CONTRACTOR (Science Applications Inc) INPUT TO CLASS 9 ACCIDENTS ANALYSIS. From: Ed. Rumble, SAI A-4

DRAFT

Received approximately in the 2nd Week of June, 1982.

Probabilities of Severe Accidents

The probabilities of severe (Class 9) accidents at CRBR involving core disruption and containment failures are related to three phases of such accidents. First, initiation of core disruption must be considered, and this typically requires simultaneous failures of redundant safety systems to function as required. Secondly, there are variations in the release to containment which are dependent on the energetics associated with core disruption and the nature of the response of the primary coolant boundary. Finally, the mode of containment failure must be considered. The probabilities of these variations are discussed below.

Initiators of Core Disruptive Accidents

Core disruption could be initiated by: (1) failure to adequately cool the fuel as exemplified by a loss of heat sink (LOHS), loss of coolant accident (LOCA), or massive flow blockage; (2) failure to terminate the fission chain reactions, as exemplified by a failure to scram during a loss of flow event (ULOF) or a transient overpower event (UTOP); (3) core-wide fuel failures as exemplified by propagation of local fuel faults (FFP).

As discussed on page 7-2 and 7-7 of the FES, accident prevention requirements will be imposed on the CRBR design to assure that initiation of core disruptive accidents is very improbable. Consequently such accidents are not included in the CRBR design basis accident spectrum.

LOHS events at CRBR would have to involve simultaneous loss of availability of the main condenser-feedwater train, of all three trains of the steam generator-auxiliary heat removal system (SGAHRS) and of both trains of the direct heat removal system (DHRS). The CRBR SGAHRS system, which is similar in many respects to the steam generator-auxiliary feedwater systems included in PWR designs, consists of one steam driven and two electrically driven auxiliary feedwater trains. The DHRS employs a diverse heat removal concept. Although the staff review of these systems is not complete it is our judgement that there is sufficient inherent redundancy, diversity, and independence in the SGAHRS and DHRS systems to achieve an unavailability on demand of less than 10^{-4} per reactor year. This estimate is based on a general consideration of typical achievable PWR auxiliary feedwater system reliabilities, potential for common cause failures, and the potential for achieving high reliability in final design and operation through an effective reliability program. A significant contributor to the LOHS probability for CRBR would be from simultaneous loss of offsite and onsite AC electrical power and the steam driven auxiliary feedwater train.

Because of the high boiling point of sodium the CRBR primary coolant system may operate at significantly lower pressures than LWR primary coolant systems. This reduces the possibility of large ruptures in the primary coolant system. To further assure that large breaks cannot occur and cause core damage, implementation of pre-and in-service inspection of the primary coolant boundary, and a leak detection system will be required. In addition a guard vessel will be included to prevent unacceptable leakage from large portions of the primary coolant system. For these reasons LOCAs are not considered credible events at CRBR. The probability assumed for LOHS adequately bounds the LOCA contributions to core disruption probability.

The design approach being utilized in the coolant inlet region of the CRBRP core will prevent large sudden flow blockage, such as that which led to extensive damage to two subassemblies in the Enrico Fermi Reactor. Multiple inlet ports at different planes, with interposed strainers will prevent large pieces of debris from significantly reducing coolant flow to a subassembly module. Although sources of particulate debris in sufficient quantity to produce significant flow blockage have not been mechanistically identified, it may be postulated that this might occur. Such debris would be expected to be distributed rather generally throughout a large region of the core, so that it would be detectable by the core outlet thermocouples if it significantly reduced core flow. The probability assumed for LOHS adequately bounds the flow blockage contribution to core disruption probability.

UTOP and ULOF events involve simultaneous failure of both of the reactor shutdown systems. Each of these systems will be required to meet the high standards normally applied to LWR shutdown systems. For example, as specified by IEEE Standard 279, each shutdown system will be automatically initiated, will meet the single failure criterion and will be tested regularly. Each system consists of three independent electrical actuation channels of diverse logic and diverse components. The mechanical portions of the two systems are based on diverse mechanisms and materials. Although the staff review of these systems is not complete it is our judgement that there is sufficient inherent redundancy, diversity and independence in the overall shutdown system designs to achieve an unavailability on demand of less than 10^{-5} per reactor year. This estimate is based on a general consideration of LWR shutdown system unavailability rates, ATWS precursors, potential for common cause failures, and the feasibility of implementing an effective reliability program to achieve high reliability in the final design and in operation. By factoring in a conservative assumption that an average of ten transient overpower or loss of flow events combined might occur per year of operation over the life of the plant, we arrive at the conclusion that the combined probability of core melt accidents initiated by ULOF and UTOP events is less than 10^{-4} per reactor year.

