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ATTACHMENT 3

REVIEW OF THE INTERACTION OF SODIUM WITH CONCRETE AND OTHER MATERIALS

Task 1 Report

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1. Introduction

The formal licensing review of the Clinch River Breeder Reactor (CRBR) was initiated with the submittal of the Preliminary Safety Analysis Report in 1975. This reactor differed from commercial light water reactors in several important areas, one of which is the use of sodium coolant. In the initial application permit, emphasis was placed on reliability analysis in the development of the safety-related design considerations for the plant.

One of these safety analyses considered the accident scenario for a hypothetical core disruptive accident (HCDA). One scenario postulates that the reactor vessel is breached along with the guard vessel and cavity cell liner. This postulated series of events places bare concrete in contact with hot sodium metal. An understanding of sodium-concrete interactions was necessary to assess such safety related concerns as hydrogen generation, with its attendant explosion risk, gas generation which could over-pressurize the containment building, degradation of the concrete which could lead to penetration or structural collapse, and, finally, the energy release from sodium reactions which under some circumstances would rival the decay heat.

Several organizations initiated studies of sodium-concrete interactions. The two principal investigators were Sandia National Laboratories and the Hanford Engineering Development Laboratory (HEDL). Altogether, more than 100 significant ma[illegible] experiments have been conducted to study sodium-concrete interactions. These tests examined concretes similar to those used at the Fast Flux Test Facility (FFTF) and proposed for use at Clinch River. The experimental results varied considerably, even under what appeared to be similar initial conditions. This variation is not completely understood and is the principal reason for concern about predicting long-term effects of a large sodium spill.

2. Review of Sodium-Concrete Experiments

Several years ago, the authors had the opportunity to review the sodium-concrete experimental data available at that time (Reference 1). Since then, a substantial number of additional experiments have been conducted but only a few of these have examined limestone concrete. In three years Sandia has conducted only four additional large scale limestone concrete tests and HEDL has conducted only six tests.

Results from the limestone concrete tests at HEDL are summarized in Table I and results from Sandia's tests are shown in Table II. Blanks in some columns for certain tests indicate that no information has been made available. Some Sandia tests show two numbers due to changes in the numbering system for these tests. The specimen thickness, area and orientation are provided. Sodium temperatures, masses and depths are also shown. Information is given on the test duration, whether or not an exothermic reaction was observed, hydrogen evolution, the extent of penetration and whether or not noises were observed suggesting a vigorous interation.

Other, smaller scale tests, conducted elsewhere are not listed although information concerning these tests can be found in References 2-4. Only limestone concrete data is listed in these tables. While many experiments have been conducted with other concretes, such as basalt concrete and magnitite concrete, it is felt that the various types of concrete are significantly different materials which must be treated separately. It is

Table I. HEDL Sodium-Limestone Concrete Interaction Tests

Test Number	Concrete Type/ Thickness, cm	Surface Area, m^2 (Orientation)	Temperature Ave, °C (Max)	Sodium Mass, kg/ (Depth, cm)	Exothermic Reaction	Length of Test (hr)	Hydrogen Evolven kg	Penetration Max. cm	Noises (Pops, Bumps, etc)
SC-4	Limestone 30	0.092 (Horizontal)	677 (802)	22.7 (25.4)	Yes	8	0.18	8.4 (6.1)	Yes
SC-5	Limestone 30	0.092 (Horizontal)	871 (871)	22.7 (25.4)	No	2	0.10	4.3 (2.5)	No
SC-6	Limestone 30	0.092 (Horizontal)	871 (871)	24.4 (27.3)	No	8	0.26	4.8 (3.8)	No
SC-8	Limestone 30	0.092 (Horizontal)	871 (871)	23.6 (26.4)	Yes	24	0.28	5.3 (4.4)	No
SC-10	Limestone 30	0.092 (Vertical)	871 (1093)	18.1 (20.3)	Yes	8	0.25	13.0 (8.9)	Yes
SC-12	Limestone 30	0.092 (Vertical)	871 (954)	19.1 (21.4)	Yes	24	0.23	14.0 (9.2)	Yes
SC-13	Limestone 30 +3.15 Kg NaOH	0.092 (Horizontal)	871 (932)	22.7 (25.4)	Yes	26	Yes	7.6 (4.4)	Yes
SC-14	Limestone 30 +3.15 Kg NaOH	0.092 (Horizontal)	871	22.7 (25.4)	-	24	Yes	7.6 (5.7)	?
SC-1[illegible]	Limestone 30	0.092 (Horizontal)	871	3.15	-	8	No	5.0 (2.5)	?
SC-1[illegible]	Limestone 30	0.092 (Horizontal)	677	22.7 (25.4)	-	8	Yes	6.4 (3.8)	?

Table I. HEDL Sodium-Limestone Concrete Interaction Tests (Continued)

Test Number	Concrete Type/ Thickness, cm	Surface Area, m^2 (Orientation)	Temperature Ave. °C (Max)	Sodium Mass, kg/ (Depth, cm)	Exothermic Reaction	Length of Test (hr)	Hydrogen Evolven kg	Penetration Max. cm	Noises (Pops, Bumps, etc)
LFT-4	Limestone, 30 cm MgO, 10 cm Steel, 1 cm (5 cm hole)	0.092 (Horizontal)	871	22.7 25.4	?	8	Yes	2.5/1.8	
LFT-6	Limestone, 61 cm MgO, 10 cm Steel, 1 cm (15 cm hole)	0.836 (Horizontal)	820	454/ (70)	Yes	15	Yes	7.5/5	
LSC-2	Limestone, 61 cm	0.836 (Horizontal)	475 (801)	454 (70)	Yes	100	Yes	7.5/5	
LFT-12	Limestone, 61 cm 10 cm Pearlite	0.092 (Horizontal)	593 (871)	46/?	?	48	Yes	1.3 cm + 10 cm Pearlite	
[illegible]T-1	Limestone, 61 cm (Pressure=17 ft head)	0.092 (Horizontal)	893 (871)	46 (51)	Yes	100	Yes	19 cm	
[illegible]T-2	Limestone, 61 cm [illegible] dehydrated concrete (sodium limited)	0.092 (Horizontal	593 (871)	46 (51)	Yes	70	No	32.5 cm	

Table II. Sandia Sodium-Limestone Concrete Interaction Tests

Test Number	Concrete Type/ Thickness, cm	Surface Area, m^2 (Orientation)	Temperature Avg, °C (Max)	Sodium Mass, kg/ (Depth, cm)	Exothermic Reaction	Length of Test (hr)	Hydrogen Evolven kg	Penetration Max. cm	Noises (Pops, Bumps, etc)
P1,LS1	CRBRP Limestone 30.5	0.29 (Horizontal)	550 (800)	21 (8.6)	Yes	22 min	Yes	8.3	Explosion
P2,LS2	CRBRP Limestone 38.1	1.17 (Horizontal)	550 (800)	108 (11.2)	Yes	45 min	Yes	9.1 (7.6)	Yes
P3,LS3	CRBRP Limestone 38.1	1.47 (Horizontal)	550 (740)	186 (15)	Yes	3 hr	Yes	15.2	Explosion Lrg spalled chunk
P4,LS4	CRBRP Limestone 38.1	0.65 (Horizontal)	540 (450)	188 (30)	No	4 hr (8 min)	Yes	0.5	Yes (once)
LS5	CRBRP Limestone	1.17 (Horizontal)	540 (460)a	186	No	2 hr	?	Slight (0.5)	No
LS6	LS5 reused + 36 Kg NaOH		0 (700)	186	No	- 8 hr	?	Slight (< 1)	No
LS8	CRBRP Limestone	1.17 (Horizontal)	550 (450)	127 (15)	Moderate	52 min	?	1.0	-
LS9	CRBRP Limestone	0.65 (Horizontal)	600	182 (35)	Yes	3.4 hr (5 min)	Yes-H_2 exp T + 2 min	4.5	-
LS10	CRBRP Limestone	0.65 (Horizontal)	600	182 (38)	Yes	- 25 sec	Yes-H_2 exp T + 30 sec	-	Explosion terminated exp. after 25 sec
LS18	Limestone 61 cm Flawed Liner	0.65 (Horizontal)	665/575	182 (38)	No	4.2	?	1	
LS19	Limestone	0.65	695/575	68+68/29.1cm	Yes	?	?	6.4	

Table II. Sandia Sodium-Limestone Concrete Interaction Tests (Continued)

Test Number	Concrete Type/ Thickness, cm	Surface Area, m^2 (Orientation)	Temperature Avg, °C (Max)	Sodium Mass, kg/ (Depth, cm)	Exothermic Reaction	Length of Test (hr)	Hydrogen Evolved kg	Penetration Max. cm	Noises (Pops, Bumps, etc)
SET1	Limestone	0.072	600/?	4.5/0.3	No	?	?	0	
SET2	Limestone	0.072	650/?	4.5/0.3	No	?	?	0.5	
SET3	Limestone	0.072	700/?	4.5/0.3	No	?	?	0.5	
SET4	Limestone	0.072	750/?	4.5/0.3	No	?	?	0.5	
SET5	Limestone	0.072	73/?	4.5/0.3	No	?	?	?	
SET6	Limestone + perforated liner	0.072	73/?	4.5/0.3	No	?	?	?	

Notes: a=pool set point

particularly dangerous to attempt to infer the properties of one concrete from another concrete.

Some of the earliest experiments tested very small samples of concrete, typically 1 cm or 1 inch cubes. These samples were immersed in liquid sodium. Limestone, basalt and magnitite aggregate concretes were tested. It was discovered that all three types of concrete would react with sodium and totally disintegrate after several hours. In each case a certain minimum temperature had to be exceeded before a reaction would occur. It was determined that both the cement and aggregate reacted with sodium at temperatures above the 500 to 600°C. range. These reactions were exothermic with some differences noted which depended upon the kind of aggregate used and the water content of the concrete (Reference 2-4).

A larger group of tests was conducted at HEDL using 1 ft. diameter concrete surfaces which were exposed to heated sodium (Reference 5). Usually, 22.7 kg. (50 lb.) of liquid sodium were poured on these intermediate scale specimens covering them to a depth of 30.5 cm (1 ft.). Typically, the HEDL experiments were performed at higher temperatures than those at Sandia; in a majority of the HEDL experiments the tests were conducted at 871°C.

HEDL has recently conducted two large scale tests, LSC-2 and LFT-6. In these tests, 454 kg. of sodium were poured limestone concrete specimens with a surface area of 0.84m (one square yard), forming a pool with a depth in excess of two feet. Test LFT-6 included a steel liner and a 10.2 cm. thick layer of MgO aggregate. All tests were conducted with the surface under attack oriented horizontally, except for vertical tests SC-10 and SC-12.

During the same period when the tests described above were being performed, Sandia conducted a number of experiments on a significantly larger scale. Typical experiments have involved pouring approximately 180 kg of liquid sodium on specimens with areas of about 0.5-1.5 sq.m. The sodium pool depths have generally been shallower than in the HEDL tests and have ranged from 8 to 38 cm. The pouring temperature of the sodium in the experiments has varied from 450°C to 760°C. Some experiments have included steel liners.

Historically, there appeared to be a substantial difference in the results obtained by Sandia and HEDL. In the first three large-scale Sandia tests, highly energetic reactions were observed after an initial relatively quiescent phase. The energetic reaction quickly consumed all the sodium and penetrated as far as 15 cm into the limestone concrete within 3 hrs. In the first four HEDL small-scale tests (1 ft^3) on limestone concrete, penetration of less than half this depth occurred over 24 hr with an excess of sodium. All tests were made on horizontal surfaces.

The differences in results were initially attributed to scale effects, since the surface areas in the Sandia tests ranged from 3 to 15 times greater than those of HEDL. Significantly, the sodium pool depths in the initial Sandia sodium pours were approximately half those used by HEDL. Also, the initial HEDL sodium temperatures were 300°C higher than those used by Sandia;

it was thought at the time that the higher temperatures would produce more energetic reactions.

Following these initial tests, the experimental results changed. The fourth Sandia limestone concrete test employed a 0.5 m^2 surface area and twice the previous sodium depth. A penetration depth of only 0.5 cm was observed. This was followed by six more tests that produced little penetration but some experimental difficulties, such as hydrogen deflagrations or explosions. HEDL conducted two limestone concrete tests using vertical test surfaces and observed penetrations that came close to 15 cm. HEDL observed that the penetration into a vertical surface exceeded that into a horizontal surface by a factor of two. On the other hand, Sandia observed greater penetration downward than radially.

At this point, HEDL, on the basis of its tests only, proposed that concrete reaction products were responsible for the limited interactions observed in the HEDL tests. Past examinations of the HEDL specimens showed that the upper surface of the concrete was covered by a hard, strong layer of reaction products. It was HEDL's position that, during the experiment, this layer was a viscous liquid which separated the unreacted concrete from the sodium. Under generally similar conditions, experiments with vertically oriented concrete surfaces showed greater penetration into the surface, suggesting that the viscous protecting liquid slumped under the influence of gravity thereby exposing an unprotected concrete surface to the sodium. On this basis, HEDL proposed that horizontal concrete surface reactions would not progress beyond 3 or 4 inches. The problem with this interpretation was that it did not explain the results of Sandia tests P1, P2 and P3.

Next, Sandia proposed a mechanism to explain the different results obtained at Sandia and HEDL. It was suggested that NaOH rather than sodium was primarily responsible for the observed attack on concrete. The NaOH formed when sodium reacted with water driven from the concrete by heat. It was suggested that the concrete would experience little attack during the period before the sodium became saturated with NaOH. When additional NaOH was formed, the liquid NaOH formed a separate layer, which in turn attacked the concrete. It was argued that the shallow pools used in the first three Sandia limestone concrete tests quickly saturated with NaOH allowing the energetic reaction to begin promptly. The deeper sodium pools used by HEDL, and in Sandia test No. 4, required time to reach saturation with NaOH. In addition, it was suggested that reaction products would provide some shielding from further attack for the concrete. This proposed mechanism led both Sandia and HEDL to perform tests in which NaOH was added to the liquid sodium before it came into contact with limestone concrete in order to achieve immediate saturation. Sandia's test with NaOH (test No. 6) showed a lesser reaction than before, and HEDL's tests (tests SC-13, SC-14 and SC-19) showed about the same penetration. These results certainly did nothing to confirm the NaOH attack hypothesis.

Later Sandia tests (LS8, LS18, LS19) did not reproduce the high rates of erosion experienced in the early tests. As compared to the early tests, the later ones employed deeper sodium pools, specimens with smaller areas and somewhat higher temperatures. Our interpretation of these tests has been [illegible]ered by a lack of published data about them.

More recent tests at HEDL have shown a limited penetration in rather long tests. In HEDL test LSC-2, a penetration of only 7.5 cm was observed in a 100 hour test when sodium was poured on concrete at 475°C and later maintained at several different temperatures (Reference 6). However, when a similar test (LCT-1) on a smaller scale was conducted under pressure simulating a 17 foot head of sodium, the erosion increased to 19 cm. Most of the erosion occurred during the first five hours.

HEDL's last test, LCT-2, with 80% dehydrated concrete, showed an erosion of 32 cm in 70 hours. This erosion was limited by consumption of all of the sodium and presumably would have continued further if additional sodium had been present.

3. Models For Sodium Concrete-Interactions

Two models have been developed by HEDL and Sandia to explain the results observed in the liquid sodium-limestone concrete tests conducted until now. Unfortunately, the models make totally different assumptions about the role of NaOH in the reaction process. Neither model at this time can successfully explain all of the observed results.

A. HEDL's Conceptual Model of Sodium Concrete Reactions

A model for the interaction between liquid sodium and concrete has been developed at HEDL for limestone concrete [Reference 4]. The model postulates the existence of a threshold temperature of approximately 500°C which must be attained before sodium-concrete reactions occur. The existence of a threshold is well supported by experimental evidence.

The concrete erosion in this model is controlled by a reaction product layer, composed mainly of NaOH, formed by the reaction between liquid sodium and water from the concrete. Water release from the concrete begins at temperatures of about 100°C, with the following reactions occurring as a result [Reference 7].

$$2Na + H_2O \rightarrow NaOH(l) + NaH \tag{1}$$

$$4Na + H_2O \rightarrow Na_2O + 2NaH \tag{2}$$

$$2NaH \rightarrow 2Na + H_2 \tag{3}$$

A chemical analysis of the reaction products has shown that they are 37% NaOH by weight, which suggests that reaction path (1) is favored. NaOH from reaction (1) then attacks concrete:

$$2NaOH + CaCO_3 \rightarrow Na_2CO_3 + Ca(OH)_2 \qquad (4)$$

Calcium hydroxide, in turn, decomposes at 580°C, thereby replenishing the water supply for further reactions with sodium:

$$Ca(OH)_2 \rightarrow CaO + H_2O \qquad (5)$$

Sodium aggregate reactions will not occur until the temperature level reaches a threshold value of 500°C. The following reactions are principally responsible for the degradation of concrete integrity:

$$4Na + 3Ca\ CO_3 \rightarrow 2Na_2\ CO_3 + 3CaO + C \qquad (6)$$

$$4Na + 3SiO_2 \rightarrow 2Na_2SiO_3 + Si \qquad (7)$$

A provision in the model is also made for the following reaction:

$$4Na + CaCO_3 \rightarrow 2Na_2O + CaO + C \qquad (8)$$

In HEDL's view, the NaOH produced by the aqueous reaction does immediately attack concrete but the rate of attack is very slow and is dependent on the dilution of NaOH with other reaction products. Sodium-concrete reactions are much more energetic but do not occur until the 500°C threshold temperature is attained; it is these reactions which are responsible for the degradation of concrete integrity. Further, at temperatures in excess of 700°C calcium carbonate in the limestone will decompose to yield CO_2. The reaction of CO_2 with sodium is highly exothermic.

Since HEDL views sodium concrete reactions as the major reactions responsible for concrete erosion, transport of sodium to fresh concrete is essential for concrete attack. However,

should be most extensive in concretes with little water available for sodium hydroxide formation. This has in fact been observed in test LCT-2 where the penetration, 32 cm, is the largest observed to date.

B. Comments on the HEDL Model

The HEDL model is essentially a retread of a model first proposed 3 or 4 years ago. At that time, concern was expressed about the inability of the model to adequately explain the results of Sandia tests P1, P2 and P3 in which greater than average concrete erosion was observed. In the case of Sandia test P3, this erosion reached 15 cm in a three hour test. However, in test P3 there is a possible explanation. It can be argued that the substantial erosion observed in that test was due to spallation or cracking which resulted in the separation of a large chunk of concrete, 25 cm in diameter and 7 cm thick, which was found lying above the cavity floor. The spallation could have disrupted the protective layer, thereby increasing erosion.

Another problem area is HEDL tests SC-10 and SC-12, both of which exposed vertical surfaces to a sodium pool. If reaction products are the controlling mechanism in concrete erosion, then a vertical surface should be eroded substantially since the reaction products are free to fall away under the force of gravity. In tests SC-10 and SC-12, greater erosion was observed than in horizontal tests but the erosion was not nearly so great as might be expected if reaction products are in fact the mechanism controlling concrete erosion. The observed erosion was an average of 9 cm in these tests as compared to 2.5-6.1 cm in a number of other tests. In fact, there is one horizontal test, LCT-1, with a 17 foot head of sodium, in which the erosion, 19 cm, was greater than in test SC-10 and SC-12. It should be noted that test LCT-1 was a 100 hour test whereas SC-10 and SC-12 were 8 and 24 hour tests, respectively; however, test length should not be particularly relevant in view of HEDL's assertion that nearly all concrete attack occurs within the first few hours. The point to be made is that if reaction products are a controlling mechanism in concrete erosion, then the average erosion should have been much greater in tests SC-10 and SC-12. Since the reaction products were free to fall away, erosion should have continued until the sodium supply was exhausted. This clearly did not occur.

The results obtained in HEDL tests SC-13, SC-14 and SC-19 also are of concern. In these tests, NaOH was added to the test initially. If sodium, rather than sodium hydroxide, is responsible for most of the attack on concrete, why was the attack on concrete in these tests as great as it was? Average erosions of 4.4, 5.7 and 3.8 cm have been reported, which are typical of tests where sodium hydroxide was not present initially. One might reasonably expect that the erosion would be substantially less than reported if indeed a sodium hydroxide layer is less reactive than sodium.

Finally, there is concern about the inability of sodium to wet and react with concrete directly. It has been reported in several places that sodium does not appear to wet hydrated concrete nor does it enter small cracks, presumably from a combination of surface tension effects and the counter flow of steam emerging from the cracks. The sodium hydroxide formed from the reaction of water and sodium does wet and react with the concrete surface and is responsible for the formation of the reaction product layer. It is hard to see how sodium would be able to react directly, as proposed, with a material that it does not wet.

C. Sandia's SCAM Model

A model and computer program, SCAM, has been developed at Sandia National Laboratories to describe the interactions between liquid sodium and <u>basalt</u> concrete [Reference 8]. An effort is being made to extend the SCAM model to limestone concrete at Sandia. Two "working hypothesis" have been developed; these consist of a "gas-phase model" and a "liquid-phase model". At present, this work has not been completed and is undocumented. However, it will be briefly described below.

In the SCAM model for limestone concrete, the sodium pool and concrete are divided into five regions [Reference 9]. The upper region consists of a sodium pool containing saturated sodium vapor and hydrogen bubbles. The second region consists of porous reaction products and liquid sodium. The third and fourth regions collectively constitute a "dry zone" in the concrete.

In the third region, closest to the sodium pool, sodium hydroxide and the concrete aggregate can react. Sodium vapor from the overlying pool of sodium diffuses downward passing hydrogen, CO_2 and water, which simultaneously diffuse upward through the layer.

In the upper part of the fourth region (the bottom of the dry zone), calcium carbonate decomposes due to heat, forming calcium oxide and releasing carbon dioxide. In the lower part of this region, bound water is released from the concrete.

Finally, the fifth region is separated from the others by a liquid evaporation plane and constitutes a "wet zone". In this region water migration occurs.

The gas phase reactions occurring in these regions are listed below:

$$2H_2O + 2Na \rightarrow NaOH + H_2$$

$$CO_2 + Na \rightarrow Na_2O + C$$

In the gas phase model, sodium enters the concrete pores, reacts and NaOH, Na_2O and carbon condense on the concrete aggregate. These reactions create an energy pulse in a localized zone. They are assumed to proceed at a steady rate and the model provides information on the rate of reaction but not the total extent of reaction. The model does not include a provision for either initiation or termination of concrete attack. The reactions are regarded as "secondary," following the initial set of reactions described in the liquid phase model discussed below.

In the liquid phase, the following reactions occur:

$$\text{Step 1:} \quad 2Na + 2H_2O \rightarrow H_2 + (NaOH)_{Na} \tag{9}$$

In this initial reaction, during the mild phase of the attack, sodium hydroxide is dissolved in sodium. In step 2, a sodium hydroxide layer forms when the sodium becomes saturated with NaOH:

$$\text{Step 2:} \quad (NaOH)_{Na} \rightarrow NaOH(l) \tag{10}$$

In the Sandia hypothesis, liquid sodium hydroxide is essential to the process by which concrete is dissolved. It has been suggested that the need for a NaOH layer, which will be present only at the bottom of the sodium pool, predicts an absence of sideward attack. The next step is:

$$\text{Step 3:} \quad NaOH + CaCO_3 \rightarrow (CaCO_3)_{NaOH} \tag{11}$$

Since the NaOH layer does not form immediately, the reaction of Step 3 is delayed. A delay time has been experimentally observed. In the next step, energetic attack occurs with the evolution of heat.

$$\text{Step 4:} \quad (CaCO_3)_{NaOH} + 6Na \rightarrow 3Na_2O + CaO + C \tag{12}$$

The reaction accounts for the observed free carbon found during post-test analysis.

The NaOH can be regenerated so long as water is available from the concrete, as indicated.

Step 5: $Na_2O + H_2O \rightarrow 2NaOH$ (13)

In this model, the presence of NaOH is essential for concrete attack to continue. Consequently, tests can be misleading if the tests are either water or sodium limited.

If water is not available from the concrete, there will be unreacted Na_2O present which could not react in Step 5. When this occurs, the reaction below results:

Step 6: $Na_2O + NaOH \rightarrow (Na_2O)_{NaOH}$ (14)

When all of the NaOH is gone, further $CaCO_3$ dissolution ends and the reaction terminates. The sodium oxide dissolved in NaOH will eventually precipitate out as the solution cools.

SCAM employs the above information together with the continuity equation, the momentum and energy equations, the gas laws, diffusion equations and chemical kinetics to arrive at a solution. No attempt has been made to include the effects of spalling and cracking.

D. Comments on the Sandia Model

If sodium hydroxide is principally responsible for the erosion of concrete, as proposed by Sandia, then the greatest erosion should have occurred in Sandia test LS-6, and HEDL tests SC-13, SC-14 and SC-19. In these tests, sodium hydroxide was added to the sodium at the beginning of the test in order to provide immediate saturation. However, the penetration actually observed (less than 1 cm, 4.4 cm, 5.7 cm and 3.8 cm, respectively) was only about average.

Another problem area concerns HEDL tests SC-10 and SC-12. These are the only tests conducted with limestone concrete oriented vertically. If the Sandia hypothesis is correct, then the erosion in these tests should have been less than for horizontal concrete surfaces. The reasons is that the more dense sodium hydroxide layer will be at the bottom of the sodium pool in the Sandia model. Since it is primarily responsible for erosion, there should have been little erosion in a vertical test because very little sodium hydroxide would be in contact with the concrete. However, as has been mentioned earlier, the erosion observed was greater than average by a factor of approximately two. On the other hand, it must be noted that Sandia has observed relatively little attack on the crucible sidewalls in its own experiments; this observation supports its model.

Another problem area for this model is the observed interaction between liquid sodium and predehydrated concrete in HEDL test LCT-2. Since the water content of this concrete is low, little sodium hydroxide will form and, consequently, the attack on the concrete should be limited. However, the observed penetration was the greatest of any limestone concrete test. It could be that this penetration resulted from increased porosity arising from the dehydration process.

4. Discussion

In the preceeding section, the models proposed by HEDL and Sandia were both discussed. As was indicated, both models have problems in adequately describing the experimental results. In this section, the positions purported to be held by each organization on a number of issues will be discussed and further comments will be provided.

A. Erosion of Vertical Surfaces

Sandia and HEDL have proposed models which predict the opposite results for erosion of vertical walls. It should be possible, in principle, to eliminate one of the models on this basis.

Sandia's model predicts that sodium hydroxide attack on concrete is the dominant factor in concrete erosion. A NaOH layer, because of its density will be found at the bottom of a sodium pool. If it is the dominant reactant then there should be little erosion of a vertical concrete surface because the wall will be exposed over most of its area to sodium rather than NaOH. In support of their hypothesis, Sandia cites the limited erosion of the vertical surfaces of their concrete crucible tests.

On the other hand, HEDL's model predicts that sodium-concrete reactions are terminated by the accumulation on horizontal surfaces of a passivating layer containing reaction products and sodium hydroxide. Such products would be expected to fall away from a vertical surface and thus would expose fresh concrete to attack by sodium. In support of its contention, HEDL can cite its vertical surface tests, SC-10 and SC-12, in which greater than usual concrete erosion was observed.

Each group has its own test data to support its position. It should be noted that in some of the shallow pools used by Sandia, very limited sidewall erosion would be expected because they were so shallow. Other Sandia tests have employed deeper pools. It is not clear which of these experiments were the source of the data in question. Although HEDL's vertical surface tests do show greater erosion than horizontal tests, the erosion is less than might be expected if fresh concrete is continually being exposed to sodium if the latter is the dominant reactant. It is not apparent that there would be any limit to the extent of penetration of a vertical surface, other than sodium exhaustion, under the HEDL hypothesis.

The preceding discussion suggests that it should be possible to eliminate one or the other of the models by means of a test exposing both a vertical and horizontal concrete surface to the same sodium environment. If the vertical surface is eroded to a greater extent than the horizontal surface, HEDL's model is probably confirmed. If the opposite situation occurs, Sandia's model will be confirmed. While in principle, the answer could be inferred from a comparison of various HEDL tests, the experiment to experiment variability has been sufficiently great to make such a procedure dubious. Such a comparison is best done with both surfaces present in the same sodium pool so that both will be exposed to an identical initial environment.

Hanford has indicated that they believe that vertical wall erosion is irrelevant to the CRBR. The vertical wall will be protected by a layer of pearlite concrete and a steel liner. In HEDL test SET-12, the pearlite concrete reacted completely but appeared to protect the underlying limestone concrete from extensive erosion. HEDL further points out that they do not believe that there is a scenario in which the steel liner and pearlite would not be present, thereby exposing the vertical wall to direct attack. It is possible to conceive of a situation where core debris piles up against a wall and destroys the steel liner and pearlite concrete so that the limestone concrete is exposed to sodium. However, an assessment of the probability of such an event is beyond the scope of this work.

Finally, in regard to HEDL's last contention, it should be noted that there was concern some years ago about the possibility of cracks developing in the steel liner over a period of time. There was also concern that cracks might be present initially unless care was taken in fabrication. Such cracks could provide a path for sodium to enter a liner and attack a vertical wall. Whether cracks can be present initially or develop later should be examined further.

B. Role of Sodium Hydroxide

All parties agree that substantial quantities of NaOH will form due to the interaction between liquid sodium and water driven from the concrete by heat. As just discussed, the subsequent behavior of NaOH is disputed by HEDL and Sandia. HEDL has proposed that NaOH is less reactive than sodium, and thus form a passivating layer between the sodium and concrete. On the other hand, Sandia believes that NaOH is the dominant reactant species. Evidence from vertical wall erosion has already been discussed in this regard. However, there is additional information from HEDL test LCT-2, that is relevant here.

