to the RHR system.

4.1.10 Additional Diesels

Three diesel generators are available for each I.P. unit. Two generators are adequate to meet engineered safeguards load.

Zion has two diesel generators per unit with a shared diesel generator which swings to the demand unit.

4.1.11 Gas Turbine Generators at I.P.

One gas turbine located onsite and two others located in close proximity provide diverse energy sources. Interconnections exist for supplying power from these gas turbine units to IP-2 and IP-3.

4.1.12 Additional Service Water Pumps

Three 100% capacity service water pumps are available per unit. In addition, at Indian Point, three additional pumps provide BOP cooling requirements and facility exists to use these pumps for safety systems.

Our next topic of discussion is those actions under consideration as means to reduce the probability of an accident thereby reducing risk.

4.2 PROBABILITY REDUCTION

In general, improvements in reliability can be considered a function of system hardware, operator actions, and testing and maintenance programs. The Westinghouse/Utility Short-term-WASH-1400 studies have identified significant contributors to the accident sequence probabilities. These items were included in our benefit assessment and are summarized as follows:

1. HF's - Recirculation/containment spray on recirculation failure, may be improved by

    a. Testing recirculation containment sump suction valve control circuits on monthly basis (affects Zion only).

    b. Reviewing procedures for transferring to recirculation mode - Offsite Review.

    c. Operator training - Transfer to recirculation mode(has been accomplished).

2. V - Interfacing Check Valve Failure may be improved by: following RHR system operation, prior to returning to power operation, test the two check valves nearest the reactor coolant system in each of the RHR cold leg injection lines.

The improvement in HF sequences from testing and training is at least an order of magnitude. The testing of the check valves in the V sequence changes the probability from leak-rupture to rupture-rupture with at least two orders of magnitude improvement.

Other selection of "fixes" to reduce probability of a severe accident will be evaluated when the more detailed probabilistic risk assessment study is completed.

In the meantime, we will be spending considerable time considering directions for conceptual designs for mitigating features.

4.3 DIRECTIONS OF CONCEPTUAL DESIGN

Based on the analytical studies discussed earlier which were used to identify and scope areas where design features could be employed to mitigate the effects of various degraded core scenarios, four major areas of design activity were defined. These areas are defined as those which deal with the phenomena of steam explosion, combustible gases, concrete erosion and containment overpressure.

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 preliminarily 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 or adjusting 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 earlier. A concept to assure water coverage of the concrete is under consideration in order to assure (based on earlier discussions) that quench occurs. Typical concept is shown in this viewgraph. The space available below the reactor vessel of the IP

(Slide 14a)

and Zion plants will readily accommodate a water supply. Heat removal from the containment via refluxing with water is being evaluated through the use of fan coolers and/or direct feed and bleed using an external water source to dissipate heat by conversion to steam at a nominal rate of 100,000,000 BTU's/hr for four hours. This assumes that the total water volume beneath the core is 7,000 cubic feet.

The pressure transients caused by the release of energy from phenomena associated with the core debris quench results in peak pressures which may exceed the design pressure of the containment. Concepts for relieving this pressure are under investigation together with trigger mechanisms which actuate in a controlled manner. A discussion of two candidates will be presented as examples. It should be noted that all concepts, layouts and sizes are preliminary. It should also be noted that modifications to these concepts such as recycling non-condensibles will be given consideration.

Sparger Condenser and Water Scrubber

The purpose of this system is to provide a means to vent the containment building 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. A typical, but very preliminary, conceptual design is shown on this viewgraph.

(Slide 14b)

Gases enter the 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 to approximately 18,000 CFM. 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 at a peak rate of 100 million BTU per hour. The sparger tank has the capacity to absorb a total of 2 billion BTU. Water saturated gas is vented from the sparger tank to atmosphere at temperatures ranging from $50^{O}F$ to $180^{O}F$. Condensate from the process gases remains in the sparger tank.

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. This system has no provisions for reducing noble gas and organic iodine releases to the atmosphere.

A tank, as shown on this slide, with a usable volume of 1,875,000 gallons is required. The tank would be approximtely 80 feet in diameter by 50 feet high. Other dimensional combinations with the same capacity are also being considered.

Submerged Gravel Scrubber

The purpose of this system is to provide a means to vent the containment building 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 gravel scrubber submerged in water. Again, a typical but very preliminary conceptual design is shown.

(Slide 14c)

Gases enter the 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 to approximately 18,000 CFM. When the containment pressure drops, the containment isolation valves will close.

The vent line discharges into a manifold which is located in the bottom of the submerged gravel scrubber. Gases exit the manifold and pass up through the water-gravel mixture where heat is removed at a peak rate of 100 million BTU per hour The submerged gravel bed has the capacity to absorb a total of 2 billion BTU. Water saturated gas is vented from the scrubber tank to atmosphere at temperatures ranging from $50^{o}F$ to $180^{o}F$. Drain lines in the bottom of the tank provide for condensate removal after the accident. The gravel scrubber is estimated, on a preliminary basis, to provide a decontamination factor of 100 for any particulates in the process gas and a decontamination factor of 10 for molecular iodine. This system has no provisions for reducing noble gas and organic iodine releases to the atmosphere.

The size of the submerged gravel scrubber as shown on this slide is 130 feet square by 25 feet deep. This is equivalent to nearly 10 acre feet of filter media. To fill the gravel bed, almost 25,000 tons of gravel are required.

Other Potential Features

In addition to the concepts discussed above, other features are under consideration for Zion and Indian Point. These are:

1. Means to ensure water flow to the area beneath the reactor vessel at Zion and Indian Point.

2. Means to achieve AC independence for the turbine driven auxiliary feedwater pump at Zion.

3. Means to add a diesel driven containment spray pump at Indian Point.

These features may offer some degree of risk reduction and will be studied in more detail as our work progresses.

It is necessary to pause for a few moments and put the entire concept of risk in perspective. Earlier, we showed you graphs of the relative total category probabilities and risks for Zion and Indian Point. We also discussed means of probability reduction.

(Slide 15)

The risk remaining after the probability reduction is dominated by overpressure failure of the containment in release categories 2 & 5. In turn, the major contributers in this regard are the LOCA sequences. The risk due to TMLB' is a second order effect for both sites. We have evaluated the transient resulting from TMLB' and have determined that it is limiting but that, due to the nature of TMLB', it also requires two major design constraints:

1. The vent has to be designed to open below containment design pressure.
2. The filtered vent system has to be essentially a passive system.

The first constraint poses the possibility of vent actuation when such actuation is not desired thereby actually increasing risk in the event of a lessor transient of higher probability. For example, it is possible to envision the vent opening for a small LOCA and releasing activity to the environment when such releases are neither necessary nor desirable. The second constraint makes design solutions for the first constraint very difficult, if they are possible at all, due to the loss of AC power and restricts vent system design to vent, filter and release or vent filter & third containment concepts.

Based on these considerations and concerns we may well, in our preliminary design work, conclude that our efforts are appropriately directed toward system concepts which address the LOCA or dominant risk sequences to the exclusion of TMLB'.

(Slide 16)

The viewgraph you now see shows the mean probabilities for Zion and Indian Point Stations. The base probabilities shown illustrate the total category probabilities prior to probability reduction and prior to a filtered, vented containment system. The hatched portion of the graph illustrates the probability remaining after these improvements are factored in. Where benefits differ by more than a very small amount, a separate indication is shown. The effect of excluding or including TMLB' in the vent system design is so small that it cannot be graphically depicted.

Our next topic of discussion is those future actions planned by the utilities regarding this work. Mr. William Bennett will make this presentation.

5.0 FUTURE ACTIONS

5.1 SCHEDULE FOR RELIABILITY ANALYSIS

(Slide 17)

The detailed reliability evaluation program should be complete during the summer. All event and fault trees will be initially completed by spring but refinements are likely to be required after detailed review. The consequences model will also go through a thorough review. The scope of this work has also been expanded to include Failure Mode Effects Analysis for active components on the reactor coolant pressure boundary, and minor departures from operating, maintenance and emergency procedures.

5.2 DEVELOP CONCEPTUAL DESIGN

We would now like to discuss those future utility actions which will take place once certain NRC actions, our last topic, have been completed.

Our next action in the mitigation program is further review of containment transients including combustible gases. This will be followed by a review with you of your actions and our findings to achieve technical agreement in these areas. We then plan to prioritize our alternatives and begin preliminary design work as required. Such design work would include P&ID development, general arrangement drawings, design criteria, and initial physical piping layouts. System sizing and initial cost estimates will also be performed.

These tools will serve as vital inputs for our final evaluation process leading to a decision regarding the value impact of selected mitigating features and to the selection, if required, of a preferred design concept (or concepts) for each site. We expect that the characteristics of the individual sites and units may make themselves more apparent in this phase of the effort.

As you know, we started this work at your request, with relatively undefined objectives. We were to see what could be done to mitigate the effects of a core melt no matter how improbable. We understood that your interest stemmed from a feeling that the two sites in question had to present a greater risk, due to population density, than other sites. We further understood that, insofar as a goal was defined, it was to reduce this risk to some reasonable degree.

We feel we have been fully responsive to your request. The program we have set forth today has involved a tremendous effort on our part and has resulted in as thorough a study as time allowed. The results have been enlightening. Zion and Indian Point offer far less risk to the public than original estimates might indicate. Also, the mitigating feature of current regulatory interest, a filtered, vented containment, has proved to be a much more substantial undertaking than originally contemplated. In our original meeting on December 5, 1979, Mr. Brian Grimes remarked that the staff felt the investment of a couple of million dollars would be worthwhile if we could extend the allowable evacuation time by around 10 hours as a result. Even rough scoping estimates would indicate that we are well beyond that point for the most simple concepts.

As a result of these considerations, we feel it is appropriate to complete the design development just discussed only after a more refined benefit evaluation has been completed. We do not feel that it is appropriate to commit to any plant changes until these efforts have been combined into a realistic assessment of the risk benefit to be derived compared to a defined safety goal which relates to other risks faced by society. Furthermore, a value-impact analysis is also required.

Thus, an essential part of any such effort is the establishment of a uniform safety goal for these plants which is expressed quantitatively in unambiguous terms.

We believe such steps are absolutely vital to an orderly, well conceived reactor safety program. They are also essential to any sound engineering process.

(Slide 18)

This slide illustrates the process described above. You can see that all the steps mentioned are included. The time frame for this effort is as short as we feel is reasonable and presumes an active, aggressive program on the part of all organizations involved.

We would now like to discuss those areas where the efforts of the NRC are seen to be vital to the continuation of this program.

6.0 NEEDED NRC ACTIONS

Since the directions we intend to pursue in this continuing program are based on the analytical approach chosen, the concepts derived from that approach, and the technology developed in our studies and in other work; it is necessary to come to agreement on these matters.

We have, today, outlined the entire program and demonstrated a strong and rapid response to the concerns first articulated by Mr. Eisenhut on December 5, 1979, which included a 60 day limitation; and to the concerns further developed in meetings on December 20, 1979, and January 11, 1980.

