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ESTIMATING EMERGENCY PLANNING ZONES FOR THE SHOREHAM NUCLEAR REACTOR: A REVIEW OF FOUR SAFETY ANALYSES

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Summary

This Note reviews four safety analyses submitted for the purpose of estimating emergency planning zones for the Shoreham nuclear reactor located in Suffolk County, New York. The analyses are assessed in terms of both how and why they differ in their data base and their assumptions regarding nuclear reactor design; in their supporting computer models; and in the threshold dose criteria they employ. This work, performed for the Assistant Secretary for Energy Emergencies, U.S. Department of Energy, is intended to clarify the methods and the results contained in the four safety analyses.

Introduction

The joint position of the U.S. Environmental Protection Agency (EPA) and the U.S. Nuclear Regulatory Commission (NUREG) regarding emergency response to nuclear accidents is exposed in NUREG-0396 and EPA 301/1-78-016 and was adopted as policy on October 23, 1979 (44 FR 61123). The purpose of these reports was to determine the most severe accidents for which off-site agencies should develop radiological emergency response plans.

Since NUREG-0396 was published, a major research and development program has been initiated to characterize more precisely the probability and consequences of using the results of that program. Four such studies have been applied specifically to the Shoreham nuclear power reactor in Suffolk County, New York. Those studies differ in their calculations of the appropriate radius for the emergency planning zone (EPZ). They also differ in their estimates of the amounts, probability, and consequences of radiological releases.

Drawing from a progressively increasing body of knowledge about accident initiators, sequences, and risks, we make a comparative evaluation of the differences in those studies, identifying their implications for required emergency response strategies.

The four safety analyses differ in their EPZ estimates because they differ in the data and design assumptions, the computer models, and the threshold dose

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criteria they employ. One analysis, performed by F.C. Finlayson and Associates for the consultants for Suffolk County, recommends a larger EPZ than do the other analyses. It reaches that conclusion because it does not take into account a key structural modification made in the Shoreham containment structure and because it assumes more stringent threshold dose criteria than those required by the federal government. However, if the structural modification were accounted for, even the more stringent dose criteria required by the county could be met with the smaller, 10-mile-radius EPZ recommended by the other analyses.

1. Introduction to emergency planning

The major objective of emergency planning is to provide dose savings or, in some cases, immediate life savings in the event of a release of radioactive material from a nuclear power plant. To this end, various studies have attempted to estimate a radius for an emergency planning zone (EPZ), outside which the effects of an accidental release would be minimal.

Some of the controversy and disagreement regarding the licensing of the Shoreham nuclear reactor have been sparked by differences in values estimated for the size of the EPZ, as determined by four sets of analyses. In the present review, we try to understand how and to what extent these analyses differ, and to resolve these differences, to the extent we are able. Our evaluation is based on a progressively increasing body of knowledge about accident initiators, sequences, radioactive material releases, and risks. As such, our review may be a useful background to all sides of the licensing discussions.

The remainder of this Introduction has two parts. First, we highlight several relevant political events specific to the Shoreham nuclear reactor to establish the appropriate policy setting for our present review. Second, to place these political events in the proper perspective, we review the roles of the Nuclear Regulatory Commission (NRC) and the Environmental Protection Agency (EPA) over the past several years in defining the need for the sort of calculations and analyses that we are reviewing.

1.1 The recent political arena

In February 1985, a New York Supreme Court Justice rendered a decision that Long Island Power and Lighting (LILCO) (the owner/operator of the Shoreham reactor) lacks legal authority to take certain steps necessary to implement an emergency response plan. In March 1985, a U.S. District Court Judge rejected LILCO's arguments that Suffolk County was required by the federal government to participate in the process of radiological emergency planning. LILCO has appealed both decisions.

On April 22, 1985, an Atomic Safety and Licensing Board (ASLB) of the NRC decided that LILCO's off-site radiological emergency response plan for

Shoreham is both feasible and adequate in substantially all respects. Further, it was found that LILCO has established an organization that can effectively execute its responsibilities under that plan. However, the ASLB decided that LILCO lacked the legel authority to implement the plan. LILCO appealed this decision to the NRC's Atomic Safety and Licensing Appeal Board. The State of New York and the County of Suffolk have also filed appeals.

