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CONTEXTUAL INDEX

FOR

FFTF PSAR

SUPPLEMENTARY DOCUMENTATION^{AT}

MASTER

AUGUST 1971

HANFORD ENGINEERING DEVELOPMENT LABORATORY

UNITED STATES ATOMIC ENERGY COMMISSION

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HANFORD ENGINEERING DEVELOPMENT LABORATORY

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United States Atomic Energy Commission Under Contract No. AT(45-1)-2170

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- INTRODUCTION -

The Preliminary Safety Analysis Report (PSAR) for the Fast Flux Test Facility presents a description of the reactor plant and an evaluation of the plant relative to safety features and considerations which support the conclusion that the facility may be safely built and operated. Since submittal of the PSAR in September 1970, substantially more work has been completed and documented in the form of supplemental information. In order to provide a convenient orientation of the supplemental documentation to the initial submittal, this contextual index has been prepared to identify each supplemental document, to provide a brief description of each document's content, and to identify the sections of the initial submittal to which the supplemental document primarily applies.

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SUPPLEMENTS APPLICABLE TO SECTION 2, SITE

1. "SITE INVESTIGATION REPORT FOR THE FAST FLUX TEST FACILITY, RICHLAND, WASHINGTON FOR THE UNITED STATES ATOMIC ENERGY COMMISSION," Bechtel Corporation, December 1970

The purpose of this study is to investigate the FFTF site area, determine the surface and subsurface conditions and determine the most advantageous facility location.

The investigation consisted of several parts and included the following:

- a. Review of literature covering existing facilities with respect to geology, soil conditions, and hydrology in the Hanford Reservation.
- b. Drilling and sampling of exploratory borings.
- c. Excavating test pits for conducting field soil tests.
- d. Laboratory testing of samples that were obtained from the drill holes and test pits.
- e. Seismic survey.
- f. Determination of soil dynamic properties and liquefaction potential.
- g. Preliminary water-well testing utilizing exploratory drill holes.
- h. Aerial survey and topographic mapping.

The investigation indicates that from the point of view of geology foundation engineering and groundwater level, there is no preferred facility location within the general area considered. The final FFTF location is therefore based on topographical features of the terrain, economy of earthwork for site preparation, railroad and road access facilities.

2. "REVIEW OF TORNADO CONSIDERATIONS FOR FFTF," HEDL-TME 71-35, D. E. Simpson, January 1971

In response to Regulatory Review questions, and reconsideration of AEC licensing practices regarding tornadoes in the western United States, additional evaluation of tornado considerations applicable to the FFTF site were carried out. Information made available by the Jersey Nuclear Corporation was included. The probability of a tornado occurrence at the site is extremely low. In addition,

both statistical data and the meteorological and orographic characteristics of the region show that tornadoes which do occur are of a relatively low intensity. A conservative assessment of tornado characteristics consistent with the region leads to a maximum design basis of 150 mph rotational wind speed plus a 25 mph translational wind speed plus associated pressure transients and missiles.

3. "A DISCUSSION OF SAND "DIKES" IN THE FFTF EXCAVATION," Bechtel Incorporated, HEDL-TME 71-101, June 7, 1971

The purpose of this report is to discuss the sedimentary structure and structural features observed in the containment bowl excavation, with special emphasis on the dike-like features present.

Initial inspection of the glaciofluvial deposits exposed in the excavation indicates they are essentially horizontal, relatively clean sands, predominantly fine to medium grained, with little to no cementation. The most unique structural features seen in the excavation are near-vertical and apparently continuous clastic dikes. These dike features have been described by others, elsewhere, and are apparently common to the Touchet formation, and its coarser-grained facies, as exposed at the site.

4. "SEISMIC AND GEOLOGIC SITING EVALUATION FAST FLUX TEST FACILITY NEAR RICHLAND, WASHINGTON," John A. Blume & Associates, Engineers, August 1970

This report presents the results of a comprehensive seismic and geologic siting evaluation for the Fast Flux Test Facility site near Richland, Washington. Data and analysis procedures used in establishing the location and magnitude for the Design Basis Earthquake (DBE) are presented.