We now need both guidance and agreement on the following areas in order to develop the conceptual designs of the systems which could mitigate or delay the possible release of radioactivity following a postulated extremely severe accident:

1. The acceptance by the NRC of the analytical approach described earlier including the use of a WASH-1400 short term study and steam explosions.

2. In the absence of the results of the probabilistic risk assessment study now in progress, the acceptance by the NRC of the conceptual design approach we indicated today is essential. This relates to refining our transient analysis and evaluating the effects of filtered containment venting. Acceptance of our assessments regarding lack of penetration of the containment base mat is also essential.

3. The definition of a uniform, quantitative, unambiguous safety goal for the Zion and Indian Point plants.

This NRC support is essential if we are to achieve the common technological base which is necessary to select design features which will mitigate consequences or reduce the probability of a severe accident. We are asking the NRC for the technical expertise of its staff and that of its contractors. This input will then be factored into our analysis and decision making process so that we all may achieve a rational objective in reactor safety.

NEAR IN SITE STUDY FOR INDIAN POINT & ZION

Mitigation

1. Selection of Dominant Sequences and Transient Development

2. Selection of Dominant Transient Conditions, Development of Site Consequence Model

3. Identification & Evaluation of Feasible Mitigation Features

4. Evaluation of Effects of Feasible Features on Risk
    (Input From ANL on Steam Explosion & Containment Base Mat Penetration) (Input From Dr. Lewis On Gas Combustion)

5. Ranking of Feasible Features and Selection of Those For Further Investigation

Probability Reduction

1. Short Term (Phase "A") Review of Accident Sequences & Probabilities

2. Review of Phase "A" Work to Identify Potential Means to Reduce Probability

3. Selection of Measure to Reduce Probability - Near Term

4. Initiation of Long Term (Phase "B") Risk Assessment Program

5. Completion of Phase "B" (Summer 1980) and Review of Results For Additional Means to Reduce Probability

(Slide 1)

PRESENTATION OUTLINE

Analytical Approach

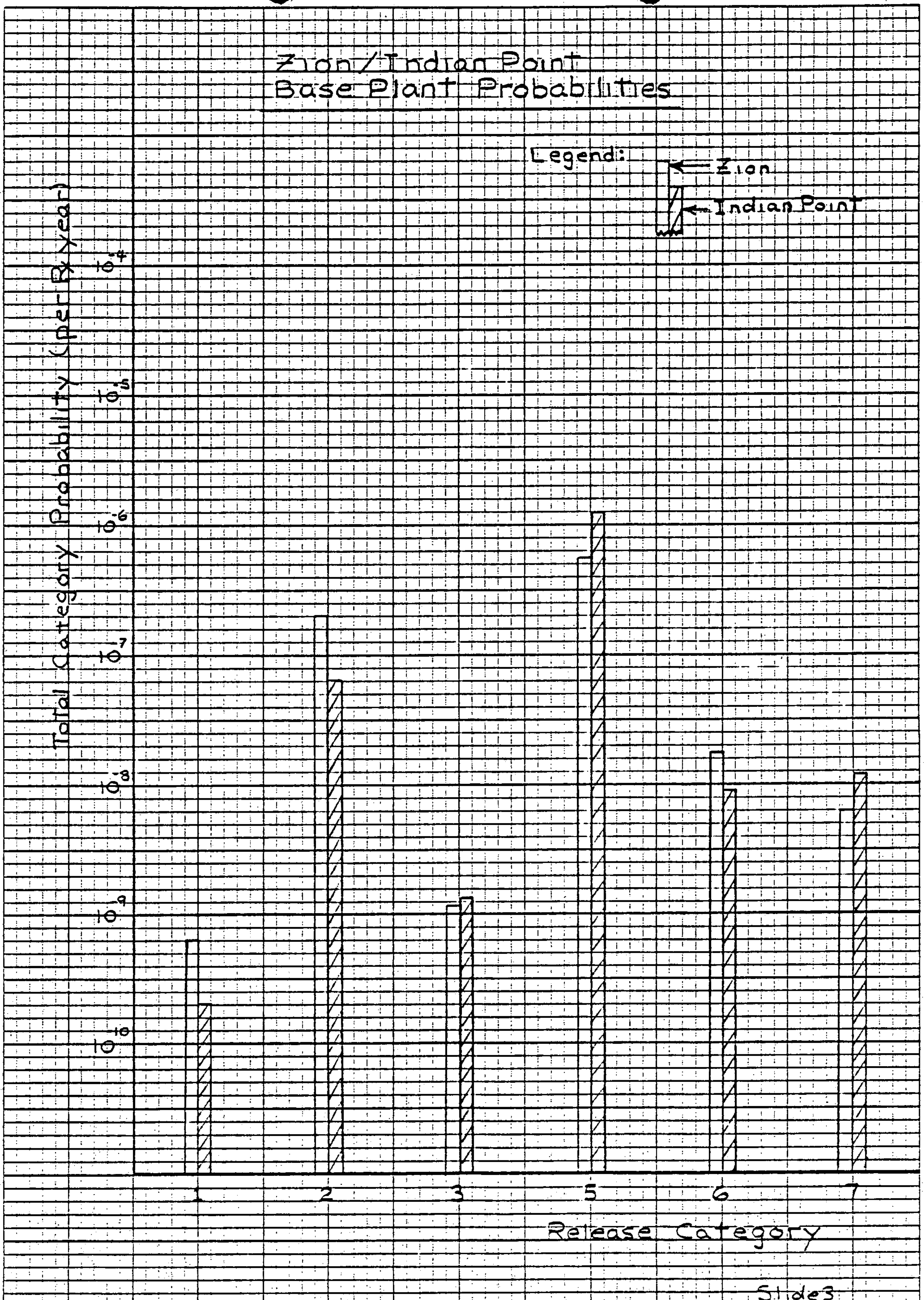
Technological Conclusions

Mitigation Concepts

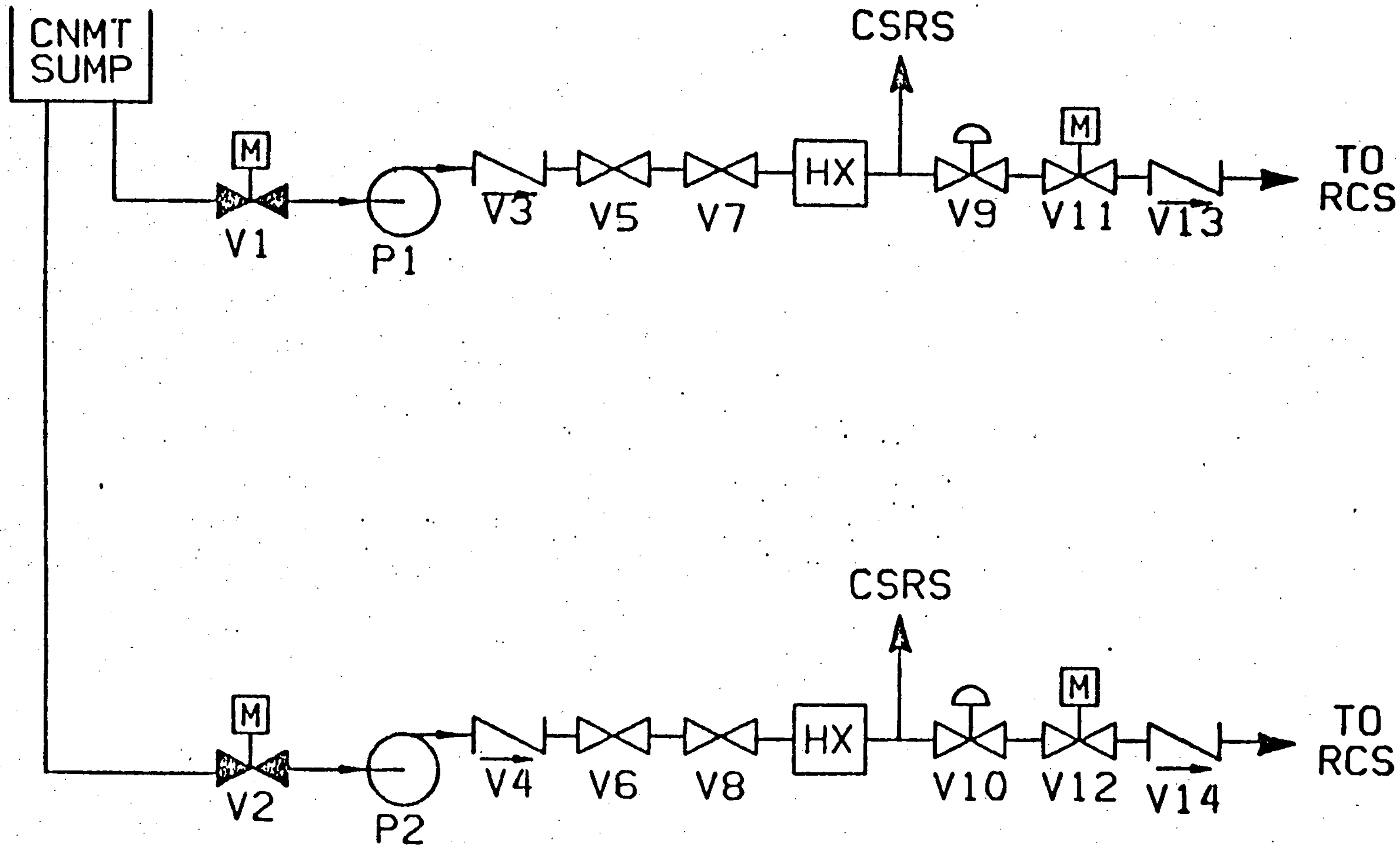
Proposed Future Actions

Needed NRC Input

(Slide 2)

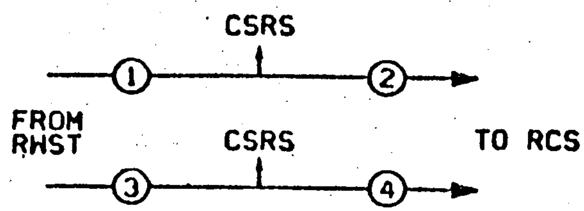


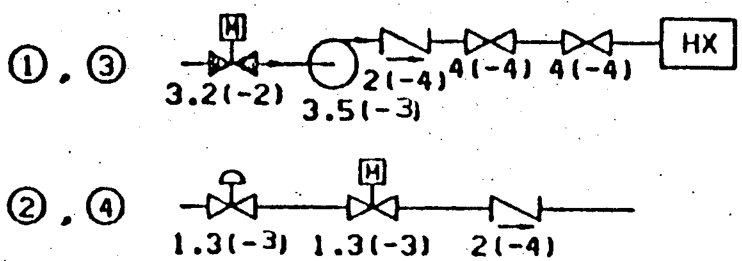
Zion/Indian Point
Base Plant Probabilities
Legend:
Zion
Indian Point
Total Category Probability (per Rx year)
10^{-4}
10^{-5}
10^{-6}
10^{-7}
10^{-8}
10^{-9}
10^{-10}
1
2
3
5
6
7
Release Category
Slide 3