Another ASLB had authorized LILCO to proceed with low-power testing from 0.001 percent to 5 percent of full power, but the Appeal Board remanded for further proceedings before the ASLB certain security issues relating to alternative emergency power sources that the ASLB had found to be adequate. LILCO, in addition to preparing for the remanded proceeding, asked the NRC to review this Appeal Board decision expeditiously, and the review took place in late May 1985.

On May 30, 1985, the Suffolk County Executive directed the Suffolk County Police and Planning Commissioners to participate in off-site emergency response planning. Discussions between county officials and the Federal Emergency Management Administration (FEMA) about county participation in a required federally graded exercise of LILCO's off-site emergency response plan have taken place. However, 12 members of the 18-member Suffolk County Legislature and four of the county's 10 towns - Southampton, East Hampton, Rivershead, and Southold - began Article 78 proceedings in New York Supreme Court, Suffolk County, seeking to annul the directive of the County Executive. In those proceedings, the plaintiffs as well as the Governor and the Attorney General of the State of New York have argued that the County Executive lacked the unilateral power to change the position of the county which, between February 1983 and May 30, 1985, had opposed the operation of Shoreham. On June 10, 1985, a Justice of the New York Supreme Court declared the Suffolk County Executive's Executive Order null and void and enjoined the County Executive from participating in off-site emergency response planning activities for Shoreham. The County Executive appealed that decision with the New York Supreme Court, Appellate Devision for the Second Department. Under New York law, the enforcement of the decision is stayed pending an appeal by the County Executive. Thus, despite the lower court decision, LILCO believes that the County Executive is currently empowered to order county participation pending the appellate review.

NRC regulations require on-site standby generating capability which would be used in the unlikely event that all off-site power sources failed at a time when it was necessary to shut down Shoreham. On June 14, 1985, an ASLB issued a decision allowing construction of permanent emergency power sources (i.e., the three rebuilt and retested Transamerica DeLaval, Inc. (TDI) diesel generators).

The State of New York and other opponents to the operation of Shoreham, including Suffolk County before the decision by the County Executive to participate in off-site emergency response planning, want an approved off-site emergency planning response plan to be a prerequisite to low-power testing. Consequently, they have argued that there is no need at this time to proceed with low-power testing until the off-site emergency response plan issues have been resolved. The position of Suffolk County on this issue has been rendered uncertain as a result of the dispute between the County Executive and those County Legislators and towns in the county who feel that the County Executive has no power to force participation in such response planning. Regardless of the position taken by Suffolk County, the opposition of the State of New York and others to low-power testing continues. On June 17, 1985, an appeal from the ASLB decision was filed with the Appeal Board on behalf of Suffolk County. To preserve its jurisdiction in the matter, the Appeal Board stayed the issuance of the authorization to test at up to 5 percent of full power. LILCO filed its response with the Appeal Board on June 19, 1985. The decision of the Appeal Board is subject to review by the NRC and the decision of the NRC is subject to judicial review [1].

1.2 The roles of the NRC and the EPA

The joint position of the EPA and the NRC regarding emergency response to nuclear accidents is expressed in NUREG-0396 and EPA 301/1-78-016 [2], and was adopted as policy on October 23, 1979 (44 FR 61123). The purpose of these reports was to establish the planning basis for state and local radiological emergency response. For the NRC analysis, the radioactive releases from severe accidents, as well as the frequencies of these accidents, were based on the 1975 Reactor Safety Study [3].

Since the publication of NUREG-0396, major research and development programs have been initiated to characterize more precisely the frequency and consequences of severe accidents. Numerous plant-specific safety analyses have been completed during the course of these programs. Four such studies have been applied specifically to the Shoreham nuclear power reactor in Suffolk County, New York. In particular, those four studies differ in their calculation of the appropriate radius for the EPZ.

2. The four analyses

2.1 Analysis 1

In 1978, the NRC staff, as part of its work in establishing a basis for the development of state and local radiological emergency planning, used the 1975 Reactor Safety Study to estimate the magnitude and frequency of the release of radioactive material from Shoreham [3]. Considerations drawn from this analysis support the estimate of 10 miles for the EPZ.

2.2 Analysis 2

Consultants to Suffolk County [4] performed a second analysis in the fall of 1982. They obtained their data on accident frequencies and associated radiological releases from a draft version of the Shoreham Probabilistic Risk Assessment (PRA) performed by Science Applications International (SAI) [5].[1] In addition, the County consultants adopted more stringent threshold dose criteria for predicting acute mortality than did SAI. Using this version of the PRA, the County consultants estimated an EPZ radius of 20 miles.

