See folder for
Att. to Attached List
AM. OLSON 5-31-84 101.2

Summary Meeting Notes

DOE/NRC Meeting on the BWIP Barrier Materials Test Plan

Gaithersburg, Maryland
May 8-9, 1984

Agenda: See Attachment 1

Attendees: See Attachment 2

Developments:

BWIP presented an overview of the Barrier Materials Test Plan (BMTP) (see Attachment 3). In the absence of established specific performance requirements and reliability goals for the waste package, the barrier materials test plan was prepared based on applicable regulatory criteria, OCRWM criteria, BWIP functional design criteria, BWIP conceptual designs, sound scientific/engineering judgement/and peer review. It is the intent of BWIP to add a discussion on the relationship of design and performance assessment activities to the materials testing efforts to the test plan in its next revision. BWIP is developing an integrated performance allocation plan to address the relation of data types (including the waste package data) to performance goals. This plan is not available at this time.

NRC presented the elements against which the review was conducted (attachment 4). As explained in the BMTP, the current draft presents test plans for materials and waste package environment but does not cover performance assessment or design. NRC noted that these omissions sharply limited the scope of the review; without an understanding of the performance objectives assigned by DOE to components of the engineered system a review of the test plans is incomplete. NRC has stressed, in interactions with DOE, the importance of early attention to performance objectives for system components in order to direct test plans and site characterization toward licensing information needs. To illustrate, NRC distributed materials developed at a workshop on August 4-5, 1983 (see attachments 5 and 6). NRC comments on the DOE Mission Plan also cover the importance of setting performance objectives for system components (see attachment 7).

NRC preliminary comments on the BMTP comprise attachments 8 and 9. The handwritten changes represent clarifications developed during the course of the technical meeting.

BWIP responses to NRC comments comprise attachment 10.

The presentations served to clarify the meaning of "test plan" as used by BWIP: a BWIP test plan, like the BMTP, deals mainly with the overall logic, rationale and general schedule; details such as the matrix of test conditions, selection of specific test methods, test procedures and detailed schedules are contained in other documents such as the annual test instructions and QA documents such as MA-4. NRC stated that to carry out its review and comment function both the general material and the specifics are needed. This matter will be covered further in the NRC follow-up comments.

The principles underlying NRC's draft "Fault Trees Depicting Failure of High-Level Radioactive Waste Packages" and "Event Trees Depicting Release of Radionuclides from High-Level Radioactive Waste Packages" were discussed. The documents had previously been given to BWIP for review. They form attachments 11 and 12.

NRC and DOE participants discussed the technical state-of-the-art on system characterization and testing. NRC representatives offered suggestions on program alternatives to broaden the data collection effort. It is the consensus of participants that the meeting provided clarification on technical issues and approaches involved in the waste package system testing.

Follow-up Items:

NRC will send follow-up comments on BMTP to DOE in mid-June.

James E. Mecca 5/10/84
James E. Mecca, DOE

Robert J. Wright 5/10/84
Robert J. Wright, NRC

Attachments:

1. Meeting agenda
2. Listing of attendees
3. BWIP overview of BMTP
4. Considerations in NRC review of BMTP
5. Extract from summary meeting notes of the DOE/NRC technical management meeting August 4-5, 1983
6. NRC statement on reliability - August 4, 1983
7. Extract from NRC comments on DOE Mission Plan - February 1983
8. Topics covered in NRC preliminary comments
9. NRC preliminary comments on BMTP
10. BWIP responses to NRC preliminary comments
11. "Fault Trees Depicting Failure of High-Level Radioactive Waste Packages" (Draft) - December 1983
12. "Event Trees Depicting Release of Radionuclides from High-Level Radioactive Waste Packages" (Draft) - January 1984

PROPOSED AGENDA FOR NRC/BWIP MEETING TO DISCUSS THE BARRIER MATERIALS TEST PLAN

MAY 8-10, 1984

CLIMAT DE FRANCE HOTEL

GAITHERSBURG, MD.

Tuesday, May 8, 1984

✓8:00 am	Introductory Remarks, NRC	NRC
✓8:10 am	Introductory Remarks, DOE	DOE
✓8:20 am	Background and Basis for the BWIP Test Plan (BMTP)	BWIP
8:40 am	Preliminary NRC Review Comments on the BMTP - DOE/BWIP Responses	NRC
12:00 noon	Lunch	
1:00 pm	Reconvene the NRC Comments and BWIP/DOE Responses	NRC
5:00 pm	Adjourn	

Wednesday, May 9, 1984

8:00 am-noon	Reconvene NRC - DOE/BWIP Dialogue on the BMTP	NRC
12:00 noon	Lunch	
1:00 pm-5:00 pm	Conclude Dialogue**	NRC

Thursday, May 10, 1984

8:00 am	Wrap-up and Adjournment	NRC/DOE/BWIP

* Breaks at approximatly 10:00 am and 3:00 pm.
** Time permitting, the open items remaining from the Geochemistry Workshop will be discussed.

ATTACHMENT 2

LIST OF ATTENDEES

Name	Affiliation	Telephone
Dave Dahlem	DOE-Richland	509 376 3022
Jim Mecca	" "	" " 5038
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TRIS McCALL	" "	509-373-4012
Bob Fish	" "	509-373-1598
Marc Wood	" "	509 373 4013
John H Smithson	DOE HQ	202 252 9433
WARREN REHFELDT	NRC-WMRP	301-427-4535
Claudio Pescatore	BNL	FTS 666-2853
Everett Wick	NRC WMEG	FTS 427-4393
Michael McNeil	NRC RES	FTS 427-4636
Michael Tokar	NRC WMEG	FTS 427-4119
Robert J. Wright	NRC WMRP	FTS 427-4697
Kien C Chang	NRC WMEG	FTS 427-4608
CHARLES H PETERSON	NRC WMEG	FTS 427-4242
M.J. FURMAN	DOE-Richland	509-376-7062
DEAN TOUSLEY	YAKIMAS	202-328-3500
Scott Barney	Rockwell-BWIP	509-373-2634
David Brooks	NRC-WMGT	FTS 427-4603
Maxine Dunkelman	NRC-WMRP	FTS 427-4300
E.R. Wiot	WESTON	301-963-6596
Jay Rhoderick	DOE-HQ	FTS 233-5204
R John Starmer	NRC/WMGT	FTS 427-4541
Don Alexander	DOE-HQ	FTS 233-5596

②

Ed Benz	Weston	301-963-5246
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NAIEM S TANIOUS	NRC/WMEG	(301) 427-4211
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H. C. Claiborne	ORNL	FTS 626-2578
CARL NEWTON	DOE-HQ	FTS 233-485
FRANK DICKSON	T.R.G., DOE/BWIP	(615) 482 6879
John T. Greeves	NRC/WMEG	(301) 427-4612
John W. Bradbury	NRC-WMGT	(301) 427-4591
Peter Soo.	BNL	FTS 666-4094
Mark W. Frei	DOE-HQ	233-3013

BWIPO

OVERVIEW OF

ENGINEERED BARRIERS TEST PLAN

(EBTP)

P.F. SALTER

MAY 8, 1984

NRC/BWIP REVIEW OF EBTP

Rockwell Hanford Operations

ATTACHMENT 3

PURPOSE OF THE MATERIALS TESTING EFFORT:

TO ACQUIRE THE LABORATORY MATERIALS PERFORMANCE DATA NECESSARY TO DEVELOP WASTE PACKAGE DESIGNS AND TO EVALUATE THEIR PERFORMANCE RELATIVE TO THE ESTABLISHED SYSTEM PERFORMANCE REQUIREMENTS/RELIABILITY GOALS

IN THE ABSENCE OF ESTABLISHED SPECIFIC PERFORMANCE REQUIREMENTS AND RELIABILITY GOALS FOR THE WASTE PACKAGE, THE LABORATORY BARRIER MATERIALS TESTING EFFORT HAS BEEN BASED ON APPLICABLE REGULATORY CRITERIA, OCRWM CRITERIA, BWIP FDC, BWIP CONCEPTUAL DESIGNS, SOUND SCIENTIFIC/ENGINEERING JUDGEMENT AND PEER REVIEW.

PURPOSE OF THE BARRIER MATERIALS TEST PLAN

- DEFINE WHAT THE LABORATORY MATERIALS DATA NEEDS ARE AND WHY THEY ARE NEEDED (I.E., RELATE TO PERFORMANCE ASSESSMENT AND DESIGN REQUIREMENTS AS ESTABLISHED BY THE SYSTEM REQUIREMENTS TREE)

- DEFINE HOW THE DATA WILL BE ACQUIRED AND THE BASIS FOR EVALUATING DATA QUALITY (I.E., PRECISION, ACCURACY AND DATA VALIDITY)

- PROVIDE LOGICAL SEQUENCE/SCHEDULE FOR DATA ACQUISITION TO SUPPORT DESIGN, SRR, AND LA

OMISSIONS IN CURRENT TEST PLAN

- DESCRIPTION OF THE DESIGN/ENGINEERING PROCESS FOR DEVELOPMENT OF WASTE PACKAGE DESIGNS
- DESCRIPTION OF THE WASTE PACKAGE DESIGN PERFORMANCE ASSESSMENT EFFORT AND ITS RELATIONSHIP TO THE LABORATORY TESTING EFFORT
- DESCRIPTION OF WASTE PACKAGE PERFORMANCE REQUIREMENTS/RELIABILITY GOALS (REQUIRED TO ESTABLISH HOW MUCH AND HOW GOOD THE DATA MUST BE)
- RELATIONSHIP OF DEFINED DATA NEEDS TO OVERALL BWIP MISSION (SYSTEM REQUIREMENTS TREE)

- TESTING REQUIRED FOR DESIGN/ FABRICATION DEVELOPMENT
- ENGINEERING SCALE AND FIELD/IN-SITU TESTING
- TEST PROCEDURES/TEST INSTRUCTIONS

CONTENTS SUMMARY OF THE BMTP

- BACKGROUND AND OBJECTIVE
- SUMMARY OF DATA NEEDS AND TESTING TO DATE
- TEST REQUIREMENTS
    - PURPOSE
    - DESCRIPTION OF TEST EQUIPMENT
    - SUMMARY OF TEST TECHNIQUE
    - DATA TO BE GENERATED
    - VARIABLE TO BE INVESTIGATED
    - APPLICATION OF DATA
    - SCHEDULE/LOGIC OF DATA ACQUISITION
- QA
- TEST REPORTING

OBJECTIVES

FULLY-RADIOACTIVE WASTE FORM TESTING
(325 BUILDING, HOT CELL FACILITY)

WP DESIGN

- DETERMINE NEED FOR ADDITIONAL BARRIERS
- EVALUATE EFFECT OF RADIATION ON BARRIER STABILITY
- CONFIRM COMPATIBILITY OF BARRIERS
- DETERMINE NEED FOR TAILORING AGENTS TO CONTROL RADIONUCLIDE RELEASE

WP PERFORMANCE ANALYSIS

- MEASURE RADIONUCLIDE RELEASE RATES
- IDENTIFY SOLIDS CONTROLLING RADIONUCLIDE RELEASE
- PROVIDE RADIONUCLIDE SOURCE TERM DATA FOR FAR-FIELD PERFORMANCE MODELING

OBJECTIVE OF W/B/R PROGRAM

- EXPERIMENTALLY DETERMINE STEADY-STATE RADIONCULIDE CONCENTRATION DATA FOR USE AS SOURCE TERM INPUT TO PA MODELS AND DESIGN DEVELOPMENT
- DETERMINE SYNERGISTIC EFFECTS BETWEEN BARRIER MATERIALS UNDER EXPECTED HYDROTHERMAL CONDITIONS
- DEVELOP Eh AND pH SENSORS FOR USE IN HYDROTHERMAL INTERACTION EXPERIMENTS
- DETERMINE OXYGEN BUFFERING CAPACITY OF BASALT FOR WASTE PACKAGE DESIGN
- IDENTIFY ALPHA AND GAMMA RADIOLYSIS EFFECTS
- INVESTIGATE NATURAL ANALOGS FOR WASTE PACKAGE

WP8401-257A

WASTE/BARRIER/ROCK RADIONUCLIDE RELEASE TESTING

QUANTIFY RADIONUCLIDE RELEASES TO THE HOST ROCK

DETERMINE RADIONUCLIDE STEADY STATE CONCENTRATIONS AS A FUNCTION OF:

- TEMPERATURE
- PRESENCE OF BARRIER MATERIALS
- FLOW RATE

IDENTIFY ALTERATION PHASES WHICH CONTROL RADIONUCLIDE CONCENTRATIONS AS A FUNCTION OF:

- TEMPERATURE
- PRESENCE OF BARRIER MATERIALS
- FLOW RATE

ASSESS CAPABILITY TO CONTROL RADIONUCLIDE RELEASE FROM THE WASTE PACKAGE RELATIVE TO CRITERIA

WP8405-15

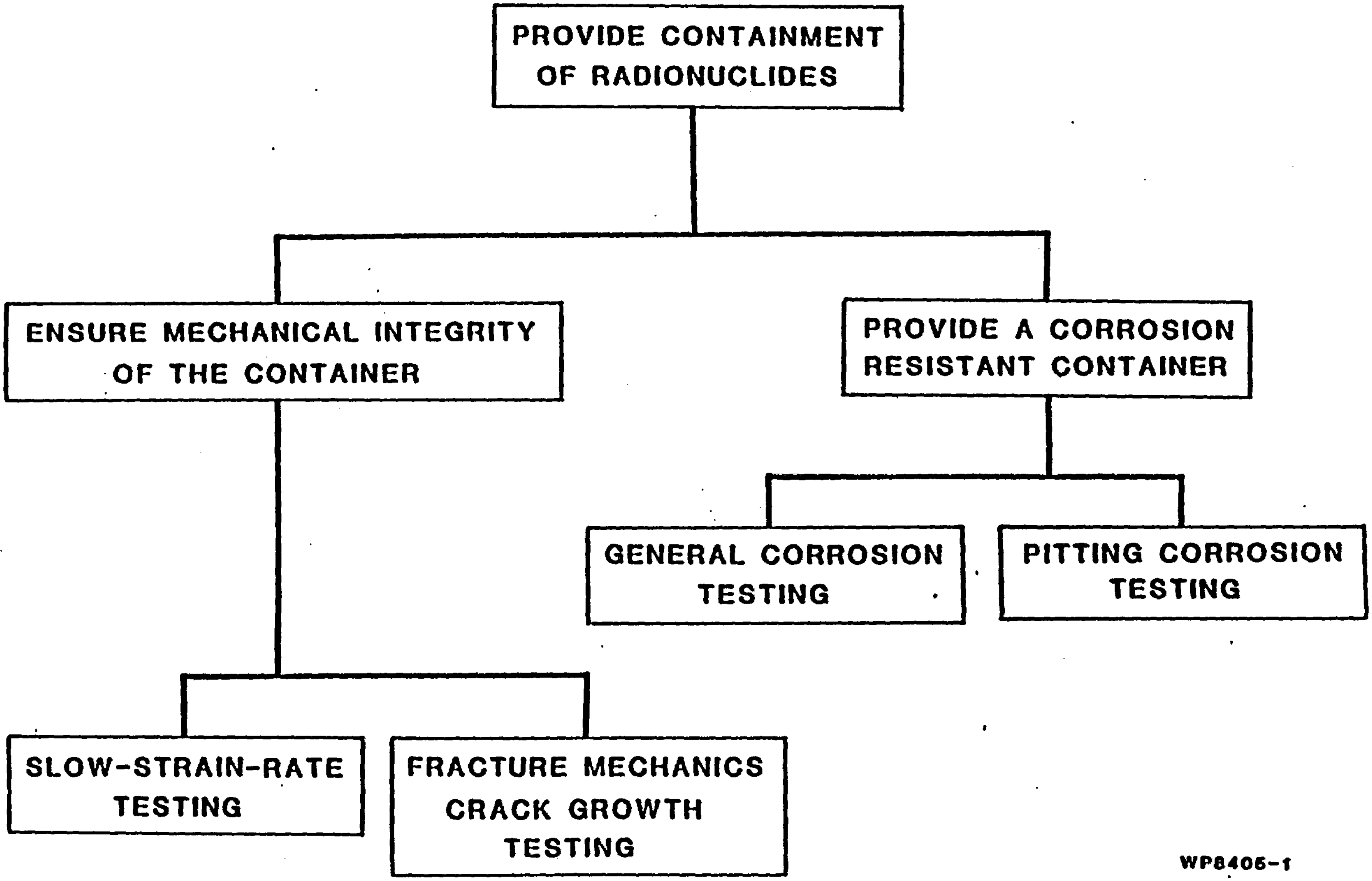
BWIP CONTAINER MATERIAL STUDIES

OBJECTIVE

- IDENTIFY AND CHARACTERIZE A CONTAINER MATERIAL THAT WILL PROVIDE REASONABLE ASSURANCE OF NUCLEAR WASTE CONTAINMENT IN THE ENVIRONMENT OF A DEEP GEOLOGIC REPOSITORY CONSTRUCTED IN BASALT FOR 300 TO 1,000 YEARS.

WP8301-74B

WASTE PACKAGE CONTAINER MATERIALS TESTING



WP8405-1

CONTAINER MATERIALS CRACK GROWTH TESTING

SLOW-STRAIN-RATE TESTS

- ASSESS RELATIVE SUSCEPTIBILITY OF CONTAINER MATERIALS TO ENVIRONMENTALLY ASSISTED CRACKING FOR RANKING AND SCREENING

FRACTURE MECHANICS CRACK GROWTH TESTS

- QUANTITATIVELY ASSESS ENVIRONMENTALLY ASSISTED CRACKING TO ESTABLISH LIMITING STRESS AND FLAW SIZE COMBINATIONS

- MECHANISTIC UNDERSTANDING
- STATISTICAL TREATMENT OF DATA

CRACK PROPAGATION CORRELATIONS

- DESIGN RELIABILITY ASSESSMENT
- PERFORMANCE ASSESSMENT

WP8405-2

CONTAINER MATERIALS CORROSION TESTING

GENERAL CORROSION TESTS

- CHARACTERIZE UNIFORM (INTERGRANULAR IF APPLICABLE) CORROSION BEHAVIOR OF CONTAINER MATERIALS

PITTING CORROSION TESTS

- CHARACTERIZE PITTING (CREVICE) CORROSION BEHAVIOR OF CONTAINER MATERIALS

- MECHANISTIC UNDERSTANDING
- STATISTICAL TREATMENT OF DATA

CORROSION CORRELATIONS

- DESIGN RELIABILITY ASSESSMENT
- PERFORMANCE ASSESSMENT

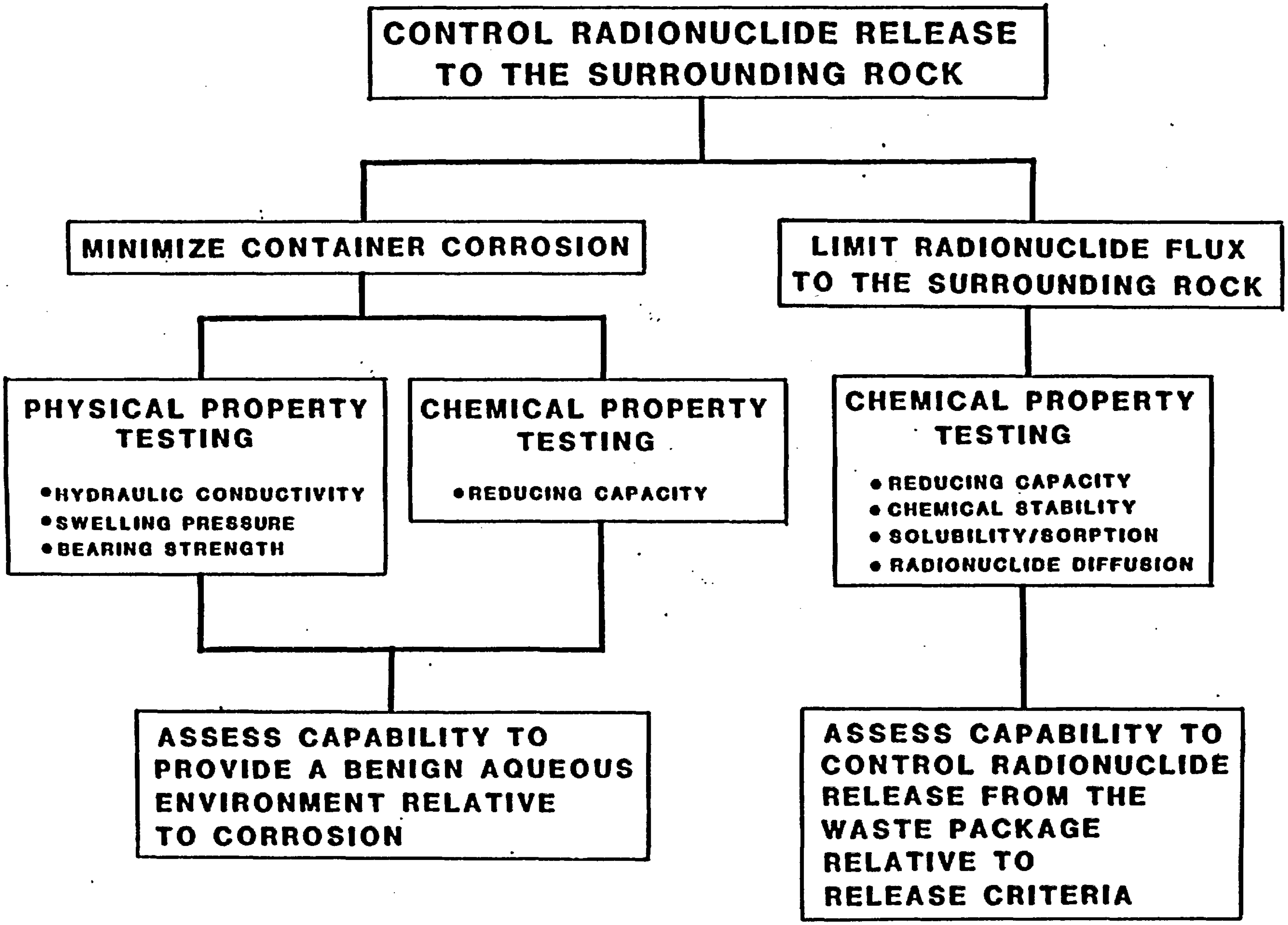
WP3405-3

PACKING MATERIALS TESTING OBJECTIVE

QUANTIFY PHYSICAL AND CHEMICAL PROPERTIES OF PACKING MATERIALS WHICH WILL MINIMIZE CANISTER CORROSION AND RADIONUCLIDE FLUX TO THE SURROUNDING ROCK

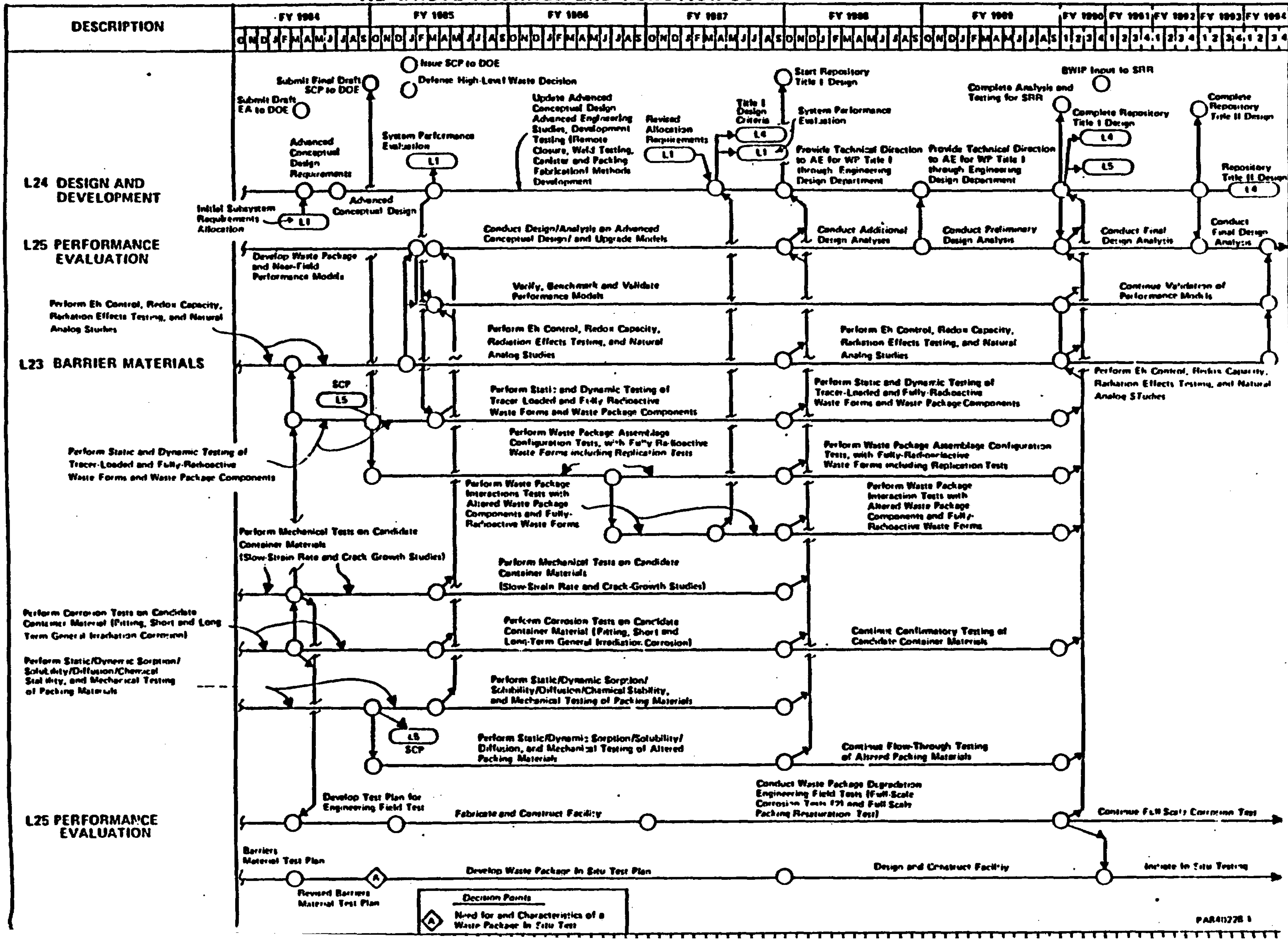
WP8405-16

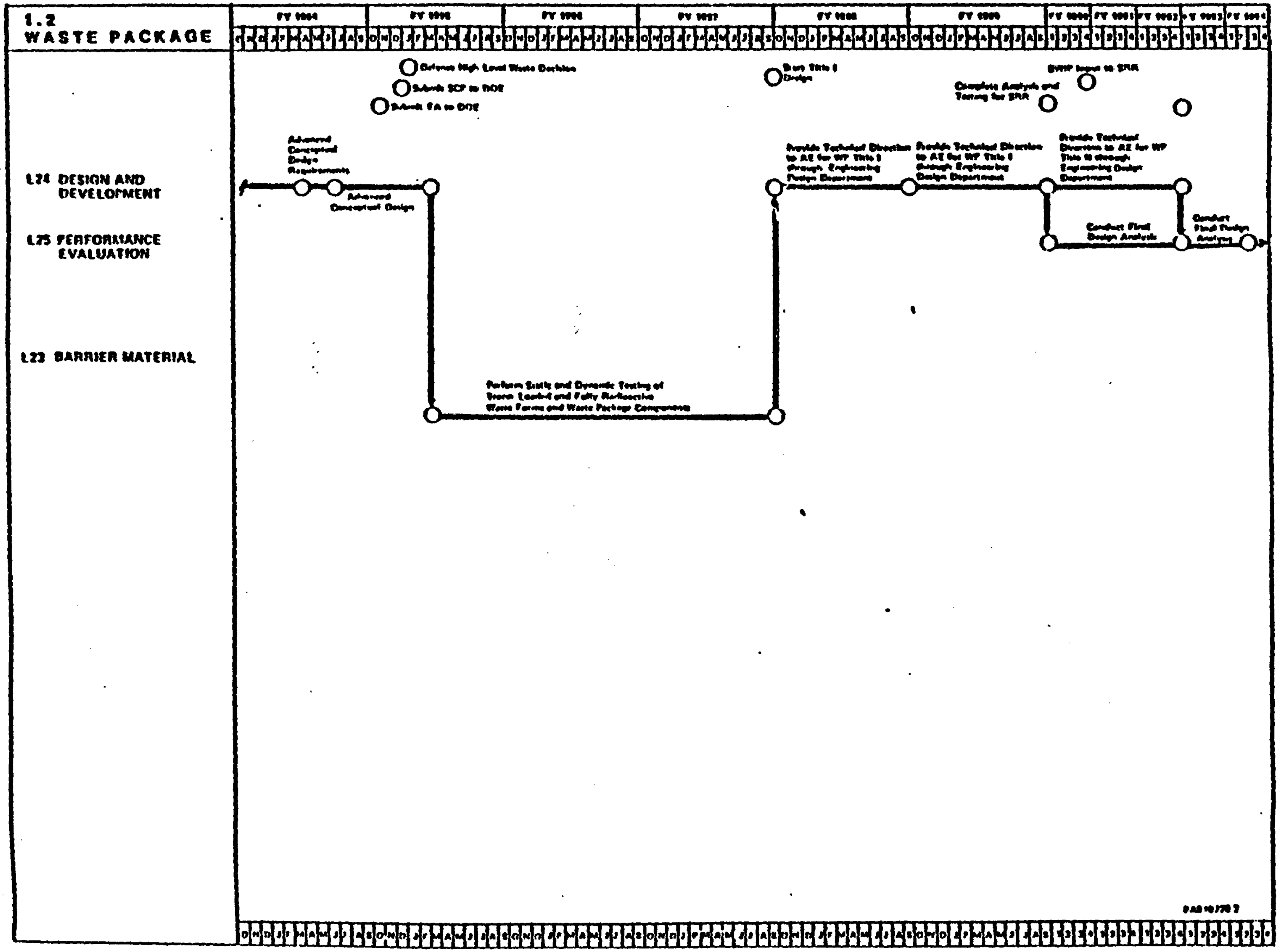
PACKING MATERIALS TESTING



WP8405-12

1.2 WASTE PACKAGE END FUNCTION SCHEDULE





WP8402-58A-CONT

CONSIDERATIONS IN NRC REVIEW OF BMTP

1. RELEVANCE OF TESTS AND DATA TO PERFORMANCE OBJECTIVES FOR THE WASTE PACKAGE STATED IN 10 CFR 60 AND TO THE WASTE PACKAGE ISSUES STATED IN NRC'S SCA.

2. WHAT SCENARIO AND WASTE PACKAGE DESIGN(S) IS THIS PLAN BASED ON? DETERMINE IF THE PLANNED TESTS WILL PROVIDE SUFFICIENT DATA TO ADDRESS THE DOMINENT FAILURE SCENARIOS FOR THE WASTE PACKAGE.

3. DETERMINE IF THE TESTS WILL YIELD SUFFICIENT DATA FOR AN INDEPENDENT ASSESSMENT OF WASTE PACKAGE PERFORMANCE AND AN INDEPENDENT DEVELOPMENT OF A PERFORMANCE MODEL.

ATTACHMENT 4

CONSIDERATIONS IN NRC REVIEW OF BMTP

4. DOES NRC AGREE WITH THE TEST DETAILS AND EXPERIMENTAL SET-UPS DEFINED IN THE PLAN? IF NOT, IDENTIFY THE SHORTCOMINGS AND PROPOSE IMPROVEMENTS.

5. DETERMINE IF THE TEST SCHEDULE IS REALISTIC.

6. DETERMINE IF THE TEST SEQUENCE IS LOGICAL.

7. REVIEW DETAILS OF EXPERIMENTAL SET-UPS AND TEST CONDITIONS.

BWIP engeneering planned for inclusion in the SCP will not have progressed to the level necessary for NRC to review and comment on these asspects.

3. As a result of (2), DOE will wish to reassess the SCP preparation process in light of NRC information needs.

4. Considerable discussion centered on the need, in the SCP, for interim, quantitative performance objectives for the components of the engineered system. NRC indicated that an acceptable way of specifying performance is through the standard approach of establishing performance reliability and the confidence level therein, as well as performing design analyses to determine these values. DOE pointed out that while this approach is being attempted, its usefulness may be limited because of the many variables involved. Additonally, an NWTS program-wide approach needs to be established by DOE.

5. NRC urged DOE to move aggressively in forging a concensus on the problem of modeling and testing the thermal-hydrological-mechanical-geochemical response of the repository host rock to waste emplacement. This could be the critical path item in site characterization because of its effect on time and scale of testing.

