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JUL 1 1982

Docket No.: 50-537

MEMORANDUM FOR: Paul S. Check, Director
Clinch River Breeder Reactor Program Office, NRR

FROM: R. W. Houston, Assistant Director
for Radiation Protection
Division of Systems Integration

SUBJECT: REVISED EVALUATION OF ACCIDENTS FOR THE CRBR ENVIRONMENTAL REVIEW

In response to your request to L. G. Hulman, dated 3/31/82, the Accident Evaluation Branch (AEB) re-evaluated the risks resulting from a Class 9 accident at CRBR site, and provided you with our evaluation of the risks from a single Class 9 accident discussed in the FES. Subsequently, Bill Morris verbally requested us to perform additional risk analyses for a spectrum of four categories of severe accidents. Our revised evaluation of these accidents, which is being included as Appendix J to the proposed FES Addendum, is enclosed.

Our analysis is based on the information for a spectrum of four Core Disruptive Accident (CDA) classes, their probabilities, and the associated release fractions provided by Bill Morris and Ed Rumble (SAI). In this evaluation we have added a new section discussing the economic risks of the accidents on the facility itself.

Since our evaluation is based on the methodologies of the Reactor Safety Study and the related follow-on work on calculation of light water reactor (LWR) consequences, our methods at present do not account for the large quantities of sodium present in the CRBRP in place of the large quantities of water present in the LWRs. We have, however, bounded the consequences of sodium in our assessment.

The results of the AEB analyses indicate that the calculated risks for the selected CRBRP accidents are not different from the risks that the staff has presented in the environmental statements of light water reactors which have been licensed since the issuance of the Commission's June 1980 Statement of Policy.

The accident probabilities and release fractions were provided by Bill Morris and Ed Rumble, input for the economic impacts of the facility loss were provided by Argil Toalston, ~~L. Soffer~~ provided population distribution information. The

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NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO section on liquid pathway considerations was prepared by R. Codell. The evaluation of the accident risks was performed by Mohan Thadani, x28941, who also coordinated the preparation of attached Appendix J.

Original signed by L G Hulman
R. Wayne Houston, Assistant Director
for Radiation Protection
Division of Systems Integration

Enclosure:
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APPENDIX J

ACCIDENT EVALUTION BRANCH INPUT TO THE FINAL ENVIRONMENTAL STATEMENT UPDATE FOR CLINCH RIVER BREEDER REACTOR PLANT

Addendum to Section 7.1

J~~7~~.1 PLANT ACCIDENTS INVOLVING RADIOACTIVE MATERIALS

The staff has examined the Clinch River Breeder Reactor Plant (CRBRP) Final Environmental Statement (FES) with a view to updating the FES to reflect any plant-site-feature or regulatory framework changes that have occurred since the FES was issued in February 1977. The staff finds that since the issuance of the FES, no significant plant-site changes have occurred which are relevant to environmental concerns, nor is their significant new information relevant to environmental concerns which bears on the project or the environmental impacts or risks of accidents as reported in the FES. Since the issuance of the FES, however, the Commission has issued a Statement of Interim Policy (June 13, 1980) that provides guidance on the considerations to be given to nuclear power plant accidents under NEPA. Among other things the Commission's statement indicated that: "this change in policy is not to be construed as any lack of confidence in conclusions regarding the environmental risks of accidents expressed in any previously issued (Environmental Impact) Statements, nor, absent a showing of...special circumstances, as a basis for opening, reopening, or expanding any previous or ongoing preceeding."

The staff in its environmental review of the CRBRP application concluded that the CRBRP did constitute a special circumstance that warranted consideration of Class 9 accidents in the Environmental Statement. Because the CRBRP reactor was very different from the conventional light water reactor plants for which the safety experience base is much broader, the staff included in the CRBRP FES a discussion of the potential impacts and risks of such accidents. As noted in the Statement of Interim Policy, the fact that the staff had identified this case as a special circumstance was one of the considerations that led to the promulgation of the June 13, 1980 Statement.

IR In examining the CRBRP FES, as issued in 1977, the staff has considered the guidance of the Interim Policy Statement which was provided for "Future NEPA Reviews." We have concluded that the discussion of accidents as presented in the FES generally meets that guidance, except for consideration of the risks due to liquid pathways. A discussion of the liquid pathway risks is included ~~below~~ in Section J.1.2 .