The CRBR fuel design will be required to have an inherent capability to prevent rapid propagation of fuel failure from local faults. Systems to detect more slowly developing faults will also be required. Each of these features is considered feasible and in fact have been achieved on similar fuel designs to that of CRBR. Therefore, the probability of fuel failure propagation is considered very remote. The probabilities attributed to LOHS, UTOP, and ULOF events adequately bound the contribution to core disruption probability from fuel failure propagation.

In summary the probabilities of core disruption from LOHS, UTOP, ULOF, LOCA and FFP are all considered to be less than 10^{-4} per reactor year. Even when combined, these events and variations on them are estimated to have a net probability no greater than 10^{-4} per reactor year.

This net probability does not reflect the variations in response of the primary coolant system which might be associated with the various initiators. Some initiators may result in more severe response than others. This is taken into account as follows.

Response of the Primary Coolant System

The response of the primary coolant system to core disruption depends on the amount of energy associated with the disruption. Four categories have been identified and are listed here in order of increasing potential threat to containment integrity and increasing release of radio isotopes into containment:

I. Primary system remains intact; no major release of radioactive materials.

II. Primary system initially intact but ultimately fails due to ineffective long term decay heat removal (of the order of hours or more). Core debris and sodium are initially released into the reactor cavity but eventually reach outer containment through the reactor cavity vents at a slow rate relative to the initial releases of III and IV below.

III. Primary system seals fail due to excessive mechanical and/or thermal loads. Some sodium fuel vapor and fission products are expelled into the head access area. As discussed in the FES Class 9 accident description 1% of core Pu and solid fission products, and 100% of noble gases and 10% of volatiles would be released into upper containment immediately.*

IV. Primary system fails due to excessive mechanical loads. Outlet piping (three loops) fails and sodium is expelled into the reactor guard vessel. Substantial quantities of fuel, sodium or sodium vapor and fission products are released to the outer containment. Initial failure of the containment due to ~~these effects~~ Spray fires or missiles is ~~possible~~ likely. 10% of core Pu and solid fission products and 100% of

*See next page

may change

noble gases and volatiles would be released to upper containment immediately.*

Most core disruptive accidents are expected to be non-energetic and to culminate in effects such as Categories I and II above.

The applicants have proposed to incorporate features to mitigate the above behavior indicated in Categories II and III to reduce the probability of subsequent failure of the containment. These include a filtered vent system to relieve containment pressure, a containment purge system to reduce the potential for hydrogen explosions, fans to cool the annulus between the steel containment shell and the confinement structure, and vents to relieve pressure from gasses generated behind the reactor cavity cell liners. These provisions are currently under review by the staff.

If the provisions proposed by the applicants are determined to be inadequate these may be upgraded, but the staff is also aware of other feasible design features which separately or in combination could reduce the probability of containment failure to an acceptable level. These include a cooling system to transfer decay and reactant heat from the core debris and sodium deposited in the reactor cavity to outside containment and installation of protective materials in the reactor cavity to reduce the production of reactants and heat from interaction of sodium with concrete.

The Class 9 accident release described in the FES corresponds to Category III. The staff considered such an energetic release because the staff had

*Note: Longer term release to containment via the reactor cavity and vents would be as in II.

determined that for the old CRBR homogeneous core an energetic core disassembly could not be precluded. The staff is reviewing the new heterogeneous core to determine the magnitude of energy release anticipated for that design. If the conclusion of this review is that an energy release beyond primary system capability cannot be precluded, the staff will require that the vessel be strengthened or that head restraints and sodium spray deflectors be installed to eliminate the possibility of early containment failure from missiles or spray fires. The staff believes that the technology exists to design and build such devices; similar devices and/or measures were utilized in the design of the FERMI reactor, as well as in Atomics International's design studies of the 500 MWe LMFBR demonstration plant.

On this basis the probability that energetic core disruption could lead to immediate containment failure due to missiles or spray fires as outlined in event Category IV is considered very small compared with the probability of Categories I and II or even that of Category III.

Assuming that a core disruptive accident occurs the conditional probabilities of event Categories I through IV subsequently occuring are estimated as follows:

Primary System Failure and Category I & II combined; ∿.9

Primary System Failure and Category III; ∿.10

Primary System Failure and Category IV; ∿~~.0001~~ → 10^{-2} — 10^{-6}

These estimated are based on the relative probabilities of increasing energetics and the feasibility of incorporating enhanced primary coolant system designs.