<u>Management Developments:</u>

DOE presented a preview of a release system for BWIP site characterization data and information (Attachment 9), which is expected to be implemented in October 1983. NRC indicated that the scheme appears to be working in the right direction because it appears to accommodate the main requirements, viz.

1. The system needs to capture all site characterization data, whether collected by the present or predecessor contractors.

2. The system needs to permit access to all data by NRC and other affected parties.

3. The system needs to explain to a user what is in the system and how it can be traced.

NRC STATEMENT ON RELIABILITY

1

Reliability of Engineered Barrier System

- 10 CFR Part 60 does not require a specified quantitative level of confidence or reliability.

  - Reasonable assurance is the standard.

  - However, the Commission expects that the information considered in a licensing proceeding will include probability distribution functions...

- Consideration of uncertainties in the data and the models is a major concern that will need to be addressed quantitatively in the license application.

  - The data to address these uncertainties must be gathered during the site characterization program.

  - Testing and data collection must consider in a systematic way all important interactions of the system affecting waste package performance.

  - One way to do this that would be acceptable to NRC and that could be factored into the analysis of the waste package design for the license application is by using the reliability assessment techniques in the draft technical position we are developing in combination with fault tree analysis.

substantially reduced. But in any event, the Commission anticipates that a high standard of engineering will be necessary—not only to compensate for geologic uncertainties at even the best reasonably available sites, but perhaps also to mitigate the consequences of unanticipated processes and events (including potential intrusion) during the years when fission product inventories remain high.

Although the Commission agrees with the underlying appraisal of the commenters that the isolation capabilities of the site play a key role in assuring that the performance objectives will be met, it finds no reason to change the rule's approach.

Reasonable Assurance

The proposed rule stated that with respect to the long-term objectives and criteria under consideration, "what is required is reasonable assurance, making allowance for the time period and hazards involved, that the outcome will be in conformance with those objectives and criteria." A number of commenters took exception to this formulation on the ground that it provides inadequate guidance as to the required level of proof. Others were concerned that "reasonable assurance" was too weak a test and that the Commission should not license DOE activities without a "high degree of confidence" that releases would be very small. Some commenters suggested that a statistical definition of acceptability should be employed. For the reasons set forth below, the Commission has not modified the language.

In the Commission's view, the "reasonable assurance" standard neither implies a lack of conservatism nor creates a standard which is impossible to meet. On the contrary, it parallels language which the Commission has applied in other contexts, such as the licensing of nuclear reactors, for many years. See 10 CFR 50.35(a) and 50.40(a). The reasonable assurance standard is derived from the finding the Commission is required to make under the Atomic Energy Act that the licensed activity provide "adequate protection" to the health and safety of the public; the standard has been approved by the Supreme Court. *Power Reactor Development Co. v. Electrical Union,* 367 U.S. 396, 407 (1961). This standard, in addition to being commonly used and accepted in the Commission's licensing activities, allows the flexibility necessary for the Commission to make judgmental distinctions with respect to quantitative data which may have large uncertainties (in the mathematical sense) associated with it.

The Commission has not modified the language, but has explained elsewhere (see *Anticipated/Unanticipated Processes and Events,* above) how the concept will be applied. The Commission expects that the information considered in a licensing proceeding will include probability distribution functions for the consequences from anticipated and unanticipated processes and events. Even if the calculated probability of meeting the Commission's standards is very high that would not be sufficient for the Commission to have "reasonable assurance"; the Commission would still have to assess uncertainties associated with the models and data that had been considered. This involves qualitative as well as quantitative assessments. The Commission would not issue a license unless it were to conclude, after such assessments, that there is reasonable assurance that the outcome will in fact conform to the relevant standards and criteria.

It is important to keep in mind this distinction between, first, a standard of performance and, second, the quality of the evidence that is available to support a finding that the standard of performance has been met. In principle, there is no reason why the first of these—the performance standard—cannot be expressed in quantitative terms. The rule does this in several places—notably, in including as performance objectives a designed containment period, a radionuclide release rate, and a pre-waste-emplacement groundwater travel time. Similarly, EPA's standard will establish limits on concentrations or quantities of radioactive material in the general environment.

Expressing a requisite level of confidence in quantitative terms is far more problematical. To be sure, measurement uncertainties are amenable to statistical analyses. Even though there may be practical limitations on the accuracy and precision of measurements of relevant properties, it is possible to make some quantitative statement as to how well these values are known. The licensing decisions which the Commission will be called upon to make involve additional uncertainties—those pertaining to the correctness of the models being used to describe the physical systems—which are not quantifiable by statistical methods. Conclusions as to the performance of the geologic repository and particular barriers over long periods of time must largely be based upon inference; there will be no opportunity to carry out test programs that simulate the full range of relevant conditions over the periods for which waste isolation must be maintained.

The validity of the necessary inferences cannot be reduced by statistical methods to quantitative expressions of the level of confidence in predictions of long-term repository performance. Similarly, the Commission will not be able to rigorously determine the probability of occurrence of an outcome that fails to satisfy the performance standards. It must use some other language, such as "reasonable assurance," to characterize the required confidence that the performance objectives will be met. In practice, this means that modeling uncertainties will be reduced by projecting behavior from well understood but simpler systems which conservatively approximate the systems in question. Available data must be evaluated in the light of accepted physical principles; but, having done so, the Commission must make a judgment whether it has reasonable assurance that the actual performance will conform to the standards the Commission has specified in this rule.

It should also be borne in mind that the fact-finding process is an administrative task for which the terminology of law, not science, is appropriate. The degree of certainty implied by statistical definition has never characterized the administrative process. It is particularly inappropriate where evidence is "difficult to come by, uncertain or conflicting because it is on the frontiers of scientific knowledge." *Ethyl Corp. v. EPA,* 541 F.2d 1, 28 (D.C. Cir. 1976).

Population vs. Individual Dose

Some commenters noted that the performance objectives are derived from an assumed EPA standard that is based upon consideration of doses to populations as a whole rather than to the maximally exposed individual. Several other analyses of repository design have examined prospective requirements in terms of keeping individual doses below specified values, and as a consequence have led to different conclusions. The differences represent a source of potential uncertainty regarding the overall goal for safety performance. However, the resolution of this question is a matter within the province of EPA. The Commission has assumed that the EPA approach will be based upon population dose, since that is the direction reflected in its working documents and its

Specification of Interim Reliability Goals

- Under final 10 CFR Part 60, DOE has great flexibility in design of the waste package.

  - Components that will be relied on to meet the performance objective.

  - Performance objectives for the engineered barrier system.

  - The level of reliability in each part of the system so that there is reasonable assurance that the overall system will meet the EPA standard.

  - DOE can optimize the design of individual components to provide the most cost effective system.

- Reliability goals for the overall system.

  - Overall system performance standard is the EPA standard.

  - While NRC does not believe EPA should specify numerical probabilities of releases in their standard, it is appropriate for DOE to use them in designing the repository and for NRC to consider in its review.

  - An overall performance goal is that releases having more than one chance in one hundred of occuring in 10,000 years should not exceed the table in EPA standard. This can be used to derive the needed reliability in parts of the system, taking into account the uncertainty in the data and the models used to assess performance.

ATTACHMENT 7

EXTRACT FROM NRC COMMENTS ON DOE MISSION PLAN
(JOHN G. DAVIS (NRC) TO MICHAEL J. LAWRENCE (DOE)),
LETTER OF FEBRUARY 8, 1984

This is an excerpt of a letter dated 2/8/84 from John G. Davis to Michael J. Lawrence of the DOE regarding Mission Plan Comments.

Paragraph No. 5:

"Over the past year, the NRC staff has informed DOE of the need to establish, as soon as possible, the intended performance requirements for repository system components on a site specific basis. 10 CFR Part 60 gives DOE flexibility, on a site by site basis, to propose trade-offs among system components (natural and engineered). We believe that it is essential that decisions be made promptly by DOE for these intended component performance requirements. These decisions are essential to provide focus to the repository investigation programs. Without this focus, the programs may not provide an adequate or timely basis for DOE decisions or for NRC reviews. Also, NRC's ability to give timely guidance to DOE on licensing information needs may be hindered and in some cases made impossible.

TOPICS TO BE PRESENTED BY NRC IN NRC/BWIP MEETING OF MAY 8-10, 1984

- O BWIP'S STATISTICAL TEST DESIGN
- O APPROACH AND RATIONALE
- O CORROSION TESTS AND DATA
- O PACKING TESTS AND DATA
- O OVERALL DATA SUFFICIENCY FOR PERFORMANCE ASSESSMENT
- O TEST SCHEDULE AND TEST LOGIC
- O GEOCHEMISTRY

ATTACHMENT 8

SUMMARY OF PRELIMINARY NRC

REVIEW OF BMTP

1. THE PRESENT DRAFT BMTP IS STRUCTURED ON THE SUPPOSITION OF A STABILIZED REDUCING NEAR-FIELD ENVIRONMENT. IT MAY NOT BE POSSIBLE TO VALIDATE THIS ENVIRONMENT. THEREFORE, THE TEST SCOPE ~~FOR THE WASTE PACKAGE MUST~~ SHOULD BE BROADENED TO INCLUDE TESTS IN AN OXIC ENVIRONMENT.

ATTACHMENT 9

2. THE APPLICABILITY OF THE OSTWALD STEP RULE MUST BE ~~VALIDATED TO SHOW ITS APPLICABILITY FOR~~ DEMONSTRATED FOR PROJECTING WASTE/BARRIER/ROCK INTERACTIONS. NRC HAS IDENTIFIED MANY PROBLEM AREAS. THE ~~TESTS~~ APPROACH PRESENTED IN THE DRAFT BMTP IS INADEQUATE TO ~~DO THE VALIDATION~~ ESTABLISH GENERAL APPLICABILITY.

3. ~~THERE ARE SERIOUS OMISSIONS OF PARAMETERS IN THE TEST APPROACH DESCRIBED IN THE BMTP, INCLUDING TIME OF FAILURE OF WASTE PACKAGE COMPONENTS, RANGE OF WATER CHEMISTRY AND TRANSPORT OF COLLOIDS/PARTICLES.~~

It is not clear that the variations of test conditions due to time of failure of waste package components, range of water chemistry and transport of colloids/particles were considered in the waste/barrier/rock interaction test plans.

4. THE "STATISTICAL TEST DESIGN" PROPOSED IN THE DRAFT BMTP IS COMPATABLE WITH NRC'S ~~POSITION~~ VIEW ON WASTE PACKAGE RELIABILITY. DETAILS OF TEST MATRIX ARE NOT PROVIDED IN THE PRESENT DRAFT. WE THEREFORE CANNOT ASSESS DATA SUFFICIENCY EXCEPT FROM A QUALITATIVE STANDPOINT.

BWIP'S STATISTICAL TEST DESIGN

APPENDIX A: "STATISTICAL TEST DESIGN"

COMMENTS

THE BMTP PROPOSES USING FULL AND FRACTIONAL FACTORIAL DESIGN OF EXPERIMENTS FOR SCREENING TESTS, FOR TESTS TO OBTAIN EMPIRICAL MODELS, AND FOR ACCELERATED AGING TESTS.

IN PRINCIPLE BOTH FULL AND FRACTIONAL FACTORIAL DESIGN OF EXPERIMENTS CONSTITUTE ACCEPTABLE APPROACHES AS LONG AS THEY ARE BACKED BY SOUND STATISTICAL ANALYSES. IN PARTICULAR FACTORIAL TECHNIQUES ARE ESPECIALLY USEFUL AT THE SCREENING STAGE. MODELING REQUIRES HIGHER LEVEL EXPERIMENTING AND AN ANALYSIS OF THE STATISTICAL UNCERTAINTY OF THE DATA AND MODELS. THE BMTP DOES NOT ADDRESS UNCERTAINTY ANALYSIS.

STATISTICAL DESIGN IS CERTAINLY ACCEPTABLE FOR ACCELERATED AGING TESTS. IN THESE CASES IT IS IMPORTANT TO SHOW THAT THE MECHANISMS TO BE INVESTIGATED ARE NOT REPLACED BY OTHER MECHANISMS DUE TO THE SELECTED, ABNORMAL STRESS LEVELS.

SIGNIFICANCE

IT IS NOT CLEAR FROM THE DOCUMENT WHETHER UNCERTAINTY ANALYSIS WILL BE PERFORMED. THE PROPOSED METHODOLOGY HAS THE POTENTIAL FOR SUCH AN ANALYSIS.

NRC WILL NEED AN UNCERTAINTY ANALYSIS OF THE DATA AND MODELS IN ORDER TO JUDGE COMPLIANCE WITH 10CFR60.

SUGGESTIONS

ALONG WITH TEST RESULTS DOE SHOULD ALSO PROVIDE TEST PLANS AND ANALYSES DETAILING

A. FACTORS SCOPED FOR, INDICATING AS WELL IF SOME PARTICULAR FACTORS WERE DISMISSED A PRIORI AND WHY

B. RATIONALE WHY A SPECIFIC NUMBER OF LEVELS WAS SELECTED

C. RATIONALE IN ASSIGNING LEVEL VALUES

D. ERROR TOLERANCE FOR EACH LEVEL VALUE

E. EXPERIMENTAL ERROR ANALYSIS WHERE APPROPRIATE. THIS SHOULD ADDRESS ACCURACY AND PRECISION OF MEASUREMENTS. FURTHERMORE, ANY REGRESSION OR FIT SHOULD HAVE AN ANALYSIS OF CONFIDENCE OF FIT.

APPROACH AND RATIONALE

SCIENTIFIC RATIONALE FOR WASTE/BARRIER/ROCK

INTERACTION TESTING (APPENDIX B)

COMMENTS

- THE OSTWALD STEP RULE MAY EVENTUALLY PROVE TO BE AN ACCEPTABLE APPROACH FOR ESTIMATING LONG TERM CONTROLLED RELEASE.

- HOWEVER, POSSIBLE PROBLEMS ASSOCIATED WITH THE STEP RULE WHICH SHOULD BE ADDRESSED IN A TEST PROGRAM INCLUDE:

  A) THE STEP RULE, STRICTLY SPEAKING, IS ONLY VALID FOR A CLOSED ISOTHERMAL SYSTEM. IN A BASALT REPOSITORY THERE WILL BE MASS FLOW IN AND OUT OF THE WASTE PACKAGE. FLOW RATE EFFECTS SHOULD, THEREFORE, BE EVALUATED.

  B) AS A WASTE PACKAGE COOLS, THE SOLUBILITIES OF SOME RADIONUCLIDES COULD INCREASE BECAUSE OF INVERSE TEMPERATURE EFFECTS.

  C) THE STEP RULE, AS STATED IN (A) ABOVE IS VALID FOR A CLOSE SYSTEM CONTAINING A CONSTANT INVENTORY OF ATOMS. THIS IS NOT THE CASE IN A WASTE PACKAGE BECAUSE OF TRANSMUTATION. THUS IT IS POSSIBLE THAT A PHASE RICH IN RADIONUCLIDE X COULD BECOME MORE SOLUBLE AS X BEGINS TO TRANSMUTE TO Y.

D) ALPHA RADIATION DAMAGE IN A PRECIPITATED PHASE COULD CAUSE AN INCREASE IN SOLUBILITY BECAUSE OF THE FORMATION OF AMORPHOUS MATERIAL. ALSO IF ALPHA DAMAGE GIVES A LARGE INCREASE IN THE STORED ENERGY IN A PHASE IT MAY CAUSE DISSOLUTION.

E) THE STEP RULE DESCRIBES THE PROGRESSION OF PRECIPITATION IN GROUNDWATER AND IS THEREFORE CONCERNED WITH MAJOR ELEMENT GEOCHEMISTRY. RADIONUCLIDES ON THE OTHER HAND ARE PRESENT AS TRACE ELEMENTS IN PRECIPITATING PHASES. THUS THE SOLUBILITY OF VARIOUS RADIONUCLIDES WILL PROBABLY BE DETERMINED BY THE MAJOR NON-RADIOACTIVE ELEMENTS PRESENT IN A PHASE.

SIGNIFICANCE OF COMMENTS

THE COMMENTS ABOVE SHOW THAT THE FUNDAMENTAL BASES FOR THE APPLICABILITY OF THE STEP RULE ARE QUESTIONABLE FOR THE BWIP REPOSITORY SYSTEM. COMPREHENSIVE TESTING WILL BE NEEDED TO ESTABLISH SHORTCOMINGS IN THE STEP RULE SO THAT DEFICIENCIES CAN BE ACCOMMODATED AND CONSERVATIVE ESTIMATES OF SOLUBILITY ARE OBTAINED FOR PERFORMANCE ASSESSMENT AND LICENSING.

SUGGESTIONS

BWIP SHOULD INCLUDE IN THEIR TEST PLAN SPECIFIC PROGRAM TO DETERMINE HOW GROUNDWATER FLOW, ALPHA RADIATION OF PRE-CIPITATION PHASES, TRANSMUTATION IN PRECIPITATED PHASES, AND POSSIBLE INVERSE TEMPERATURE EFFECTS CONTROL SOLUBILITIES.

APPROACH TO WASTE/BARRIER/ROCK INTERACTION TESTING

(APPENDIX C)

COMMENTS

- FIGURE C-1 (APPENDIX C) OUTLINES THE BWIP APPROACH TO WASTE/BARRIER/ROCK INTERACTION TESTING.

- APPROACH IS LOGICAL BUT IT IS NOT CLEAR WHETHER, IN FACT, THE BMTP SPECIFIES TESTS UNDER THE CONDITIONS DESCRIBED IN FIGURE C-1.

- MUST BE RECOGNIZED THAT THERE IS A SPECTRUM OF WATER CHEMISTRIES THROUGH THE WASTE PACKAGE AND THIS SPECTRUM VARIES WITH TIME AND TEMPERATURE. FOR EXAMPLE, FIGURE C-1 SEEMS TO INDICATE THAT THERE IS A UNIQUE WATER B CHEMISTRY IN THE PACKING. THIS IS INCORRECT.

- VERY IMPORTANT TO ENSURE THAT CORRECT WATER CHEMISTRIES ARE USED FOR EVALUATION AND PREDICTION OF CONTAINER CORROSION, SOURCE TERM SPECIATION, LIMITING CONCENTRATIONS (SOLUBILITIES) AND PACKING MATERIAL SORPTION EFFECTS, ETC.

- PAGE 5-81 (TOP) STATES "TESTS WILL BE CONDUCTED IN AUTOCLAVES CONTAINING SYNTHETIC GRANDE RONDE BASALT GROUNDWATER, CRUSHED BASALT ROCK, AND BENTONITE." THIS IS NOT APPROPRIATE SINCE THE WATER FIRST CONTACTING AN ACTUAL CONTAINER WILL BE SIGNIFICANTLY DIFFERENT THAN GRANDE RONDE WATER.

SIGNIFICANCE OF COMMENTS

- THE USE OF TEST PROCEDURES WHICH DO NOT EMPLOY ANTICIPATED WATER CHEMISTRIES, ETC., MAY YIELD DATA ON CORROSION, SOURCE TERM SPECIATION, SOLUBILITIES, AND SORPTION WHICH DO NOT CONSERVATIVELY ESTIMATE WASTE PACKAGE PERFORMANCE. SUCH DATA MAY NOT BE ACCEPTABLE FOR LICENSING.

SUGGESTIONS

NEED TO ESTABLISH THE APPROPRIATE WATER CHEMISTRIES FOR DIFFERENT LOCATIONS IN THE PACKAGE AND USE THESS AS STAN-DARDS FOR EVALUATING INDIVIDUAL BARRIER PERFORMANCE. A POSSIBLE SEQUENCE OF TESTS TO ESTABLISH WATER CHEMISTRIES IS:

A) CONDUCT HIGH-TEMPERATURE BASALT/GRANDE RONDE H_2O TESTS AT 1 ATMOS. TO ALLOW PRECIPITATION OF DISSOLVED SALTS TO SIMULATE PRE-CLOSURE EVENTS.

B) REACT THIS MODIFIED BASALT WITH GRANDE RONDE WATER TO DETERMINE ACTUAL CHEMISTRY OF WATER FIRST CONTACTING PACKING MATERIAL.

C) CONDUCT HYDROTHERMAL COLUMN TESTS USING WATER CHEMISTRY FROM (B) AND PACKING MATERIAL TO DETERMINE HOW WATER CHEMISTRY EVOLVES DURING PASSAGE THROUGH THE PACKING TOWARDS CONTAINER. CORRELATE THESE WATER CHEMISTRY SPECTRA WITH SOLUBILITY AND SORPTION TESTS AND USE THE APPROPRIATE WATER FOR CONTAINER CORROSION STUDIES.

A SIMILAR SEQUENCE OF WATER CHEMISTRY EVALUATIONS AND WASTE PACKAGE COMPONENT TESTS CAN BE DEVISED FOR REGIONS "DOWNSTREAM" FROM THE WASTE FORM.

COMMENT: The BWIP testing program is based on the assumption that the groundwater will be anoxic except for a brief period immediately following closure and that for corrosion (and other) testing purposes one can assume that all significant processes are governed by a very reducing E_h which will prevail in the near-field environment.

There are a number of factors which make thisassumption questionable:

a) Gamma radiation and, after breach, alpha radiation, will continuously generate oxidizing species by radiolysis of water.

b) Hydrogen produced by radiolysis and corrosion may dissolve in the containers and may also escape because of its high diffusivity.

c) The presence of methane is irrelevant because of its very sluggish oxidation kinetics in aqueous environments.

d) The buffering capacity of basalt is limited to surfaces. New surfaces generated during mining will have been exposed to air during the operational period; old surfaces (pores and existing cracks) will have been exposed to groundwater over geologic times.

e) It is not clear that the concept of a system master E_h will be useful in predicting localized corrosion behavior.

SIGNIFICANCE: If the testing program is based on the assumptions that the groundwater is anoxic and that all significant processes are controlled by a very reducing E_h, there is considerable risk that DOE may not have sufficient data to support their license application because it may not be possible to provide reasonable assurance of the validity of the assumptions. In this case, in the absence of test data under more oxidizing conditions, the ability of the waste package to meet the requirements of 10 CFR 60 will not have been demonstrated.

CORROSION TESTS AND DATA

SUGGESTION: Corrosion data for each potential failure mechanism (uniform corrosion, pitting corrosion, hydrogen embrittlement, etc.) should be collected under the most unfavorable conditions for that particular failure mechanism consistent with what is certainly known about the environment (e.g., hydrogen damage should be collected under the most reducing plausible conditions, whereas pitting corrosion data should in general be taken under the most oxidizing plausible conditions). At the same time, of course, DOE retains the option of demonstrating that the environment will be characterized by some fixed E_h, that there is sufficient buffering capacity to maintain this E_h during the relevant storage period, and that this E_h does in fact control the significant processes.

Furthermore, DOE needs to collect more mechanistic data so as to permit confidence in extrapolations and interpolations. It should be borne in mind that any extrapolation or interpolation involves the assumption that there is no significant change in mechanism, and that such an assumption must almost always be based on considerable detailed mechanistic knowledge.

COMMENT: The plan for experimental studies on localized corrosionand hydrogen-related degradation mechanisms needs to be described in detail.

SIGNIFICANCE: It appears that if reasonable care is taken with the choice of site and with container design and manufacture, the most probable cause of container failure will be some sort of localized corrosion (or possibly hydrogen-related degradation), quite likely at a weldment. Unlike general corrosion, these types of failure mechanisms often depend strongly on variations in the concentrations of alloying elements (e.g., the stress corrosion cracking behavior of low-carbon steels in some environments depends strongly on C content; pitting behavior of some steels is severely aggravated by minor quantities of Cu, S, and possibly P), as well as on metal cleanliness, thermal history, and texture.

SUGGESTION: The plan should be developed and communicated to NRC. In developing the QA system for the plan careful attention should be paid to characterizing the alloy samples.

COMMENT: The plan for experimental studies on localized corrosionand hydrogen-related degradation mechanisms needs to be described in detail.

SIGNIFICANCE: It appears that if reasonable care is taken with the choice of site and with container design and manufacture, the most probable cause of container failure will be some sort of localized corrosion (or possibly hydrogen-related degradation), quite likely at a weldment. Unlike general corrosion, these types of failure mechanisms often depend strongly on variations in the concentrations of alloying elements (e.g., the stress corrosion cracking behavior of low-carbon steels in some environments depends strongly on C content; pitting behavior of some steels is severely aggravated by minor quantities of Cu, S, and possibly P), as well as on metal cleanliness, thermal history, and texture.

SUGGESTION: The plan should be developed and communicated to NRC. In developing the QA system for the plan careful attention should be paid to characterizing the alloy samples.

PACKING TESTS AND DATA

PACKING MATERIAL CONSIDERATIONS

COMMENTS

THE PACKING MATERIAL PROGRAM NEEDS CLARIFICATION WITH RESPECT TO DEFINING TESTS SPANNING ANTICIPATED CONDITIONS. ALSO IT SEEMS THAT SOME BWIP TESTS ARE OF LIMITED IMPORTANCE TO LICENSING. SPECIFIC COMMENTS INCLUDE:

- NEED TO DETERMINE THE PHYSICAL CHARACTERISTICS AND UNIFORMITY OF FULL SCALE EMPLACED PACKING TO ENSURE THAT LABORATORY TESTS ARE CONDUCTED ON APPROPRIATE MATERIAL.

- ALPHA IRRADIATION DAMAGE MAY CHANGE THE PERMEABILITY AND SORPTIVE CAPACITY OF PACKING.

- BWIP SHOULD ADDRESS THE DIFFERENCES IN THE PERFORMANCE OF PACKING FOR A RANGE OF CONTAINER FAILURE TIMES. IMPORTANT BECAUSE 1000 YEAR OLD PACKING MAY BEHAVE BETTER OR WORSE THAN 300 YEAR OLD PACKING WITH RESPECT TO PERMEABILITY, SOLUBIITY OF RADIONUCLIDES, SORPTIVE CAPACITY, ETC.

- NEED TO ADDRESS THE ADEQUACY OF DRY PACKING MATERIAL CONDITIONS TO COVER POTENTIAL PROBLEMS ASSOCIATED WITH LOW THERMAL CONDUCTIVITY AND THE PEAK TEMPERATURE OF $\sim 300^{\circ}C$ IN THE PACKING AND WASTE FORM FOLLOWING CLOSURE AND BEFORE RESATURATION.

- One set of tests considered unnecessary is that for static diffusion/transport tests for radioactive tracers in a gamma field with hydrothermally altered packing (Figure 5-19 of BMTP). Gamma effects during the controlled release period should be negligible.

SIGNIFICANCE OF COMMENTS

THE USE OF HIGH INTEGRITY LABORATORY SAMPLES OF PACKING MATERIAL WILL PROBABLY GIVE BETTER PERFORMANCE COMPARED TO ACTUALLY EMPLACED PACKING. DATA OBTAINED WILL THEREFORE BE NON-CONSERVATIVE AND OF LIMITED VALUE FOR LICENSING.

ALPHA DAMAGE MAY BECOME SIGNIFICANT AFTER MANY THOUSANDS OF YEARS SO THAT SOLUBILITIES AND CONTROLLED RELEASE RATES FOR RADIONUCLIDES ARE INCREASED IN THE VERY LONG TERM.

ALSO, THE PERFORMANCE OF UNNECESSARY TESTS WILL PROBABLY REDUCE EFFORTS IN AREAS WHERE IMPORTANT LICENSING DATA ARE NEEDED.

PERFORMANCE ASSESSMENT

DATA SUFFICIENCY FOR PERFORMANCE ASSESSMENT

COMMENT

- CURRENT BMTP DRAFT DOES NOT ESTABLISH THE LINK BETWEEN PERFORMANCE ASSESSMENT AND THE TEST PROGRAM.

SIGNIFICANCE OF COMMENT

- PERFORMANCE ASSESSMENT IS NECESSARY FOR LICENSING.
- PERFORMANCE ASSESSMENT REQUIREMENTS INFLUENCE NATURE OF TEST PROGRAM.
- UNDERSTANDING HOW TEST DATA WILL BE USED IN PERFORMANCE ASSESSMENT IS ESSENTIAL FOR NRC REVIEW OF BMTP.
- WAITING UNTIL SEPTEMBER REVISION OF BMTP WOULD HAMPER NRC REVIEW.

PERFORMANCE ASSESSMENT
(CONTINUED)

SUGGESTIONS

- PROVIDE NRC WITH ADVANCE SUMMARY OF BWIP PERFORMANCE ASSESSMENT APPROACH (ESPECIALLY FOR WASTE PACKAGE).
  - TOPICS COVERED IN PERFORMANCE ASSESSMENT PLAN (2/28/83). IS IT SUPERSEDED?
  - DETERMINISTIC / PROBABILISTIC
  - ROLE OF UNCERTAINTY AND SENSITIVITY ANALYSIS
  - CONSOLIDATED LARGE MODELS (E.G., WAPPA) VS SMALL MODELS
  - CODES
  - LINK BETWEEN PERFORMANCE ASSESSMENT AND TEST PROGRAM
- IF SPECIFIC PERFORMANCE ASSESSMENT TECHNIQUES HAVE NOT BEEN CHOSEN, PROVIDE A DISCUSSION OF THE ALTERNATIVES AND THEIR CHARACTERISTICS.

DATA SUFFICIENCY

COMMENTS

- CURRENT BMTP DRAFT PROVIDES NEITHER A SUMMARY LIST OF PLANNED TESTS NOR TEST PLAN SCHEDULES THAT CAN BE READILY INTERPRETED.

- USE OF BOUNDING OR CONSERVATIVE VALUES IS NOT MENTIONED; TESTING APPEARS TO BE ORIENTED TOWARD DEVELOPING BEST-ESTIMATES.

SIGNIFICANCE OF COMMENTS

- NRC NEEDS SPECIFIC TEST INFORMATION, E.G.:

  - WHEN WILL INDIVIDUAL TEST PLANS BE AVAILABLE TO NRC FOR CONSTRUCTIVE COMMENT BEFORE TESTS ARE PERFORMED?

  - WHEN AFTER TEST COMPLETION WILL DATA BE AVAILABLE FOR SIMILAR NRC COMMENT PRIOR TO START OF FOLLOW-ON TESTS?

- KNOWLEDGE OF THE BOUNDS ON DATA ACCURACY IS ESSENTIAL FOR NRC PERFORMANCE ASSESSMENT.

- IF IT IS NOT POSSIBLE TO DEVELOP DEFENSIBLE BEST-ESTIMATES, BOUNDING OR CONSERVATIVE VALUES WILL BE EASIER TO OBTAIN.

DATA SUFFICIENCY
(CONTINUED)

SUGGESTIONS

- PROVIDE LIST OF TESTS, TEST MATRICES, AND SCHEDULES FOR TESTS AND REPORTS.
- WITH RESPECT TO DATA COLLECTION AND PRESENTATION:
  - DEVELOP BEST-ESTIMATE.
  - DEFINE BOUNDS.
  - USE CONSERVATISM IF A BEST ESTIMATE IS NOT DEFENSIBLE.

TEST SCHEDULE-TEST LOGIC

LOGIC/SCHEDULAR DIAGRAMS

COMMENTS

- IT IS RECOGNIZED THAT LOGIC DIAGRAMS ARE VERY DIFFICULT TO PREPARE WITH RESPECT TO DEFINING APPROPRIATE TEST PROGRAMS AND TEST CONDITIONS. THIS ARISES BECAUSE WASTE PACKAGE DESIGNS DETERMINE THE TEST MATERIALS, TEST TEMPERATURE, WATER CHEMISTRIES, ETC., BUT THESE DESIGNS CAN ONLY BE OPTIMIZED IF APPROPRIATE TEST DATA ARE OBTAINED. THUS THERE WILL NEED TO BE MANY INTERACTIONS IN DESIGN AND TESTING PROGRAMS BEFORE A FINALIZED DESIGN CAN BE SPECIFIED AND ITS PERFORMANCE DETERMINED.

- THE BWIP LOGIC DIAGRAMS, HOWEVER, DO NOT DETAIL DECISION POINTS AT WHICH MODIFIED TEST CONDITIONS WILL HAVE TO BE USED. IT SEEMS THAT THE CORROSION, PACKING MATERIAL, AND WASTE FORM PROGRAMS WILL PROCEED INDEPENDENTLY WITH NO FORMAL INTERACTION.

- IT IS ALSO NOT CLEAR HOW BWIP WILL CHOOSE BETWEEN THE MANY POSSIBLE COMBINATIONS OF EXPERIMENTAL PARAMETERS AND ANALYTICAL TECHNIQUES (GIVEN IN BMTP SUMMARY TABLES) AS THE PROGRAM PROCEEDS.