5. "GEOLOGIC INVESTIGATION OF FAULTING IN THE HANFORD REGION, WASHINGTON," James W. Bingham, Clark J. Londquist, et al, with a section on the "OCCURRENCE OF MICROEARTHQUAKES," A. M. Pitt, 1970

This report consists of investigations of three principal areas at and near the Hanford Reservation.

1. A study of stratigraphic and structural details at places along the Saddle Mountains fault and Smyrna Bench north of the Hanford Reservation.
2. A study of the stratigraphy and geologic structure of the Gable Mountain and Gable Butte anticlinal folds on the Hanford Reservation.
3. A reconnaissance study of the Horse Heaven Hills and the Rattlesnake Hills-Wallula lineament south and southeast of the Hanford Reservation.

These areas were selected for study to resolve differing interpretations of previous investigators concerning the existence of faults and the recency or antiquity of tectonic movements on some of these geologic structures. Although the investigations confirmed the presence of faults in each area, most of the faults are relatively old -- the latest surface displacements on all but one apparently occurred more than 12,000 years ago and probably more than 100,000 years ago.

As part of this project the U.S. Geological Survey's National Center for Earthquake Research established a network of seven seismographs in the Hanford area. The network, interpretation of the data, and locations of the epicenters of located seismic events are discussed.

6. "A SUMMARY REPORT SEISMIC EVALUATION AND DEVELOPMENT OF GROUND ACCELERATION AND RESPONSE SPECTRA FOR FFTF SITE," John A. Blume & Associates, JABE-WADCO-03, February 1971

This report presents a summary of the August 1970 Blume geological report. It also presents additional information concerning the development of the maximum ground acceleration and associated response spectra for the FFTF site. Studies by the USGS and by Blume geologists indicate that the Rattlesnake Wallula fault is the largest potential earthquake generator near the site. Site work conducted by others determined the geo-technical characteristics and physical properties of the materials upon which the facility will be built.

The Rattlesnake Wallula fault is considered to be a potential generator of earthquakes based upon data presented in a USGS report. The fault is shown on a tectonic map of the Hanford and adjacent areas. Field investigations by Blume geologists of the site, a line of hills near Horn Rapids and the area in the vicinity of Badger Coulee, and a detailed literature review indicated that there are no significant faults closer to the site. Examination of historic records and calculation of peak ground motion possibly felt at the site from earthquakes which have occurred within 200 miles of the site showed that an earthquake on the Rattlesnake Wallula fault would cause maximum ground motion at the site.

7. "SUPPLEMENTARY GEOLOGIC INVESTIGATIONS FOR SEISMIC EVALUATION OF THE FFTF SITE NEAR RICHLAND, WASHINGTON," John A. Blume & Associates, Engineers, February 1971, JABE-WADCO-04

This document contains results of two recent geologic investigations and updates a report on an earlier investigation, all of which were located in or near the Hanford Atomic Energy Commission reservation

in Washington. The investigations were required for seismic evaluation of the Fast Flux Test Facility (FFTF) site. The primary purpose was to identify and delineate faults which are near enough to the FFTF site to be considered in its seismic evaluation.

8. "DETERMINATION OF SPRING CONSTANTS FOR SEISMIC ANALYSIS OF FAST FLUX TEST FACILITY," H. Bolton Seed, January 1971

This report describes the procedures used to determine appropriate values of foundation spring constants for FFTF. Spring constants for vertical translation, horizontal translation, rotation, and coupling between rotation and translation are determined for both the DBE and the OBE for both Type I & Type II structures.

9. "DETERMINATION OF DAMPING FACTORS FOR SEISMIC ANALYSIS OF FAST FLUX TEST FACILITY," H. Bolton Seed, January 1971

This report describes the procedures used to determine appropriate values of soil damping factors for the FFTF. Damping due to soil deformation and radiation of waves away from the structure are presented for both the OBE and the DBE.