2.3 Analysis 3

SAI performed a third analysis as part of the final version of the Shoreham PRA in 1985 to incorporate significant design changes made in the Shoreham facility and to account for the improved understanding of radioactive material behavior following an accident. This third analysis, performed for LILCO, led to release magnitudes lower than in previous estimates [6]. This analysis estimated an EPZ radius of 10 miles.

2.4 Analysis 4

A fourth analysis, also performed in support of the Shoreham PRA, was conducted by Morton and Potter in 1985 [7]. This was the only one to use the data base from experiments performed since the 1970s and to employ the methodology developed in the IDCOR program [8], which includes a better understanding of mechanisms than was possible at the time of the earlier analyses. The results of the fourth analysis are consistent with those in the first and third studies – the EPZ radius is given as 10 miles or less.

These studies differ in three basic ways:

1. The data base and reactor design assumptions used to estimate radiological releases and the probabilities of those releases under different accident conditions;
2. The primary computer program used to estimate the consequences of a radioactive release (various versions of the CRAC[2] computer code); and
3. The threshold dose criteria used to approximate the EPZ, once the magnitude of release and its probability are estimated.

3. Design assumptions regarding the Mark II containment

3.1 Introduction

The Mark II containment was developed by the General Electric Company for Boiling Water Reactors (BWRs) as a replacement for the Mark I contain-

[1]The Shoreham design assumed by the County consultants was extracted from this draft study, which was eventually finalized as Ref. 6.

[2]Calculation of Reactor Accident Consequence (see Section 4).

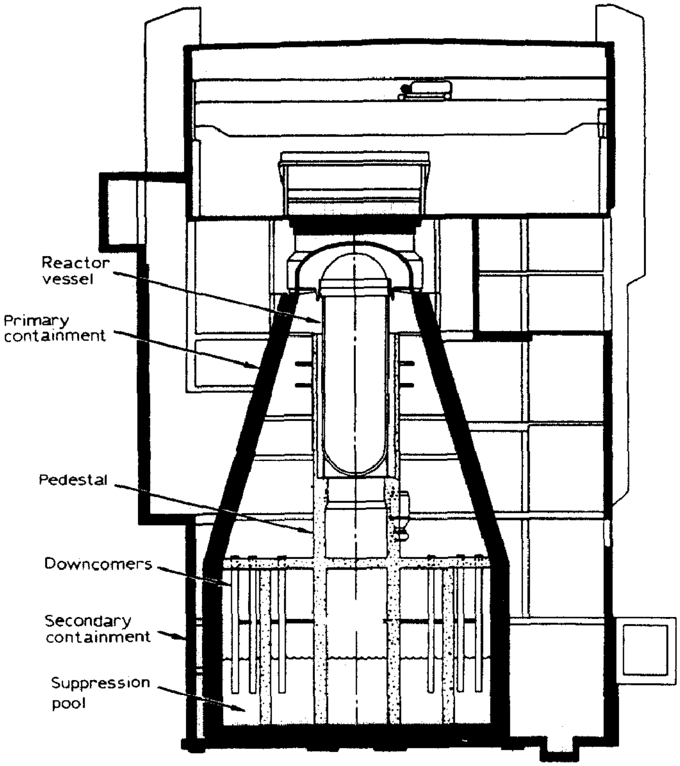

Fig. 1. Mark II Primary and Secondary Containments (source Ref. 10).

ment design. Approximately 10 plants in the United States, in various stages of construction, licensing, and operation, fall into this category. The Mark II has an over-under pressure-suppression pool whose general configuration is shown in Fig. 1.

The suppression pool, or wetwell, is located below the truncated cone-shaped drywell containing the reactor vessel and is designed to accommodate safety relief valve (SRV) actuation after a turbine trip. Suppression pool water is used to condense the steam released during a transient of this type, and in doing so, prevents an unacceptably large pressure in the containment. In addition to accommodating SRV actuation, a number of downcomers (see Fig. 1) control pressure in the event of a steam line pipe break in the drywell portion of the containment.

In this section, the unique features of the Shoreham Mark II containment are discussed, as well as the recent design modification which plays a major role in the different assessments of the EPZ.