J.1.1 IR Design Basis Accidents ~ All Caps

IR The results of the staff's analyses of the realistic consequences of design-basis accidents were presented in the FES Table 7.2. The reported values appear to the staff to be reasonable. This conclusion is based upon comparison of the realistic dose consequences of CRBRP design-basis accidents with the corresponding doses for some recently evaluated light water reactors (LWRs) such as Comanche Peak, Callaway, and Palo Verde plants, as shown in Table J-1. The CRBRP doses are within the range of dose values of some of the LWRs, and the radiological health effects and the environmental impacts of such postulated accidents would be comparable to those from postulated LWR accidents.

Although the staff analysis of the design-basis accidents does not treat in detail the probabilities of accident occurrence, except as implied in a general way in the development of the accident classification scheme of the previously proposed annex of Apendix D to 10 CFR 50, the estimated doses are so small that in the staff's judgement no unreasonable radiological risk to the public health and safety, and to the environment could arise as a result of these accidents.

Table J.1 Comparison of design-basis accident (Classes 2-8) site boundary doses reported in the CRBRP FES with corresponding doses reported in the environmental statements of some recent LWR operating license reviews.

Accident	CRBRP FES	Comanche-Peak FES	Callaway FES	Palo Verde FES
Fuel-handling accidents				
Rems thyroid	0.4	2.0	4.0	0.002
Rems whole body	0.5	0.05	1.0	0.07
Large-break LOCA or site suitability source term				
Rems thyroid	1.0	85.0	91.0	8.0
Rems whole body	0.1	1.2	2.2	0.4
Rems lung	0.2	-	-	-
Rems bone	1.2	-	-	-

Included in this judgement is acknowledgment that accidents of the types represented by those described in FES Table 7.2 for Classes 2-8 have a finite and relatively larger likelihood of occurrence during the operating lifetime of the CRBRP than the occurrence of Class 9 accidents. Furthermore, their consequences are required not to exceed the dose guideline values of 10 CFR 100. This acknowledgment ensures that an assessment of the adequacy of the engineered safety features and operating requirements to mitigate and limit the consequences of such accidents will be considered in the safety evaluation of the CRBRP. Such considerations at all contemporary LWRs have resulted in a combination of engineered safety features and operating procedures so that the contribution of these accidents to total risk to the environment is judged to be negligible. The staff will reexamine the radiological risk contribution of the design-basis accidents at both the construction permit stage and the operating license stage of CRBRP, giving consideration to the probabilities of occurrence of accidents and to their consequences. The purpose of this reexamination at each stage of licensing will be to require that the plant safety and mitigation systems be designed adequately and operated to offset the uncertainties arising from a limited national and international LMFBR operating experience base, and to ensure that the radiological risks of accidents up to and including the severity of design basis events are not greater than those of the LWRs.

II.1.2 <u>Evaluation of Class 9 Accidents</u> All caps.

The staff has also performed some new calculations to provide additional perspective on the risk associated with the hypothetical Class 9 accidents at the CRBRP. Presented below is a discussion of the Class 9 accident sequences, estimates of accident probabilities, release of radioactive material to ~~atmosphere~~ environment, risks due to the atmospheric and liquid pathway exposures, economic costs of the loss of the facility, the uncertainties in predictions, and the conclusions.

All Caps.

(1) Probabilities of Severe Accidents

The Class 9 accident discussed in the FES involved a sequence and release representative of possible core disruptive accidents (CDAs). Additional sequences are included here to provide better perspective regarding the risks of CRBRP severe accidents.

The frequencies of severe (Class 9) accidents at CRBRP involving potential core disruption and containment failure 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. Secondly, there are variations in the release to containment that are dependent on the energy associated with core disruption and the nature of the response of the primary coolant boundary. Finally, the potential for containment failure must be considered. The probabilities of such events 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 when necessary, as exemplified by a failure to scram during a loss of flow event (ULOF) or a transient overpower event (UTOP); and (3) core-wide fuel failures as exemplified by propagation of local fuel faults (FFP).

As discussed on pages 7-2 and 7-7 of the FES, requirements* for prevention of severe accidents will be imposed on the CRBRP design to ensure that initiation of core disruptive accidents is made very improbable. Consequently such accidents are not included in the CRBRP design-basis accident spectrum.

LOHS events at CRBRP 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 CRBRP SGAHRS system, which is similar in many respects to the steam generator/auxiliary feedwater

* The staff has required in the FES that the design basis accidents envelope extend to accidents with probabilities of one chance in one million per reactor year.