10. "MICROEARTHQUAKE ACTIVITY NEAR WOODED ISLAND AND ITS EFFECT UPON SEISMIC EVALUATION OF THE FFTF SITE," John A. Blume & Associates, Engineers, April 1971, JABE-WADCO-05

This report is an evaluation of the impact of microseismic activity in the Hanford area upon the seismic and geologic siting evaluation for the FFTF completed in August 1970. It is concluded that the Wooded Island cluster of microseismic events is not indicative of any geologic processes indicating probable strong seismic activity or the presence of a concealed fault which would require modifications in the seismic and geologic siting evaluation of the FFTF site.

SUPPLEMENTS APPLICABLE TO SECTION 3, REACTOR

1. "MATERIAL CONSIDERATIONS IN SUPPORT OF THE FFTF PRELIMINARY SAFETY ANALYSIS REPORT - RM 9, VOLUMES I AND II," T. T. Claudson, April 9, 1971, HEDL-TME 71-53

This report contains materials properties data obtained before, during, and after neutron irradiation on selected alloys. Long-term and short-time test results on base metal and weldments are included. Materials properties for both annealed and cold worked austenitic stainless steels are provided. Swelling data and microstructural effects are summarized and compared with the interim equations currently used in design. Recent data in this area suggest that the current equations over-predict the amount of swelling which will occur under FTR conditions thereby providing an added margin of safety. Similarly, the creep data on stainless steel under the combined effects of neutron irradiation, thermal history, and stress are presented and creep behavior is described by interim design equations. Special topics such as high strain rate effects, fatigue crack growth rates, friction and wear, creep-fatigue interactions, and notch effects are discussed briefly and current data reviewed with respect to FFTF applications. Data analyses which relate to specific FFTF materials considerations are included as appendices.

2. "THE IMPACT OF IRRADIATION INDUCED SWELLING, CREEP AND DUCTILITY LOSS ON THE DESIGN OF FFTF CORE COMPONENTS," W. E. Pennell, FRA-425, March 1971

This report provides a summary of the treatment of irradiation induced swelling, creep and ductility loss in the design of the FFTF core. Details of the design equations in use together with their data base are given. Analyses performed in support of the design of the fuel assembly, the core restraint system and the control rod system are presented in outline, and the results obtained are given.

SUPPLEMENTS APPLICABLE TO SECTION 4, REACTOR COOLANT SYSTEM

1. "FFTF OUTLET REGION FEATURE MODEL 300 SERIES AIR ENTRAINMENT TESTS REPORT," J. Spalek, D. L. Ballard, and L. R. Sweetin, February 1971, HEDL-TME 71-25

A series of water model tests have been performed as a part of the FFTF development program. This document reports the results of the 300 Series tests, performed in a dynamic water model of the FFTF, for the evaluation of potential gas entrainment problems. Tests were conducted to show the effectiveness of a vortex suppression plate positioned just below the liquid pool surface. Both visual and ultrasonic data were obtained and recorded for conditions with and without the suppression plate being in position.

Gas entrainment in the liquid pool is reduced to essentially zero in the current reference design configuration.

2. "GAS BUBBLE DISPERSION TEST REACTOR INLET MODEL," J. Muraoka, J. W. Hagan, and D. L. Ballard, HEDL-TME 71-69, May 1971

Gas bubble dispersion tests were performed using a hydraulic model of the FFTF inlet reactor plenum. These tests were performed to evaluate effective breakup and dispersion of noncondensable gases entering that region. The tests and results described in this report indicate that dispersion will effectively preclude any significant reactivity effects due to gas entering the reactor inlet plenum region.