3.2 General description

As shown in Fig. 2, the primary containment of the Mark II plant is made up of the drywell and wetwell portions. The reactor vessel, recirculation pumps,

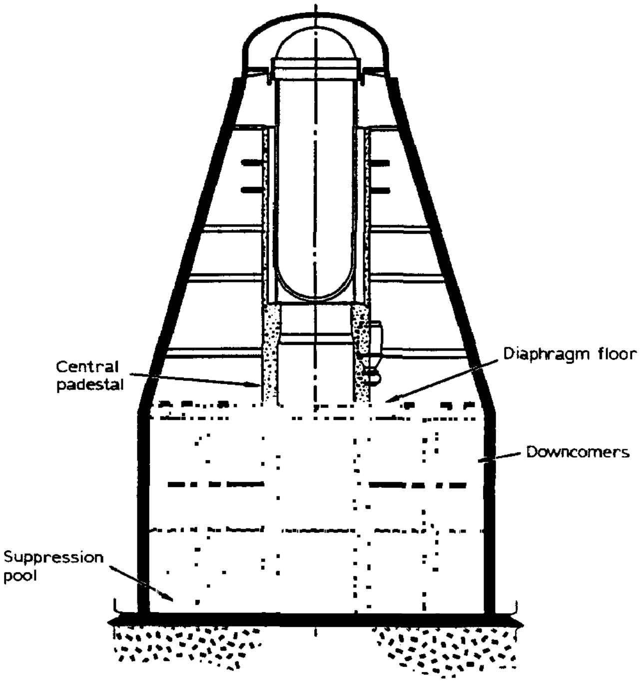

Fig. 2. Typical Mark II Containment (source Ref. 10).

emergency core cooling equipment, and other safety features are located within the upper drywell region, and the suppression pool is located below in the wetwell.

The suppression pool is designed to accommodate "design basis accidents" – accidents up to, but not including, degraded cores or core melt. Steam released during a design basis accident to the drywell is conveyed into the suppression pool by multiple vertical steel downcomer pipes. The downcomer pipes penetrate the diaphragm floor separating the drywell and wetwell. The pool functions to condense steam and reduce primary containment pressure and temperature accordingly. It is also a water reservoir that can be tapped for emergency core cooling when necessary.

The reactor vessel is supported on a concrete pedestal extending down to the concrete basemat of the primary containment, and the diaphragm floor passes through the pedestal. Of particular interest here is the unique configuration of the pedestal for the Shoreham plant.

As shown in Fig. 3, the Shoreham plant has several downcomers in the pedestal region. This design feature has profound implications for core melt accidents and in determining the EPZ.

3.3 Behavior of the Mark II in core melt accidents

In addition to coping with the pressure generated during a design basis accident, the Mark II suppression pool mitigates the effects of accidents beyond

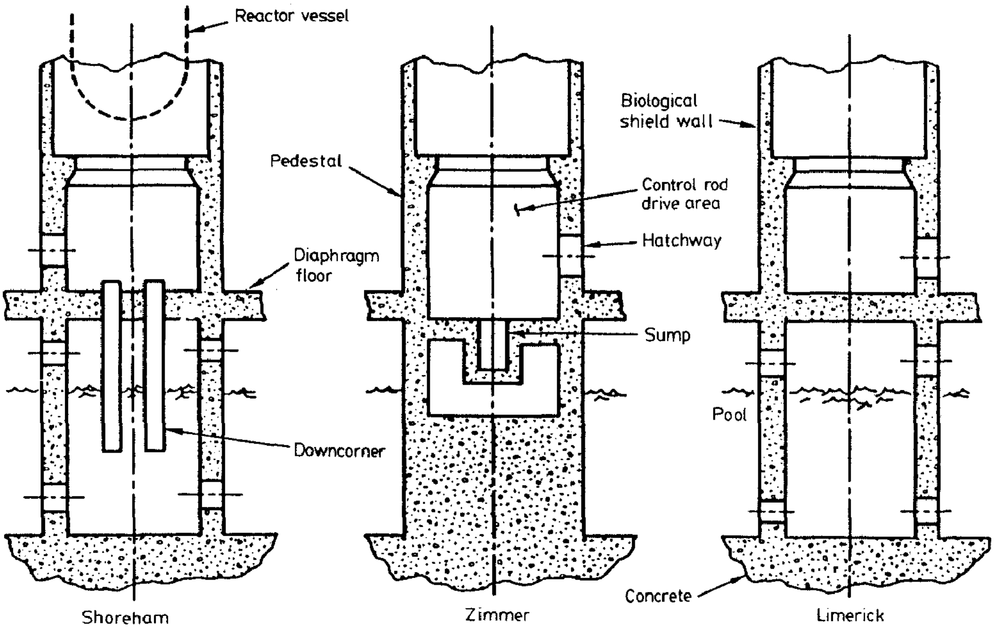

Fig. 3. Variations in the Mark II Pedestal Configuration (source Ref. 10).