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 the judgment of the staff that there is sufficient inherent redundancy, diversity, and independence in the SGAHRS and DHRS systems to achieve a core degradation frequency due to LOHS events of less than per 10^{-4} reactor year. This estimate is based on a general consideration of typically achievable PWR auxiliary feedwater system reliabilities, the 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 the CRBRP 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 CRBRP primary coolant system will operate at significantly lower pressures than LWR primary coolant systems. This reduces the frequency of large ruptures in the primary coolant system. To further ensure that large breaks cannot occur and cause core damage, implementation of preservice and inservice 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 (i.e. design basis) events at CRBRP. The frequency assumed for LOHS adequately bounds the LOCA contributions to core disruption frequency.

The coolant inlet region of the CRBRP core is being designed to 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 and would be detectable by the core outlet thermocouples if significantly reduced core flow were to result. The frequency assumed for LOHS core degradation sequences adequately bounds the flow blockage contribution to core disruption frequency.

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 gailure 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 employ diverse mechanisms and materials. Although the staff review of these systems is not complete, it is the judgement of the staff that there is sufficient inherent redundancy, diversity, and independence in the overall shutdown system designs to expect an unavailability of less than 10^{-5} per demand. 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. Using the assumption, based on LWR experience, that an average of about 10 transients (required scram) might occur per year of operation over the life of the plant, the staff concludes that the combined frequency of degraded core accidents initiated by ULOF and UTOP events is less than 10^{-6} per reactor year.

The CRBRP 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 has been achieved on fuel designs similar to that of the CRBRP. Therefore, the frequency of fuel failure propagation is considered very low. The frequencies attributed to LOHS, UTOP, and ULOF events adequately bound the contribution to core disruption frequency from fuel failure propagation.

In summary, the frequencies of core disruption from LOHS, UTOP, ULOF, LOCA, and FFP events are all considered to be less than 10^{-4} per reactor year. Even when combined, the overall combined probability of these types of events are estimated to have a frequency of 10^{-4} per reactor year or less. This net frequency does not reflect the variations in response of the primary coolant system that might be associated with the various initiators. Some initiators may result in more severe response than others. This is taken into account as described in the following paragraphs.

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. Three categories have been identified and are listed here in order of increasing potential threat to containment integrity and increasing release of radioisotopes into containment:

I. Primary system remains intact; no significant release of radioactive materials to the containment atmosphere.

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~~ and eventually reach the containment atmosphere through the reactor cavity vents at a slow rate relative to the initial releases of Category III below.

III. Primary system seals fail due to excessive mechanical and/or thermal loads. Core Pu, solid fission products, noble gases, and volatile material would be released into upper containment immediately.*

Most core disruptive accidents are expected to be nonenergetic and to culminate in effects such as described for 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

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

containment failure. These include a filtered vent system to relieve containment pressure, a containment purge system to reduce the potential for hydrogen explosions, fans in the annulus between the steel containment shell and the confinement structure to cool the two structures, and vents to relieve pressure from gases generated behind the reactor cavity cell liners. These provisions are currently under review by the staff.

The Class 9 accident release described in Category III corresponds to a core disruption of sufficient energy, due to recriticality, to cause mechanical damage to the primary coolant system. The staff is reviewing the potential for energetic recriticalities to determine the magnitude of energy release anticipated. If the conclusion of this review is that an energy release beyond primary system capability cannot be precluded, the staff will require some action be taken (e.g., that the vessel be strengthened or that head restraints and sodium spray deflectors be installed) to prevent 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 Atomic International's design studies of a 500 MWe LMFBR demonstration plant.

Assuming that a core disruptive accident occurs, the conditional frequencies of event Categories I through

III subsequently occurring are estimatyed as folows:

Primary System Failure - Category I & II combined:

0.9 per CDA

Primary System Failure - Category III:

0.1 per CDA

These estimates reflect the lower frequencies expected for core disruption accidents of increasing energetics.

Response ~~to~~ of Containment

For the purpose of estimating risk given the threats to containment identified above, the following two containment failure modes leading to airborne releases are identified:

(A) Failure of Containment Caused by Overpressure.

(B) Failure of Containment to Isolate.

The frequency and consequences of releases to the ground by basemat penetration are considered to be overshadowed by airborne releases, as discussed under the subsection entitled "LIQUID PATHWAYS."

The staff will require that the containment annulus cooling and vent/purge systems be designed with sufficient redundancy and quality, and be tested and inspected during operation with sufficient frequency, that it can be assumed that their unavailability for anticipated mission times will not exceed 10^{-2} per demand. Such systems will not be needed to prevent overpressure conditions 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. Should the containment systems be required ater a temporary loss of all ac power initiating event, failure to recover power ac before containment failure occurs is estimated to have a frequency of about 10^{-2}/demand.