3. "EVALUATION OF POSSIBILITIES FOR GAS ENTRAINMENT INTO THE FFTF CORE," FPS-930, WADCO Corporation and Westinghouse Advanced Reactors Division, February 1971

This report investigates the potential effects of sodium voiding on the operation of the Fast Flux Test Facility. A series of water model tests were performed to determine the behavior of large coherent bubbles swept into the vessel inlet plenum and to evaluate means to avoid cover gas entrainment in the liquid pool in the outlet plenum. The tests and the results are described. Analyses are presented which determine the extent of gas solution in the sodium and which assess the capability of the core design to accommodate voids. It was concluded that the configuration of the hydraulic system promoted bubble breakup and that core is sufficiently tolerant of large voids.

4. Refer also to item 1 of Supplements to Section 3.

SUPPLEMENTS APPLICABLE TO SECTION 15, SAFETY ANALYSIS

1. "AN ANALYTICAL EVALUATION OF FUEL FAILURE PROPAGATION FOR THE FAST FLUX TEST FACILITY," R. D. Coffield and P. L. Wattelet, WARD-AT(45-1)-2171-11, December 1970

The basic objective of this report is to demonstrate the margin of safety that is designed into the Fast Flux Test Facility with respect to fuel failure propagation. The criterion selected for this demonstration is the capability of the design to prevent possible fuel element failure in a single affected driver fuel subassembly from causing fuel element failure in neighboring driver fuel subassemblies. To meet this objective a realistic appraisal was made of the major fuel failure and failure propagation phenomena of concern in liquid metal cooled fast reactors.

In the present analysis the various possible events leading to accident progression have been categorized into the following major propagation phenomena:

1. Fission gas release from failed fuel pins
2. Molten fuel release from failed fuel pins
3. Molten fuel/sodium interaction and resulting pressure pulses.

The response of the major barriers to failure propagation was analyzed by formulating appropriate analytical models and utilizing the latest available design parameters, materials data and experimental safety information. The severity of fuel failure phenomena required to violate the successive barriers is demonstrated with the result that even in the event of this assumed level of severity, the adjacent subassembly duct remains intact.

2. "SAFETY ANALYSES FOR THE IMPROVED CORE CONDITIONS," FFTF PSAR Supplementary Section 15.A, March 1971

In Section 15 of the PSAR, results were presented for studies of "realistic" accidents based on initial core operating conditions. Analyses of such accidents for "improved" core conditions have now been completed and are presented in this supplement. The analyses were carried out using the same models as those employed in the original Section 15 studies. Significant parameters used in the analysis are listed in Table 15.A-1. It should be noted that "improved" core conditions relate to operating temperatures above those previously specified for initial operation. The mechanical and physical compositions of the improved core are the same as for the initial core.

Note: This information is found in Section 15 of Vol. II, Preliminary Safety Analysis Report for FFTF.

3. Refer also to item 1 of Supplements to Section 3.
4. Refer also to item 3 of Supplements to Section 4.

SUPPLEMENTS APPLICABLE TO APPENDIX A, FFTF QUALITY ASSURANCE
PROGRAM DESCRIPTION

1. "FAST FLUX TEST FACILITY QUALITY ASSURANCE MANUAL," WADCO Corporation, August 1970, WHAN-M-12

This manual provides detailed description of the Quality Assurance program procedures to be applied, describes how they will be used, identifies responsibilities for their application, and indicates the project phases during which they will be applied.

Included are:

1. The FFTF Fuels Quality Assurance Manual, including the quality assurance procedures which are in use for the FFTF fuel.
 2. Indices of the program plans and procedures for quality assurance utilized by the other principle contractors and national laboratories.
 3. A discussion of the planning and administration of development testing in support of the FFTF design.
2. QUALITY ASSURANCE PROGRAM PLAN FOR THE FFTF, WHAN-M-14

This document sets forth the elements of the quality assurance program and methods of quality control and verification which will be used during all project phases, from initial design through operation of the Fast Flux Test Facility.