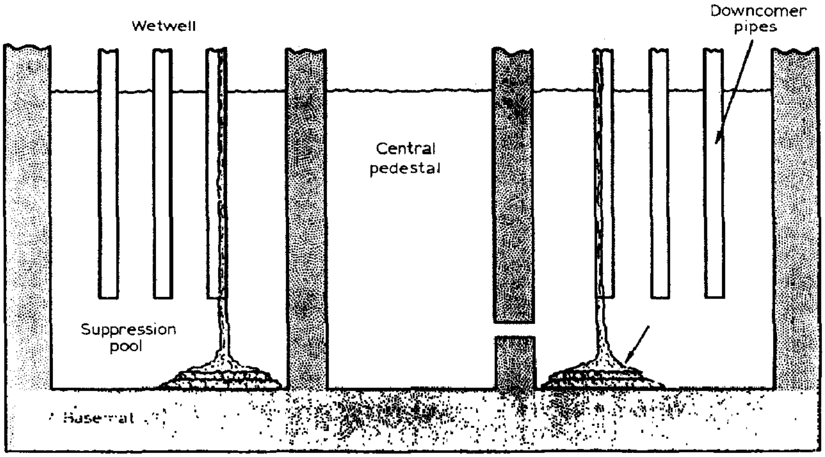

Fig. 4. Core Debris on Basemat Below Downcomer Ducts (source Ref. 10).

the design basis (i.e., degraded core or core melt accidents). Analyses [8] have shown that:

1. Fission products can be scrubbed from steam and other gases bubbling through the suppression pool during serious accidents.

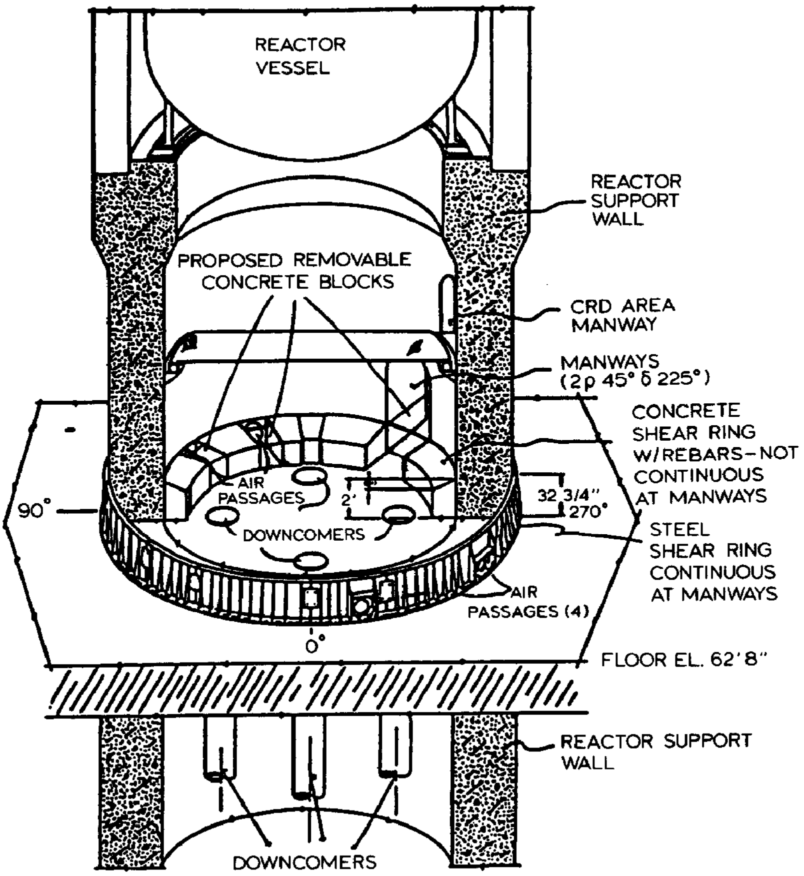

Fig. 5. Control Rod Drive Room Pedestal Shear Ring Detail (source Ref. 10).