LPRS FLOW DIAGRAM

Slide 4





P 1 = P 3 = 3.6×10^{-2}

P 2 = P 4 = 2.8×10^{-3}

COMBINATION OF BLOCKS	PROBABILITY FORMULATION	RESULT: CSRS ALSO FAILS	RESULT: CSRS DOES NOT FAIL
1 AND 3	3.6(-2)X3.6(-2)	1.3×10^{-3}	
1 AND 4	3.6(-2)X2.8(-3)		1×10^{-4}
2 AND 3	2.8(-3)X3.6(-2)		1×10^{-4}
2 AND 4	2.8(-3)X2.8(-3)		7.8×10^{-6}
	P(LPRS AND CSRS) =	1.3×10^{-3}	
	P(LPRS) =		2×10^{-4}

LPRS BLOCK DIAGRAM & FAILURE COMBINATIONS

Slide 5

Failure of LPRS (Core Only) =

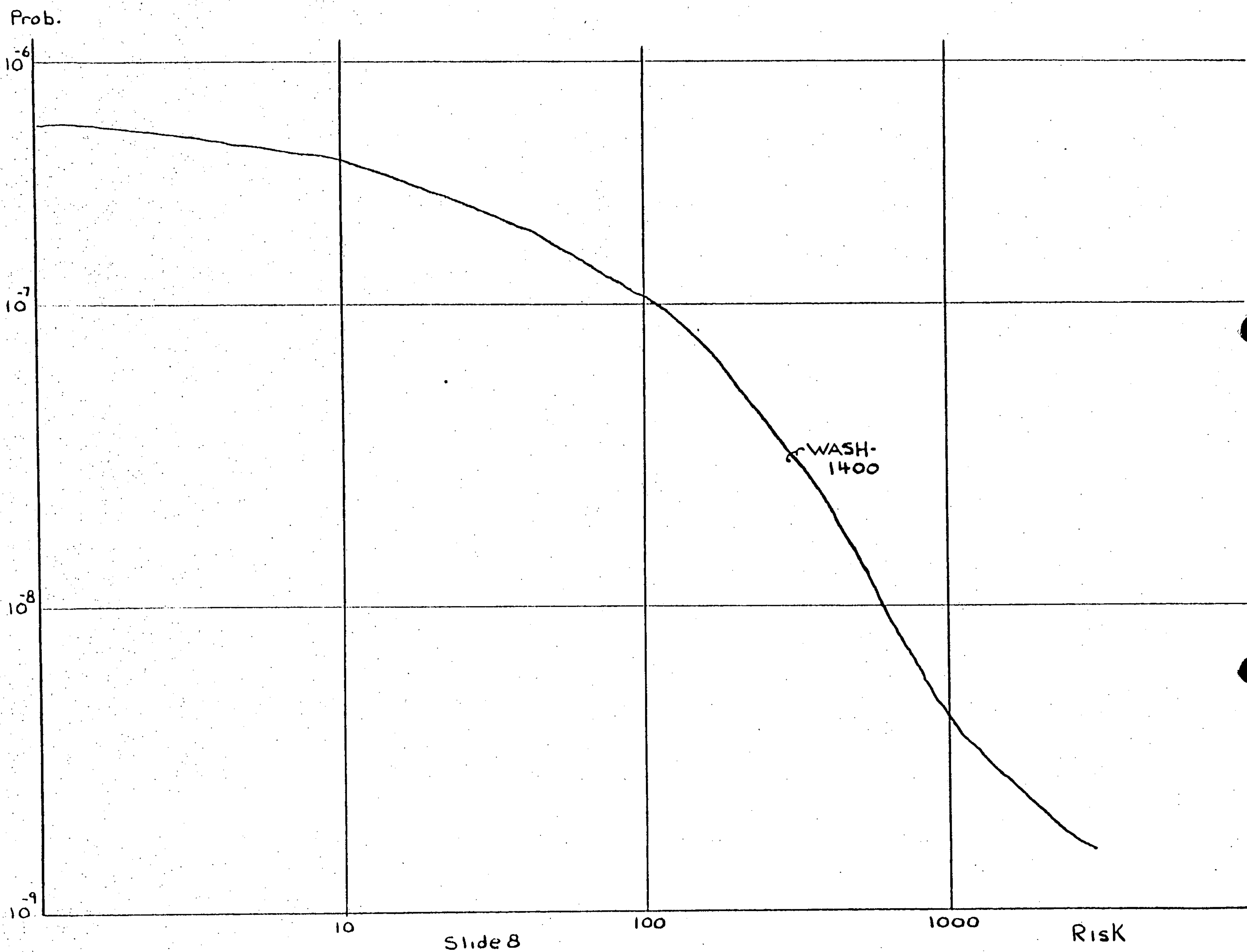
	p(H)	2×10^{-4}
	p(A)	1×10^{-4}
	p(AH)	2×10^{-8}

Failure of LPRS and CSRS		1.3×10^{-3}
p(Operator error during		3×10^{-3}
shift from INJECTION to RECIRC)	p(HF)	4.3×10^{-3}
	p(A)	1×10^{-4}
	p(AHF)	4.3×10^{-7}

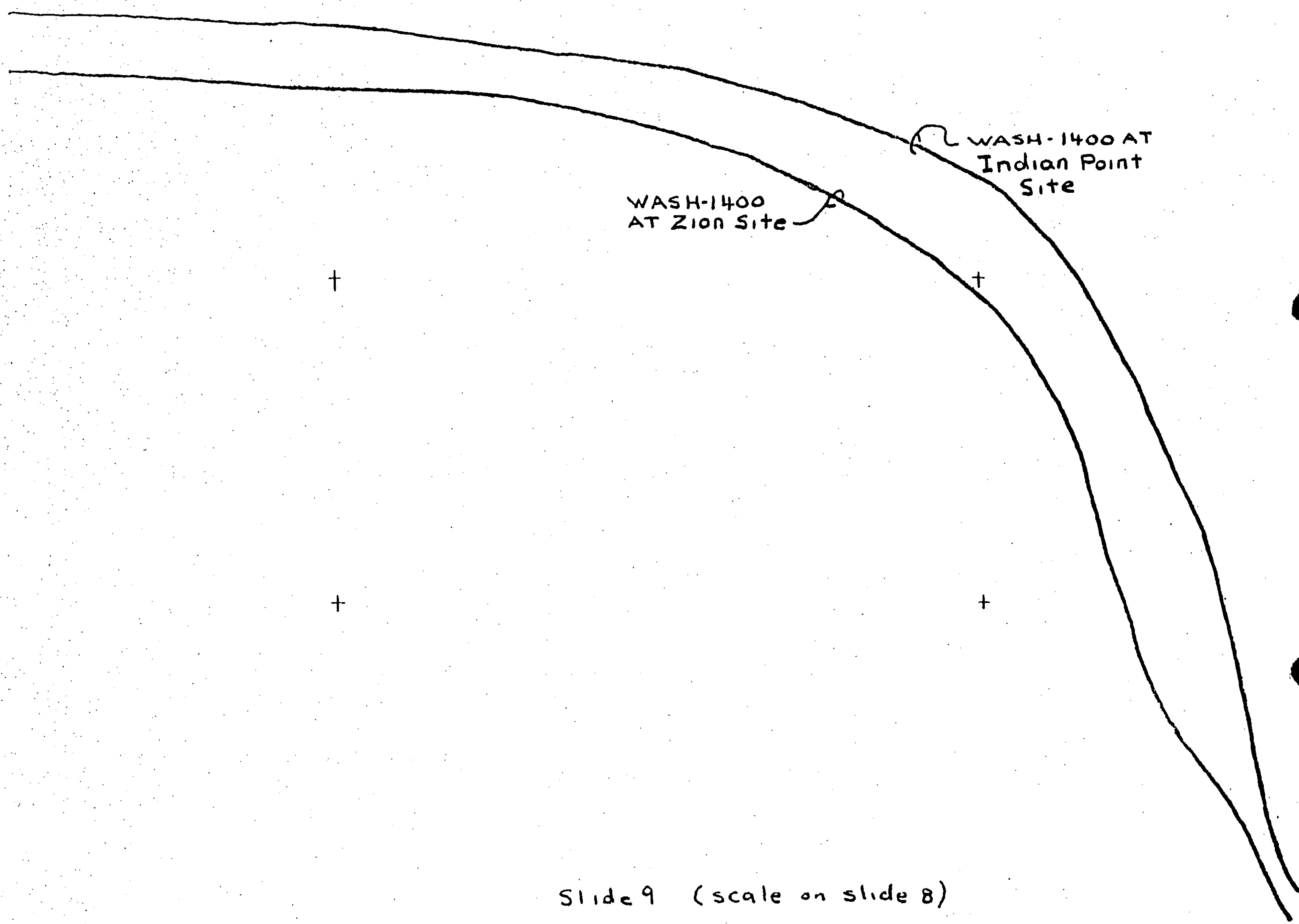
(Slide 6)

Containment Failure Modes	Symbol
Steam explosion in reactor vessel	α
Isolation leakage	β
Overpressure from hydrogen burning	γ
Overpressure - generally (less hydrogen)	δ
Containment melt-through	ϵ