Containment isolation is an engineered safety feature at CRBRP. Such systems are designed to high quality standards and with redundancy. An unavailability of less than 10^{-2} per demand is feasible for such systems and is expected to be attained at CRBRP given that implementation of an adequate reliability program will be required.

In summary, the conditional unavailabilities for the containment failure modes are as follows:

Containment Failure Mode A (Mitigating System Failure): $\leq 10^{-2}$ per demand

Containment Failure Mode B (Containment Isolation Failure): $\leq 10^{-2}$ per demand. All Caps

(2) Release of Radioactive Material

Estimations 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 J.3, stet four release classes are considered and releases to the environment are defined for three containment modes:

1. Design leakage and filtered venting;

2. Overpressure failure (at about 24 hours); and
3. Containment isolation failure (24" diameter ventillation 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 (which forms in the reactor cavity after reactor guard and 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; and
- 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, filtration, decay, and deposition) rate. Removal by aerosol agglomeration and settling, considered the dominant deposition mechanism, is modeled as an exponentially varying 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 coolant 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.

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Referring to Table J.~~6~~, 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.

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Leakage From the RCB

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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 atmosphere (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. X o.k X X

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<u>Release to the RCB</u>

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For the purposes of this analysis head release fractions were selected as indicated in Table J.2. The fission product inventory remaining in the vessel after the head release constitutes the pool inventory after vessel meltthrough. K X

Table J.2

HEAD RELEASE 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
III	100	30	30	10	10	3	3
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 Cecium, for example, boil off 10 to 20 times the ~~fractional~~ rate of sodium. 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 dry debris will increase in temperature and attack the concrete basemat. Additional 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

* See footnotes to Table J.4

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.* The overpressure failure mode drops the containment pressure to 1 atmosphere, thereby releasing 57% of its atmosphere. Since this release ~~does~~ would not occur until about 24 hours after the head release, and about 14 hours after pool boiling begins, considerable deposition of the airborne material would occur~~s~~. The remaining releases after overpressure relief are similar to those occurring after containment isolation failure.

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

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

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* Design leakage rates of 10-4 to 10-5/hour correspond to 10-5 to 10-~~6~~7 long term relese fractions. Filtered venting is 97% to 99% efficient.

After considering the above factors, releases to the environment for each CDA Class were esitmated for vessel head releases, pool releases and dry cavity releases. 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 ~~is~~ are shown in Table J-7. stet

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<u>Comparison of Accident Sequence Frequencies</u>

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The most probable class of CDA accident sequences is that in which containment systems function ~~is~~ as designed. Releases to the environment would occur becasue of design leakage and controlled, filtered venting at about 24 hours after CDA initiation. The likelihood of this accident class is estimated to be less than 10^{-4}/per reactor year. The doses associated with this accident class ~~is~~ are not expected to exceed 10 CFR 100 guidelines. The two most probable classes of CDA accident sequences for which the doses are expected to exceed 10 CFR 100 guidelines are as follows. First, a CDA is initiated (less than 10^{-4}/per reactor year), a primary system failure of Category I and II or III (combined conditional frequency $\sim$1) occurs, and containment failure mode A, containment cooling or vent/purge failure at approximately 24 hours (less than 10^{-2}/ demand) follows. This class of CDA accident sequences corresponds to the FES Class 9 accident. Second, a CDA is initiated (less than 10^{-4}/per reactor year), a primary system failure of Category

I and II (combined conditional frequency ~1) occurs, and containment failure mode B, failure to isolate (less than 10^{-2}/demand) follows. Both of these classes of CDA accident sequences would, therefore, have an estimated bounding frequency of less than 10^{-6} per reactor year. Furthermore, the frequency of 10^{-6} per reactor year bounds each CDA accident class sufficiently such that the combined frequency of the two classes is estimated to be less than 10^{-6} per reactor year.

A less probable class of CDA sequences for which doses ~~are expected to~~ could exceed 10 CFR 100 guidelines would be initiation of a CDA (less than 10^{-4}/year), primary system failure category I and II (about 1), or III (10^{-1} per year), and containment failure mode B, and failure to isolate (less than 10^{-3}/demand). The event ~~with a~~ has an estimated combined frequency of less than 10^{-6} to 10^{-7} per reactor year.

These CDA sequence classes correspond to releases to the environment of four different magnitudes, ~~and~~ and their probabilities represent an estimate of the frequency of each release mode.