SIGNIFICANCE OF COMMENTS

IT SEEMS THAT THE BWIP LOGIC AND SCHEDULAR DIAGRAMS ASSUME FULLY SUCCESSFUL PROGRESS IN ALL ASPECTS OF THE INDIVIDUAL PROGRAMS. NRC IS CONCERNED THAT THE DISCOVERY OF A NEW IMPORTANT TEST ENVIRONMENT MAY INVALIDATE MUCH OF THE EARLY DATA. THIS WOULD DETRIMENTALLY AFFECT THE BWIP SCHEDULE AND NRC'S ASSOCIATED LICENSING ACTIVITIES.

SUGGESTIONS

NRC SUGGESTS THAT IN PERIODIC BWIP/NRC WORKSHOPS THAT BWIP GIVE STATUS REPORTS ON PROGRESS TO DATE. BWIP SHOULD OUTLINE POSSIBLE CONTINGENCY PLANS FOR MINIMIZING SCHEDULAR DELAYS IF THERE ARE SIGNIFICANT CHANGES IN TESTS THAT ARE CURRENTLY SPECIFIED IN THE BMTP.

. BWIP SHOULD CONSIDER THE INCLUSION OF TEST MATRICES AND "DECISION TREES" IN THE FINAL VERSION OF THE BMTP.

KEY RADIONUCLIDE LIST

COMMENT

- O SOME RADIONUCLIDES WILL BE MUCH MORE IMPORTANT THAN OTHERS IN CONTRIBUTING TO RADIOACTIVITY RELEASE TO THE ENVIRONMENT
- O BWIP HAS NOT PUBLISHED THE SUPPORTING DOCUMENTATION SHOWING HOW THIS LIST WAS DEVELOPED

SIGNIFICANCE OF COMMENT

- O THE BMTP CONTAINS A KEY RADIONUCLIDE TABLE ON PAGE 5-49
- O THE LIST WILL HELP DIRECT THE BWIP EXPERIMENTAL EFFORT

SUGGESTIONS

- O THE ASSUMPTIONS AND CALCULATIONS USED TO DEVELOP THIS LIST SHOULD BE DOCUMENTED FOR REVIEW AND ANALYSIS.

GEOCHEMISTRY

OSTWALD STEP DIAGRAM AS A MODEL FOR RADIONUCLIDE SOLUBILITY LIMITS

COMMENT

- O THE OSTWALD STEP DIAGRAM IS CLEARLY NON-CONSERVATIVE FOR THE PREDICTION OF RADIONUCLIDE SOLUBILITY LIMITS

SIGNIFICANCE OF COMMENT

- O APPENDIX B OF THE BWIP ATTEMPTS TO USE THIS AS PROOF THAT ANY RADIONUCLIDE CONCENTRATION VALUE MEASURED AT ANY TEST TIME WOULD EITHER STAY THE SAME OR DECREASE WITH TIME, I.E., THAT ANY TEST VALUE WILL BE CONSERVATIVE

SUGGESTIONS

- BWIP SHOULD CONSIDER:
  - O THAT THE OSTWALD STEP DIAGRAM HAS ONLY BEEN VALIDATED FOR SILICATE MATERIALS
  - O THAT THE OSTWALD STEP DIAGRAM IS ONLY APPLICABLE IF A SOLID PHASE OF THE RADIONUCLIDE PRECIPITATES
  - O THAT IF RADIONUCLIDES ARE REMOVED FROM SOLUTION BY INTERSTITIAL SUBSTITUTION INTO OR SORPTION ONTO POORLY CRYSTALLINE SILICATES, THE RADIONUCLIDES MAY BE RELEASED TO THE SOLUTION AT A LATER TIME WHEN THE SILICATES RECRYSTALLIZED TO MORE STABLE, BETTER ORDERED MINERALS, I.E., THE RADIONUCLIDE CONCENTRATION MAY <u>INCREASE</u> WITH TIME, NOT DECREASE

- THESE ARGUMENTS ARE STRICTLY VALID ONLY FOR A CLOSED SYSTEM, AND THE OSTWALD STEP DIAGRAM MAY BE INAPPLICABLE TO FLOWING OPEN SYSTEMS

REDOX POTENTIAL OF BASALT/GROUNDWATER SYSTEMS

COMMENT

- O IN MANY PLACES THE BMTP STATES THAT THE TEST EH WILL BE MEASURED AND/OR CONTROLLED

SIGNIFICANCE OF COMMENT

- O THE SYSTEM EH WILL AFFECT THE VALENCE OF THE RADIONUCLIDES
- O IN GENERAL, REDUCED RADIONUCLIDES MAY BE LESS SOLUBLE AND/OR MORE STRONGLY ADSORBED
- O IT IS UNCERTAIN IF A SYSTEM MASTER EH IS EFFECTIVE, OR EVEN IF MEANINGFUL EH VALUES CAN BE MEASURED IN SOME EXPERIMENTS
- O THE SYSTEM EH OR REDOX CONDITION TO BE EXPECTED IN THE ENGINEERED FACILITY OR THE SITE HOST ROCKS THROUGH REPOSITORY TIME HAS NOT BEEN WELL ESTABLISHED
- O EMPLOYMENT OF ADDED CHEMICAL REDUCTANTS MAY POORLY MODEL HETEROGENEOUS BASALT/GROUNDWATER SOLUTE REACTIONS

SUGGESTIONS

BWIP SHOULD RECONSIDER:

- O IF IT IS CONSERVATIVE TO ATTEMPT TO CONTROL TEST E_H AT SOME PRECONCEIVED VALUE
- O IF IT CAN BE PROVEN THAT A MASTER E_H EXISTS FOR TESTS, I.E., THAT ALL REDOX COUPLES IN THE TEST ARE AT THE SAME E_H CONDITION
- O IF ANOXIC CONDITION (AIR EXCLUDED) TESTS WITHOUT DELIBERATE E_H CONTROL MAY NOT BETTER MODEL EXPECTED REPOSITORY CONDITIONS THAN TESTS HELD AT PREDETERMINED E_H VALUES
- O IF MEASUREMENT OF E_H PROVES DIFFICULT OR UNCERTAIN, IS SUCH MEASUREMENT NECESSARY FOR LICENSING
- O IF THE EXPECTED REDOX CONDITION (OR E_H) CAN BE PREDICTED FOR THE ENGINEERED FACILITY AND THE SITE THROUGH TIME WITH THE PRESENT INFORMATION ABOUT BASALT PHASES, RADIOLYTIC EFFECTS, ETC.
- O IF EXPERIMENTAL RESULTS PREVIOUSLY OBTAINED IN THE PRESENCE OR HYDRAZINE (RADIONUCLIDE SORPTION VALVES) ARE CONSERVATIVE AND ACCEPTABLE FOR LICENSING APPLICATION

Packing Material Considerations:

Response to Bullet 1

Laboratory Tests of packing materials' physical properties are necessarily limited and will be supplemented by engineering scale tests to more realistically simulate large scale emplacement.

Response to Bullet 2

The effects of alpha irradiation on packing materials are expected to be minimal. However, we will consider the effects of alpha radiation experimentally.

Response to Bullet 3

Changes in packing material properties as a function of time are being considered by performing experiments as a function of temperature and degree of alteration.

Response to Bullet 4

Experimental determination of dry packing material thermal conductivities and their effect on thermal histories have been completed.

Response to Bullet 5

It is necessary to perform this kind of test in the event that premature container failure occurs during the period of high gamma flux.

Comment on Ostwald's Step Rule

Concerns about the strict applicability of Ostwald's Step Rule to radionuclide solution behavior are legitimate concerns. These concerns will be carefully considered and addressed in the test plan update. The need for extensive testing to demonstrate the validity of Ostwald's Step Rule with regard to radionuclide solution and precipitation behavior is recognized.

Approach to Waste/Barrier/Rock Interaction Testing

Response to All Bullets:

BWIP believes the need to do separate experiments to determine the effects of barrier materials on water chemistry is largely unnecessary because these effects will occur during the course of an experiment where barrier materials are included. The initial solution composition will change during the course of the experiment to the appropriate composition. However, the suggestion for doing experiments to determine the effects of salt deposits is well founded. It will receive serious consideration in the test plan update. Flow through experiments (suggestion (C)) are already planned.

CORROSION TESTS AND DATA

FRACTURE MECHANICS TESTING APPROACH

- - J - Integral will be assessed for potential application to container materials evaluation

More Detailed Description of Testing Plans on Localized Corrosion and Hydrogen - Degradation Modes

- Rationale for localized corrosion/hydrogen related degradation testing effects will be clarified in the revised BMTP.

- Effects on container material alloying elements (e.g., C, S, P, Cu) on localized degradation modes will continue to be investigated.

CORROSION TESTS AND DATA (continued)

Anoxic Groundwater Assumption

- Groundwater oxygen concentration effects on corrosion behavior will continue to be investigated over a range of levels anticipated at the container surface.

- Gamma radiation effects on corrosion behavior will continue to be investigated.

- Data will continue to be collected under worst "plausible" conditions for each corrosion mode, including information on the nature of corrosion products to help develop an understanding of the corrosion processes in support of predictive models.

OVERALL DATA SUFFICIENCY FOR PERFORMANCE ASSESSMENT

- The revision of the BMTP will provide the link between performance assessment and the materials testing program. It also will discuss our performance assessment activities/approach. It will be available with the SCP.

- The level of detail provided in this draft of the BMTP in terms of planned test descriptions/test schedules and reporting will not be increased in future revisions to the test plan. Test instructions will be referenced in the semi-annual updates of the SCP. Test data will be made available to NRC after QA Clearance.*

- The BWIP will continue to use bounding values in addition to a best estimate wherever possible in its performance assessment efforts. This will be made clearer in the BMTP revision.

- Supporting documentation has been prepared describing the calculations used to determine the BWIP key radionuclides list and should be available by July 1984. This list will continue to be used in prioritizing the materials testing program.

* Also applies to logic/schedular diagrams comments

TEST SCHEDULE/TEST LOGIC

- The revision to the BMTP will describe the relationship of design activities to materials testing.
- Consideration will be given to revising the logic diagrams in the BMTP to reflect the interactions that occur between the various testing activities and to reflect key decision points in the testing program.

EH CONTROL IN TESTING

Statement

In many places the BMTP states that the test EH will be measured and/or controlled. However, a system Eh may not be effective in controlling radionuclide valence, Eh measurement is difficult or uncertain, and the expected redox conditions in the engineer facility or site host rocks has not been well established.

Response

BWIP recognizes the importance of oxidizing potential in establishing valence states of radionuclides and in evaluating the corrosion behavior of containment materials. We evaluate redox states in experiment where possible to assist in our interpretation of the behavior of radionuclides and other materials in the basalt system. As new analytical techniques become available for evaluating Eh, these will be incorporated into the testing program. However, our experimental approach is to allow the natural system, i.e., the fluids and solids present, to establish the appropriate oxidation potential for the experiment.

BWIP'S STATISTICAL TEST DESIGN

Comment

The BMTP does not address uncertainty analysis of data and models which will be required for licensing.

Response

Appendix A of the BMTP will be modified by adding a section on uncertainty treatment of data and models, including a description of techniques used to define probability distributions for data for use in models.

ATTACHMENT 11

DRAFT

DRAFT

FAULT TREES DEPICTING FAILURE OF HIGH-LEVEL RADIOACTIVE WASTE PACKAGES

December 1983

Prepared for

Office of Nuclear Material Safety and Safeguards

U.S. NUCLEAR REGULATORY COMMISSION

Washington, D.C.

Prepared by

Eastern Technical Division

THE AEROSPACE CORPORATION

Washington, D.C.

PREFACE

The Nuclear Regulatory Commission (NRC) must pass independent judgment on the adequacy of high-level radioactive waste package designs developed by the Department of Energy (DOE). To determine whether the packages meet the requirements of 10 CFR 60, NRC must be able to estimate the lifetime of the package and to quantify the rate of radionuclide release should a package failure occur. The program to develop this capability consists of research projects to (1) develop an understanding of the failure modes and material processes and (2) develop the analytical methodology needed to relate the research to the licensing decisions that ultimately must be made.

The Aerospace Corporation "Preparation of Engineering Analysis for High Level Waste Packages in Geologic Repositories" project is one of several that collectively will achieve these objectives. The project has four main tasks: (1) evaluation of the methodology for assessing long-term performance of high-level waste packages, (2) construction of fault trees and event trees depicting package failure and transport of radionuclides from the package, (3) assessment of the performance of the Department of Energy waste package designs, and (4) general technical assistance associated with waste package assessments. The Aerospace project covers a period of 3 years, with all four tasks to be accomplished within the first year (fiscal year 1984), specifically for a basalt repository. The same basic scope for repositories in tuff and salt formations will be covered in the remaining 2 years, concurrent with further refinement of the analytical techniques.

CONTENTS

FAULT TREES DEPICTING FAILURE OF HIGH-LEVEL RADIOACTIVE WASTE PACKAGES

INTRODUCTION

This report presents the first version of fault trees for high-level radioactive waste package failure in a basalt repository. This work is part of the overall project described in the Aerospace Program Plan (1983). The complete project will include preparation of event trees, an examination of other methods for analyzing waste package reliability, and quantitative assessment of Department of Energy waste package designs. The fault trees presented here are intended to provide a medium for discussion of the events resulting in waste package failure and a starting point in developing a methodology for quantifying waste package reliability.

The fault trees are presented in the appendix to this document. They address the failure of commercial high-level waste (CHLW) and spent fuel (SF) waste packages, in a basalt repository. This analysis is unique in that it considers the value of the barriers provided by the waste package. Other fault tree analyses conducted to date have primarily considered the radionuclide barrier provided by the geologic formation.

The reference engineering conceptual designs for CHLW and SF waste packages for basalt (Westinghouse, 1982) were used as the basis for the trees. The main barriers to radionuclide release afforded by the CHLW (waste form, canister, overpack, and packing) and SF (waste form, overpack, and packing) designs were used to structure the fault trees. The rationale behind the structure and specifics on the events resulting in barrier failure are discussed later in this report.

The fault trees presented should be viewed only as a starting point. Work remaining in refining the trees is identified in the final section of the report. As comments are received and as developmental work continues, the fault trees will be modified to incorporate new information and to correct oversight errors. This will be an iterative process that will continue at least throughout FY 1984.

On the basis of experience gained during development of the fault trees, it is believed that the technique can be an effective tool in assessing waste package performance. Fault tree analysis provides a concrete method to illustrate the processes and events that have a bearing on waste package failure. However, it is widely recognized that successful fault tree analyses require extensive review and participation by as many knowledgeable persons as possible and that there must be sufficient time for reflection, incubation, and reiteration. In this regard, NRC and its contractors should provide as much comment and participation as possible.

There is by no means unanimity in the scientific community regarding the relationships diagramed in the trees, especially in terms of the significance of individual failure modes. However, the trees provide a medium for discussion and, as they evolve, will serve to clarify the body of knowledge. For this reason, the fault trees presented would be a valuable tool even if never used to quantify waste package reliability.

GENERAL FAULT TREE/EVENT TREE METHODS

Fault Tree Versus Event Tree

This report presents the fault trees developed for waste package failure in a basalt repository. A discussion of general fault tree methodology is provided as background. In addition, event tree methodology is presented, because event trees are to be developed in the next phase of this project.

Prior Work

Fault tree analysis was introduced in 1961 to perform safety evaluations of the Minuteman missile program and has been used since then for a variety of complex systems analyses, the best known of which is the Reactor Safety Study (NRC, 1975). The technique is in widespread use today as a major tool for probabilistic risk assessment of nuclear power plants, and is a proven tool for systems analysis in a wide range of other applications. There is no real controversy with respect to the tool itself—it is a means of graphically showing the logical relationships among events and then analyzing the probability of the outcome using Boolean algebra and other accepted mathematical techniques.

Fault tree analysis is not known to have ever before been applied to the problem of high-level radioactive waste packages. The approach used in this project is to build on the existing body of fault tree knowledge and make whatever changes are necessary to accommodate any unusual characteristics of the waste package problem.

Several studies have applied fault tree methods to analyze the geologic formation as a barrier to the release of radionuclides (d'Alessandro and Bonne, 1980; Bertozzi et al., 1977; Logan and Berbano, 1977; Lee et al., 1978; Bhaskaran and McCleery, 1979). Although these studies applied to geologic formations that could act as barriers to the release of radionuclides, they have not incorporated an analysis of the waste package. In each study, the same basic approach was used—define the events, prepare the trees, and make whatever assumptions and simplifications are necessary to complete the analysis.

Fault Trees

Fault tree analysis is a technique whereby an undesired state of the system is specified (usually a state that is critical from a safety standpoint). The system is then analyzed in the context of its environment and operation to find the ways in which this undesired event can occur. The fault tree itself is a graphic model of the various parallel and sequential combinations of faults that will result in the occurrence of the predefined undesired event. The faults can be events associated with component failures, human errors, or any other pertinent event that can lead to the undesired event. The fault tree thus depicts the logical interrelationships of basic events that lead to the undesired event, which is the top event in the tree. For the purpose of this project, the top event is any release of radionuclides outside the engineered waste package (waste form, canister, overpack, and packing material).

As discussed in the NRC "Fault Tree Handbook" (NRC, 1981), it is important to realize that a fault tree is not a model of all possible system failures or all possible causes. A fault tree includes only those faults that contribute to the top event. Therefore, a fault tree includes only the faults assessed by the analyst as being pertinent. As the systems and processes are better understood over a period of time, additional information can be screened for possible incorporation into or deletion from segments of the tree. New information may, however, confirm that an existing tree sufficiently covers the pertinent faults and precursors.

Minimal cut sets are the smallest set of primary events, inhibit conditions, or undeveloped fault events or any combination of these that must occur in order for the top event to occur. The fault tree work has shown that when the minimal cut sets are determined, the result can be used to simplify the rest of the analysis. .Note that the cut sets are functions only of the logic structures and are independent of the event probabilities. This feature means that once the cut sets have been found, the investigation can concentrate primarily on those portions of the tree that are most relevant to the top event.

Persons desiring background information on fault tree analysis in general should consult references such as McCormick, 1981; Reilly, 1978; and Larsen, 1974. They describe the technique and how to apply it.

Event Trees

Fault trees are developed by starting with an end event as the top of the tree and working backward through the precursor events. Event trees, on the other hand, begin with a defined initiating event and then examine the consequences of the event, the factors influencing mitigation of its effects, and the results of the sequence of events. Figure 1 depicts a sample event tree with an initiating event and two safety systems, the successful operation of which will mitigate the effects of the initial event. The operation

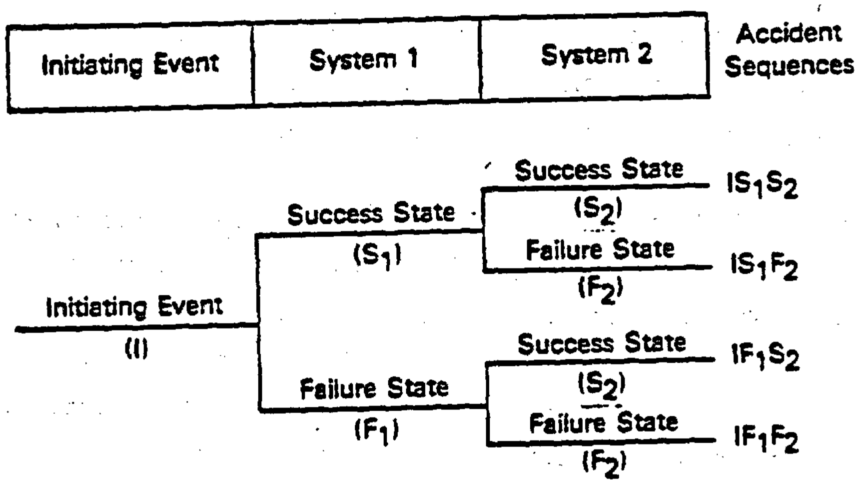


Figure 1. Sample Event Tree (NRC, 1976)

of the safety systems is described as either a success or a failure. The resulting accident sequence is thus identified by the possible paths that can be taken. For a situation in which the safety system can fail partially, but not necessarily totally, the success and failure states can have more branches, with each representing a specifically defined type of failure (McCormick, 1981).

Event trees provide a means to define particular degrees of failure and to assess the resulting radionuclide releases. Event trees also provide an easily understood way to display particular failure sequences.

In summary, fault trees and event trees, when used in concert, are valuable tools for analyzing system safety problems. There is sufficient diversity in each technique to provide a check on the results generated using the other technique. Although fault trees and event trees may be effective for analyzing waste package performance, they are certainly not the only techniques to be used. Later in this project other methods will be examined and compared with fault trees and event trees to ascertain an appropriate combination of tools for waste package analysis.

FAULT TREES FOR BASALT WASTE PACKAGES

Approach

This section of the report presents the initial version of the fault trees prepared to depict waste package failure. Text discussions regarding how the trees were developed and the sources of information that serve as their basis are provided.

Early in the project it became apparent that there is a natural division with respect to what work should be described by fault trees and by event trees. Because 10 CFR 60 is oriented toward preventing any radionuclide release during the containment period (e.g., at least 1000 years), it is logical to use fault trees to depict waste package failure, where failure is defined as any release of radionuclides by the waste package to the basalt repository. The fault trees

trees depict the failure mechanisms over whatever time is required for failure. The event trees depict the radionuclide releases associated with particular failure sequences.

The fault trees have been structured to show the significant items that will affect the likelihood of package failure, whether it occurs before or after 1000 years following repository closure. At any point in the lifetime of a package (before or after failure), the probability of any event in the trees occurring may vary. When the failure mechanisms illustrated by the trees are quantified, the intitial strategy probably will be to choose a time interval after waste package emplacement (e.g., 50 years), apply the probabilities believed to represent the conditions that have transpired up to that time, and calculate the probability of the top event--waste package failure to contain radionuclides. The process will be repeated for succeeding time periods until package lifetime is sufficiently characterized.

Similarly, the event trees and their associated probabilities will be used to calculate radionuclide releases. The radionuclide release, decay, and transport models used by the nuclear industry can assist in quantification, but a decision has not yet been made regarding whether to attempt integrating such models into the fault tree/event tree codes. It may be more desirable to perform calculations separately and apply the results to the overall analysis.

Reference Package Designs

For use with the fault trees, the reference engineering conceptual designs for CHLW and SF waste packages for basalt are shown schematically in Figures 2 and 3, respectively. The schematics define the components of the engineering packages and were taken from the reference designs presented by Westinghouse (1982). The CHLW form resembles the borosilicate glass form developed for fuel reprocessing at Barnwell, South Carolina, and is for waste resulting from reprocessing of spent fuel that originally contained only uranium. The CHLW form is poured, while molten, into stainless steel canisters. The canisters, filled to about 85 percent capacity, facilitate handling and interim storage. A steel overpack is required for repository containment and is surrounded

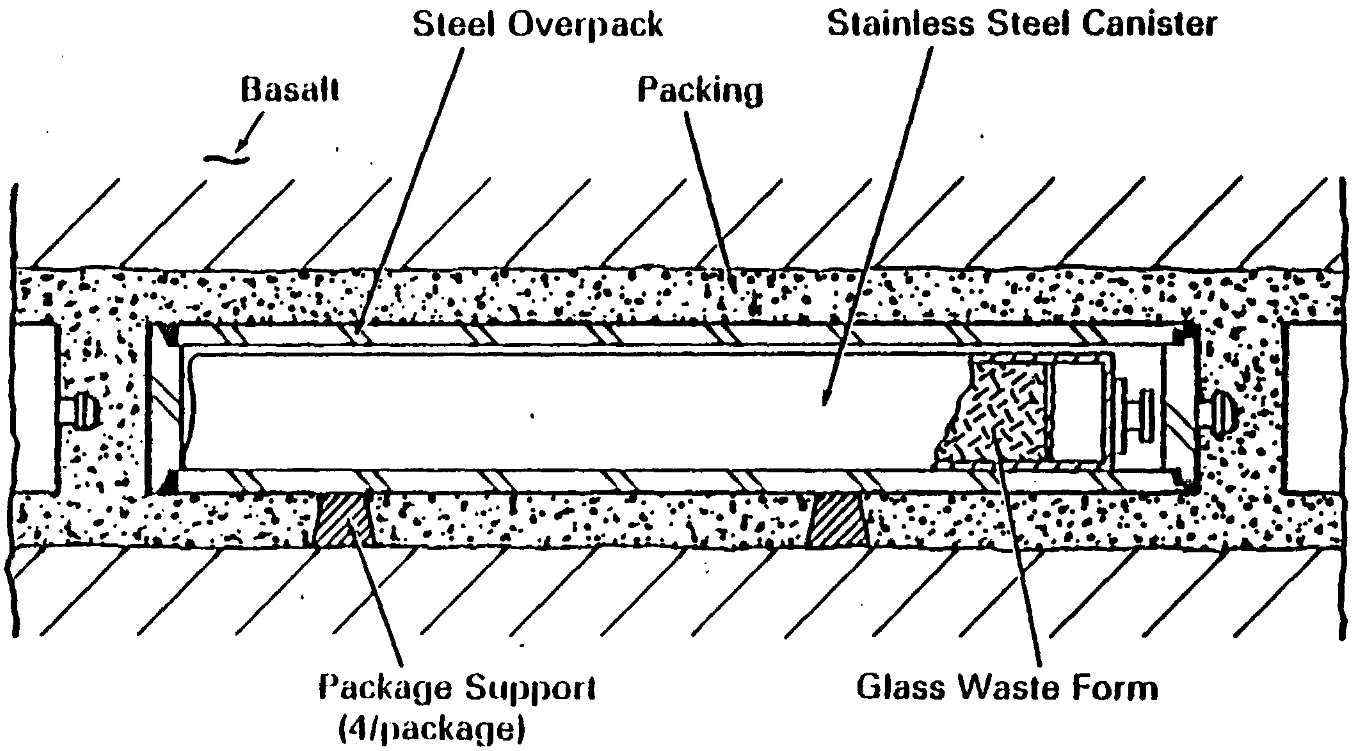


Figure 2. Reference Waste Package Conceptual Design for Commercial High Level Waste in Basalt (Westinghouse, 1982)

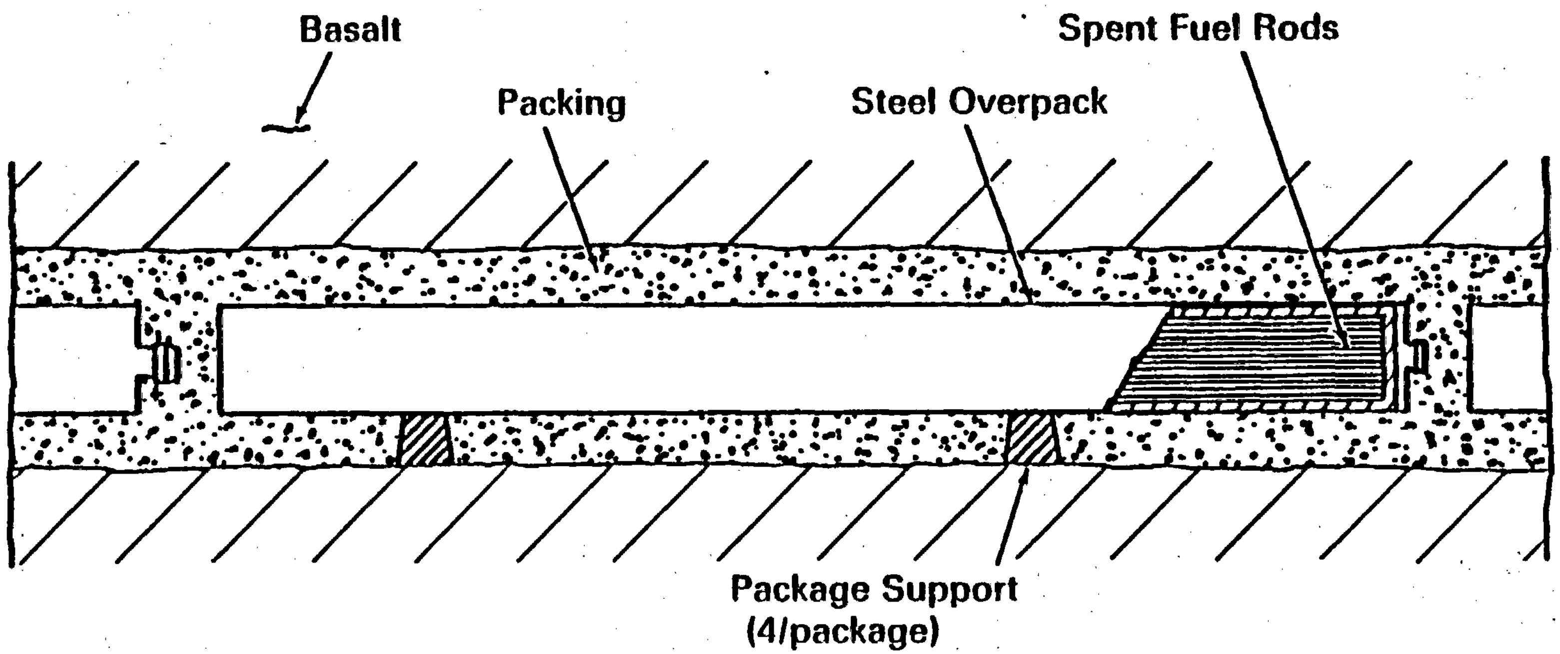


Figure 3. Reference Waste Package Conceptual Design for Spent Fuel in Basalt (Westinghouse, 1982)

surrounded by a mixture of bentonite clay and crushed basalt (25/75 percent by weight), sized to be suitable for pneumatic emplacement in reference horizontal boreholes. Therefore, the CHLW package provides four barriers to radionuclide release (waste form, canister, overpack, and packing).

The spent fuel form consists of spent fuel rods (fuel pellets encased in Zircaloy cladding) removed from intact assemblies and consolidated into a closely packed array. In the reference conceptual design, the numbers of rods per waste package is to be 792 pressurized water reactor rods (3 assemblies) or 441 boiling water reactor rods (7 assemblies). The steel container used as the overpack for the consolidated rods will be designed for the 1000-year containment time. A pressurized water reactor fuel rod is shown schematically in Figure 4 (Woodley, 1983) to show the components in a typical configuration for use with the spent fuel fault trees (specifically Figure B-2). Note that a canister is not used for the SF form as in the CHLW. Thus, the SF waste package only provides three barriers to radionuclide release (fuel waste form, overpack, and packing).

The waste package designs for basalt are, at the time of this report, in the conceptual stage with numerous backup alternatives and placement options still under serious consideration. As these designs mature and proceed into the preliminary and detailed design stages, adjustments will be made to the fault trees to reflect current design status.

Terminology

The NRC "Fault Tree Handbook" (1981) was used as a guide in developing the fault trees presented in this report. As discussed previously, a fault tree is a qualitative model of the events resulting in system failure. The trees diagram the logic of the failure by using a hierarchy of "gates" and events. The gates permit or inhibit the passage of fault logic up the tree and show the relationships of events needed for the occurrence of "higher" events. A "higher" event is the "output" of the gate; "lower" events are the "input" to the gate.

