CONF-770611--14

OVERVIEW OF THE CRBRP SAFETY STUDY

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American Nuclear Society Annual Meeting
June 1977

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Prepared for the U.S. Energy Research and Development Administration
Under Contract No. E(11)-2395

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ABSTRACT

A Safety Study has been conducted for the Clinch River Breeder Reactor Plant (CRBRP). This paper presents a review and discussion of objectives, methods, techniques, and results of that study. The CRBRP Safety Study was conducted to; (1) provide a realistic assessment of accident risks associated with operation of the CRBRP, (2) place those identified risks in perspective with other societal risks, and (3) aid in determining whether accident risks from the CRBRP are comparable to those of previously licensed reactors.

Achievement of the objectives of this study has required identification of significant contributors to risk in a logical and orderly manner. Consideration of a comprehensive set of accident initiators, inclusion of experience data with a conservative bias, reliance upon proven methods and techniques, evaluation of a wide range of radionuclide releases and associated health effects, and utilization of experienced risk analysts are the salient elements employed in the systematic approach to this study. This together with heavy reliance on experience gained during years of Light Water Reactor (LWR) design, licensing, and operation provides reasonable assurance that the study objectives have been achieved.

Results of the CRBRP Safety Study indicate that the risk arising from the operation of CRBRP is small in comparison to other local societal risks, and that CRBRP risk is comparable to the risk from previously licensed nuclear power plants as identified in the Reactor Safety Study.

1.0 INTRODUCTION

The Clinch River Breeder Reactor Plant (CRBRP) Safety Study was performed to evaluate the risk to the public associated with the operation of CRBRP, to provide a perspective on this risk in relation to other societal risks, and to develop a basis for assessing the comparability between risk from the CRBRP and that from previously licensed nuclear power plants. Although the CRBRP design is not yet finalized the design of systems significant to plant safety are sufficiently developed to allow a meaningful risk assessment.

In this study risk has been taken to be a combination of the probability of potential accidents associated with the operation of the CRBRP and the resulting health effects on the population surrounding the Clinch River site.

Since the Reactor Safety Study (RSS) (1) was performed with a similar but considerably broader objective (to assess the risk to the public associated with 100 LWRs), the approach employed in the present study parallels that used in the RSS. This similarity of approach has facilitated a comparison between the risk associated with operation of the CRBRP on its own site and the risk associated with previously licensed nuclear power plants.

The key inputs to the evaluation are summarized below.

1. First, possible sources of radioactive material were screened to evaluate their potential for contribution to CRBRP risk. This screening process focused attention on core related accidents as being the most significant potential contributors to risk.

2. A comprehensive list of accident initiators was formulated and the probability of important accident sequences emanating from those initiators was characterized. Event trees and fault trees were employed in the development of these accident sequences and probabilities.

3. The accident sequences leading to core disruptive accidents (CDAs) were evaluated and generic classes of CDAs were defined. In this work a CDA has been defined as an accident in which loss of core coolable geometry occurs. All CDAs have been assumed to lead to core meltdown and to various degrees of mechanical damage to the primary system resulting from energetics derived from the CDA.

4. The probabilities that important generic classes of CDAs will lead to various degrees of mechanical damage to the reactor vessel and head were estimated.

5. Release categories, which define radioactive material releases from containment to the environment were established.

6. The radioactive material releases to the environment associated with each release category were evaluated and the effects on public health associated with these releases were calculated.

7. The occurrence probability of each release category was calculated. These probabilities were combined with the health effects for each release category to develop a representation of the risk associated with the CRBRP.

A flow diagram depicting the major steps of the CRBRP study is shown in Figure 1. The four major steps are:

1. Accident sequence definition and quantification

2. Core accident analysis and evaluation

3. Consequence modeling

4. Risk qualification

Chronologically an accident initiator leads to an accident sequence involving plant protective features. The states of these protective features, and thus the possible accident sequence paths, are described using event trees. The probability of occurrence of each branch of the event tree is estimated using fault tree methodology. Radionuclide release from the core to the reactor containment building (RCB) resulting from the accident are then evaluated. Characteristics of the behavior of the material released to the RCB and the releases to the environment are assessed. Finally, the public health consequences of the radionuclide releases to the environment are evaluated.

2.0 ACCIDENT SEQUENCE DEFINITION AND QUANTIFICATION

2.1 Potential Accident Initiator Definition

This portion of the study consisted of defining a list of potential accident initiating events, developing event trees to define the accident sequences which could result from these initiators, and quantifying the likelihood of these sequences. The study considered core-related accident sequences as well as ex-core accident sequences. This paper focuses on core-related accident sequences with the plant operating at or near full power since it is these sequences which were determined to dominate the risk from CRBRP.

The basic philosophy used to develop a comprehensive list of initiators was that to release a significant quantity of radioactivity, the core must overheat and melt. A list of 12 causal categories with the potential to lead to this condition was prepared. From these 12 causal categories, a list of 33 accident initiators which are potentially significant contributors to risk was developed. These intitators have been identified through years of safety analysis related to the LMFBR and particularly the CRBRP as well as for LWRs and other commercial and experimental nuclear plants. The RSS, the CRBRP Preliminary Safety Analysis Report (PSAR)(2), and the Draft Environmental Statement (DES)(3) were specific sources used to develop the initiator list. Consultation with design engineers and independent consultants was also used to assure a comprehensive initiator list. An accident initiator can be either internal to the plant (e.g. pumps coastdown) or external to the plant (e.g. loss of offsite power). When combined with safety system failures an initiator could lead to release of significant amounts of radioactivity to the environment.

2.2 Event Tree Use in Accident Sequence Development

Event trees have been used to provide a logical ordering of systems or functions designed to respond to an initiating event to prevent fuel and clad melt and radioactivity release to the environment. The event trees allow a description of the states (available or not available) of these systems and functions. Thus, a comprehensive set of accident sequences describing the possible system/function combinations evolved from the event trees.

Figure 2 is a functional event tree used as an aid in developing the core-related accident event trees and accident sequences. The three functional event tree headings are reactor shutdown, decay heat removal, and containment of radioactivity (if a CDA occurs). These headings represent the three basic safety functions in CRBRP which are designed to respond to potential accident initiators. The order of these functions was determined by timing considerations, i.e., reactor shutdown is a fast response function, followed by decay heat removal which begins after shutdown and extends for many days. Containment of radioactivity is last since it is required only if one or both of the first two functions fail resulting in a CDA. Success and failure for the first two functions is in general defined as whether or not core coolable geometry is maintained. Failure to maintain core coolable geometry is assumed to produce a CDA. For the containment of radioactivity function, success is defined as the case in which all containment systems work as designed and the CDA energetics are not large enough to breach containment. Failure is defined as one or more containment systems not working as designed or core accident conditions causing containment failure. Note in Figure 2 that for the sequence in which reactor shutdown and decay heat removal function successfully there is no branching for containment of radioactivity since no radioactivity is released and thus sequences involving containment are physically meaningless.

Since several safety systems are associated with each of the three basic safety functions depicted in Figure 2, a single event tree which represents all possible system combinations would be quit complex. To avoid the complexity three individual types of event trees corresponding to the three basic safety functions were constructed. These three event tree types were linked in series thus forming a multitude of paths, each of which represents an accident sequence.

The event tree types are as follows:

1. Initiator Event Trees - These event trees describe the plant response to the accident initiating events discussed above. The initiator event trees start with the initiating event and describe the accident sequence through successful or unsuccessful reactor shutdown.

2. Shutdown Heat Removal System (SHRS) Event Trees - These event trees give the response of the SHRS to the demand for decay heat removal following a successful shutdown. The shutdown could be due either to a reactor scram in which case the SHRS event tree is linked to the initiator event tree, or to normal shutdown in which case the SHRS event tree serves as an initiator event tree. The SHRS event tree starts with successful reactor shutdown as the initiator and describes the accident sequence through success or failure of the SHRS. SHRS failure is assumed to lead to a CDA.

3. Containment System Event Tree - The containment system event tree gives the response of the reactor containment building isolation and associated cooling and cleanup systems to a CDA resulting from a failure to shutdown or a failure to remove decay heat.

2.3 Accident Sequence Selection

The 33 potential accident initiators together with the physically meaningful paths through the series of three event trees yield on the order of 10^4 potential accident sequences. Probability discrimination among sequences having similar outcomes reduced the number of accident sequences to 69. This was accomplished in two steps. First, the approximately 10,000 physically meaningful sequences were reduced to about 200 by eliminating sequences on the basis of initiator recurrence frequency or probability of containment failure state. The second step in the probability discrimination process was to use system unavailability estimates to determine the probability of each of the 200 remaining sequences. In this screening process, those sequences with a probability of 10^{-10} per year and higher were retained. Those less than 10^{-10} per year were considered in relation to their impact on their associated release category. In cases where this impact was non-negligible, the sequence was retained.