Table J-~~2~~4 gives the inventory of activity of radionuclides in the CRBRP core at the time of shutdown. The CDA sequence classes and their releases to the environment are summarized in Table J-~~3~~ J-3. The first class in the table, which involves no containment failure, is expected to

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Table J.3 CRBR CDA sequence classes

CDA class	Initiation	Primary system failure category	Containment failure mode	Bounding estimate of containment release frequency (per reactor year)	Percent of core inventory release to environment[1]						
					Xe-Kr	I	CS-Rb	Te-Sb	Ba-Sr	Ru[2]	La[3]
1	Generic Core Disruption	I and II or III	None	10^{-4}	100	0.01	0.01	0.01	0.01	0.001	0.001
2	Generic Core Disruption	I and II or III	A (Overpressure)	10^{-6}	100	1.0	1.0	0.6	0.6	0.08	0.08
3	Generic Core Disruption	I and II	B (Containment Isolation)	10^{-6}	100	1.3	1.3	0.8	0.8	0.06	0.06
4	Generic Core Disruption	III	B (Containment Isolation)	10^{-7}	100	4.0	4.0	1.7	1.7	0.35	0.35

[1]Background on the isotope groups and release mechanism is presented in Appendix VII of "Reactor Safety Study," WASH-1400, NUREG-75/014, October 1975.
[2]Includes Ru, Rh, Mo, Tc.
[3]Includes Y, La, Zr, Nb, Ce, Pr, Nd, Np, Pu, Am, Cm.

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Table J.~~1~~ 4 Activity of radionuclides in the CRBR reactor core at 1121 MWt

Group/radionuclide	Radioactive inventory in millions of curries	Half-life (days)
A. NOBLE GASES		
Krypton-85	0.1	3,950
Krypton-85m	5.0	0.183
Krypton-87	8.0	0.0528
Krypton-88	11.4	0.117
Xenon-133	52.3	5.28
Xenon-135	56.5	0.384
B. IODINES		
Iodine-131	30.0	8.05
Iodine-132	40.8	0.0958
Iodine-133	51.5	0.875
Iodine-134	54.7	0.0366
Iodine-135	50.4	0.280
C. ALKALI METALS		
Rubidium-86	0.14	18.7
Cesium-134	0.66	750
Cesium-136	2.7	13.0
Cesium-137	1.7	11,000
D. TELLURIUM-ANTIMONY		
Tellurium-127	3.7	0.391
Tellurium-127m	0.54	109
Tellurium-129	9.7	0.048
Tellurium-129m	2.7	34.0
Tellurium-131m	4.5	1.25
Tellurium-132	40.0	3.25
Antimony-127	3.8	3.88
Antimony-129	10.3	0.179
E. AKALINE EARTHS		
Strontium-89	16.0	52.1
Strontium-90	0.7	11,030
Strontium-91	21.0	0.403
Barium-140	42.0	12.8

Table J.~~2~~4 (Continued)

Group/radionuclide	Radioactive inventory in millions of curies	Half-life (days)
F. NOBLE METALS		
Molybdenum-99	46.6	2.8
Technetium-99a	40.3	0.25
Ruthenium-103	52.6	39.5
Ruthenium-105	38.5	0.185
Ruthenium-106	19.6	366
Rhodium-105	38.5	1.50
G. RARE EARTHS, REFRACTORY OXIDES, AND TRANSURANICS		
Yttrium-90	0.71	2.67
Yttrium-91	20.4	59.0
Zirconium-95	36.2	65.2
Zirconium-97	40.9	0.71
Niobium-95	34.8	35.0
Lanthanum-140	42.2	1.67
Cerium-141	42.9	32.3
Cerium-143	34.8	~~35.0~~ 1.38
Cerium-144	20.2	284
Praseodymium-143	34.8	13.7
Neodymium-147	17.0	11.1
Neptunium-239	1100	2.35
Plutonium-238	0.38	32,500
Plutonium-239	0.11	8,900,000
Plutonium-240	0.10	2,400,000
Plutonium-241	13.0	5,350
Americium-241	0.16	150,000
Curium-242	14.0	163
Curium-244	0.01	6,630

Note: The above grouping of radionuclides corresponds to that in Table J.3.

should this include Tritium?

produce doses not exceeding the guidelines ~~meet the requirements~~ of 10 CFR 100.* The second class in the table is the FES Class 9 accident sequence. Although the sequences represented by the third and fourth classes would involve earlier releases than the FES Class 9 accident, it is expected that they would involve risks (product of probability and consequences) of about the same ~~order~~ as the FES Class 9 accident risk.

(3) CR Atmospheric Pathways Risks — All Caps.