A set of symbols (Table 1) is used in the fault trees to graphically represent the types of relationships between input and output events. The symbols

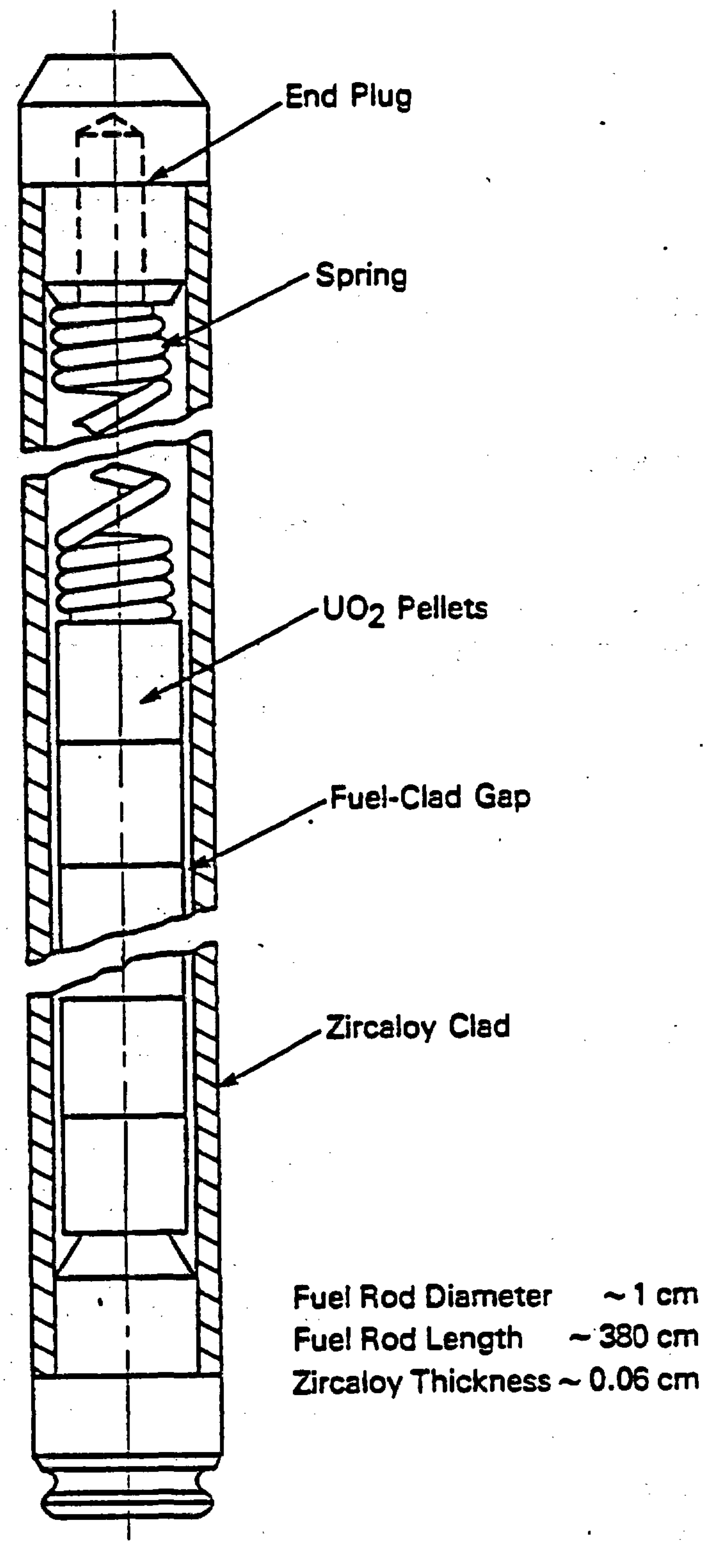
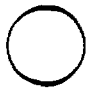
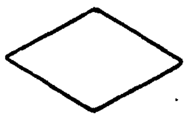
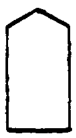


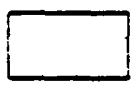
Figure 4. Typical PWR Individual Spent Fuel Rod (Woodley, 1983)

Table 1. Fault Tree Symbols (NRC, 1981)

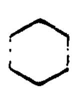
Primary Event Symbols

	BASIC EVENT	A basic initiating fault requiring no further development
	CONDITIONING EVENT	Specific conditions or restrictions that apply to any logic gate (used primarily with PRIORITY AND and INHIBIT gates)
	UNDEVELOPED EVENT	An event that is not further developed either because it is of insufficient consequence or because information is unavailable
	EXTERNAL EVENT	An event that is normally expected to occur; displays events that are not, of themselves, faults

Intermediate Event Symbols

	INTERMEDIATE EVENT	A fault event that occurs because of one or more antecedent causes acting through logic gates

Gate Symbols

	AND	Output fault occurs if all of the input faults occur
	OR	Output fault occurs if at least one of the input faults occurs
	EXCLUSIVE OR	Output fault occurs if exactly one of the input faults occurs
	PRIORITY AND	Output fault occurs if all of the input faults occur in a specific sequence (the sequence is represented by a CONDITIONING EVENT drawn to the right of the gate)
	INHIBIT	Output fault occurs if the (single) input fault occurs in the presence of an enabling condition (the enabling condition is represented by a CONDITIONING EVENT drawn to the right of the gate)

Transfer Symbols

	TRANSFER IN	Indicates that the tree is developed further at the occurrence of the corresponding TRANSFER OUT (e.g., on another page)
	TRANSFER OUT	Indicates that this portion of the tree must be attached at the corresponding TRANSFER IN

generally used in fault tree analysis are in four categories: primary event, intermediate event, gate, and transfer.

Primary events are those that have not been developed further and that will have probabilities of occurrence assigned when the model is quantified. The primary events used most frequently in the appendix are the basic (circle) and undeveloped (diamond) events. A basic event is that which requires no further development; thus, a circle represents the lowest level to which a failure can be taken. Undeveloped events have not been developed further either because the event is considered insignificant or because sufficient information is not available to describe the basic input events. As the project progresses, the intent is to remove as many of the diamonds as possible, either by adding more detail or by developing a consensus that the event does not merit further attention.

Intermediate events, represented by rectangles, occur because of one or more preceding events. Logic gates are used to identify the relationships between the antecedent and intermediate events. The two basic types of logic gates are the OR and the AND gates. OR gates are used to show that the output events occur only if one or more of the input events occur. It is important to note that ". . . causality never passes through an OR gate. That is, for an OR gate, the input faults are never the causes of the output fault. Inputs to an OR gate are identical to the output but are more specific as to how the output occurs" (NRC, 1981). An AND gate shows that the output event occurs only when all of the input events exist. Therefore, an AND gate does show a causal relationship between input and output.

Transfer symbols can be used to avoid duplication in the figures, thus simplifying the trees; they are indicative of system interfaces. Additionally, it is frequently not feasible to show an entire fault tree on a single sheet of paper, so transfer symbols can be used to divide system trees into subtrees. A numbering convention, using a five character code identifier placed inside the transfer symbols, has been implemented in this report. This numbering system enables the user to maintain a record of transfers and is represented by the following format:

X	XXXX
alphabetic character represents system	four numeric characters identify the subtree

Table 2 identifies the alphabetic characters used to identify the systems where each system is representative of a single package barrier. The numeric characters used to identify the subtrees start at 10000 and are incremented by one for each additional subtree.

Table 2. System Identification Code

Code	System
A	Glass Waste Form
B	Stainless Steel Canister
C	Steel Overpack
D	Packing (bentonite/basalt)
E	Spent Fuel Waste Form

Structure of the Trees

The overall structure of the fault trees has two parts: one set of trees for CHLW (Appendix, Part A) and a separate set of trees for spent fuel (Appendix, Part B). Although there are items common to both types of waste packages, separate presentation reduces the potential for confusion. Within each of these two sets of trees, the basic structure centers on the main barriers (waste form, canister, etc.) provided by the reference package designs.

The next lower level makes the distinction between aqueous and nonaqueous conditions. Unless otherwise stated, for the packing material, aqueous conditions equate to any conditions partially to fully saturated, and nonaqueous conditions represent the nonsaturated case. For the other parts of the waste package, nonaqueous or no-groundwater conditions were assumed unless the next outer barrier had been breached.

The next level of detail in the trees deals with the mechanisms for releasing radionuclides that travel through a breach. Finally, the level of detail is carried down to the point at which the very basic events (depicted by circles) are identified or to the point at which the analysis is stopped at an event that is undeveloped (depicted by diamonds). However, it should be recognized that the lower an event is on the tree, the less is its overall impact. An exception would be a single event that occurs in several places in the lower parts of the tree (e.g., presence of water).

One important observation regarding the fault tree structure is that the trees do not depict (from the bottom up) the exact order in which events occur. (That would not be practical, because some events cycle or occur repeatedly.) Rather, the fault trees show the events that must have transpired for higher level events to occur. The event trees will be used to illustrate important event sequences and will be based on specific scenarios chosen from the myriad of possible paths illustrated by the fault trees.

Fault Tree Discussions

Packing

The reference waste package design (Westinghouse, 1982) was used as the basis for the development of the fault trees for packing failure. The conceptual design uses a 25 percent bentonite/75 percent crushed basalt (by weight) packing mixture to provide an additional barrier between the waste form and basalt host rock. The packing primarily serves to (1) control the groundwater flow both to and from the waste form and (2) retard the migration of radionuclides (BNL, 1983c). The packing is also designed to chemically modify or buffer the groundwater, provide a mechanical stress barrier, and maintain good heat transfer from the container to the host rock. Failure of the packing to provide these functions may result in the alteration of the physical, chemical, and mechanical properties of the packing and therefore can affect its ability to control groundwater flow and radionuclide release.

Information on the mechanisms contributing to the failure of the packing to meet its major objectives was provided by Oak Ridge National Laboratory

(ORNL) (1983) and supplemented by the work of Brookhaven National Laboratory (BNL) (BNL, 1983c; Davis and Schweitzer, 1982; Bida and Eastwood, 1983; BNL 1983b; BNL, 1983a). The Oak Ridge report (ORNL, 1983) assumed that the ". . . waste package will not be subjected to catastrophic events of low probability . . . ," and dropped these as possible failure modes. The portion of the fault tree depicting the aqueous transport of radionuclides through the packing (Figure A-22) does not specifically identify failure resulting from events such as meteorite impact, volcanic eruption, tunnel collapse, and tectonic activity. However, because it may be possible to minimize the probability of failure of the engineered barrier by "over-engineering" the design, these catastrophic events may become more critical. It may be necessary to incorporate these unanticipated events in future iterations of the trees. Figure A-25 addresses the nonaqueous transport of radionuclides as solids, liquids, or gases. The concept of catastrophic events could also be incorporated as lower events resulting in nonaqueous transport.

The literature reviewed focuses on the movement of radionuclides as species dissolved in the groundwater. Therefore, processes that fail to control this method of transport provide the primary structure for the packing portion of the tree. The aqueous transport of radionuclides as insoluble, fine particulates and as gases were also identified as possible means of exceeding the release criteria. However, the events resulting in the release of radionuclides as particulates and gases were not developed in detail during this phase of the analysis because information on these modes of transport was not available.

For groundwater transport of radionuclides to occur, it was assumed that the packing is saturated and the radionuclides are dissolved in the water. Failure of the packing to completely retain the radionuclides by sorption contributes to their presence in the groundwater. Using the BNL and ORNL documentation, the events or processes resulting in reduced sorption capabilities by chemical poisoning, mineral alteration, and selective dissolution and leaching have been diagramed (Figure A-22). These events are a function of the chemical and physical conditions present in the repository (i.e., temperature, water chemistry, pressure, and radiation). However, the causes of and the relationships between the conditions are not well understood, and therefore, the fault trees were not expanded to include them. Research efforts are under way, and additional work is needed to define the sorptive properties of the basalt and bentonite and when possible to identify the interrelationships among the environmental conditions (BNL, 1983c). Additionally, the interactive effects between the packing and other package components must also be considered.

In addition to retarding the movement of radionuclides by sorption mechanisms, the packing is also designed to act as a water controlling barrier that limits movement to diffusion of the dissolved species (Westinghouse, 1982). Darcy's law for one-dimensional flow in a homogeneous, isotropic medium has been used as a basis for identifying the factors to be considered (BNL, 1983b):

$$Q = -K\ A\ (dh/dl)$$

where Q = flow rate, m^3/s;
K = hydraulic conductivity, m/s;
dh/dl = hydraulic gradient, dimensionless; and
A = cross-sectional area, m^2.

For diffusion to be the principal mechanism of transport through the clay mixture, the water movement is controlled by the hydraulic conductivity of the packing and the hydraulic gradient. A hydraulic conductivity of 10^{-7} m/s and regional hydraulic gradient of 10^{-3} have been used together as the basis for the reference waste design (Westinghouse, 1982). Therefore, events resulting in changes to the hydraulic gradient and conductivity should be considered when evaluating the failure of the packing to act as a barrier. Figures A-23 and A-24 apply to these concerns.

Hydraulic conductivity is a function of both the properties of the material and the properties of the fluid (Chow, 1964):

$$K = Cd^2 \rho g/\mu$$

where K = hydraulic conductivity, m/s;
C = constant of proportionality, dimensionless;

d = representative pore diameter, cm;
ρ = fluid density, kg/m^3;
g = gravitational constant, m/s^2; and
μ = dynamic viscosity of fluid, kg/m-s.

In the literature, the emphasis tends to be placed on the properties of the packing material and thus has been carried through in developing the trees. At this time, events resulting in changes in the fluid characteristics have not been evaluated in detail.

In considering the characteristics of the packing, the specific permeability (k) has been defined as (Chow, 1964)

$$k = Cd^2$$

Generally, a ". . . material is considered permeable if it contains interconnected pores, cracks, or other passageways through which water can flow" (Cedergren, 1967). The constant of proportionality, C, is a function of the medium that includes packing, porosity, grain-size distribution, and shape. Therefore, any changes to the material that might alter these characteristics may influence its ability to act as a filtering medium and barrier to water movement. In addition, because porosity is affected by the degree of cementation and compaction of the packing materials, and the presence of solution openings, joints, or fractures, these factors must also be considered. As a result, the fault tree has been developed to include the impact of such things as wet/dry cycling, hydrologic erosion, leaching, dehydration, and mineral alteration on the characteristics of the packing. The physical and chemical conditions influencing the packing and fluid characteristics (i.e., pH, Eh, temperature) and the interactions between them have not been detailed in this phase of the analysis.

Overpack

The overpack is a steel barrier used for encapsulating either (1) a canister with its glass matrix waste form CHLW or (2) a packet of spent fuel rods. The dimensions of the overpack for the CHLW containment package differ from those of the SF containment package (see Westinghouse, 1982). Consideration was given only to the reference waste package design for the CHLW and SF overpacks (and not to other design alternatives), which currently favor the use of carbon steel to provide positive containment for 1000 years. The overpack forms a barrier that should prevent penetration of groundwater and other chemicals to the canister zone in the CHLW overpack or to the SF (Zircaloy cladding and fuel pellets) in the SF overpack and should also prevent the transport of radionuclides to the packing. The overpack must resist being breached by corrosion, mechanical mechanisms, and radiation.

Failure of the overpack can occur in either saturated or nonsaturated packing conditions. These situations are represented in Figure A-2 for the CHLW storage package and in Figure B-2 for the SF storage package. It is recognized that the packing (bentonite and basalt) around the overpack is never completely dry, so some moisture, whether in the liquid or gaseous (steam) state, will always surround the outer surface of the overpack. It appears that the effect on the overpack from saturated versus nonsaturated packing differs with regard to the potential for overpack corrosion and mechanical failure.

The saturation condition of the packing also affects the mode of transport of radionuclides though an overpack breach to the packing. If a CHLW overpack is in a saturated packing situation, water would enter the breached CHLW overpack and allow available radionuclides (those that had escaped the canister) inside the overpack to migrate through the breach. The water also could corrode and eventually breach the canister and reach radionuclides. In the nonsaturated packing, water is less likely to flow into a breached overpack, so the flow of radionuclides out of the overpack and corrosion of the canister are less probable than for the saturated situation.

For an SF overpack in the saturated packing situation, water would enter the SF overpack and immediately attack the spent fuel, eventually allowing radionuclides to flow out of the overpack. In the nonsaturated packing situation, water is less likely to enter a breached overpack; consequently, the flow of radionuclides to the packing is less probabilistic than in the saturated situation.

In structuring many of the fault tree branches, Westinghouse (1982), BNL (1983c), Stahl and Miller (1983), Ahn and Soo (1982), Claiborne (1983), and Siskind (1983) were used as sources for failure mechanisms used in the fault trees.

Before radionuclides can flow through the overpack wall, two important events need to occur: (1) a breach must exist in the overpack wall and (2) radionuclides must be in a position to flow through the wall as indicated in Figures A-14 and A-19 for the CHLW and Figures B-7 and B-12 for SF. These figures note mechanisms by which the overpack can be sufficiently breached in order to allow radionuclides to flow through the overpack wall. Mechanisms for breaching the overpack include (1) placing lithostatic forces on the overpack, (2) degrading the overpack in either a saturated or nonsaturated packing, or (3) subjecting the overpack to a catastrophic event such as a volcanic eruption with a lava flow through the package zone or a meteorite penetrating the package zone. An earthquake scenario has been included as a mechanism for crushng or severing the overpack. Catastrophic mechanisms will not be pursued further in the current effort and will be expanded in future iterations only to the degree warranted relative to other fault event probabilities.

Both the C1000 (Figure A-14) and C2000 (Figure A-19) fault tree branches of the CHLW and the C3000 (Figure B-7) and C4000 (Figure B-12) branches of the SF fault tree, which cover aqueous and nonaqueous conditions, respectively, include overpack breach mechanisms by mechanical, chemical, and other means. The degradation and breach mechanisms cited in the BNL and ORNL references (BNL, 1983c; Ahn and Soo, 1982; Siskind, 1983; and Claiborne, 1983) are among the lower events shown in Figures A-15, A-16, A-17, A-18, A-20, and A-21 for CHLW and Figures B-8, B-9, B-10, B-11, B 13, and B-14 for SF. The interaction of degradation mechanisms, such as pitting corrosion, ductile rupture, and thermal enchancement, will be developed further, where warranted, in future iterations of the fault tree. Thermal interactions with corrosion have been incorporated into the tree.

The overpack can also be breached due to radiation and human intrusion into the package after repository closure, overpack fabrication deficiencies, and handling abuses. These mechanisms will be further enhanced as the fault trees are developed. Underground explosions, oil and water drilling, tunneling, and other human activities can damage packages in the repository after closure. All these events are considered to be part of the event categorized as "Human Intrusion After Closure" on the last cited group of figures.

Another branch in CHLW subtree C1000 (Figure A-14) considers how the radionuclides would be available to flow through a breach. For CHLW subtree C2000 (Figure A-19), there are likewise corresponding breach and available radionuclide flow branches. Subtrees C3000 (Figure B-7) and C4000 (Figure B-12) of the SF are similar to C1000 and C2000, respectively. It was postulated that the radionuclides could be in gaseous, liquid, or solid states and could also be available to flow through a breach. It is assumed that if gaseous radionuclides are available, they can flow through any adequate breach to which they can gain access. Radionuclides in the liquid or solid/granular state in a nonaqueous situation could flow through a breach only in certain geometric configurations, as presented in Figure A-19 and Figure B-12. If water floods through an overpack breach, as considered in an aqueous situation, then radionuclides possibly can be released by several mechanisms, including water reaction with radionuclides already out of the canister or breaching of the canister with subsequent release of radionuclides in the CHLW case, or with the Zircaloy cladding and SF radionuclides in the SF case.

When radionuclides are in a solid or liquid state, water can either suspend the radionuclides in a mixture and transport the radionuclides through Brownian motion or convection of the water, or the radionuclides could be dissolved by the water and the ions then transported through the overpack breach. After radionuclides are transported through the overpack, the problem becomes one in which the packing acts as the final barrier to radionuclide transport to the basalt.

Canister

The canister is the barrier that contacts and encloses the glass waste form and is enclosed in the overpack in the CHLW package, as described by

Westinghouse (1982). The canister is fabricated from Type 304L stainless steel. The canister serves as a barrier to prevent penetration of groundwater and other chemicals to the glass waste form zone and to prevent transport of radionuclides to the overpack. The canister also reduces the radiation intensity to the overpack and packing used in the CHLW package. A canister is used only with the CHLW package, not in the SF package.

Failure of the canister can occur in the presence of an aqueous medium that has entered the overpack and consequently contacts the canister, or the canister can deteriorate when an aqueous medium is not present as indicated in Figures A-6, A-11, B-7, and B-12. This latter case is designated as nonaqueous transport, but only implies that water has not entered the overpack cavity. Any water trapped in the cavity between the canister and overpack or inside the canister with the glass at the time of repository closure is classified as being in the nonaqueous category. A major concern with water is transporting radionuclides dissolved in an aqueous solution or radionuclide granules suspended in the aqueous solution.

Many of the branches of the fault tree for the canister were developed based on the mechanisms cited in Westinghouse (1982), BNL (1983c), Stahl and Miller (1983), Ahn and Soo (1982), Claiborne (1983), and Siskind (1983).

Two important events need to occur before radionuclides flow through the canister wall: (1) a breach must exist in the canister wall and (2) the radionuclides must be in a position and state to flow through the breach as seen in subtrees B1000 and B2000, Figures A-6 and A-11, respectively. The various mechanisms by which the canister can be sufficiently breached such that radionuclides flow through the canister wall are shown. These include a path for breaching by (1) crushing the canister with the overpack, (2) degradation with water (aqueous solution) of the canister, or (3) a catastrophic event, such as discussed for the overpack.

In both the aqueous and nonaqueous cases, shown in Figures A-6 and A-11, the canister breach mechanism by mechanical, chemical, and other modes was persued. The mechanisms for degradation and breach cited in the ORNL and BNL references are shown as bottom events in Figures A-7, A-8, A-9, A-10, A-12, and A-13. These events may be developed further to show interactions already discussed for the overpack.

In addition to considering a breach in the canister, consideration was given to the manner in which radionuclides can escape. It was postulated that the radionuclides could be in a gaseous, liquid, or solid state and available to flow through a breach. If gaseous radionuclides are available, it is assumed that they can flow through any adequate breach to which they can gain access. Because of the limitation created by gravity, liquefied or solid/granular radionuclides would require the breach to occur at a canister level below the upper surface of the radionuclides in order for radionuclides to flow through a breach in a nonaqueous situation (see Figure A-11). In a situation in which water floods (aqueous situation) the canister to the breach level, there are many additional mechanisms for transporting the radionuclides through the breach (see Figure A-6).

In situations where water is able to transport radionuclides, it is suggested that the radionuclides could be freed from the glass matrix and ready for transport either (1) prior to a canister breach, and then further mobilized by water after a canister breach, or (2) after a canister breach through the action of water releasing the radionuclides from the glass matrix.

Once radionuclides are transported through the canister, the problem is considered as one relating to the overpack, whether or not the overpack was previously breached.

Glass Waste Form

Figures A-3 and A-4 of the tree focus on the aqueous transport of radionuclides from the glass waste form to the canister. The release of radionuclides from the glass to the groundwater is generally considered to occur through leaching and dissolution. The concept of radionuclides transported as particulates in suspension is shown as an undeveloped event. As information becomes available on the events resulting in the transport of radionuclides as solids, the tree will be modified. Figure A-5 addresses the concept of nonaqueous methods of mobilizing the radionuclides from the glass to the canister; however, this branch has not been developed completely.

The information on the events contributing to dissolution and leaching, and hence the failure of the glass to completely contain the radionuclides, was provided by ORNL (1983) and Battelle Columbus Laboratory (Stahl and Miller, 1983). The influence of pH has been shown to affect dissolution and leaching where, "for pHs greater than approximately 9, congruent dissolution occurs. For pHs less than 9, selective leaching is the dominant mechanism" (Stahl and Miller, 1983). For this reason, the mechanisms of dissolution and leaching were separated into two branches of the tree; however, the events leading up to each are essentially the same.

For leaching and dissolution of the glass matrix and for removal of the radionuclides to occur, the groundwater must be in contact with the waste form, and the environmental conditions necessary for removal must be present. In addition, the radionuclides must be soluble in water; thus, the initial composition of the waste form is important. At this point in the analysis, the required physical and chemical conditions have not been presented in the tree.

The characteristics of glass also influence its performance as a barrier to radionuclide release. Increases in the surface-area-to-volume ratio results in higher concentrations of species in solution. Changes in the chemical composition of the glass, as the chemical and physical environment changes, may also contribute to leaching and dissolution, although this has not been verified. Devitrification, hydration, and radiation-induced microfracturing have been identified as processes that contribute to increased surface area. Of these, devitrification is considered the most significant (ORNL, 1983). Devitrification is primarily temperature dependent with recrystallization limited at temperatures below 500°C (ORNL, 1983). The amount and identity of the crystals are also dependent on the original composition of the glass (ORNL, 1983). When the volume fraction of crystals formed by devitrification ". . . approaches approximately 10 percent, additional cracking of the monolithic waste form may occur" (Stahl and Miller, 1983). The formation of crystals will also provide grain boundaries to act as additional surfaces for leaching.

Spent Fuel Waste Form

Historically in waste repository studies, spent fuel has not been credited as a barrier against radionuclide release. However, there is reason to believe that, at least to some degree, the Zircaloy cladding and perhaps the fuel matrix itself would prevent or retard release. The fault tree addresses the aqueous and nonaqueous degradation methods of the Zircaloy cladding and UO_2 pellets in Figures B-3 and B-6.

Breach of the cladding could occur either prior to emplacement (from reactor operation or from handling) or through a degradation process after emplacement. As shown in the fault tree, pre-emplacement clad defects might be discovered by inspection. This could be important if the assumption of little or no defects is essential to the credit claimed for the cladding as a barrier. In recent years, the failure rate of fuel rods has been quite low. One study performed for the basalt repository effort (Claiborne, 1983) cited a failure rate less than 0.01 percent. Such failed rods could be treated as special cases. Additionally, the possibility of failure of the cladding by mechanical means, such as might result from loads exceeding the design loads, has also been incorporated in the tree.

Cladding degradation after emplacement in the repository has also been examined (Claiborne, 1983). As shown in Figure B-3, two degradation mechanisms that occur under aqueous conditions are hydrogen embrittlement and stress corrosion cracking. Hydrogen embrittlement can occur as a hydride phase forms subsequent to saturation with hydrogen at high temperatures. Stress corrosion cracking of the cladding can occur if chloride solutions are present and the free corrosion potential is exceeded.

In addition to the cladding, the UO_2 fuel matrix itself provides a barrier to the release of radionuclides. As shown in Figure B-4, fuel that has reached a temperature in the reactor greater than approximately 1000°C can result in migration of fission products to the gap between the fuel pellets and the cladding. If the cladding is breached in such fuel by either aqueous or nonaqueous means, radionuclides could be directly released. There are, however, ways of identifying such fuel, either by the recorded operating history or by testing. If detected, the fuel could be treated as a special case and, therefore, does not appear in the tree. This high-temperature release phenomenon is discussed in Woodley (1983). The conclusion is that the majority of fuel rods will not have been operated at this high temperature.

If the fuel matrix has not released the fission products directly to the gap, then there is the possibility that the matrix could be disrupted, perhaps over time, by either extensive oxidation or dissolution of the matrix (leaching). These items are discussed by Woodley (1983) and Claiborne (1983). Existing information seems to indicate that, at least in principle, the applicant for a repository license may be able to justify the fuel matrix as either a partial barrier or a delayed release mechanism. On the other hand, work such as the Canadian leaching study cited in Claiborne (1983) indicates that under certain conditions, leaching of some radionuclides may be relatively rapid (e.g., 4 percent of the cesium was known to leach in a few days). Mechanical failure of the matrix that would place radionuclides physically in conjunction with the overpack and mechanisms resulting in radionuclides as solids or colloids suspended in water have not been discussed by ORNL (1983). However, because the fault trees are intended to provide a basis for discussion, these failure modes have been treated as undeveloped events at this point in the analysis.

In summary, research on cladding failures and the UO_2 matrix itself is not yet definitive. As new information develops, the fault trees can be modified accordingly. If there is not a significant increase in information by the time the fault trees are quantified, the best available data will be used even if they are conservative.

WORK TO BE DONE

There are several areas related to the fault trees that will require further examination and refinement. As can be seen from the fault trees, there are a number of events that ideally should be expanded. The approach will be to continue research using literature, specialists, and comments from reviewers to expand and refine these events.

A second area meriting further consideration is the question of synergistic interactions among the failure modes. To the extent possible in this first round of fault trees, relationships involving more than one failure mode have been diagramed. However, the literature is almost totally devoid of information that would enable the structuring of specific mechanisms that would result in synergism. Efforts to develop this information will be continued.

A third area of interest is the issue of data for quantification of the trees. Although quantification is not scheduled until later in the fiscal year, an exploratory search to see whether sufficient information will be available has begun. The initial finding was that although many researchers have worked in the areas of interest, the results in general have not been reported in probabilistic form. Consequently, the initial quantification may require considerable approximation and individual judgments. As better information becomes available, the analysis can be refined.

In addition to the above items, a number of analytical areas involved in the mathematics of fault tree and event tree analysis and quantification are being explored. Items such as kinetic tree theory and phased mission analysis are under examination. This effort will help optimize the quantification of the trees.

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DRAFT

FAULT TREES DEPICTING FAILURE OF HIGH-LEVEL RADIOACTIVE WASTE PACKAGES

APPENDIX FAULT TREES

DRAFT

December 1983

DRAFT

Prepared for
Office of Nuclear Material Safety and Safeguards
U.S. NUCLEAR REGULATORY COMMISSION
Washington, D.C.

Prepared by
Eastern Technical Division
THE AEROSPACE CORPORATION
Washington, D.C.