Response of Containment

For the purposes of estimating risk given the threats to containment identified above the following three containment failure categories are identified:

(A) Failure of Containment Annulus Cooling or Vent/Purge Systems.

(B) Failure of Containment to Isolate.

(C) Failure of Containment From Missiles or Spray Fires.

The containment annulus cooling and vent/purge systems will be designed with sufficient redundancy and quality, and will be tested and inspected during operation with sufficient frequency that we conclude that their unavailability will not exceed 10^{-2} per demand. Such systems will not be needed until many hours after initiation of a CDA, and would not be expected to be affected by loss of offsite and emergency onsite power unless such power loss should be a long term outage.

Containment isolation is an engineered safety feature at CRBR. Such systems are designed to high quality standards and with redundancy. An unavailability rate of 10^{-3} per demand is feasible for such systems and is expected to be attained at CRBR given that implementation of an adequate reliability program will be required.

The probability that core disruption will result in sufficient energy to generate a missile or spray fire has been evaluated in the previous section. It is estimated that ~~even~~ if missiles or spray fires should be generated ~~only 10% of such situations would result in~~ containment failure. would occur.

In summary, the conditional probabilities of containment failure for the containment failure categories is as follows:

Containment Failure Category A (Mitigating System Failure); $\sim 10^{-2}$

Containment Failure Category B (Containment Isolation Failure); $\sim 10^{-3}$

Containment Failure Category C (Missile/Spray Fire Induced Failure); $\sim$ ~~10^{-1}~~ 1

Comparison of Accident Sequence Probabilities

The most probable accident sequence for which early or latent fatalities would be possible is the initiation of a core disruption accident (less than 10^{-4} per reactor year), primary system failure of Category I, II or III, (combined conditional probability ~ 1), and containment failure category A, containment annulus cooling or vent/purge failure at approximately 24 hours ($\sim 10^{-2}$ per demand). This sequence which corresponds to the FES Class 9 accident, would therefore have an estimated probability less than 10^{-6} per reactor year.

A less probable sequence would be initiation of a CDA (10^{-4}/year) primary system failure Category III (10^{-1}), and containment failure Category B, failure to isolate, (10^{-3}/demand), with a combined probability of $\sim 10^{-8}$ per reactor year.

Even less likely would be a CDA involving ~~(10-4/year) wit~~h primary system failure Category IV (~~10-4~~) and containment failure Category C. (~~10-1~~) ~~, and combine~~d proba~~bility of ~ 10-9 per reactor year~~. The probability of such a sequence is believed to be very low but the uncertainties are large.

These sequences correspond to releases to the environment of three different magnitudes or times, and are the most probable sequences for each release type. Other sequences would be of smaller probability or smaller release to the environment. This conclusion is based in part on the fact that all containment failure modes are not to be combined with all primary system failure modes. Primary system failure Categories I through III would not result in containment failure Category C, for example.

These accident sequences and their releases to containment are summarized in Table ? . The first entry in the table, which is the FES Class 9 accident, is more probable than the other entries by factors of 100 and 1000 respectively. Although these sequences would involve earlier releases or earlier and larger releases respectively than the FES Class 9 accident it is not expected that they would involve risks (product of probability and consequences) significantly greater from the FES Class 9 accident risk.

TABLE

DOMINANT CRBR SEVERE ACCIDENT SEQUENCES

INITIATION	PRIMARY SYSTEM FAILURE CATEGORY	CONTAINMENT FAILURE CATEGORY	ESTIMATED RELEASE TO CONTAINMENT	PROBABILITY
Generic Core Disruption (10^{-4}/Years)	I + II + III (~ 1)	A. (10^{-2}/Demand) at 24 Hours	1% Pu 1% Solid Fission Products 100% Noble Gases 10% Volatiles	10^{-6}
Generic Core Disruption (10^{-4}/Years)	III ($\sim 10^{-1}$)	B. (10^{-3}/Demand) at 0 Hours	"	10^{-8}
Generic Core Disruption (10^{-4}/Years)	IV ~~($\sim 10^{-4}$)~~ (10^{-2} – 10^{-6})	C. (~~10^{-1}~~ 1 /Demand) at 0 Hours	10% Pu 10% Solid Fission Products 100% Noble Gases 100% Volatiles	~~10^{-9}~~ 10^{-6} – 10^{-10}
" "	II (~ 1)	None Venting occurs at ~ 24 Hours	≤ 1% Pu ≤ 1% Solid Fission Products 100% Noble Gases ≤ 10% Volatiles	10^{-4}

Uncertainties and Conclusions

The foregoing estimates of probabilities and risks associated with CRBR have included allowances for uncertainties. Unavailability estimates for shutdown and heat removal systems have been set high enough to include allowances for potential common cause failures, for example. In general probabilities have also been set high to account for external events such as earthquakes, tornados, or floods, for deliberate acts of sabotage, and for human error. However, the impact of all these factors on risk is difficult to predict with high accuracy because associated uncertainties are difficult to quantify.