The very recently reported results of HEDL test LCT-2 describe a test made with a sodium pour on dehydrated limestone concrete. This concrete was heated at 1000°F for 24 hours to achieve 80% dehydration. When sodium was poured on the concrete the reaction was described as being very benign with virtually no hydrogen release and little energy generation. A pour of 46 kg of sodium was totally consumed in 70 hours producing an average penetration of 13 inches. Presumably, penetration would have continued if additional sodium had been present. The penetration of 13 inches is substantially greater than the 3 inches typically found in the HEDL tests with normal limestone concrete.

Since this concrete was dehydrated, there was little water available to react with sodium to form NaOH. Thus, the extensive erosion seems to support the position of HEDL that erosion by sodium is the dominant process. However, it is possible that the observed erosion was enhanced by physical and chemical changes induced in the concrete by the dehydration process. It is not certain that the additional penetration was caused by the absence of a NaOH layer.

Another issue worthy of some exploration is the water loss observed in the Sandia tests. The larger tests conducted initially by Sandia had open outside surfaces through which, it was reported, came substantial quantities of water. With water escaping through the outside surfaces, one would expect that less sodium hydroxide would be generated thereby slowing the formation of a NaOH layer. This observation could enable one to explain some of the early energetic reactions in the Sandia tests on the basis of HEDL's model. By contrast, in HEDL's well constrained tests, water driven from the concrete could exit only through the sodium pool where it would contribute to HEDL's proposed passivating NaOH layer.

C. Nature of the Layer Between the Sodium and Concrete

As discussed previously, the layer between the sodium pool and the concrete surface consists of a mixture of sodium hydroxide and reaction products. While everyone agrees that both sodium and sodium hydroxide both can attack concrete, there is disagreement concerning the relative importance of the two reactants. In HEDL's views, sodium is responsible for most of the attack on concrete and it is prevented from reaching fresh concrete by the presence of the NaOH-reaction product layer. Thus the reaction product layer serves to limit the extent of sodium-concrete reactions as it develops. On the other hand, Sandia regards NaOH as principally responsible for the reactions with concrete. If this is true, then the only limit to the extent of reaction will be exhaustion of the reactants (sodium and water driven from the concrete). It is our purpose here to examine what is known about this layer.

One of the difficulties in studying the reaction product layer is that we can only observe it after the test is over and the material has frozen. When a test specimen is sectioned, this layer appears to be homogeneous and gives the appearance of having been a liquid. The layer looks similar to sandstone. There is no aggregate or included sodium visible. At room temperature the layer appears to have a substantial mechanical strength.

When analyzed, the layer is found to contain significant quantities of sodium hydroxide. Samples of this layer will melt at the temperature of the sodium used during the reaction. The properties of the fluid are disputed and are relevant here. HEDL believes that the liquid is viscous while Sandia believes that it flows like water, based on experiments in which some of the material was remelted. The fluid does wet the concrete and forms a layer separate from the sodium pool.

The volume of the reaction product layer is quite substantial. Contributing to this volume is the substantial increase [illegible] in the volume of concrete as it reacts with sodium hydroxide. When one observes this thick layer after a test, it is easy to believe that it would prevent contact between sodium and concrete in the HEDL model. However, its character when molten, as noted, is disputed.

It has been suggested that this layer might be displaced by convection currents in the deep sodium pool of the reactor. Other factors that could displace the layer include gas evaporation and spallation. Whether the layer is readily displaced by any of these mechanisms will depend on its viscosity, which is disputed.

Finally, it should be noted that if the view of Sandia is correct, this intermediate layer is the cause of most of the erosion of concrete. If this is true, the viscosity and displacement of the layer are not relevant in the erosion process.

D. Temperature of the Sodium

It is the view of most observers that there is a temperature threshold for sodium-concrete reactions. However, a range of values has been suggested and the threshold is probably strongly dependent on the nature of the concrete in question.

There are two difficulties in this area with HEDL's tests. First, the sodium in some of their tests has been allowed to cool to a fairly low temperature immediately after being poured on concrete. Sandia has asserted that this initial quenching process might in some manner affect the chemical process occurring and may inhibit the development of an energetic reaction. There is some merit to this point of view.

A more serious objection can be made to HEDL's conduct of nearly all experiments at a very high temperature, near the boiling point of sodium. While it is true that reaction rates usually increase rapidly with temperature, there are exceptions, which can result from competing reactions as well as in other ways. In particular, a review of Na-NaOH phase diagram data [Reference 9] suggests that the composition of the sodium pool and the development of the NaOH rich liquid layer are dependent on the pool temperature. Consequentially, the results of tests conducted at lower temperatures may be significantly different. It is suggested that some of HEDL's future tests should be conducted at lower temperatures.

E. Cracking and Spallation

The subject of cracking is highly complex in such a non-uniform material as concrete. The stress patterns in a given sample depend not only pre-existing stresses, but also on temperature distribution, the effects of chemical reactions, and external mechanical forces. With all other factors the same, the size of cracks in a sample would be expected to increase with sample size.

Cracking and spallation can provide a means for reactants to reach fresh concrete and thus can enhance erosion. HEDL has expressed the view that the effects of cracking could be serious in terms of their model. The protective layer of reaction products, which they have proposed limits the reaction, could drain away into any cracks thereby exposing fresh concrete to sodium attack.

In tests up to this time, no tendency has been observed in experiments at Sandia for sodium to enter cracks in blocks of concrete, even when there has been energetic attack. However, at some point sodium will enter a crack if it is wide enough or if the pressure is great enough.

HEDL's concern over the effects of cracking and spallation has led them to be concerned about the method of restraint of specimens in their tests. The HEDL test specimens are usually very firmly restrained in a surrounding collar of concrete which is intended to more closely simulate the reactor cavity floor. By contrast, the Sandia crucibles have been unrestrained and may consequently be more prone to cracking. This does not necessarily make the Sandia tests unrepresentative since the reactor cavity clearly does have corners. The Sandia tests should represent the corners of the reactor cavity better than HEDL's and HEDL's tests should provide a better representation of the center region of the cavity floor. The effect of rebars on cracking and spallation was examined in HEDL test LSC-2. Some local cracking was observed due to differential thermal expansion but it was concluded that rebars would not greatly increase cracking.

HEDL believes that any spallation that occurs will tend to be limited to the region close to the top surface of the concrete. While a spalled layer may spall again, they believe that spalling is a surface phenomenon which will not proceed indefinitely. When the thermal gradients in concrete level out, they feel that spallation will end.

Sandia has observed the appearance of a large chunk of concrete in one test, P-3. The chunk was approximately 25 cm in diameter and 7 cm thick. Its origin is uncertain but it probably separated from the main mass of the concrete crucible by either cracking or spallation.

Spalling also is a scale dependent phenomenon. Certainly, one would not expect to see a chunk of material of the size produced in Sandia test P-3 in a small scale test. It is by no means certain, in our view, that any tests conducted to date have been on a scale sufficiently large to demonstrate conclusively that extensive cracking and spallation will not occur. Unfortunately there is currently no way to make theoretical predictions regarding either cracking or spallation that would be meaningful.

The method of restraint probably should be a variable in some tests in order to assess its importance in promoting cracking and spallation processes. This type of information could provide some data that might be useful in assessing the extent of scaling effects.

F. Effect of Core Debris

The are at least two scenarios for the attack of core debris on concrete. In one, the core debris does not form a coolable debris bed and melts into the concrete. The formation of a crust on top of the core debris could prevent cooling once melting is initiated. The development of crusts in the presence of coolant has been observed in experiments with water and molten metals attacking concrete in experiments conducted by Dr. Peehs in Germany [Reference 10]. Data developed in experiments at Sandia [Reference 11] suggests that the erosion rate for steel on concrete at 1700°C could be as high as 25 ± 15 cm/hr. For oxide fuels at 2800°C, an erosion rate of 130 ± 50 cm/hr has been measured in tests.

Even if the core debris is initially coolable, it will tend to sink into sodium-concrete reaction products due to its greater density. It is not clear that initially coolable core debris mixed with concrete reaction products would remain coolable. Once melting begins, crust formation could prevent cooling by sodium. Substantial concrete erosion could result, either because of direct attack by core debris or exposure fresh concrete to either sodium or sodium hydroxide.

In view of the potential for substantial erosion of concrete induced by core debris, it is urged that studies should be initiated to review the feasibility of core retention devices constructed from refractory materials such as MgO. Emphasis should be placed on the selection of materials that are compatible with sodium.

C. Reactant Limited Tests

Most of the tests conducted by both HEDL and Sandia have been reactant limited. In this context, both sodium and water form the concrete reactants.

Sandia argues that HEDL's tests should have used thicker concrete specimens so that additional water would be driven from the concrete into the sodium pool where it could react and form sodium hydroxide. If Sandia's model is correct, the added NaOH would increase erosion of concrete. On the other hand, if HEDL's model is correct, the added NaOH would have little effect. This suggests that a comparison of tests with identical conditions except for concrete layer thickness could provide evidence for or against Sandia's model.

HEDL argues that Sandia's tests have been sodium limited and that erosion would have stopped by itself soon after the Sandia tests ran out of sodium. Clearly, the Sandia tests were sodium limited and more sodium should have been available because the determination of the total extent of penetration is an important issue.

Arguments have been made in the past against those Sandia experiments that used shallow pools. It is felt that pool depth should be a variable. Our understanding of the phenomena in these tests is limited and the investigation of the influence of various parameters, such as pool depth, should not be precluded.

For example, Sandia has tried to explain the delay in the onset of the energetic reactions that they observed in terms of pool depth. If the concrete is principally attacked by NaOH, rather than by sodium, a longer time will be required for attack to occur in a deeper pool. A longer time is required in order to saturate the sodium with NaOH when the pool is deep. The NaOH has only limited contact with the concrete until the pool is saturated. At that time, a separate NaOH layer forms in contact with the concrete allowing the attack to proceed expeditiously in the Sandia model. In the Sandia shallow pool test, this saturation would have occurred relatively quickly, leading to rapid penetration. In tests with deeper sodium pools, a longer time should be required for saturation.

H. Erosion Rate and Total Penetration

The greatest total penetration observed with hydrated concrete occurred in the recent HEDL test LCT-1. The test employed 46 kg of sodium covering a 60 cm (two foot) thick layer of limestone concrete with a diameter of 35 cm (13.5 in.). The test lasted for 100 hours but HEDL believes that nearly all of the erosion occurred in the first 5 hours.

A similar total penetration was observed in Sandia test P-3 several year ago. An erosion of 15.2 cm was observed in a test that lasted only 3 hours. Whether erosion would have continued is unknown.

As discussed earlier, the HEDL model predicts a self-limiting reaction in which reaction products, mixed with sodium hydroxide, prevent sodium from directly attacking concrete. The apparent termination of attack in the 100 hour tests at HEDL supports the hypothesis that reaction products can limit concrete erosion.

On the other hand, the Sandia model explains these results on the basis of the test being limited in an essential reactant, water. The Sandia model would have the NaOH continue to attack concrete until all of the NaOH is consumed. With the relatively thin concrete layer employed (compared to the quantity of concrete available in CRBR), the attack is viewed as limited by a lack of water that can be driven from concrete.

For CRBR, HEDL has indicated that they expect that the total concrete erosion would not exceed 6-9 in., based on their worst case test, LCT-1. Sandia on the basis of its model has calculated that the water present in the CRBR concrete could support an erosion of about 30 in.

The problem with accepting the total penetration that HEDL has proposed is that it is based on a small number of empirical observations from experiments that are much smaller in scale than the conditions that would exist in an actual accident. If the erosion process was better understood or if experiments could be conducted on a scale similar to reactor accident conditions, then the proposed total penetration could be more readily accepted.

In view of the uncertainties imposed by scale effects, it is our view that the most appropriate limit for sodium penetration into concrete is that imposed by exhaustion of water, or 30 in. of concrete. It must be noted, however, that even this value may not be conservative, given the data base and controversy over the basic phenomena.

There is even less reliable data to use as a basis for selection of an erosion rate. An initial rate of 2 in/hr has been suggested by HEDL based on their experimental observations. Other have suggested a 7 in/hr rate for 3 hours with a 1 in/hr rate thereafter. Given the scale effect uncertainties, a rate as low as 2 in/hr is difficult to support. On the other hand, the 7 in/hr rate may be overly conservative. There is no really good basis for making a choice, given the kind of data available and the controversy over the phenomena responsible for erosion. If it is necessary to select a rate, then the more conservative 7 in/hr rate for 3 hours, followed by a 1 in/hr rate seems a better choice.

It is also our view that the interaction between core debris, reaction products and concrete may be potentially quite serious as discussed earlier. Core debris-concrete interactions could cause very large erosion rates. For this reason the feasibility of a core retention device that is compatible with sodium should be examined.

1. Comments on Past and Future Experiments

Additional experiments should not be initiated at those laboratories that are behind in documentation of experiments until the backlog of undocumented work has been eliminated and the data have been made generally available. One difficulty encountered in the preparation of this report was a lack of documentation for experiments particularly those conducted at Sandia. There does not appear to be a consistent scheme for numbering the Sandia experiments so that confusion can be avoided. HEDL has generally documented its experiments quite well and has prepared an excellent review [Reference 4].

Experimentalists in the past have failed to obtain all the information that could be obtained from their experiments. In view of the controversy regarding the erosive processes, this is unfortunate. Information from specimens taken in the reaction product layer and in the concrete (particularly at the interface) could be helpful in developing an understanding of the interactions occurring. In addition to the conventional chemical analysis, specimens should be studied in an optical microscope and in a scanning electron microscope. Instrumental analysis, by ion microprobe mass analysis, X-ray diffraction and other techniques should be possible for concrete specimens, if not in the reaction products due to the presence of sodium. An example of a microstructural examination of concrete from sodium-concrete interface can be found in Reference 2.

We would like to be able to say that additional experiments should be performed to improve our understanding of the phenomena. However, in view of the limited progress that has been made towards resolution of the issues in this area over the past few years, it is difficult for us to be optimistic.

The issue of scale effects is especially difficult. It is not financially feasible to conduct a statistically acceptable number of experiments of a size that precludes significant scaling effects.

A few limited areas for tests are suggested. Elsewhere, a vertical concrete wall test was suggested since the models predict different results for this configuration. Future tests by both laboratories should have an adequate supply of sodium and a sufficiently thick concrete layer to properly represent water release from concrete in the CRBR.

Separate effects test to examine the effect of both sodium and sodium hydroxide on the cement, aggregate and other constituents would be useful. Dehydrated concrete probably should also be included. The purpose of these tests would be to isolate the relevant chemical reactions and establish the relevant chemical processes. It should be possible in principle to resolve issues concerning chemistry in small scale tests.

Finally, there is a need for greater cooperation and coordination of work between the two groups to avoid the current pattern of a scattered group of experiments that cannot be related to each other.

J. Selection of a Model for Sodium-Concrete Interactions

Discussion of this issue has been delayed until this point because there currently is no confirmed model that can reliably predict the extent of sodium concrete interactions.

Although a number of experiments have been conducted to study liquid sodium-limestone concrete interactions over the last 7 years, the situation really has changed little since our earlier review in Reference 1. Many of the same arguments are being made and the same points of view are being advocated by the same parties. The inconclusive nature of the experimental data lends itself to varying interpretations. As a consequence, reasonable people can disagree as to the conclusions that can be reached in interpreting the experimental results currently available.

As the discussion in the preceding sections has shown, arguments can be made for and against each of the two models. Neither model is supported by a preponderance of evidence. Consequently, it is not possible at present to select one and eliminate the other. Of the two models, the HEDL model currently seems to best fit the available data. However, there are observations and problems noted elsewhere in this report, which are not adequately explained by it. Additional experimental data may change this conclusion. Several years ago, the Sandia model seemed to fit the data better on the basis of the data available then.

In addition to the concerns raised earlier regarding the HEDL model, there are two others. First, the model remains speculative. Reaction products probably do tend to inhibit concrete erosion. Unfortunately, although this model was proposed some time ago, progress towards a resolution of outstanding issues has been limited. The proposed chemistry remains unconfirmed and the properties of the reaction product layer are not understood. Some possibilities that might be considered include studies to confirm the proposed chemical reactions and examine their chemical kinetics, determination of the viscosity of the proposed viscous protective layer and investigations of the permeability of the layer to sodium. It is recognized that it may be impractical to conduct such studies in a high temperature sodium environment; on the other hand, some information on sodium properties does exist in the scientific literature which suggests that studies are possible. Given the controversy that has surrounded this area, some confirmation of the elements of the HEDL model is needed before it will be generally accepted.

The second concern with regard to attempting predictions of sodium-concrete behavior based on HEDL's model concerns the effects of scale, especially in regard to spallation and cracking. Neither the HEDL or Sandia models account for either. HEDL believes that cracking and spallation will be limited in a reactor cavity. This may in fact be true, but there is evidence from Sandia test P-3 that at minimum raises questions about that assumption. Given the present state of knowledge, it seems possible that cracking and spallation will be limited in an accident in a reactor cavity, but we cannot be certain on the basis of the evidence currently available.

In conclusion, it is not possible at present to conclude that either of the models is significantly better than the other one. It is also unlikely that this situation will change in the near future. Since the issues probably cannot be adequately resolved in a time relevant for CRBR, strong consideration should be given to the use of alternate mat rials in the reactor cavity. The use of high alumina cement o mag sia might reduce the threats posed by sodium-concrete reactions. A program should be initiated to examine alternative materials as soon as possible.

K. Hydrogen Generation

As mentioned earlier, the chemistry of sodium-concrete reactions is the subject of considerable controversy. Until the chemistry is understood, there will be no way to adequately predict the extent of hydrogen production.

The problem is further complicated by the potential for a reaction between sodium and hydrogen, forming NaH, at lower temperatures. Above 500°C, the equilibrium favors dissociation back into sodium and hydrogen. In addition, NaH will react with any water present over a wide range of temperatures, thereby regenerating hydrogen.

5. Conclusions

After reviewing sodium-limestone concrete interactions, we have reached the following conclusions:

1. A total sodium-limestone concrete penetration of 30 inches is recommended for the CRBR. There is no basis for recommending an erosion rate.

2. The chemistry of sodium-limestone concrete interactions remain speculative. Hypotheses have been proposed that seem reasonable but are as yet unconfirmed. The role of sodium hydroxide has not been established.

3. Sodium pool temperature may affect the chemical reactions that occur and the reaction products that form. It is not clear that a lower temperature sodium environment is more benign than a high temperature environment.

4. Sodium pool depth may affect the time required for the occurrence of a reaction and the extent of reaction.

5. Dehydration of the concrete may affect the extent of sodium penetration.

6. Reaction products may limit the extent of concrete penetration.

7. The physical aspects (cracking, spallation) of sodium-limestone concrete interactions are poorly understood.

8. The questions of the effects of scale, geometry and mode of restraint remain unresolved.

9. The interactions between core debris, reaction products and concrete are potentially serious and require additional research. As a consequence, the possibility of using other materials to protect the concrete from sodium and core debris should be assessed in the near future.

10. In view of the limited progress that has been made in understanding sodium-concrete interactions during recent years, it is probable that the controversies will not be resolved in a time that is meaningful for CRBR.

11. Future tests should employ sufficient sodium and concrete so that they will not be either sodium or water limited.

12. A comparison of erosion on horizontal and vertical surfaces in a single test may provide a test of the two proposed models for sodium-concrete interactions.

13. Greater emphasis should be placed on post-test analysis of specimens from the concrete and reaction product layers.

14. All experiments conducted to date should be fully documented before additional work is initiated.

15. Further research is needed on both chemical and physical aspects of sodium-concrete interactions. Given the differences between the experimental conditions, it is difficult to coherently correlate the results. More detailed coordination between the laboratories in selecting experiments and experimental paramaters for study would be desirable.

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A.5 RADIOLOGICAL ASSESSMENT

This section presents the staff's evaluation of the applicants' postulated core disruptive accident (CDA) scenarios as presented in CRBRP-3 (Reference 1) with regard to the calculation of radiological consequences. The evaluation begins with the path of the radionuclides from the disrupted core to the environment and then looks at the dose calculations and their sensitivity to alternative circumstances. Our review and evaluation have been conducted within the framework of the general guidelines discussed in Section A.1. More specifically we have considered whether the realistically evaluated doses resulting from venting containment after CDAs are likely to exceed the dose guidelines of 10 CFR 100.

Whether specific radionuclides are released to the environment depends on the mode of their release to the containment atmosphere, on the conditions of the containment atmosphere, and on the containment system mode of operation during the period the radionuclides are in the containment atmosphere available for release. Figure A.5-1 shows the relative timing of these conditions for the core disruptive accident which the applicants have designated their base case in CRBRP-3 (Reference 1). The approximate timing for the staff's realistic upper bound case would have hydrogen ignition at about eight hours, venting at about 22 hours and boildry at some time greater than 70 hours.

The applicants have modeled the reactor and containment structures for calculations with the CACECO code. The results of CACECO code calculations form a significant part of the basis for their scenario of conditions and events within the Reactor Containment Building (RCB) following the time of the initial release from the CDA.

A.5.1 Modes of Containment Atmosphere Release

The release from the containment atmosphere has three design modes, all through efficient filtering systems. The first is design basis leakage, a function of pressure but established as 0.1 volume percent per day at 10 psig, with an accompanying bypass (i.e., unfiltered) leakage one-one hundredth as large. The principal (non-bypass) part of the design basis leakage, leaks into the annular space between the steel containment shell and the confinement structure. When there is a non-routine release of radioactivity to the containment atmosphere, the signal from radiation monitors initiates containment isolation and an increase of flow in the annulus ventilation from about 3000 cfm to 14000 cfm. The flow is then drawn through the annulus filtration system. About one quarter of the filtered air is exhausted to the environment from the top of the reactor containment building and the remainder is recirculated to the annulus. The air flow exhausted to the environment is that amount needed to maintain the annulus at a slight negative pressure (1/4 inch water gauge) so as to assure capture of leakage from the steel shell containment.

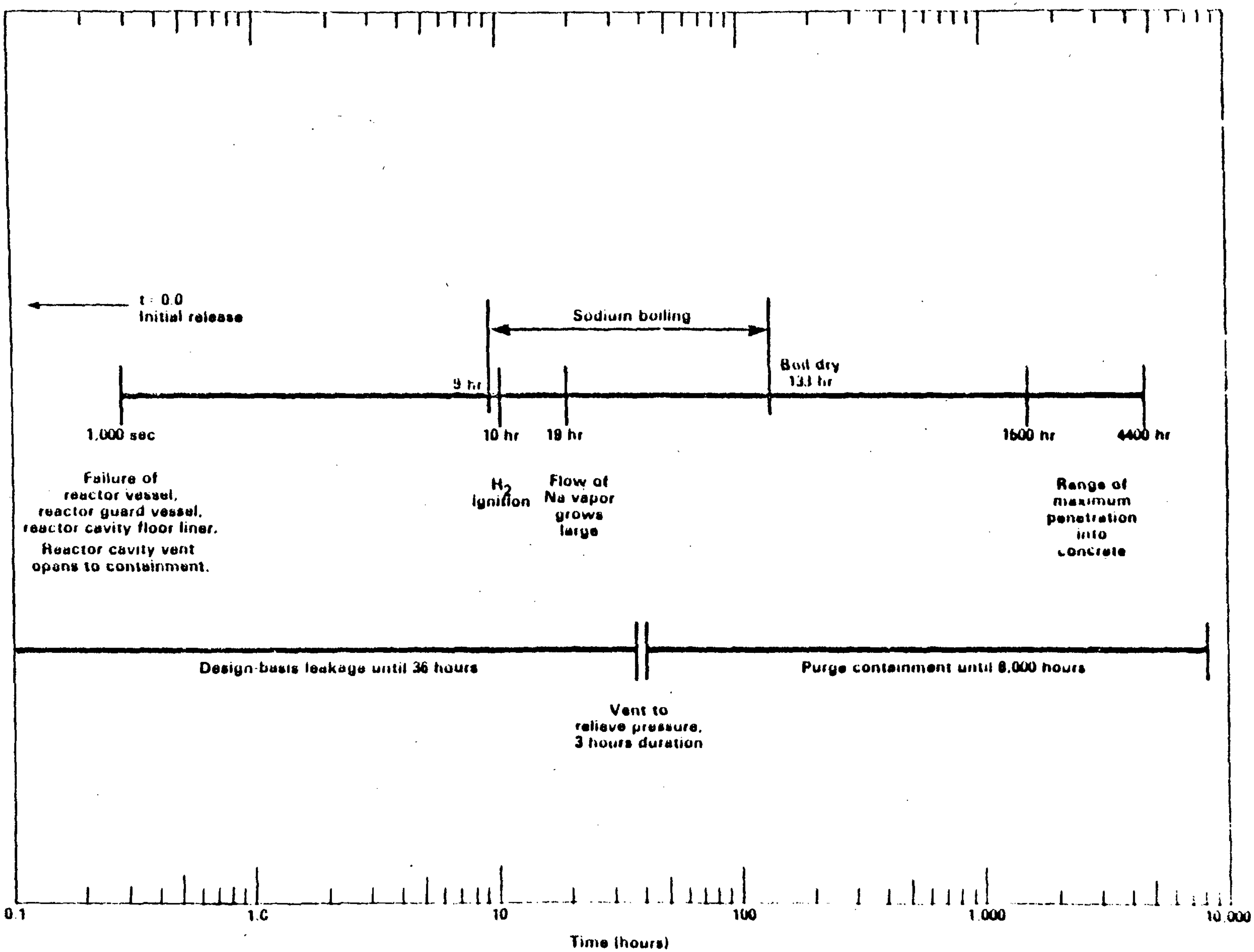

Figure A.5.1 Chronology of CDA release model applicants' base case

The second mode of release, for beyond-the-design-basis events only, is by controlled venting of the steel shell containment. When the operator decides to vent, he opens the vent system isolation valves. The decision to vent may be based on either high pressure or hydrogen build-up. Venting occurs at a controlled rate through a wet filtration system located outside the containment building. When the controlled venting is initiated, the annulus filtration system is turned off. The exhaust from the containment cleanup system is released to the environment from the top of the RCB.

The third mode of release, also for beyond-the-design-basis events only, occurs after venting drops the pressure of the atmosphere in the steel shell containment to ambient atmospheric. Then, in order to maintain low hydrogen concentration, the containment atmosphere may be purged by admission of outside air into the containment. Simultaneously, the containment atmosphere is exhausted through the containment cleanup system (the wet filtration system). This exhaust is also released to the environment from the top of the RCB.

A.5.2 Sequence of Conditions Within Containment

The postulated CDA also initiates a sequence of conditions within the subcompartments of the containment building which determine the source term for the radiological releases. Initially there is a period in which there can be some releases to containment atmosphere (especially the noble gases) without large concentrations of aerosols. When sodium boiling begins and the concentration of sodium vapor in the vented gases becomes large enough, the vented gases ignite, forming a large flame at the vent exit in the containment. Then, as sodium boilup progresses, the reactor cavity and pipeway cells are flushed of their initial atmospheres and are heated so that eventually there occurs a strong flow of sodium vapor (with hydrogen from the reaction of sodium and water from the concrete) which is vented to the containment atmosphere.

The burning of this large sodium vapor flow fills the containment atmosphere with a large concentration of aerosols, primarily of sodium oxide and other sodium compounds. This atmosphere is kept well mixed by the large flame. Within a short time the containment atmosphere reaches a quasi-equilibrium in which the rate of depletion of the aerosols by fallout and other processes is about equal to the rate of aerosol production from the sodium vapor injection. This continues until the sodium pool is boiled dry, after several days. This period, with a large flow of sodium vapor to the containment atmosphere, is perturbed by the ventdown of containment pressure. This pressure relief causes a flashing of sodium. Thus, for a few hours, a greater flow of sodium vapor occurs, followed by a few hours with a diminished flow. The applicants estimate that about three quarters of the 500 metric ton sodium pool will be boiled up to the containment. The staff concurs that this is a reasonable amount of the sodium to be involved.

At the time of sodium boildry, the remaining materials will begin to heat up due to decay heat from the core debris. The staff expects that thereafter there is a release of most of the remaining fraction of the materials that are only somewhat less volatile than sodium.

After boildry, the staff considers that the pool of core materials and sodium-concrete reaction products will heat up until it is further diluted with melted concrete and products of continuing core debris reactions with the concrete. During this period there is a greatly reduced flow of aerosols to the containment atmosphere, and the aerosol composition is changing. Eventually, the aerosols consist only of particles of the least volatile materials, transported by the steam, carbon dioxide and other gases (possibly including hydrogen and carbon monoxide) and vapors resulting from the reaction of the molten pool with the underlying concrete. Over a period of several months, the molten pool decay heating rate diminishes and the pool approaches a stable condition; aerosol release to the containment atmosphere gradually decreases to zero.

About 11 months after the CDA, the conditions within the RCB may become static, permitting purge termination.

A.5.3 Radionuclide Release Groups

At the time the core is initially disrupted, the most volatile radionuclides will be released from the disrupted core materials into the reactor vessel. The noble gases Kr and Xe will pass through the sodium to the cover gas. Other volatiles, the I, Br, Cs, Rb, Se, Te and Sb, will be released into the primary sodium. When a CDA occurs, it may involve the whole reactor (fuel, blankets and other materials) or perhaps, for example, only one third. For the purposes of radiological assessment, it is usually conservative to consider the whole reactor involved; e.g., release of 100% of the noble gases is considered to occur at once. The applicants' base case release model also includes an initial release of 1000 lb sodium containing 100 ppb plutonium, plus 0.026% of the reactor inventory of fuel radionuclides, solid fission products and halogens. This initial release is considered to pass through the head seals at the beginning of the CDA, t = 0.0.