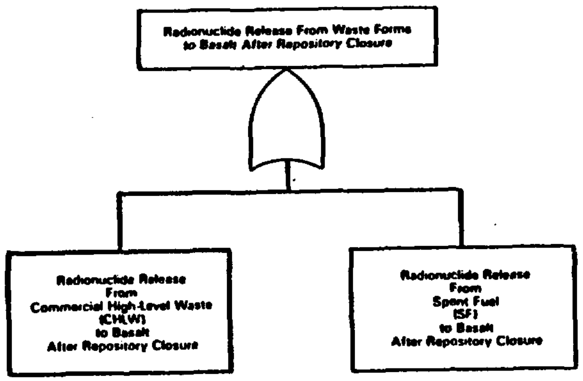

Figure A-1. Top Events for Radionuclide Release From Waste Forms
to Basalt After Repository Closure

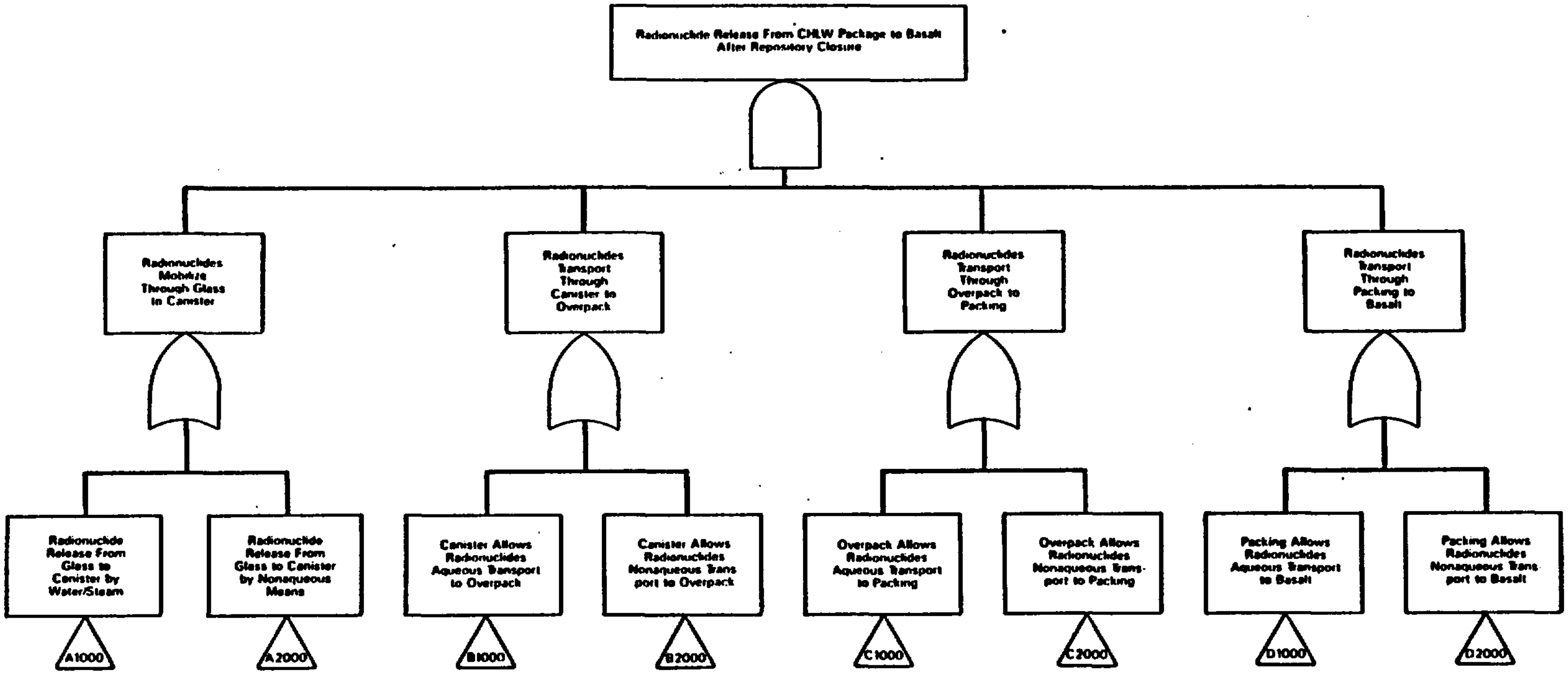

Figure A-2. Top Event Fault Tree for Radionuclide Release From Commercial High-Level Waste Package to Basalt After Repository Closure

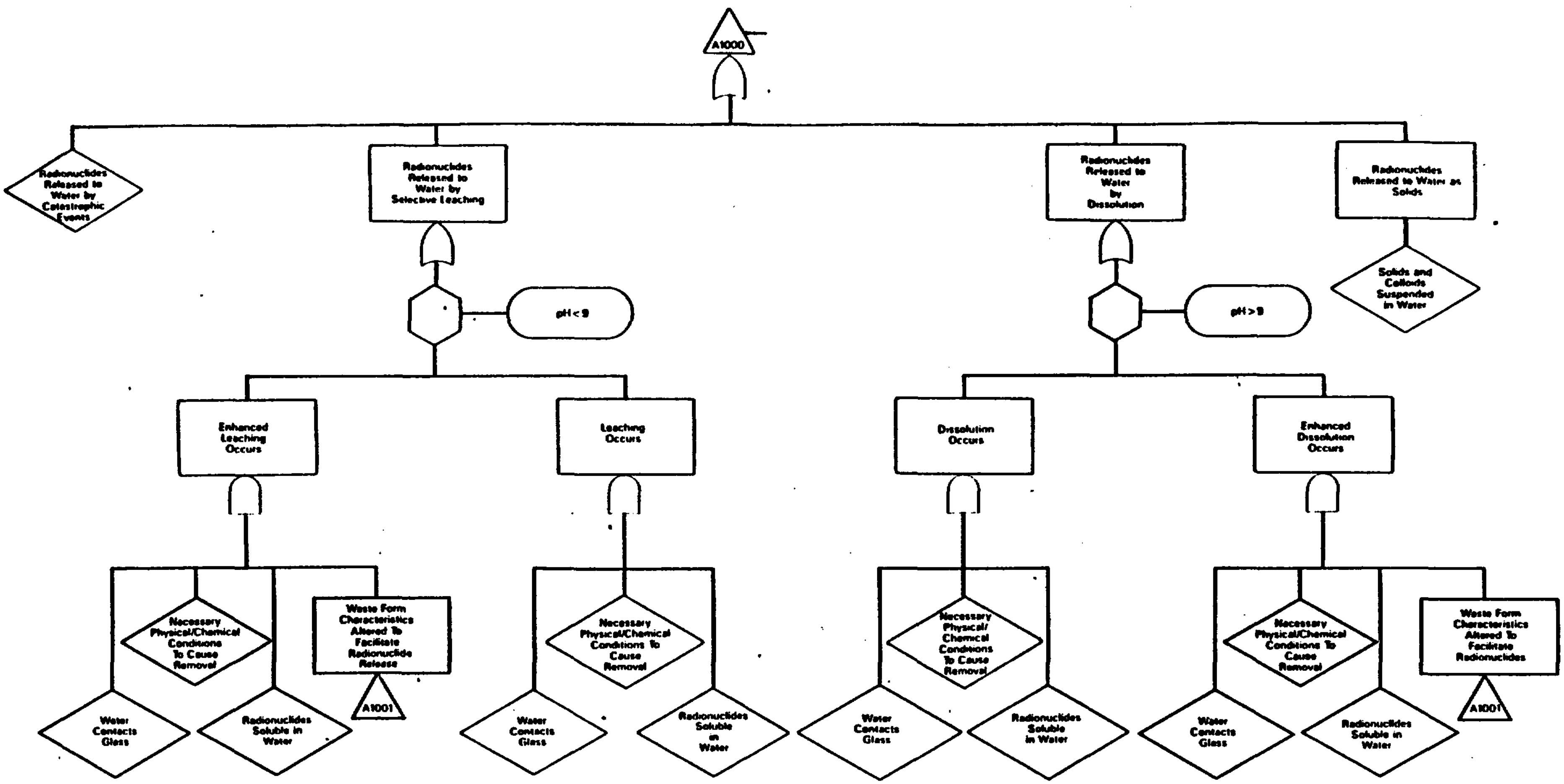

Figure A-3. A1000 – Radionuclide Release From Glass to Canister by Water/Steam

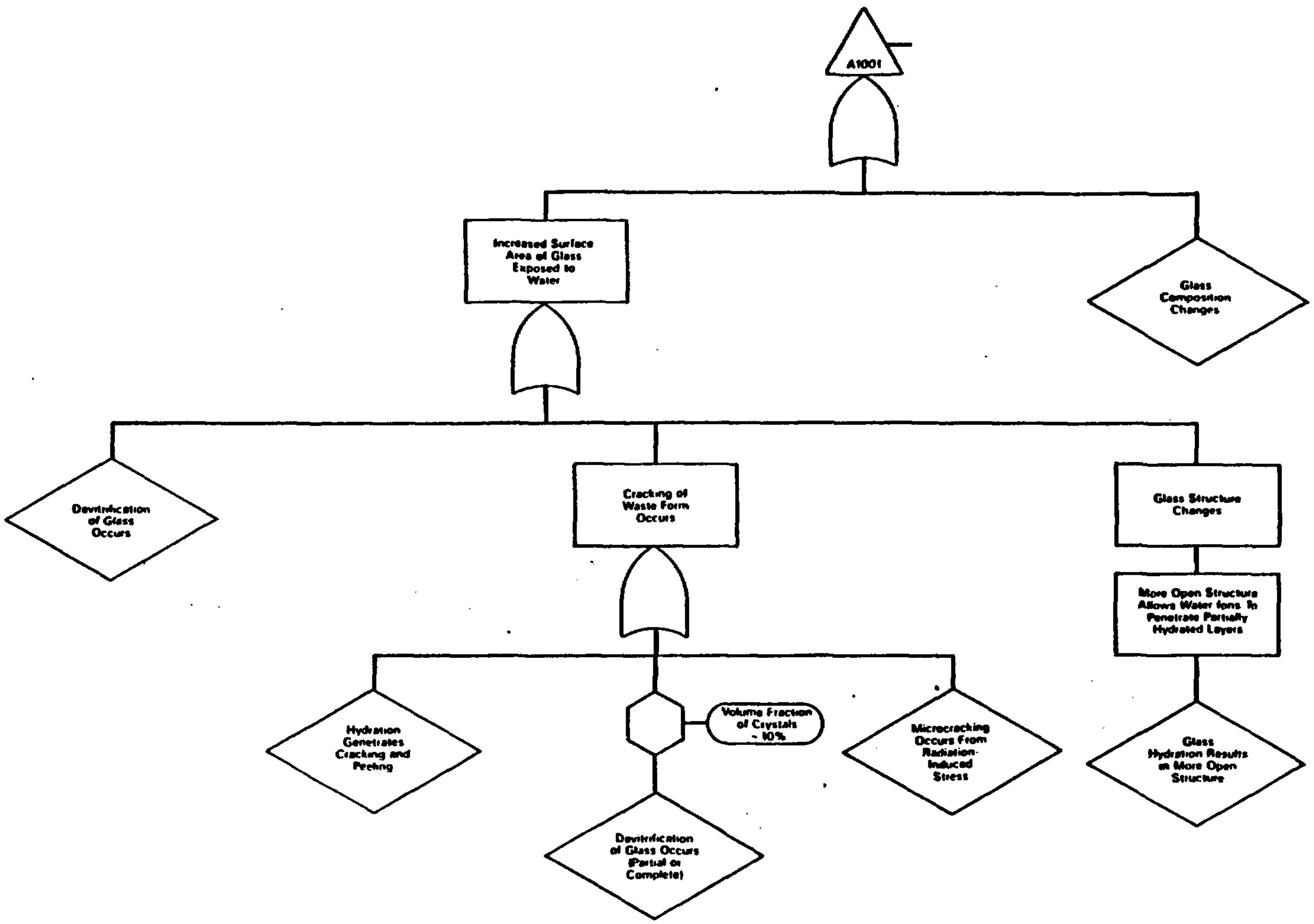

Figure A-4. A1001—Waste Form Characteristics Altered To Facilitate Radionuclide Release

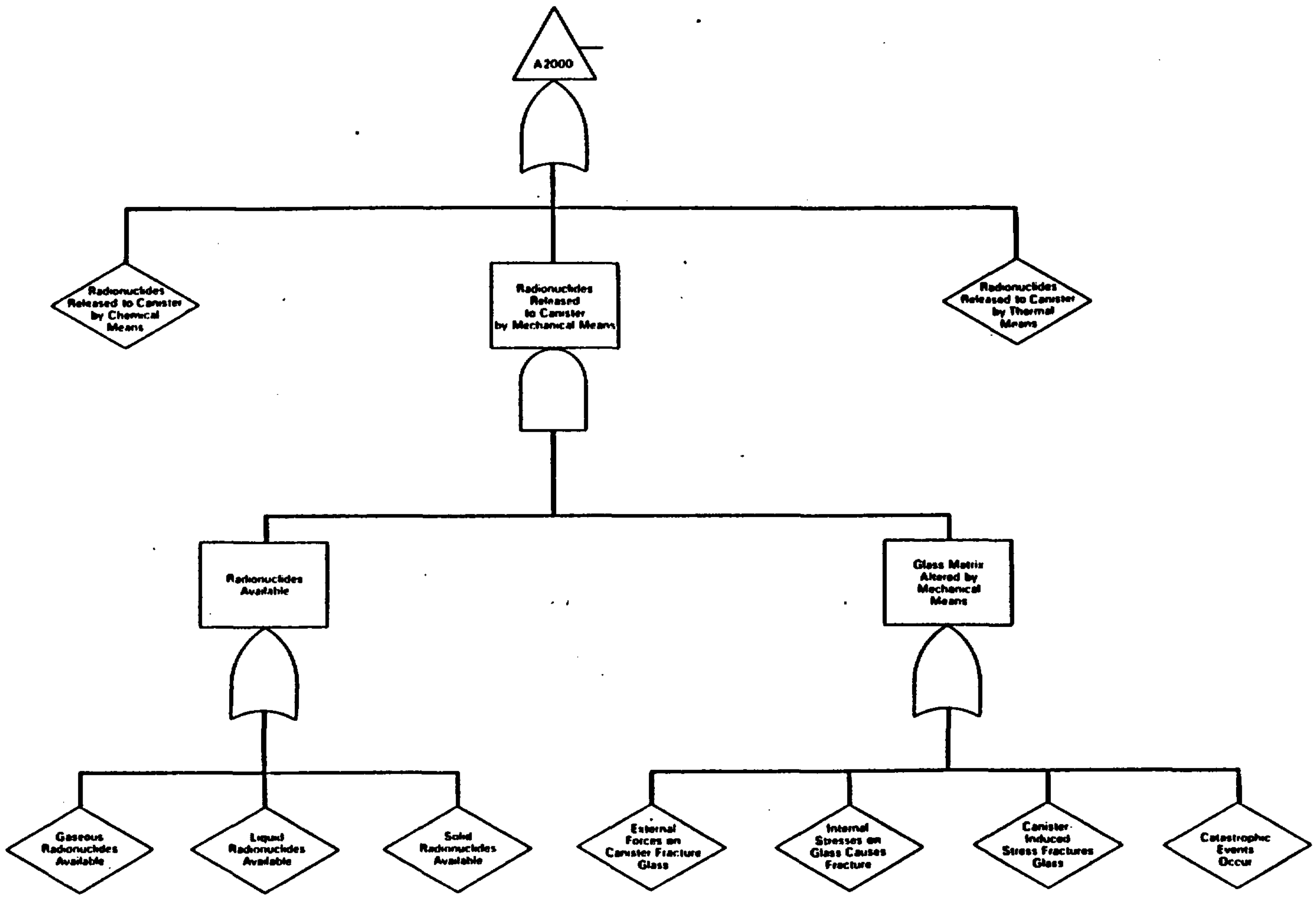

Figure A-5. A2000—Radionuclides Released From Glass to Canister by Nonaqueous Means

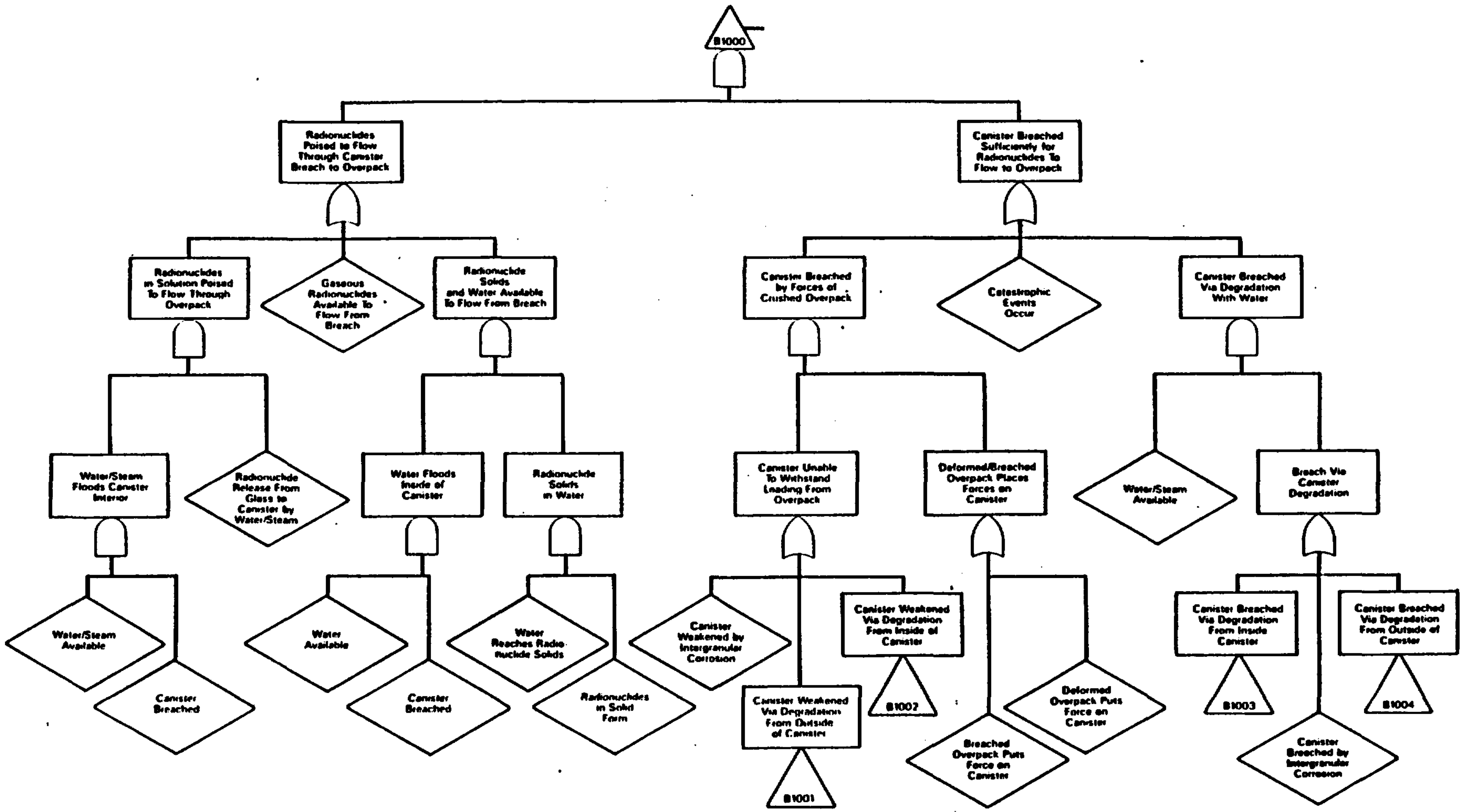

Figure A-6. B1000—Canister Allows Radionuclides Aqueous Transport (Through Canister) to Overpack

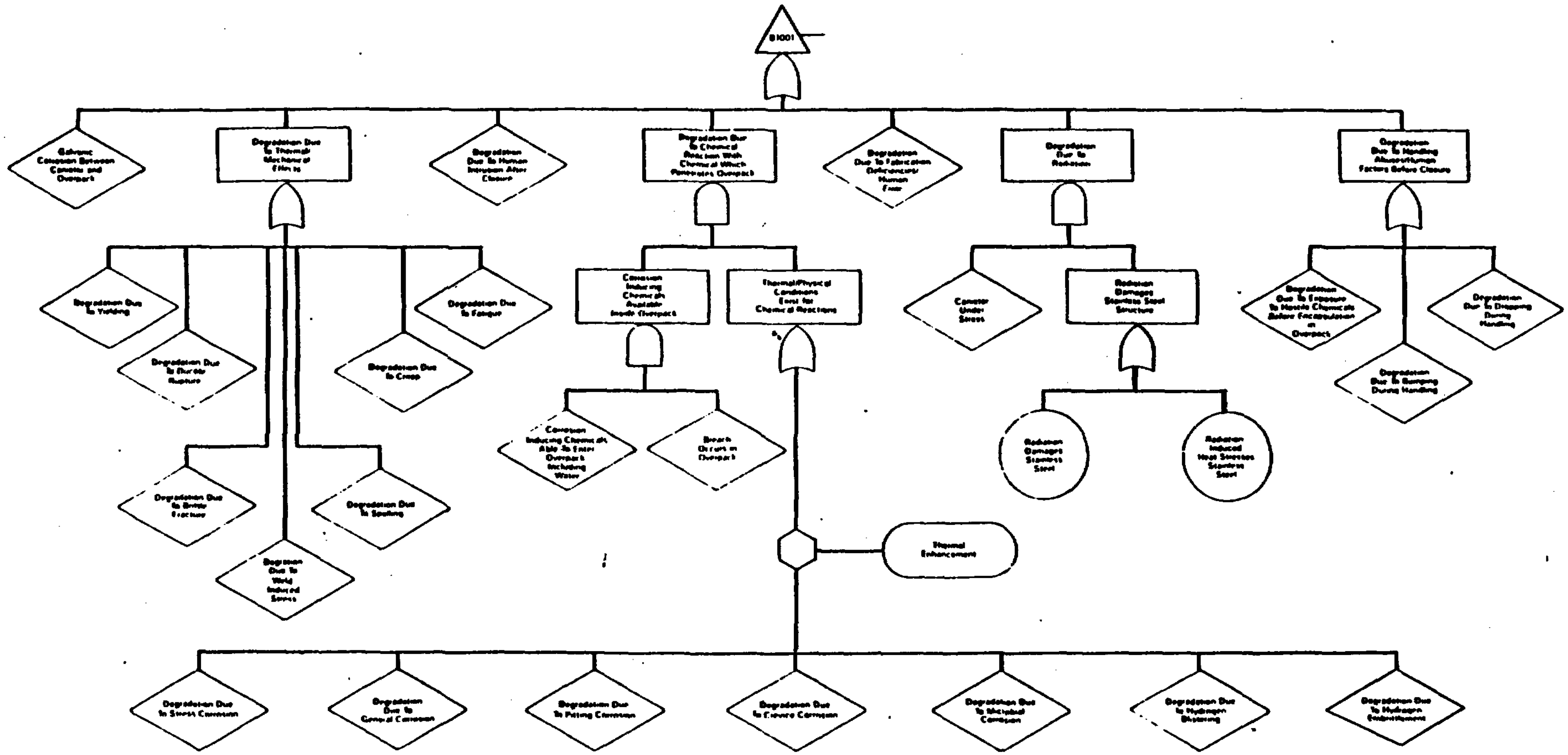

Figure A-7. B1001—Canister Weakened Via Degradation From Outside Canister

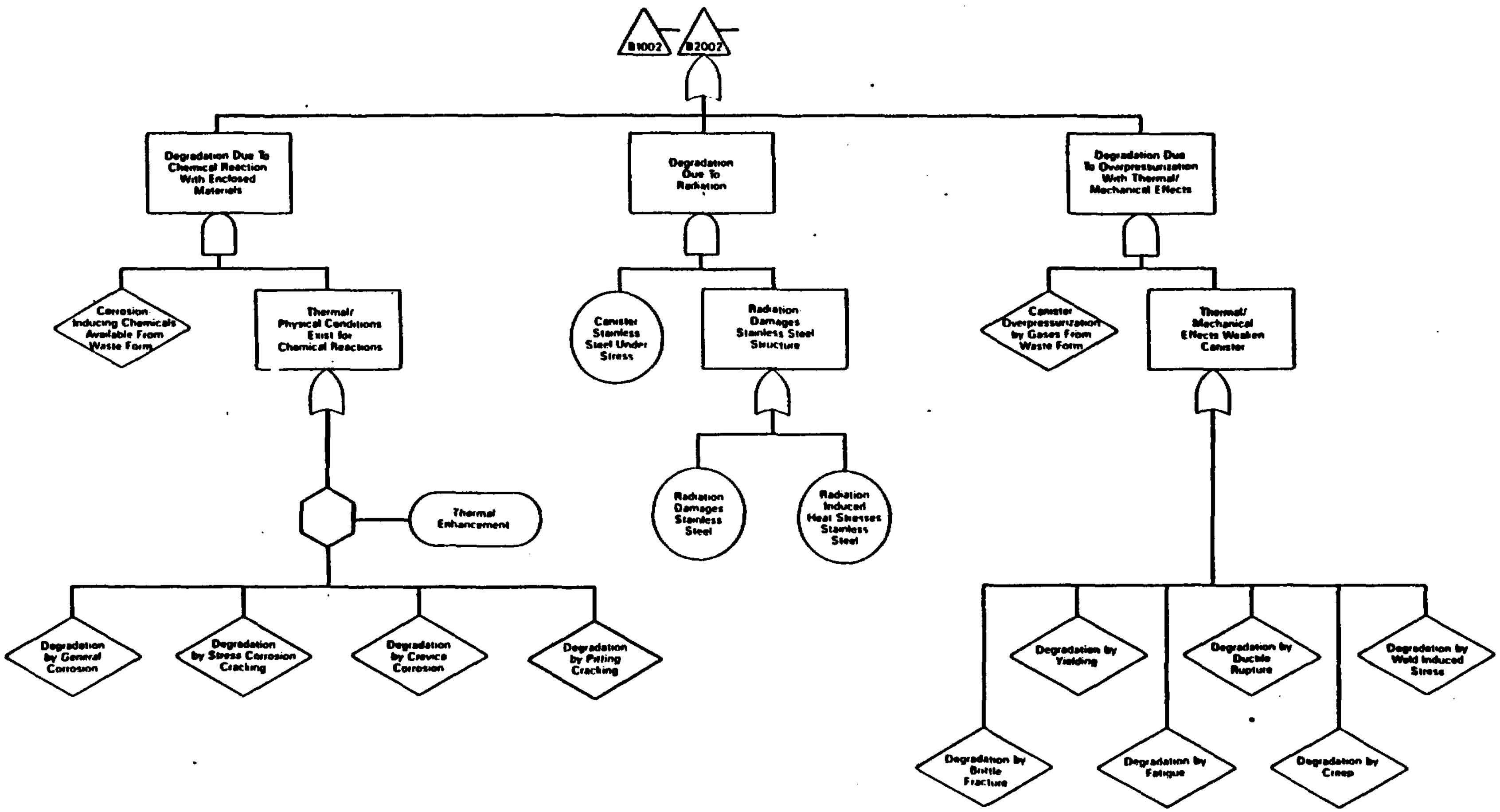

Figure A-8. B1002 and B2002—Canister Weakened Via Degradation From Inside Canister

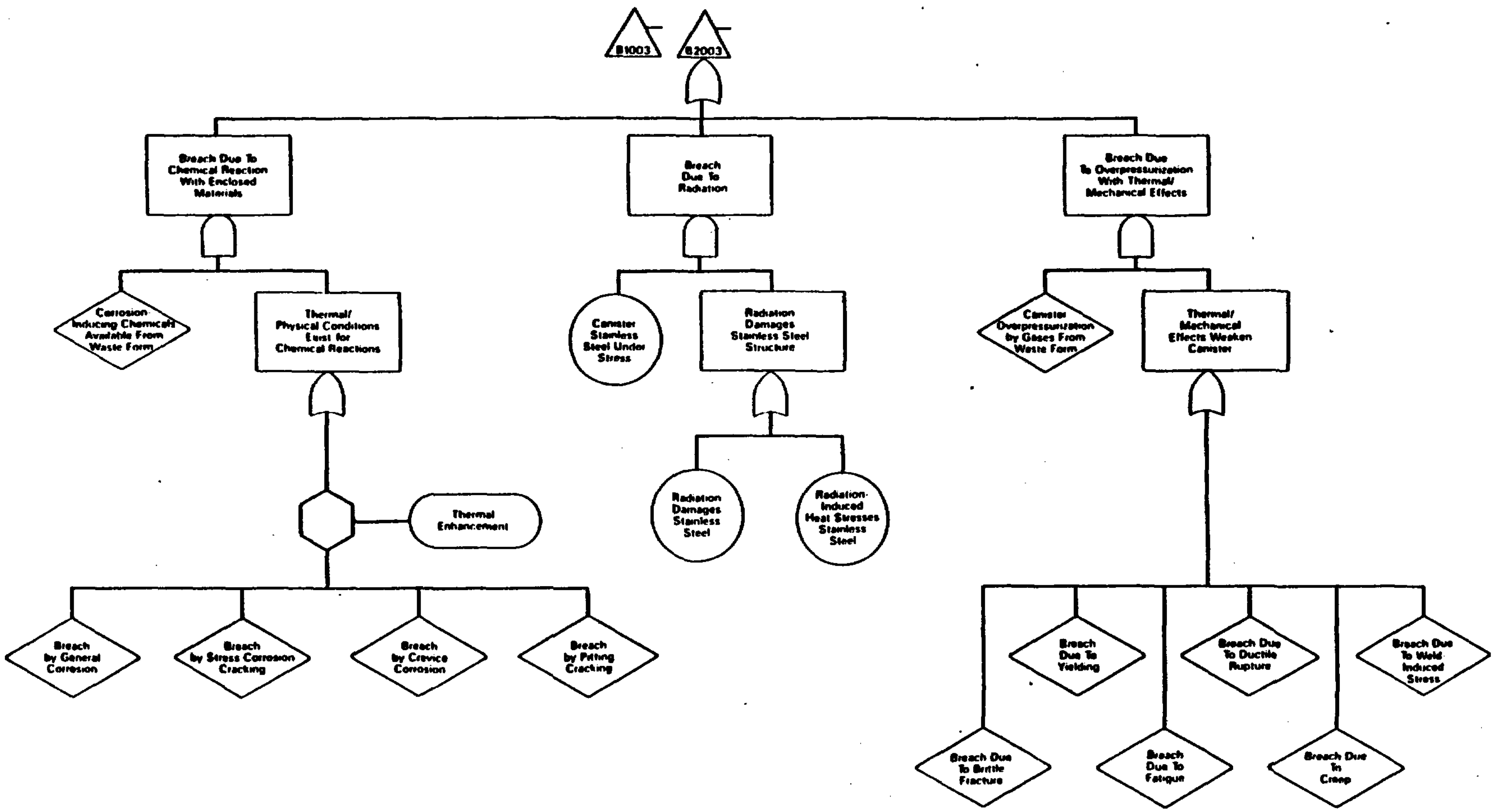

Figure A-9. B1003 and B2003 — Canister Breached Via Degradation From Inside Canister

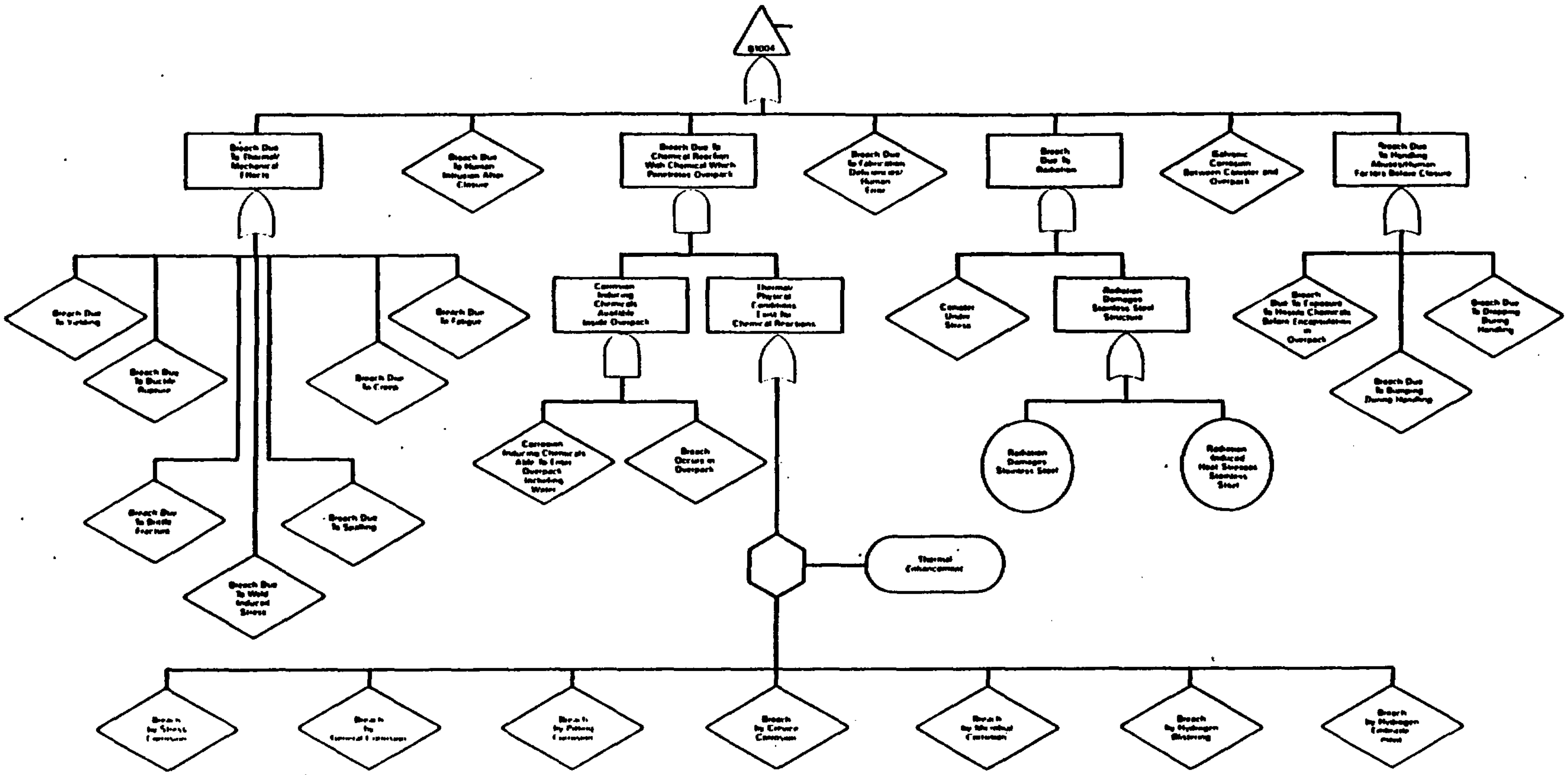

Figure A-10. B1004—Canister Breached Via Degradation From Outside Canister

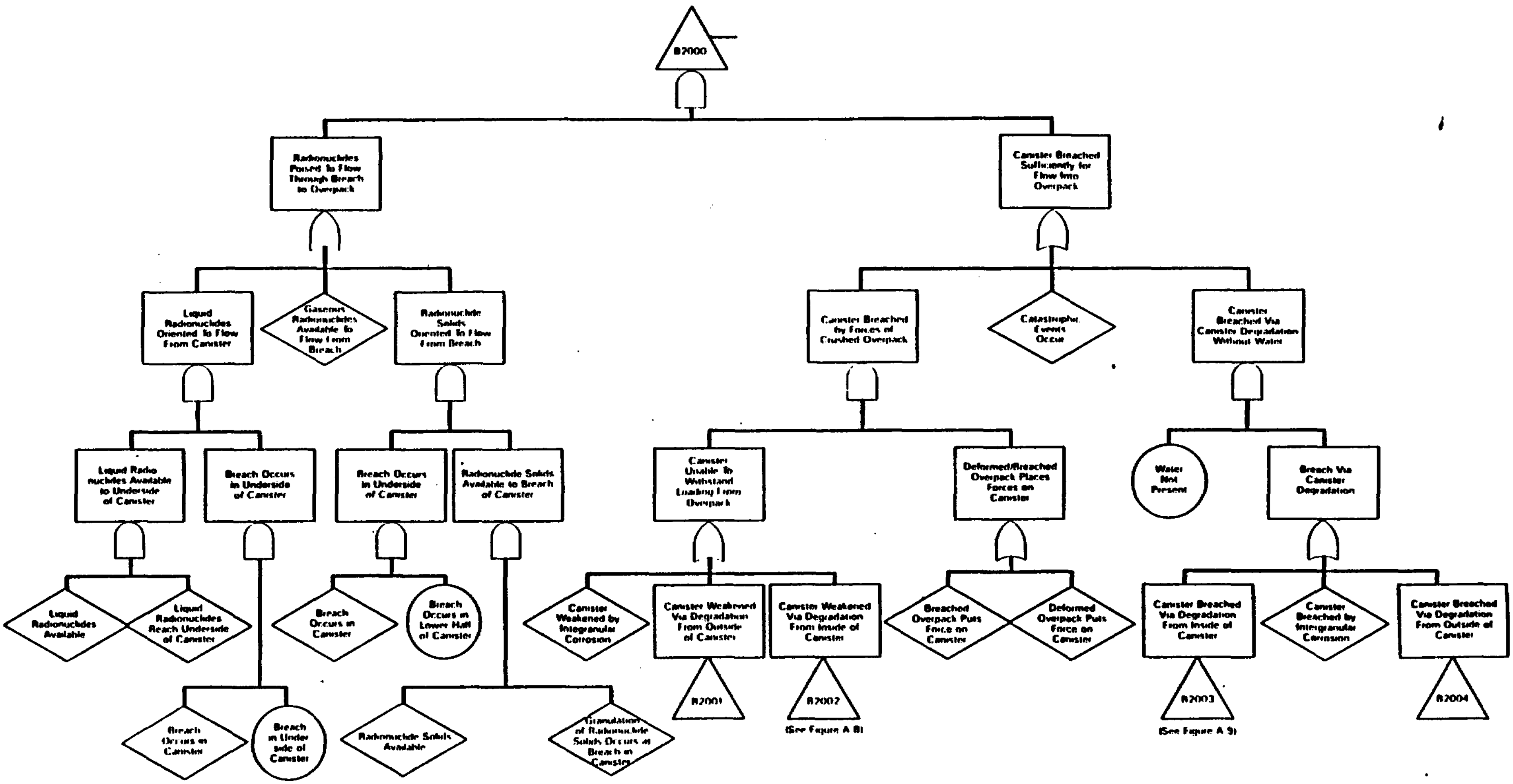

Figure A-11. B2000 – Canister Allows Radionuclide Nonaqueous Transport (Through Canister) to Overpack