2.4 Accident Sequence Qualification

To quantify the probability of accident sequences, estimates are required of the probability of success or failure of the systems and functions which make up the event tree branches. One ground rule of the safety study was to utilize RSS data, including system failure probability derivations, where applicable. Accordingly, many of the unavailability values were obtained from Appendices II and III of the RSS. Generally, the CRBRP systems were not anlayzed to the component level of resolution as was done in the RSS. Instead, system and subsystem level probability values derived in the RSS were applied to the CRBRP systems since, for single train systems (no redundancy), system failure probability values typically are not a strong function of individual component failure probability values. The system values are, however, dependent on the logic makeup of the system (e.g., how much redundancy is built into the system and how independent the redundant parts are of one another and of potential common cause failure events). Where possible the selection of RSS system probabilty values for application to the CRBRP systems was based on the similarities in the logic makeup of the systems.

Factors which affect a system's availability are introduced at each stage of design, fabrication, installation, checkout, and operation. Since the CRBRP is still in the final design stage, operational data are not available. However, by applying RSS systems analysis results to the CRBRP systems, common mode, operational procedures, test and maintenance activities, and other related events, identifiable only from a complete plant, are included to the extent possible in the CRBRP safety study. Furthermore, because the CRBRP follows the same stringent licensing process as do LWR's and a reliability program more detailed than any previously implemented is being applied, overall system reliability at least equivalent to LWRs is assured.

Sources of component failure data other than RSS were used, particularly for sodium systems and components. Such data as the LMEC Failure Data Handbook(4) and documented sodium plant operating experience(5) were utilized.

The main protection systems which are involved in the significant accident sequences are the reactor shutdown system (consisting of plant protective system and the control rod system the shutdown heat removal system (SHRS), and the containment system. Fault trees were constructed for the SHRS and containment system because of the need to define interfaces and dependencies for these two systems. The PPS and SCRAM unavailabilities were estimated using experience data from water reactors and design information from CRBRP. The fault trees were evaluated using the PREP computer code (described in the RSS, Appendix II) to identify important minimal cut sets, i.e., the minimum number of failure events which would lead directly to system failure.

The event trees together with the fault trees form the basis for the sequence fault diagrams. A sequence fault diagram was constructed for each of the 69 significant accident sequences. These diagrams combine the logic of the event trees and fault trees with the failure data discussed above to provide a representation of each accident sequence. The sequence fault diagrams display the necessary information on the initiating event, protection system conditions, and containment failure mode to completely define the accident sequence and facilitate quantification of the sequence probability. An example of a sequence fault diagram is shown in Figure 3.

The probability of an accident sequence is calculated by combining the individual fault event probabilities shown on the sequence fault diagram for the accident sequence. This was done using SAMPLE, a computer code used in the RSS and described in Appendix II of that study report.

For each fault event appearing in the function for sequence fault diagram, a failure probability for that fault event is obtained by a random sampling of the assigned log-normal probability distribution. Next, all of the fault event probabilities are used to compute a value for the sequence occurrence rate. This process is repeated for a large number of trials to obtain a distribution for the overall sequence occurrence rate.

The computed sequence occurrence rate distribution can be summarized by the median value and the 90% distribution interval. The upper and lower bounds of the 90% simulation interval are determined by the 95% and 5% percentiles of the empirical probability distribution.

2.5 Completeness and Common Mode Considerations

Consideration of a comprehensive set of accident initiators, inclusion of experience data with a conservative bias, reliance upon proven methods and techniques, and utilization of experienced risk analysts have provided reasonable assurance that the study objective of completeness has been achieved.

Identification of common mode failure was an integral part of the work. For example, accident initiating events (e.g., loss of offsite power, earthquakes, and fires) which could have a common mode effect by impacting more than one system or component were considered. Also, where potential dependencies between multiple systems or components existed, the extent of the dependency was estimated and reflected as a common mode failure probability.

3.0 CORE ACCIDENT ANALYSIS AND EVALUATION

Plant protection and safety systems are provided to prevent the occurrence of core accidents. If an initiator should occur these features are expected to function as designed and no loss of core coolable geometry would result. Should a sufficient number of these systems fail to function as designed fuel integrity may be lost followed by the possibility of release of radioactive material to the environment. Any accident in which there is a loss of core coolable geometry has been termed a core disruptive accident (CDA). The accident sequences developed in this work were evaluated and generic classes of CDAs were defined. Among these generic CDA classes are loss of decay heat removal capability following reactor shutdown, loss of heat removal capability at power, loss of piping integrity, and loss of flow with failure to shutdown.

Each of the generic CDA classes has been investigated to ascertain the ensuing progression of events within the core during the accident and the condition of the core and reactor primary system at the conclusion of the accident. Three accident progression paths are possible depending on the accident initiator and the assumptions made about the accident phenomenology. First, an "early termination" path is possible for some accidents. In this case, removal by melting and dispersal of small amounts of fuel terminates the accident, leaving the bulk of the core sufficiently intact so that it can be cooled in place. Second, the initiating phase of some CDAs could lead directly to a hydrodynamic disassembly. Various levels of energetics may be associated with this accident progression path. Possible damage to the primary system arising from the energetics has been assessed as described below. Third, a path exists which leads to a "transition phase". This phase is entered through a more gradual core meltdown which terminates the initiating phase, leaving the core in a subcritical but uncoolable configuration. The transition phase has been assumed to result in further fuel dispersal via relatively gradual fuel removal processes. However, since scenarios which lead to hydrodynamic disassembly conditions from the transition phase cannot be absolutely ruled out, recriticallity arising from the transition phase has been considered by treating events both within the reactor vessel following fuel melting and within the reactor cavity following vessel melt-through.

As may be inferred from the above discussion, CDA's can lead to a wide spectrum of effects. For most CDAs, the highest probability event is relatively slow melting of several fuel assemblies followed perhaps by collection of the molten fuel and steel in the bottom of the reactor vessel. If a sufficient fraction of the core is involved, peneteration by the molten fuel material through the bottom of the reactor vessel and guard vessel may occur. This is termed thermal damage. It has been conservatively assumed that all events leading to loss of core coolable geometry also lead to melt-through of the reactor and guard vessels.

For each type of CDA probabilities have been assigned to the various degrees of mechanical damage which might result. The degree of mechanical damage to the vessel as a result of a CDA can be related to the amount of fuel vapor formed during the accident. In general, the larger the fraction of the core vaporized, the greater will be the energy transmitted to the reactor vessel and head and the greater will be the potential for mechanical damage. Three classes of mechanical damage have been defined and the expected occurrence of each of the damage classes is given below.

- No seal damage (for approximately 90% of the CDAs),
- Moderate seal damage (for approximately 10% of the CDAs),
- Massive seal failure (for approximately 1% of the CDAs),

These categories of mechanical damage are related to three expected ranges of accident energetics:

- Non-energetic termination (energetics so low that seal design performance is not impaired),
- Energetics less severe than structural capabilities of head seals,
- Energetics exceeding the structural capabilities of the head seals.

A fourth mechanical damage class has been defined:

- Highly energetic CDA.

The highly energetic CDA has been defined, non-mechanistically, to be an accident with sufficiently high energetics to cause significant damage both to the reactor vessel head and to the reactor containment building. Thus, a significant fraction of the radionuclides released from the core could escape into the environment through the damaged containment. Scientists and engineers with detailed knowledge and experience in LMFBR safety analysis judge that CDAs resulting in such severe levels of mechanical damage are extremely improbable. However, the current status of CDA analysis techniques is such that occurrence of such a severe event cannot be ruled out. Therefore, the highly energetic CDA has been considered in this risk assessment. The probability of the highly energetic CDA has been estimated to be 10% of the probability of the massive seal failure class (the third class noted above). Table 1 is an example of the format in which the probabilities of various degrees of mechnical damage to the reactor vessel head have been tabulated. Such tables were prepared to specify mechanical damage probabilities for each of the nine (9) generic CDAs.

Analytical results from several computer codes developed for the purpose have been employed in evaluating the accident progressions and in determining the resultant mechanical damage to the reactor, vessel and primary system.