2. The reduction in containment pressure by condensation action lowers the leakage rate through the containment.
3. The suppression pool serves as a large heat sink. Hence, any molten fuel that escapes from the vessel and finds its way to the pool (e.g., through the downcomers) will be cooled (see Fig. 4).

The effect is that the presence of the suppression pool significantly delays containment failure and reduces the quantity of fission products available for release, when failure eventually takes place.

3.4 Design features unique to Shoreham

One unique design feature incorporates four downcomer pipes, which lead to the suppression pool in the floor immediately beneath the reactor vessel (i.e., in the pedestal region). These four downcomer pipes provide direct access to the suppression pool from the diaphragm floor directly below the vessel.

In some severe accidents involving core melt and thermal failure of the bottom head of the reactor vessel, the molten core debris would land on the diaphragm floor. Some fraction would flow through the four downcomers into the pool and would be quenched and remain cool. The rest would flow through the openings (doorways or windows) and spread on the diaphragm floor (see Fig. 3, Shoreham diagram, and Fig. 4). Again, some would flow to the pool through other downcomers; the rest would cause thermal attack of concrete and produce noncondensible gas whose pressure ultimately could not be contained. In the draft PRA [5] for the original Shoreham plant, it was determined that about 90 percent of the molten debris would spread to the outer pedestal floor through the open doors and windows, and only 10 percent would drain to the pool and be quenched. Consequently, the containment was predicted to reach its failure pressure about an hour after the initiation of core-concrete interaction [5]. This formed the basis for the Suffolk County consultants' results (Analysis 2).

The final PRA [6] for the Shoreham plant is based on a significant design modification in the pedestal region. A concrete and steel barrier, which acts as a confinement ring to prevent horizontal flow of core debris, has been incorporated as shown in Fig. 5. In this case, the barrier enhances the flow through the four downcomer pipes in the pedestal region, thus delaying core-concrete attack and containment failure. Using a prediction of 90 percent flow of the molten core debris directly into the suppression pool, containment integrity is estimated to be maintained for about 24 hours [9].

The smaller radionuclide release in the final PRA can then be attributed to two factors [9]:

1. The suppression pool provides a heat sink for a major portion of the molten core debris, preventing further fission product release from the fuel,
2. Containment integrity is maintained for a longer period of time allowing natural removal processes (scrubbing, settling, etc.) to deplete airborne fission products before its ultimate failure.

The new design feature forms a basis for the third and fourth analyses.

4. Differences in computer programs

The computer model most often used in probabilistic consequence assessment is CRAC. Three versions of CRAC were used for the four Shoreham safety analyses.

- Analysis 1 used CRAC;
- Analysis 2 used CRAC2;
- Analysis 3 used CRACIT;
- Analysis 4 used both CRACIT and CRAC2.

The CRAC code, developed by the Atomic Energy Commission for use in the Reactor Safety Study, was the first program used to perform a comprehensive probabilistic assessment of consequences from a severe reactor accident. It treats the effects of plume rise, wet and dry deposition, and changes in meteorological conditions (except wind direction). It also simulates the effects of evacuation and other mitigative measures, and it models dose and health effects from both acute and chronic phases of exposure. The CRAC code was intended for use on composite sites. The CRAC2 code refined some of the calculated models in CRAC and is intended for application to specific nuclear reactor sites throughout the United States. CRAC2 employs a straight line Gaussian plume model to represent the transport and dispersion of radionuclides released in reactor accidents. The model allows for changes in weather but assumes that wind direction remains constant.

The CRACIT program was developed by Pickard, Lowe, and Garrick [6] to overcome perceived deficiencies in the CRAC code. CRACIT incorporates features such as variable-direction wind trajectories and variable-direction evacuation trajectories to simulate site conditions more realistically. Another important improvement enables simulation of multiple plume trajectories as a result of changes in meteorological conditions during the course of long duration releases. Other improvements include more realistic modeling of plume characteristics during conditions such as inversions and turbulent internal boundary layers.

As mentioned above, CRAC2 is a more site-specific version of CRAC but less site-specific than CRACIT.