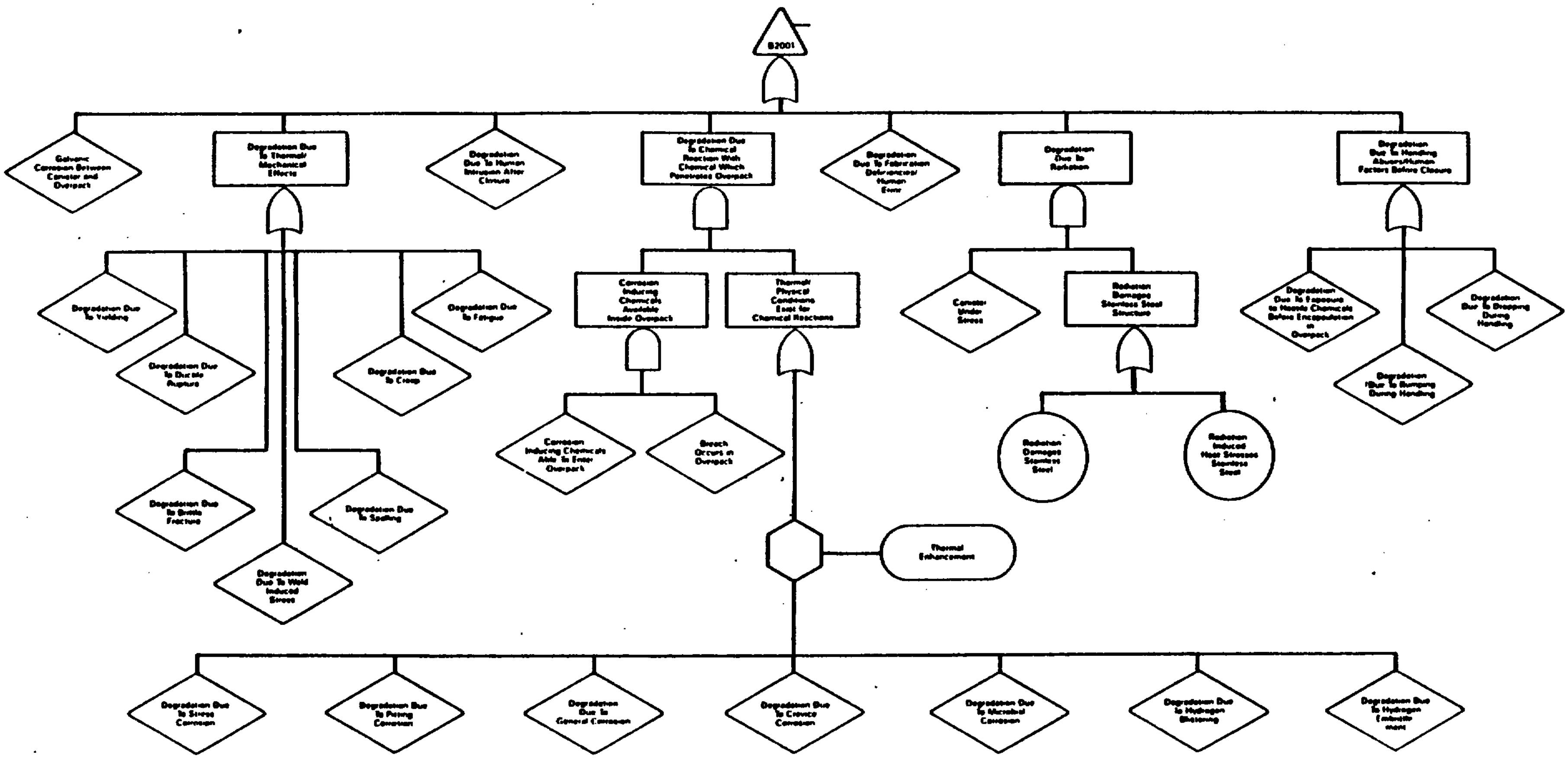

Figure A-12. B2001—Canister Weakened Via Degradation From Outside Canister

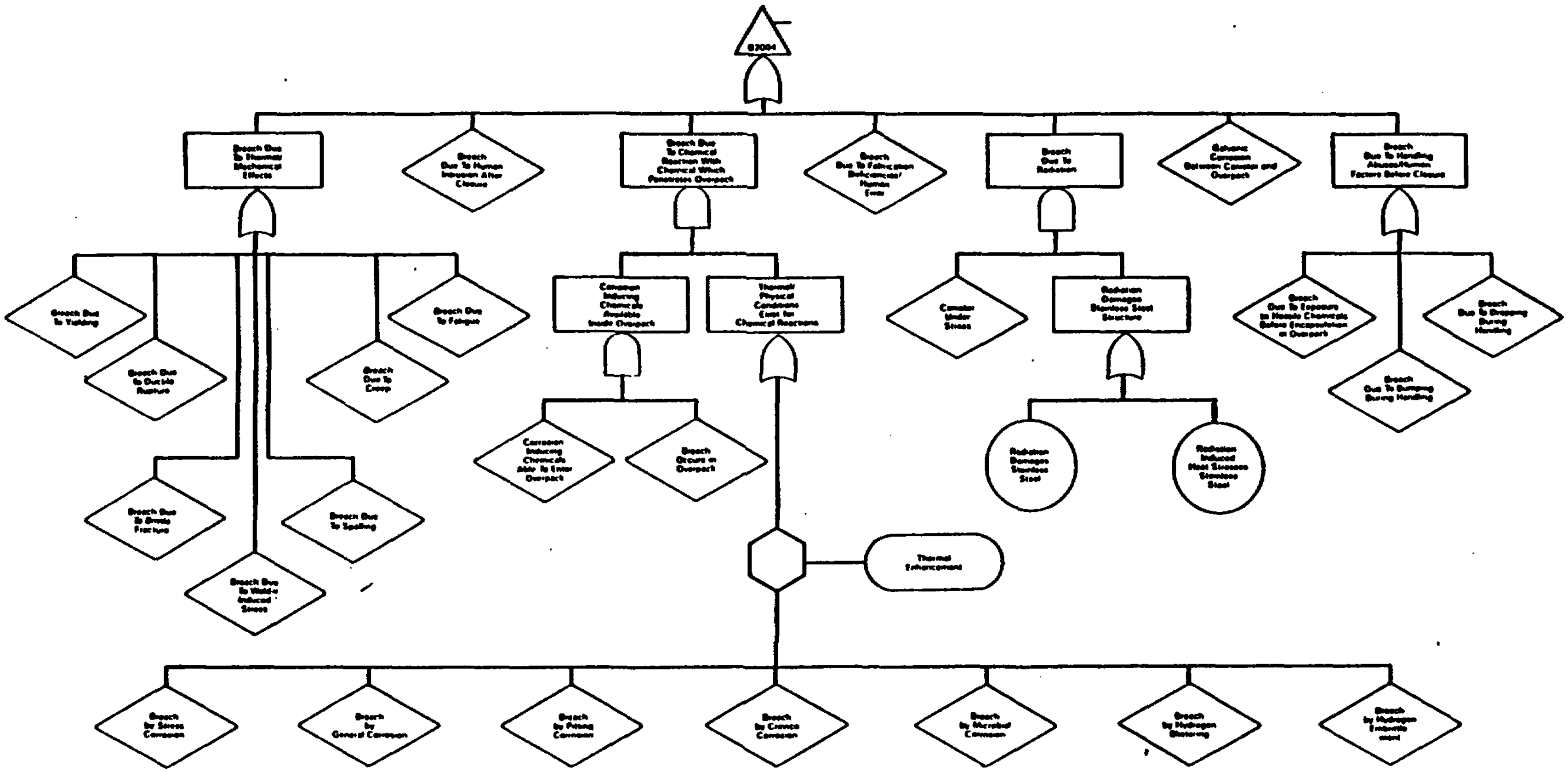

Figure A-13. B2004—Canister Breached Via Degradation From Outside Canister

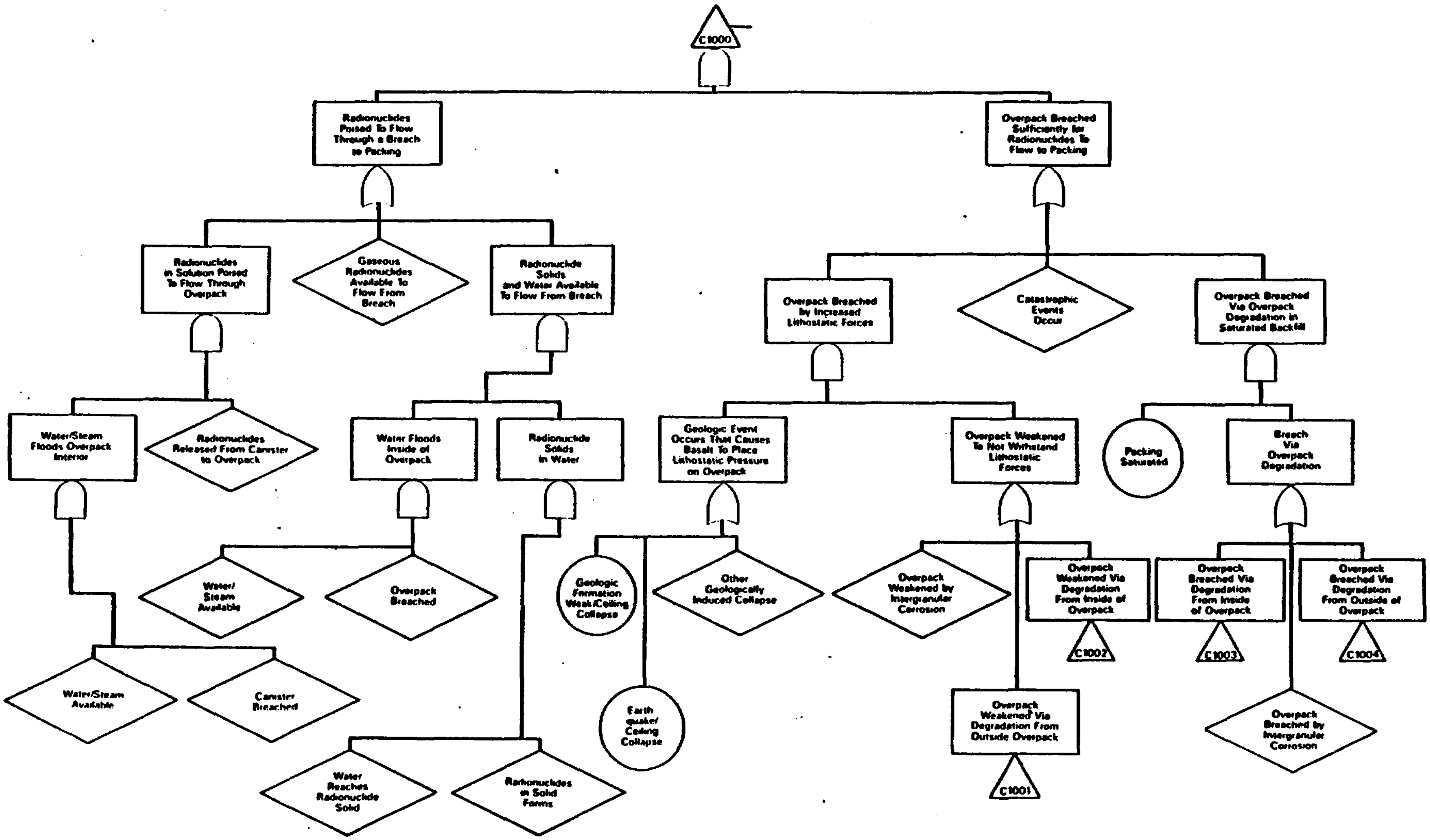

Figure A-14. C1000—Overpack Allows Radionuclides Aqueous Transport (Through Overpack) to Packing

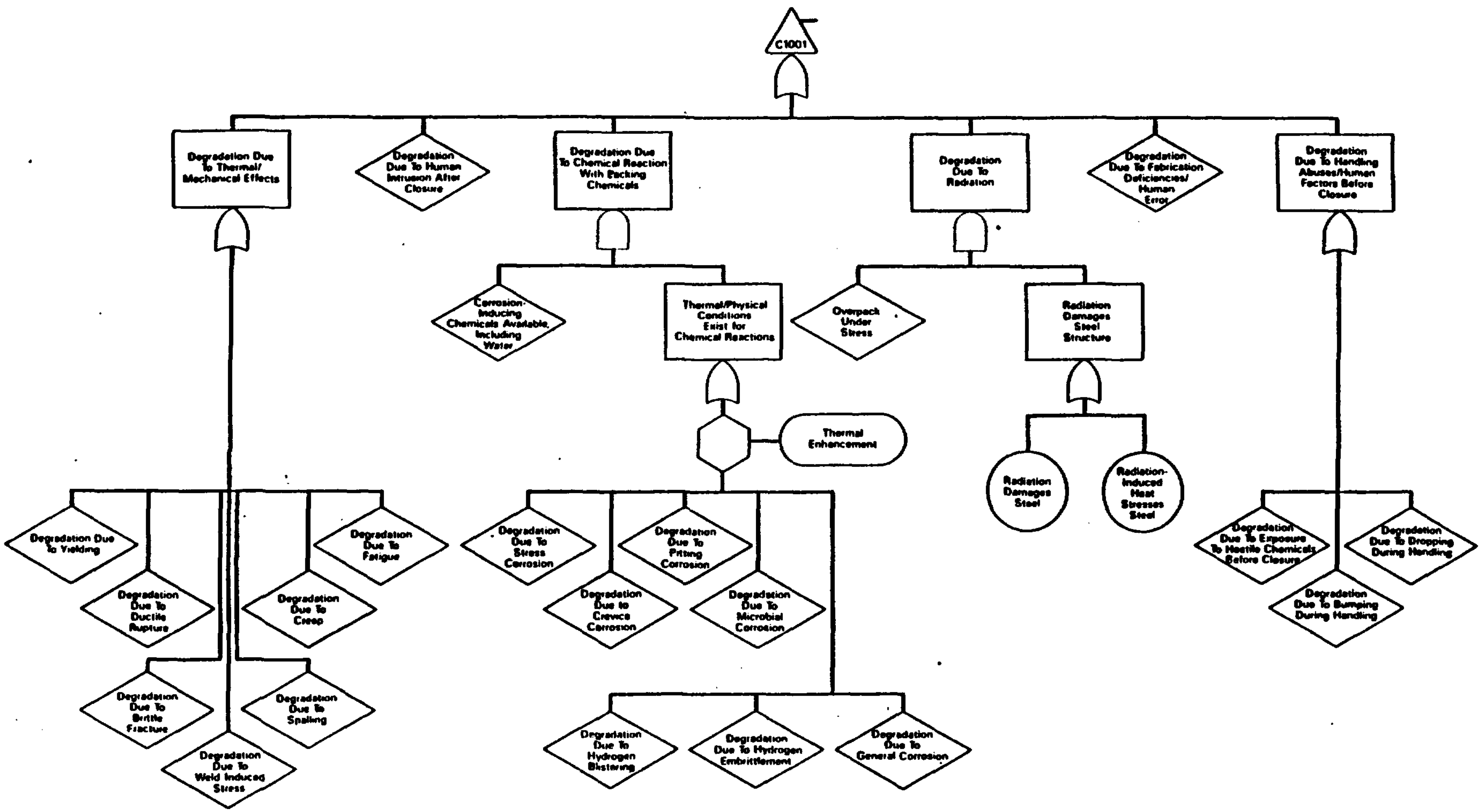

Figure A-15. C1001—Overpack Weakened Via Degradation From Outside Overpack

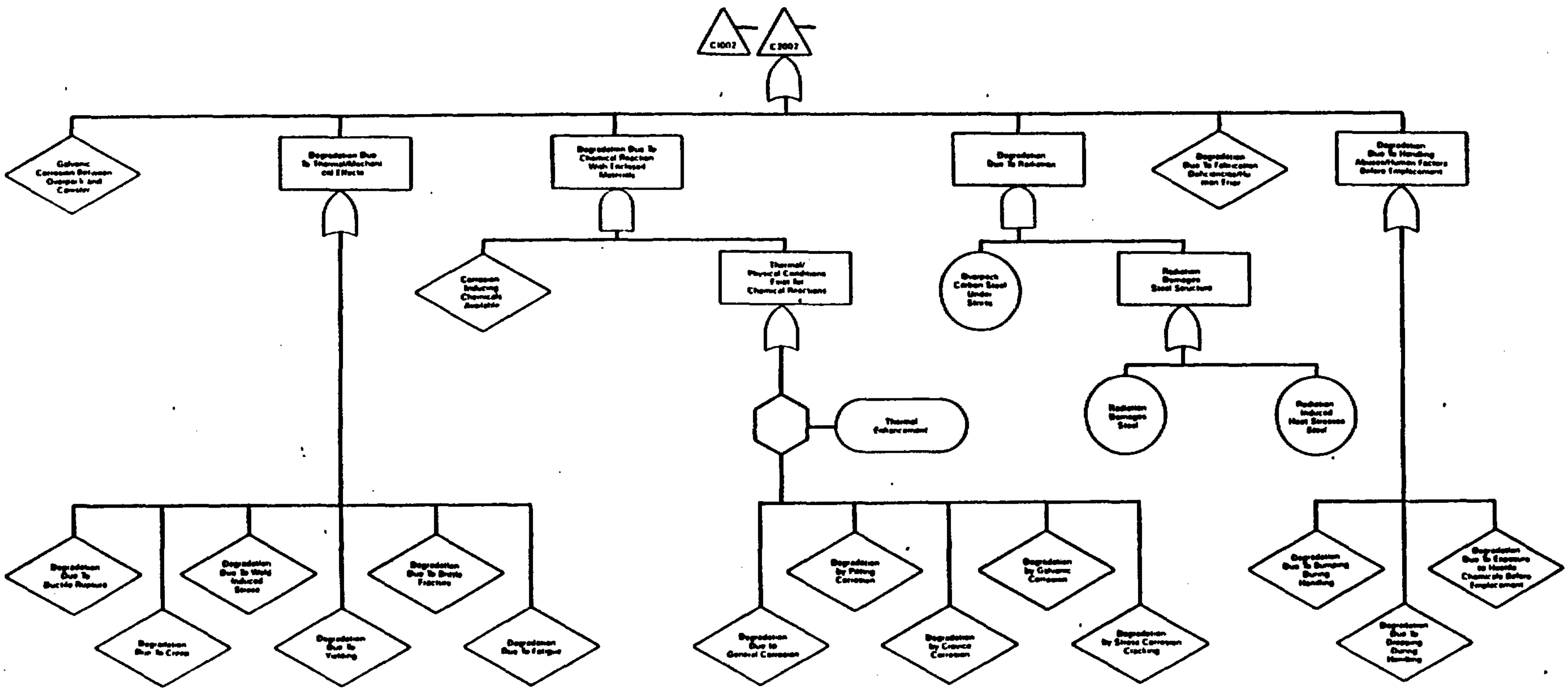

Figure A-16. C1002 and C2002—Overpack Weakened Via Degradation From Inside Overpack

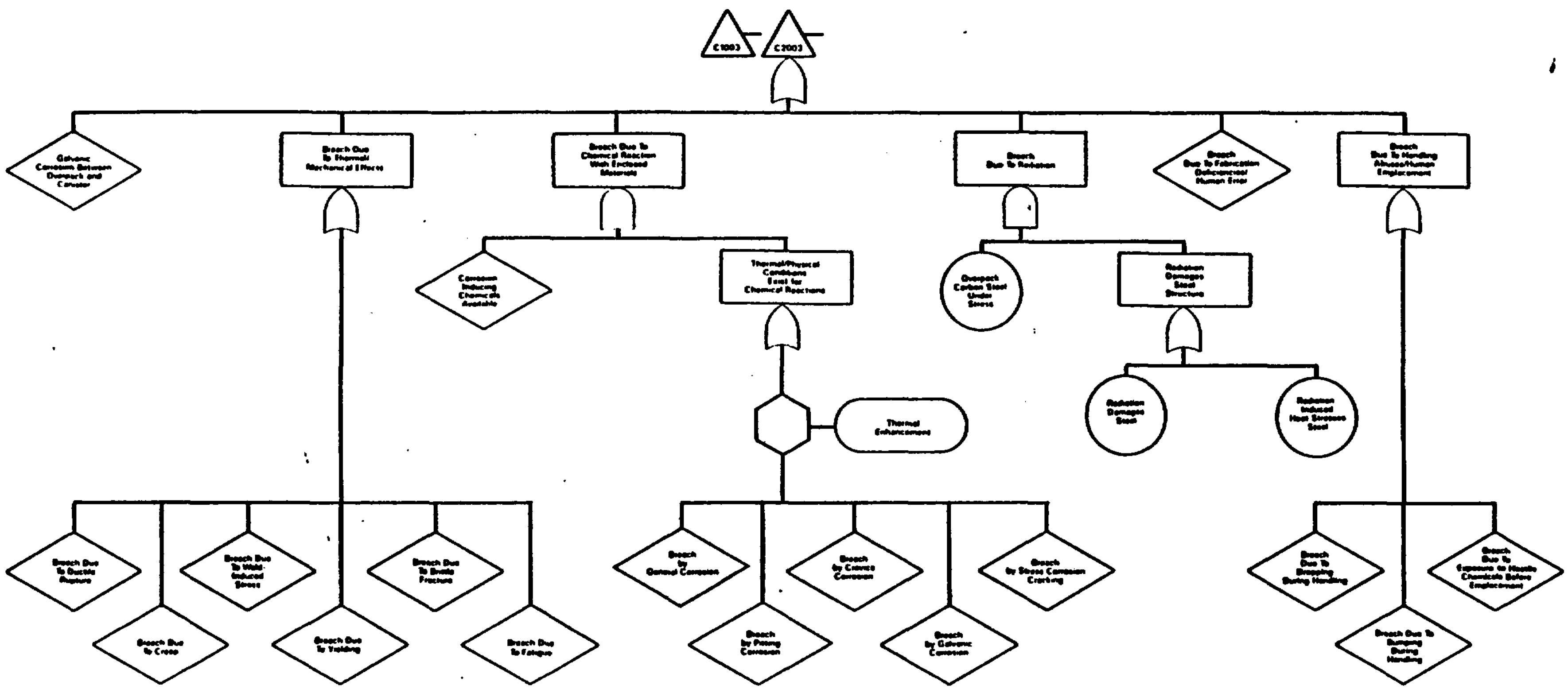

Figure A-17. C1003 and C2003—Overpack Breached Via Degradation From Inside Overpack