Upon failure of the reactor guard vessel, the sodium enters and reacts with the reactor cavity atmosphere, causing the reactor cavity venting rupture discs to open, and the Kr and Xe may be vented into the containment atmosphere. The staff considers that, alternatively, they may be trapped in the PHTS to be released later. In either case, it is conservative to assume that they are released to the containment atmosphere at the initiation of the CDA where the Kr and Xe will be available to be leaked to the environment in accordance with the design basis leak rate until the time of venting, when a major fraction would be released to the environment. The remainder would be released with the subsequent purge flow, within the next 24 hours.

One hundred percent of the volatile halogens, iodine (I) and bromine (Br), are assumed released into the primary sodium at the beginning of the CDA. Their fractional release to the containment atmosphere is modeled by the applicants as being directly proportional to the fraction of sodium boiled up into the containment. The staff considers this a conservative approach because the halogens would tend to be retained in the sodium pool, providing time for appreciable radioactive decay of most halogen radionuclides. Further, most of the sodium is boiled up to containment after the venting initiation time, so that the mode of release of the accompanying iodine to the environment is by the TMBDB vent and purge flow, which passes through the TMBDB cleanup system before release.

The elements other than the noble gases and halogens which are considerably more volatile than sodium are cesium (Cs) and rubidium (Rb). When released from the core materials at the beginning of the CDA, the Cs and Rb would be likely to be dissolved in the sodium. The applicants have, however, modeled their release as an immediate 100% release to the containment atmosphere. This approach results in the Cs and Rb being considered available for design-basis-leakage release until they are depleted from the containment atmosphere by fallout. The following discussion shows this to be a reasonable approach. The Cs and Rb dissolved in the sodium, because of their greater volatility, will be released from the sodium pool when the sodium begins to boil. However, the factors which delay a large flow of sodium to the containment atmosphere will also delay the Cs and Rb. The staff considers that they may be released to the containment atmosphere at about the time a strong flow of sodium vapor first arrives there, at 19 hours in the applicants' base case, about 17 hours in advance of venting. The Cs and Rb enter at the vent flame and are oxidized along with the sodium. They will then be depleted with the rapid fallout due to the large sodium aerosol concentrations.

Other volatile fission products are selenium (Se), tellurium (Te) and antimony (Sb); these are considered dissolved in the primary sodium at the beginning of the CDA. Their fractional release to the containment atmosphere is modeled by the applicants as being directly proportional to the fraction of sodium boiled up into containment, 100% all together. These elements are generally less volatile than sodium and the staff considers that they would largely be retained in the sodium pool until about the time of boildry; at that time when the remaining mass of core debris and reaction products heats up, they would be among the first to be released.

The applicants have taken the approach of modeling the release of Ba, Sr, and all other non-volatile fission products, such that one percent of the inventory of each is released to the containment atmosphere along with the boiled up sodium. The elements, Ba and Sr are only slightly more volatile than the uranium and plutonium oxides and the remaining fission products. Because of their low volatility, little will be released to the containment atmosphere. That part not released into the sodium will eventually be available for sparging from the pool of core debris, molten concrete and reaction products. The staff considers that appreciably less than one percent of the Ba and Sr would be sparged to the containment atmosphere, and therefore the applicants' approach may be regarded as reasonable.

The largest group of radionuclides are those which can be classified as least volatile. Two modes of release to the containment atmosphere are considered of possible significance for these solids. At the time of core disruption, when molten or vaporized core materials are quenched in the primary sodium, they fragment into particles ranging in size downward from about one millimeter diameter. An appreciable fraction, estimated at 15%, may form particles so small that they remain in suspension in the sodium. A small part of this material, about one part in one thousand, is carried over with boiled up sodium to the containment atmosphere. The other mode of release is by gas sparging of the post-boildry pool of core debris, molten concrete and reaction products. In this mode, gases released from the underlying concrete reaction zone, bubble through the molten pool and entrain a small fraction of the solids, which then is carried to the containment atmosphere. The staff considers the applicants' modeling approach, which releases 1% of these solids to the containment with the boiled up sodium, to be conservative.

Plutonium is among the most radiologically significant of the radioactive materials in the reactor, present as plutonium oxide in the core fuel and in the blankets. Plutonium oxide and uranium oxide have low volatility and their release fractions will be among the smallest. Their releases to containment are modeled by the applicants as 0.015% carried over with the sodium vapor during sodium boiling and another small fraction, much less than one percent, sparged from the post-boildry pool of materials and carried up with the gases released from the concrete. For plutonium, the applicants estimate that these amount to about 320 grams and 26 grams, respectively, that are released to the containment atmosphere.

The staff considers that the applicants' estimates of plutonium releases to containment involve the principal uncertainties in the radiological consequences. The estimates consist of only the above 320 grams and 26 grams, plus about 546 grams in the 0.026% initial release of fuel and about [illegible] grams in the initial release of 1000 lb sodium. These are each small fractions of the core inventory of roughly 2.1 million grams; an error in the retention factors could impact estimated consequences significantly. The staff believes that the most significant area of uncertainty is the amount of plutonium carried up with vapor from the boiling sodium. The experimental results of Jordan and Ozawa

(Reference 2) are in general directly applicable and show partitioning at the boiling surface of more than a factor of 1000. However, the staff considers that the sodium boiling rate (Reference 3), and potential chemical differences between experiment and CDA circumstances, which may influence the formation and release of sodium plutonates (Reference 4), introduce substantial uncertainty into the estimate of the 320 grams of plutonium boiled up with the sodium. Based on review of References 3 and 4, the staff believes this uncertainty to be not more than a factor of ten. Therefore, the staff has used 0.16% of the inventory of fuel radionuclides and non-volatile fission products as the fraction boiled up to containment with the sodium. This fraction is considered large enough to also account for the minor amounts of the same radionuclides that might be sparged up after boildry.

A.5.4 Calculation of Radiological Consequences

The applicants have presented doses calculated with computer codes HAA-3 and COMRADEX (Reference 1). The release of radioactivity to the environment depends on the concentration of radioactive materials in the containment atmosphere and upon the rates at which they are added and removed. The applicants have calculated the time dependent suspended aerosol concentration in the containment and the rate of aerosol depletion with the computer code HAA-3, taking into consideration the source generation rate, the aerosol deposition rate, and the rate of removal by leakage, venting or purge flow. The output of the HAA-3 code serves as input to the COMRADEX code. The staff has found that the HAA-3 code tends to overestimate the suspended aerosol concentration and therefore provides conservative estimates of the amounts leaked.

The applicants have used the COMRADEX computer code to calculate radiation doses. COMRADEX includes radioactive decay within containment in the calculation of the rates of release of radioactivity to the environment. Using 50% frequency atmospheric dispersion parameters, the applicants used COMRADEX to calculate radiation doses at the exclusion area boundary and at the low population zone boundary. The calculation includes doses from direct gamma shine, inhalation of radioactive material and submersion in the radioactive cloud. Table A.5-1 lists some of the data and assumptions used in the dose calculations. Table A.5-2 presents the calculated doses; as shown the applicants calculated CDA doses, based on use of best estimate rather than conservative assumptions, are smaller than the 10 CFR Part 100 dose guidelines for design basis accidents.

A.5.5 Comparison of Dose Calculations

In Table A.5-2 the doses from the CRBR Site Suitability Source Term calculation are shown with the applicants' TMBDB base case doses.

The Site Suitability Source Term (SSST) results are from Reference 8. The SSST consists of 100% of the core inventory of noble gases, 50% of the iodines, and 1% of the solid fission products and plutonium; all are released instantaneously into the containment atmosphere, in a non-mechanistic manner. Releases to the environment and doses were calculated conservatively, e.g., using 5% meteorology, and the calculation considered only the design basis leakage release mode, because the calculation was made to satisfy the requirements of 10 CFR Part 100.

Table A.5-1 PARAMETERS IN DOSE CALCULATIONS [a]

Power Level, MWt - 975		
Core Inventory - End of Equilibrium Cycle		
Initial Plutonium Composition - FFTF Grade		
Containment Volume, ft^3	3.6×10^6	
Containment Leak Rate, %/day at 10 psig	0.1	
Bypass Fraction	0.001	
Filtration Efficiencies, %		
Particulates	99	
Chemically Reactive Vapors	97	
Flow Rates		
Annulus Filtration System, scfm	14,000	
Recirculated, scfm	11,000	
Exhausted, scfm	3,000	
Containment Release Cleanup System		
Venting, cfm	24,000	
Purging, cfm	17,000	
Atmospheric Dilution Factors, 50% X/Q (sec/m^3)		
Exclusion Area Boundary, 0.42 miles	Applicants' Values	Staff's Values
0-2 Hours	1.01×10^{-3}	1.3×10^{-4}
Low Population Zone, 2.5 miles		
0-2 Hours	1.59×10^{-4}	(1.1×10^{-5})
2-8 Hours	2.30×10^{-5}	1.1×10^{-5}
8-24 Hours	3.58×10^{-6}	1.0×10^{-5}
1-4 Days	2.29×10^{-6}	8.0×10^{-6}
4-30 Days	2.60×10^{-6}	5.7×10^{-6}

a. From References 1 and 5.

Table A.5-2 Radiological Consequences

	10 CFR 100 Dose Guideline[b]	Doses in rem[a]: Staff's SSST Doses[e]	Doses in rem[a]: Applicants Base Case[c]
2 Hour Exclusion Area Boundary			
Bone Surface	300	31[d]	0.19
Red Marrow	—	24[d]	0.040
Bone (total)	—	10	—
Lung	75	0.4	0.032
Liver	—	1	0.060
Thyroid	300	12	0.020
Whole Body	25	0.6	0.82
30 Day Low Population Zone Boundary			
Bone Surface	300	27[d]	0.95
Red Marrow	—	2[d]	0.19
Bone (total)	—	9	—
Lung	75	0.4	1.6
Liver	—	1	0.36
Thyroid	300	7	85
Whole Body	25	0.3	2.1

Notes:

a. Bone surface and marrow doses calculated with dose conversion factors from NUREG/CR-0150 (Reference 6), all others from NUREG-0172 (Reference 7).

b. For comparison purposes. These are the 10 CFR Part 100 dose guidelines as supplemented for CRBRP. As specified in "Site Suitability Report in the Matter of Clinch River Breeder Reactor Plant," NUREG-0786, June 1982 (Reference 8), there is an additional guideline of 34 rem whole body mortality risk equivalent. The requirements of 10 CFR Part 100 dealing with these dose guidelines apply only to accidents within the design basis and not to TMBDB cases.

c. Reference 5.

d. Reference 9.

e. Reference 8.

A.5.6 Sensitivity to Time Venting Initiated

Due to the uncertainties associated with the potential rate of sodium-concrete reaction, the applicants have assessed several cases involving different rates. Bounding these is the Margin Assessment Case (Reference 10), a case based on the venting initiation time 10 hours after the beginning of the CDA and uses sodium-concrete reaction rates beyond any experimentally observed rates. The time for venting initiation of 10 hours is predicated on the basis of allowing 10 hours for the decision to recommend protective actions offsite and for implementation of those protective actions.

For the Margin Assessment Case, in order to achieve projected conditions within the containment which could call for venting initiation at 10 hours, the applicants assumed a very conservative sodium-concrete reaction rate of 7 inches of concrete per hour for 3 hours, followed by 1 inch per hour until sodium boildry.

Table A.5-3 presents the applicants' calculated results, comparing the Margin Assessment Case to their base case; the applicants' base case assumes one-half inch per hour for 4 hours. The Table shows that the containment conditions do not differ greatly from the applicants' base case. The applicants assert feasible modifications can increase the design margins so that the plant design will accommodate the Margin Assessment Case. Table A.5-4 presents calculated radiological consequences of these two cases. The staff's evaluation of these results and those for intermediate reaction rates and venting times is that these results are reasonable and that the radiological consequences are relatively insensitive to venting initiation times between 10 and 36 hours.

A.5.7 Sensitivity to Initial Release

The applicants' base case includes an initial release to containment at the beginning of the CDA of 1000 lb sodium containing 100 ppb plutonium, plus 0.026% of the reactor inventory of fuel radionuclides, solid fission products, and halogens. The applicants have also calculated radiological consequences for a selection of such initial releases, ranging from none up to 50% of fuel and fission products, and including sodium releases ranging up to 7000 lb. In each case, the containment system is considered to perform as designed. Table A.5-5 presents some of the bone surface doses calculated by the applicant (Reference 5); the bone surface dose is the only organ dose that exceeds the 10 CFR Part 100 dose guidelines, and then only in the 50% release case, a physically unrealistic case. The narrow range of the bone surface doses is the result of the small leak rate of the containment combined with the rapidity with which the source, suspended in the containment atmosphere, is depleted by fallout and plateout. The initial release to containment is almost totally depleted before the time of venting, due to the rapid depletion rate that results when large masses of aerosols are injected into the containment atmosphere.

TABLE A.5-3

SUMMARY OF APPLICANTS' MARGIN ASSESSMENT CASE RESULTS

	Base Case	Margin Assessment
Initial Hydrogen Ignition		
Time (hrs.)	10.0	1.4
RCB Atmosphere Temperature (^oF) (before/after)	120/845	145/570
RCB Pressure (psig) (before/after)	2.2/22	2.4/14
Hydrogen Concentration (Vol. %) (before/after)	4.5/0.0	2.5/0.0
Initiation of RCB Venting		
Time (hrs.)	36	10
RCB Atmosphere Temperature (^oF)	617	710
RCB Steel Shell Temperature (^oF)	400	390
RCB Pressure (psig)	13	19
RCB Hydrogen Concentration (%)	0.0	2.6
RCB Oxygen Concentration (%)	8.4	7.4
Maximum Conditions During Venting		
Maximum Venting Rate (CFM)	24,000	28,000
Purge Rate Assumed (SCFM)	8000	8000
Peak Hydrogen Concentration (Vol. %)/Time (hr.)	4.0/40	8.7/14
RCB Atmosphere Temperature (^oF)/Time (hr.)	915/40	1020/15
Aerosol Comparisons		
Maximum Rate to the RCB Cleanup System (lb/hr)	4400	5100
Total Aerosols to the RCB Cleanup System to Boildry (lb)	260,000	170,000

TABLE A.5-4

COMPARISON OF APPLICANTS' RADIOLOGICAL CONSEQUENCES, MARGIN ASSESSMENT CASE

Organ	36 Hour Vent Base Case	10 Hour Vent Margin Assessment
	2 Hour Exclusion Area Boundary Doses (rem)[a]	
Bone[c]	0.028	0.44
Lung	0.0055	0.082
Thyroid	0.0096	0.023
Whole Body	0.16	1.9
	30 Day Low Population Zone Doses (rem)	
Bone[c]	55[b]	55[b]
Lung	4.0	3.9
Thyroid	99	95
Whole Body	3.5	13

a. Doses calculated with dose conversion factors from NUREG-0172 (Reference 7), and the radionuclide inventory of the homogeneous core design.

b. Results include earlier, extremely conservative estimate of 13 Kg Pu (Reference 11) sparged to containment atmosphere, vs 26 g (Reference 5).

c. Bone surface doses would be about 3X as large, if calculated with dose conversion factors from NUREG/CR-0150 (Reference 6).

TABLE A.5-5

BONE SURFACE DOSE CHANGES WITH INITIAL RELEASE SIZE

	Bone Surface Doses (rem)	
Initial Release Size[a]	Exclusion Area Boundary	Low Population Zone
Zero	0.027	0.92
Base Case	0.19	0.95
1% Fuel	6.5	2.5
5% Fuel	32	8.2
10% Fuel	64	15
50% Fuel	320	70

a. Includes 100% noble gases and Cs, Rb in all cases, 1000 lb sodium in all but the zero case, and 100% halogens and volatile fission products in the 1%, 5%, 10% and 50% cases.

The thyroid doses, due primarily to iodine releases, are also affected by the aerosol depletion in containment. The thyroid doses of the applicants' base case, 0.02 rem at the exclusion area boundary and 85 rem at the low population zone boundary, change to about 23 rem and 8 rem respectively when the release mode for halogens is changed from 100% boiled up with the sodium (as in the base case) to 100% in the initial release.

It is the staff's assessment that, given a CDA, initial releases will be either zero or, as in applicants' base case, small. Further, the radiological consequences will not be greatly changed by initial releases within a reasonably expected range.

A.5.8 Sensitivity to Alternative Scenarios

Several alternative scenarios have been considered. In the principal alternative, the applicants have presented information indicating that after the core debris penetrates the reactor vessel and reactor guard vessel, and has dropped into the reactor cavity with the sodium, it will form a uniformly-distributed bed on the reactor cavity floor liner (Reference 1). The uniform bed of particles of fuel, blanket and structural materials would be stable for the interim because the sodium would remove sufficient heat to keep the bed from changing character or penetrating the steel floor liner. The staff considers that in this scenario, initial releases, if any, would be the same as in the base case scenario, and releases during the sodium boilup phase would also be substantially the same. Exceptions are that the time to boiling and the time to boildry would be lengthened due to the absence of heat generation by sodium-concrete reactions, and the time to venting would be lengthened, hydrogen production would be less. Also, there would be no sodium-concrete reaction products and little NaOH and Na_2O with the debris bed at boildry. Once failure of the liner occurs and interactions with the concrete begin, the circumstances would be similar to the base case, with the exception of the absence of sodium-concrete reaction products. If the liner failed at some intermediate time, e.g., after 50 hours, the ensuing sequence would also be similar to the base case.

In another scenario, the sodium could be drained into the reactor cavity in advance of core disruption , leaving the core to overheat and enter the sodium pool later. Halogens and other volatiles released in the reactor vessel could either be dissolved in the sodium or be trapped in the PHTS. Once the core debris enters the sodium, this scenario would be similar to the base case.

Other alternative sequences are possible, but it is the staff's assessment that it is highly unlikely that there would occur circumstances such that the radiological consequences of the base case would be greatly exceeded.

A.5.9 Uncertainties

Although a great amount of experimental data relevant to the TMBDB scenario has been accumulated over the years, the staff believes there still remain substantial uncertainties in the estimation of the radiological consequences. The estimate of plutonium releases is judged to contribute the most to the uncertainty attributable to release quantities, possibly as much as a factor of 20. However, in Reference 1, the applicants have evaluated CRBRP's beyond-the-design-basis margins for a number of variations of their basic TMBDB scenario and have shown that the calculated range of radiation doses are limited. It is the staff's assessment that the applicants have adequately shown that the increase in dose, due to the range variations of circumstances, is small.

In a number of instances, the modeling for calculation of consequences includes conservative assumptions and use of values conservatively selected from the maximum of experimental observations; the result is that, even though these are called "realistic" calculations, there is a basis for belief that they tend to overestimate the consequences. An example of such conservatisms is considering immediate 100% release to containment of noble gases and Cs and Rb rather than considering them trapped in the PHTS above the sodium pool, and rather than considering that the Cs and Rb would be retained in the sodium and in the vent system for some hours before being vented into containment.

Of course, the dose estimates also include all the other uncertainties normally found in such estimates, e.g., in meteorological dispersion, in filter efficiencies, etc.

A.5.10 Conclusions

The staff has done scoping analysis for radiological consequences and while the staff conclusions are not identical to the applicants, but show that the dose guidelines of 10 CFR 100 as augmented for CRBR, may be exceeded, the staff has determined that sufficient improvements in the containment cleanup system filtration efficiency are easily achievable and therefore this is acceptable for the CP stage. However, conclusions presented here are based on projected performance of the proposed design of the TMBDB systems. As stated in section A.4.10, satisfactory equipment qualification and demonstration of performance of the TMBDB systems will be required at the OL stage. The staff advises that in the meantime, before installation of the TMBDB system are undertaken, that the applicants should provide for review of the parameters, and ranges of values, on which testing for operational qualification of the TMBDB systems will be performed.

References

1. CRBRP-3, Hypothetical Core Disruptive Accident Considerations in CRBRP, Volume 2, Assessment of Thermal Margin Beyond the Design Base, Revision 4, June 1982.

2. S. Jordan and Y. Ozawa, "Fuel Particle and Fission Product Release from LMFBR-Core Catcher," in Proceedings of the International Meeting on Fast Reactor Safety and Releated Physics, Chicago, Illinois, October 5-8, 1976, CONF-761001, Vol. IV, p. 1924-1929.

3. M. Berlin, E. de Montaignac, J. Dufresne and G. Geisse, "Evaluation of the Sodium Retention Factors for Fission Products and Fuel," in Proceedings of the L.M.F.B.R. Safety Topical Meeting, Lyon, France, July 19-23, 1982, Vol. III, p. III-369-380.

4. Letter report from E. Randich and J. E. Brockmann, Sandia National Laboratories, to T. J. Walker, NRC, dated January 18, 1983.

5. Letter: HQ:S:83:140 John Longenecker to Paul Check, Submittal of Information on Thermal Margins Beyond the Design Base (TMBDB), dated December 7, 1982.

6. D. E. Dunning, Jr., G. G. Killough, S. R. Bernard, J. C. Pleasant, and P. J. Walsh, "Estimates of Internal Dose Equivalent to 22 Target Organs for Radionuclides Occurring in Routine Releases from Nuclear Fuel Cycle Facilities, Vol. III," NUREG/CR-0150, Vol. 3, October 1981.

7. G. R. Hoenes and J. K. Soldat, "Age-Specific Radiation Dose Commitment Factors for a One-Year Chronic Intake," NUREG-0172, November 1977.

8. "Site Suitability Report in the Matter of Clinch River Breeder Reactor Plant," NUREG-0786, June 1982.

9. "NRC Staff's Supplemental Answers to Natural Resources Defense Council, Inc. and the Sierra Club Twenty-Sixth Set of Interrogatories to Staff," August 5, 1982.

10. Letter: HQ:S:82:112 John Longenecker to Paul Check, Thermal Margin Beyond Design Base (TMBDB) Margins Assessment Document, dated October 20, 1982.

11. CRBRP-3, Hypothetical Core Disruptive Accident Considerations in CRBRP, Volume 2, Assessment of Thermal Margin Beyond the Design Base, Revision 0, March 1980.

A.6 Summary Conclusions

The conclusions summarized here are contingent on:

(1) Satisfactory completion of the required fuel pin design modification studies and testing directed at mitigation of the effects of plenum fission gas during CDA progressions (p. A.2-6). The applicants have agreed to provide this modification.

(2) Satisfactory completion of the commitments associated with modifications of the rotating plugs in the reactor closure head and related efforts including evaluation of the SRI tests and updating of appropriate documentation (p. A.3.5 and p. A.3-14). The applicants have agreed to modify the plugs.

(3) Satisfactory resolution of the cell liner criteria based on forthcoming analyses and testing or the adoption of satisfactory fall-back positions as discussed in Section A.4.10. The applicants have agreed to this resolution.

(4) A scoping equipment qualification program has been developed by the applicants and reviewed by the staff. The staff finds this acceptable for the CP. However, the staff requires confirmation of the specific values of the parameters (temperature, pressure, etc.) during the OL review.

The staff's conclusions are developed as a result of the evaluation of the CDAs in terms of the general criteria as discussed in Section A.1.3. Based on the independent evaluation of core disruptive accident energetics described in Section A.2 and on the mechanical capability of the reactor vessel discussed in Section A.3 we conclude that, assuming a CDA occurs, containment failure from spray fires or missiles is not of concern, and further that no significant leakage of vaporized fuel will occur from the RCB. This means that the radiological consequences of CDAs are principally determined by the degree to which TMBDB features prevent containment failure from thermal phenomena such as aerosol generation, sodium fires and hydrogen burning. As discussed in Section A.4, containment failure from such phenomena is unlikely because the containment vent-purge system can relieve internal pressures and effectively control hydrogen in the RCB. The doses from venting, to which an individual would be exposed, if he remained 30 days at the plant's low population zone boundary, can be brought below the CRBR version of the 10 CFR 100 dose guidelines (realistically calculated). This, taken in conjunction with the low probability of such events, leads us to conclude that the risks from such events at CRBR will be very small, and not significantly different from the risks from typical LWRs.

APPENDIX B

UNRESOLVED SAFETY ISSUES

NUREG-0606, "Unresolved Safety Issues Summary," lists several safety issues which are undergoing NRC study before the staff can make judgments as to whether existing requirements should be modified. These issues are sometimes called "generic safety issues" because they are related to a particular class or type of nuclear facility rather than a specific plant. The staff has screened the unresolved safety issues relative to their applicability to CRBR and asked the applicants to respond as to how they plan to treat each applicable issue during the licensing activity. The applicants' response and the staff's assessment of each issue are detailed in this section.

B.1 WATERHAMMER (Unresolved Safety Issue (USI) A-1)

Applicants' Assessment of Applicability to CRBRP

Waterhammer and its equivalent, sodium hammer, are applicable to the CRBRP. Waterhammer events introduce a range of hydraulic loads, or pressure pulses, into a fluid system and are the result of rapid condensation of steam pockets, steam-driven slugs of water, pump startup into voided lines, and improper (or sudden) valve closures. Where waterhammer has occurred in water lines, the principal damage in most instances has been to pipe hangers and snubbers. Occasionally pipe welds have experienced small cracks. In none of the reported LWR waterhammer incidents has there been a release of radioactive material or a disabling of safety systems.

Applicants' Suggested Resolution for CRBRP

This issue has been technically resolved for the CRBRP. The water and steam systems of the CRBRP (i.e., the steam generator auxiliary heat removal system (SGAHRS) are described in PSAR Sections 5.5 and 5.6.1, respectively. Design resolution of waterhammer will be accomplished by including fill and vent holes in the auxiliary feedwater sparger in the steam drum to preclude waterhammer effects resulting from steam-driven slugs of SGAHRS water and by including hydraulic dampers in the actuators of the water and steam isolation valves to preclude waterhammer effects resulting from the overly rapid closing of a valve. The vent holes are described in revised PSAR Section 5.5.2.3, and the hydraulic dampers are discussed in Section 5.5.3.1.5.2.

Protection against the effects of pipe breaks and waterhammer loads are incorporated in ASME design codes that require consideration of impact loads and dynamic loads in the structural design. The ASME codes are applied to the sodium systems of CRBRP, that is, the primary heat transport system, the intermediate heat transport system (including the steam generator), and the sodium-water reaction pressure relief system, as well as to the water-steam systems.

The occurrence of sonic pulses, similar to those produced in waterhammer incidents, has been considered in the design of the intermediate heat transport system (IHTS), described in PSAR Section 5.4. Sonic pulses may occur as a result of a large sodium-water reaction caused by a postulated steam generator tube rupture. In addition, the effects of accelerated sodium slug flows in the component and piping design has been considered in the design of the sodium-water reaction pressure relief subsystem, described in PSAR Sections 5.5, 7.5.6, and 15.3.3.3.

The absence of sodium isolation valves in the IHTS precludes high decelerations of sodium that could cause waterhammer effects in sodium. The high normal boiling point and high heat of vaporization of sodium make vapor-driven sonic pulses extremely unlikely.

NRC's Position

The staff concurs that the applicants are addressing the waterhammer/sodium-ammer phenomena analytically and in the proper manner relative to the CRBRP application. However, the applicants must verify that unacceptable feedwater hammer will not occur by performing acceptability tests, approved by NRC, as described in BTP ASB 10-2. (Standard Review Plan, Section 10.4.7-7)

B.2 STEAM GENERATOR TUBE INTEGRITY (USIs A-3, A-4, and A-5)

Applicants' Assessment of Applicability to CRBRP

This issue is applicable to CRBRP. The design designates steam generators in each of the three heat transport system loops for the transfer of heat from the secondary sodium loop to the water systems. The issue concerns the capability of steam generator tubes to maintain their integrity under normal operation and accident conditions, should mechanisms exist that could result in tube degradation.

Applicants' Suggested Resolution for CRBRP

This issue has been technically resolved for the CRBRP.

The CRBRP steam generator design has minimized the potential for corrosion/erosion degradation common to steam generators in pressurized-water reactors (PWRs). The tubes in the CRBRP steam generator will be exposed to the water environment only on their inside surface. The water side will consist of smooth wall tubes terminated in spherical plena. This will greatly reduce the potential for tube degradation by corrosion-induced wastage, cracking, and denting. Preferential corrosion product formation or deposition will be minimized because there will be no restrictions, crevices, water levels, or structure-related concentration sites. Water-side chemistry will be maintained by state-of-the-art, all-volatile chemistry control, which has been modified from PWR practice and which will incorporate fossil plant experience with 2¼% Cr-1% Mo tube material. Full flow demineralizers, a 2:1 full-power recirculation ratio (for each two parts of water flowing into the steam generator, one part will be recirculated and one part will be fresh feed), and 10% blowdown will contribute to minimizing the potential for water-side corrosion-related problems.

Steam generator tube integrity has been properly addressed in the CRBRP design by specifying that a total of 29% of the 0.109-in. tube wall thickness (Section 5.5.2.3.4 of the PSAR) will be allocated for corrosion, cleaning, and wear allowances. The reduced thickness will be used for all stress and strain calculations, and the full thickness will be used for weight and seismic calculations. In addition, allowances will be provided to compensate for material strength degradation by postweld heat treatment, thermal aging, and decarburization. In spite of these reductions in thickness and material conservatively based on the end-of-life condition, the tube will have a 38% margin over the ASME Code, Class 1, criteria for pressure retention.

Erosion of tubes as a result of tube vibration is being addressed in three ways, as discussed in PSAR Section 5.5.

(1) The design and material selection of the shell (sodium-containing) side of the steam generator (SG) will provide for acceptable accommodation of tube vibrations; all known flow-induced vibration mechanisms have been evaluated. Tube-to-spacer plate gaps will be consistent with guidelines used throughout the heat exchanger industry. Tube-spacer plate material (Inconel 718) has been chosen, since it has a low coefficient of friction when coupled with the tube material (2¼% Cr-1% Mo).