5. Differences in threshold dose criteria

5.1 Introduction

In all four analyses being discussed, the emphasis is on defining an EPZ based on:

1. The results of a PRA for Shoreham, and
2. Determining some radius outside which the effects of the accident would be minimal.

As discussed in the previous sections the two main differences in the four analyses with respect to the PRA are: (1) the inclusion (or omission) of a major containment design change, and (2) the computer code used to calculate consequences. In this section we discuss the third main difference – the threshold dose criteria, which establish a radius for the results of a given PRA.

5.2 Results of a PRA

A PRA is a mathematical formalism that combines the frequency of accidents and their consequences to yield various measures of risk. These measures include the expected value of core melt frequency, early and latent (cancer) deaths, injuries, and property damage (in dollars or acres lost). Of particular interest to emergency planning is the radiation dose (measured in rem) to an individual as a function of distance for a given accident. On the other hand, emergency planning is not based upon a single accident sequence but upon a spectrum of accident sequences ranging from low frequency/high consequence events to relatively high frequency/low consequence events. Of particular interest then are the results of the PRA that show the frequency with which various doses occur as a function of distance. The specification of a threshold dose criteria could include the required radius, if some determination of frequency (or probability) is made.

5.3 Differences in assumptions

The four analyses used three different sets of threshold dose criteria:

- Analysis 1 used the criteria NUREG-0396 [2].
- Analysis 2 reflected the more stringent criteria of the County consultants [4].
- Analyses 3 and 4 used the criteria in NUREG-0654 [10].

The objective of the two NUREGs is to prevent doses in excess of Protection Action Guides (PAGs). The PAGs are defined as incipient projected radioactive doses at which protective actions should be taken to safeguard potentially exposed members of the public. That is, protective actions would be triggered by the anticipation of reaching PAG dose levels.

For the general population, the EPA recommended that protective actions should be implemented when projected whole-body doses of 1 to 5 rem or 5 to 25 rem to the thyroid were foreseen to the public. The NUREGs specify the following [7]:

1. Projected doses from traditional design basis accidents should not exceed PAG levels outside the zone.
2. Projected doses from most core melt sequences should not exceed PAG levels outside the zone.
3. For the worst core melt sequences (early containment failure or adverse meteorological conditions, immediate life-threatening doses would generally not occur outside the zone.
4. Detailed planning within the EPZ would provide a substantial base for expansion of response efforts in the event that this proved necessary.

The Suffolk County consultants established four criteria for the EPZ [4]:

1. The probability of exceeding 30 rem to the whole body beyond the EPZ, given a core melt accident, is less than 0.01.
2. The probability of exceeding a whole body dose of 200 rem beyond the EPZ, given a core melt accident is "negligible."
3. Beyond the EPZ, if protective action was required at all, shelter would be adequate to assure that the goal of limiting doses to approximately PAG levels would be achieved.
4. The probability of exceeding 200 rem beyond the EPZ, given a "worst case" release, is less than 0.01.

TABLE 1

Distance beyond which criteria are satisfied for four safety analyses (in miles)

Criteria	Safety analyses				
	Analysis 1 NUREG-0396[a]	Analysis 2 Suffolk County Consultant[b]	Analysis 3 SAI/PLG Final Study[c]	Analysis 4A Potter Study[c]	Analysis 4B Potter Study[b]
NUREG-0396:					
1. Prob. of dose exceeding high PAG is approx. 0 (given design basis accident)	10	—	<1	<1	<1
2A. Prob. of dose exceeding low PAG is less than 0.3 (given core melt)	10	40	2	10	12
2B. Prob. of dose exceeding high PAG is less than 0.03 (given core melt)	7	30	1	6	7
3. Prob. of dose exceeding 200 rem is less than 0.03 (given core melt)	10	<1	1	<1	2
NUREG-0654:					
1. Prob. of dose exceeding low PAG is approx. 0 (given design basis accident)	?	—	<1	<1	<1
2A. Prob. of dose exceeding low PAG is less than 0.5 (given core melt)	5	33	1	3	7[d]
2B. Prob. of dose exceeding high PAG is less than 0.5 (given core melt)	1	25	<1	2	5
3. Prob. of dose exceeding 300 rem is approx. 0 (given worst release)	10	20	10	10	10
COUNTY CONSULTANTS:					
1. Prob. of dose exceeding 30 rem is less than 0.01 (given core melt)	>40	20	10	10	5
2A. Prob. of dose exceeding 200 rem is less than 0.001 (given core melt)	17	20	5	2	2
2B. Prob. of dose exceeding high PAG is low (given core melt)	>20	20	<10	<10	<10
3. Prob. of dose exceeding 200 rem is less than 0.01 (given worst release)	17	20	6	6	6

Sources: The information given here in its present form is adapted primarily from Tables 1, 2, and 3, and read from Figs. 1 through 9 of Ref. 7. Information contained in columns 1, 2, and 3 was further checked against Refs. 2, 4, and 6, respectively.