The SAS-3A (6) code was used to analyze the initiating phase of each accident, and the VENUS-II (7) code was used to analyze any hydrodynamic disassemblies which were predicted to occur. In the transition phase analysis for CRBRP, key phenomena were examined as separate effects, judgment was employed to construct the scenario. The damage evaluations for events that produce structural loadings have been accomplished using the REXCO-HEP (8) and ANSYS (9) computer codes.

Each category of vessel mechanical damage has been conservatively analyzed to determine the release of radionuclides and sodium from the reactor vessel and into the reactor containment building (RCB) as a result of the initial accident energetics phase. The radionuclide and sodium releases into the RCB arise from two sources. First, radionuclides are released and some sodium may be released (depending upon the degree of seal damage) to the RCB through the head during and shortly after the energetic CDA, and second, material is released to the RCB during the approximately two hundred hours following the CDA as a result of sodium boiling in the reactor cavity (RC) which is vented to the RCB. This boiling is caused by decay heat generated in the molten fuel following melt-through of the reactor vessel and guard vessel.

Table 1 shows the fractional releases of non-volatile core material and sodium entering the RCB in each of the four classes of mechanical damage associated with a core disruptive event.

4.0 RADIOACTIVE MATERIAL TRANSPORT AND RELEASE ANALYSIS

As shown in Figure 4, three principal paths exist for transport of radionuclides and sodium from the reactor vessel: one path through the head, one path involving melt-through of the reactor vessel/guard vessel combination and boil-off of sodium from the RC into the RCB, and one path involving possible penetration of the molten fuel through the concrete base mat into the ground water and eventually to the surface water.

The first two of these three paths lead to releases of radionuclides to the RCB where they can either fallout, decay, be removed by scrubbing and filtration, or be released to the environment. This section will discuss the analysis of these release paths. The third path has been evaluated and, as in the RSS (1), was shown to contribute insignificantly to the overall risk from the CRBRP.

4.1 Description of Release Paths

A significant fraction of the gaseous fission products, a smaller fraction of volatile solids, and an even smaller fraction of non-volative solids could be released through the head shortly after the energetic CDA occurs. Estimates of these fractional releases (see Table 1) were derived for each of the categories of vessel/head mechanical damage by the following procedure:

- o It was assumed that a partial failure of the head seals occurs instantanteously upon initiation of the CDA.
- o Using a pressure-time history within the reactor vessel and time required for the fuel vapor bubble to rise from the core to the reactor head, the release of sodium through the reactor head and the fraction of fuel and fission products released was estimated.

Following the initial releases of sodium and radionuclides through the reactor head (occurring immediately after the CDA), some fraction of the core would be expected to melt and to collect in the bottom of the vessel. If this fraction collecting in the bottom of the vessel is greater than approximately 4% of the core, eventual melt-through of the reactor vessel and guard vessel would be expected (perhaps occurring as early as 15 minutes after the CDA). Breach of the reactor vessel and guard vessel would lead to draining of the available sodium from the primary system into the RC. This release of sodium would be followed shortly by melting of the remainder of the core and its accumulation in the bottom of the RC. Following vessel melt-through, the fission product decay energy would heatup and boil-off sodium in the RC into the RCB. During this boil-off of sodium, a significant fraction of volatile fission products would follow the sodium into the RCB. The material in the RCB would be gases, vapors, and aerosols from the boil up material source as well as from the source initially released through the head.

Natural depletion mechanisms including plate-out, condensation, and aerosol agglomeration and settling exist to remove fission products, core material, and sodium from the RCB atmosphere. These depletion mechanisms do not rely on any active system to initiate them, but are benevolent phenomena which occur in the presence of sodium/sodium oxide aerosols, dust particles, and metal surfaces. Pressurization of the RCB, caused by heat evolved in the oxidation of sodium and fission product decay leads to leakage of radionuclides into the environment, and eventually, assuming availability of containment systems, venting of the RCB atmosphere through a scrubber/filter system. In the analysis performed for this assessment, the radionuclide releases which occurred over a period of hundreds of hours were integrated to determine the total release during the course of the accident. These total releases are assumed to issue from the RCB as a single puff. This is considered to be a conservative assumption in that if the release were uniformly distributed as a function of time, the effects would be diminished due to dilution and environmental processes.

4.2 Analytical Tools for Release Analysis

A number of computer codes were used to analyze radionuclide transport and release from containment. These codes have been developed over many years, during which considerable experimental verification of their predictive capabilities has been performed. The key assumptions employed in the radionuclide transport and release analyses are shown in Table 2. The SOFIRE (10) and SPRAY (11) computer codes have been used to evaluate the temperature and pressure history in the RCB following the initial (Path 1 on Fig. 4) release of sodium and radionuclides. This pressure history was used in the RCB leakage analysis during the time between initial material release through the head and the onset of sodium boiling in the RC.

The CACECO computer code (12) has been used to calculate the pressure and temperature in the RC and RCB following vessel melt-through. The results of this analysis were used to evaluate the radionuclide source term to the RCB and, together with the aerosol analysis code HAA-3 (13), to evaluate the release of radionuclides from the RCB to the environment.

CACECO is a computer code used to calculate the heat and material transport within the RC and RCB during the time following melt-through of the vessel until the sodium has been completely boiled away from the RC. All of the important heat sources (e.g., fission product decay heating and sodium vapor combustion) and heat sinks (e.g., the walls of the RC and the containment cooling system) were modeled in the CACECO analysis.

The COMRADEX code has been used to calculate the total radionuclide release based on its calculated time histories of radionuclide releases from the RCB.

4.3 Cases Analyzed for Radionuclide Releases

Radionuclide releases to the environment have been characterized as dependent only on the initial release of radioactive material through the reactor vessel head to the RCB and on the state of availability of the containment systems. The releases of radioactive material through the reactor vessel head during the accident energetic phase have been discussed and the results presented in Table 1. The possible availability states of containment systems are shown in the containment event tree, Figure 5. Analysis of the various possible sequences representing system availability states to determine expected probabilities and radionuclide releases led to the selection of four containment event tree paths as bounding sequences for detailed analysis using the methods discussed earlier. Those paths are A, C, I, and L. For each of these four paths, the three categories of primary system mechanical damage shown in Table 1 have been considered. Analysis of the releases represented by the eleven combinations (L2 and L3 were combined and represented as L3) of containment system availability state and degree of primary system mechanical damage shown in Figure 5 has resulted in release conditions shown in Table 3 and in the fractional releases of core inventory shown in Table 4. The twelth release evaluated, shown in Table 4, is that for the highly energetic CDA (HECDA).

These values represent the fraction of the radionuclide inventory in the CRBRP core at the end of an equilibrium cycle calculated to be released to the environment. These radionuclide releases were used in evaluating the public health consequences.

5.0 HEALTH CONSEQUENCE ANALYSIS

Following a radioactivity release from the containment building, material would be transported downwind according to the prevailing meteorological conditions. Ultimately, the radioactivity is either uniformly mixed with the global atmosphere or is deposited on the surface by wet (rain) or dry (fallout) deposition.

The radiation exposure to man caused by the release of radioactivity to the atmosphere was divided into three components:

- o Direct γ -ray exposure from the passing cloud
- o Irradiation from material deposited on the ground
- o Deposition of radioactive material in the body by inhalation of the passing cloud

Possible medical consequences considered include:

- o Early death from acute whole body exposure
- o Respiratory impairment from acute lung exposure

- Growth of thyroid nodules from cumulative thyroid exposure

- Fatal latent cancer from cumulative whole body exposure

A computer model similar to that used in the RSS was employed to characterized potential consequences. The block diagram for this model is shown in Figure 6. The radionuclide releases associated with each of the twelve release categories were assumed to occur as a "puff" releases. Health effects arising from each release were calculated using a spectrum of atmospheric dispersion conditions characteritics and the population distribution surrounding the site. The computer model used the International Committee on Radiation Protection (ICRP) dose/damage model to compute health effects.

6.0 RISK QUANTIFICATION AND RESULTS

6.1 Results

The final step in assessing the CRBRP risk was to evaluate the probability of various degrees of health effects resulting from the radionuclide releases discussed earlier. By utilizing the probability of occurrence of each release of radioactivity and the spectrum of health effects resulting from the probabilistic treatment of meteorological phenomena for these releases, complementary cumulative probability curves were developed for each of the important health effects. A summary of some of the more important results drawn from this work is given below:

1. The frequency of core accident sequences leading to release of radioactive material is estimated to be one in forty-five thousand per year of reactor opertion. Most of these accident sequences would have an insignificant effect on public health.

2. The frequency of higher consequence core accident sequences potentially leading to simultaneous failure of the primary system boundary and the reactor containment building is estimated to be one in 200 million per year of reactor operation.