(2) To confirm that all flow-induced vibration mechanisms are considered, a flow-induced vibration program has been implemented using both a full-scale model closely representing the prototype unit and a 0.42 scale model. The scale model flow-induced vibration tests will ensure that mechanisms of unexpected origin in the plant unit design do not exist.

(3) The applicants have developed an ultrasonic tube inspection technique that can detect the tube wear well before the tube wall is thinned beyond that spcified for the design. This technique is discussed in PSAR Appendix G.

NRC's Position

The staff agrees that the actions proposed by the applicants will minimize the probability of steam generator tube degradation resulting from wastage and flow-induced vibration. The inservice inspection technique (volumetric eddy current and continuous monitoring) and the intervals for volumetric examination of the tubing suggested in Section 5.11 are adequate to ensure that no major undetected steam generator tube degradation will take place during the life of the plant.

The scale model flow test results will be reviewed during the operating license review process to ensure that the described design has been successful in meeting flow-induced vibration requirements.

All-volatile chemistry control techniques proposed for the CRBR water treatment system have proven to be an effective method for reducing wastage and stress-corrosion cracking in LWRs.

Finally, the staff recognizes the less severe safety role a failure of a CRBR steam generator tube imposes relative to that of an LWR steam generator tube failure. However, the same ASME Code design rules are being imposed on the CRBR steam generators. Large and small steam generator tube failures have been

addressed in the Chapter 15 review, and the proposed accommodation criteri
cluding immediate reactor shutdown, have been found acceptable.

B.3 ANTICIPATED TRANSIENTS WITHOUT SCRAM (USI A-9)

Applicants' Assessment of Applicability to CRBRP

This issue is applicable to CRBRP. The issue is the potential for a commo failure to reduce the reliability of protection systems in such a way that reactor might not shut down as required when an anticipated transient occu

Applicants' Suggested Resolution for CRBRP

The applicants' view is that this issue is resolved for CRBRP because CRBR incorporates into its design two independent shutdown systems, either of w will have the capability, of itself, to terminate reactor power transients to effect rapid shutdown of the reactor automatically, and further, becaus strict attention to the diversity and independence of the two shutdown sys will reduce the likelihood of their simultaneous failure to such a low lev that additional design features to improve reliability of shutdown will no necessary.

NRC's Position

The Commission has initiated a rulemaking on this anticipated transient wi scram (ATWS) issue. The ATWS issue is discussed for light-water reactors in "Anticipated Transients Without Scram for Light Water Reactors," NUREG- Volume 4, March 1980, where specific design features and analyses are pres for LWRs. These prescriptions are, however, tailored to each type of LWR thus generally not appropriate for CRBRP.

The staff's conclusions on this issue for CRBRP are based on its review of redundancy, independence, and diversity embodied in the proposed designs f CRBRP's two shutdown systems (as discussed in Sections 4 and 7 of this SER on the acceptability of the applicants' Reliability Assurance Program (as cussed in Appendix C of this SER), and on the assessment that even if an A event should occur at CRBRP and lead to core disruption, the risks would b acceptably low (Appendix A of this SER). The staff concludes that the ATW issue will be resolved for CRBR upon implementation of the design as modif by the findings of this SER. However, additional insight in this area may gained from continued evaluation of operating experience at LWRs and other nuclear reactors. Therefore, the staff will expect the applicants to addr in the FSAR those measures taken in response to lessons learned from react operating experience during the period from the issuance of the constructi permit to the issuance of the FSAR, and specifically the implications of t ATWS at the Salem reactor in February 1983.*

*It should be noted that at the time this SER was issued, the staff had de mined that the ATWS event at Salem was caused by failure of the scram bre to open when required. The scram breakers to be used on CRBR are of diff design than those used at Salem and will be identical to those used on FF and in the Naval Reactors Program. These breakers have undergone extensi testing and have operated successfully. Therefore, they are not expected lead to an ATWS event at CRBR. Nevertheless, they will be included in th applicant's Reliability Assurance Program.

B.4 FRACTURE TOUGHNESS OF STEAM GENERATOR AND REACTOR COOLANT PUMP SUPPORTS (USI A-12)

Applicants' Assessment of Applicability to CRBRP

This issue is applicable to CRBRP. It concerns the low fracture toughness and potential lamellar tearing in materials used for heat transport system component supports.

Applicants' Suggested Resolution for CRBRP

The design of that portion of the CRBRP steam generator supports that is in accordance with the ASME Code requires that impact testing (Charpy V-notch) of all materials of construction be performed according to Paragraph NR-2311 of ASME Code, Section III. The acceptance standards of Paragraph NR-2330 must be met at 50°F, maximum. Since the lowest operating temperature for the steam generator support will be 125°F, there will be adequate margin for protection against nonductile failures. In addition to the materials fracture toughness requirements, postulated defects will be evaluated using the procedure in Appendix G of ASME Code, Section III, for all applicable conditions plus shipping, lifting, and installation. Therefore, the concern relating to fracture toughness of steam generator supports has been properly and adequately addressed in the CRBRP design.

The building structural steel that supports steam generators will be designed in accordance with the requirements of the American Institute of Steel Construction Code using American Society for Testing Materials (ASTM) A-36 steel and SA-540 bolting material. Sandia Laboratories report SAND78-2348 (Appendix C to NUREG-0577, "Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports - Resolution of Generic Technical Activity A-12 for Comment") classifies USI A-36 as falling within material group II, that is, intermediate susceptibility to brittle fracture, and identifies that group II materials have been judged adequate. SAND78-2348 classifies SA-540 bolting material as falling within material group III, which has also been judged adequate.

The supports for reactor coolant pumps and intermediate heat exchangers will be type 304 stainless steel, connected to ASTM A-36 embedded plate with SA-540 bolting material.

CRBRP design criteria applied to the reactor vessel and steam generator supports will preclude conditions leading to lamellar tearing (e.g., material selection, welded joint orientation, and fabrication sequence).

NRC's Position

The staff agrees. The heat exchanger supports, steam generator supports, and the primary and secondary pump supports will be in cells or local regions where temperatures will not fall below 125°F before reactor operations. This temperature limit could conceivably be a technical specification even though the CRBR primary coolant contains relatively low stored energy.

B.5 SYSTEMS INTERACTION IN NUCLEAR POWER PLANTS (USI A-17)

Applicants' Assessment of Applicability to CRBRP

This issue is applicable to CRBRP. It concerns the sufficiency of integration of divided responsibilities for design, analysis, and installation of systems among teams of engineers with functional specialities, such as civil, electrical, mechanical, and nuclear, to ensure that adverse operational interactions between plant systems will be minimized.

Applicants' Suggested Resolution for CRBRP

The applicants have implemented a combination of programs and activities directed toward ensuring an integrated design that has considered the potential for and will provide protection against adverse operational interactions between plant systems.

These include the CRBRP quality assurance program, a comprehensive design control program, specialized design reviews, and reliability and probabilistic risk assessment programs.

The plant has been designed to requirements that support a defense-in-depth philosophy. These requirements ensure physical separation and independence of redundant safety systems, diversity of safety features, and protection against hazards such as sodium leaks, sodium-water reactions, line ruptures, missiles, tornadoes, floods, seismic events, fires, human errors, and acts of sabotage. These requirements are described in PSAR Section 1.1.2 and Chapter 3.

To ensure that these requirements will be properly implemented, the CRBR Quality Assurance Program addresses the design process. This program requires that during the design process emphasis be placed on the control of interfaces between systems. This interfacing is described in PSAR Section 17A.3.1. Independent design reviews, with interdisciplinary memberships and objectives, are required at various stages of the design process. Requirements for these independent design reviews are described in PSAR Chapter 17, Appendix G.

Extensive key systems reviews (KSRs) cutting across system boundaries have been conducted. Multidisciplined groups of individuals conducted these reviews with objectives that included assessments of plant and operator responses during offnormal and accident events. Interactions between systems were explicitly considered as part of these reviews. Evaluations of the results of these reviews addressed the potential for adverse systems interactions, including consideration of human, spatial, and functional coupling effects. A summary report of these KSRs was provided in response to an NRC question.

The CRBRP safety-related reliability program is described in PSAR Appendix C. The results obtained in this program provide additional confidence that the requirements for the systems designs will minimize the potential for adverse operational interactions.

In response to an NRC question, the applicants developed the CRBRP Probabilistic Risk Assessment (PRA) Program Plan, which will include tasks to demonstrate that the risks at CRBRP will be acceptably low. The planned methodology will use event trees and fault trees to identify the component failures combinations program is expected to employ analytical methods, visual inspections, experience feedback, and experiments for dependencies. The light-water-reactor industry's current experience with systems interaction reviews is fragmented. Experience like that gained by the Phase I study is an essential ingredient to the staff's considerations of a comprehensive systems interaction program.

CRBRP has been evaluated against current licensing requirements that are founded on the principle of defense in depth. Adherence to this principle results in requirements such as physical separation and independence of redundant safety systems and protection against hazards such as high-energy-line ruptures, missiles, high winds, flooding, seismic events, fires, human factors, and sabotage. These design provisions are subject to review against the Standard Review Plan (NUREG-0800) which requires interdisciplinary review of safety-grade equipment and addresses different types of potential systems interactions. Also, the quality assurance program that is followed during the design, construction, and operational phases for each plant contributes to the prevention of introducing the potential for systems interactions by error. Thus, the current licensing requirements and procedures provide an adequate degree of plant safety.

In addition the applicants have described a program, Key Systems Reviews, that separately evaluates all structures, systems, and components important to safety for the three categories of adverse systems interactions, that is, spatially coupled, functionally coupled, and humanly coupled. Also, the applicants have committed to a reliability program that will include failure modes and effects analyses, system level fault trees or their equivalent to account for event sequences, and systematic searches (such as plant walk-throughs) for adverse systems interactions. This program is discussed in more detail in Appendix C of this SER.

Although the staff at this time cannot be sure that the applicants' programs will be equivalent to the methodology eventually arrived at for resolution of USI A-17 for LWRs, the staff believes it is acceptable for the CP stage of licensing. The staff expects the applicant to monitor the ongoing generic effort on USI A-17 during the intervening period between CP issuance and OL licensing review and to develop an equivalent program to that adopted for LWRs.

B.6 ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED ELECTRICAL EQUIPMENT (USI-A-24)

Applicants' Assessment of Applicability to CRBRP

This issue is applicable to CRBRP. CRBRP design will include Class 1E equipment that must be qualified for the environmental conditions in which it may be required to perform.

Applicants' Suggested Resolution for CRBRP

The technical resolution of this issue for LWRs and NRC's position are contained in NUREG-[illegible] "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment." The issue is [illegible] for CRBRP through a program for environmentally qualifying safety-related electrical equipment that is consistent with the objectives and requirements [illegible] Revision [illegible] applied to CRBRP technology. This program [illegible] in response to NRC [illegible] and [illegible] PSAR Section 3.11.

NRC's Position

The staff has reviewed PSAR Section 3.11 and supplemental letters from the applicants relative to the proposed CRBRP environmental qualification program and finds it acceptable as detailed in Section 3.11 of this SER. Major highlights of the review are:

(1) The applicants' list of systems and components that are required to perform safety-related functions, as presented in WARD-D-165, Revision 6, was found acceptable.

(2) The temperature, pressure, and humidity conditions, for both inside and outside containment, were properly specified by the applicants.

(3) The applicants' approach to qualifying the equipment for a sodium aerosol environment was found acceptable.

(4) The applicants have committed to follow the recommendations in RG 1.33, Revision 2, to identify and prevent significant age-related degradation of electrical equipment.

(5) The applicants have defined the design methodology used to calculate the radioactive environment based on three different source terms (site suitability source term, sodium storage tank failure source term, and cover gas release source term). The staff has reviewed the proposed methodology and finds it acceptable for use in the qualification of electrical equipment.

(6) The applicants have committed to meet the documentation requirements identified in IEEE Std. 323, 1974. The staff finds this plan for documentation acceptable and in accordance with 10 CFR 50.49.

B.7 RESIDUAL HEAT REMOVAL REQUIREMENTS (USI A-31)

Applicants' Assessment of Applicability to CRBRP

This issue is not applicable to CRBRP. It concerns the capability of PWRs to go from hot to cold shutdown without the availability of offsite power.

A safe shutdown condition equivalent to a PWR cold shutdown condition will be achieved in CRBRP when the plant will be brought down from operating temperature to 600°F using the plant shutdown heat removal systems. At the 600°F temperature the plant will be in a safe and stable state, and long-term cooling will be in effect. There is no subsequent requirement to proceed to another mode or state to effect long-term shutdown.

The normal decay heat removal path will be through the use of the main condenser and feedwater train. However, since the main condenser and feedwater train will not be available on loss of offsite power, the steam generator auxiliary heat removal system, which is a safety-related system, will be provided for shutdown heat removal and long-term decay heat removal, and will not depend on the availability of offsite power. The initial heat load will be dissipated through the use of power relief valves in the steam generator loops.

NRC's Position

The NRC concurs with the applicants' assessment.

B.8 CONROL OF HEAVY LOADS NEAR SPENT FUEL (USI A-36)

Applicants' Assessment of Applicability to CRBRP

This issue is applicable to CRBRP. Although the design of CRBRP does not designate spent fuel pools, this concern is applicable to the control of heavy loads over the ex-vessel storage tank closure head and striker plate, and over the fuel-handling cell.

Applicants' Suggested Resolution for CRBRP

The technical resolution of this issue and NRC's position are contained in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

The issue is resolved for CRBRP by the application of a single-failure-proof crane (in accordance with NUREG-0554, "Single Failure Proof Cranes for Nuclear Power Plants") in both the reactor service building and reactor containment building for all critical lifts. The project application of NUREG-0612 is presented in response to NRC Question CS410.3.

NRC's Position

The applicants have applied the two proper criteria to resolve this issue for CRBRP, namely, NUREG-0554 and NUREG-0612. These two criteria will be the basis during the OL licensing review.

B.9 SEISMIC DESIGN CRITERIA (USI A-40)

Applicants' Assessment of Applicability to CRBRP

This issue is applicable to CRBRP. It concerns the conservatism of certain aspects of the overall seismic design criteria.

Applicants' Suggested Resolution for CRBRP

This issue has been technially resolved for CRBRP. The seismic design bases and the seismic design of CRBRP conform to the current NRC criteria. CRBRP seismic design criteria are described in PSAR Section 3.7. NRC has not established any other bases that would render conformance to the current criteria inadequate.

NRC's Position

Pending the formal supportive documentation relating to the adequacy of assumptions used in the rock structure interaction model, the NRC concurs that the seismic design criteria and procedures used by the applicants for CRBRP are adequate, namely:

(1) assigning two levels of earthquake (SSE at 0.25 g and OBE at 0.125 g) which reflects appropriate consideration for the most severe earthquake recorded for the site with an appropriate margin

(2) appropriate combinations of the effects on normal and accident conditions with the effect of the natural phenomena

(3) appropriate consideration of the safety functions to be performed--the use of a suitable dynamic analysis or a suitable qualification test to demonstrate that structures, systems, and components can withstand the seismic and other concurrent loads, except where it can be demonstrated that the use of an equivalent static load method provides adequate consideration.

B.10 STATION BLACKOUT (USI A-44)

Applicants' Assessment of Applicability to CRBRP

A loss [illegible] ac power involves a loss of both the preferred and backup sources of offsite power.

If offsite ac power is lost, three diesel generators and their associated distribution systems will be designed to deliver emergency power to safety-related equipment.

If both offsite and onsite ac power are lost, CRBRP will be designed to remove reactor-generated decay heat on natural circulation with the heat sink provided by the steam generator auxiliary heat removal system. This capability will ensure that adequate cooling can be maintained for at least 2 hours, which will allow time for restoration of ac power from either offsite or onsite sources.

The decay heat generated in the spent fuel in the ex-vessel storage tank (EVST) will be capable of being removed by natural circulation. This will be provided by the third EVST cooling loop which will be designed to remove all decay heat produced in the EVST during natural circulation.

The ex-vessel transfer machine will be designed to ensure that cladding temperature will be maintained within limits by a natural convection cooling system. This will ensure cooling of a fuel assembly in transit between the reactor and EVST during a station blackout.

Two-hour station blackout during the handling of a bare fuel assembly during normal fuel-handling cell (FHC) operations would result in release of fission products to the environment. The potential radiation doses at the site boundary resulting from such a release have been calculated to be below established limits.

NRC's Position

The NRC concurs with this assessment provided the applicants demonstrate that adequate natural circulation capabilities exist in the main heat transport systems and the ex-vessel storage tank natural circulation heat removal loop.

Also, the adequacy of the circuit breaker realignment capability during a station blackout must be demonstrated during the prestartup test program.

On the basis of the staff's confidence that these provisions will be met, the staff concludes that the CRBRP will have the capability to withstand a station blackout comparable to that of a PWR. The final generic resolution of this issue for PWR will probably be determined after the CP license is issued. On the basis of the similarity of the CRBR electric power system and auxiliary feedwater system to those of PWRs, the staff anticipates that the generic PWR station blackout resolution will be generally applicable to CRBR. The applicant should adopt that generic resolution for CRBR or develop an equivalent resolution in time for the OL review.

B.11 SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS (USI A-45)

Applicants' Assessment of Applicability to CRBRP

This issue is applicable to CRBRP. It concerns the sufficiency of plant capability to remove decay heat. CRBRP must have a highly reliable capability to remove decay heat from the reactor.

Applicants' Suggested Resolution for CRBRP

This issue has been resolved for CRBRP by incorporating into the design, multiple, independent, and highly reliable heat transport paths, any one of which will have sufficient capacity to remove the reactor decay heat by itself. The various heat removal paths and their operating modes embody substantial diversity.

The CRBRP heat transport system (HTS) will use three independent loops, each of which will provide a separate path from the reactor vessel to the ultimate heat sinks. The normal heat removal path includes the main condenser and feedwater train, which is used for normal operation and some shutdown heat removal conditions. However, for each path an alternative safety-related path will be provided through the SGAHRS, which will provide its own heat sinks. Thus, it will not be necessary to rely on the main condenser and feedwater train, since SGAHRS will be available for all anticipated plant events.

The SGAHRS will include the auxiliary feedwater subsystem (AFWS) and protected air-cooled condensers (PACCs), which will serve as alternative heat sinks.

Also, to ensure that the operation of safety system equipment will not be impaired, the single-failure criterion has been applied in the plant design. PSAR Section 7.2.2 discusses plant protection system (PPS)-control system interaction. The CRBRP PPS will be composed of two independent subsystems, either of which will be capable of bringing the plant to a safe shutdown condition.

Further, these two subsystems will employ diverse trip functions for PPS activation. Therefore, for any design-basis transient, there will always be more than one trip function provided by these two totally independent subsystems to activate the PPS and terminate the ensuing transient. Details of this design are described in PSAR Section 7.2 and Table 7.2-2.

A wide range of bounding transients and accidents currently is being analyzed to ensure that the postulated events would be adequately mitigated by the safety systems. In addition, systematic reviews of safety systems have been performed with the goal of ensuring that the control system failures will not defeat safety system action. The worst conditions for each given type of transient are assumed in the accident analyses. This information is provided in PSAR Chapter 15.

The AFWS provides water makeup to the closed loops between the steam generators and the PACCs. The AFWS includes two motor-driven pumps and one steam-turbine-driven pump.

The sodium in the primary and intermediate systems of the HTS loops will always be at temperatures well below the flash point. Thus, in the unlikely event of a sodium pipe leak in any loop, there will not be a loss of heat removal capability resulting from loss of coolant inventory through flashing. Also, degradation of one loop will not affect heat removal capability in either of the other two loops.

Thus, the plant configuration will provide multiple independent paths through the heat transport system, which will contribute to the high reliability of the plant systems for removing reactor decay heat. These capabilities are discussed in PSAR Sections 5.6 and 5.6.1.

In CRBRP there is an additional path for decay heat removal, the direct heat removal service. This system provides a diverse heat removal path to yet another redundant and diverse set of air-cooled heat exchangers. This is described in PSAR Section 5.6.2.

NRC's Position

The acceptance criteria for CRBR shutdown decay heat requirements are more stringent than those for LWRs, namely, the probability of loss of all ultimate heat sink must be sufficiently low so as to allow treatment of the consequences of the event beyond the design basis. These consequences are discussed in Appendix A of the SER. Principal design criterion (PDC) 35 on residual heat removal has been developed for CRBR and contains requirements more conservative than those for LWRs. In addition, the plant will be designed to remove decay heat via natural circulation in the main HTS loops so that even a total loss of offsite and onsite ac power will not prevent decay heat removal. The applicants have also committed to perform a realiability risk assessment of the AFWS during the operating license review.

Furthermore, the direct heat removal service goes a long way toward resolving USI A-45 for CRBRP. However, this issue has not yet been generically resolved for LWRs. The staff will, therefore, reconsider this issue in the OL review. The applicant should consider the applicability of the LWR resolution to USI A-45 to the CRBR and provide justification in the FSAR that a comparable level of safety has been achieved. As noted in Appendix D, the PRA to be performed by the applicants will include consideration of enhancements in the heat removal capability.

B.12 SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS (USI A-46)

Applicants' Assessment of Applicability to CRBR

This issue is not applicable to CRBRP. The issue is whether operating plants must be reassessed to ensure the adequacy of their seismic qualification of equipment. Construction of the project has not yet commenced and thus, it is not an operating plant. CRBRP resolution of USI A-40 ensures the adequacy of seismic design criteria applied to it.

NRC's Position

The NRC concurs with the applicants' assessment.

B.13 SAFETY IMPLICATIONS OF CONTROL SYSTEMS (USI A-47)

Applicants' Assessment of Applicability to CRBR

This issue is applicable to CRBRP. CRBRP will depend on the proper functioning of control systems to maintain the plant in a safe condition for all normal operations and accidents. This issue concerns the potential for transients or accidents being made more severe as a result of control system failures or malfunctions. These failures or malfunctions may occur independently or as a result of the accident or transient under consideration.

Applicants' Suggested Resolution for CRBR

This issue has been technically resolved for CRBRP. Design features ensure that control system failures will not prevent automatic or manual initiation and operation of any safety system equipment required to trip the plant or to maintain the plant in a safe shutdown condition following any anticipated operational occurrence or accident. This will be accomplished by providing independence and physical separation between safety system trains and between safety and nonsafety systems. For the latter, as a minimum, isolation devices will be provided. These devices will preclude the propagation of nonsafety equipment faults to the protection systems.

NRC's Position

A number of concerns have been expressed regarding the adequacy of safety systems in the mitigation of the kinds of control system failures that could actually occur at nuclear plants, as opposed to those analyzed in PSAR Chapter 15 safety analyses. Although the Chapter 15 analyses are based on conservative assumptions regarding failures of single control systems, systematic reviews have not been reported to demonstrate that multiple control system failures beyond the Chapter 15 analyses could not occur because of single events. Among the types of events that could initiate such multiple failures, the most significant are, in the staff's judgment, those resulting from a failure or malfunction of power supplies or sensors common to two or more control systems. To provide assurance that the design-basis event analyses adequately bound multiple control system failures, the applicants were asked to provide the following information:

(1) Identify those control systems whose failure of malfunction could seriously impact plant safety.

(2) Indicate which, if any, of the control systems identified under Item (1) receive power from common power sources. The power sources considered should include all power sources whose failure of malfunction could lead to failure or malfunction of more than one control system and should extend to the effects of cascading power losses resulting from the failure of higher level distribution panels and load centers.

(3) Indicate which, if any, of the control systems identified under Item (1) receive input signals from common sensors, common hydraulic headers or common impulse lines.

Section 7 of this SER verifies that the design criteria for the control systems will be such that simultaneous malfunctions of control systems that could result from failure of a power source, sensor, or sensor impulse line supplying power or signals to more than one control system will be bounded by the analysis of anticipated operational occurrences in Chapter 15 of the Final Safety Analysis Report.

B.14 HYDROGEN CONTROL MEASURES AND EFFECTS OF HYDROGEN BURNS ON SAFETY EQUIPMENT (USI A-48)

Applicants' Assessment of Applicability to CRBRP

This issue is not applicable to CRBRP. Design-basis accidents within the CRBRP containment will not lead to the generation of hydrogen. Accordingly, there will be no effect of hydrogen burns that could impact the capability of safety-related equipment to perform its intended safety function. However, accidents beyond the design basis involving hypothetical core disruptive accidents may produce hydrogen as a result of sodium-concrete interactions. The control and burning of the hydrogen from a hypothetical core disruptive accident is addressed in the CRBRP Thermal Margin Beyond Design Basis (TMBDB) (Westinghouse, CRBRP-3, Vol. 2). In the TMBDB scenario, the hydrogen is ignited in the containment atmosphere by sodium burning with the oxygen in containment. CRBRP-3, Volume 2, also demonstrates how containment integrity will be maintained.

NRC's Position

The staff agrees with this position. The control and burning of hydrogen from a core disruptive accident is further discussed in Appendix A of this SER.

13.5 REFERENCES

American Institute of Steel Construction (AISC), "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," New York, Sixth Edition, 1969.

American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Appendix G.

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---, RG 1.33, "Quality Assurance Program Requirements (Operation)," Rev. 2, Mar. 1978.

Westinghouse Electric Corporation, "Hypothetical Core Disruptive Accident Considerations in CRBRP," Vol. 2, "Assessment of Thermal Margin Beyond the Design Base," Pittsburgh, PA.

---, "Requirements for Environmental Qualification of Class 1E Equipment," WARD-D-0165, Rev. 6.

APPENDIX C

RELIABILITY ASSURANCE PROGRAM

C.1 INTRODUCTION

The applicants have identified several activities which are under way or are to be performed as part of final design to enhance and assess the reliability of certain CRBRP systems considered important to safety and to estimate the risk associated with CRBRP operation.

The objective of these reliability assurance activities, as stated by the applicants in a January 11, 1983 letter (J. R Longenecker to P. S. Check, #HQ:S:83:184), is to "provide additional assurance that the inherent reliability in the CRBRP design concept is achieved and that the likelihood of exceeding the offsite radiological dose guidelines of 10 CFR 100 is acceptably low. The overall aiming point of these activities is to ensure that the risk to the public from CRBRP is comparable to that of a current LWR." This represents an effort beyond that which is required for the licensing of an LWR and in the staff's judgment is a positive step toward enhancement of CRBRP reliability.

Traditionally the reliability of nuclear power plant safety systems has been enhanced by the application in the design of the principles of:

(1) single-failure criterion
(2) redundancy
(3) diversity
(4) independence

For CRBRP these principles and requirements are specified in 10 CFR 50 and the principal design criteria.

However, application and implementation of 10 CFR 50 and the design criteria has been and continues to be guided by engineering judgment. Additionally, reliance on operator action to terminate or mitigate accident conditions has been minimized in CRBRP to limit, to the extent practical, the potential for operator error.

More recently greater emphasis has been placed on more qualitative and quantitative attention to reliability and risk assessment as an additional tool with which to improve designs and to assess the risk of plant operation. Major developments in this area have been:

(1) Following the accident at Three Mile Island, Unit 2, a requirement for each applicant to perform a probabilistic risk assessment and factor the results of this assessment into the design was issued in NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permit and Manufacturing License," August 1981.

(2) The current Standard Review Plan for the auxiliary feedwater system (Section 10.4.9 of NUREG-0800) has included a requirement to demonstrate that the reliability of the system meets certain quantitative goals.

(3) In January 1983 the NRC issued a policy statement on safety goals which embody the principle of acceptable risk.

In the case of CRBRP, however, limited experience with the design and operation of similar facilities (relative to experience with LWRs) has been accumulated. Thus while there are no regulatory requirements (other than discussed above) to conduct a reliability program within the licensing process, it is the staff's judgment that additional reliability assurance activities should be applied to CRBRP to compensate for the lack of an experience base comparable to that available for an LWR.

The staff has reviewed the various activities outlined by the applicants in the January 11, 1982 letter as contributing to the CRBRP Reliability Assurance Program. As part of this review it was considered necessary to develop criteria addressing what constitutes an acceptable Reliability Assurance Program for CRBRP and against which the applicants' activities could be judged. The criteria developed are considered as requirements for CRBRP and are discussed in detail in Section 2.

The staff believes that the overall objective of the CRBRP Reliability Assurance Program (henceforth called the Program) should be to evaluate and enhance the potential safety-related reliability inherent in the application of 10 CFR 50 and the principal design criteria. This evaluation and enhancement should provide further assurance that the CRBR design will be capable of providing for accident prevention, termination, and mitigation so that the likelihood of a core disruptive accident or of exceeding 10 CFR 100 guidelines is extremely low. In general terms, the activities under the Program should be performed to ensure that the risk to the public from CRBRP is at least no greater than that from a current LWR. It is envisioned that such a Program be conducted to provide reliability feedback information comprehensively during design, fabrication, construction, and operation (including maintenance and surveillance testing) of CRBRP. This feedback should also cause the plant design and operating, surveillance, and maintenance procedures to be changed where considered appropriate.

As stated previously, the staff has developed criteria for the program specifying the appropriate breadth and depth of activities to be performed. The applicants' Program should be structured to meet these criteria. Further, the staff intends to review the applicants' Program through audits and reviews of the process and results to ensure that the overall objective is being met. The staff may perform independent reliability-oriented studies to gain additional confirmatory understanding of the Program. This latter activity is, however, of secondary importance to the staff audits and reviews of the applicants' Program based upon traceable and auditable documentation.

C.2 EVALUATION CRITERIA

In this section, the staff's criteria regarding what constitutes an acceptable program are presented. These criteria define the nature and extent of the Program in broad terms.

The staff in conjunction with its consultant, Science Applications, Inc., has developed an outline of a comprehensive Program with the potential for assessing and impacting CRBRP reliability. The elements of this Program are described below as criteria and are considered by the staff as requirements for a reliability program for CRBRP.