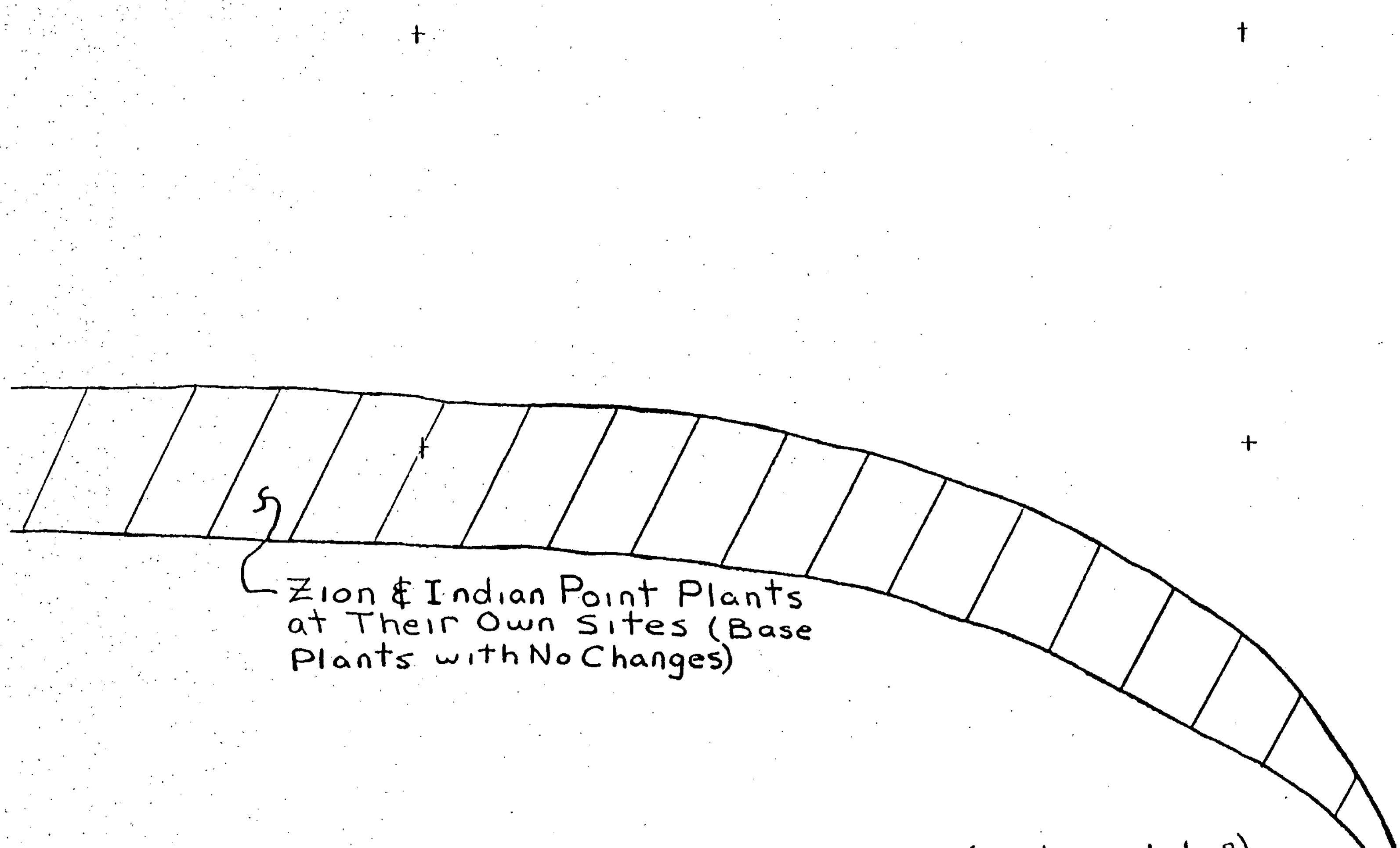
(Slide 7)



Slide 8



Slide 9 (scale on slide 8)



Slide 10 (scale on slide 8)

Z ion with probability reduction (2.9×10^{-11})(I.P. similar)
Z ion & I.P. with probability reduction & vent (3 to 4×10^{-11})

Slide 11
(scale on slide 8)

STEAM EXPLOSIONS

ROBERT E. HENRY

ARGONNE NATIONAL LABORATORY

(Slide 12a)

STEAM EXPLOSIONS

- ISSUES ADDRESSED -

1. UNDER WHAT CONDITIONS CAN STEAM EXPLOSIONS OCCUR?
2. CAN SUCH EVENTS RESULT IN FAILURE OF THE CONTAINMENT BUILDING?

STEAM EXPLOSION-CONTAINMENT FAILURE MECHANISM IN WASH-1400

STEAM EXPLOSION WITHIN THE VESSEL ACCELERATED A CONTINUOUS OVERLYING LIQUID LAYER IN A PISTON-LIKE MANNER UNTIL IT IMPACTED UPON THE VESSEL HEAD. THIS IMPACT FAILED THE VESSEL HEAD AND PROPELLED IT UPWARD WITH SUFFICIENT ENERGY TO FAIL THE CONTAINMENT BOUNDARY UPON IMPACT.

<u>KEY QUESTION</u> : CAN SUCH A CONTINUOUS OVERLYING LIQUID LAYER EXIST?

PROBABILITIES CONSIDERED

	DESCRIPTION	PROBABILITY
A	PROBABILITY OF A LARGE MASS OF MOLTEN UO_2 (20% OF THE CORE) CONTACTING A SIMILAR MASS OF WATER DURING THE MELTDOWN PROCESS	1
B	PROBABILITY OF A DISPERSAL EVENT OCCURRING WHEN MOLTEN UO_2 FALLS INTO WATER (EFFECTIVE PARTICLE SIZE 4000)	0.1
C	FRACTION OF STEAM EXPLOSIONS LEADING TO CONTAINMENT FAILURE	$\frac{0.1}{10^{-2}}$

(Slide 12d)

IN-VESSEL STEAM EXPLOSIONS

I. ELEVATED SYSTEM PRESSURES
(GREATER THAN ~ 150 PSIA)

II. LOW SYSTEM PRESSURES
(LESS THAN ~ 150 PSIA)

(Slide 12e)

Experiments Demonstrating A High Pressure Cutoff

Laboratory	Materials Used	Explosive Pressures Measured (MPa)	System Pressure Required to Eliminate Explosion (MPa)	Reduced Pressure
Argonne	Freon-22 and Mineral Oil	2.5	0.2	0.04
Argonne	Sodium and Water	2.0	1.0	0.05
Ispra	Sodium chloride and Water	6.0	1.0	0.05
Sandia	Corium and Water	1.5	0.75	0.04

(Slide 12f)

IN-VESSEL STEAM EXPLOSIONS

II. Low System Pressure

A. Probability Of A Steam Explosion Should Be Assumed To Be Unity.

B. Can A Continuous Overlying Liquid Layer Exist?

WATER SWELL DURING FILM BOILING AND FRAGMENTATION

VAPOR FLOW RATE $\dot{M}_V = \frac{h \Delta T}{h_{fg}} A_S$

SURFACE AREA $A_S = N\, 4\pi r^2$

NUMBER OF PARTICLES $N = M_F/(\rho_F 4/3 \pi r^3)$

$$\dot{M}_V = \frac{h \Delta T}{h_{fg}} \frac{3M_F}{\rho_F r}$$

$$U_\infty = \frac{3M_F h \Delta T}{\rho_V A_l h_{fg} \rho_F r}$$

$r = 1$M

$U_\infty = 5.7$ CM/SEC

(EXCEEDS BUBBLY FLOW LIMIT)

IN-VESSEL STEAM EXPLOSIONS

I. ELEVATED SYSTEM PRESSURES (> 150 PSIA)

- STEAM EXPLOSIONS WILL NOT OCCUR.

II. LOW SYSTEM PRESSURES (<150 PSIA)

- PROBABILITY OF A STEAM EXPLOSION SHOULD BE ASSUMED TO BE UNITY.
- A CONTINUOUS OVERLYING LIQUID LAYER CANNOT BE FORMED FOR ANY REASONABLE LEVELS OF FRAGMENTATION.
- A STEAM EXPLOSION IN THIS ENVIRONMENT WOULD RESEMBLE A SHALLOW UNDERWATER EXPLOSION, I.E., EXPANDING STEAM BUBBLE WOULD QUICKLY BREAK THROUGH A LIQUID LAYER.
- PROBABILITY OF VESSEL FAILURE BY A STEAM EXPLOSION IS INSIGNIFICANT - CONSEQUENTLY, THE PROBABILITY OF CONTAINMENT FAILURE IS ALSO INSIGNIFICANT.

EX-VESSEL STEAM EXPLOSIONS

- LOW PRESSURE - THE PROBABILITY OF A STEAM EXPLOSION SHOULD BE ASSUMED TO BE UNITY.
- SHALLOW UNDERWATER EXPLOSION ANALOGY IS PARTICULARLY RELEVANT HERE.
- SHORT LENGTH FOR A SLUG ACCELERATION.
- IN-CORE INSTRUMENT SHAFT IS A VENT FOR THE EXPLOSION - I.E., NO LONG TERM ACCELERATION (MISSILE) POTENTIAL.
- SHOCK WAVES FROM A COHERENT EXPLOSION WITHIN THE REACTOR CAVITY WOULD NOT BE SUFFICIENT TO FAIL THE CONTAINMENT WALL.

STEAM EXPLOSIONS

- CONCLUSIONS -

I. IN-VESSEL

- ELEVATED SYSTEM PRESSURE-STEAM EXPLOSIONS WILL NOT OCCUR.
- LOW SYSTEM PRESSURES - STEAM EXPLOSIONS CAN OCCUR, BUT WOULD NOT FAIL THE REACTOR VESSEL-NO CONTINUOUS OVERLYING LIQUID LAYER CAN BE FORMED.

II. EX-VESSEL

- STEAM EXPLOSIONS CAN OCCUR, BUT THE SHOCK WAVES GENERATED WOULD BE MUCH LESS THAN THE CONTAINMENT DESIGN PRESSURE.