3. Studies indicate that major variations in important input data and assumptions do not alter the main risk assessment conclusions.

4. Because of the number of different accident sequences which are important contributors, there is no single component or system failure which dominates the risk from the CRBRP.

6.2 Comparison with Societal Risk

To obtain a proper perspective on CRBRP accident risks, comparisons between these risks and those emanating from other sources are presented. Figure 7 shows the comparison between the frequency of early fatalities (occurring within one year of the postulated accident) and the frequency of fatalities resulting from man-caused events. The data for early fatalities resulting from man-caused events have been derived from results presented in the RSS and renormalized to the expected occurrence rate within ten miles of the CRBRP site. Ten miles has been selected because all of the acute fatalities resulting from potential CRBRP accidents have been predicted to occur within ten miles of the site. The population within ten miles of the plant is approximately 42,000. Selection of a larger radius would lead to a somewhat more favorable comparison between CRBRP risks and other societal risks.

Risk can also be expressed in terms of individual risk of death per year. Table 5 presents individual risks from both natural and man-caused sources for comparison with individual risk from CRBRP. This table shows that a person living within 10 miles of the CRBRP site is 100,000 times more likely to be fatally injured by lightning and a million times more likely to drown than to be fatally injured by a CRBRP accident.

6.3 Comparison with Commercial Nuclear Power Plant Risk

Figure 8 shows a comparison between the early fatality risk from CRBRP and that from a single current generation LWR as determined by the Reactor Safety Study. Figure 9 shows a similar comparison for latent fatalities.

Potential consequences associated with the CRBRP accidents are comparable to those of the LWR. Care must be exercised, however, in drawing direct comparisons from these curves. Three important considerations are listed below:

1. The LWR curve is the average of LWR sites throughout the United States whereas the CRBRP curve is specific to the Clinch River site in Tennessee.

2. The LWR curve is for a 1000 MWe plant whereas CRBRP is 380 MWe. This means that the fission product inventory in CRBRP, at an equivalent fuel burnup, is approximately one third that in a current generation LWR in which the plutonium inventory is about one third that of the CRBRP.

3. Even though the CRBRP consequences appear to be significantly less than those for an LWR, there are larger uncertainties associated with the CRBRP curve.

Some general observations can be made regarding the risk comparison. The probability of core melt in an LWR (5×10^{-5} per year) and the probability of a low energetic CDA in CRBRP (2×10^{-5}) are comparable. There does appear to be a difference in failure probability for the containment for the two reactor types. The RSS indicates about 10^{-1} for containment failure given a core melt. The CRBRP Safety Study calculates about 10^{-2} for a delayed overpressure failure of containment. These lower numbers result because the pressurization of the CRBRP containment occurs over a longer period of time than that for an LWR. The longer time results in a higher probability that electric power can be restored to operate CRBRP containment to reduce RCB pressure before containment failure pressures are reached.

6.4 Summary of Important Release Categories

Of the twelve release categories for which radionuclide releases have been developed only four contribute significantly to the overall CRBRP risk. Table 6 shows the four release categories, A1, L1, I1 and the highly energetic CDA, which contribute 100% to the risk of acute fatality and 99% to the risk of latent fatality arising from the operation of the CRBRP. Also shown on Table 6 are the types of accident sequences which contribute most signficantly to the four release categories. As shown, loss of decay heat removal capability following seismic events, loss of electric power events, and shutdown heat removal system failures contributes a very significant fraction to the overall CRBRP risk. Other important accident sequences include loss of flow without shutdown and loss of piping integrity.

7.0 Conclusions

To logically draw conclusions from this work the objective of the study should be restated:

- o The objective of the study is to perform a realistic but conservatively biased assessment of the risk involved with the operation of the CRBRP.

Having performed such an assessment a determination not only of the relative risk from the plant but also of the comparability of this risk from CRBRP with the risk from previously licensed nuclear plants can be made.

The conclusions therefore are:

- o The risk associated with postulated CRBRP accidents is small when compared to non-nuclear risks to which the local population is already exposed.

- o CRBRP risks are comparable to those from current generation LWRs as characterized in the RSS (1).

- o The results of this study and those presented in the RSS reveal that the risks arising from both systems (LWRs and CRBRP) is extremely small.

8.0 REFERENCES

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(5) G. O. Collins, "Development of Large Electromagnetic Pumps for Main Heat Transport Systems of LMFBRs", GEAP-13965, June 1973.

(6) M. G. Stevenson, et. al., "Current Status and Experimental Basis of SAS LMFBR Accident Analysis Code System", CONF-740401-P3, 1303-1321.

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(8) Y. W. Chang and J. Guildys, "REXCO-HEP: A Two Dimensional Computer Code for Calculating the Primary System Response in Fast Reactors," ANL-75-19, June 1975.

(9) ANSYS, Swanson Analysis Systems, Inc., 870 Pine View Drive, Elizabeth Pa., 15037. (ANSYS is a proprietary computer code of Swanson Analysis Systems, Inc.).

(10) P. Beiringer, et. al., "SOFIRE-II Users Report", AI-AEC-13055, March 1973.

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(12) R. D. Peak, "User's Guide To CACECO Code," Hanford Engineering Develments Laboratory, Richland, Washington, HEDL-TC-859, 1977.

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TABLE 1

TYPICAL VESSEL MECHANICAL DAMAGE AND RELEASE POTENTIAL MATRIX FOR A CORE DISRUPTIVE EVENT

Vessel Damage / Core Damage	Mechanical Damage			
	No Head Seal Damage (Case 1)	Moderate Head Seal Damage (Case 2)	Massive Head Seal Failure (Case 3)	Highly Energetic CDA
Whole Core Melting	0.9	0.1	0.01	0.001
Non-Volatile Core Material Release to Containment (% of Core)	0.01	1.0	10.0	**
Sodium Released to Containment (lb)	10.0	100.0	1000.0	

**Releases to Environment:

	Noble Gases	100%
	Halogens	70%
	Volatile Solids	50%
	Non-Volatile Core Material	10%

TABLE 2

ASSUMPTIONS IN RADIOISOTOPE RELEASE ANALYSIS

- o Hydrogen Recombination Was Assumed
- o Sodium Penetrtion into Concrete
  - * 1/2 inch per hour
  - * Maximum penetration is 2 inches
- o RCB Failure at 20 PSIG
- o RCB Purge and Vent at 10 PSIG
- o 99% Scrubber/Filter Efficiency
- o Containment Leakage Rate is 0.1 Volume Percent Per Day at 10 PSIG

TABLE 3

BASIC FEATURES OF RADIOISOTOPE RELEASE CASES

CONTAINMENT EVENT TREE PATH	ANNULUS COOLING	SCRUBBER/ FILTER	VENT TYPE	VENT TIME	PURGE TIME
C	YES	NO	10 PSIG	20.7 HR	24.1 HR
A	YES	YES	10 PSIG	20.7 HR	24.1 HR
I	YES	NO	HOLE	NONE	22.0 HR
L	NO	NO	OVERPRESSURE FAILURE	31.2	NONE

TABLE 4

SUMMARY OF CRBRP ACCIDENT RELEASES TO ENVIRONMENT

CONTAINMENT EVENT TREE PATH	PROBABILITY[1]	Release (percent of Core Inventory) HALOGENS	NOBLES	VOLATILE SOLIDS	Pu, AM, Cm
HECDA[2]	4.9-9	70	100	50	10
I3	1.5-10	3.3	61.	3.3	3.2
L3	2.5-8	0.91	45.	0.96	1.0-3
L1	1.2-6	0.78	34.	0.82	1.2-6
I2	2.3-9	0.48	52.	0.50	0.22
I1	8.5-8	0.48	51.	0.30	0.0021
C1	5.5-9	0.38	76.	0.40	1.7-6
C2	1.3-10	0.38	76.	0.40	1.7-4
C3	7.7-12	0.35	76.	0.37	1.0-3
A3	4.9-8	0.0075	76.	0.0078	9.4-4
A2	6.1-7	0.0075	76.	0.0078	1.4-4
A1	2.0-5	0.0074	76.	0.0077	1.4-6

(1) These numbers should be interpreted as follows 4.9-9 is 4.9×10^{-9}

(2) This HECDA case refers to the highly energetic CDA

TABLE 5
INDIVIDUAL RISK OF FATALITY BY VARIOUS CAUSES[(a)]