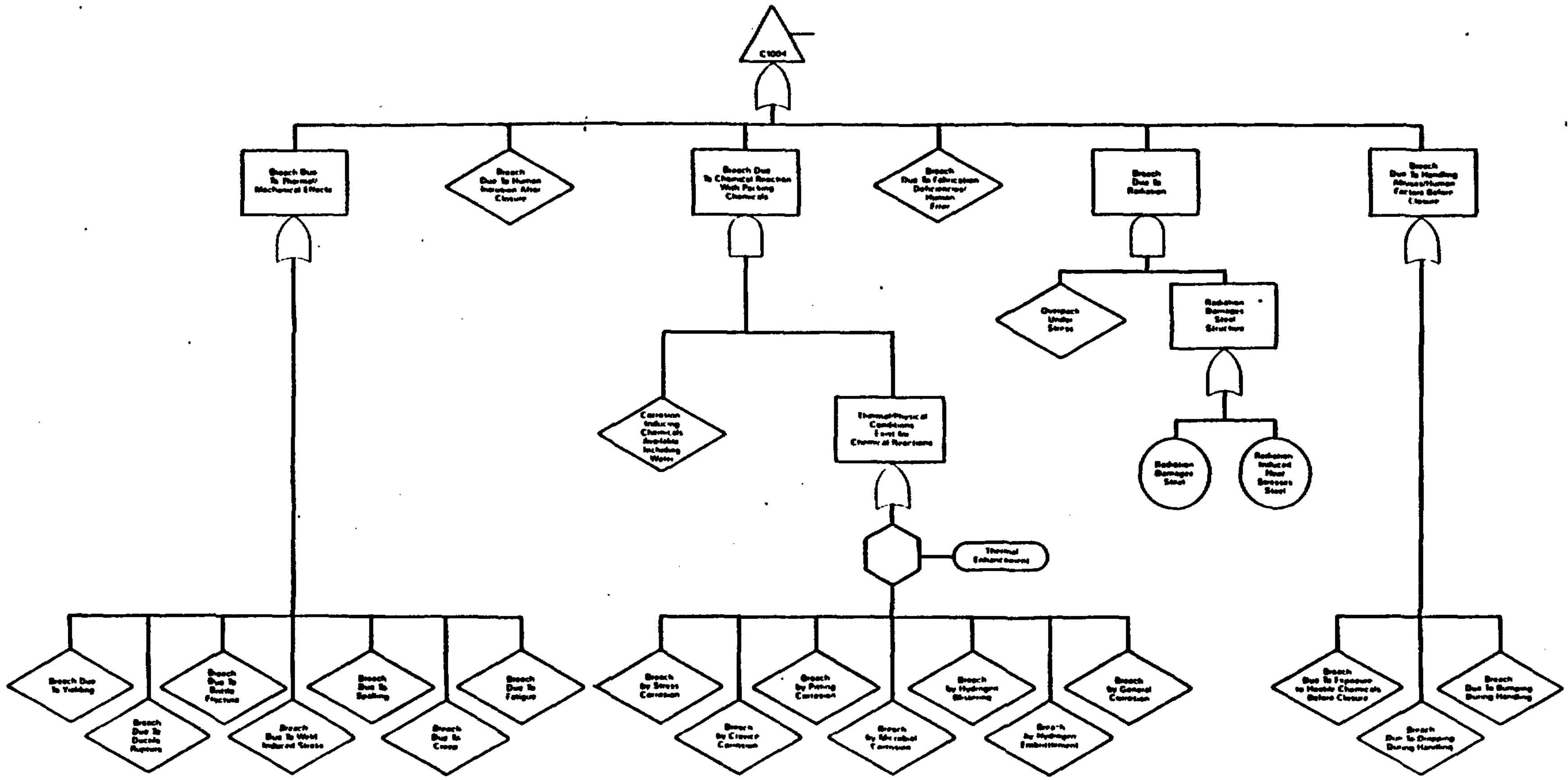

Figure A-18. C1004—Overpack Breached Via Degradation From Outside Overpack

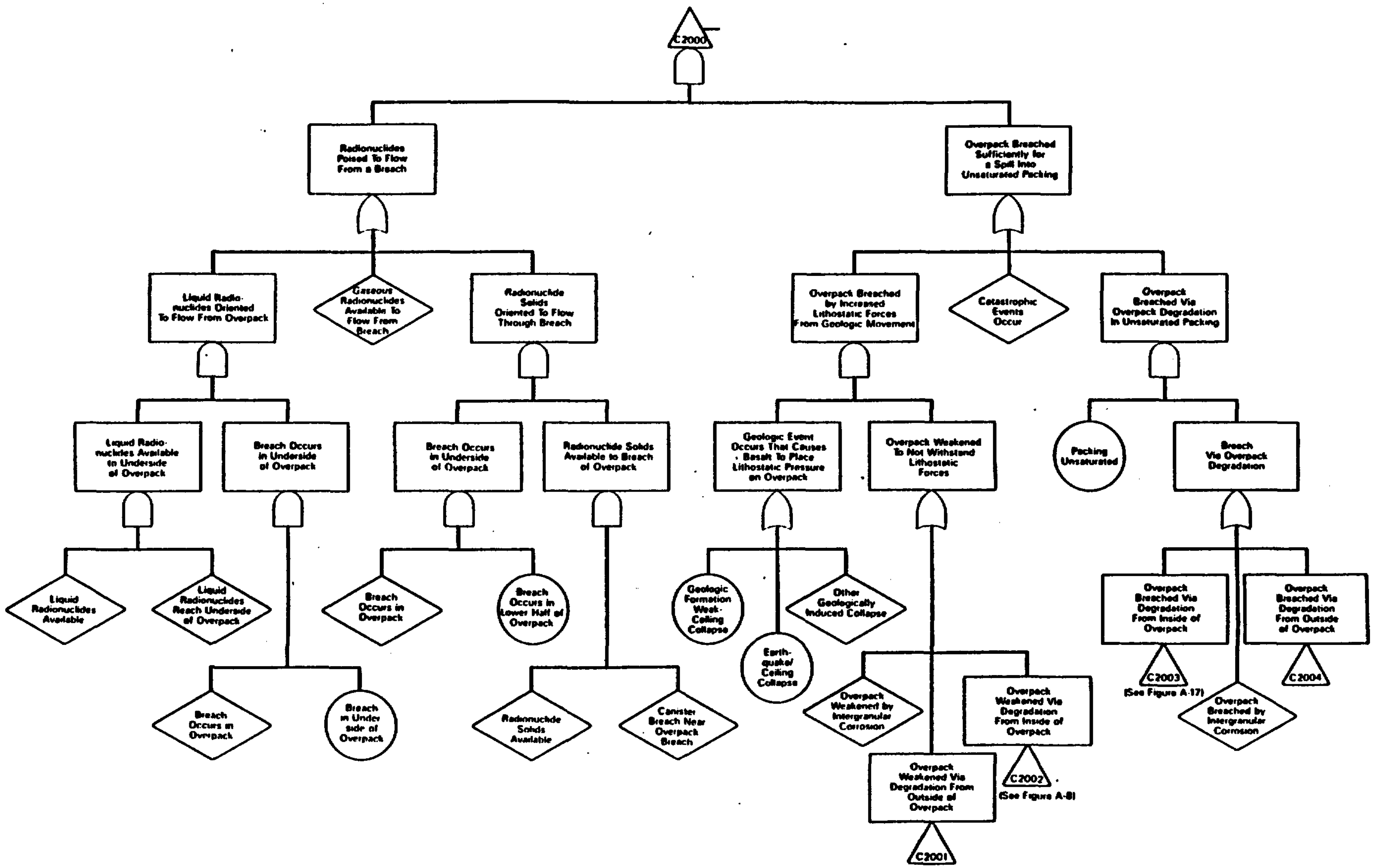

Figure A-19. C2000—Overpack Allows Radionuclides Nonaqueous Transport to Packing

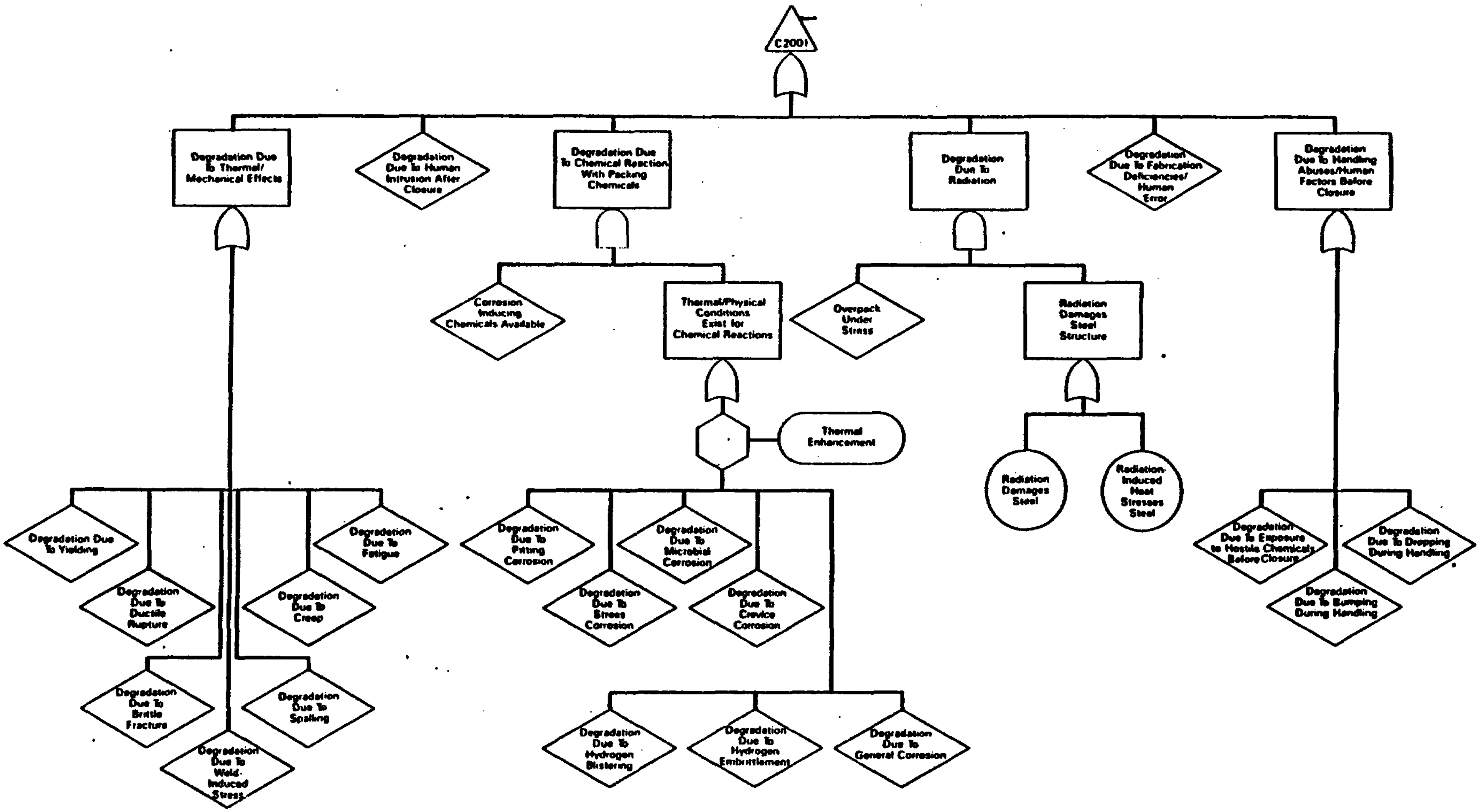

Figure A-20. C2001—Overpack Weakened Via Degradation From Outside Overpack

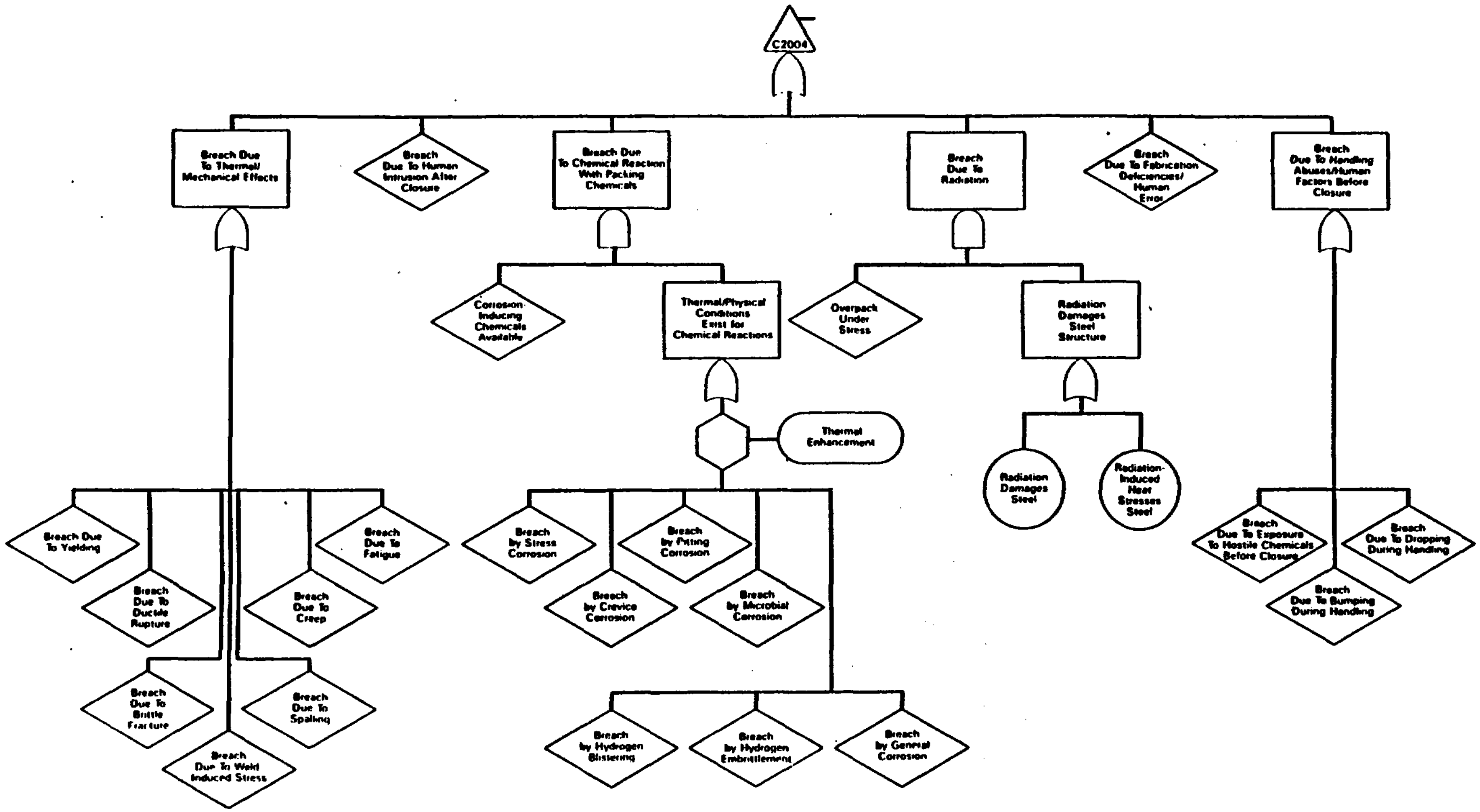

Figure A-21. C2004—Overpack Breached Via Degradation From Outside Overpack

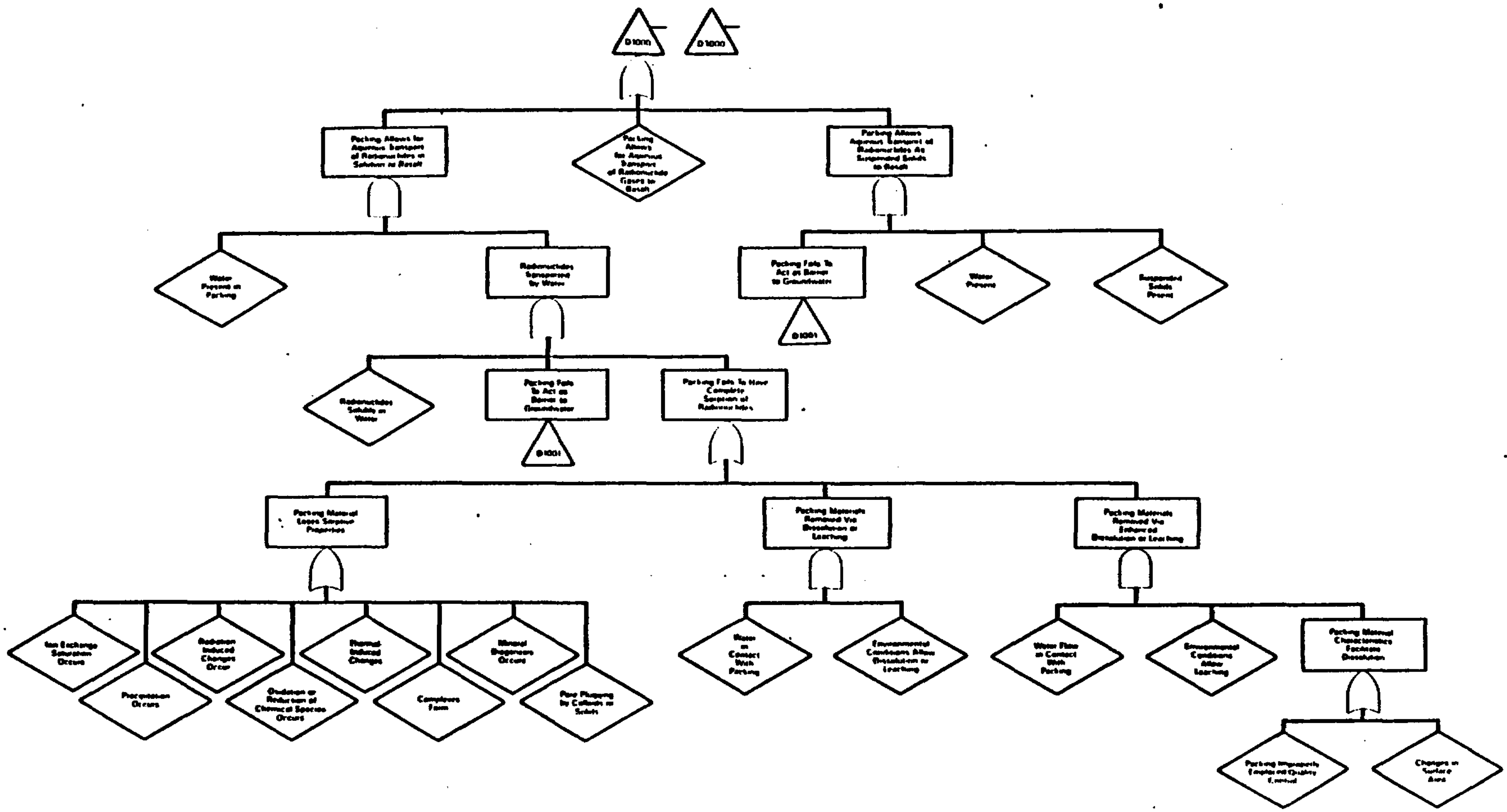

Figure A-22. D1000 and D3000—Packing Allows Radionuclides Aqueous Transport to Basalt

RESPONSES TO

NRC COMMENTS

ON BMTP

MAY 9, 1984

ATTACHMENT 10

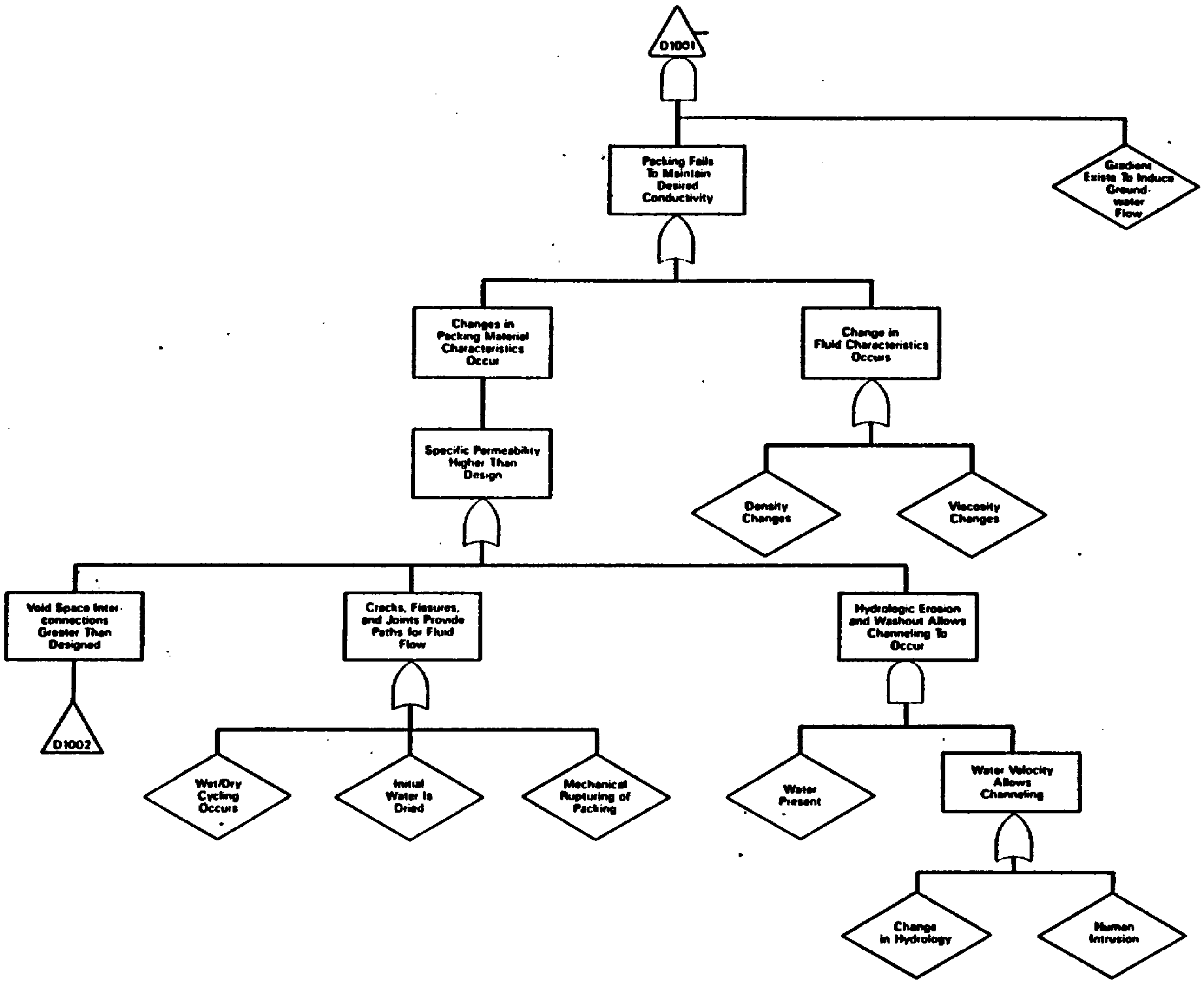

Figure A-23. D1001—Packing Fails To Act as a Barrier to Groundwater

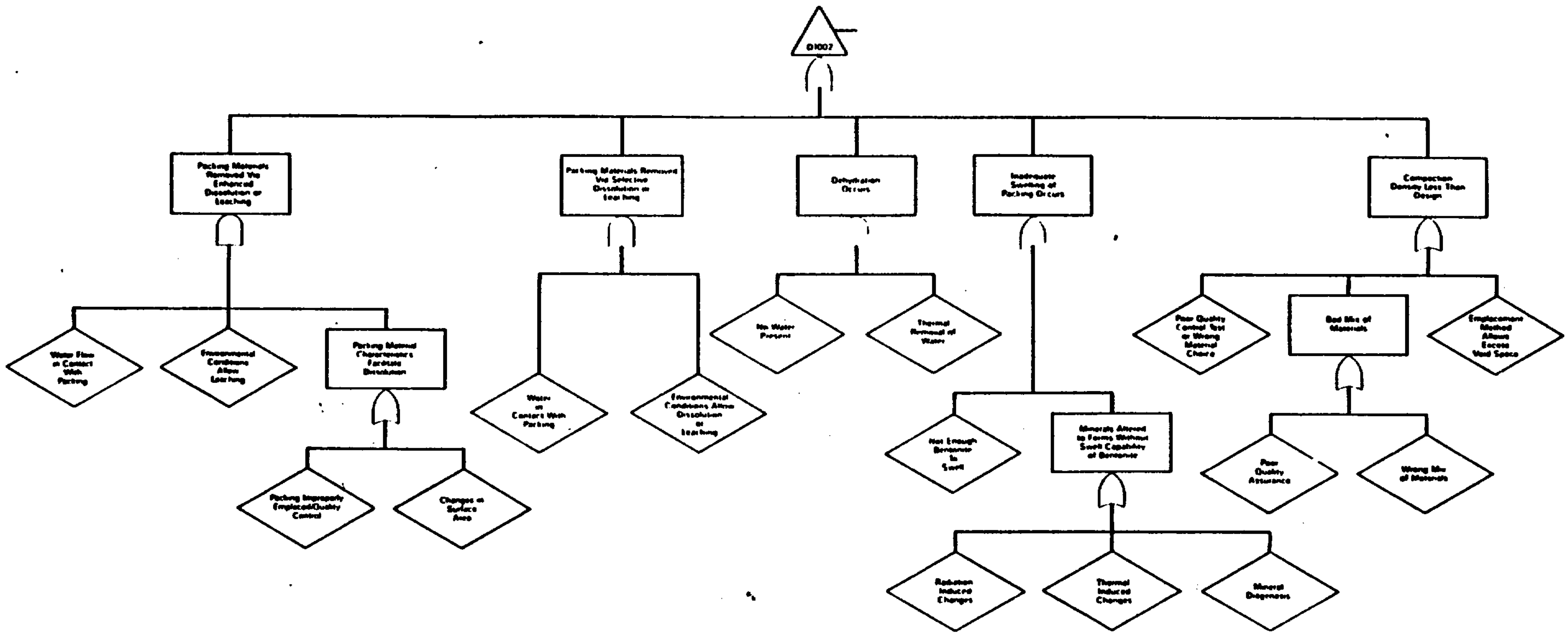

Figure A-24. D1002—Void Space Interconnections Greater Than Desired

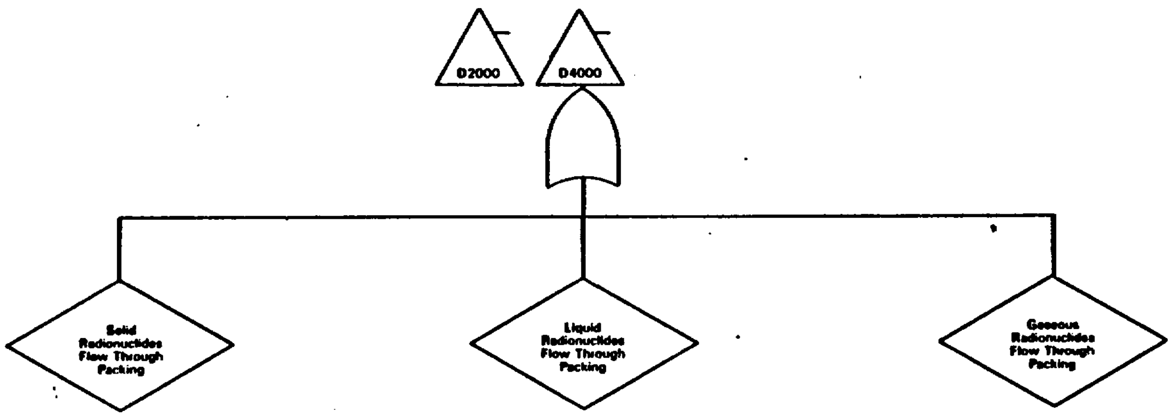

Figure A-25 D2000 and D4000—Packing Allows Radionuclides Nonaqueous Transport to Basalt

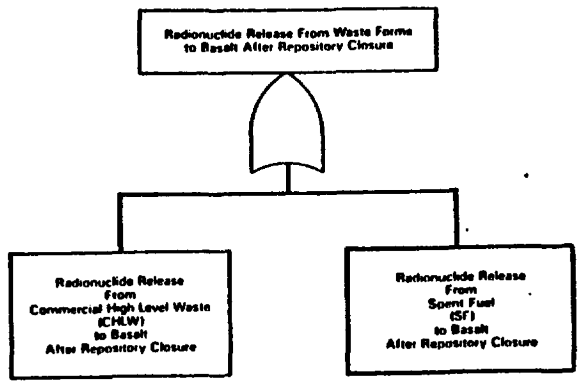

Figure 8-1. Top Events for Radionuclide Release From Waste Forms to Basalt After Repository Closure

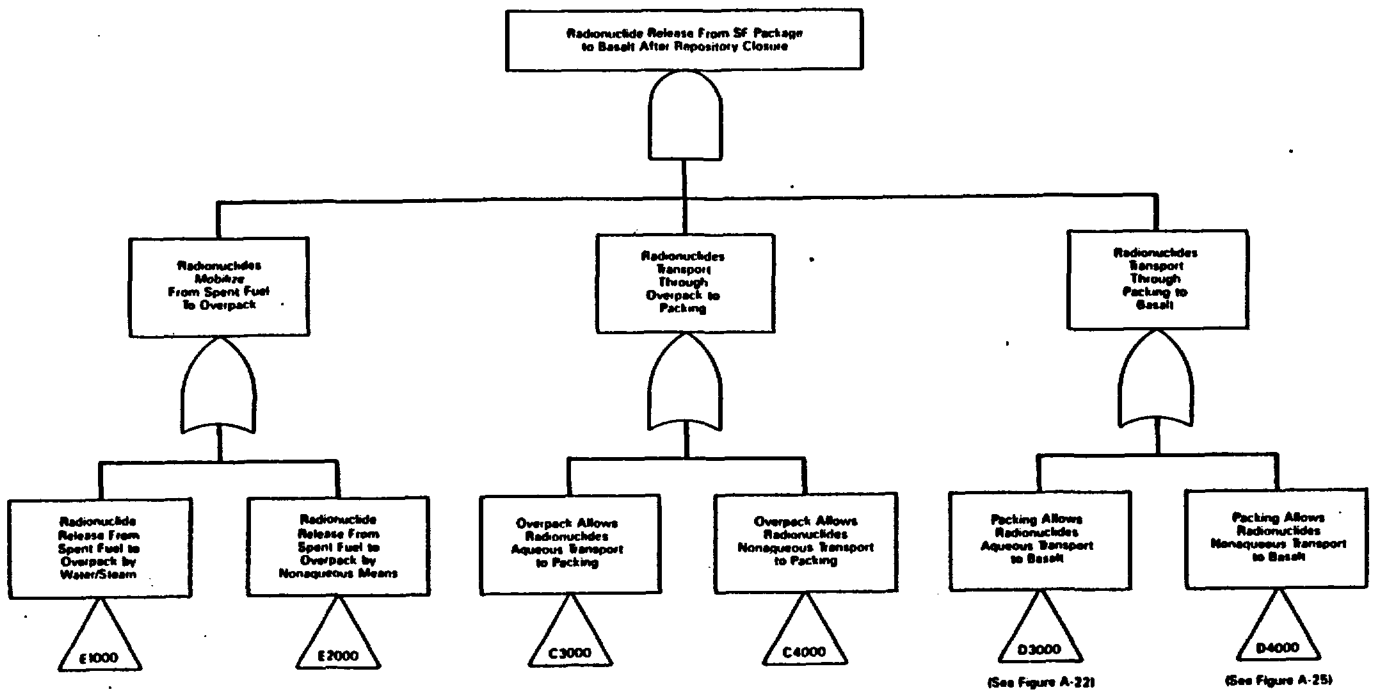

Figure B-2. Top Event Fault Tree for Radionuclide Release From Spent Fuel Package to Basalt After Repository Closure

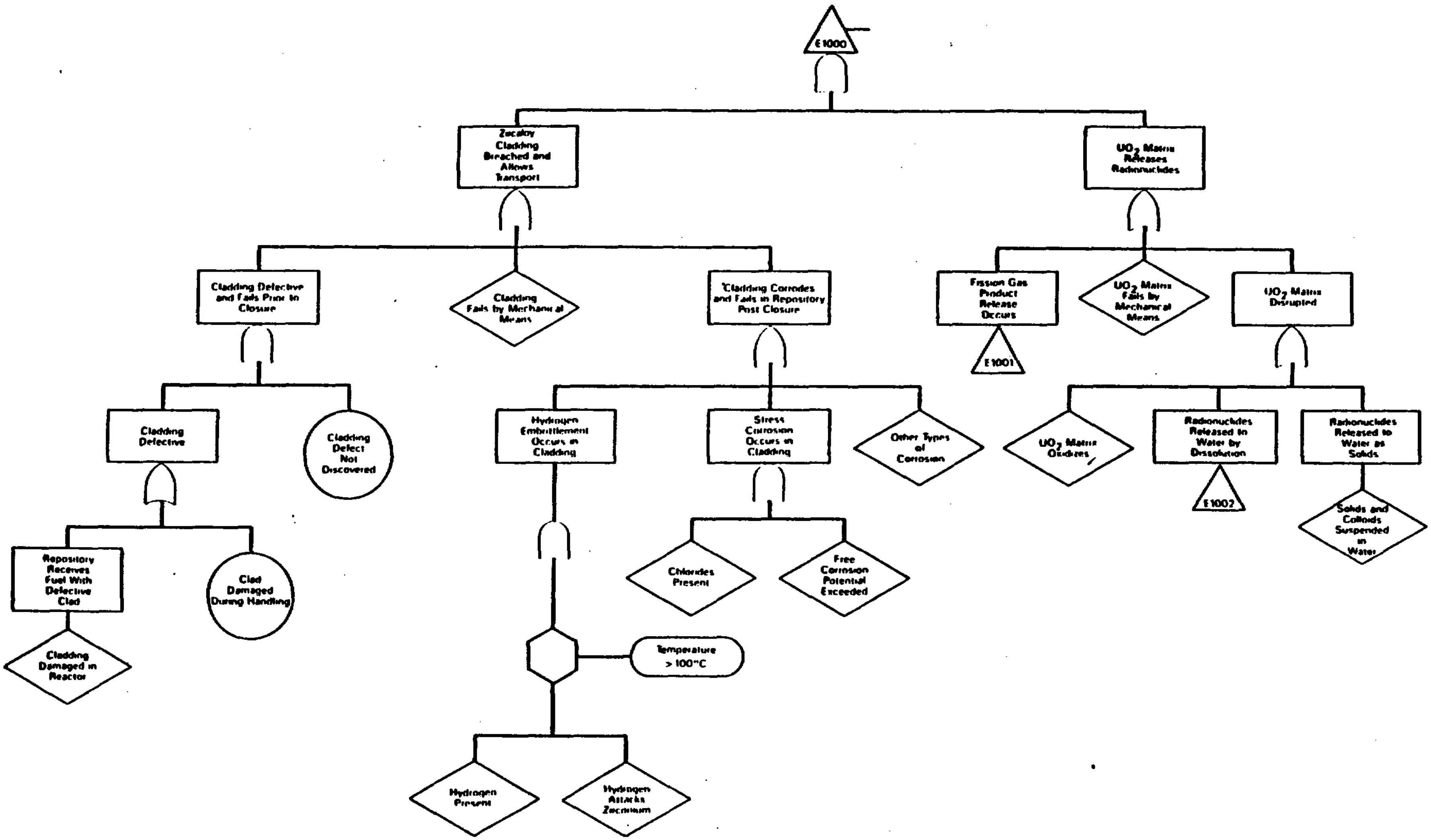

Figure B-3. E1000—Radionuclides Released From Spent Fuel to Overpack by Water/Steam

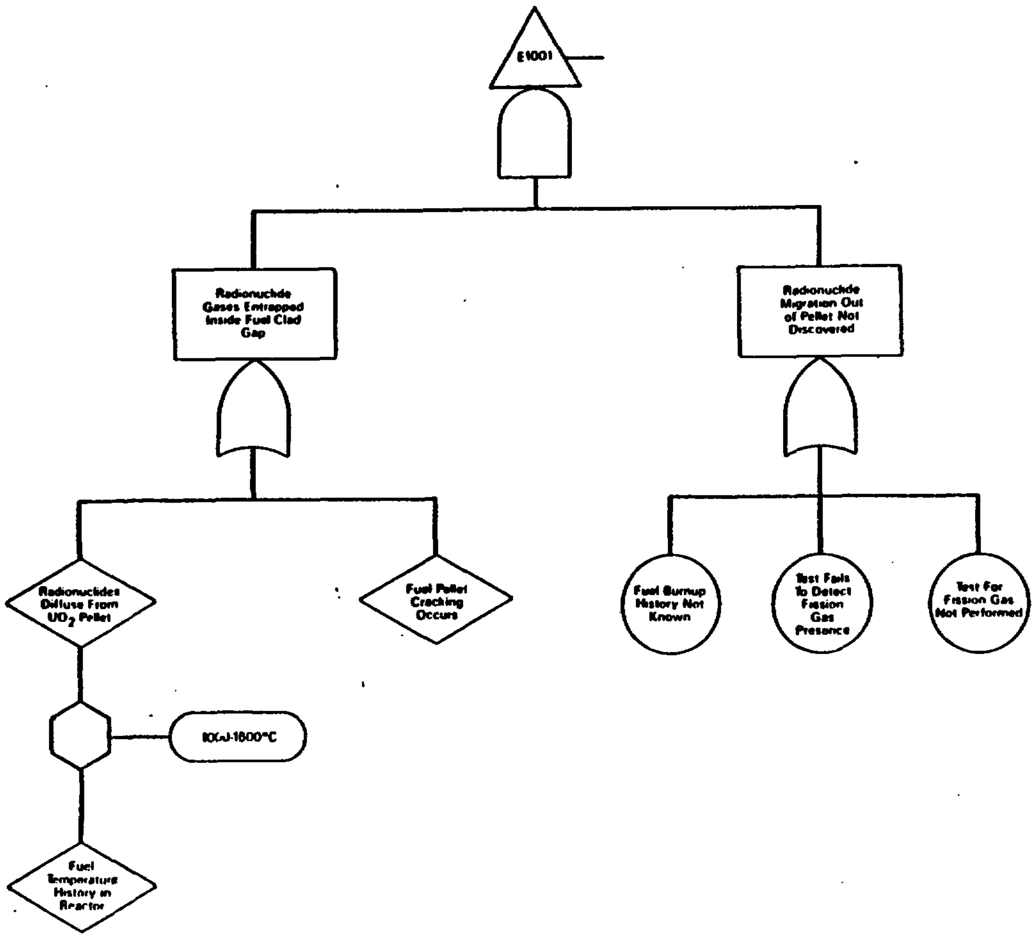

Figure B-4. E1001—Fission Gas Product Release Occurs

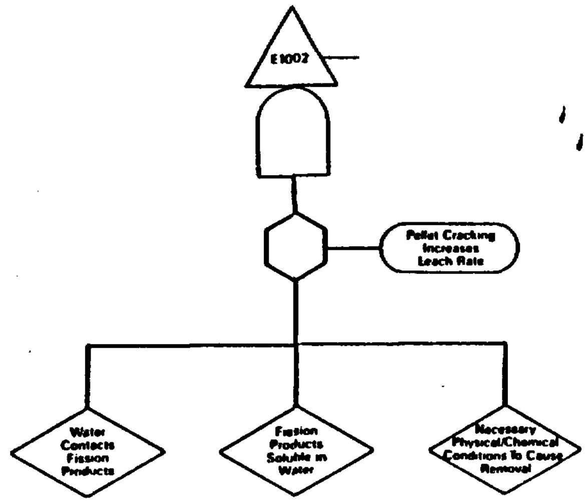

Figure B-5. E1002 — Radionuclides Released to Water by Dissolution

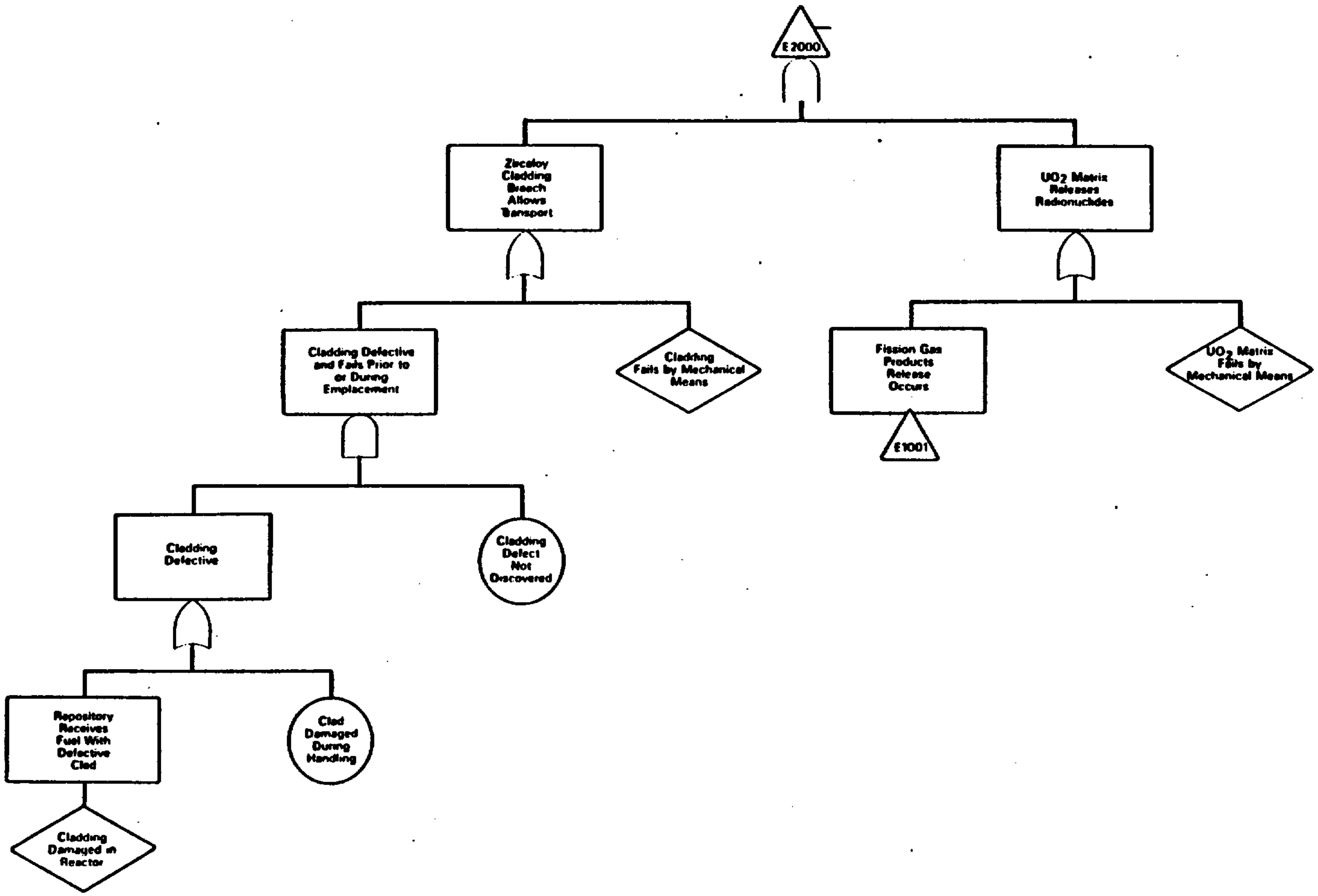

Figure B-6. E2000—Radionuclides Released From Spent Fuel to Overpack by Nonaqueous Means