CORE COOLABILITY

Robert E. Henry

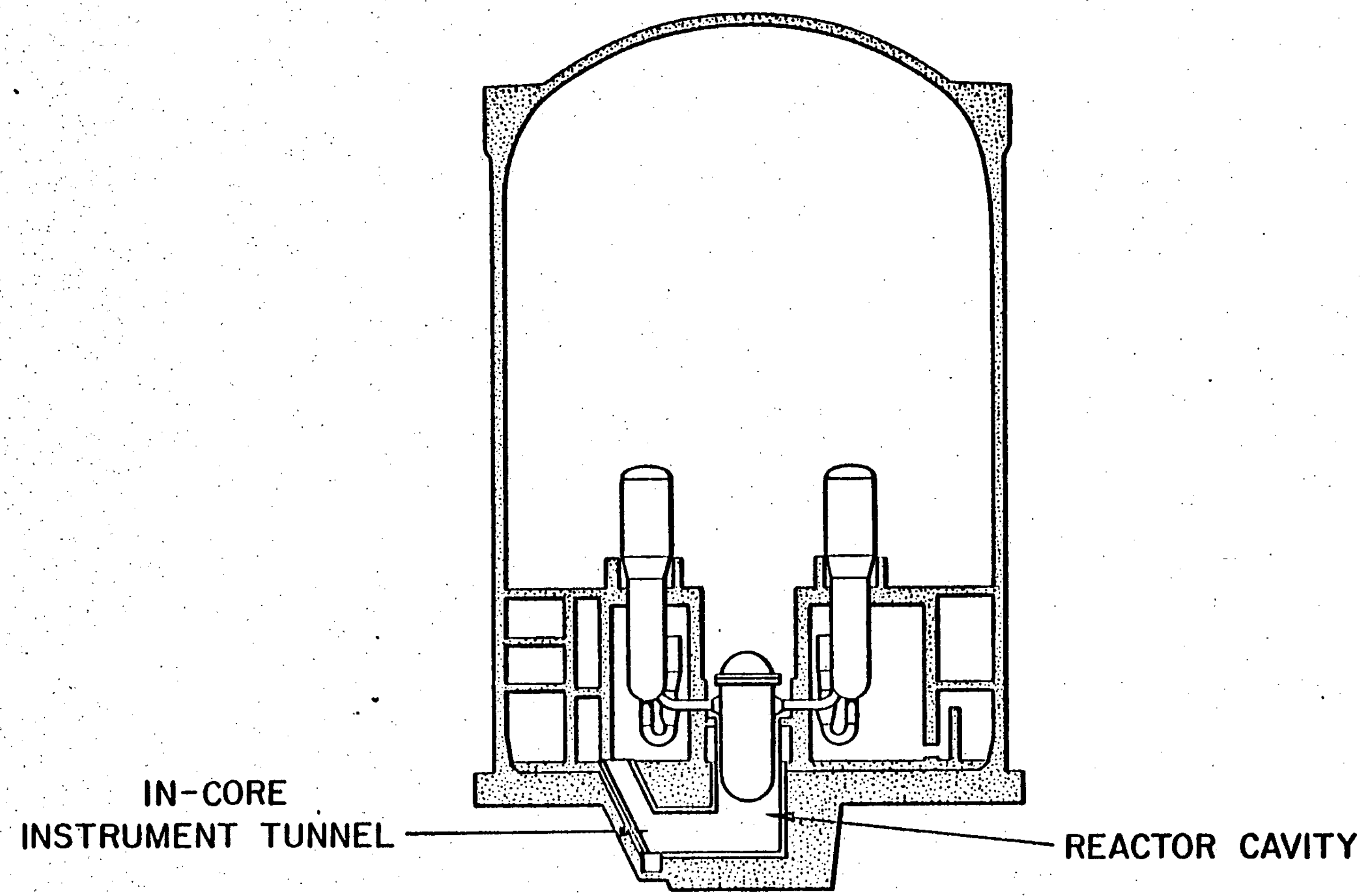
ARGONNE NATIONAL LABORATORY

(Slide 13a)

CORE COOLABILITY

- CONDITIONS TO BE EVALUATED -

1. MINIMUM WATER INVENTORY LOST BEFORE DAMAGE CAN BE INITIATED
2. PERMANENTLY COOLABLE SIZES
3. GOVERNING MECHANISMS FOR IN-VESSEL COOLING
4. GOVERNING MECHANISMS FOR EX-VESSEL COOLING



ZION REACTOR BUILDING

(Slide 13c)

MAXIMUM COOLABLE SIZE

$$\frac{\dot{Q}R^2}{K} = -\frac{d}{dR}\left(R^2 \frac{dT}{dR}\right)$$

$$\frac{\dot{Q}R^2}{6K} = T_M - T_S$$

$$R = \left\{ \frac{6K\,(T_M - T_S)}{\dot{Q}} \right\}^{1/2}$$

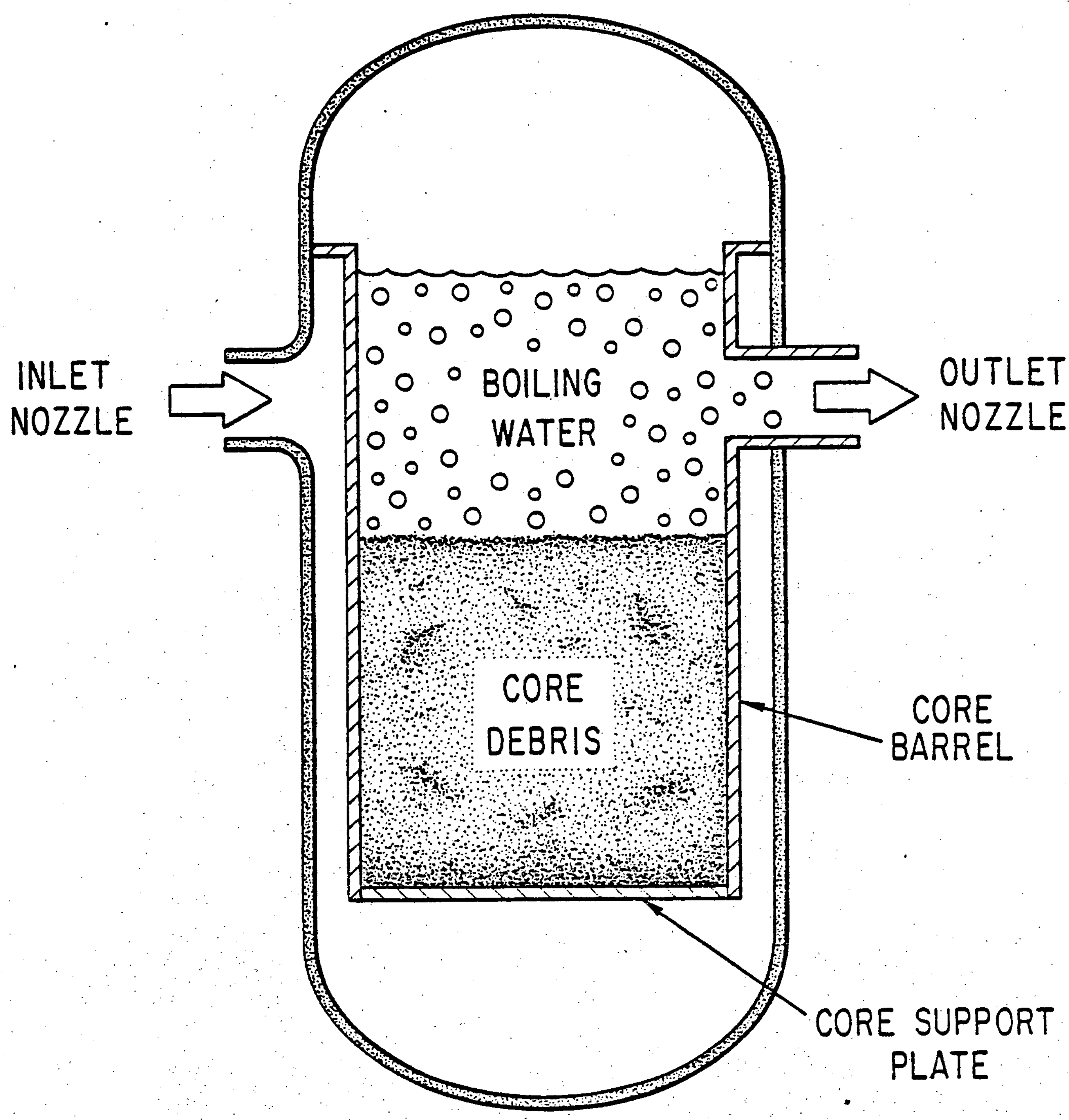
$T_M = 2800°C$ $\quad$ $T_S = 280°C$

$K = 0.0025$ KW/M°C $\quad$ $Q = 2550$ KW/M^3 (30 MW)

$R = 12$ CM $\quad$ $D = 24$ CM $= 9.6$ IN.

- THIS SIZE IS PERMANENTLY COOLABLE -

(Slide 13d)


INLET
NOZZLE
BOILING
WATER
OUTLET
NOZZLE
CORE
DEBRIS
CORE
BARREL
CORE SUPPORT
PLATE

(Slide 13e)

CORE COOLABILITY

I. CHF CRITERION - VAPOR REMOVAL RATE UPWARD MUST NOT PRECLUDE LIQUID RETURNING TO THE SURFACE

$$Q/A = 0.14\, H_{fg}\sqrt{\rho_g}\left[\, g\sigma\left(\rho_f - \rho_g\right)\right]^{1/4}$$

II. DEBRIS BED LIMITATION - PRESSURE GRADIENT INDUCED BY VAPOR FLOW THROUGH THE BED MUST NOT EXCEED THE STATIC HEAD OF THE LIQUID. THIS LIMITATION APPLIES AT THE TOP OF THE BED SINCE THIS IS THE LOCALE OF MAXIMUM VAPOR VELOCITY.

$$-\frac{dP}{dZ} = 2\, C_f\, \frac{\rho_g J_g^{\,2}}{D}\; \frac{1-\epsilon}{\epsilon^3} < \rho_f\, g$$

$$C_f = \frac{75}{R_E} + 0.875$$

$$R_E = \frac{\rho_g J_g D}{(1-\epsilon)\mu_g} \qquad J_g = \frac{Q}{\rho_g A\, H_{fg}}$$

CRITICAL HEAT FLUX

$$Q/A = 0.14\, H_{FG} \sqrt{\rho_G}\left[g\sigma(\rho_f - \rho_g)\right]^{1/4}$$

$P = 15$ MPA $\sigma = 0.005$ N/M $\rho_g = 100$ KG/M^3

$H_{FG} = 1000$ KJ/KG $\rho_f = 602$ KG/M^3

$Q/A = 3118$ KW/M^2 $A = 12$ M^2 $Q = 37.4$ MW

$P = 7$ MPA $\sigma = 0.018$ N/M $\rho_g = 36.5$ KG/M^3

$H_{FG} = 1505$ KJ/KG $\rho_f = 741$ KG/M^3

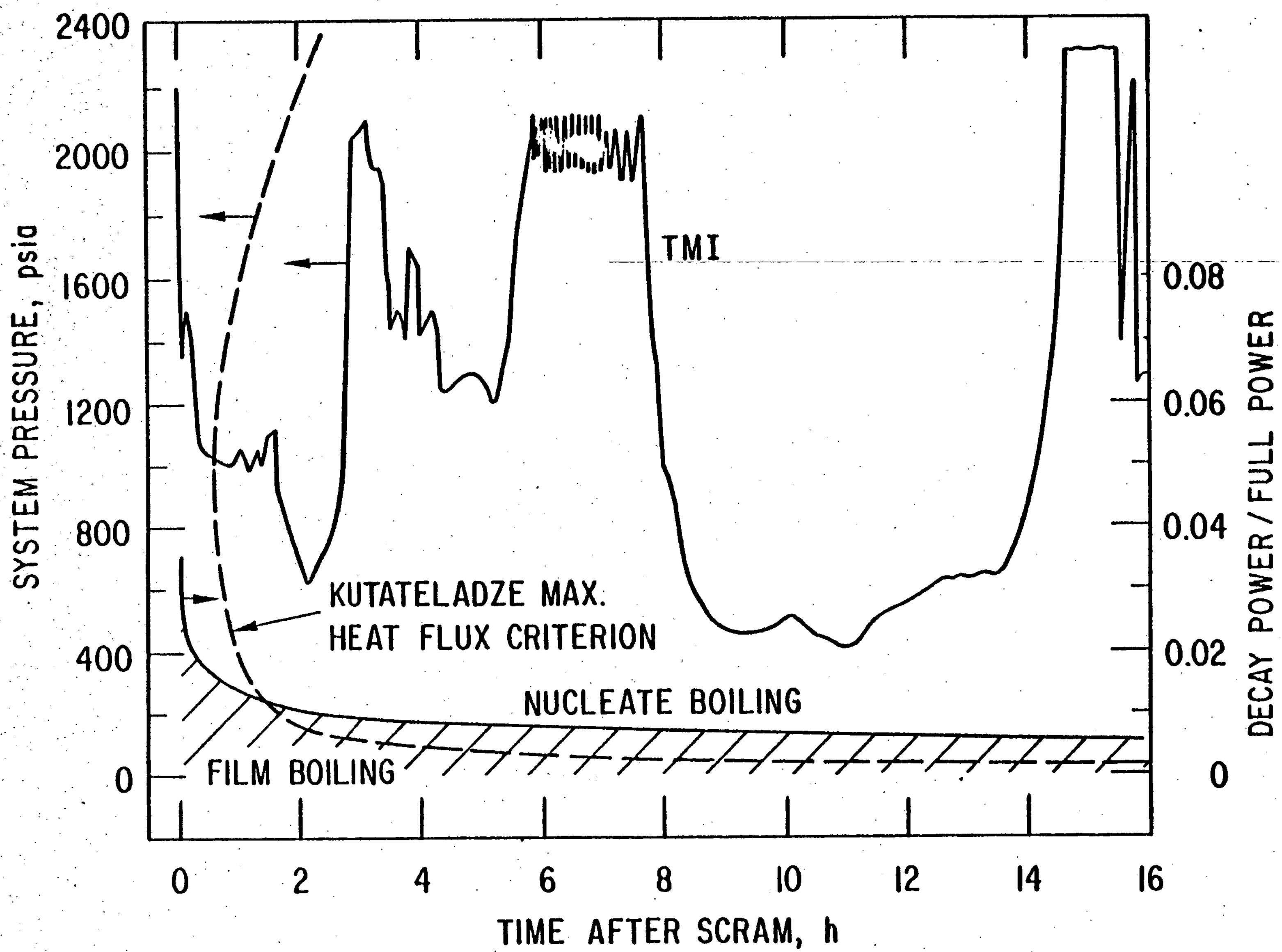
$Q/A = 4220$ KW/M^2 $A = 12$ M^2 $Q = 50.6$ MW