Source of Fatality	Probability of Death per Resident per year
Motor Vehicles	3.7×10^{-4}
Falls	8.1×10^{-5}
Fires and Burns	3.1×10^{-5}
Drowning	2.6×10^{-5}
Poison	2.5×10^{-5}
Firearms	1.3×10^{-5}
Cancer	1.8×10^{-3}
Water Transport (b)	7.6×10^{-6}
Air Transport (b)	5.2×10^{-6}
Railroad Transport (b)	3.3×10^{-6}
Farm Accidents (c)	1.7×10^{-4}
Electricity Usage	5.2×10^{-6}
Lightning	3.2×10^{-6}
Tornadoes	3.0×10^{-7}
Suicide	1.2×10^{-4}
Homocide	1.4×10^{-4}
All Accidents	7.2×10^{-4}
LWRs (d)	4.3×10^{-11}
CRBRP Accidents: Early (e)	2.9×10^{-11}
Latent (f)	4.5×10^{-12}

(a) These probabilities have been derived from data for the population within 50 miles of the CRBRP site for the Year 1973 (approximately 700,000). Year to year variations are expected to be small as are local population variations within this region.
(b) Exludes persons on duty.
(c) Per farm resident only.
(d) Based on RSS estimate of early fatalities for one reactor and all affected population within 25 miles.
(e) This number is the estimated probability of early fatalities per resident per year based on the assessment presented in this report for the population within 10 miles of the CRBRP site (approximately 42,000).
(f) This number is the estimated probability of latent fatality per resident per year based on the assessment presented in this report for the population within 10 miles of the CRBRP site.

TABLE 6

SUMMARY OF IMPORTANT RELEASE CATEGORIES AND CONTRIBUTING ACCIDENT SEQUENCES

RELEASE CATEGORY	PROBABILITY	ACCIDENT SEQUENCE	PERCENT CONTRIBUTION TO RELEASE CATEGORY
A1 INSIGNIFICANT ENERGETICS- Containment Works as Desinged	2.0×10^{-5}	Loss of Power	57.
		Seismic	19.
		Loss of Flow W/O Shutdown	8.
			84.
L1 INSIGNIFICANT ENERGETICS- Containment Over-Pressure Failure	1.2×10^{-6}	Seismic	86.
		Loss of Power	12.
			98.
I1 INSIGNIFICANT ENERGETICS- Failure to Isolate Containment	8.6×10^{-8}	Seismic	62.
		Loss of Flow W/O Shutdown	28.
		Loss of Piping Integrity	3.
			93.
HIGHLY ENERGETIC CDA- Containment Failed by Event	4.9×10^{-9}	Loss of Flow W/O Shutdown	35.
		Loss of Electric Power	23.
		Decay Heat Removal Failure (with loss of offsite power)	20.
		seismic	10.
			88.