CR The potential atmospheric pathway radiological consequences of these accidents have been calculated by the consequence model used in the RSS (NUREG-0340) adapted and modified to the ~~specific~~ CRBRP site. The model used 1 year of site meteorologic data, projected population for the year 2010 extending throughout a ~~regions of 80-km (50-mi)~~ radius ~~and~~ of 563-km (350-mi) ~~radius~~ from the site, and habitable land fractions within the 563-km (350-mi) radius. The essential elements of the atmospheric pathways model are shown in schematic form in Figure J.1.

CR To obtain a probability distribution of consequences, the calculations were performed assuming the occurrence of each accident-release sequence at each of 91 different "start" times throughout a 1-year period. Each calculation utilized the site-specific hourly

*The comparison to 10 CFR 100 guidelines is made to indicate that this class of CDA does not have such severe consequences as other Class 9 accidents. The 10 CFR 100 guidelines were developed for siting analysis and are often applied in design basis accident analysis.

Figure ~~7~~J.1 Schematic outline of atmospheric pathway consequence model

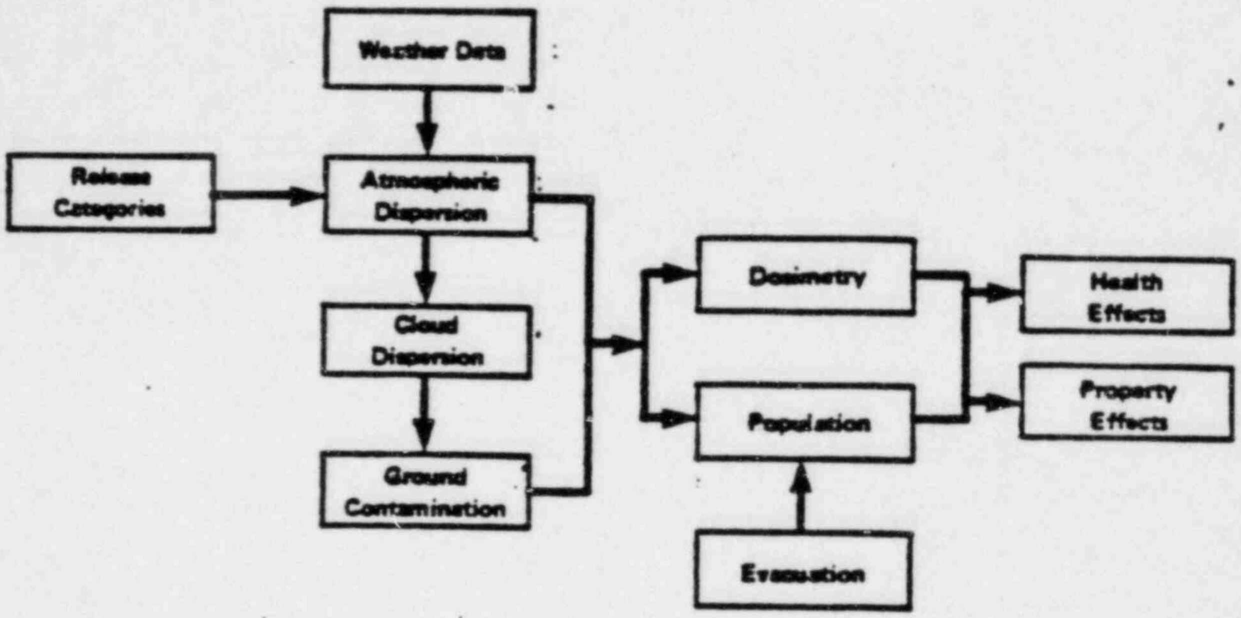

Figure 7.1 Schematic outline of atmospheric pathway consequence model

meteorological data and seasonal information for the time period following each "start" time. The consequence model also contains provisions for incorporating the consequence-reduction benefits of evacuation, relocation, and other protective actions, since early evacuation and relocation of people would considerably reduce the exposure from the radioactive cloud and from the contaminated ground in the wake of the cloud passage. The evacuation model used has been revised from that used in the RSS for better site-specific application. The quantitative characteristics of the evacuation model used for the CRBRP site ~~are~~ include conservative estimates made by the staff of key parameters. since These estimates were ~~made~~ The applicant's estimates are in a preliminary state of preparation. Included among the key parameters was the ~~one~~ assumption of 12 hours delay in starting evacuation after operator identification of a severe accident. There normally would be some facilities near a plant--such as schools or hospitals--where special equipment or personnel may be required to effect evacuation, and there may be some people near a site who may choose not to evacuate. Several facilities of this type have been identified near the CRBRPsite, such as the Loudon County Memorial Hospital, Roane County High School, and facilities related to national security. Therefore, actual evacuation effectiveness could be greater or less than that characterized, but would not be expected to be ~~very much~~ significantly less.