$P = 1$ MPA $\sigma = 0.045$ N/M $\rho_g = 5.2$ KG/M^3

$H_{FG} = 2015$ KJ/KG $\rho_f = 887$ KG/M^3

$Q/A = 2857$ KW/M^2 $A = 12$ M^2 $Q = 34.3$ MW

- NOT LIMITED ON THE UPPER SURFACE -



SYSTEM PRESSURE, psia
2400
2000
1600
1200
800
400
0
TMI
KUTATELADZE MAX.
HEAT FLUX CRITERION
NUCLEATE BOILING
FILM BOILING
0.08
0.06
0.04
0.02
0
DECAY POWER / FULL POWER
0
2
4
6
8
10
12
14
16
TIME AFTER SCRAM, h

(Slide 13h)

DEBRIS BED

$$-\frac{dP}{dZ} = 2\, C_f \frac{\rho_g J_g^2}{D} \frac{1-\epsilon}{\epsilon^3} < \rho_f g$$

$Q = 30$ MW $\qquad \dot{M}_V = Q/H_{fg}$

ASSUME $P = 7$ MPA $\qquad \dot{M}_V = \dfrac{30{,}000}{1505} = 19.9$ KG/SEC

$\rho_g = 36.5$ KG/M^3 $\qquad A = 12$ M^2 $\qquad J_g = 0.046$ CM/SEC

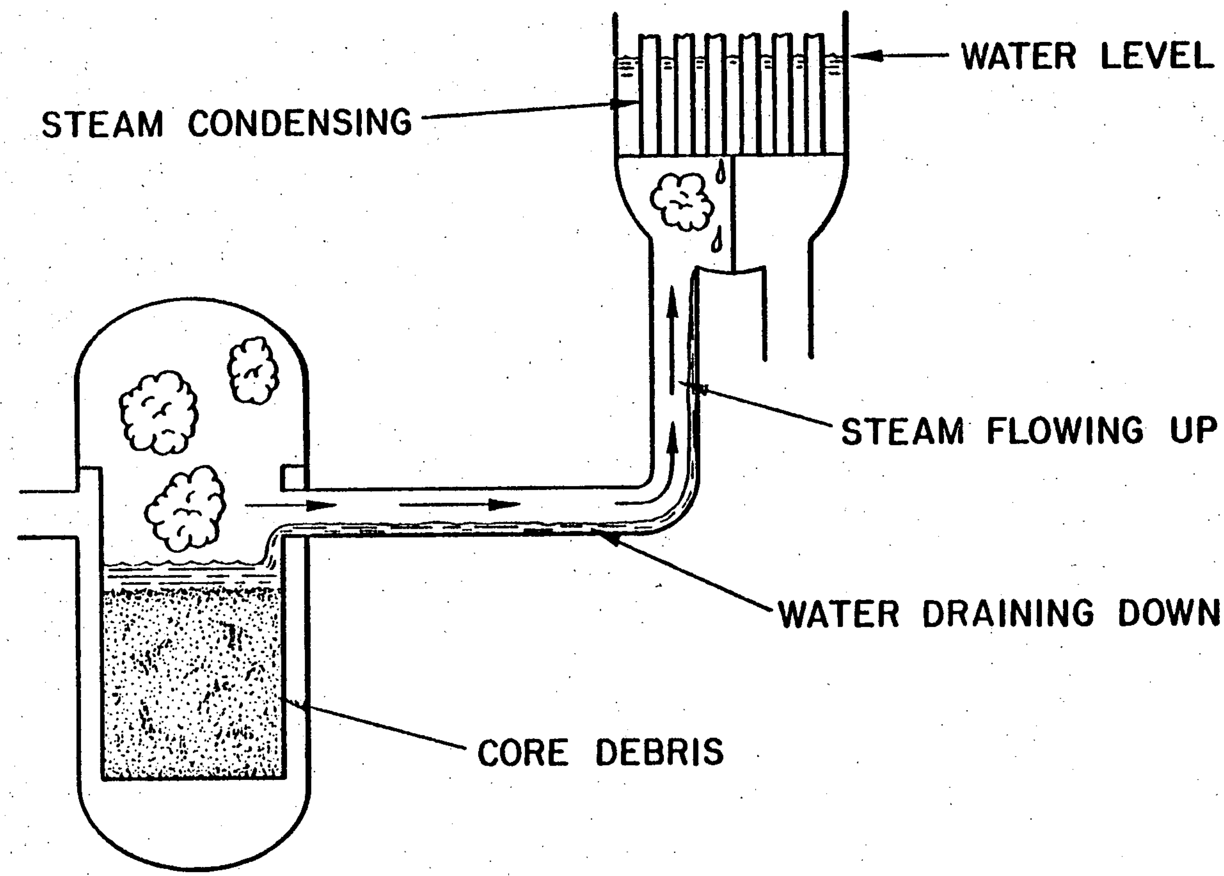
ASSUME $D = 0.01$ M $\qquad \epsilon = 0.5$

$R_E = 1687$ $\qquad C_f = 0.92$

$-\dfrac{dP}{dZ} = 57$ PA/M $\qquad \rho_f g = 10^4$ PA/M

- NOT LIMITED WITHIN THE BED -

(Slide 13i)



REFLUX PATH

(Slide 13j)

CONCLUSION

IN VESSEL COOLABILITY

IF WATER IS AVAILABLE ABOVE THE CORE,

A. IT WILL COME IN CONTACT WITH THE UPPER SURFACE OF THE CORE.

B. IT WILL PENETRATE DOWN THROUGH A BADLY DAMAGED CORE.

C. REFLUXING CAN OCCUR WITH A FRACTION OF ONE STEAM GENERATOR REMOVING THE HEAT. THIS HEAT REMOVAL PATH CAN BE ESTABLISHED IN THE PRESENCE OF HYDROGEN.

(Slide 13k)

EX-VESSEL CORE COOLABILITY

	ZION	INDIAN POINT
DECAY POWER	30 MW	30 MW
FLOOR AREA - REACTOR CAVITY & INSTRUMENT TUNNEL	53.6 M^2	39.1 M^2
REQUIRED HEAT FLUX	560 KW/M^2	767 KW/M^2