Considering CRBRP characteristics, a unique set of evaluation criteria has been defined for CRBRP. This set contains three elements:

(1) reliability information gathering
(2) feedback to design, operation, surveillance, and maintenance
(3) traceability and auditability

The following activities will generate reliability information:

(1) component level evaluations
(2) system level evaluations
(3) accident sequence level evaluations
(4) common cause failure analyses
(5) system interaction analyses
(6) equipment testing
(7) equipment qualification
(8) failure evaluation

It is the staff's opinion that the existence of these activities in appropriate depth (as discussed later) ensures completeness of a reliability program. The first three activities above--evaluation of the component, system, and overall accident sequence levels--ensures that potential malfunctions at all levels are examined. The common cause failure analyses help provide assurance that built-in design redundancies and mitigative functions are not defeated by common environmental factors, common support systems, or common initiating malfunctions. System interaction analyses are needed to identify system malfunctions, which may be acceptable by themselves, but which could propagate to other systems with unacceptable consequences. Equipment testing is used in a developmental program to verify design. In some cases component failure mode or failure rate data can also be generated. Equipment qualification is a standard requirement for the nuclear industry to ensure performance under required environmental conditions. Failure evaluation is a necessary ingredient to ensure appropriate design feedback and corrective action.

Given these activities of a program, the next essential step is to apply them to the correct components, systems, features, and operational aspects of the plant. In this regard the staff concluded that the Program should be applied to those systems and features whose functions are necessary to prevent core disruptive accidents and to ensure that the likelihood of exceeding 10 CFR 100 dose guidelines is acceptably low. It was judged that if these functions are performed in a reliable manner, then the risk to public health and safety from CRBR operation would be acceptably low and comparable to that from an LWR.

The extent of the reliability activities performed for each system depends upon (1) whether or not the system has active components or features, (2) the accumulated base of directly applicable experience in LWRs or other LMFBRs, (3) whether the system is designed for prevention or mitigation, and (4) the judged importance to protection of public health and safety. In ranking the CRBR systems the reactor shutdown and shutdown heat removal functions are considered of primary importance, and thus those systems utilized in fulfilling these two functions should receive emphasis in the Program. Furthermore, it was concluded that both the front line and support systems necessary to perform each function and feature should be included in the Program. The functions and features judged to fall in this category are:

(1) reactor shutdown
(2) shutdown heat removal
(3) coolant system boundary integrity
(4) features to prevent core flow blockage
(5) features to prevent failed fuel propagation
(6) containment
(7) spent fuel cooling
(8) active features to mitigate core disruptive accidents

In addition to the information-gathering activities mentioned above, the Program includes two additional elements. Feedback to design and operation provides a means of improving the design or operating, surveillance, and maintenance procedures should this be judged appropriate. Traceability and auditability enables determination of the status and appropriateness of the Program. Each of the three elements are discussed in the following sections.

C.2.1 Content of Safety-Related Reliability Information-Gathering Activities

The set of activities for gathering information within the Program are described in more detail below.

(1) Component Level Evaluations

Failure modes and effects analysis (FMEAs) are the basic tools of reliability evaluations applied at the component (pumps, valves, sensors, and so forth) level which form the foundation upon which higher level evaluations are built. Emphasis regarding FMEAs should be placed on components unique (or unique in application) to CRBRP or those components for which a statistically significant reliability data base has not been established. Documented reliability data from previous experience can be used in connection with or instead of FMEAs. Components of this nature which are incorporated in the reactor shutdown and shutdown heat removal functions are of primary importance. Component failures critical to operational success should be systematically identified and evaluated as to both severity and likelihood of occurrence. The principal output of the component-level FMEA is

(a) a comprehensive list of failure mo[illegible]
(b) a list of potential causes
(c) the effect of the failure and [illegible]
(d) qualitative or quantitative [illegible]

This output provides the initial asse[illegible] and can also be used in defin[illegible] a te[illegible] [illegible] FMEA also a source of data for det[illegible] to failure from a common ca[illegible]

(2) [illegible]

[illegible] to relate failure information to its impact [illegible]. In this context examples of systems are [illegible] shutdown system, auxiliary feedwater system, [illegible], and so forth. Fault trees or a combination [illegible] block diagrams are acceptable methods to produce [illegible]. From this evaluation, combi- [illegible] that lead to system failure can be derived. [illegible] qualitatively or quantitatively eval- [illegible] complete set of combinations of component [illegible]. As part of the process [illegible] resulting from common supporting [illegible] operations, and maintenance or human inter- [illegible] should be an evaluation of all system failure [illegible] of occurrence.

(3) Accident Sequence [illegible] Evaluations

A main [illegible] accident prevention is the identification of combinations [illegible], or features which when failed in combination lead [illegible] or radiation releases in excess of 10 CFR [illegible] classes of accident initiation event trees should [illegible] to display [illegible] combinations of systems and features that effect the likelihood [illegible] core damage. System-level logic displays of accident sequences, similar to system-level evaluations only on a larger scale, that [illegible] or radiation releases in excess of 10 CFR 100 should [illegible] for each significant event tree branch.

As part of the process [illegible] accident sequences known intersystem dependencies [illegible] interfacing systems) should be modeled as well as potential adverse [illegible] other failed systems and operator errors. The output [illegible] a comprehensive list of all accident sequences [illegible] or radiation releases in excess of 10 CFR [illegible] and [illegible] of occurrence.

(4) [illegible]

The [illegible] the extent and significance of common [illegible] redundancy designed into systems and [illegible] by internal events such as [illegible] and pressure extremes, common [illegible] to degrading influences, fires, human [illegible], floods, light- [illegible] or missile impacts. [illegible] potential causative [illegible] each causative factor, [illegible] determine two factors:

[illegible] the causative factor

[illegible] likely to affect

Based on this two-step process, the cut sets can be screened to determine the extent and significance of common cause events.

For specific common cause events, especially external events, a two-phase process is useful in which the first phase bounds the problem and checks its significance. Should this preliminary study indicate that one or more accident sequences may contribute to the risk in a significant manner, a more detailed analysis to ascertain its risk significance more realistically may be warranted. The output from CCFAs should be a system by system comprehensive list of common cause failures and should feed into the system and accident sequence evaluations. Although CCFA is inherently part of component, system, and accident level evaluation, it is highlighted as a separate criterion to emphasize its inclusion in the overall program.

(5) Systems Interaction Analyses

One or more independent components or components of a redundant grouping of components may fail or become more unreliable because of the interaction with other adjacent or nearby system failures. For example, a high-energy-feed-line or steamline break could cause rotating machinery in proximity to fail, or a nonseismically qualified structure adjacent to one train of a seismically qualified system might collapse on the train degrading the system's redundancy.

Although consideration of these dependency conditions is made during design and construction, an organized approach to reviewing the facility for potential systems interaction is warranted. The methods employed may use appropriate lists of component cut sets found in the system and accident sequence level evaluations and the common cause failure derived in the CCFA. In-plant walk-throughs on a compartment basis are needed to check the potential of systems interaction causative factors such as seismic and high-energy-line breaks. A program to accomplish the above should be developed with the output being a a comprehensive list of potential interactions. This information may also be used as input to the system and accident sequence evaluations. A generic investigation of systems interaction is being pursued by the staff for LWRs, and a discussion of the relation of this program to CRBR is provided in Appendix B of this SER.

(6) Equipment Testing

Testing should be performed at the component and subsystem level to explore failure modes, equipment performance, and extended limits of operation in a qualitative reliability sense. Accelerated life testing can be employed to provide early feedback concerning potential failures. A test program should be developed and documented which provides data to demonstrate performance and support reliability assumptions. Emphasis regarding equipment testing should follow the guidelines as described under component-level evaluations. Namely, emphasis should be placed on equipment unique (or unique in application) to CRBRP. Well-documented reliability data can be used in conjunction with or in lieu of equipment testing. Equipment associated with the reactor shutdown system or shutdown heat removal system are of primary importance. In addition, it is expected that the natural circulation and direct heat removal service (DHRS) testing described in Section 4.4 of this SER will also contribute to the overall plant reliability assessment.

(7) Equipment Qualification

Qualification should be conducted to ensure that components and systems can perform their intended safety functions under the anticipated service conditions in which they are required to perform. Section 3 of the SER provides the staff's evaluation of the applicants' Equipment Qualification Program.

(8) Failure Evaluation

Procedures should be established to provide assurance that the cause and mode of each failure during development and operation of CRBRP are identified, that the potential safety and availability implications are evaluated, and that corrective action is taken.

Although quality assurance is not a reliability-gathering activity as defined in this Program, it is an integral part of reliability assurance and should be considered in the applicants' evaluations. Section 17 of the SER discusses the staff's evaluation of this program.

Considering the above activities and the various safety functions and features in CRBRP, a matrix showing specific elements, which in the staff's opinion are required for each function, is shown in Figure C.1.

C.2.2 Feedback to Design, Operation, Surveillance, and Maintenance

The above activities will provide a large and varied amount of reliability-oriented information regarding the safety functions of CRBRP.

The second element of the Program is that this information must be fed back into the design, operation, surveillance, and maintenance documentation in time to support final design as well as remain in place during the lifetime of the facility as a tool with which future changes and the impact of operating experience can be assessed. As a result of this requirement, there will be a number of decisions to be made by the applicants regarding whether changes should be implemented. Thus, there is a need to ensure that the process by which the information is fed back into the design and the criteria or rationale used to control this process are documented and auditable. The key criteria to be used by the applicants to determine whether or not design changes will be implemented need to be documented. Generic criteria applicable project wide are preferred with additional considerations or criteria on a case-by-case basis. For example, specific reliability information may be compared against the principal design criteria, against comparable performance in modern LWRs, or against NRC's safety goals. Further, the reliability information may be compared internally to identify specific large contributors to risk. The probabilistic risk assessment may also be an acceptable tool to help guide judgments regarding design and operational improvements. In any event, the final decisions will be based on engineering judgment using some of the above or other considerations as appropriate. Regardless of the specific considerations utilized, it is important that the applicants provide clear documentation to assist the staff in understanding these considerations and how they are applied in the feedback on the design, operation, surveillance, and maintenance of CRBRP.

C.2.3 Traceability and Auditability

The third element of the Program, traceability and auditability, allows determination of the reliability function performed and verification of the appropriateness of its performance. This element requires clear documentation of all elements of the Program. The staff desires documentation of the Program plan before the operating license review so that the program can be audited before completion of final design.

An example of a required traceable, auditable function would be performance tests of reactor components. Documents must be available indicating the tests performed, the test conditions, and the test results.

C.2.4 Schedule Requirements for Program

The basic design features of the plant which contribute to reliability are those of redundancy, diversity, and independence of safety equipment. These are established by the principal design criteria and construction permit review. The intent of this Program is to enhance and evaluate the reliable performance of the plant safety functions. It is the staff's judgment that implementation of the Program should be on a time scale which allows impact on design, operation, maintenance, and surveillance, if the results indicate change is warranted.

It is also the staff's judgment that this Program should not end at the completion of final design, but rather, should continue throughout the life of the plant as a tool for assessing the impact and acceptability of plant design and procedure changes and the impact of plant operating experience on overall plant risk.

The applicants' schedule for implementing the Program consistent with the above is required.

C.3 APPLICANTS' RELIABILITY ASSURANCE PROGRAM

The applicants' overall Reliability Assurance Program was outlined in a letter dated January 11, 1983 (J. R. Longenecker to P. S. Check, #HQ:S:83:184). This letter describes all of the efforts under way by the applicants to ensure reliable plant operation. The applicants' Program is composed of the following elements:

(1) Design approaches used in ensuring reliability--this includes design reviews, development, and environmental qualification testing, quality assurance, and safety analysis.

(2) The Safety-Related Reliability Program for the reactor shutdown system (RSS) and the reactor residual heat removal system (RRHRS) as described in Appendix C of the PSAR.

(3) The probabilistic risk assessment (PRA) for the entire plant as described in Appendix J of the PSAR.

(4) Key system reviews--review of the interfacing and safety aspects of all systems required for reactor residual heat removal. The review considers failure modes and effects, operation, maintenance, and testing. These

reviews are documented in a letter from J. R. Longenecker to P. S. Check, dated February 19, 1982 (#HQ:S:82-005).

(5) Systems interaction analysis--review of plant systems associated with maintaining high plant availability.

(6) Equipment testing--includes development testing on first-of-a-kind components to verify their performance. Is not intended to develop a statistical data base but may identify failure modes.

(7) Equipment qualification--a test program designed to qualify safety-related equipment to the environment and conditions under which it has to perform. This program is documented in Westinghouse report WARD-D-0165, "CRBR Requirements for Environmental Qualification of Class 1E Equipment."

(8) Failure evaluation--a program for the evaluation of failures resulting from the Equipment Testing Program.

(9) Quality assurance program--the applicants have described an approriate quality assurance program including the following programmatic practices:

    (a) program management
    (b) design control
    (c) procurement control
    (d) manufactoring and construction control
    (e) operation control

    and the following work-oriented practices:

    (a) inspection
    (b) examination
    (c) testing

Details of the quality assurance program are included in Chapter 17 of the PSAR.

For a more detailed description of the applicants' program, the reader is referred to the documents referenced in this section.

C.4 ASSESSMENT OF APPLICANTS' PROGRAM

The applicant's Program, as outlined in the January 11, 1983 letter, contains many of the activities described in the staff's criteria in Section 2. Additionally, the applicants have reviewed the staff's criteria and have committed to revise their Program to comply with all of the staff's criteria (see letter J. R. Longenecker to J. N. Grace, "CRBRP Reliability Assurance Program," dated March 2, 1982, HQ:S:83:229). Based upon this commitment, the staff concludes that the applicants' Reliability Assurance Program is acceptable for a construction permit.

As part of the assessment of the reliability of those systems and features that prevent CDAs, it is essential that the effect of human error be considered. In this regard it is suggested that, as part of this program, the benefit of maintaining diversity in the operation, surveillance, and maintenance of those diverse plant systems associated with prevention of CDAs be explored. If maintaining diversity in this area would contribute significantly to maintaining the reliability of the functions, then it is suggested that this diversity be adopted in the plant operating philosophy or justification be provided as to why this is not desirable.

Because of the unique nature of this Program it is the staff's goal to work with the applicants to ensure development of a meaningful well-documented Program which can be used by both the applicants and the staff as a tool in assessing CRBRP reliability and risk.

It is the staff's plan that as design proceeds and the applicants' activities are further defined and implemented, the staff will periodically audit the Reliability Assurance Program to determine if it is accomplishing the intent of the above criteria and is being implemented in a fashion which contributes to the reliability of CRBRP.

PROGRAM ACTIVITIES / CRBR SAFETY FUNCTIONS (1)	REACTOR SHUTDOWN	SHUTDOWN HEAT REMOVAL	COOLANT SYSTEM BOUNDARY INTEGRITY (2)	PREVENTION OF CORE FLOW BLOCKAGE	PREVENTION OF FAILED FUEL PROPAGATION (3)	CONTAINMENT	EVST HEAT REMOVAL	BEYOND BASIS FEATURES
COMPONENT LEVEL EVALUATIONS	X	X	X	X	X	X	X	X
SYSTEM LEVEL EVALUATIONS	X	X	X	(6)	X	X	X	(6)
ACCIDENT SEQUENCE LEVEL EVALUATIONS	X	X	X	(5)	X	X	X	(6)
COMMON CAUSE FAILURE ANALYSIS	X	X	X	X	X	X	X	(6)
SYSTEMS INTERACTION ANALYSIS	X	X	X	(5)	X	X	X	(6)
EQUIPMENT TESTING (4)	X	X	X	X	X	X	X	X
EQUIPMENT QUALIFICATION	X	X	X	X	X	X	X	X
FAILURE EVALUATION	X	X	X	X	X	X	X	X

NOTES:
(1) The applicable front line and support systems for each function heading are those that are necessary to fullfill the specific safety function.
(2) Leak detection system is an active part of this function.
(3) Delayed neutron detection system is an active part of this function.
(4) Reliability testing of passive features is not required.
(5) Not required because these features are not a system.
(6) Not required because reliability emphasis should be on systems which prevent core disruptive accidents.

Figure C.1 Reliability assurance program activities required for each safety function

APPENDIX D

PROBABILISTIC RISK ASSESSMENT --
CLINCH RIVER BREEDER REACTOR PLANT

D.1 INTRODUCTION

The CRBRP Probabilistic Risk Assessment is one of the principal components of the applicants' Reliability Assurance Program. The PRA provides a mechanism for integrating the deterministic analyses (e.g., failure mode effects analysis, common cause failure analysis) into a complete model of the plant that can be used to obtain an understanding of the relative importance of individual systems and components to overall plant reliability and risks.

Since the Reactor Safety Study (published as WASH-1400, now NUREG-75/014) was performed in the early 1970s, probabilistic risk assessment (PRA) has increasingly been accepted as a means of assessing relative risks in nuclear power plant operations. One of the earliest such safety studies, after WASH-1400, was published as "CRBRP Safety Study, An Assessment of Accident Risks in the CRBRP," CRBRP-1, March 1977 (a Westinghouse document now out of date and not a docketed item). Acceptance of PRA has since reached the level where, in "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License," NUREG-0718, Revision 1, June 1981, Requirement II.B.8(1) states:

> Applicants shall: (1) commit to performing a site/plant-specific probabilistic risk assessment and incorporating the results of the assessment into the design of the facility. The commitment must include a program plan, acceptable to the staff, that demonstrates how the risk assessment program will be scheduled so as to influence system designs as they are being developed. The assessment shall be completed and submitted to NRC within two years of issuance of the construction permit. The outcome of this study and the NRC review of it will be a determination of specific preventive and mitigative actions to be implemented to reduce these risks. A prevention feature that must be considered is an additional decay heat removal system whose functional requirements and criteria would be derived from the PRA study.
>
> It is the aim of the Commission through these assessments to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant. Applicants are encouraged to take steps that are in harmony with this aim.

D.2 PRA PROGRAM PLAN

The applicants have prepared a probabilistic risk assessment program plan which was submitted in June 1982 and is incorporated into the PSAR as Appendix J. The program plan has been reviewed and found to be a responsive plan to meet the NUREG-0718, II.B.8 requirement. The plan includes what is, in the terminology of the PRA Procedures Guide (NUREG/CR-2300), a Level III PRA, that is, an in-depth PRA.

Major tasks under this part of the plan include initiator development, plant model development and quantification, core and containment accident modeling, and analysis of offsite consequences. Plant model development and quantification include system functional event tree development, fault tree development, analyses of plant response, accident sequence quantification, uncertainty analysis, and common cause failure analysis. Under core and containment accident modeling are the development of phenomenological event trees and the evaluation of source terms.

Other tasks in the program plan, which support application of the PRA, are the development of operator action event trees, an assessment of the effectiveness of design variations including consequence mitigation features, adaptation of the study to a continuing risk management program, providing input to site emergency procedures, and studies which aid in understanding the plant, such as evaluating sensitivities to testing and maintenance intervals.

D.3 PROGRAM PLAN IMPLEMENTATION

This PRA effort was begun about June 1981; the schedule calls for a final report in December 1984.

The PRA effort has been subdivided into two phases. Products from Phase I are a list of initiating events, a set of system event trees, a set of phenomenological event trees with heat transport states and success criteria, a package of fault trees with the data base for quantification, quantification of dominant accident sequences, a dependency analysis, and a sensitivity analysis. The products of Phase I were delivered in early February 1983.

Phase II Part A of the PRA effort includes review and validation of Phase I, plus the tasks remaining to satisfy Level III PRA requirements of the PRA Procedures Guide, NUREG/CR-2300. This includes the radionuclide release, health consequence and risk analyses, the uncertainty analysis, and the common cause failure analysis. Phase II Part B consists of PRA application tasks including adaptation of the PRA to the continuing risk management program, in which application of the PRA can continue through the operating life of the plant.

The applicants had under contract for Phase I, for the accident sequence definition and quantification, EG&G of Idaho, assisted by Wood-Leaver & Associates, Inc. The firm of Fauske & Associates, Inc. was under contract for the accident process analysis.

The Technology for Energy Corp. (TEC) of Knoxville, Tennessee, was awarded the contract for Phase II of the PRA. The results of Phase I have been transferred to TEC.

D.4 NRC REVIEW

The staff is conducting a review of the applicants' PRA effort in which the staff maintains cognizance of applicants' ongoing efforts and provides review and comment for product documents at various stages of their development. The review effort is being conducted with contracted assistance from Science Applications, Inc.

Activities of the review effort include continued monitoring of the ongoing PRA effort by review of the PRA products and by participating in interaction meetings with the applicants, detailed review of specific major elements of the study, and integrated review of the overall PRA. The applicants have committed to interactive meetings to convey early information on methodology and interim results to facilitate the staff review.

Other efforts by the staff related to the review of the PRA are the performance of selected independent assessments, that is, a risk reduction feasibility study of selected modifications to CRBRP safety systems, and a preliminary estimate of release frequencies for CRBRP potential core disruptive accidents.

D.5 FUNCTIONS OF THE PRA

D.5.1 Principal Functions

In addition to its primary function in the CRBRP Reliability Assurance Program as the integrated plant model used to determine the relative importance of individual systems and components to plant reliability and safety, the principal functions of the PRA are (1) to identify specific preventive and mitigative actions to reduce risks, (2) to feed back to the facility design process information which can permit any identified cost-effective risk reduction to be incorporated in the design, (3) to feed back to the reliability program any information needs that the reliability program can provide toward improved risk management. In addition, the PRA establishes the foundation and framework for a continuing risk management program as an aid to plant operations.

D.5.2 Safety Objective and Safety Goals

In the "Final Environmental Statement Related to Construction and Operation of Clinch River Breeder Reactor Plant," NUREG-0139, February 1977, Appendix I, is a letter of May 6, 1976, in which the following, concerning a safety objective, was stated:

> We use the further safety objective that there be no greater than one chance in one million per year for potential consequences greater than the 10 CFR 100 dose guidelines for an individual plant, for example, CRBR; this is a design objective rather than a fixed number which must be demonstrated for a given plant.

This safety objective has been used as an "aiming point" in the safety review of CRBRP.

However, the Commission will issue a Policy Statement on Safety Goals for the Operation of Nuclear Power Plants in the Federal Register. In this Policy Statement the Commission will set forth:

(1) Two qualitative safety goals:

- Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.

- Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

(2) A cost-benefit guideline:

- The benefit of an incremental reduction of societal mortality risks should be compared with the associated costs on the basis of $1,000 per person-rem averted.

(3) Three quantitative design objectives:

- The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1%) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.

- The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1%) of the sum of cancer fatality risks resulting from all other causes.

- The likelihood of a nuclear reactor accident that results in a large-scale core melt should normally be less than one in 10,000 per year of reactor operation.

These three quantitative design objectives have been taken by the staff as a candidate to replace the earlier safety objective. Conceptually, PRA can provide results to compare with the quantitative design objectives. The Commission recognized that "because of the sizable uncertainties still present in the methods and the gaps in the data base...[for PRA]...the design objectives should be viewed as aiming points or numerical benchmarks which are subjected to revision." The CRBRP PRA can, however, be of value in indicating whether these "aiming points" are being adequately approached.

The qualitative safety goals supported by the quantitative design objectives have been adopted by the Commission for use during a 2-year evaluation period. They "will not be used in the licensing process or be interpreted as requiring the performance of probabilistic risk assessments during the evaluation period. The goals and objectives are also not to be litigated in the Commission's hearings." If following the 2-year evaluation period, the Commission should elect to extend implementation of the qualitative safety goals and quantitative design objectives to specific cases, for example, CRBRP, the CRBRP PRA will facilitate such further implementation.

D.5.3 Statement of Interim Policy

The Commission's statement of interim policy regarding nuclear power plant accident considerations under the National Environmental Policy Act of 1969 (45 FR 40101, June 13, 1980) requires environmental impact statements to "include a reasoned consideration of the environmental risks (impacts) attributable to accidents at the particular facility" in which "approximately equal attention shall be given to the probability of occurrence of releases and to the probability of occurrence of the environmental consequences of those releases." The statement of interim policy is applicable to environmental impact statements rather than to the safety review, and its requirements are met by the scoping analysis of the risks of accidents at CRBRP which the staff provided in Appendix J of the "Supplement to Final Environmental Statement Related to Construction and Operation of Clinch River Breeder Reactor Plant," NUREG-0139, Supplement No. 1, Vol. 2, October 1982. The Appendix J analysis showed the risk to be similar to that from LWR plants and acceptably low. The Appendix J analysis is independent of the PRA being performed by the applicants; however, the PRA is expected to confirm the results and conclusions of the Appendix J analysis.

D.6 REFERENCES

U.S. Nuclear Regulatory Commission, NUREG-75/014 (formerly WASH-1400), "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants." The Rasmussen Report, Oct. 1975.

---, NUREG-0139 "Final Environmental Statement Related to Construction and Operation of Clinch River Breeder Reactor Plant," Feb. 1977; Supplement No. 1, Vol. 2, Oct. 1982.

---, NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License," Rev. 1, June 1981.

---, NUREG/CR-2300, "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," Jan. 1983.

---, "Statement of Interim Policy Regarding Nuclear Power Plant Accident Considerations Under the National Environmental Policy Act of 1969" (48 FR 40101, June 13, 1980).

APPENDIX E

CHRONOLOGY

October 11, 1974	Project Management Corporation (PMC) and Tennessee Valley Authority (TVA) tender application, Chapter 2 of the PSAR (Vols. 1 and 2) and Environmental Report (ER) (Vols. 1-3), for license to construct and operate the Clinch River Breeder Reactor Plant (CRBRP).
November 14, 1974	Summary of meeting with PMC to discuss the design to be presented in the PSAR.
November 19, 1974	Letter to PMC rejecting the ER for lack of sufficient information and requesting additional information.
November 27, 1974	Letter to PMC requesting additional information on site hydrology.
December 19, 1974	Site visit by staff, PMC and their consultants, and State of Tennessee.
December 27, 1974	Summary of meeting with PMC, TVA, GE, and EPA on December 12, 1974 to discuss scram reliability.
February 14, 1975	Summary of meeting with PMC, Westinghouse, GE, and ERDA on January 23, 1975 to discuss core disruptive accident analysis.
March 31 & April 1, 1975	Site visit by staff, PMC, and TVA.
April 10, 1975	Letter to PMC advising that additional material submitted to satisfy major deficiencies in ER and Chapter 2 of PSAR are acceptable for staff review.
April 11, 1975	Application docketed.
April 11, 1975	PMC submits PSAR (Vols. 3-10) for acceptance review
April 15, 1975	Summary of meeting with PMC, GE, Westinghouse, and TVA on March 18, 1975 to discuss the reliability of systems designed to remove decay heat from CRBR.
April 16, 1975	Summary of meeting with PMC, ERDA, and Westinghouse on March 20, 1975 to discuss general design criteria.

May 5, 1975	Summary of meeting with PMC, Westinghouse, and ERDA on April 16, 1975 to discuss PMC's progress in the area of radiological source terms for routine releases.
June 5, 1975	Letter to PMC accepting PSAR and requesting additional information.
June 11, 1975	Summary of meeting with PMC, GE, and Westinghouse on May 30, 1975 to discuss information needed for environmental and site suitability reviews.
June 11, 1975	Letter to PMC requesting additional information on site suitability evaluation.
June 12, 1975	Notice of Hearing issued (40 FR 25708, June 18, 1975).
July 17, 1975	ACRS Subcommittee meeting to develop information for consideration of its review of the application.
July 28, 1975	PMC submits Amendment 1 to the PSAR, consisting of responses to requests for additional information.
July 31, 1975	Summary of meeting with PMC, Westinghouse, GE, and ERDA on July 11, 1975 to discuss the status of source terms for site suitability accidents.
August 8, 1975	Summary of meeting with Natural Resources Defense Council on July 14, 1975 to discuss and clarify the scope and status of the radiological and environmental reviews.
August 22, 1975	Summary of meeting with representatives of the Oak Ridge Gaseous Diffusion Plant on July 16, 1975 to discuss postulated releases of toxic chemicals which could adversely affect operation of the plant.
August 25, 1975	PMC submits Amendment 2 to the PSAR, consisting of responses to requests for additional information.
August 27, 1975	Letter to PMC clarifying staff's presentation at July 17, 1975 ACRS Subcommittee meeting relative to preliminary radiological dose assessments on the context of suitability of the proposed site for other reactor types as well as for the CRBRP.
August 29, 1975	PMC submits Amendment 3 to the PSAR consisting of additional responses to requests for information and site suitability source term for the parallel design.
September 2, 1975	Summary of meeting with ERDA and Westinghouse on August 14, 1975 to clarify staff comments made at previous meetings in the matter of requirements for an LWA, with emphasis on radiological site suitability.

September 5, 1975	Summary of meeting with ERDA and consultants on August 15, 1975 to discuss the safe shutdown earthquake and the intensity-acceleration relationship for the plant.
September 12, 1975	PMC submits Amendment 4 to the PSAR, consisting of the site suitability source term for the reference design.
October 6, 1975	Letter to PMC requesting additional information on codes and references cited in the PSAR.
October 6, 1975	PMC submits Amendment 5 to the PSAR consisting of updated appendices for the primary pipe rupture fallback system and the core disruptive accident accommodation.
October 8, 1975	Summary of meeting with ERDA and Westinghouse on August 19, 1975 to discuss current views on apparent critical areas under discussion between PMC and staff, with specific attention to the needs and requirements associated with staff decisions on an LWA.
October 17, 1975	Summary of meeting with ERDA on September 12, 1975 to discuss the seismic design analysis.
October 23, 1975	PMC submits Amendment 6 to the PSAR consisting of responses to requests for additional information and additional design information.
October 24, 1975	Lette to PMC concerning the establishment of a review schedule.
October 29, 1975	tter to PMC requesting additional information on the erence design.
November 4, 1975	Meeting with ERDA to discuss piping integrity and associated fracture mechanics studies.
November 7, 1975	PMC submits Amendment 7 to the PSAR consisting of responses to requests for additional information and an updated seismic model.
November 13, 1975	Summary of meeting with PMC, ERDA, Westinghouse, and Burns and Roe on October 21, 1975 to discuss the quality assurance program.
November 18, 1975	Letter to PMC requesting additional information.
November 20, 1975	PMC submits an updated shutdown system reliability assessment.
November 26, 1975	Summary of meeting with PMC, ERDA, Westinghouse, and PNL consultants on November 6, 1975 to discuss fuel design and fuel design limits.