CR The other protective actions include: (1) either complete denial of use (interdiction), or permitting use only at a sufficiently later time after appropriate decontamination of foodstuffs such as crops and milk, (2) decontamination of severely contaminated environment (land and property) when it is considered to be economically feasible to lower the levels of contamination to protective action guide (PAG) levels, and (3) denial of use (interdiction) of severely contaminated land and property for varying periods of time until the contamination levels are reduced by radioactive decay and weathering such that land and property can be economically decontaminated as in (2) above. These actions would reduce the radiological exposure to people from immediate and/or subsequent use of or living in the contaminated environment. X

Early evacuation of people from the plume exposure pathway zone (EPZ) and protective actions as mentioned above are considered essential sequels to severe nuclear reactor accidents involving significant release of radioactivity to the atmosphere. Therefore, the results shown for CRBRP include the benefits of these protective actions. X x

There are uncertainties in each facet of the estimates of consequences (See Figure J.1) and the error bounds may be as large as they are for accident probabilities. The results of the calculations, based on conservative assumption of 12 hour delay in evacuation, are summarized and compared * with those for Midland Plant (LWR) in Table J.5 as expectation values, or averages of environmental risk per year of reactor operation. These averages are instructive as an aid in the comparison of radiological risks associated with potential CRBRP accidents and those risks calculated for recently evaluated LWRs, (e.g., Midland) for which calculations of radiological risks were made in essentially the same manner. The table shows the average risk associated with population dose, early fatalities, latent fatalities, and costs of protective actions and decontamination.

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Table J.5 A comparison of average values of environmental risks due to selected CRBRP accidents with those for Midland Plant

Environmental risk (per reactor year)	CRBRP	Midland
Population exposure		
Person-rems within 80 km	3.5	26
Total person-rems	5	130
Early fatalities	6x10-6	1.5x10-5
Latent cancer fatalities		
All organs excluding thyroid	0.3x10-3	7.2x10-3
Thyroid only	0.04x10-3	1.8x10-3
Cost of protective actions and decontamination	$690*	$4,800*

*1980 dollars

The population doses and latent fatality risks may be compared with those for normal operation population doses given in table 5.13 of the FES. The comparison shows that the accident risks are comparable to operating risks.

For perspective and understanding of the meaning of the early fatality risks of 6x10-6 per reactor-year, however, the staff notes that to a good approximation the population at risk within about 16 km (10 miles) of the plant, is expected to be about 80,000 persons in the year 2010. Based upon this population estimate, there is a risk of about .5 fatality per reactor year. Accidental fatalities per year for a population of this size, based upon overall averages for the United States*, are approximately 18 from motor vehicle accidents, 6.2 from falls, 2.5 from drowning, 2.3 from burns, and 1.0 from fire arms.

(4) Liquid Pathways All Caps

Surface water hydrologic properties at CRBRP should be similar to those used for the Liquid Pathways Generic Study (LPGS) small river site which was based on the Clinch - Tennessee - Ohio - Mississippi rivers system, although the river uses and population in the LPGS were based upon national averages and have not been directly compared to the CRBRP. The groundwater characteristics at Clinch River do not indicate any unusual adverse transport characteristics.

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*Based on risk to individual in "CONAES Final Report," National Research Council, Chapter 9, pp 577-534, 1979.

Additionally, the CRBRP is a considerably smaller plant than LPGS case (CRBRP is 1121 MWt vs. 3425 MWt assumed for LPGS), and contrary to the Light Water Reactors characteristics, CRBRP does not contain any large storage of water which could serve as a potential "prompt source" to the environmental liquid pathways. Therefore, only the radioactive material leached from the core debris by the local groundwater is likely to be transported to the Clinch River. This source was found in the LPGS to be considerably smaller than the "prompt source." Therefore, based on the preliminary appraisal of the liquid pathways, the staff concludes that the liquid pathways impacts of CRBRP would be probably smaller than those for the LWRs analyzed in the LPGS "Small River" site case.

(5) <u>Other Economic Risks</u> All Caps

There are other economic impacts and risks which are not included in the costs that can be given a monetary value. 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 of power. Although it is possible that the facility would simply be decommissioned rather than 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 more than the cost of restoration of the facility, then presumably the facility would not be restored and the cost impact would be less than the restoration cost, so that use of the restoration cost would represent a high side estimate. Because the worth of the facility is primarily in the nature of research and development, the actual value cannot be quantified any more accurately than it is perceived at the time.