Compliance with the principal design criteria, with the seismic criteria in the Code of Federal Regulations, and with 10 CFR Part 73 will assure that the risks from seismic events and sabotage are acceptably low. However, because of the low anticipated frequencies of such events, quantitative estimates of their risks necessarily have large uncertainties. Although the NRC is devoting significant effort to assure that risks from human error are acceptably low, for purposes of risk estimation, large uncertainty factors seem appropriate.

It is our best estimate that the probabilities of severe accidents involving core disruption and human fatalities or doses exceeding 10 CFR 100 guidelines in less than 10^{-6} per reactor year. H~~owever, to account for uncerta~~inties in pr~~obability and uncertainties in consequences we have conservatively~~ set ~~the upper limit of risks estimated for CRBR~~ from the FES Class 9 accident at ~~an order of magnitude (factor of 10)~~ higher than the best estimate ~~values. These risks appear in Table ___~~. Although there are large uncertainties on ~~the~~ some sequences, and specifically on the highly energetic CDA category, it is our best estimate that ~~the~~ CRBR risks are represented by the FES Class 9 Event. These risks appear in Table ___

The estimated probabilities of severe accidents for CRBR do not depend in a significant way on the Reactor Safety Study (RSS) which was published in 1975. However, the RSS has been reviewed to gain perspective regarding representative system unreliabilities and general aspects of methodology and uncertainties. For that reason the following discussion of the current status of WASH 1400 is provided.

In July 1977, the NRC organized an Independent Risk Assessment Review Group to (1) clarify the achievements and limitations of the Reactor Safety Study Group, (2) assess the peer comments thereon and the responses to the comments, (3) study the current state of such risk assessment methodology, and (4) recommend to the Commission how and whether such methodology can be used in the regulatory and licensing process. The results of this study were issued in September 1978 (Ref. 30). This report, called the Lewis Report, contains several findings and recommendations concerning the RSS. Some of the more significant findings are summarized below.

1. A. number of sources of both conservatism and nonconservatism in the probability calculations in RSS were found, which were very difficult to balance. The Review Group was unable to determine whether the overall probability of a core melt given in the RSS was high or low, but they did conclude that the error bands were understated.
2. The methodology, which was an important advance over earlier methodologies that had been applied to reactor risk, was sound.

3. It is very difficult to follow the detailed thread of calculations through the RSS. In particular, the Executive Summary is a poor description of the contents of the report, should not be used as such, and has lent itself to misuse in the discussion of reactor risk.

On January 19, 1979, the Commission issued a statement of policy concerning the RSS and the Review Group Report. The Commission accepted the findings of the Review Group. These findings have been considered in evaluating the potential risks from CRBR.

E. Rumble, SAI
24 June 1982
(415) 493-4326

SOURCE TERMS FOR CRBRP FES

Estimation of the release fractions of the various isotopes which can escape from the CRBRP are made using the isotope groups defined in WASH-1400. As shown in Table , four release classes are considered and releases to the environment are defined for three containment failure modes:

1. Design leakage and filtered venting
2. Overpressure failure (at about 24 hours)
3. Containment isolation failure (24" diameter ventilation line)

Releases from the primary system to the RCB can potentially occur by either leaking through the vessel head seals immediately following an energetic CDA or release from the sodium pool (formed in the reactor cavity after reactor and guard vessel meltthrough) through the reactor cavity vent system.

Chemically inert noble gases (Xe-Kr) are not removed from the RCB other than by decay or leakage to the environment. The remaining fission products, however, can be removed from the RCB by decay, leakage, filtered venting, and also by naturally occurring depletion mechanisms such as:

- Aerosol agglomeration and settling
- Thermophoretic deposition on cooler surfaces
- Plate-out

The fraction of airborne material which leaks to the environment in the long term, depends on the ratio of the leakage rate to the total removal (leakage, filter decay, and deposition) rate. Removal by aerosol agglomeration and settling, considered the dominant deposition mechanism, is modeled as an exponentially varing time dependent process.