SUPPLEMENTS APPLICABLE TO APPENDIX D -
SPECIFIC FFTF ENGINEERING DESIGN CRITERION

1. "SEISMIC DESIGN CRITERIA FOR THE FAST FLUX TEST FACILITY SITE, RICHLAND, WASHINGTON," John A. Blume & Associates, Engineers, November 1971, JABE-WADCO-02

This report presents the seismic design criteria for the Fast Flux Test Facility site near Richland, Washington. These criteria establish requirements to ensure that structures, systems, and equipment are designed in accordance with design standards commensurate with the importance of the safety function which they perform. They apply to Categories 1, 11 and 111 structures, piping and equipment.

2. Refer also to items 4 through 10 of Supplements to Sections 2.

SUPPLEMENTS APPLICABLE TO APPENDIX G, HYPOTHETICAL ANALYSES
FOR TESTS OF CONTAINMENT

1. "SUMMARY OF COMPUTER CODES USED FOR FFTF CONTAINMENT AND SITING STUDIES," compiled by Staff of Safety and Systems Analysis Engineering, HEDL-TME 71-30

Because of the complex nature of the analytical models used for analysis of postulated events it was necessary to use computer codes. The scope of these codes covers the initiation of a hypothetical accident, its immediate consequences in terms of energy, pressure and temperature effects and long term consequences involving the potential for leakage of radioisotopes from containment to the environs and the potential dose at the site boundary.

The codes used in each area of analysis are as follows:

	<u>Codes Used</u>	<u>Originator</u>
• Initiating Mechanisms, Core Disruption and Disassembly	FREADM-1	GE
	SAS1A	ANL
	MELT-II	HEDL
	VENUS	ANL
• Work Energy	SOCOOL-II	HEDL
• Mechanical Effects on Core Structures	ASPRIN	HEDL
	REXCO-H	ANL
• Pressures and Temperatures from Sodium Spills	SOFIRE-II	AI
	CACECO	HEDL
• Radiological Effects	ISOSHL	HEDL
	INDOSE II	HEDL
	EXDOSE	HEDL
	RIBD	HEDL
	COMRADEX	AI
	HAA-3A	AI

These codes form a relatively complete series of analytical models that may be used to evaluate consequences at any stage of postulated accidents of a severe type.

A brief description of complexities, uncertainties and experimental verification, if any, is given. Other information of interest includes method of analysis and items related to use of the codes.

2. "PLANT TRANSIENT FOLLOWING A LARGE PIPE RUPTURE IN THE FFTF PRIMARY SYSTEM," J. K. Martin, March 1971, WARD-AT(45-1)-2171-13

The purpose of this study was to analyze the FFTF plant transient following the unlikely event of a large pipe rupture in the primary system piping. The analysis was performed for both full temperature (1100°F core exit) operation and initial temperature (900°F core exit) operation. Representative rupture locations were chosen around the primary loop and the effects of an instantaneous double ended rupture were studied at each location except at the reactor inlet nozzle, where the inlet flow restrictor prevents a double ended rupture. Three rupture locations in the cold leg piping were analyzed, and two in the hot leg piping.

The occurrence of an instantaneous, guillotine rupture of the primary loop piping is not considered to be a realistic postulate. Ruptures of less severity than this may be envisaged, but any event approaching the magnitude of a complete severance must be regarded as an extremely low probability occurrence. However, the principal reason for carrying out analysis on the effects of an instantaneous guillotine rupture is that this postulate represents an upper bound to the range of studies which might be conceived, and not that it should be regarded as a real event. This event has been classified as a hypothetical event and included in APPENDIX G of the Preliminary Safety Analysis Report.

The analyses were done with the digital plant simulation code IANUS. A description of the aspects of IANUS pertinent to this study is given.

3. "ASSESSMENT OF MAGNITUDE AND UNCERTAINTIES OF HYPOTHETICAL ACCIDENTS FOR THE FFTF," D. E. Simpson, A. E. Waltar and A. Padilla, HEDL-TME 71-34, March 27, 1971

The purpose of the study reported in this document was to evaluate the sensitivity of the energy generated to a broad range of parameters and phenomenological models. Essential conclusions from the results of analyses carried out include the following:

The postulation of initiating mechanisms and propagating phenomena leading to an energetic nuclear disassembly accident, as well as analytical methods employed, appear to be substantially conservative in themselves; thus the potential for an accident approaching even the nominal energy release of low hundreds of MW-seconds is extremely low and quite conservative.