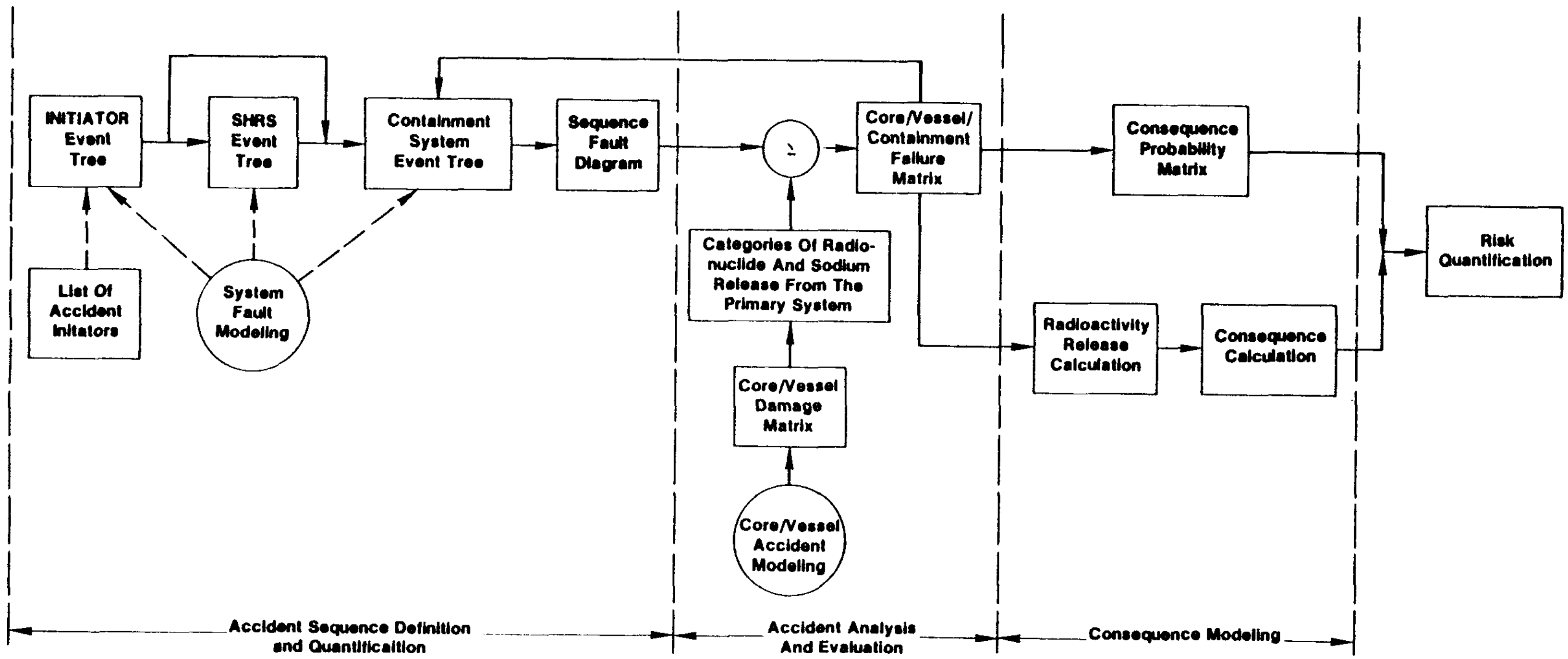

Figure 1. CRBRP Safety Study Flow Diagram

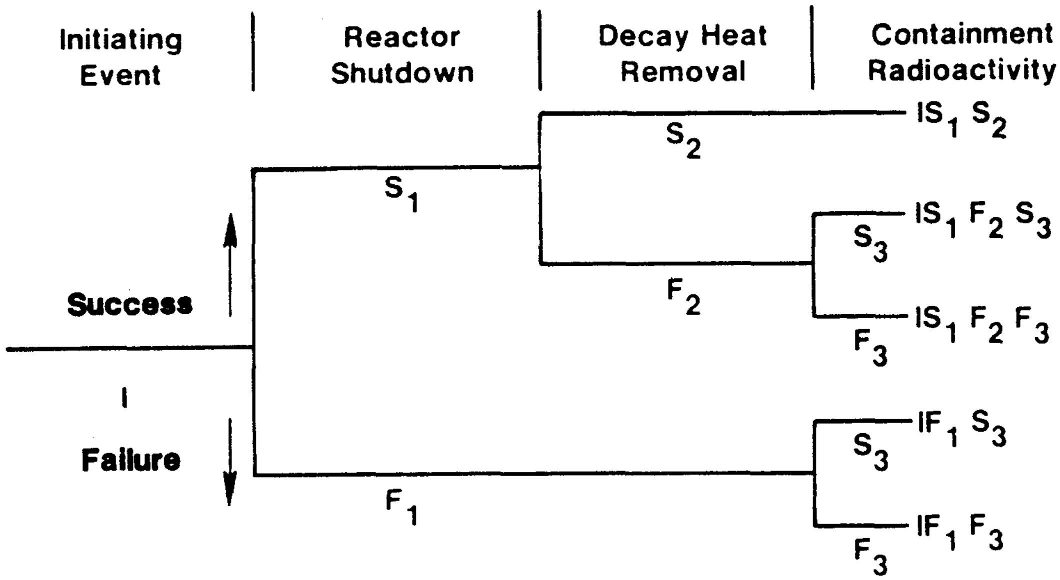

Figure 2. Functional Event Tree for Core Related Accident Sequence Definition

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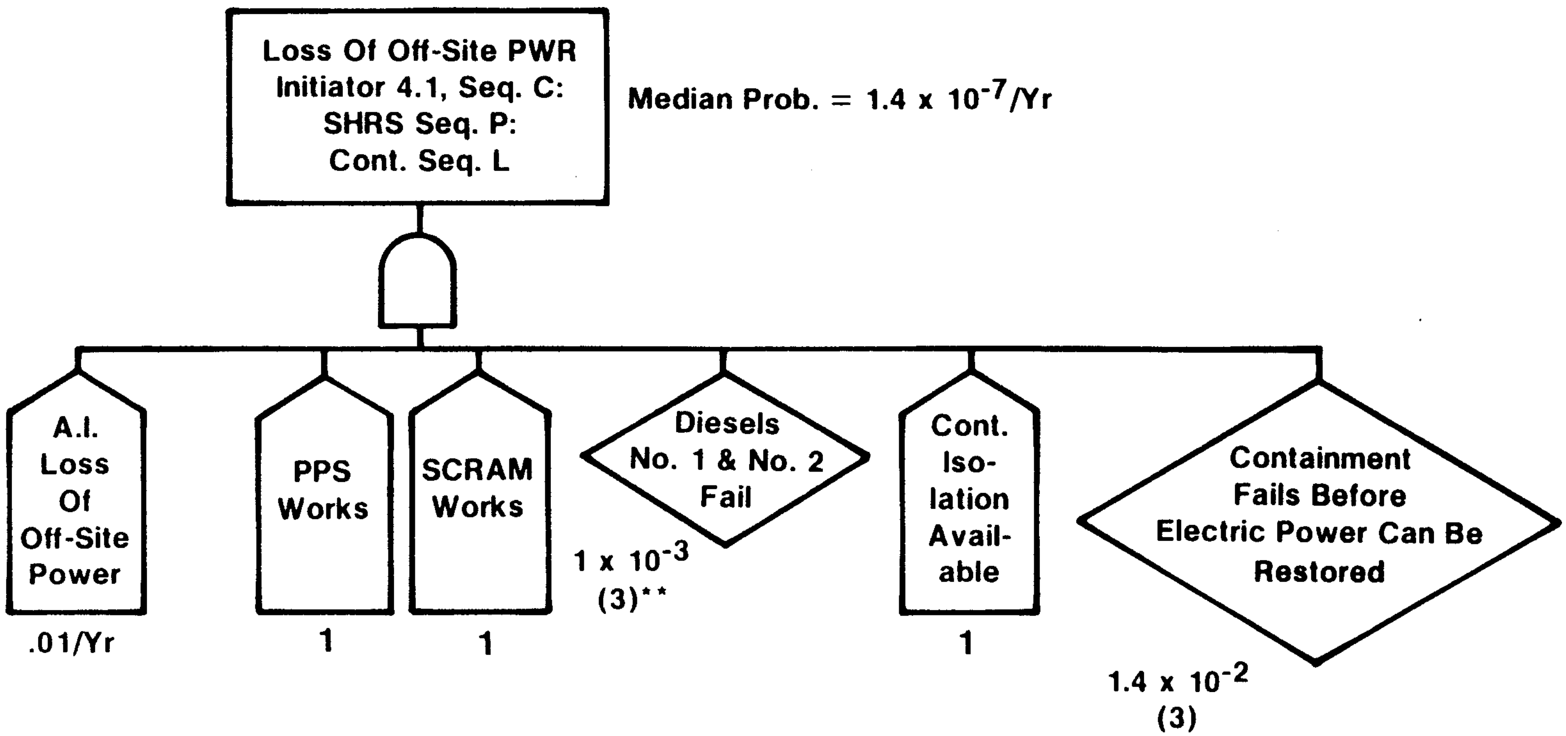


**These Numbers In Parenthesis Refer To Uncertainty Factors On The Probability Values

Figure 3. Example Sequence Fault Diagram Depicting Loss Of Off-Site Power Sequence C With SHRS Sequence P And Containment Sequence L.

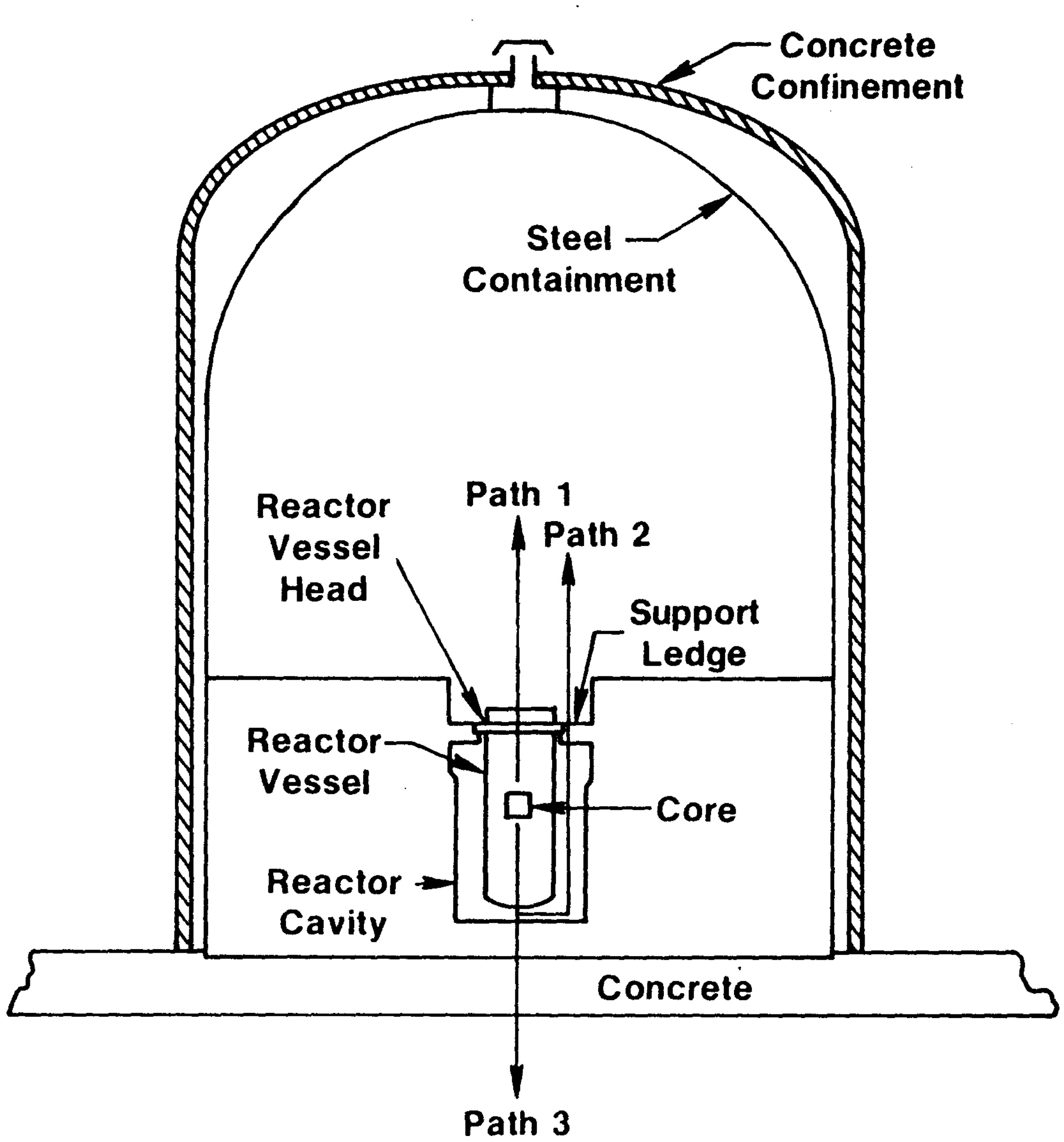
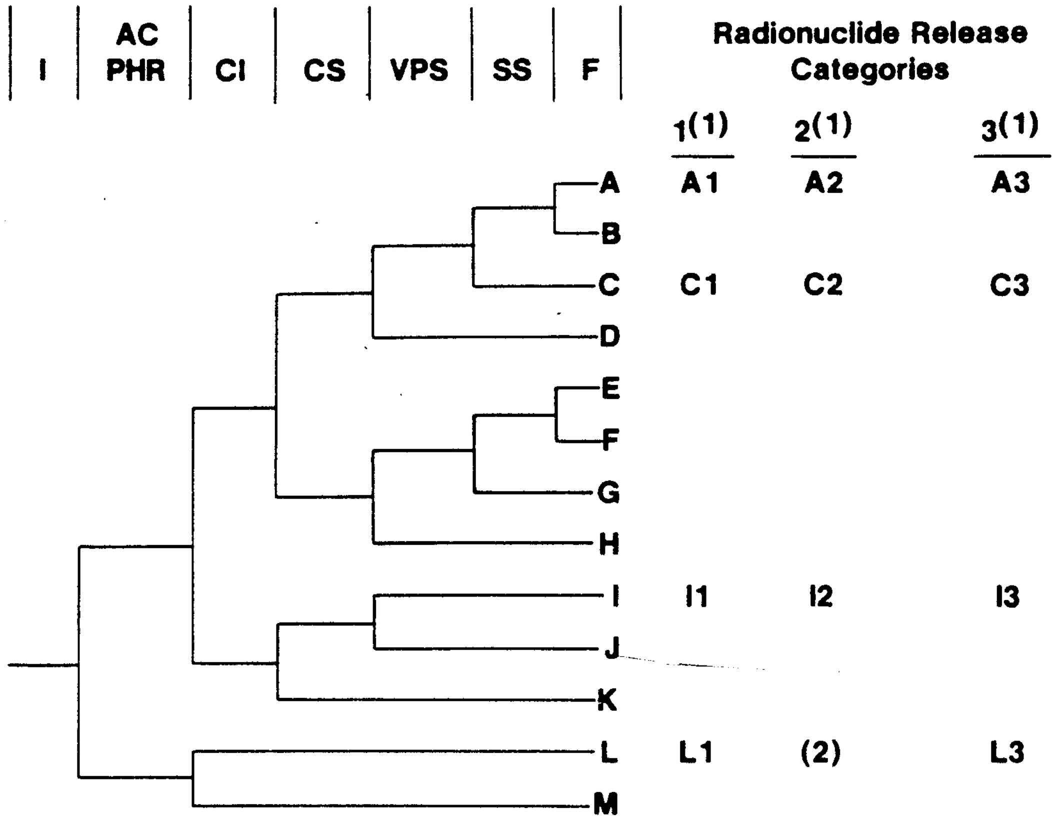