December 3, 1975	Summary of meeting with ERDA and PMC on November 5, 1975 to discuss safety system classifications and to clarify PMC's interpretation of seismograph traces recorded at ORNL during injection well operations.
December 4, 1975	Letter to PMC providing additional clarification of the requests for additional information of October 29 and November 18, 1975.
December 5, 1975	Summary of meeting with State and local officials on September 17, 1975 to discuss their specific concerns with the CRBRP.
December 5, 1975	Letter to PMC requesting additional information.
December 11, 1975	Summary of meeting with ERDA and Westinghouse on November 14, 1975 to discuss ERDA-sponsored efforts to quantitatively assess the containment thermal margins in the reference design.
December 11, 1975	PMC submits Amendment 8 to the PSAR, consisting of responses to requests for additional information, a new Appendix 2-C to Chapter 2 incorporating test grouting program report, and revisions to the quality assurance program.
December 17, 1975	PMC submits Amendment 9 to the PSAR, consisting of responses to requests for additional information.
December 17, 1975	PMC submits progress report, "Summary of CRBRP Inherent Retention Analysis," providing scoping analysis of Class 9 events for the reference design.
December 18, 1975	PMC submits a topical report on piping integrity in the primary heat transport system.
December 30, 1975	Letter to PMC requesting additional information on parallel design features.
December 30, 1975	PMC submits Amendment 10 to PSAR, consisting of responses to requests for additional information.
January 9, 1976	Letter to PMC advising of the design criteria which will be used by NRR staff in review of the application.
January 9, 1976	PMC submits a reliability plan for activities which ensure that core disruptive accidents are of sufficiently low probability to be excluded from the design basis.
January 15, 1976	PMC sumbits Amendment 11 to the PSAR, consisting of responses to requests for additional information and WARD quality assurance plans.

January 15, 1976	PMC letter furnishing responses to questions on the industrial security plan (proprietary).
January 21, 1976	Summary of meeting with PMC, ERDA, and Westinghouse on November 13, 1975 to discuss the scope and content of the reliability assessment of the reactor shutdown system.
January 23, 1976	Summary of meeting with PMC, ERDA, and Westinghouse on January 13, 1976 to discuss lack of adequate PSAR documentation for R&D in support of CRBRP.
January 27, 1976	PMC submits report, "Update of the Preliminary Reliability Prediction for CRBRP Shutdown Heat Removal System."
January 28, 1976	Letter to PMC requesting GROWS, SPRAY, FXVARI, ANSYS, TRANSWRAP, PLAP, AND SOFIRE computer codes.
January 28, 1976	Letter to PMC requesting additional information on the possible use of land near the site not presently identified in the PSAR.
January 28, 1976	Letter to PMC requesting additional information concerning injection well activities at ORNL.
January 29, 1976	PMC letter advising that NRC recommended revision to GDC is acceptable for the application and submitting recommended clarification for GDC 15, 27, 29 and 35.
January 29, 1976	Letter to PMC advising of staff position on defining PSAR terminology important to the review.
January 30, 1976	PMC submits "Interim Status Report on Inherent Retention Capabilities of the CRBRP," dated January 1976.
February 2, 1976	Summary of meeting with PMC and ERDA on January 22, 1976 to discuss the site suitability source term.
February 6, 1976	PMC submits Amendment 12 to the PSAR, consisting of responses to requests for additional information.
February 9, 1976	Summary of meeting with PMC, ERDA and Westinghouse on January 16 to discuss NRR staff questions on the parallel design.
February 18-19, 1976	ACRS Subcommittee meeting.
February 20, 1976	PMC submits Amendment 13 to the PSAR, consisting of responses to requests for additional information.
February 20, 1976	PMC submits onsite meteorology and χ/Q calculation.

February 26, 1976	PMC submits response to staff question concerning the safe shutdown earthquake.
March 3, 1976	Summary of meeting with ERDA on February 5, 1976 to discuss system safety classification, design criteria, and piping failure outside containment.
March 5, 1976	Summary of meeting with ERDA on February 3, 1976 to present current understanding of accident energetics and its basis and current and/or future R&D aimed at improving this understanding.
March 5, 1976	Letter to PMC providing the results of staff assessment of their proposed revisions to safety classification and design criteria discussed in the February 2, 1976 meeting.
March 8, 1976	PMC submits Amendment 14 to the PSAR, consisting of responses to requests for additional information and new general arrangement drawings.
March 9, 1976	Letter from PMC transmitting WARD-D-0033, "Preliminary Thermal and Hydraulic Evaluations in the Development of the CRBRP Primary Control System Design."
March 12, 1976	PMC submits response to NRC position on site suitability source term.
March 21, 1976	Meeting with ERDA to discuss injection well activities.
March 22, 1976	Letter to PMC requesting referenced report used as basis for analyses regarding turbine failure and appropriate turbine missile protection.
April 1, 1976	Letter to PMC setting forth areas of disagreement relating to the core disruptive accident and energetics discussed at February 3, 1976 meeting.
April 1, 1976	PMC submits Amendment 15 to the PSAR, consisting of responses to requests for additional information; Appendix F to Chapter 17; and description of S&W quality assurance program.
April 9, 1976	PMC submits report on turbine missile data in response to NRC request dated March 22, 1976.
April 14, 1976	PMC submits Amendment 16 to the PSAR, consisting of responses to requests for additional information.
April 19, 1976	Summary of meeting with ERDA on March 5, 1976 to discuss sodium fire codes SPRAY, CACECO, and SOFIRE.

April 19, 1976	Summary of meeting with PMC on March 11, 1976 to discuss the intensity rating of the maximum historical earthquake and selection of the attendant design ground acceleration.
April 19, 1976	Summary of meeting with PMC on March 19, 1976 to discuss meteorology.
April 22, 1976	PMC submits report, "Third Level Thermal Margins in the CRBRP."
April 23, 1976	Letter to PMC expressing concern regarding their intent and/or capability to document information so as to expedite the resolution of technical issues.
April 30, 1976	PMC submits Amendment 17 to the PSAR, consisting of responses to requests for additional information, and updates to Chapter 17.
April 30, 1976	PMC submits Amendment 18 to PSAR, consisting of additional features to provide additional margin in the reference design.
May 6, 1976	Letter to ERDA (Denise to Caffey) providing comments and guidance on the overall approaches being evaluated and requesting their response.
May 6, 1976	ERDA, PMC and TVA submit Amendment 1 to the Clinch River application to reflect the realignment of responsibilities of the several participants in the project. (ERDA becomes the lead participant.)
May 13, 1976	ERDA submits Amendment 19 to the PSAR, consisting of responses to requests for additional information.
May 14, 1976	Summary of meeting with ERDA and PMC on April 7, 1976 to discuss round two questions on quality assurance.
May 20, 1976	Summary of meeting with ERDA on April 6, 1976 to discuss the reliability program.
May 24, 1976	Letter from ERDA, responding to NRC guidance on CRBRP licensing approach dated May 6, 1976.
May 25, 1976	Letter from ERDA providing additional R&D to support the core disruptive accident analysis.
May 27, 1976	Letter to ERDA providing staff position concerning the safe shutdown earthquake.
May 27, 1976	Letter from ERDA transmitting meteorological data for the period February 11 - March 31, 1976.

May 27, 1976	ERDA submits Amendment 20 to the PSAR, consisting of responses to requests for additional information.
May 27, 1976	ERDA submits additional information on fuel penetration models and experiments.
June 1, 1976	Summary of meeting with ERDA on March 10, 1976 to discuss decay heat removal system redundancy and diversity.
June 2, 1976	Letter from ERDA transmitting "Summary of CRBRP Transient Testing Portion of the Plan for the National LMFBR Mixed Oxide Fuel Transient Performance Program."
June 3, 1976	ERDA submits Amendment 21 to the PSAR, consisting of an assessment of the additional plant margin available under various postulated HCDA mechanical loading conditions.
June 9, 1976	ERDA submits WARD report, "The Development and Application of a Cumulative Mechanical Damage Function for Fuel Pin Failure Analysis in LMFBR Systems."
June 11, 1976	Letter from ERDA advising that a determination by NRC of an appropriate factor for wind meander and agreement is needed in order to calculate the χ/Q.
June 17, 1976	ERDA submits Amendment 22 to the PSAR, consisting of responses to requests for additional information.
June 21, 1976	ERDA submits plan for verification of natural circulation.
June 23, 1976	Letter to ERDA requesting additional information concerning the industrial security and emergency plans.
June 23-24, 1976	ACRS Subcommittee meeting.
June 27, 1976	Summary of meeting with ERDA on June 17, 1976 to discuss the TLTM report, their responses to NRC May 6, 1976 position letter, and schedule considerations.
June 30, 1976	ERDA submits Amendment 23 to the PSAR, consisting of responses to requests for additional information.
June 30, 1976	Summary of meeting with PMC on March 31 to discuss injection well activities.
July 1, 1976	ERDA submits information on the reactor vessel head margin shear rirg.
July 2, 1976	Letter to ERDA transmitting staff position on safe shutdown earthquake.

July 8, 1976	Letter from ERDA transmitting correction pages to the "Summary of CRBR Transient Testing Portion of the Plan for the National LMFBR Mixed Oxide Fuel Transient Performance Program."
July 9, 1976	ACRS meeting.
July 14, 1976	ERDA letter transmitting their position on HCDA and siting problems.
July 15, 1976	ERDA letter advising of plans for core drilling and test to determine potential onsite source for concrete aggregate and Class A fill.
July 16, 1976	ERDA letter requesting NRC agreement with their position on appropriate factor for wind meander.
July 22, 1976	ERDA submits Amendment 24 to the PSAR, consisting of responses to requests for information.
July 28, 1976	ERDA letter submitting additional information supporting assessment of plant margin in HCDA mechanical loading conditions.
July 30, 1976	ERDA letter submitting additional information on feature to accommodate site suitability source term.
August 5, 1976	ERDA letter transmitting leak detection information requested at June 18, 1976 meeting.
August 12, 1976	Letter from ERDA Project Office submitting summary of the June 17, 1976 meeting on third level thermal margin report.
August 13, 1976	ERDA submits Amendment 25 to the PSAR, consisting of responses to requests for additional information.
August 17, 1976	Letter to ERDA requesting additional information.
August 20, 1976	ACRS report on hypothetical core disruptive accident for liquid metal fast breeder reactors.
August 27, 1976	ERDA submits Amendment 26 to the PSAR, consisting of responses to requests for information and withdrawal of Appendix E, "Primary Pipe Rupture Accommodation."
August 27, 1976	ERDA letter advising of investigation of previously unidentified linears in vicinity of site.
August 27, 1976	ERDA letter enclosing a plan and schedule for an alternate fuel management scheme.

August 31, 1976	Summary of meeting with ERDA on August 3, 1976 to discuss analysis of structural and mechanical response to CDA.
September 1, 1976	ERDA letter requesting clarification of NRC guidance provided in May 6, 1976 letter concerning plutonium dose guidelines.
September 3, 1976	ERDA letter advising that their position regarding appropriate SSE ground acceleration continues to be 0.189.
September 8, 1976	Meeting with ERDA to discuss containment cell liners and design of basic pipe leaks.
September 8, 1976	ERDA letter providing summary of materials properties of reactor vessel head and surrounding structures.
September 9, 1976	Meeting with ERDA to discuss structural design aspects of the plant.
September 9, 1976	Letter to ERDA documenting NRC staff evaluation of short-term atmospheric dispersion.
September 15, 1976	Letter to ERDA requesting the AYER computer code.
September 16, 1976	Summary of meeting with ERDA on June 18, 1976 to discuss leakage detection for sodium piping.
September 17, 1976	Meeting with ERDA on CACECO code.
September 17, 1976	ERDA letter transmitting additional information on sodium leak detectors.
September 17, 1976	ERDA letter appealing the NRC staff requirement for site suitability source term stated in May 6, 1976 letter.
September 20, 1976	ERDA letter stating environmental qualification of safety-related instrumentation.
September 20, 1976	ERDA letter transmitting additional information on subassembly faults.
September 22, 1976	Meeting with ERDA to discuss NRC staff's CDA analysis.
September 23, 1976	ERDA letter advising that their evaluation of events beyond the design basis are scheduled to be available in early 1977.
September 24, 1976	Letter from ERDA Project Office transmitting additional information on sodium leak detectors.
September 28, 1976	ACRS Subcommittee meeting at Oak Ridge, Tennessee.

October 1, 1976	ERDA submits Amendment 27 to PSAR, consisting of responses to requests for additional information.
October 5, 1976	Summary of meeting with ERDA on August 26, 1976 to discuss their response to EICSB acceptance review and first round questions.
October 5, 1976	ERDA submits report, "Exposure Dependent Cladding Deformation," WARD-D-0146, July 1976 in response to staff request.
October 5, 1976	Meeting with ERDA to discuss atmospheric dispersion of effluents.
October 6, 1976	Letter to ERDA requesting cell design information.
October 7, 1976	Letter to ERDA emphasizing staff position on PHTS piping integrity and requesting additional information.
October 7, 1976	Letter to ERDA advising that staff is not able to confirm that OBE ground acceleration stated in ERDA's September 3, 1976 letter is appropriate and requesting schedule for providing justification to modify the selection criteria of OBE.
October 8, 1976	Letter to ERDA requesting information on materials compatibility between core debris and refractory materials.
October 8, 1976	ERDA letter transmitting reports of experimental information on halogen attenuation and fission gas bubble breakup.
October 14, 1976	ERDA submits Amendment 28 to PSAR, consisting of responses to requests for additional information.
October 15, 1976	ERDA letter addressing NRC staff positions regarding adequacy of decay heat removal system and outlining approach for resolution of concern.
October 15, 1976	ERDA letter providing information on cell liner design.
October 19, 1976	Letter to ERDA highlighting discrepancy in CDA analysis presented in PSAR and ANL/RAS 75-29 "An Analysis of Unprotected Transients Under Cooling and Transient Overpower Accidents in CRBR."
October 20, 1976	Summary of meeting with ERDA on September 27, 1976 to discuss status of their information regarding emergency planning provisions.
October 21, 1976	ERDA sumits the AYER computer code.

October 28, 1976	ERDA sumits Amendment 29 to PSAR, consisting of responses to request for additional information.
October 29, 1976	ERDA letter transmitting additional information on SRI scale model tests.
November 2, 1976	ERDA letter concerning resolution of the site suitability of source term.
November 5, 1976	ERDA letter transmitting an updated analysis of third level margins for the first 24 hours.
November 5, 1976	ERDA letter submitting report, "Fuel Rod Bowing," WARD-D-0150, August 1976, in response to NRC request.
November 11, 1976	ERDA submits Amendment 30 to PSAR, consisting of responses to requests for additional information.
November 23, 1976	Summary of meeting with ERDA on October 21, 1976 to discuss resolution of the site suitability source term.
November 24, 1976	ERDA letter transmitting report, "FORE-2M: A Modified Version of the FIRE-II Computer Program for the Analysis of LMFBR Transients" in response to staff request for information on this subject.
November 30, 1976	ERDA letter transmitting additional information on cell liners.
November 30, 1976	ERDA submits Amendment 31 to PSAR, consisting of responses to requests for additional information.
November 30, 1976	ERDA submits information on industrial security plan.
December 1, 1976	Letter to ERDA requesting additional information.
December 1, 1976	ERDA letter transmitting information on materials compatibility between core debris and refractory materials.
December 6, 1976	Letter to ERDA concerning implementation of CRBRP 1200 MJ "appeal" decision.
December 7, 1976	ERDA letter submitting information on CDA analyses.
December 17, 1976	ERDA submits "An Analysis of Reactivity Effects of Bubble Collapse in a Boiled-up Molten Pool in CRBRP" in response to staff request for information.
December 22, 1976	ERDA submits Amendment 32 to PSAR, consisting of responses to requests for additional information.

December 23, 1976	Letter from ERDA Project Office submitting schedule for round 0, 1, and 2 questions.
December 27, 1976	ERDA submits "TRANSWRAP--A Code for Analyzing the System Effects of Large Leak Sodium-Water Reactions in LMFBR Steam Generation" in response to staff request.
January 3, 1977	Summary of meeting held on October 13, 1976 with Directors of DPM and DSE and their staffs to discuss the staff position regarding CRBRP site suitability source term.
January 3, 1977	ERDA transmits 9 of 12 references to report on third level thermal margins.
January 13, 1977	ERDA submits Amendment 33 to PSAR, consisting of responses to requests for additional information.
January 14, 1977	Summary of meeting held on October 13-14, 1976 with CRBRP representatives to discuss fuel design limits; bases and criteria; and R&D commitments related to fuel design.
January 14, 1977	Letter to ERDA advising that as a result of unscheduled receipt of all necessary information, staff unable to conduct complete review of pipe integrity in limited time suggested by them; pending satisfactory review and resolution of piping integrity issue, cannot agree that PHTS pipe breaks should not be considered a design-basis event.
January 18, 1977	Meeting to discuss structural adequacy of reactor head design.
January 18, 1977	Letter to ERDA transmitting a copy of the December 15, 1976 meeting summary and advising that timely and satisfactory course of action to resolve staff's concerns is not evident.
February 3, 1977	ERDA submits Amendment 34 to PSAR consisting of responses to requests for additional information.
February 4, 1977	Letter to ERDA providing supplementary comments regarding their summary of the December 15, 1976 meeting.
February 11, 1977	ERDA submits report "Simulation Model, DAHRS," CRP002, Revision 3, and "Flow Induced Vibration of Fuel Rods in CRBRP," WARD-0166, December 1976, in response to staff's request for additional information.
February 16, 1977	ERDA submits "Radial Blanket Power to Melt Analysis" in response to request for additional information.

February 17, 1977	ERDA submits "A Recent Evaluation of Foreign Wastage Data from Sodium-Water Reaction Investigation" and "Summary of Design and Development Status of the Liquid Metal to Gas Leak Detection System for the CRBRP" in response to a staff request for information.
February 18, 1977	ERDA submits Amendment 35 to PSAR, consisting of responses to requests for additional information.
March 7, 1977	ERDA submits additional information pertaining to the leak detection system.
March 11, 1977	ERDA submits Amendment 36 to PSAR, consisting of responses to requests for additional information.
March 14, 1977	ERDA transmits "CRBRP Risk Assessment Report."
March 14, 1977	ERDA submits revised information concerning the industrial security plan.
March 15, 1977	Letter to ERDA transmitting Sandia report regarding strength characteristics of concrete at Sandia and requesting review to determine whether this concrete can be expected to be representative of that anticipated for CRBRP.
March 17, 1977	ERDA submits report, "Seismic Evaluation Methods and Criteria for CRBR Fuel; Assembly Duct Structure," WARD-D-0158, October 1976, in response to request for additional information.
March 22, 1977	Letter to ERDA requesting additional information on SRI test programs.
March 23, 1977	Letter from ERDA Project Office informing NRC of their reevaluation of the component fabrication delays.
March 24, 1977	Followup letter from ERDA Project Office on inservice inspection, leak detection, safeguards, and load combinations.
March 25, 1977	ERDA submits Amendment 37 to PSAR, consisting of responses to requests for additional information.
March 25, 1977	ERDA submits, at request of ACRS, a document providing an overview of CRBR design.
March 30, 1977	Letter to ERDA requesting additional information on third level thermal margin report, protection against core meltdown.
April 1, 1977	ERDA submits revised description and schedule for site preparation activities.

April 5, 1977	ERDA submits letter to L. W. Coffee from R. J. Hart, OROO, concerning analysis of potential impact of CRBR operation on ORNL and Oak Ridge Gaseous Diffusion Plant.
April 7, 1977	ERDA submits drawings of models to be used in SRI tests.
April 21, 1977	ERDA submits seismic margin report.
April 22, 1977	ERDA submits Amendment 38 to PSAR, consisting of responses to requests for additional information and revisions to Chapter 14 providing test abstracts that define summary test objectives for first-of-a-kind principal design feature; also response to staff concerns about fuel design.
April 27, 1977	Summary of meeting held on March 9, 1977 with CRBRP representatives and their contractors to discuss the seismic analysis and design margins in the CRBR design.
April 28, 1977	ERDA submits "Plan for the National LMFBR Mixed Oxide Fuel Transient Performance Program."
May 5, 1977	ERDA letter requesting that staff shift review emphasis from environmental hearing preparation to resolution of so-called CP issues.
May 9, 1977	ERDA letter concerning status of agreement between ERDA and NRC relative to treatment of postulated core descriptive events.
May 11, 1977	Summary of meeting held on February 15, 1977 with CRBRP representatives at Westinghouse (WARD) to review the plant protection systems.
May 19, 1977	Letter to ERDA concerning the analysis of margin shear ring and transmitting the March 16, 1977 meeting summary on the subject.
May 20, 1977	Summary of meeting held on March 3, 1977 with CRBRP representatives and ANL to discuss additional calculations and analyses performed by ANL of the LOF accident using the SAS3D computer code.
May 27, 1977	ERDA submits Amendment 39 to PSAR, consisting of responses to requests for additional information.
May 27, 1977	Letter to ERDA (Denise to Caffey) concerning the confusion and misunderstanding which continues to exist by Project of staff's intentions and responsibilities in its technical review of CRBR.

June 14, 1977	ERDA submits, in response to request for information, report, "Impact of Fuel Densification on CRBRP Fuel Performance," WARD-D-0168, March 1977.
July 12, 1977	ERDA letter providing tentative design mix and aggregate specifications for use in test programs.
July 15, 1977	ERDA submits Amendment 40 to PSAR, consisting of responses to requests for additional information.
July 15, 1977	ERDA submits topical report, "CRBRP Closure Head Capability for Third Level Structural Margin Loading," WARD-D-0176, June 1977.
July 18, 1977	Summary of meeting with ERDA on January 26-27, 1977 at ANL, Argonne, Illinois, to discuss LOF CDA energetics.
July 22, 1977	ERDA letter requesting clarification of NRC schedules for review of CRBR application and issuance of SER.
August 5, 1977	ERDA letter advising that information requested on failed fuel on October 6, 1975 was included in analysis of fuel failure propagation furnished on September 20, 1976.
August 8, 1977	ERDA submits report, "Internal/External Cladding Degradation," WARD-D-0147, February 1977.
August 9, 1977	ERDA submits description of LIFE III code.
August 29, 1977	ERDA submits report, "Geological Investigations," S:L: 1531, August 1976.
September 30, 1977	ERDA submits CRBRP piping integrity report.
October 2, 1977	ERDA submits revised GE turbine missile report.
October 14, 1977	DOE submits Amendment 41 to PSAR, consisting of responses to requests for additional information.
November 4, 1977	DOE submits Amendment 42 to PSAR, consisting of revisions to reactivity feedback component of overall power coefficient.
January 27, 1978	DOE submits Amendment 43 to PSAR, consisting of responses to requests for additional information.
March 20, 1978	DOE submits report entitled "Active Pump and Valve Operability Verification Plan," WARD-D-0174.
April 21, 1978	DOE submits Amendment 44 to PSAR, consisting of updates to sections on reactor refueling system, emergency and normal chilled water systems, and other updates and revisions.

May 24, 1978	DOE letter requesting the status on the current staff review of their application.
July 28, 1978	DOE submits Amendment 45 to PSAR, consisting of updates to chapter on quality assurance, to Appendix A, "Computer Codes," to impurity monitoring and analysis system, as well as responses to requests for additional information contained in NRC letter dated August 17, 1976.
August 17, 1978	DOE letter advising of plan to test for determination of potential onsite source for concrete aggregate.
August 25, 1978	DOE submits Amendment 46 to PSAR, consisting of revisions to geology and seismology, seismic design, auxiliary liquid metal system, and general plant description.
September 1, 1978	DOE letter transmitting responses to seismic design questions.
October 6, 1978	DOE submits Topical Report WARD-D-0165, Revision 1, "Requirements for Environmental Qualification of Class 1E Equipment."
November 9, 1978	Letter from W. P. Gammill, NRC, to L.W. Caffey, Director, CRBRP Project, Subject: NRC Discontinuing the safety review of the CRBRP and the staff's status report on major outstanding issues.
November 14, 1978	DOE letter advising of potential industrial development adjacent to CRBRP site.
November 30, 1978	DOE submits Amendment 47 to PSAR, consisting of revisions to industrial security, communication system, compressed gas system, buckling stress criteria and other updates and revisions.
December 13, 1978	DOE submits reports, "Structural Response of CRBRP Scale Models to a Simulated Hypothetical Core Disruptive Accident" (WARD-D-0218), and "Closure Head Capability for Structural Margin Beyond Design Base Loading" (WARD-D-0178).
January 3, 1979	DOE letter concerning CRBRP licensing status.
February 16, 1979	DOE submits topical reports on loss of heat sink, WARD-D-0169 and WARD-D-0170.
February 16, 1979	DOE submits topical report on HCDA's CRBRP-GEFR-00103.
February 23, 1979	DOE submits Amendment 48 to PSAR, consisting of revisions to inert gas receiving and processing system, conventional fire protection system, and other revisions.

March 5, 1979	DOE letter evaluating the NRC staff review of CRBRP.
April 20, 1979	DOE submits Amendment 49 to PSAR consisting of revisions to heating, ventilating, and air conditioning system; radioactive waste management; radiation protection; and other revisions.
June 1, 1979	DOE letter transmitting updated information on industrial security.
June 29, 1979	DOE submitted Amendment 50 to the PSAR.
September 14, 1979	DOE submitted Amendment 51 to the PSAR.
October 19, 1979	DOE submitted Amendment 52 to the PSAR.
December 14, 1979	DOE forwards WARD-D-0050, Revision 3, "Facility Core Assembly Hot Channel Factors Preliminary Analysis."
January 31, 1980	DOE submitted Amendment 53 to the PSAR.
March 11, 1980	DOE forwards WARD-D-0210, "Predicted Steady State Thermal Hydraulic Performance of Fuel and Blanket Assemblies in Plant Heterogeneous Core, Rev. L."
March 25, 1980	DOE submitted CRBRP-3, Vol. 2, "Hypothetical Core Disruptive Accident Considerations: Assessment of Thermal Margin Beyond Design Base."
April 4, 1980	DOE submitted final report on base materials tests for cell liner steels.
April 11, 1980	DOE submitted Revision 1 to WARD-D-0218, "Structural Response of Scale Model to Simulated Hypothetical Core Disruptive Accident."
June 5, 1980	DOE submitted Amendment 54 to the PSAR.
June 27, 1980	DOE submitted Amendment 55 to the PSAR.
June 27, 1980	DOE submitted physical security plan.
June 27, 1980	DOE submitted revised responses to questions 421.3 and 421.10 regarding physical security plan.
August 22, 1980	DOE submitted CRBRP-ARD-0204, "CRBRP Fuel Assembly Structural Analysis in Support of the Final Design Review."
August 29, 1980	DOE submitted Amendment 56 to the PSAR.
November 7, 1980	DOE submitted Amendment 57 to the PSAR.

November 26, 1980	DOE submitted Amendment 58 to the PSAR.
November 28, 1980	DOE provided information concerning pre-test prediction of FFTF natural circulation.
December 30, 1980	DOE submitted Amendment 59 to the PSAR.
February 13, 1981	DOE submitted Amendment 60 to the PSAR.
August 13, 1981	Request from applicants for NRC to resume review of the CRBRP project.
September 18, 1981	Applicants submitted Amendment 61 to the PSAR which includes: updates to Section 1.4, "Identification of Project Participants"; Chapter 3, "Design Criteria--Structures, Components, Equipment and Systems"; Chapter 13, "Conduct of Operations"; Chapter 14, "Initial Tests and Operation"; Section 15.1.2, "Requirements and Criteria for Assessment of Fuel and Blanket Rod Transient Performance"; and Section 16.6, "Administrative Controls."
September 24, 1981	Letter to applicants apprising them of the steps NRC has taken in resumption of the review of CRBRP.
September 29, 1981	Summary of the general LMFBR design considerations and the specific CRB design features presented to the NRC by applicants on September 23, 1981.
October 6, 1981	Meeting notice for October 14 and 15, 1981 with applicants to discuss containment accommodation of core disruptive accidents.
October 19, 1981	Summary of the October 14 and 15, 1981 meeting with applicants.
October 23, 1981	Meeting notice for November 2 and 3, 1981 to discuss electric power systems, heat removal systems and probabilistic risk and reliability analysis,
November 9, 1981	Summary of the November 2 and 3, 1981 meeting with the applicants.
November 10, 1981	Notice of meeting with applicants for November 17, 1981 to discuss systems similar to LWR systems, unique systems, and Chapter 10 systems.
November 13, 1981	Applicants submitted WARD-D-0165, "CRBRP Requirements for Environmental Qualification of Class 1E Equipment," Revision 5.
November 13, 1981	Applicants submitted Revisions 1 and 2 of CRBRP-3, Volume 2, "Hypothetical Core Disruptive Accident Considerations in CRBRP: Assessment of Thermal Margin Beyond the Design Base".