Preliminary data has appeared indicative of major conservatism in the calculation of theoretically available work produced by the thermal interaction of heated fuel and sodium in the reactor. During late 1969 and early 1970, experiments were performed by ANL in the TREAT facility which began to shed light empirically on the relatively low efficiency of this process in producing work.

Several more experiments have been performed which continue to indicate a work energy production substantially less than that which one would calculate by the methods used in this report. The results of experiments performed by ANL in the TREAT facility are summarized.

4. "MECHANICAL CONSEQUENCES OF HYPOTHETICAL CORE DISRUPTIVE ACCIDENTS," D. D. Stepnewski, G. L. Fox, et al, April 19, 1971, HEDL-TME 71-50

The FFTF reactor plant is being designed to contain the consequences of hypothetical accidents in a manner which permits the disabled core to be effectively cooled by the primary cooling system. This report provides estimates of the loads imposed on the primary boundaries by a hypothetical core disruptive accident characterized by a theoretical work energy of 150 MW-sec. Calculations are also presented for a higher energy level of 350 MW-sec.

This report provides a description of the potential consequences of hypothetical core disruptive accidents on the primary coolant boundary. Further, it treats the effects of the postulated accidents on containment structures in the event of energy releases beyond the primary coolant boundary. The objective of this treatment of the accident mechanisms described herein is to demonstrate the capability of FFTF design features, both inherent in the primary boundary and in containment structures to effectively isolate limiting severe accidents from the environs of the facility. The basic input upon which this mechanical assessment is made is taken from hypothetical accident energy release information developed for the FFTF reactor. The output from this report in the form of containment vessel pressure and leakage transients has been used in radiological evaluations for the site.

Based on analyses of the reactor vessel, vessel head and primary loop components, it is concluded that although plastic deformation may occur, the primary boundary is designed to retain a capability to remove core decay energy following a hypothetical core disassembly accident of 150 MW-sec theoretical work.

5. "POST-ACCIDENT HEAT REMOVAL IN LMFBRs IN-VESSEL CONSIDERATIONS," Argonne National Laboratory, ANL/RAS 71-12, March 1971

This report describes the anticipated behavior of fast reactor fuel in the reactor vessel following a postulated core disassembly accident. The scope of this work includes a preliminary identification and characterization of the main features of fuel behavior following a core disassembly accident and a discussion of in-vessel processes which provide intrinsic containment for Liquid Metal Fast Breeder Reactors. Results of analytical and experimental work in key areas are described. Key areas of uncertainty requiring further work are identified. Specific numerical results are based on the Fast Test Reactor (FTR) design and the hypothetical core disassembly accidents described in the Preliminary Safety Analysis Report for that reactor.

6. "COMPARISON OF A TWO-DIMENSIONAL HYDRODYNAMICS CODE REXCO TO EXCURSION EXPERIMENTS FOR FAST REACTOR CONTAINMENT," ANL/RAS 71-14, J. E. Ash & R. T. Julke, ANL, January 1971

The Reactor Excursion Containment Code System (REXCO) is a tool that has been developed at Argonne National Laboratory to predict the effects, due to excursions originating at the core, on the components and on containment of Liquid Metal Fast Breeder Reactors. The codes include hydrodynamic and inelastic material behavior depending on the magnitude of the excursion treated. The physical phenomena gaged by the codes involve the shock wave, the sodium momentum, and the early stages of the gas bubble. Current work attempts to extend treatment to the later, semistatic phenomena of the gas bubble.