Figure 4. Schematic Diagram Of Post-CDA Radionuclide Flow Paths



I—Initiating Event (High Radiation In Containment)

AC PWR—AC Power

CI—Containment Isolation

CS—Containment Cooling System

F—Filters

VPS—Vent & Purge System

SS—Scrubber System

Notes:

(1) Vessel Damage Designation
3: Massive; 2 Moderate: 1: Insignificant

(2) The Probability Of This Release Category Has Been Conservatively Added To That Of Category L1

Figure 5. Correlation Of Release Calculations And Categories With The Containment Event Tree

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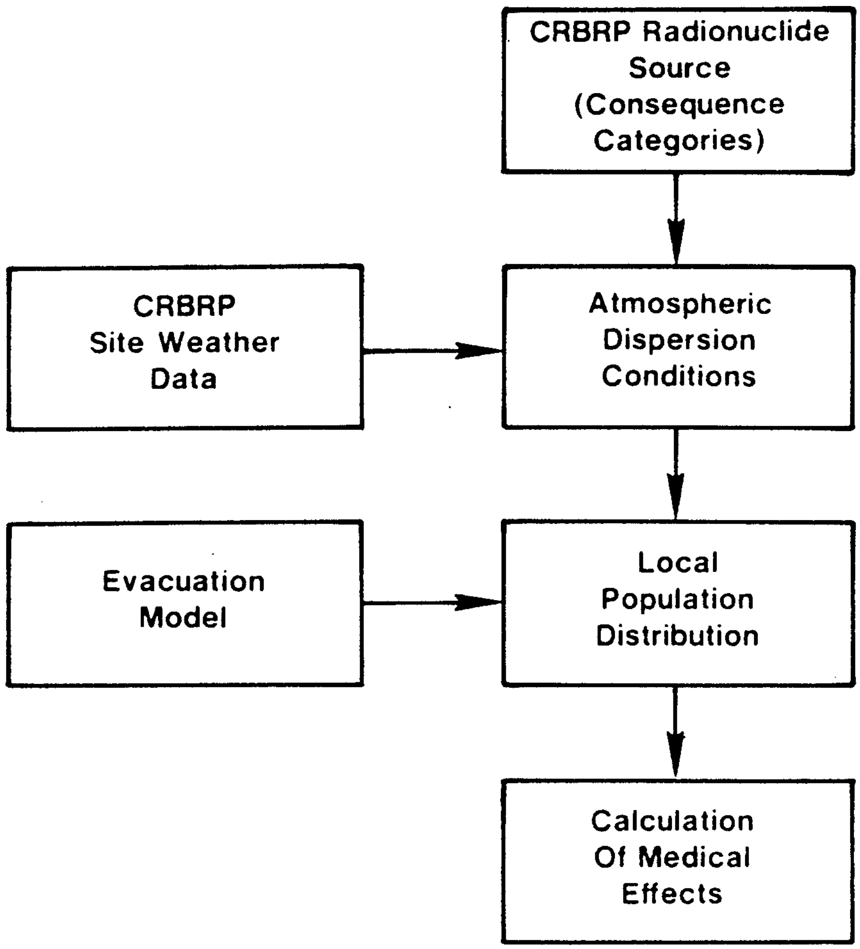

Figure 6. Schematic View Of Consequence Calculational Model

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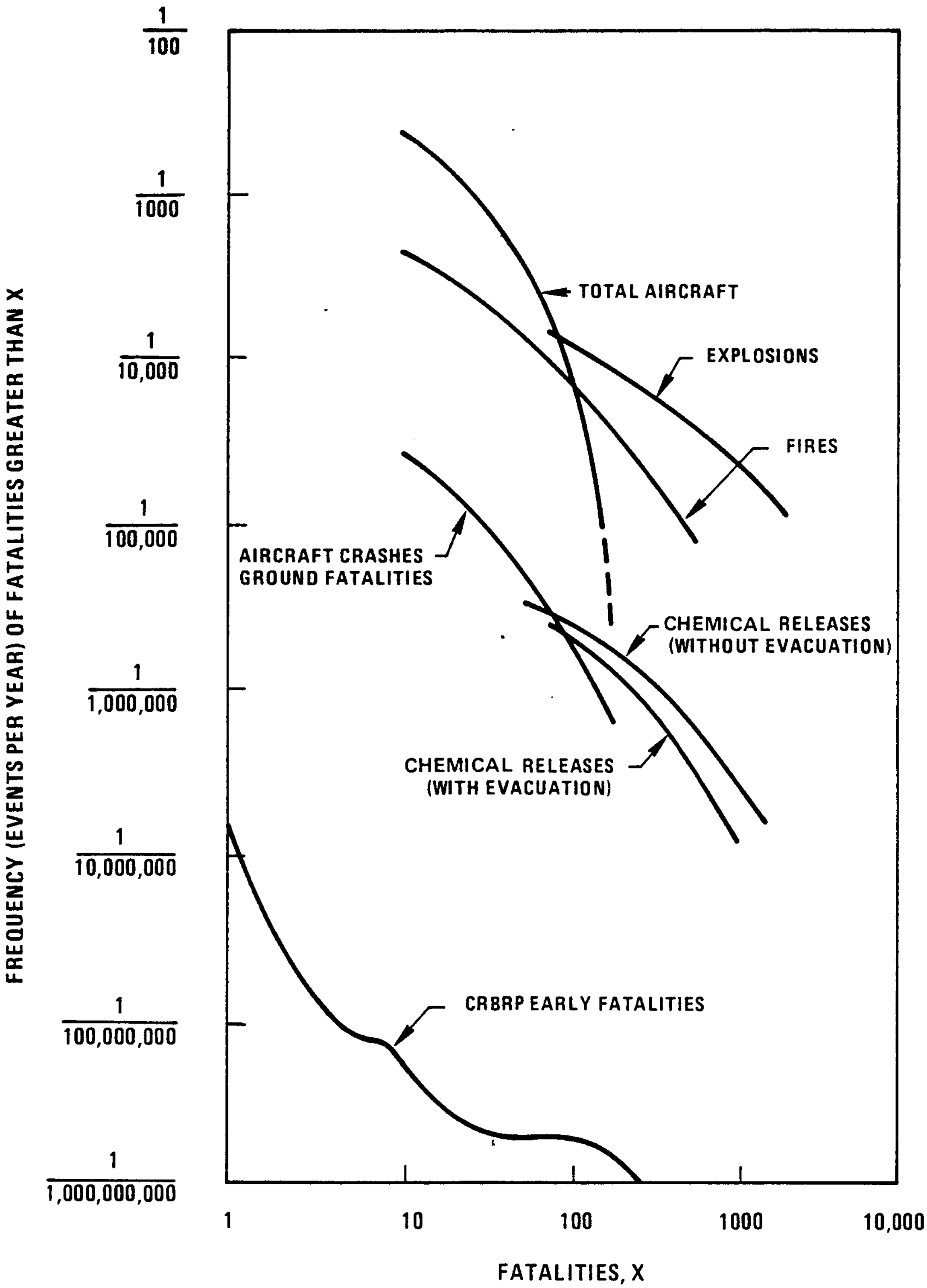

Figure 7. Frequency of Fatalities Due to Man Caused Events Occurring Within Ten Miles of the CRBRP Site.