Experience with such costs is currently being accumulated as a result of the Three Mile Island accident. Although CRBRP is considerably smaller in electrical output than the Three Mile Island plant, the physical size and complexity of CRBRP 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 CRBRP 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 (1981). 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.

The probability during each year of the units service life of sustaining a total loss of the original facility as a result of a disabling accident is taken from Table J-3 as 1.0 x 10-4. Multiplying the previously estimated costs of $2450 million for an accident to CRBRP during the initial year of its operation by the above (1.0 x 10-4) probability results in an economic risk of approximately $250,000 (in 1989 dollars) applicable to CRBRP during its first year of operation. This is also approximately the economic risk (in 1989 dollars) to CRBRP during the second and each subsequent year of its operation. Although CRBRP would depreciate in value such that the economic consequences of 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 later years.

~~The assessment of environmental risks of atmospheric pathways, assuming reasonable protective action, shows that the risks are significantly lower than similarly calculated values for light water reactors currently being licensed for operation. See, for example, FES for Callaway (NUREG-0813), DES for Seabrook Station (NUREG-0564), and DES for Skagit (NUREG-0894) for the environmental risks of light water reactos.~~

(6) UNCERTAINTIES

The foregoing estimates of frequencies and risks associated with CRBRP have included allowances for uncertainties. For example, unavailability estimates for shutdown and heat removal systems have been set high enough to include allowances for potential common cause failures. However, the risks from sabotage or from external natural events such as earthquakes, tornadoes, and floods beyond design bases for such events are difficult to quantify. This situation is generic to LWRs and advanced reactors such as CRBRP. NRC is presently devoting significant effort to developing methods for quantifying risks from such events. Compliance with current NRC siting structural, and seismic design criteria, and with 10 CFR 73 [redacted] on physical security provides assurance that reactor related risks from external events are adequately low. The CRBRP design will be required to meet all these criteria. Risks and the uncertainties in risks from the CRBRP related to sabotage and external events are not expected to differ significantly from such risks and their associated uncertainties at LWRs.

One additional potential containment failure mode not quantified above involves early containment failure and release caused by either a spray fire or missile generated from a very energetic CDA. The staff will review the potential for CDA energetics to ensure that necessary design enhancements of the primary coolant system are incorporated such that the probability of primary coolant system failure as a result of physically reasonable core rearrangement of sodium, cladding, or fuel will be very small. However, because it is possible to hypothesize nonmechanistic and speculative coherent and rapid core reconfigurations leading to high reactivity ramp rates, high energetics cannot be entirely precluded. Quantification of the frequency of this very improbable event at this time would involve such large uncertainties that the results would have no real meaning.

It should also be noted that the results do not fully account for the effects of the sodium coolant on the radioactive source term. For example, inclusion of the effects of sodium is expected to reduce the quantity of iodine available for leakage. The large mass of sodium aerosol also contributes to the agglomeration and settling of aerosols in the primary containment. On the other hand, the sodium activation products would be released together with the primary coolant, thereby adding to the amount of radioactive material released to the containment. On balance, it is expected that the risk contribution of the presence of radioactive sodium ~~be significant, and therefore,~~ would not invalidate the conclusions of these calculations. Further consideration of this subject will be included in the staff's review of the Probabilistic Risk Assessment for this plant, and in the staff's Safety Evaluation Report.

In summary, from the limited quantitative analyses discussed above, it is the best estimate of the staff that the frequency of individual classes of severe accidents resulting in fatalities or even doses exceeding 10 CFR 100 guidelines is less than 10^{-6} per reactor year. Compliance with current design criteria will ensure that risks from external events and sabotage are acceptably low. The risks estimated for CRBRP from ~~the FES~~ selected Class 9 accidents appear in Table J-5.

The estimated probabilities of severe accidents for CRBRP 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, (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. This report, commonly 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 non-conservatism 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.

OR

(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 CRBRP.

J.1.3 CONCLUSION

The foregoing sections have evaluated the environmental impacts of severe accidents, including potential radiation exposures to the population as a whole, the risk of near- and long-term adverse health effects that such exposures could entail, and the potential economic and societal consequences of accidental contamination of the environment. The assessment of environmental risk from several categories of accidents, assuming reasonable protective action, provides perspective on the overall risk from CRBRP accidents in comparison to those from LWRs. From this comparison it is concluded that there is no basis for disagreement with the FES conclusions (that the CRBRP accident risks will not be different from those of ~~recent~~ LWRs).