In this report comparisons of these codes are made with experimental excursion data. The results of these comparisons confirm the capability of the REXCO code system to predict the response of the primary system components and of containment structures to the generated pressure waves. The primary system phenomena gaged extend, but are not limited to, to reactor vessel strain and rupture, plug jump, and sodium escape past the cover seal into the secondary containment. The validating comparisons were made with published experimental data. The subject experiments were conducted at A.W.R.E., Foulness, England; at Cadarache, France, by the CEA-Euratom and the UKAEA; and in the United States by the Naval Ordnance Laboratory.

7. "SODIUM FIRES ANALYSIS FOR THE FAST FLUX TEST FACILITY," J. P. Hale and P. R. Shire, HEDL-TME 71-31, February 12, 1971

This analytical and survey report presents summaries of sodium fire cases of practical value for sodium cooled reactor plant safety.

Sodium pool fire calculations are presented in detail. Surveys of pool and spray fire calculation methods, and experimental progress, are summarized in the areas of applicability to plant design parameters.

For the pool fire calculations, various conditions are investigated: sodium spill in two-cell configuration with air atmosphere, sodium spill in air from fuel handling, and sodium spill in closed cells with low oxygen atmosphere. Spray fires in low oxygen atmospheres are also treated.

8. "RADIOLOGICAL EVALUATION OF A HYPOTHETICAL CORE DISRUPTIVE ACCIDENT," C. L. Fies, HEDL-TME 71-33, February 1971

This document evaluates the effects of radioactivity release due to the Hypothetical Core Disruptive Accident. A number of conservative assumptions are postulated in order to test the capability of the containment system to mitigate the consequences of such an accident. The following topics are discussed:

1. The radionuclide inventory, including plutonium, fission products, activation products, and sodium present in the reactor core.
 2. The amount of the radioactive inventory released from the core to the primary sodium and from the reactor vessel into the reactor vessel cavity as a result of the hypothetical core disruptive accident.
 3. The pressures and leak rates associated with the reactor cavity and the containment vessel.
 4. The aerosol behavior within the reactor cavity and within the containment vessel, and
 5. The potential exposure to an individual as a function of distance from the containment building as a result of inhalation of material leaking from the building, the dose resulting from the passage of a radioactive cloud, and the direct dose obtained from fission products suspended within the containment building.
9. "AEROSOL MODELING OF HYPOTHETICAL LMFBR ACCIDENTS," R. L. Koontz, et al, Atomics International, AI-AEC-12977, August 31, 1970

The report represents a description of an aerosol model (HAA-3), consistent with existing experimental data, developed to predict aerosol behavior. Comparison of the model theory with experiments are presented for four independent time varying parameters. A

detailed justification of the model and the application of various model assumptions and parameters to accident conditions is described. A description is presented of additional aerosol behavior characteristics, suggested by experimental data or analytical considerations, which are not completely resolved. The potential effects of these characteristics are evaluated by parameter sensitivity evaluation with the model.

10. "AN INTEGRATED MODEL FOR ANALYZING DISRUPTIVE ACCIDENTS IN FAST REACTORS," W. T. Sha and A. E. Waltar, WHAN-SA-26, February 1971

A two-dimensional (R-Z) integral model for characterizing fast reactor excursions from accident inception through core disassembly is presented. For pre-disassembly calculations, a Eulerian geometric model is used, and multichannel heat transfer computations are performed. Reactivity feedback due to Doppler broadening, coolant density change and voiding, and fuel movement are taken into account. A Lagrangian coordinate system is used in the disassembly phase, wherein the neutronics balance consists of Doppler broadening and material motion. A unique feature of the model is the ability to accommodate a pointwise Energy-Density-Dependent Equation of State according to the local sodium inventory that actually exists at the time of disassembly. By providing a consistent basis for establishing the effective reactivity ramp rate, Doppler coefficient, appropriate Equation of State, and temperature distribution at the start of core disassembly, much of the arbitrariness normally associated with large accident analyses can be removed. For most accident analyses, this model predicts a significantly lower energy yield during a super-prompt critical nuclear excursion than would be computed by using the conventional modified Bethe-Tait analysis.

11. Refer also to item 1 of Supplements to Section 3.

Revision 1

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