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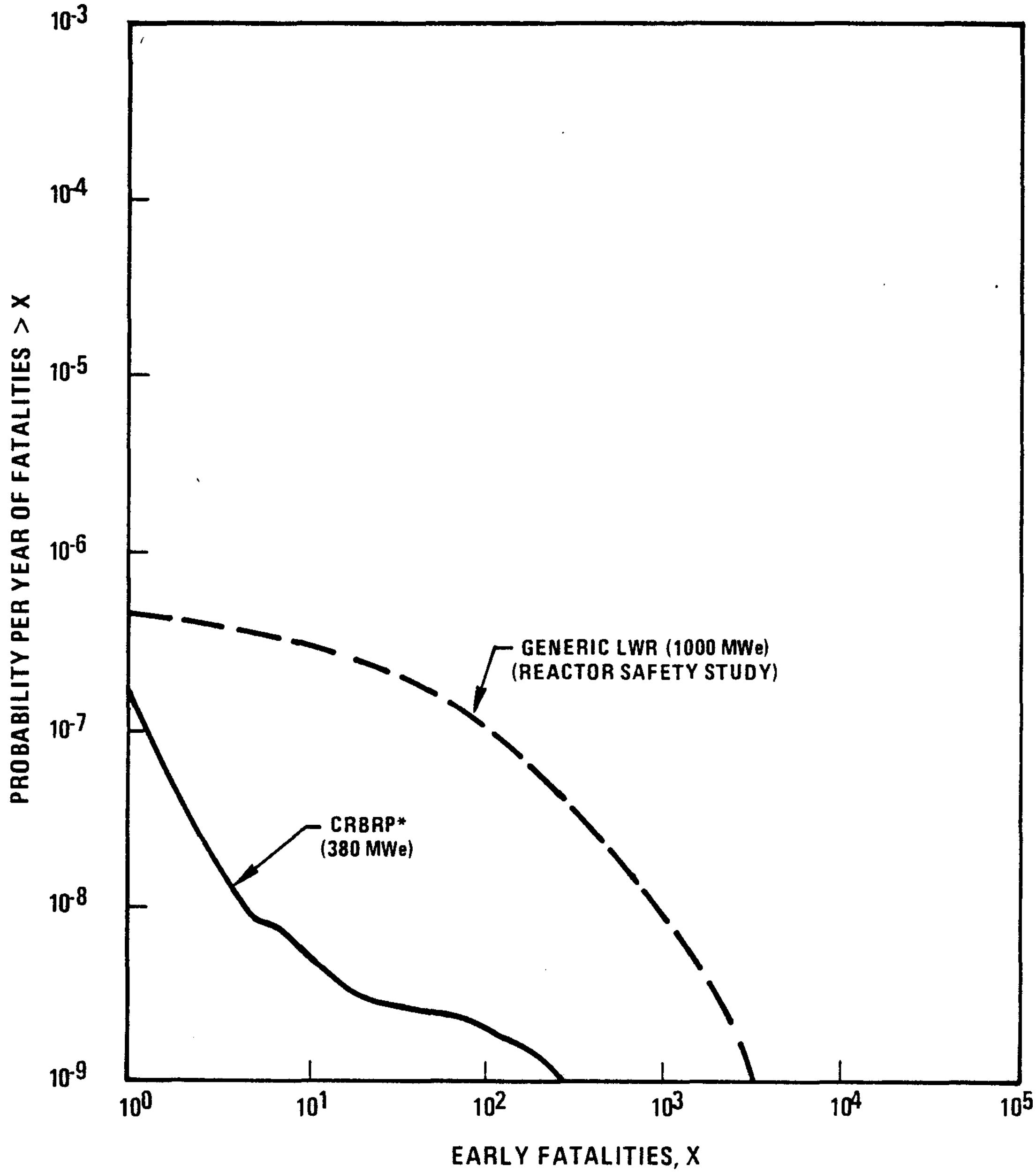

* UNCERTAINTY BOUNDS ON PROBABILITY RANGE BETWEEN
A FACTOR OF 10 AND A FACTOR OF 30

Figure 8. Comparison of Complementary Cumulative Probability Distribution for Early Fatalities due to LWR Accidents with Early Fatalities due to CRBRP Accidents

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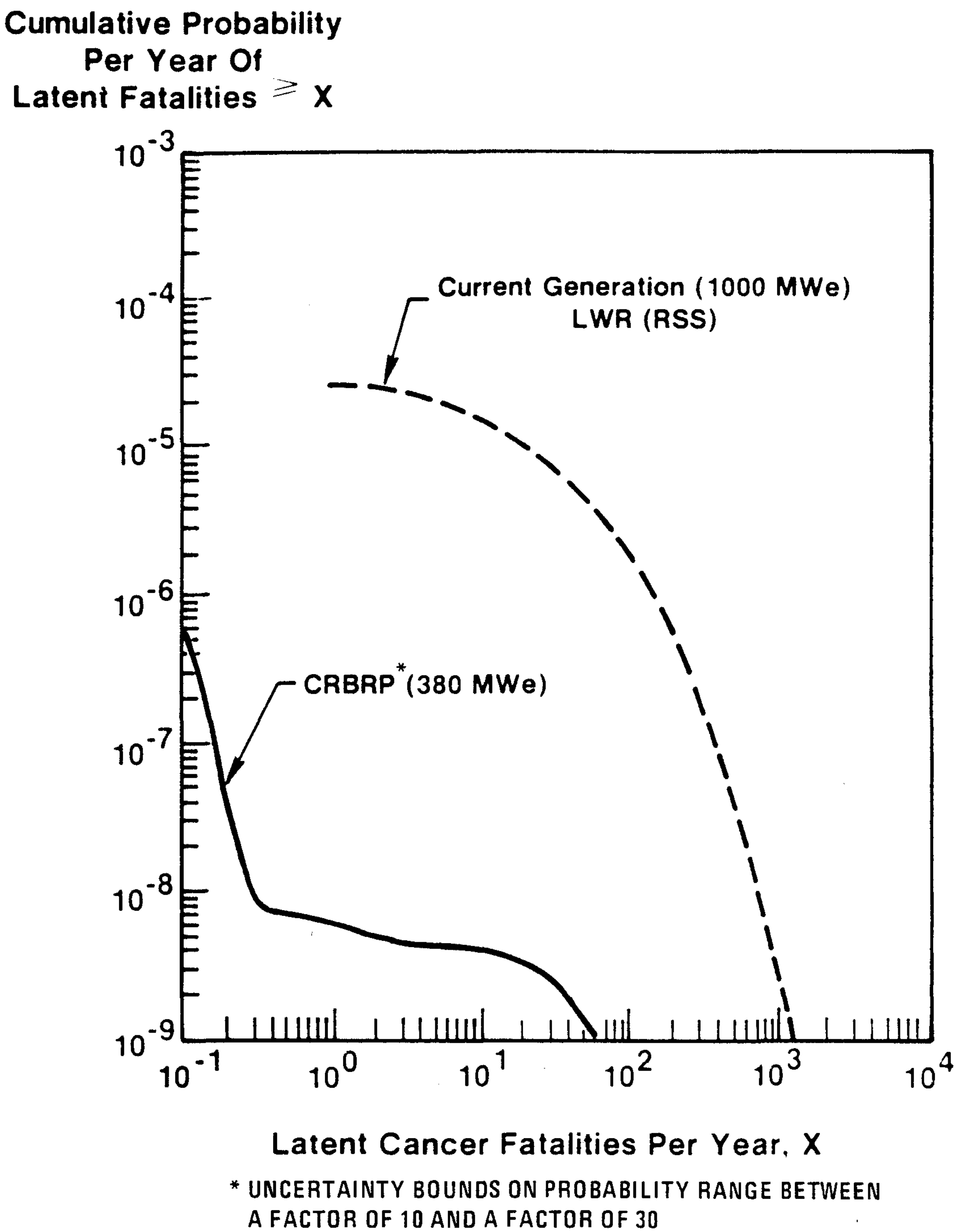

Figure 9. Cumulative Probability Distribution for Latent Cancer Fatality Incidence Per Year: CRBRP Versus LWR

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