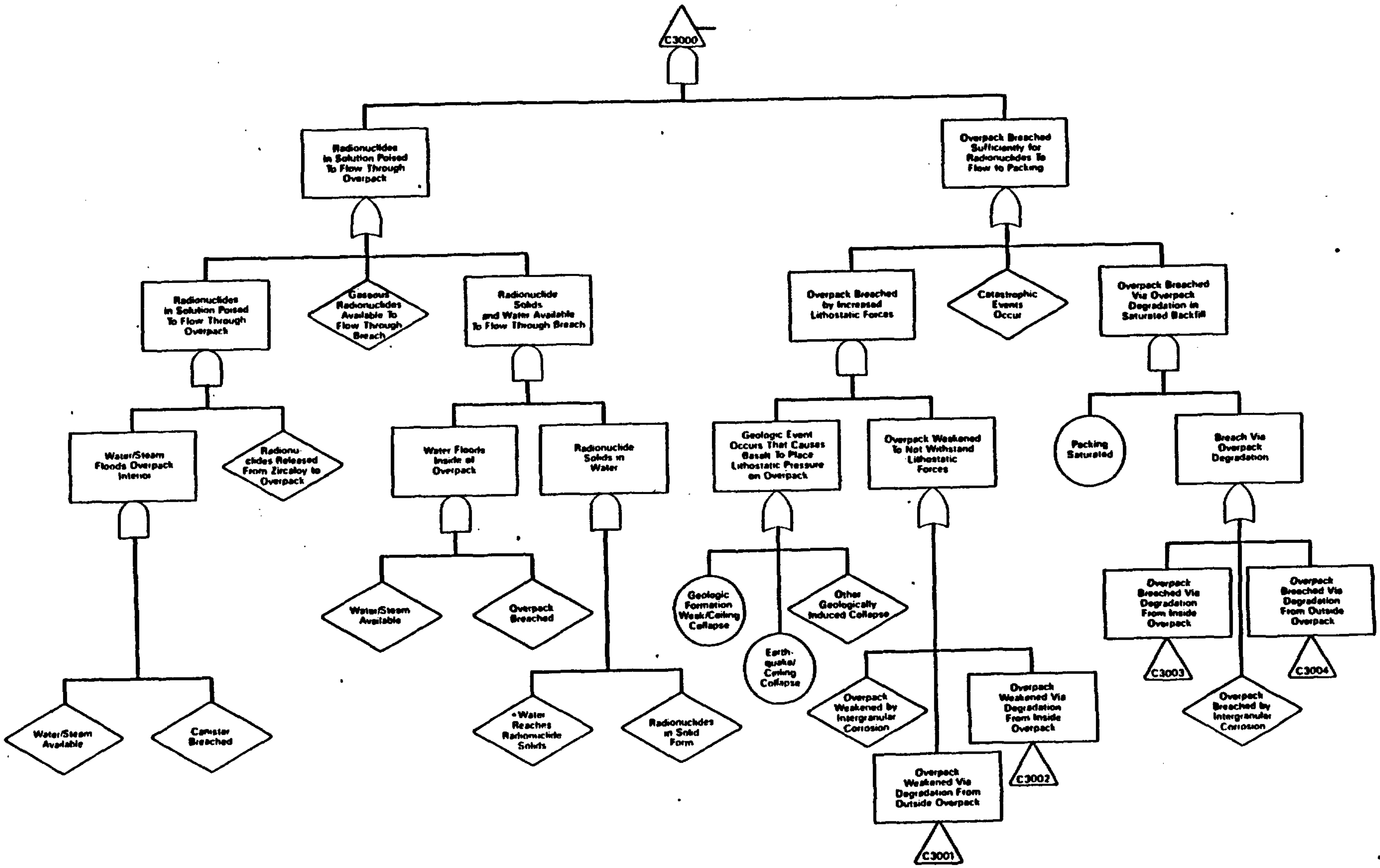

Figure B-7. C3000—Overpack Allows Radionuclides Aqueous Transport to Packing

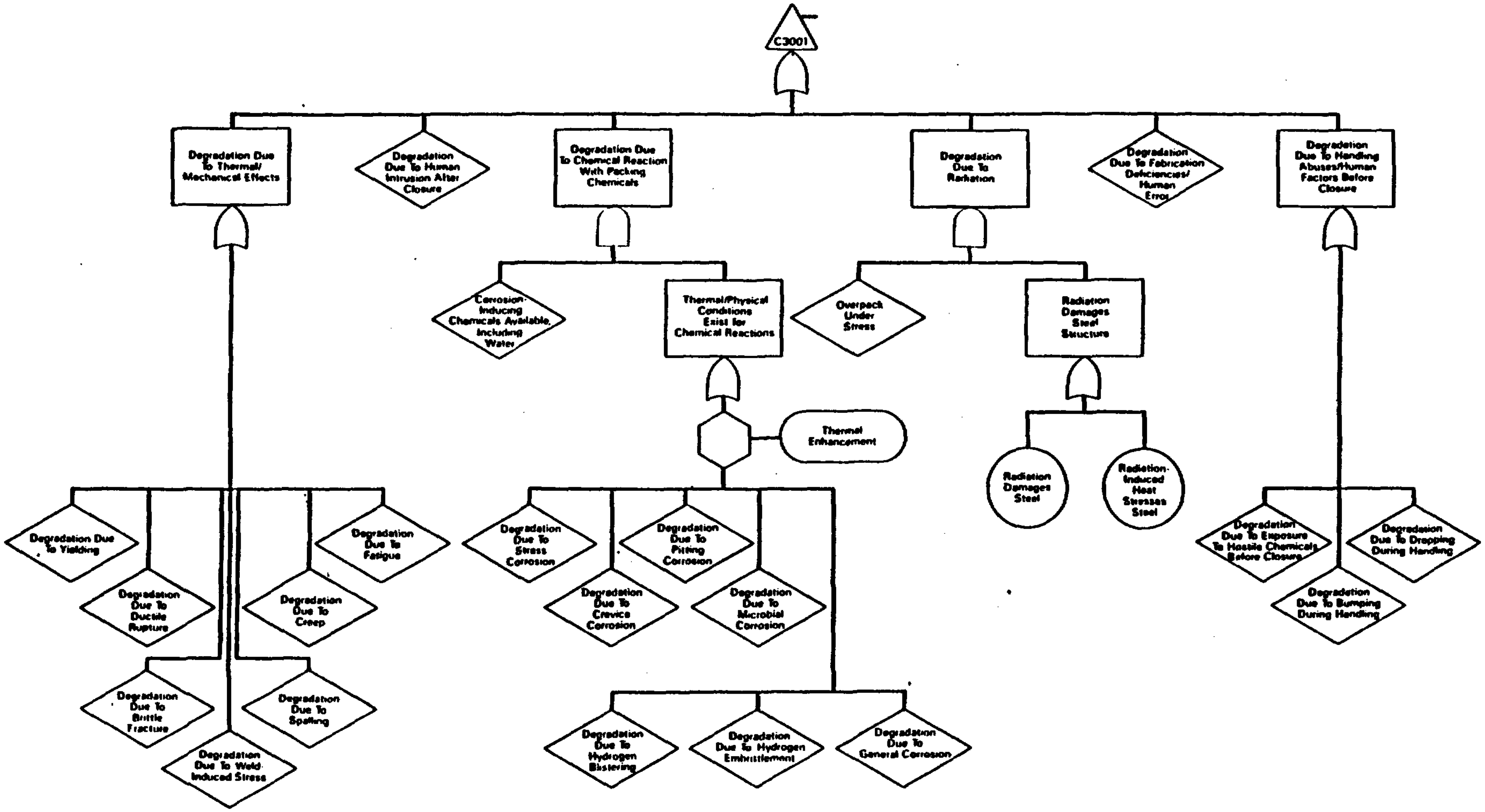

Figure B-8. C3001—Overpack Weakened Via Degradation From Outside Overpack

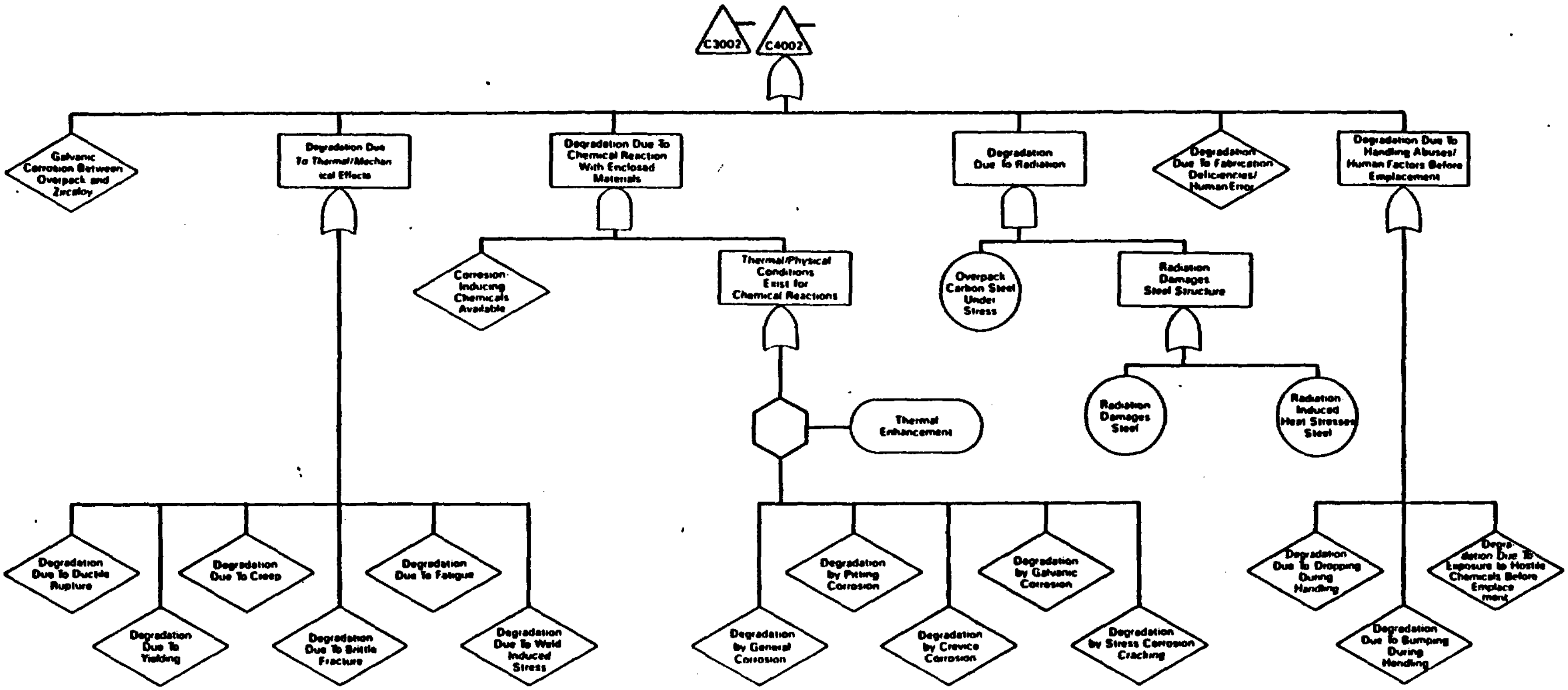

Figure 8-9. C3002 and C4002 — Overpack Weakened Via Degradation From Inside Overpack

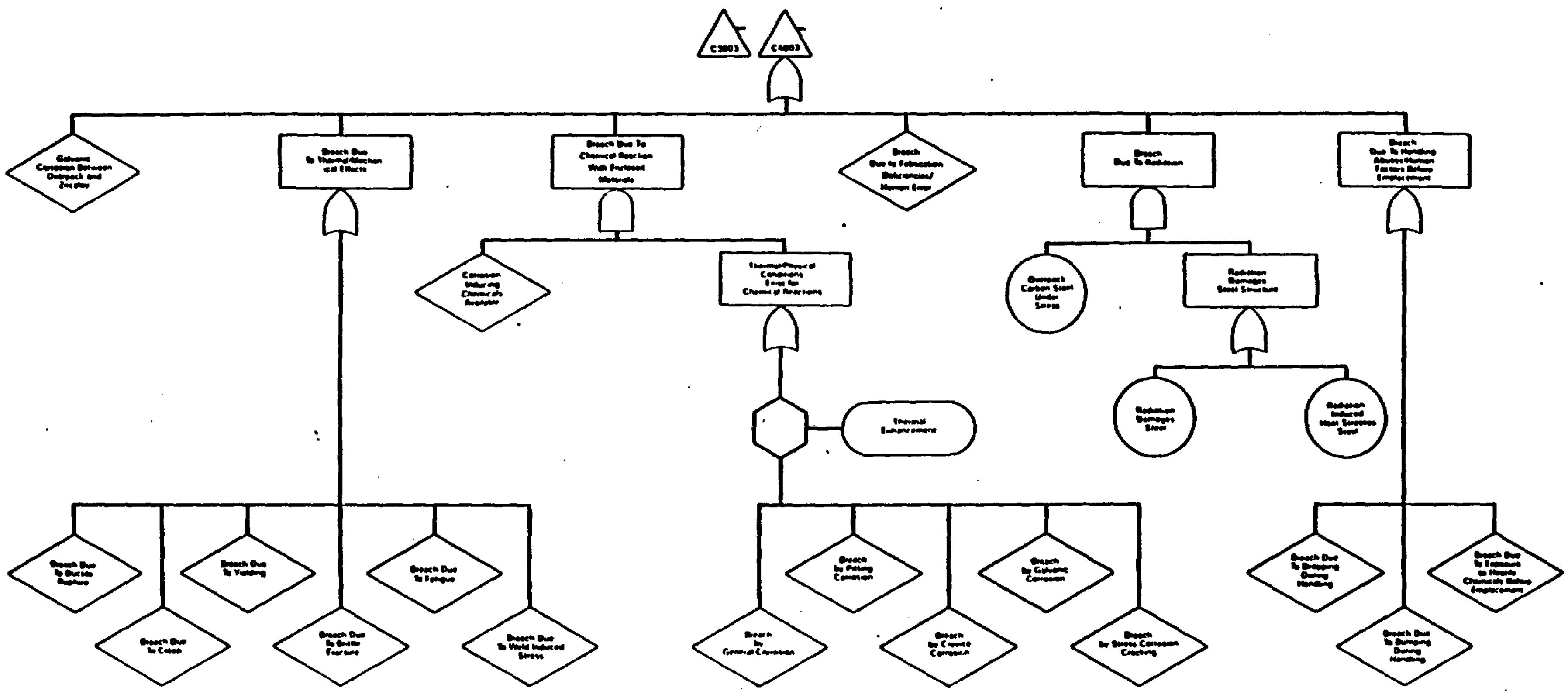

Figure 8-10. C3003 and C4003 – Overpack Breached Via Degradation From Inside Overpack

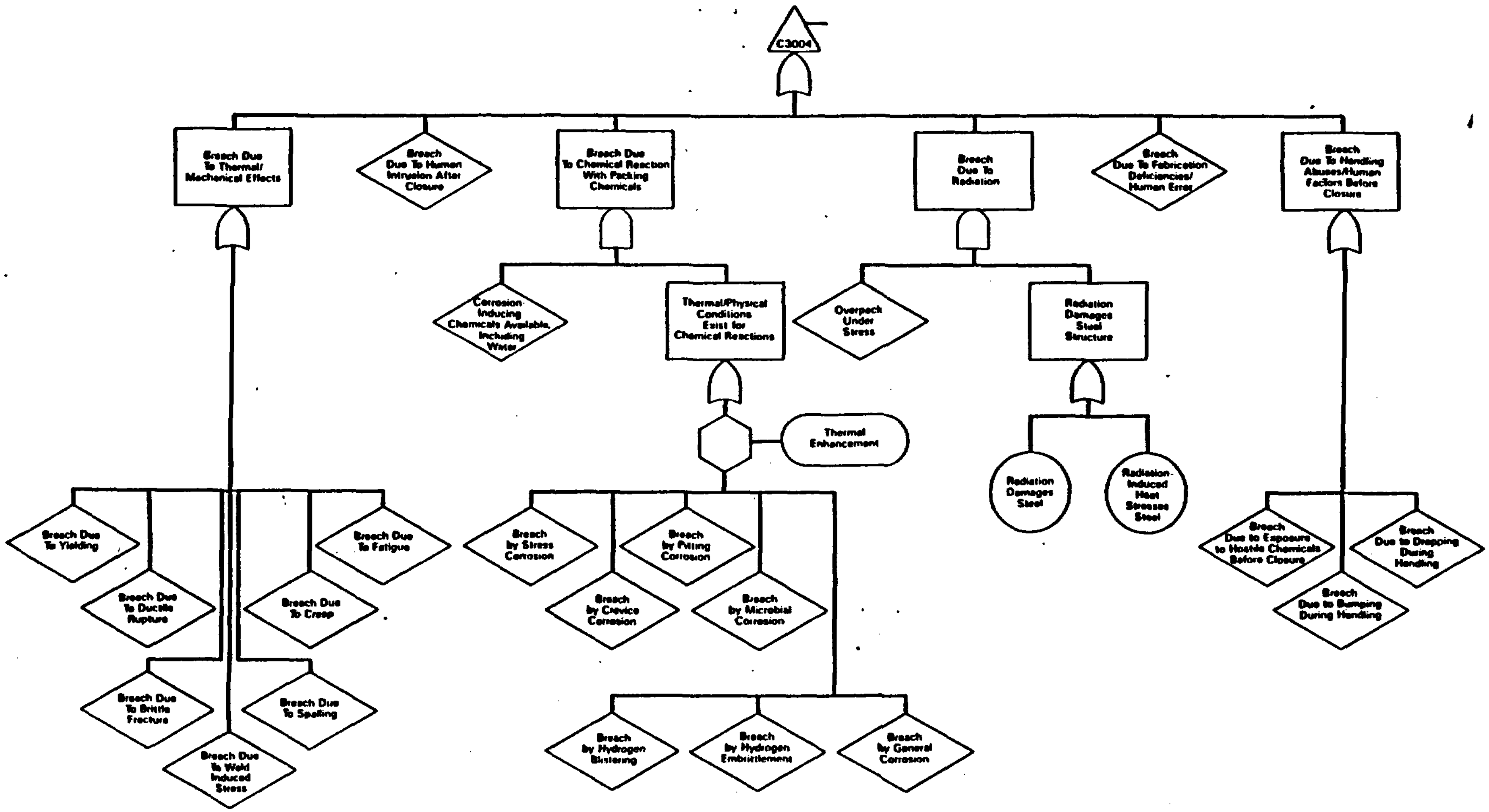

Figure B-11. C3004 — Overpack Breached Via Degradation From Outside Overpack

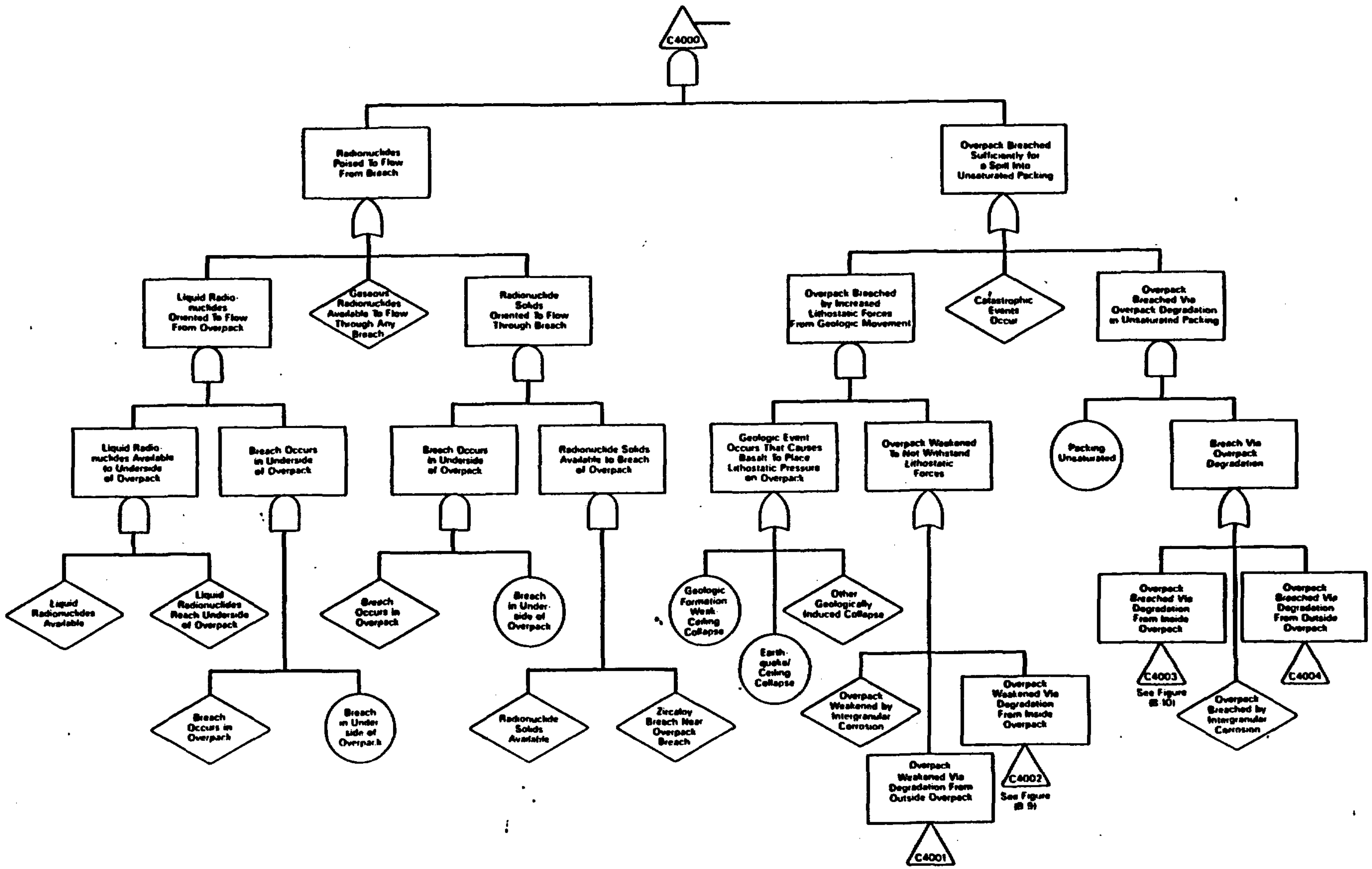

Figure B-12. C4000—Overpack Allows Radionuclide Nonaqueous Transport to Packing

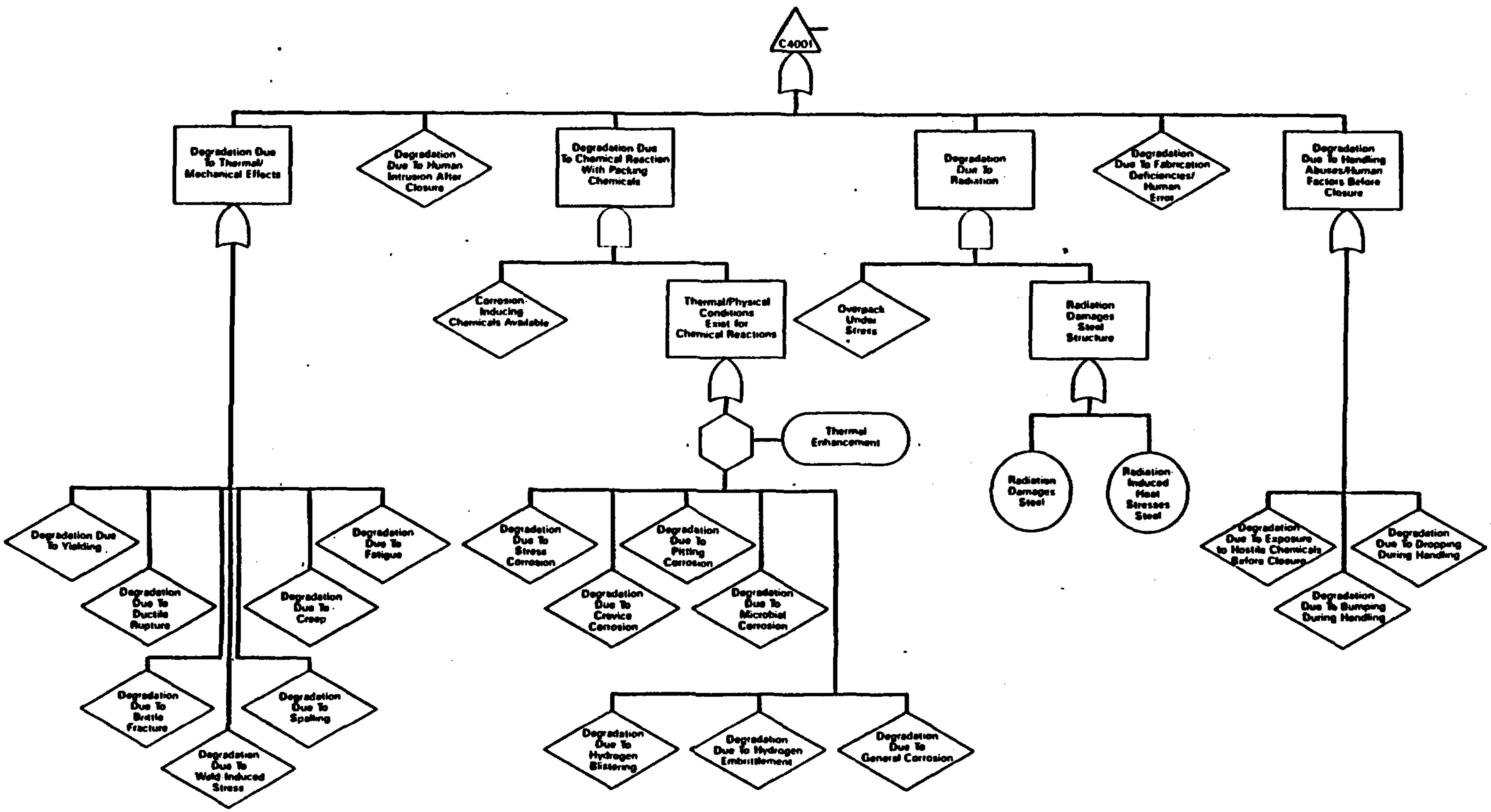

Figure B-13. C4001—Overpack Weakened Via Degradation From Outside Overpack

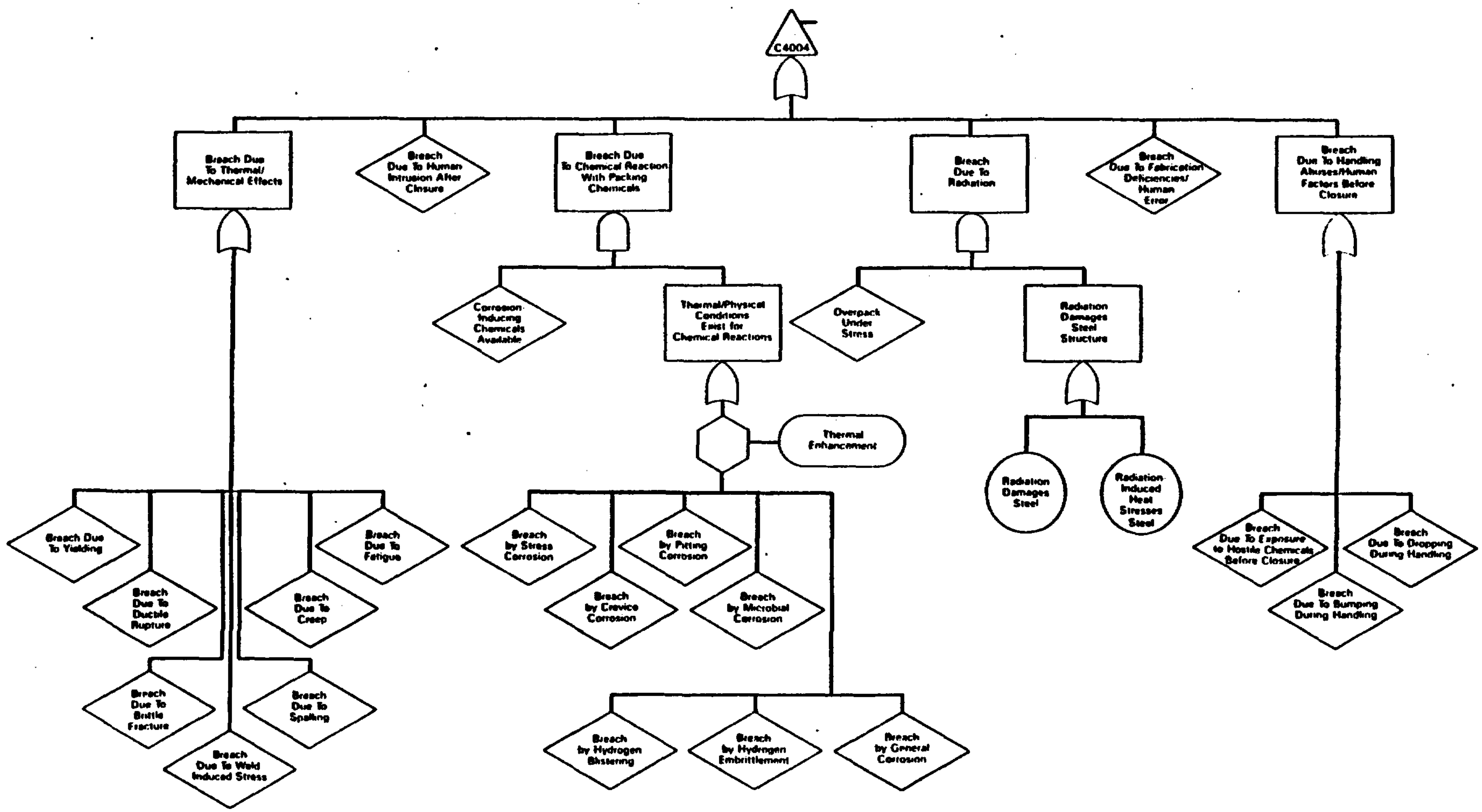

Figure 6-14. C4004—Overpack Breached Via Degradation From Outside Overpack