Primary system sodium plays an important role in removing fission products in CRBRP. First, sodium chemically combines with fission products such as iodine and bromine to form less volatile compounds. Second, sodium is maintained well below its boiling point during normal operation and thus fission product release to the RCB is retarded by the liquid sodium. Third, sodium vapor, after it becomes airborne, becomes an aerosol. When sodium vapor enters the RCB, for example, a sodium oxide aerosol is formed. Since there are over 1 million pounds of primary sodium, a dense aerosol (10-100 ug/cc) will be airborne in the RCB. The airborne fission products will interact with and essentially respond as sodium oxide aerosols. For the purpose of analysis, therefore, the airborne fission products (less noble gases) are considered to be removed at the same rate as the sodium aerosols.

Referring to Table , the variation in release fractions among isotope groups and CDA classes depends on the magnitude of competing, concomitant, rate processes (leakage from the RCB, release to the RCB, and deposition in the RCB). It should be emphasized that the indicated release fractions do not include removal by decay; this is accounted for in the consequence calculations.

LEAKAGE FROM THE RCB

Leakage from the RCB considering CDA Class 1 involves design leakage at rates of 10^{-4} to 10^{-5}/hour and filtered venting which is 97% to 99% efficient. Approximately 57% of the RCB atmosphere will be released soon after failure by overpressure (CDA Class 2) since the RCB pressure will drop from about 2.3 atmospheres (abs) to 1 atmosphere (abs). Thereafter leakage through the RCB breach is about equal to the release rates of fission products and other gases into the RCB (10^{-1} to 10^{-2}/hour). The leakage rate to the environment considering failure of the containment to isolate a ventilation supply or exhaust line (CDA Classes 3 and 4) is estimated to be on the order of 10^{-1} to 10^{-2}/hour similar to the rates after overpressure failure. Thus for each release class, several exchanges will occur during the estimated 100-200 hour period in which the sodium pool boils.

RELEASE TO THE RCB

For the purposes of this analysis head release fractions were selected as indicated in Table . The fission product inventory remaining in the vessel after the head release constitutes the pool inventory after vessel meltthrough.

Table

HEAD RELEASES SELECTED FOR SOURCE TERM ANALYSIS

PRIMARY SYSTEM FAILURE CATEGORY	PRECENT OF CORE INVENTORY RELEASED FROM THE HEAD (%)						
	Xe-Kr	I	Cs-Rb	Te-Sb	Ba-Sr	Ru	La
I $\boxed{III}$	100	30	30	10	10	3	3
~~III~~ I	100	3	3	1	1	0.1	0.1

Pool releases were estimated by considering the relative volatilities of the fission products compared to sodium. Alkali metals such as Cesium, for example, boil-off 10 to 20 times the fractional rate of sodium vaporization. Halogens such as iodine form compounds with sodium and thus are released from the sodium pool at a slower rate than the sodium. The remaining semi-volatiles and solids are released considerably slower than sodium. Insignificant amounts of the non-volatiles (including fuel) are released to the RCB before cavity dryout.

Once the sodium pool has boiled-off, the remaining debris will increase in temperature and attack the concrete basemat. Additonal release of a fraction of the remaining fission products and fuel is then possible and may be exacerbated by sparging effects caused by off-gasing from the concrete during thermal decomposition.

DEPOSITION IN THE RCB

Deposition rates for airborne fission products are a function of the assumed particle shape and size as well as concentration. Typical analysis for similar sodium aerosol conditions indicate deposition rates in a single chamber of between 0.5 and 1.0 per hour. Considering leakage rates between 10^{-2} and 10^{-1} per hour, therefore, indicates that between 1% and 20% of the airborne fission products may eventually be released to the environment.*

An overpressure failure causes a rapid drop in containment pressure thereby releasing about 57% of its atmosphere. Since this release does not occur until about 24 hours after the head release and about 14 hours after pool boiling begins, considerable deposition of the airborne material occurs prior to the release. The remaining releases after overpressure relief are similar to those occurring after containment isolation failure.

In addition to the RCB, further deposition will occur in the reactor cavity and its vent system, the annulus between the containment and confinement (overpressure failure), and the ventillation system (containment isolation failure). Each of these features present a tortuous flow path and surface area enabling condensation, plate out, and settling.

The noble gases are conservatively estimated (decay not included) to completely escape to the environment for each CDA class. This is deemed appropriate for no deposition occurs and several exchanges of the RCB atmosphere will occur.