November 13, 1981	Applicants submitted Amendment 62 to the PSAR which includes: updates from previous responses to requests for additional information; revisions to Section 1.4, "Identification of Project Participants"; Section 5.3, "Primary Heat Transport Systems"; Section 5.5, "Steam Generation System"; and an annual update to Chapter 17, "Quality Assurance."
November 13, 1981	Notice of meeting with applicants for November 24, 1981 to discuss CRBR equipment qualification program and compliance with NUREG-0588.
November 13, 1981	Notice of meeting with applicants for December 1, 1981 to discuss CRBRP physical security plan.
November 16, 1981	Notice of meeting with applicants for December 3, 1981 to discuss CRBR control room design.
November 16, 1981	Notice of meeting with applicants for December 10, 1981 to discuss CRBR emergency plans.
November 18, 1981	Letter to applicants requesting they address the informational, environmental, and programmatic changes that have occurred, and the regulatory guidance and requirements that have been promulgated since NRC's review was suspended.
November 19, 1981	Applicants submitted revised responses and revised PSAR figures to the CRBRP physical security plan.
November 20, 1981	Notice of meeting with applicants for December 8 to discuss applicability and compliance with regulatory guides.
November 20, 1981	Notice of meeting with applicants for December 9, 1981 to discuss TMI-related licensing requirements as defined in NUREG-0718, Revision 1.
November 20, 1981	Letter to applicants requesting submission of magnetic tape of onsite meteorological data for evaluation of the radiological consequences of normal and accidental releases to the atmosphere.
November 24, 1981	Notice of meeting with applicants for December 15, 1981 to discuss CRBR QA organization and QA plan.
November 30, 1981	Summary of the November 17, 1981 meeting with applicants.
November 30, 1981	Summary of the November 24, 1981 meeting with applicants.
November 30, 1981	Notice of meeting with applicants for December 14, 1981 to discuss CRBR instrumentation and control systems.
November 30, 1981	Applicants request authorization of the NRC, under 10 CFR 50.12, to conduct site preparation activities for the CRBRP project.

December 3, 1981	Notice of meeting with applicants for December 18, 1981 at Waltz Mill, Madison, Pennsylvania, for discussion and tour of Clinch River test facilities.
December 4, 1981	Summary of the December 1, 1981 meeting with applicants.
December 7, 1981	Summary of the December 3, 1981 meeting with applicants.
December 15, 1981	Summary of the December 10, 1981 meeting with applicants.
December 15, 1981	Summary of the December 14, 1981 meeting with applicants.
December 18, 1981	Applicants submitted Amendment 63 to the PSAR which includes: revisions to Section 1.4, "Identification of Project Participants"; Chapter 8, "Electric Power"; and Chapter 17, Appendix D, and Appendix E, "A Description of the Lead Reactor Manufacturer and Architect-Engineer Quality Assurance Programs."
December 28, 1981	Notice of meetings with applicants for January 11 and 12, 1982 to discuss CRBR electrical drawings and tour of electrical cabinets and prototype panels at Waltz Mill, Pennsylvania.
December 29, 1981	Summary of December 8 and 9, 1981 meetings with applicants.
December 30, 1981	Summary of December 15, 1981 meeting with applicants.
December 30, 1981	Letter to applicants requesting additional information in the geotechnical engineering area.
December 31, 1981	Applicants file with the NRC currently available documentation supporting the factual representations in the November 30, 1981, 10 CFR 50.12 exemption request.
January 6, 1982	Dircks to Commissioners: Staff Responses to Commission Requests--December 9, 1981 Briefing on CRBR Activities.
January 7, 1982	Notice of meeting with applicants for January 15, 1982 to discuss sodium-concrete interactions.
January 8, 1982	Applicants submitted topical report, "An Assessment of HCDA Energetics in the CRBR Heterogeneous Reactor Core, CRBRP-GERF-00523."
January 8, 1982	Notice of meeting with applicants for January 25, 1982 to discuss seismic and dynamic qualifications of mechanical and electrical equipment.
January 8, 1982	Notice of meeting with applicants for January 26, 1982 to discuss the natural circulation test results.

January 8, 1982	Notice of meeting with applicants for January 27, 1982 to discuss structural margin beyond design basis (SMBDB) phenomonology test programs.
January 13, 1982	Letter to applicants requesting additional information in the radiation protection area.
January 15, 1982	Notice of meeting with applicants for January 22, 1982 to discuss the impact of a possible request for an LWA-2 on the safety review schedule.
January 15, 1982	Notice of meeting with applicants for January 28, 1982 to discuss the Stanford Research HCDA scale model test.
January 22, 1982	Notice of meeting with applicants for February 11, 1982 to discuss Appendix R requirements and to discuss sodium fire protection.
January 22, 1982	Letter to applicants requesting additional information in the core energetics area.
January 25, 1982	Summary of the January 15, 1982 meeting with applicants.
January 26, 1982	Notice of meetings with applicants for February 9 and 10, 1982 to discuss the structural design within the design bases.
January 27, 1982	Notice of meeting with applicants--rescheduled from January 22, 1982 to February 8, 1982.
January 28, 1982	Notice of meeting with applicants for February 10, 1982 to discuss the ongoing sodium concrete interaction test programs at HEDL and Sandia laboratories.
January 29, 1982	Applicants submitted Amendment 64 to the PSAR which includes: new Section 6.4, "Cell Liner System"; and revisions to Chapter 4, "Reactor"; Section 6.2, "Containment Systems"; Section 9.2, "Maintenance"; Section 9.13.2, "Sodium Fire Protection System"; Section 11.23, "Gaseous Waste System"; and Section 15.6, "Sodium Spills."
February 2, 1982	Notice of meeting with applicants for February 16, 1982 to discuss CRBR structural design. (Rescheduled for February 17, 1982.)
February 5, 1982	Notice of meeting with applicants for February 12, 1982 to discuss the auxiliary liquid metal systems.
February 5, 1982	Notice of meeting with applicants for February 18, 1982 to discuss the qualification of the applicants as required by NUREG-0718.

February 8, 1982	Notice of meeting with applicants for February 24, 1982 to discuss the scope of loose parts monitoring for CRBR.
February 9, 1982	CRBR Program Office to ACRS--Providing copies of the CRBRP principal design criteria.
February 11, 1982	Notice of meeting with applicants for February 18, 1982 to discuss the structural margin beyond the design basis (SMBDB).
February 17, 1982	Notice of meeting with applicants for February 25 and 26, 1982 to discuss the CRBR accident analyses.
February 17, 1982	Letter from Los Alamos National Laboratory to NRC submitting a set of questions for PSAR Section 4.2, 15.1, and 15.2 to be responded to by applicants.
February 19, 1982	Letter to applicants requesting additional information on inservice inspection.
February 19, 1982	Applicants submitted requested information on core energetics.
February 19, 1982	Applicants submitted a "Summary Report on the Conduct of the Clinch River Breeder Reactor Plant (CRBRP) Key Systems Reviews," which provides a description and overview of system reviews conducted on the integrated performance of selected CRBRP systems.
February 24, 1982	Notice of meeting with applicants for March 4, 1982 to discuss the qualification of the applicants as required by NUREG-0718, Revision 2.
February 26, 1982	Summary of the February 18, 1982 meeting with applicants.
February 26, 1982	Letter to applicants requesting additional information on materials engineering.
February 26, 1982	Applicants submitted Amendment 65 to the PSAR which includes: revisions to Section 2.3, "Meteorology"; Section 9.3, "Auxiliary Liquid Metal System"; Chapter 11 "Radioactive Waste Management"; Chapter 12, "Radioactive Protection"; Section 13.3, "Emergency Planning"; and Appendix G, CRBRP Plan for Inservice and Preservice Inspections."
February 26, 1982	Letter to applicants requesting additional information on structural engineering.
February 26, 1982	Letter to applicants requesting additional information on mechanical engineering.

March 1, 1982	Memorandum to ACRS providing a list of special CRBR review matters that the CRBR Subcommittee is particularly interested in dealing with early.
March 3, 1982	Notice of meeting with applicants for March 10, 1982 at GE ARSD, Sunnyvale, California, to discuss structural margins beyond the design basis.
March 3, 1982	Applicants submitted a report entitled "Summary Report on the Current Assessment of the Natural Circulation Capability with the Heterogeneous Core," CRBRP-ARD-0308, which presents a description of the natural circulation event, the analysis methods, input data, and results of the current assessment of the CRBRP natural circulation capability with the heterogeneous core.
March 4, 1982	Letter to applicants requesting additional information on effluent treatment systems.
March 9, 1982	Letter to applicants requesting additional information on equipment qualification.
March 9, 1982	Summary of the February 24, 1982 meeting with applicants.
March 9, 1982	Summary of the February 25 and 26, 1982 meetings with applicants.
March 11, 1982	Summary of the February 17, 1982 meeting with applicants.
March 11, 1982	Summary of the March 2, 1982 meeting with applicants.
March 11, 1982	Letter to applicants requesting additional information on equipment qualification.
March 11, 1982	Letter to applicants requesting additional information on pipe rupture design criteria and mechanical component design.
March 12, 1982	Summary of January 25, 1982 meeting with applicants.
March 12, 1982	Summary of February 12, 1982 meeting with applicants.
March 12, 1982	Notice of meeting with applicants for March 23 and 24, 1982 to discuss the mecnanical, neutronic, and thermal-hydraulic design of the reactor core; design criteria, acceptance criteria; analysis tools; and their verification. (Postponed by applicants.)
March 12, 1982	Notice of meeting with applicants for March 25, 1982 to discuss the structural margin beyond the design basis.
March 15, 1982	Letter to applicants requesting additional information on auxiliary systems.

March 16, 1982	Letter to applicants requesting additional information on power systems.
March 17, 1982	Summary of January 26, 1982 meeting with applicants.
March 17, 1982	Summary of February 11, 1982 meeting with applicants.
March 17, 1982	Summary of March 4, 1982 meeting with applicants.
March 17, 1982	Notice of meeting with applicants for March 29, 1982 to discuss leak detection system.
March 17, 1982	Notice of meeting with applicants for April 1, 1982 to discuss containment systems.
March 17, 1982	Notice of meeting with applicants for April 6 and 7, 1982 to discuss CRBR materials and mechanical engineering.
March 17, 1982	Applicants submitted requested information in the radiation protection area.
March 19, 1982	Summary of February 9 and 10, 1982 meetings with applicants.
March 22, 1982	Applicants submitted Revision 3 of CRBRP-3, Volume 2, "Hypothetical Core Disruptive Accident Consideration in CRBRP; Assessment of Thermal Margin Beyond the Design Base (TMBDB)."
March 23, 1982	Summary of January 15, 1982 and February 10, 1982 meetings with applicants.
March 23, 1982	Summary of January 27, 1982, February 18, 1982, and March 10, 1982 meetings with applicants.
March 23, 1982	Letter to applicants requesting additional information on core performance.
March 23, 1982	Letter to applicants requesting additional information on chemical engineering.
March 24, 1982	Notice of meetings with applicants for April 5, 1982 to discuss Chapter 15, "Accident Analyses."
March 24, 1982	Notice of meetings with applicants for April 13 and 14, 1982 to discuss seismic and structural engineering. (Postponed.)
March 25, 1982	Letter to applicants requesting additional information on core performance.
March 25, 1982	Letter to applicants requesting they address the applicable safeguards regulations.

March 29, 1982	Notice of meeting with applicants for April 16, 1982 to discuss thermal margin beyond the design basis (TMBDB).
March 29, 1982	Applicants submitted requested information on core energetics.
March 31, 1982	Applicants submitted Amendment 67 to the PSAR which includes responses to NRC requests for additional information contained in a letter dated January 13, 1982; and revisions to Section 5.6, "Residual Heat Removal Systems"; Section 7.2, "Reactor Shutdown System"; and Section 7.9, "Operating Control Stations."
April 2, 1982	Summary of January 28, 1982 meeting with applicants.
April 7, 1982	Applicants submitted WARD-D-0165, Revision 6, "CRBRP requirements for Environmental Qualification of Class 1E Equipment."
April 8, 1982	Notice of meeting with applicants for April 26, 1982 at Westinghouse, Waltz Mill site, to discuss materials compatibility test facilities.
April 9, 1982	Letter to applicants requesting additional information on geology and seismology.
April 9, 1982	Letter to applicants requesting additional information on instrumentation and control systems.
April 13, 1982	Notice of meeting with applicants for May 6, 1982 to discuss CRBR management review. (Postponed.)
April 13, 1982	Notice of meeting with applicants for April 21, 1982 to discuss probabilistic risk assessment.
April 14, 1982	Applicants submitted requested information on CRBRP security systems.
April 16, 1982	Notice of meeting with applicants for April 27 at Argonne National Laboratory, Argonne, Illinois, to discuss structural margin beyond the design basis.
April 16, 1982	Notice of meetings with applicants for May 11 and 12, 1982 to discuss Chapter 14, "Reactor Design."
April 16, 1982	Notice of meetings with applicants for May 13 and 14, 1982 to discuss seismic and structural engineering. (Rescheduled from April 13 and 14, 1982.)
April 19, 1982	Applicants submitted requested information on the CRBRP inservice inspection program.

April 20, 1982	Applicants submitted a revision to CRBR-3, Volume 1, "Hypothetical Core Disruptive Accident Consideration in CRBRP: Energetics and Structural Margin Beyond the Design Base."
April 21, 1982	Applicants submitted requested information on chemical technology.
April 26, 1982	Applicants submitted a correction page to their inservice inspection response of April 19, 1982.
April 28, 1982	Summary of meeting with applicants on April 1, 1982 to discuss containment systems.
April 29, 1982	Applicants submitted requested information on chemical and mechanical engineering.
April 30, 1982	Letter to applicants requesting additional information on Chapter 15, "Accident Analyses."
May 7, 1982	Applicants submitted requested information on equipment qualification.
May 11, 1982	Summary of April 6 and 7, 1982 meeting with applicants.
May 11, 1982	Summary of March 29, 1982 meeting with applicants.
May 14, 1982	Letter to applicants requesting additional information on core disruptive accident analyses, the fuel-handling system, and sodium fire protection.
May 14, 1982	Letter to applicants requesting additional information on emergency planning.
May 14, 1982	Applicants submitted requested information on auxiliary systems.
May 14, 1982	Applicants submitted a list of topics and reports to be submitted in the near future in support of the CRBRP's assessment of thermal margin beyond the design base.
May 17, 1982	Applicants submitted requested information on equipment qualification.
May 17, 1982	Applicants submitted requested information on mechanical engineering.
May 17, 1982	Notice of meeting with applicants for June 3, 1982 to discuss licensee qualification. (Rescheduled meeting.)
May 18, 1982	Applicants submitted requested information on effluent treatment.

May 18, 1982	Summary of April 5, 1982 meeting with applicants.
May 26, 1982	Applicants submitted Amendment 2 to their "Statement of General Information."
May 28, 1982	Applicants submitted Amendment 68 to the PSAR which includes responses to CRBR Program Office's requests for additional information contained in letters dated February 26, 1982 and March 28, 1982 and revisions to Section 13.7, "Radiological Security."
June 1, 1982	Applicants submitted requested information on structural engineering.
June 1, 1982	Applicants submitted requested information on core performance.
June 1, 1982	Applicants submitted requested information on the reactor system, heat transport piping system, and Class 1E equipment qualification.
June 1, 1982	Applicants submitted requested information on core performance.
June 1, 1982	Applicants submitted requested information on power systems.
June 2, 1982	Notice of meeting with applicants for July 27, 1982 at Waltz Mill site, Madison, Pennslyvania, to discuss steam generator system.
June 2, 1982	Applicants submitted requested information on seismic qualification of mechanical components, materials engineering, reactor physics, and seismic structures.
June 2, 1982	Summary of March 25, 1982 and April 27, 1982 meetings with applicants.
June 3, 1982	Summary of May 11 and 12, 1982 meetings with applicants.
June 7, 1982	Notice of meeting with applicants for June 18 to discuss sodium fire protection.
June 8, 1982	Notice of meeting with applicants for June 22, 1982 to discuss mechanical, nuclear, and thermal hydraulic design of the CRBRP core.
June 8, 1982	Notice of meeting with applicants for June 23, 1982 to discuss fuel failure monitoring system.
June 8, 1982	Applicants submitted requested information on structural engineering.

June 8, 1982	Applicants submitted requested information on instrumentation and controls and design criteria.
June 8, 1982	Applicants submitted requested information on geology and seismology.
June 8, 1982	Applicants submitted a description of the CRBRP steam generator test program and a detailed analysis of the May 25, 1982, General Accounting Office report.
June 9, 1982	Letter to applicants requesting additional experiments to confirm the structural capability of the CRBRP vessel head to accommodate core disruptive accidents and to benchmark the analytical models used to analyze the vessel head response and failure modes.
June 9, 1982	Letter to applicants requesting additional information on nuclear design.
June 9, 1982	Letter to applicants requesting additional design layout drawings.
June 9, 1982	Notice of meeting with applicants for June 16-17, 1982 to discuss structural margin beyond the design basis. (Cancelled.)
June 9, 1982	Notice of meeting with applicants for June 22-24, 1982 at Burns and Roe, Oradell, New Jersey, to perform a CRBR seismic and structural engineering audit of calculations.
June 10, 1982	Applicants submitted a copy of the report entitled "Verification of Natural Circulation in the Clinch River Breeder Reactor Plant--An Update."
June 11, 1982	Memorandum from CRBR Program Office to ACRS transmitting NUREG-0786 "CRBRP Site Suitability Report."
June 14, 1982	Applicants submitted requested information on piping design, auxiliary systems, and instrumentation and control systems.
June 16, 1982	Notice of meeting with applicants for June 30, 1982 and July 1, 1982 to discuss structural margin beyond the design basis.
June 17, 1982	Applicants submitted the following report "ES-LPD-82-007, 008, 009, 011" and requested information on materials engineering.
June 17, 1982	Applicants submitted requested information on mechanical and structural engineering.

June 17, 1982	Applicants submitted requested information on the dynamic and static analysis used to determine the structural and functional integrity of selected seismic Category I components.
June 18, 1982	Applicants submitted requested information on power systems and core performance.
June 21, 1982	CRBR Program Office requested additional information on the core disruptive accident energetics analyses presented in GEFR-0523.
June 21, 1982	Applicants submitted requested information on the probabilistic risk assessment program plan.
June 21, 1982	Applicants submitted information requested by the CRBR Program Office Technical Review Section.
June 25, 1982	Applicants submitted a revised PSAR figure for the CRBPP physical security plan.
June 25, 1982	Applicants submitted requested isometric drawings on the piping fabrication for the direct heat removal system.
June 25, 1982	Applicants submitted information requested by the CRBR Program Office Technical Review Section.
June 25, 1982	Applicants submitted an update to PSAR Section 13.5 on plant procedures.
June 25, 1982	Applicants submitted the requested drawings P&ID BM502, "Main Steam System," and Instrument Loop Diagram BE4107, "Main Steam System."
June 29, 1982	Applicants submitted requested information on sodium fire protection.
June 29, 1982	Applicants submitted information requested by the Technical Review Section.
June 30, 1982	Notice of meeting with applicants for July 8, 1982 to discuss CRBR hydrology review.
June 30, 1982	Applicants submitted a revised response on instrumentation and control systems.
June 30, 1982	Applicants submitted information requested by CRBR Program Office Technical Review Section.
July 2, 1982	Applicants submitted requested information on core performance.
July 2, 1982	Applicants submitted the requested CACECO computer code

July 6, 1982	Applicants submitted requested information on ASME Publication PVP-63, "A Procedure to Evaluate Structural Adequacy of a Piping System in Creep Range."
July 7, 1982	Applicants submitted information requested by the CRBR Program Office Technical Review Section.
July 13, 1982	The USGS submitted input to NRR/GSB on the suitability of the CRBRP.
July 13, 1982	Letter from ACRS Chairman Shewmon to NRC Chairman Palladino with a report on the suitability of the CRBRP site.
July 13, 1982	Summary of April 16, 1982 meeting with applicants.
July 13, 1982	Summary of June 18, 1982 meeting with applicants.
July 13, 1982	Summary of June 22, 1982 meeting with applicants.
July 13, 1982	Summary of June 23, 1982 meeting with applicants.
July 14, 1982	Applicants submitted information requested by the CRBR Program Office Technical Review Section.
July 15, 1982	Applicants submitted information on the post-test analyses of the FFTF natural circulation tests--reports CRBRP-ARD-0310, "Verification of the CRBRP Natural Circulation Core Analyses Methodology with Data from FFTF Natural Circulation Tests--June 1982" and WARD-NC-94000-6 "DEMO Post Test Analysis of the FFTF Transient Natural Circulation Tests--June 1982."
July 15, 1982	Applicants submitted requested information on thermal and hydraulic design.
July 15, 1982	Applicants submitted information requested by the CRBR Program Office Technical Review Section.
July 16, 1982	CRBR Program Office asked applicants to assess the applicability of identified unresolved (some resolved) generic safety issues to CRBRP.
July 16, 1982	Notice of meeting with applicants for July 23, 1982 to discuss probabilistic risk assessment.
July 22, 1982	Applicants submitted Revision 4 of CRBRP-3, Volume 2, "Hypothetical Core Disruptive Accident Considerations in CRBRP; Assessment of Thermal Margin Beyond the Design Base."
July 26, 1982	CRBR Program Office to applicants transmitting a copy of the ACRS report on the site suitability of CRBRP to Chairman Palladino.

July 26, 1982	Notice of meeting with applicants to discuss auxiliary liquid metal system.
July 28, 1982	Applicants submitted requested information on emergency planning.
July 29, 1982	Applicants submitted a drawing as further response to emergency planning questions.
July 29, 1982	Applicants forwarded updated pages for reference 106 of PSAR Section 1.6, CRBRP-3, Volume 2, "Hypothetical Core Disruptive Accident Considerations in CRBR; Assessment of Thermal Margin Beyond the Design Base."
July 30, 1982	Applicants submitted information requested by the CRBR Program Office Technical Review Section.
July 30, 1982	Applicants submitted Amendment 69 to the PSAR which includes responses to CRBR Program Office's requests for additional information contained in letters dated February 26, 1982; March 11, 1982; March 15, 1982; March 23, 1982; March 25, 1982; and April 9, 1982; Revisions to Section 3.7, "Seismic Design"; Section 3.8, "Design of Category I Structures"; and Chapter 4, "Reactor."
July 30, 1982	Applicants submitted corrected page replacement guide to Amendment 69.
August 6, 1982	Applicants submitted design layout drawings for the containment penetrations, the containment ring stiffeners and overhead crane support, the structures within the containment-confinement annulus, cell, and cell liners, and the reactor vessel support ledge requested by CRBR Program Office.
August 6, 1982	Applicants submitted a request for authorization to proceed with LWA-2 activities.
August 10, 1982	Notice of meeting with applicants for August 17 to discuss thermal margins beyond the design base.
August 13, 1982	Summary of July 23 meeting with applicants. (Draft report on "Analysis of Nominal Heat Removal Capacity of the CRBRP in the Natural Circulation Mode.")
August 20, 1982	Applicants submitted Amendment 70 to the PSAR which includes responses to CRBR Program Office requests for additional information contained in letters dated February 26, 1982; revisions to Chapter 13, "Conduct of Operations"; Sections 17.0, 17.1, 17A, 17C, and 17F, "A Description of the Owner Assurance Program"; and Appendix C, "Safety Related Reliability Program."

August 20, 1982	Applicants submitted design layout drawings for all components and structures within, comprising, or attached to the reactor enclosure requested by CRBR Program Office.
August 23, 1982	Notice of meeting with applicants for September 15, 1982 to discuss thermal margin beyond the design base.
August 24, 1982	Notice of meeting with applicants September 8 and 9, 1982 to discuss mechanical engineering.
August 24, 1982	Summary of July 8, 1982 meeting with applicants.
August 24, 1982	Applicants submitted requested information on instrumentation and control systems and information requested by the CRBR Program Office Technical Review Section.
August 26, 1982	Applicants submitted requested information on sodium dump system, argon cover gas monitoring, reactor delayed neutron monitoring subsystem, and the effects of high temperatures in reference legs of steam drum water level measuring instruments.
August 31, 1982	Applicants informed CRBR Program Office of the initiation of site preparation activities.
September 1, 1982	Applicants submitted an action plan to resolve questions relating to monitoring component degradation in the nuclear steam supply systems.
September 7, 1982	CRBR Program Office provided ACRS with results of staff's review of potential effects of a CRBRP-type plant on the Oak Ridge Gaseous Diffusion Plant (K-25).
September 7, 1982	Summary of August 17, 1982 meeting with applicants.
September 8, 1982	Summary of August 5, 1982 meeting with applicants.
September 8, 1982	Notice of meeting with applicants for September 21 and 22, 1982 to discuss instrumentation and control systems.
September 8, 1982	Applicants submitted requested information on instrumentation and control systems not required for safety.
September 10, 1982	Notice of meeting with applicants for September 16 and 17, 1982 to discuss structural engineering.
September 13, 1982	Notice of meeting with applicants for September 15, 1982 to discuss LWA-2 TMBDB and SMBDB issues. (Revised from August 23, 1982 notice.)

September 13, 1982	Notice of meeting with applicants for September 21, 1982 at Argonne National Laboratory, Argonne, Illinois, to discuss hypothetical core-disruptive accidents and structural margins beyond the design basis.
September 14, 1982	Applicants submitted requested information on the soduium fire protection system, the fuel failure monitoring system, the design of the CRBR purge system, and the applicability of the RDT standards to safety-related instrumentation and control systems.
September 20, 1982	Summary of September 15, 1982 meeting with applicants.
September 21, 1982	Applicants submitted a summary of the September 8 and 9, 1982 meetings.
September 22, 1982	Notice of meeting with applicants for September 28, 1982 to discuss leak detection.
September 23, 1982	CRBR Program Office provided comments to DOE on the CRBRP Probabilistic Risk Assessment Program Plan.
September 24, 1982	Applicants submitted a summary of the September 21 and 22, 1982 meetings.
September 24, 1982	Notice of meeting with applicants for September 29, 1982 to discuss CRBR principal design criteria.
September 27, 1982	Notice of meeting with applicants for October 6, 7, and 8, 1982 to discuss direct heat removal system.
September 28, 1982	Notice of meeting with applicants for October 5, 1982 to discuss structural and seismic review.
September 28, 1982	Applicants submitted requested microfiche containing CACECO input/output for extreme penetration cases.
September 29, 1982	Applicants submitted information for review of CRBRP-3, Volume 2, "Letter Report, IMBDB Instrumentation Development."
September 30, 1982	Applicants submitted Amendment 71 to the PSAR which includes responses to CRBR Program Office requests for additional information contained in letters dated April 19 and 30, May 14, June 9 and 21, and July 16, 1982; revisions to Chapter 7, "Instrumentation and Controls"; and Sections 17E and 17I, "A Description of the A-E and GE-ARSD-RM Quality Assurance Programs."
October 4, 1982	Applicants submitted the additional information on instrumentation and control systems requested at the September 21 and 22, 1982 working meeting.

October 4, 1982	Applicants submitted requested information on instrumentation and control systems.
October 7, 1982	Notice of meeting with applicants for October 14, 1982 to discuss reactivity control.
October 7, 1982	Applicants submitted a report on "Preliminary Analysis of Heat Generating Blockages in CRBRP Fuel and Radial Blanket Assemblies To Determine Detection Requirements, CRBRPO-ARD-0119," in response to a request from the CRBR Program Office Technical Review Section.
October 7, 1982	Applicants submitted a summary of the September 28, 1982 meeting.
October 12, 1982	Notice of meeting with applicants for October 20 and 21, 1982 to discuss power systems.
October 12, 1982	CRBR Program Office requested additional information on thermal stress.
October 12, 1982	Notice of meeting with applicants for October 18, 1982 to discuss containment systems.
October 13, 1982	Notice of meeting with applicants for October 19 and 20, 1982 to discuss power systems. (Revised from October 12, 1982.)
October 14, 1982	Notice of meeting with applicants for October 19, 1982 to discuss reactor control room design.
October 14, 1982	Notice of meeting with applicants for October 20, 1982 to discuss CRBR thermal hydraulics.
October 15, 1982	Applicants submitted a summary of the September 21, 1982 meeting on HCDA energetics.
October 15, 1982	Applicants submitted a summary of the September meeting on ASME Code comparison.
October 20, 1982	Applicants submitted additional information requested at the September 8 and 9, 1982 meeting with the Mechanical Engineering Branch.
October 20, 1982	Applicants submitted a document entitled, "Thermal Margin Beyond the Design Base Sodium-Concrete Penetration Margins Assessment for the CRBRP."
October 21, 1982	Applicants submitted a summary of the October 19, 1982 meeting on control room design philosophy and approach.
October 21, 1982	Applicants submitted a summary of the October 20, 1982 meeting on the decay heat removal and thermal hydraulics

October 22, 1982	Notice of meeting with applicants for October 28 and 29, 1982 to discuss CRBR materials and mechanical issues, including leak before break and leak detection.
October 25, 1982	Applicants submitted requested information on instrumentation and control systems.
October 26, 1982	Applicants submitted additional information requested at the decay heat removal meeting of October 20, 1982.
October 29, 1982	Applicants submitted Amendment 72 to the PSAR, which includes responses to CRBR Program Office's requests for additional information contained in letters dated April 19 and 30, May 14, June 9 and 21, and July 16, 1982; revisions to Section 11.4, "Process and Effluent Radiological Monitoring System"; and Chapter 12, "Radiation Protection."
November 1, 1982	Notice of meeting with applicants for November 8, 1982 to discuss loose parts monitoring.
November 2, 1982	Applicants submitted additional information requested at the electrical power meeting of October 19, 1982.
November 3, 1982	Applicants submitted additional information requested at the instrumentation and control systems meeting of September 21 and 22, 1982.
November 3, 1982	Applicants submitted a summary of the October 28 and 29, 1982 meeting on piping integrity.
November 9, 1982	Applicants submitted a summary of the November 8, 1982 meeting on component degradation monitoring.
November 10, 1982	Notice of meeting with applicants for November 15, 1982 to discuss control rod logic design and function.
November 12, 1982	Notice of meeting with applicants for November 16 and 17, 1982 to discuss instrumentation and control.
November 12, 1982	Notice of meeting with applicants for November 17, 1982 to discuss structural engineering.
November 12, 1982	Notice of meeting with applicants for November 18 and 29, 1982 at the Project Office, Oak Ridge, Tennessee, to discuss CRBR control room design.
November 12, 1982	Applicants submitted a summary of the September 15, 1982 meeting on thermal margin beyond the design base.
November 12, 1982	Applicants submitted a report entitled, "Supplementary Manual for the FØRE-2M Computer Program, CRBRP-ARD-0257."