[a]Using CRAC code. [b]Using CRAC2 code. [c]Using CRACIT code. [d]Based on Final SAI/PLG Report - 27 percent of core melts result in no containment failure.

In summary, the objective of the two NUREGs is to prevent off-site doses in excess of PAGs. The primary objective of the Suffolk County consultants' criteria is to minimize the probability of public exposures leading to early mortalities. The consultants use a 200 rem maximum of acceptable dose, which is less than one-half the LD_{50} [1] commonly used by the industry. At distances beyond the zone in which lives could be immediately threatened, the consultants' planning objective is to minimize the probability of exposures leading either to early injuries or to substantial increases in the probability of latent cancer induction. They use 30 rem for this latter threshold.

The effects of these differences are summarized in Table 1 [2, 4, 6, 7]. It displays the EPZ radius that would result for each combination of criteria and analytical approach.[2] The EPZ is especially sensitive to the criteria selected. Those selected by the County consultants result in EPZs much larger than those assumed by either NUREG. (Of course, these EPZ discrepancies do not imply discrepancies in estimates of accident risk at the plant itself.) We conclude from Table 1 that the County consultants' criteria are significantly more stringent than those in the two NUREGs.

6. Summary of findings and conclusions

6.1 Design assumptions

Analysis 2 (by the County consultants) does not reflect a significant design change in Shoreham's pedestal well that was made *after* the completion of the draft SAI study on which that analysis was based. To understand the implications of this change, some background on Shoreham's containment is important. Shoreham's containment is unique among Mark II containments in that the downcomer pipes leading to the suppression pool are in the pedestal floor directly below the reactor vessel. In a severe accident at Shoreham involving core melting and the breaching of the reactor vessel bottom, some of the molten core would flow through the pipes to the pool, be cooled, and remain cool, retaining most of the fission products not yet released from the fuel. Only this portion remaining in the pedestal floor would thermally attack the concrete, producing noncondensible gases that would thermally attack the concrete, producing noncondensible gases that could threaten the integrity of the containment vessel. However, before the design change, there was another avenue of escape for the core: About 90 percent of the core debris could have spread to the outer pedestal floor through open passageways in the wall, instead of

[1] LD_{50} is the dose at which 50 percent of the exposed population would suffer early fatality. It is typically assumed to be 450 to 500 rem. According to NUREG-0396 [2], 200 rem is the dose at which significant early injury starts to occur.

[2] For comparative purposes, each approach is applied to all criteria, not just to those assumed in the analyses in which that approach was used.

running into the suppression pool. Consequently, in this earlier design, the containment vessel was predicted to reach failure pressure about an hour after the molten core reached the concrete.

The design change entailed adding concrete and steel barriers around the pedestal wall. In Analysis 3, which took account of that change, it was estimated that 27 percent of the core melt sequences would result in no containment failure. When failure does occur, the added design features would increase the failure time from one hour to about twenty.

The analysis of the integrity of the containment vessel and the fission product release and transport analysis are documented in detail in Appendices C and D of Safety Analysis 3 [6]. These results suggest that a lower radionuclide release and more time to take emergency measures to protect the public should be expected from the modified Shoreham design than from the original design, for the two reasons described above.

6.2 Computer programs

Analysis 2 used the CRAC2 code. As shown in the last two columns of Table 1, if the CRAC2 dose is applied instead of CRACIT to an analysis based on assumptions that are otherwise the same, the EPZs tend to be somewhat greater. However, the effect does not appear to be large enough to account for the difference in EPZs between Analysis 2 and the others.

6.3 Threshold dose criteria

As noted above, the criteria used by the County consultants are more stringent than those used in the other analyses. However, as Table 1 shows, even applying the consultants' criteria to the results of Analyses 3 or 4 produces an EPZ of less than 10 moles. The varying dose criteria thus do not appear to be a critical factor.

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