RESULTS

After considering the above factors, releases to the environment for each CDA Class were estimated for vessel head releases, pool releases and releases after sodium boil-off. These three release components for each CDA class were then combined into a single set of constant rate releases for input into the consequence model. The results of this analysis are shown in Table .

---

* Design leakage rates of 10^{-4} to 10^{-5}/hour correspond to 10^{-5} to $10^{-\cancel{6}\,7}$ long term release fractions. Filtered venting is 97% to 99% efficient.

Mohan
A-5
Walt
FYI

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Docket No. 50-537

MAY 10 1982

MEMORANDUM FOR: L. G. Hulman, Chief
Accident Evaluation Branch, DSI

FROM: Daniel R. Muller, Assistant Director
for Environmental Technology
Division of Engineering

SUBJECT: INPUT TO CRBR FES (CP)

In accordance with Mr. Thadani's telephone request, we are submitting the Antitrust and Economic Analysis Branch input to the accident impact section of the captioned DES starting with the sentence:

"There are other economic impacts..."

This was prepared without reference to other parts of the accident impact section and it will, therefore, be necessary for someone to carefully check the references to other parts of the Section, i.e., 6.1.4.4 and Table 6.1.4-2. Also, you may want to show the reference to the Comptroller General's report at the end of the Section rather than at the bottom of the page.

Daniel R Muller

Daniel R. Muller, Assistant Director
for Environmental Technology
Division of Engineering

Enclosure:
As stated

cc: M. Thadani

8209210193
PAR

There are other economic impacts and risk which are not included in the cost calculations discussed in Section 6.1.4.4 that can be monetized. These are accident impacts on the facility itself that result in added costs to the public, primarily taxpayers. These costs would be for decontamination and repair or replacement of the facility, and replacement power. Although, it is possible that the facility would simply be decommissioned and not actually restored following a serious (core-melt) accident, an assumption of restoration is considered conservative (high cost) in reflecting the cost impact of an accident. If the worth of the facility at the time of an accident is perceived to be worth more than the cost of restoration, then presumably the facility would be restored and the restoration cost would represent the cost impact. If the worth of the facility at the time of the accident is perceived to be less than the cost of restoration of the facility, then presumably the facility would not be restored and the cost impact would, at least be perceived to be less than the restoration cost such that use of the restoration cost would represent a high side estimate. Since the worth of the facility is primarily in the nature of research and development the actual value cannot be quantified any more accurately than as it is perceived at the time.

Experience with such costs is currently being accumulated as a result of the Three Mile Island accident. Although CRBR is considerably smaller in electrical output than Three Mile Island, the physical size and complexity of CRBR is comparable and the cost of decontamination and restoration is estimated to be about the same as that for Three Mile Island. If an accident occurs during the first full year of CRBR operation (1989), the economic penalty associated with the initial year of the unit's operation is estimated at $2250 million for decontamination and restoration, including replacement of the damaged nuclear fuel. This is based on a $952 million value in 1980 dollars as reported to Congress by the Comptroller General.[1] The $952 million in 1980 dollars has been escalated at 10% to 1989. Although property damage insurance would cover part of this, the insurance is not credited because the insurance payment times the risk probability would theoretically balance the insurance premium.

In addition, staff estimates average additional production costs of $25 million (1989 dollars) for replacement power during each year the CRBR is being restored. This is based on applicant's net projections of operating savings during the first six years of operation, discounted at 10% to 1989. Assuming the nuclear unit does not operate for 8 years due to shutdown, the total additional replacement power cost should be approximately 200 million in 1989 dollars.

If the probability of sustaining a total loss of the original facility is taken as the sum of the occurrences of a core melt accident (the sum of the probabilities for the categories in Table 6.1.3-2) then the probability of

---

1 Report to the Congress, by the Comptroller General of the United States, EMD-81--106, August 26, 1981.

a disabling accident happening during each year of the unit's service life is 1.0×10^{-5} . Multiplying the previously estimated costs of $2450 million for an accident to CRBR during the initial year of its operation by the above 1.0×10^{-5} probability results in an economic risk of approximately $25,000 (in 1989 dollars) applicable to CRBR during its first year of operation. This is also approximately the economic risk (in 1989 dollars) to CRBR during the second and each subsequent year of its operation. Although CRBR would depreciate in value such that the economic consequences due to an accident becomes less as the unit becomes older, this is considered to be offset by a higher cost of decontamination of the unit in the later years.