CRITICAL HEAT FLUX

0.1 MPA	1131 KW/M^2
0.2 MPA	1515 KW/M^2
0.3 MPA	1794 KW/M^2

- NO HEAT FLUX LIMITATION -

CONCLUSION

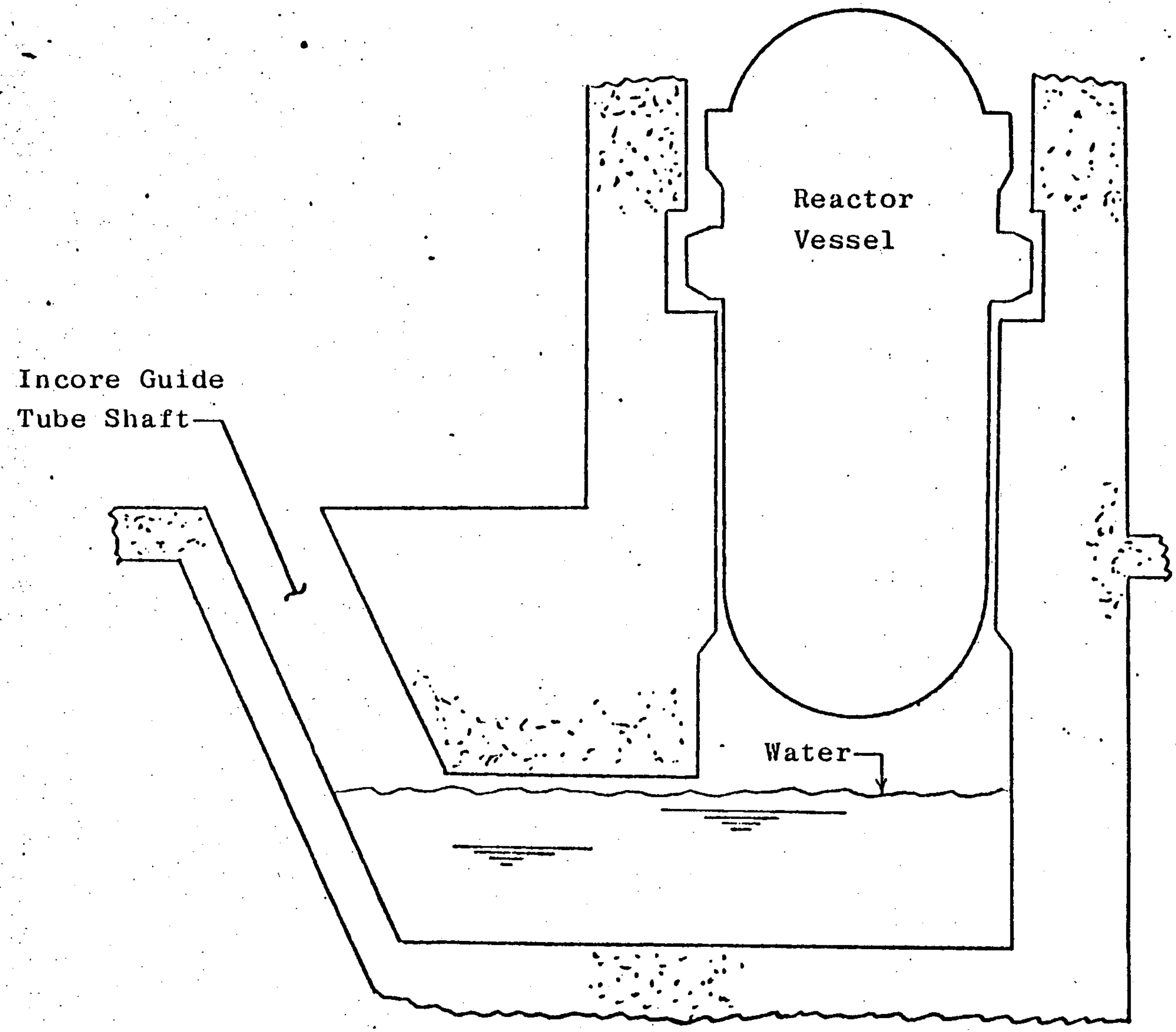
EX-VESSEL COOLABILITY

IF WATER CAN BE SUPPLIED TO THE REACTOR CAVITY,

A. IT WILL COME IN CONTACT WITH THE UPPER SURFACE OF THE CORE MATERIAL.

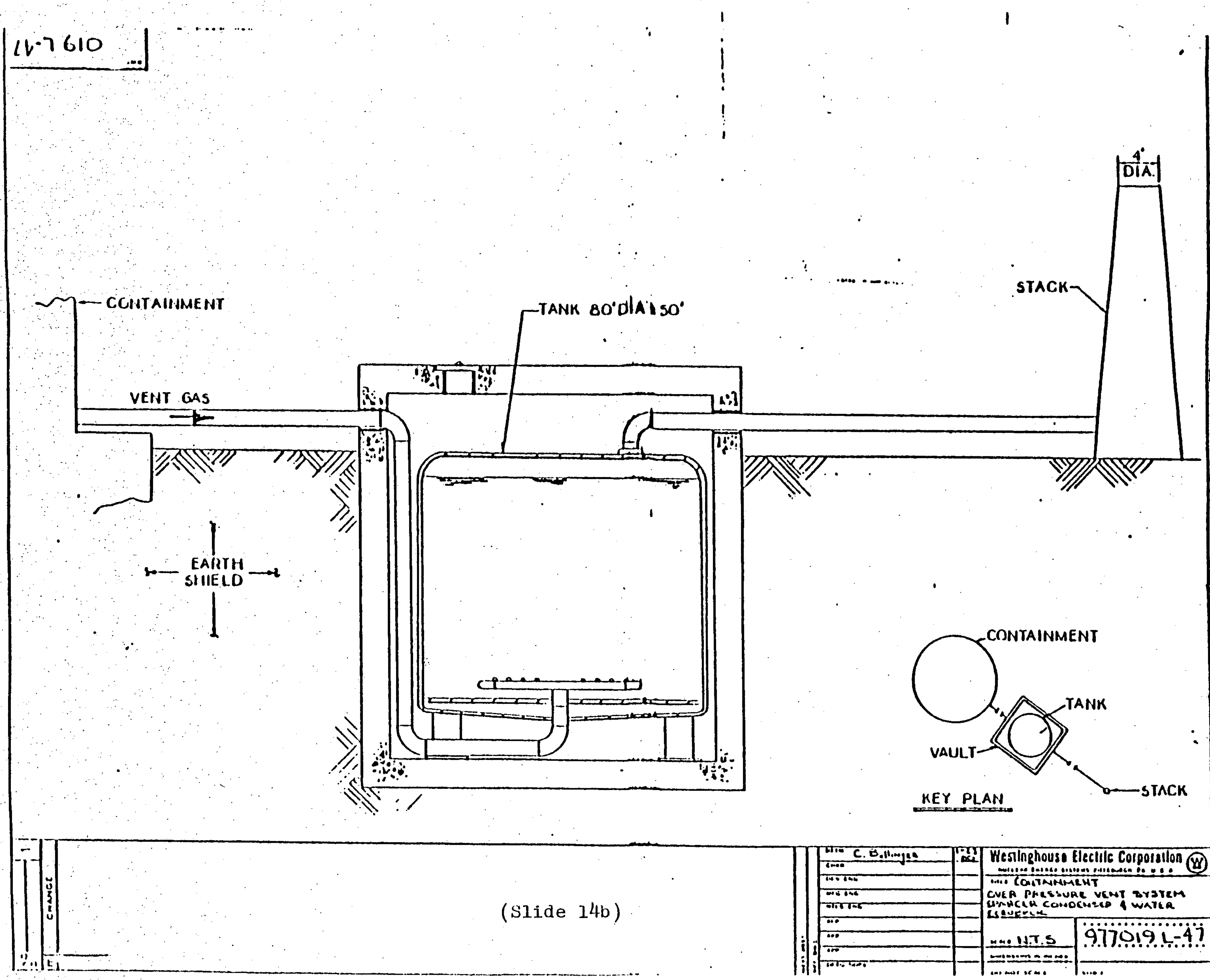
B. IT WILL PENETRATE THROUGH THE COARSELY FRAGMENTED DEBRIS.

C. REFLUXING CAN BE ESTABLISHED WITH THE BUILDING FAN COOLERS.

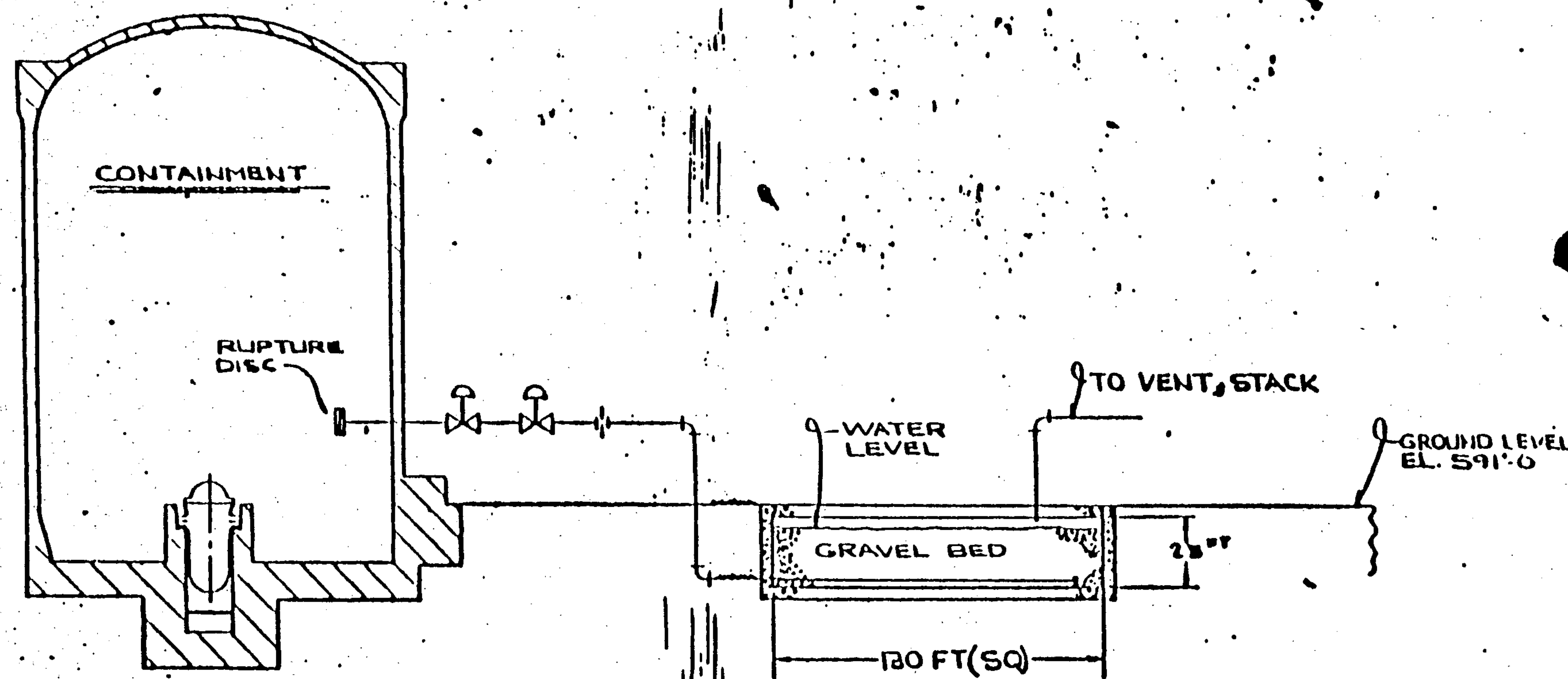


Ex-Vessel Core Cooling - Water Filled Cavity

(Slide 14a)


CONTAINMENT
TANK 80'DIA x 50'
STACK
4' DIA.
VENT GAS
EARTH SHIELD
CONTAINMENT
TANK
VAULT
STACK
KEY PLAN
C. Ballinger
Westinghouse Electric Corporation
CONTAINMENT
OVER PRESSURE VENT SYSTEM
SPARGER CONDENSER & WATER
N.T.S
977019 L-47

(Slide 14b)