November 16, 1982	Notice of meeting with applicants for November 22 and 23, 1982 at Waltz Mill, Madison, Pennsylvania, to discuss mechanical engineering calculations.
November 19, 1982	Notice of meeting with applicants for November 22 and 23, to discuss reactor design.
November 23, 1982	Applicants submitted the additional information requested at the September 8 and 9, 1982 meeting with the Mechanical Engineering Branch
November 23, 1982	Applicants submitted a document entitled, "TMBDB Melting Scenario."
November 30, 1982	Applicants submitted Draft B of the report entitled, "Fire Hazard Analysis (FHAR)."
November 30, 1982	Applicants submitted Amendment 73 to the PSAR which includes revisions to Section 2.4, "Hydrologic Engineering"; Section 7.2, "Reactor Shutdown System"; and Section 17J, "A Description of the ESG-RM Quality Assurance Program."
December 1, 1982	Applicants submitted a response to item 6 of the action items from the October 18, 1982 meeting on containment systems.
December 1, 1982	Applicants submitted a summary of the November 23, 1982 meeting on equipment qualification.
December 2, 1982	Notice of meeting with applicants for December 8, 1982 to discuss inservice inspection review items.
December 2, 1982	Notice of meeting with applicants for December 8 and 9, 1982 to discuss structural engineering review items.
December 2, 1982	Notice of meeting with applicants for December 16, 1982 to discuss shutdown heat removal systems.
December 6, 1982	Applicants submitted a summary of the November 25 and 26, 1982 meeting on reactor design.
December 6, 1982	Applicants submitted requested information on electric power and mechanical systems.
December 6, 1982	Applicants submitted requested information on instrumentation and control systems.
December 6, 1982	Applicants submitted additional information requested at the December 2 and 3, 1982 meetings on control room design.
December 7, 1982	Applicants submitted requested information on thermal margins beyond the design base.

December 13, 1982	Summary of the September 15, 1982 meeting with applicants on thermal margins beyond the design base.
December 13, 1982	Applicants submitted a revision to "CRBRP-3 Volume 1, Structural Margin Beyond the Design Base."
December 14, 1982	Applicants submitted requested information on instrumentation and control systems.
December 14, 1982	Applicants submitted additional information requested by Mechanical Engineering Branch.
December 14, 1982	Applicants submitted a summary of the December 9, 1982 meeting on structural margin beyond the design base
December 17, 1982	Applicants submitted additional clarification of CRBRP training program.
December 20, 1982	Applicants submitted a summary of the December 16, 1982 meeting on shutdown heat removal.
December 20, 1982	Applicants submitted additional information on instrumentation and control systems.
December 20, 1982	Applicants submitted additional information requested at the December 15, 1982 meeting on plant auxiliary systems.
December 21, 1982	Applicants submitted additional information requested at the December 8, 1982 meeting on thermal margin beyond the design base.
December 21, 1982	Applicants submitted a summary of the December 20, 1982 meeting on the reliability program.
December 21, 1982	Applicants submitted requested information on the secondary control rod system.
December 21, 1982	Applicants submitted additional information requested at the December 9, 1982 meeting on structural margin beyond the design base loads on the reactor support ledge.
December 21, 1982	Applicants submitted a summary of the December 20, 1982 meeting on environmental qualification of equipment.
December 22, 1982	Applicants submitted additional information regarding emergency planning.
December 22, 1982	Applicants submitted requested information on the intermediate heat transport system tee.
December 22, 1982	Applicants submitted additional information requested at the November 22-24, 1982 meeting with the Mechanical Engineering Branch.

December 23, 1982	Applicants submitted requested information on instrumentation and control systems.
December 23, 1982	Applicants submitted requested information on containment systems.
December 23, 1982	Applicants submitted requested information on energetics analysis.
December 28, 1982	Applicants submitted requested information on margin in the plant protection system setpoints.
December 28, 1982	Applicants submitted requested information regarding the plant procedures.
December 28, 1982	Applicants submitted additional information on seismic qualification.
December 28, 1982	Applicants submitted requested information concerning project technical resources, training, and utilization of industry experience.
December 29, 1982	Applicants submitted a summary of the SER open-item meeting held on December 21, 1982.
December 29, 1982	Applicants submitted updated information on the environmental design of mechanical and electrical equipment.
December 29, 1982	Applicants submitted a summary of the December 8, 1982 meeting on containment vessel/code case(s) analysis.
December 30, 1982	Applicants submitted Amendment 74 to the PSAR which includes revised responses to NRC question CS430.1 through 104; revisions to Section 3.2, "Classifications of Structures, Systems and Components"; Section 6.2, "Containment Systems"; Chapter 7, "Instrumentation and Controls"; Chapter 8, "Electric Power"; and Section 9.14, "Diesel Generator Auxiliary Systems."
January 5, 1983	Applicants submitted a revision to Section 17D, "A Description of the Nuclear Steam Supply System (NSSS) Supplier Quality Assurance Program."
January 5, 1983	Applicants submitted the DEMO code output assumptions used for the pipe break analysis.
January 5, 1983	Applicants submitted requested information on seismic margin beyond the design base criteria and a writeup on the benchmarking analyses against the SM-1 test.
January 6, 1983	Applicants submitted requested information on the direct heat removal service.

January 7, 1983	Applicants submitted a report on the "Methodology for CRBRP's Application of Radiological Source Terms in Containment."
January 7, 1983	Applicants submitted additional information on the reactor vessel and ex-vessel storage tank non-destructive examination.
January 7, 1983	Applicants submitted responses to questions concerning auxiliary liquid metal systems and plant fire protection system.
January 10, 1983	EG&G report to CRBR Program Office entitled "Comparison of Clinch River Breeder Reactor Design Basis Accidents With Those for Light Water Reactors and Liquid-Metal-Cooled Fast Reactors," EGG-NTAP-6152.
January 11, 1983	Applicants submitted additional information on steam generator nondestructive examination and reactor vessel core support cone structural integrity.
January 11, 1983	Applicants submitted additional information on mechanical engineering.
January 11, 1983	Applicants submitted additional information on material surveillance.
January 11, 1983	Applicants submitted additional information on core instrumentation.
January 11, 1983	Applicants submitted personnel résumés of key positions for CRBRP management organization.
January 11, 1983	Applicants submitted clarifying information on the selection of the groundwater level for use in seismic design of Category 1 structures.
January 11, 1983	Applicants submitted the "CRBRP Reliability Assurance Activities" program.
January 12, 1983	Applicants submitted a summary of the November 17, 1982 meeting of seismic/structure/cell liner analysis and responses to questions brought up at the meeting.
January 12, 1983	Applicants submitted a modification of PSAR Section 14 clarifying the application of operational and test experience from similar operating reactors to the CRBRP test program
January 12, 1983	Applicants submitted additional and revised information on the plant auxiliary systems.

January 20, 1983	Applicants submitted additional information on inerted cells in the reactor service building.
January 20, 1983	Applicants submitted additional information on the composition of NaK and its solidus temperature.
January 21, 1983	Applicants submitted additional information on CRBRP engineered safety features and maintenance system.
January 25, 1983	Applicants submitted additional and revised information on CRBRP auxiliary systems.
January 26, 1983	Applicants submitted additional information on the electric power system.
January 26, 1983	Applicants submitted additional information on the instrumentation and control systems.
January 26, 1983	Applicants submitted additional information on the primary heat transport system hot-leg piping code evaluation.
January 27, 1983	Applicants submitted additional information on sodium spill volumes for inerted cells.
January 27, 1983	Applicants submitted additional information on reactor material surveillance.
January 27, 1983	Applicants submitted additional information on heat removal service temperature limits.
January 27, 1983	Applicants submitted additional information on qualification of mechanical equipment.
January 27, 1983	Applicants transmitted CRBRP-ARD-0315, "Clinch River Breeder Reactor Plant Verification of FØRE-2M Computer Code."
January 27, 1983	Applicants submitted additional information on stainless steel and insulation properties of engineered safety features and on welding qualification in areas of limited accessibility.
January 28, 1983	Notice of meeting with applicants for February 9, 1983 on Phase II of the Probabilistic Risk Assessment effort.
January 29, 1983	Applicants submitted additional information on mechanical engineering.
February 2, 1983	Applicants' response to the recently issued NRC CRBRP principal design criteria.

February 2, 1983	Applicants submitted two pages that were inadvertently left out of response dated December 14, 1982 on instrumentation and control systems.
February 2, 1983	Applicants submitted additional information on ex-vessel storage tank cooling.
February 3, 1983	Notice of meeting with applicants for February 9, 1983 with the Mechanical Engineering Branch.
February 4, 1983	Applicants submitted additional information resulting from open-items meeting of December 21, 1982.
February 4, 1983	Applicants submitted additional information on sodium spills.
February 4, 1983	Applicants submitted additional information on mitigation of waterhammer in the steam generator system.
February 4, 1983	Applicants submitted information on precautions that preclude assembly blockages.
February 8, 1983	Applicants submitted Amendment 75 to the PSAR which includes: Revisions to Section 1.4, "Identification of Project Participants"; Section 5.0, "Heat Transport and Connected Systems"; Chapter 7, "Instrumentation and Controls"; Chapter 9, "Auxiliary Systems"; and Section 17D, "A Description of the Westinghouse Quality Assurance Program."
February 10, 1983	Applicants forward correction pages to "PSAR Amendment 75 page replacement guide.
February 10, 1983	Applicants submitted information on confirmatory high temperature design programs.
February 14, 1983	Applicants submitted additional information on potential highway accidents with resulting toxic plumes that could impact CRBRP.
February 14, 1983	Applicants submitted additional information on nitrogen gas services system.
February 14, 1983	Letter from applicants on reactor closure head capability to meet margin requirements.
February 15, 1983	Applicants submitted additional information on instrumentation and control.
February 15, 1983	Applicants submitted additional information requested by the Mechanical Engineering Branch at the February 9, 1983 meeting.

February 15, 1983	Applicants submitted the revised Section 3.1 of the PSAR that incorporates the final principal design criteria.
February 23, 1983	Applicants submitted additional information on the circulating water system.
February 23, 1983	Applicants submitted additional information on the secondary control rod system.
February 24, 1983	Applicants submitted supplemental information to the Mechanical Engineering Branch.
February 25, 1983	CRBR Program Office transmits criteria requirements that the CRBRP Reliability Assurance Program must meet.
February 25, 1983	Applicants submitted additional informaton on plant emergency planning.
February 28, 1983	Applicants submitted additional information on nondestructive examination procedure.
February 28, 1983	Applicants submitted additional information on primary sodium gas entrainment and assembly flow blockage criteria.
March 2, 1983	Applicants provide further responses on the CRBRP Reliability Assurance Program.

APPENDIX F

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APPENDIX G

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---, RDT F9-4T, "Requirements for Construction of Class 1 Elevated Temperature Nuclear System Components," Sept 1974 and Jan. 19[illegible]

---, RDT F9-5, "Guidelines and Procedures for Design of Class 1 Elevated Temperature Nuclear System Components."

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---, RDT M2-6.

APPENDIX H

U.S. GEOLOGICAL SURVEY FINAL SEISMOLOGY AND GEOLOGY REVIEW OF THE CLINCH RIVER BREEDER REACTOR SITE

United States Department of the Interior

GEOLOGICAL SURVEY
RESTON, VA 22092

January 19, 1983

Mr. Robert E. Jackson
Chief, Geosciences Branch
Division of Engineering
U.S. Nuclear Regulatory Commission
Washington D.C. 20555

Dear Bob,

Enclosed is the Geological Survey final Seismology and Geology Review of the Clinch River Breeder Reactor Site.

Sincerely,

Robert H. Morris for

James F. Devine
Assistant Director
for Engineering Geology

Enclosure

Final Review
Geology
D. D. Dickey
R. C. McDowell
Seismology
D. M. Perkins
S. T. Algermissen
January 19, 1983

U.S. Department of Energy
Clinch River Breeder Reactor Plant
Oak Ridge, Tennessee
NRC Docket No. 50-537

Introduction

The U. S. Geological Survey has reviewed the geological and seismological data and analysis in the Preliminary Safety Analysis Report (PSAR) for the Clinch River Breeder Reactor Plant site located about 25 miles west of Knoxville, Tennessee

Geology

Introduction

The U.S. Geological Survey (USGS) reviewed the geologic data and analysis in the Preliminary Safety Analysis Report (PSAR) for the Clinch River Breeder Reactor plant site and has compared it to the geologic literature of the area including the Phipps Bend PSAR, Docket Nos. 50-553, 50-554, and the Watts Bar SER Docket Nos. 50-390 and 50-391. A field inspection of the site and surrounding area was made June 2 and 3, 1982.

The Clinch River site, in the southwest corner of the U.S. Department of Energy Oak Ridge Reservation, Roane County, Tennessee, is inside a meander loop of the Clinch River at the upper end of Watts Bar Lake.

The site is in the Valley and Ridge Physiographic Province, which extends about 500 miles northeastward from Alabama to Virginia and is about 25 to 50 miles wide. The northeast-trending valleys are underlain by easily erodible shale and mudstone and soluble limestone, whereas the ridges are supported by more resistant sandstone, siltstone, and siliceous limestone and dolomite.

The topography in the vicinity of the site is characterized by such northeast-trending ridges with intervening valleys. Normal lake level in the valley is about 740 ft above mean sea level and Chestnut Ridge to the northwest of the site stands at about 900 ft.

Stratigraphy

The Rome Formation and the Conasauga, Knox, and Chickamauga Groups constitute most of the bedrock of the Valley and Ridge Province in Tennessee. The Rome Formation, of Middle Cambrian age, consists mainly of red, green, and yellow shale, siltstone, sandstone, and minor gray dolomite, with a maximum exposed thickness of 1,200 ft. The Conasauga Group, of middle and Late Cambrian age, consists of about 2,000 ft mainly of alternating gray shale and limestone. The amount of limestone decreases northwestward where, at the province boundary, the Conasauga is nearly all shale. The 2,500-to 3,000-ft-thick Knox Group of Late Cambrian and Early Ordovician age is predominantly chert-bearing dolomite and lesser amounts of limestone. The Chickamauga Group of Middle and Late Ordovician age consists of alternating layers of gray and maroon limestone, calcareous siltstone, and shale. The thickness ranges from 8,000 ft in the southeastern part of the province to 2,000 ft in the northwest (PSAR, p. 2.5-4).

Calcareous mudstone and limestone of the Knox and Chickamauga Groups underlie the plant site. Typical strike and dip of the beds are N. 52° E. and 37° S.E. The bedrock is covered by a veneer of residual soil, through which scattered outcrops protrude in the central part of the site. The southern part of the site, and terrain near the river, are covered with alluvial soil. Weathered rock and soil attain a maximum thickness of 78 ft in the northeastern part of the site (PSAR, p. 2.5-15a). The applicant prepared a contour map of the top of "continuous rock," (unweathered rock which shows no significant discontinuities) based on 129 borings and seismic refraction work (PSAR, fig. 2.5-16).

Structure

During Paleozoic time, northwest-southeast compressional forces thrust rocks from the southeast over rock to the northwest. A succession of such thrust faults in the site area characteristically dip southeastward near the ground surface and flatten with depth (Harris and Milici, 1977, fig. 1, plate 5, 6). Swingle (1973, fig. 1) postulated a flat sole fault, which the thrust faults join, at a depth of about 9,000 ft. Harris and Bayer (1979, fig. 3) put the depth of the decollement at closer to 15,000 ft. This later work benefited from the COCORP seismic profiling (Cook and others, 1979). Rodgers (1970, p. 64) believes that the deformation and major structural features in the southern Appalachians were completed well before Late Triassic time.

The CRBRP site is located between two of these thrust faults--the Copper Creek and Whiteoak Mountain Faults. The Copper Creek fault at its closest point to the site is about 3,000 ft to the south. The strike and dip are N. 52° E. and 25° S.E. The site is near the midpoint of the 100-mile mapped length of the fault. The Rome Formation was thrust over rocks of the Chickamauga Group for a horizontal distance estimated in miles and a stratigraphic displacement of about 7,200 ft (PSAR, p. 2.5-21).

The Whiteoak Mountain Fault system consists of a main thrust fault with several subsidiary branch faults, the nearest trace being 1.7 miles northwest of the site. This northeasterly-trending fault is tens of miles long and is estimated to dip 45 to 50 degrees S.E. near the site (PSAR, p. 2.5-22).

Discussion

The general concept of the geology, presented by the applicant, is based upon a survey of the literature supplemented by drill core, radiometric dates, and geophysical work by them and their contractors. Although it is a simplistic presentation, we are in general agreement with the conclusions.

The items of major concern arising in our review of the PSAR were (1) the possibility of a limestone cavern underlying some portion of the site, because caverns are known to be present nearby, and (2) identification of active faulting, because seismicity is present in the province, although at a relatively low level.

Examination of the drill-core and the geologic cross-sections of the site drawn by the applicant, limitation of known caverns to the Knox Group (PSAR, p. 2.5-7), and the concept of "continuous rock" based on core-hole data and seismic refraction work, makes reasonable the applicant's contention that the presence of a major undetected cavity beneath a site structure is unlikely (PSAR, p. 2.5-15a).

Seismic events occur infrequently in the site area. The applicant states (PSAR, p. 2.5-25) without supporting data, that the "normal" focal depth for seismicity is 50,000 to 65,000 ft, well below the decollement and, therefore, unrelated to the shallow structure. Although data from the literature indicate that this is a reasonable hypothesis (for example, Bollinger and others, 1973), complete independence of seismicity and shallow structures has not been demonstrated, and the focal-depth range cited appears to be much too limited.

Recent thrust faulting in the Appalachians was the subject of a study by Schäfer (1979) (PSAR, p. 2F-3). His evidence for recent thrusting was offsets along subhorizontal fractures and bedding planes, of holes drilled during construction of roads. He noted such offsets, 12 years after roadway construction, at several locations, the closest to the Clinch River site being on Interstate Highway 40 between Harriman and Rockwood. Further study by Hatcher and Webb (1981) allowed them to conclude that offset is not a result of recent tectonism. Evidence of two kinds led them to this conclusion: (1) in multiple offsets amount of offset increases upward, and (2) offsets are not consistent in direction and favor a displacement direction toward the center of the highway. Stress relief, a factor noted in other studies in the Appalachians (Wyrick and Borchers, 1981) may be called upon as an explanation even for those offsets in directions parallel to directions of past thrust faulting.

Further, the applicant supports assignment of an ancient age for movement on the Copper Creek Fault with a radiometric age of mylonite from the fault zone of 285 million years (PSAR, p. 2.5-22). Although this is a reasonable date for major movement of the thrust faulting in the Valley and Ridge Province, such dating techniques do not preclude subsequent movement on the fault after erosion reduced the confining cover so that mylonite would not be formed. Evidence such as that from mapping and/or trenching of alluvial terraces across critical faults was not obtained. Such evidence could have demonstrated conclusively that the Copper Creek fault and Whiteoak Mountain fault are not capable.

Conclusion

In conclusion, although there has not been as definitive a demonstration as possible of noncapability for faulting in the area, the analysis of site geology by the applicant results in reasonable conclusions based upon current theories of Appalachian tectonics and upon the data available. It may be appropriate to note that to date no active faults have been recognized throughout the Appalachian region.

Seismology

Introduction

The U. S. Geological Survey has reviewed the seismological analysis in the Preliminary Safety Analysis Report (PSAR) for the Clinch River Breeder Reactor Plant site and compared it with the seismological literature for the region and with some results of on-going research in the Survey.

Applicant's safe-shutdown earthquake (SSE)

On seismicity maps prepared without specialized relocation techniques the Clinch River Breeder Reactor Plant (CRBRP) lies in the midst of a diffuse band of earthquake epicenters running roughly from Alabama to West Virginia. Because this band is spatially associated with the Appalachian mountains, it is natural, in the absence of more specific knowledge about the seismotectonics of the region, to consider that this seismicity is a feature of a so-called Southern Valley and Ridge seismotectonic province. The applicant has taken the largest historical earthquake in this province, the Giles County earthquake of 1897, and hypothesized a similiar eavent in the vicinity of the site. The applicant accepts an assessment of the maximum epicentral intensity of this earthquake as being a modified Mercalli intensity VII or VII+ (PSAR, p. 2.5-25). The safe-shutdown earthquake ground motion (SSE) is taken to be .25 g, corresponding to epicentral intensity VIII on a correlation of intensity with near-field strong motion acceleration (PSAR, p. 2.5-26). This intensity and the corresponding acceleration value are reasonable results of the application of Appendix A procedures to the regional seismic history. We point out that the recent analysis of Bollinger (1981) of the last 20 years of epicentral data in the Giles County area suggests the possibility of the existence of a structure [illegible] of a [illegible] magnitude between $M_s = 6.0$ and M_s [illegible] 7.0 that might [illegible] greater than VIII.

The conservatism of the applicant's SSE

Appendix A to 10 CFR Part 100 defines a deterministic procedure the purpose of which is to arrive at an assessment of maximum ground motion at a site. In assessing the conservatism of this SSE, we have looked at the exceedance probability of .25 g, when considered in the light of the assumptions we have made in producing probabilistic ground motion maps for the eastern United States (Algermissen and others, 1982). Included in the assumptions for these maps were that the seismicity was diffuse and uniform over an Appalachian province and that the earthquakes were crustal earthquakes that could be modelled as point sources near the surface. For these assumptions, .25 g has an annual exceedance probability of 2×10^{-4} if statistical variability in the attenuation function is not taken into account, or 4×10^{-4} if the standard deviation of the attenuation variability is taken to be 0.6.

Maximum magnitude is important in the above results. If the maximum magnitude in the model is assumed to be as low as $M_s = 5.8$, the above exceedance probabilities are expected to decrease by a factor of 3.

A possible local seismic source

The above results depend upon a model of diffuse uniform seismicity in a broad Appalachian source zone. Hadley and Devine (1974) show the Appalachian seismicity to have a "hot spot" in eastern Tennessee. Much more recently, Dewey (personal communication) and Gordon (personal communication) have relocated a large number of instrumentally recorded eastern U.S. earthquakes. (A list of these earthquakes and their relocated coordinates have been sent to the NRC and the applicant's consultants). Nine of these relocated earthquakes can be seen to make up a zone 15 km wide and 180 km long, extending from about 34°57'N lat., 84°36'W long., to 36°25'N lat., 83°40'W long. A line connecting these points runs through Knoxville and forms an azimuth of nearly 20 degrees more northerly than the surface trend of the Appalachians. This may represent a concentration of seismicity in eastern Tennessee. Although there is insufficient evidence of a specific structure, it is possible that this alignment represents a basement seismic source zone or fault analogous to the proposed structure for the Giles County earthquake (Bollinger, 1981).

The conservatism of the SSE, assuming a local source

It might be asserted that the consequences of the existence of the hypothetical local source is already anticipated by the movement to the CRBRP site of a hypothetical earthquake of epicentral intensity VIII. The applicant acknowledges (CRBRP PSAR, amendment 71, page 2F-5) that Bollinger suggests that magnitudes up to $M_s = 7.0$ are possible on the hypothetical Giles County structure. The hypothetical structure considered for the vicinity of Knoxville is significantly longer than that proposed by Bollinger for the Giles County structure. Accordingly, there is some possibility of an earthquake on this hypothetical structure having an epicentral intensity greater than VIII. Because this structure is not proven, under Appendix A it would be inappropriate to bring an intensity greater than VIII to the site on the basis of this structure. However an assessment of the significance of such a structure is addressable through a probabilistic ground motion analysis.

As before, assessment of the conservative nature of the SSE depends upon a calculation of the exceedance probability of .25 g, given this hypothetical structure. Accordingly we assumed a line source, 140 km long, on which earthquakes were modelled as ruptures, with the rupture lengths depending on magnitude. (The 140 km length for the line source was chosen because it is the average of the different lengths obtained by removing zero, one, or two earthquake events from either end of the alignment). The maximum magnitude assumed was M_s = 7.0 and the b-value was 0.9. This source was assumed to lie at a radial distance of 15 km from the CRBRP. A major uncertainty in modelling this source is the determination of a suitable annual rate of seismicity attributable to this source. More precisely, the rate may be represented by the annual rate of earthquakes of magnitude greater than 4 (epicentral intensity equal to or greater than V) in the source. Two estimates for this rate were made. For the first estimate, one-quarter of the seismicity of the Appalachian seismic source zone (zone 100 from the national model of Algermissen and others, 1982) was considered attributable to the hypothetical line source. This fraction was an approximation resulting from inspection of seismicity maps, considering the contiguity of the seismicity to the hypothetical structure. For the second estimate, a list appearing in Bollinger and others (1976) for historical earthquakes occurring in the vicinity of the Maryville, Tennessee, earthquake of 1973 was used. All of these events were attributed to the hypothetical structure, and those with magnitude greater than 4, or intensity V or greater, were counted. The annual rates derived from the two procedures agreed within 15 percent of one another. The use of their average in the model yielded an annual exceedance probability for .25 g of 17×10^{-4} for no attenuation variability or 21×10^{-4} for a standard deviation of attenuation variability equal to 0.5. These results do not change significantly if the maximum magnitude is reduced to M_s = 6.4. Exceedance probabilities of these sizes may be legitimate cause for concern. However, if the maximum magnitude on this source is 5.8, the exceedance probability is 0 for no attenuation variability and 7×10^{-4} with attenuation variability.

In a more formal and complete probabilistic assessment of the exceedance probability of the CRBRP SSE, the following items would be found to be most important and would have to be treated probabilistically: (1) the seismic rate assigned to the fault, (2) the likelihood of the existence of this fault, and (3) the distance of the fault to the plant. The exceedance probability is directly proportional to factors (1) and (2) above and, over a limited distance range, inversely proportional to some power of (3). (If the distance is taken to be 20 km instead of 15 km, the exceedance probability decreases by about a factor of 1.6). It is, of course, factor (2) which is most in dispute. For the remainder of the review we address those facts relating to the credibility of the structure.

Evidence for and against the hypothetical structure

The seismological evidence which best addresses the hypothetical structure, other than the relocated epicenters themselves, is that information generated by investigations into the Maryville, Tennessee, earthquake of 1973. Maryville is on the apparent alignment of the relocated epicenters, and it is reasonable to expect that this earthquake should give evidence of a structure associated with the alignment, if indeed the alignment exists on a real structure. Most of the information about this earthquake appears in Bollinger and others (1976). The authors, however, believe that their evidence is not definitive enough to support any particular interpretation.

The map of epicenters of aftershocks of the Maryville earthquake shows a NNE trend. The main shock P-wave first motion focal-mechanism solution shows a NE-striking nodal plane, as does Herrmann's (1979) combined P-wave and surface-wave solution. However, the former solution is consistent with normal faulting down to the southeast, and the latter solution is consistent with reverse faulting on a northwest dipping plane. Both solutions have strike-slip motion component of the same sense. However, Bollinger and others (1976), on the basis of in situ stress measurements and well-hole data, prefer an alternative focal mechanism solution for the same P-wave arrivals. Their alternate main shock solution yields reverse faulting on a NW-trending fault plane. The composite mechanisms for the aftershocks give solutions inconsistent with the main shock mechanisms proposed by Bollinger and others (1976).

The major geophysical feature in the vicinity of the alignment is the "New York-Alabama lineament", which runs NE through Knoxville. This lineament has been interpreted (King and Zietz, 1978) as a fault juxtaposing different basement rock types along which strike-slip movement may have taken place. King and Zietz authors reject the interpretation of normal faulting along this line.

The alignment of epicenters does not coincide with the "New York-Alabama lineament" but rather has a more northerly strike. If the epicenters of the alignment are plotted on the Bouguer anomaly map of figure 2 of Keller and others (1982), the epicenters are found to lie not on the regions of strong gravity gradient, but rather on the tops or flanks of small gravity highs of length 40 to 60 km. The alignment, if it is as much as 180 km long must span three of these local highs. This may argue against a single structure and may limit the potential maximum magnitude.

Summary

The selection by the applicant of the Giles County earthquake in the Southern Valley and Ridge Province as the controlling earthquake at the site is reasonable. We also concur with the assessments of the maximum intensity and SSE, and the anchoring of a Regulatory Guide 1.60 response spectrum to this 0.25 g SSE. Furthermore, the CRBRP SSE has a conservative exceedance probability if one can confidently adopt a diffuse seismicity model to an Appalachian province. However there is evidence of a more concentrated local source in the vicinity of the

CRBRP. This source appears to have sufficient linear extent to generate large magnitude events. Furthermore the seismicity in the vicinity of this source, if attributable to a fault, is sufficient to imply that the CRBRP SSE has an exceedance probability notably higher than 1×10^{-4}. At the present time, the data are insufficient to establish the situation one way or the other. Accordingly, we believe that although the CRBRP SSE is reasonable on the basis of present data, a definitive seismological investigation would be required to address the problem of a possible concentrated seismic source in eastern Tennessee. This probably would require a local network, velocity models, and source mechanism determinations.

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