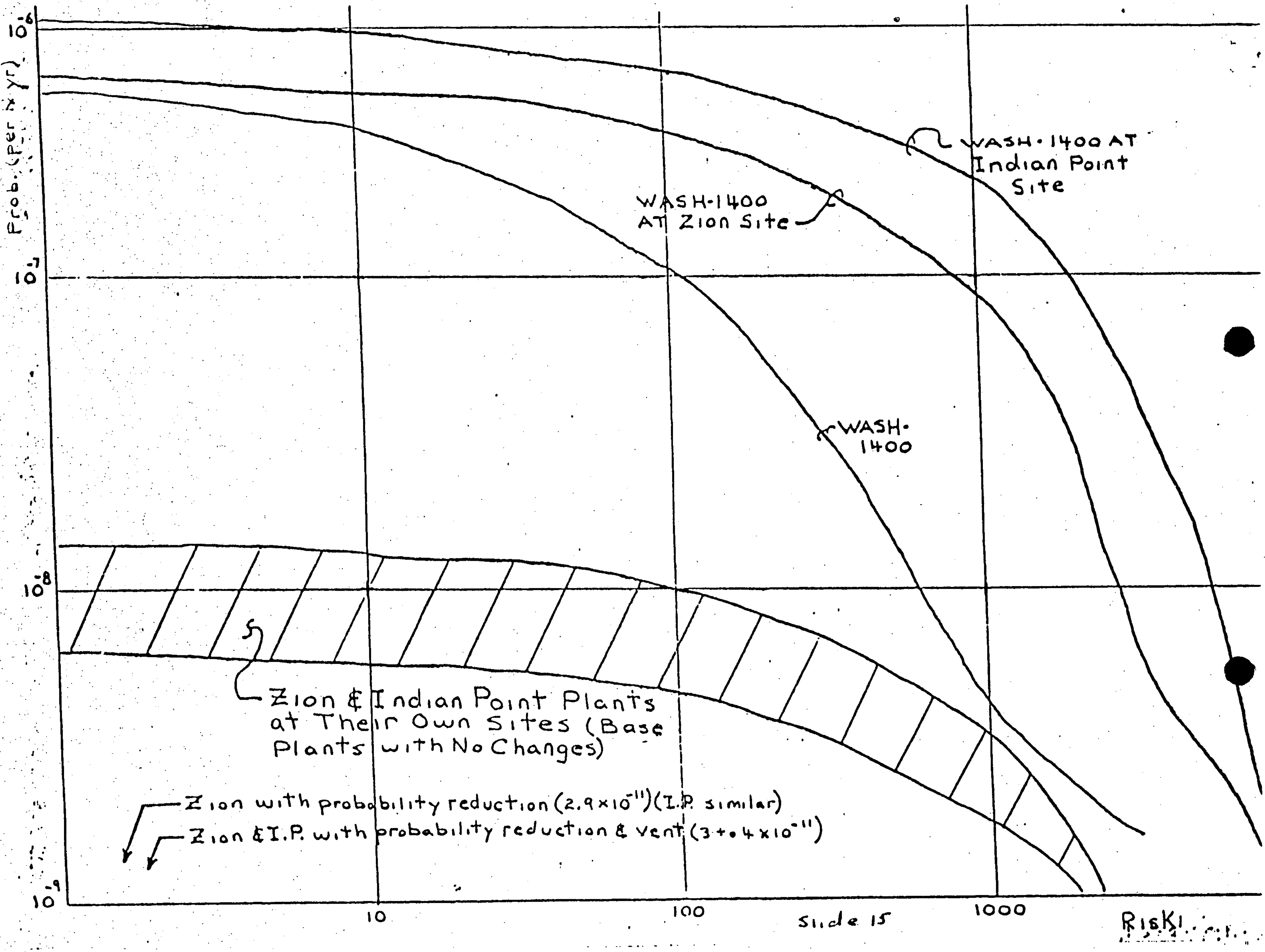


(Slide 14c)

W VANDRIAK

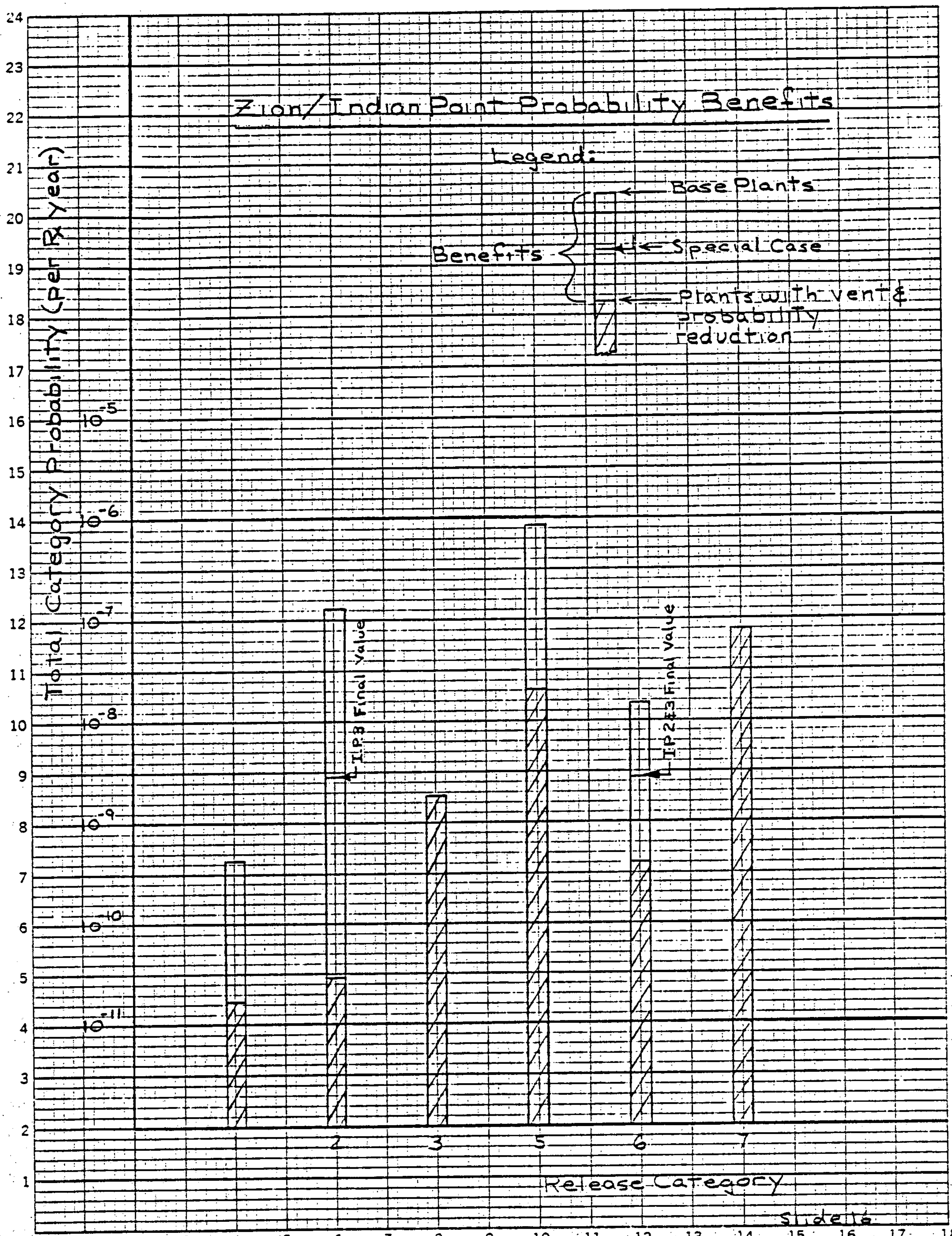
Westinghouse Electric Corp

CONTAINMENT
VENT SYSTEM
GRAVEL



Prob. (per x yr)
10^{-6}
10^{-7}
10^{-8}
10^{-9}
WASH-1400 AT Indian Point Site
WASH-1400 AT Zion Site
WASH-1400
Zion & Indian Point Plants at Their Own Sites (Base Plants with No Changes)
Zion with probability reduction (2.9×10^{-11}) (I.P. similar)
Zion & I.P. with probability reduction & vent ($3 \to 4 \times 10^{-11}$)
10
100
1000
Risk

Slide 15



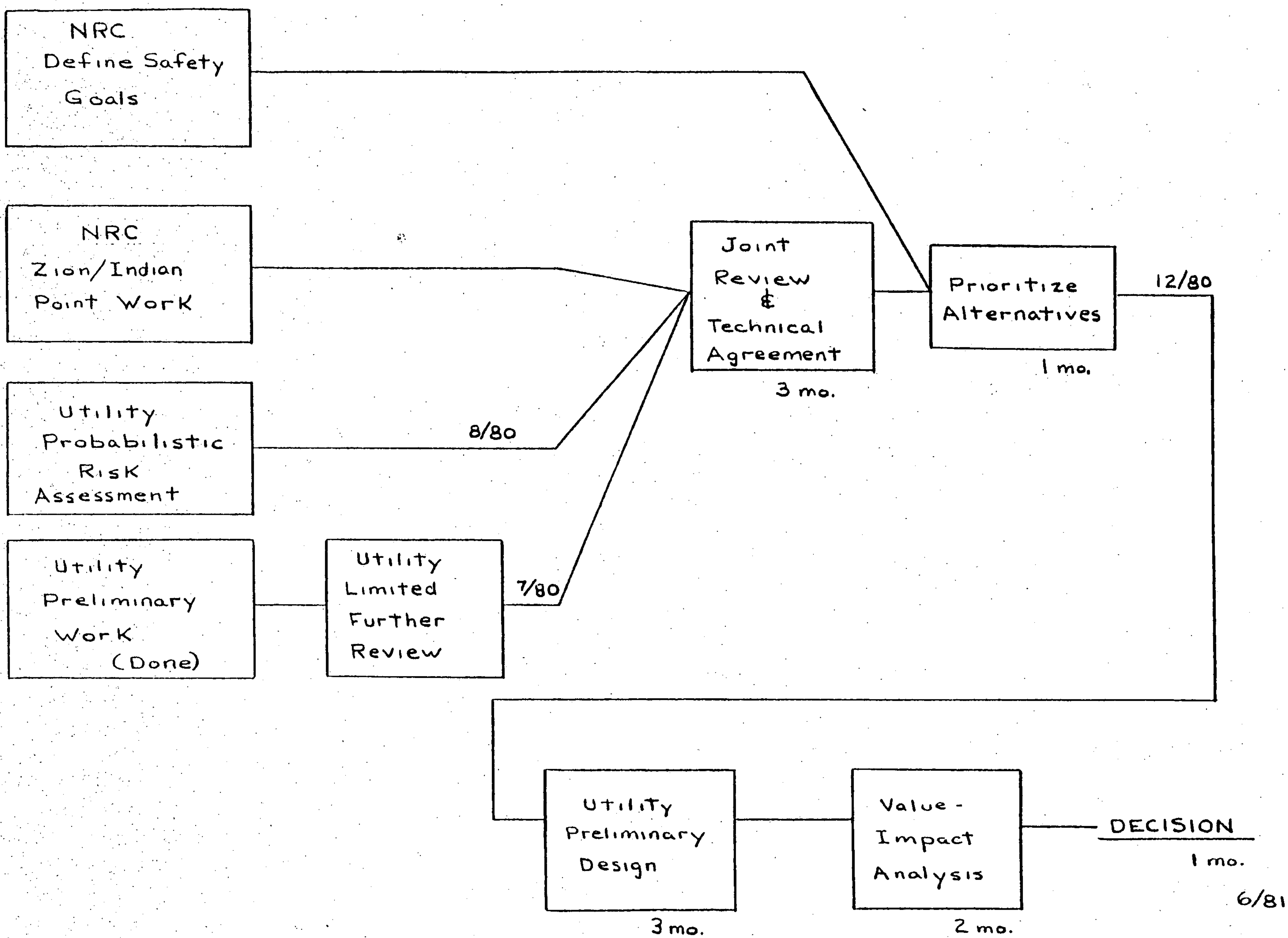
5 Squares to the Centimeter
12.189
5789

FUTURE ACTIONS

- SCHEDULE FOR RELIABILITY ANALYSIS
- DEVELOP CONCEPTUAL DESIGN
  - FURTHER REVIEW OF TRANSIENTS
  - TECHNICAL AGREEMENT
  - PRIORITIZE ALTERNATIVES
  - PRELIMINARY DESIGN WORK
- VALUE IMPACT

(Slide 17)

Future Actions



(Slide 18)