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April 3, 2003
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Account No. 20.06002.01.081

U.S. Nuclear Regulatory Commission
ATTN: Tamara Bloomer
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Subject: Submittal of Intermediate Milestone—Assessment of Mechanisms for Early Waste Package Failures, IM 06002.01.081.310

Reference: Completion of Intermediate Milestone—IM 06002.01.081.310 (Assessment of Mechanisms for Early Waste Package Failures) dated February 28, 2003 from T. Bloomer to V. Jain

Dear Ms. Bloomer:

Enclosed is the subject report incorporating NRC comments provided via e-mail on February 27, 2003. The revised report also includes additional information on human error probabilities in Chapter 4. Early failure of some waste packages may lead to an early release of radionuclides and provide a mechanism for water ingress into the failed waste packages. This review of the U.S. Department of Energy (DOE) report indicates that although the qualitative aspects of waste package degradation may be obtainable from historical data, any quantification of degradation rates or waste package failures will necessarily contain large uncertainties. Furthermore, special care may be required to adequately quantify uncertainties related to potential degradation mechanisms with long incubation times. Factors with the potential to cause early failures include human error and equipment failure. These factors could lead to, among other pitfalls, the generation of flaws during manufacturing and welding, improper use of weld material or heat treatment, and contamination that could result in early failures. These failure modes could have significant consequences. The review also indicates that most analyses contained in the DOE report require significant revision and additional analyses that may result in significantly larger probabilities for early failures.

If you have any questions regarding this report, please feel free to contact me (210) 522-5439.

Sincerely yours,

Vijay Jain
Element Manager
Corrosion Science & Process Engineering

VJ:jg

Attachment

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CNWRA 2003-05

ASSESSMENT OF MECHANISMS FOR EARLY WASTE PACKAGE FAILURES

Prepared for

U.S. Nuclear Regulatory Commission
Contract NRC–02–02–012

Prepared by

Center for Nuclear Waste Regulatory Analyses
San Antonio, Texas

March 2003

CNWRA 2003-05

ASSESSMENT OF MECHANISMS FOR EARLY WASTE PACKAGE FAILURES

Prepared for

U.S. Nuclear Regulatory Commission
Contract NRC–02–02–012

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PREVIOUS REPORTS IN SERIES

Number	Name	Date Issued
CNWRA 91-004	A Review of Localized Corrosion of High-Level Nuclear Waste Container Materials—I	April 1991
CNWRA 91-008	Hydrogen Embrittlement of Candidate Container Materials	June 1991
CNWRA 92-021	A Review of Stress Corrosion Cracking of High-Level Nuclear Waste Container Materials—I	August 1992
CNWRA 93-003	Long-Term Stability of High-Level Nuclear Waste Container Materials: I—Thermal Stability of Alloy 825	February 1993
CNWRA 93-004	Experimental Investigations of Localized Corrosion of High-Level Nuclear Waste Container Materials	February 1993
CNWRA 93-014	A Review of the Potential for Microbially Influenced Corrosion of High-Level Nuclear Waste Containers	June 1993
CNWRA 94-010	A Review of Degradation Modes of Alternate Container Designs and Materials	April 1994
CNWRA 94-028	Environmental Effects on Stress Corrosion Cracking of Type 316L Stainless Steel and Alloy 825 as High-Level Nuclear Waste Container Materials	October 1994
CNWRA 95-010	Experimental Investigations of Failure Processes of High-Level Radioactive Waste Container Materials	May 1995
CNWRA 95-020	Expert-Panel Review of the Integrated Waste Package Experiments Research Project	September 1995
CNWRA 96-004	Thermal Stability and Mechanical Properties of High-Level Radioactive Waste Container Materials: Assessment of Carbon and Low-Alloy Steels	May 1996
CNWRA 97-010	An Analysis of Galvanic Coupling Effects on the Performance of High-Level Nuclear Waste Container Materials	August 1997
CNWRA 98-004	Effect of Galvanic Coupling Between Overpack Materials of High-Level Nuclear Waste Containers—Experimental and Modeling Results	March 1998

PREVIOUS REPORTS IN SERIES (continued)

Number	Name	Date Issued
CNWRA 98-008	Effects of Environmental Factors on Container Life	July 1998
CNWRA 99-003	Assessment of Performance Issues Related to Alternate Engineered Barrier System Materials and Design Options	September 1999
CNWRA 99-004	Effects of Environmental Factors on the Aqueous Corrosion of High-Level Radioactive Waste Containers—Experimental Results and Models	September 1999
CNWRA 2000-06 Revision 1	Assessment of Methodologies to Confirm Container Performance Model Predictions	January 2001
CNWRA 2001-003	Effect of Environment on the Corrosion of Waste Package and Drip Shield Materials	September 2001
CNWRA 2002-01	Effect of In-Package Chemistry on the Degradation of Vitrified High-Level Radioactive Waste and Spent Nuclear Fuel Cladding	October 2001
CNWRA 2002-02	Evaluation of Analogs for the Performance Assessment of High-level Waste Container Materials	March 2002
CNWRA 2003-01	Passive Dissolution of Container Materials—Modeling and Experiments	October 2002
CNWRA 2003-02	Stress Corrosion Cracking and Hydrogen Embrittlement of Container and Drip Shield Materials	October 2002

ABSTRACT

Early failure of some waste packages may lead to an early release of radionuclides. Early failure also provides a mechanism for water ingress into the failed waste packages, which may increase the potential for criticality. This review of the U.S. Department of Energy (DOE) report (CRWMS M&O, 2000) indicates that the evaluation of potential failures and early failure mechanisms for waste packages can be based on historical data obtained from components with similar operating conditions or built using similar fabrication processes, if an adequate understanding of the origin and applicability of such information is available. Factors with the potential to cause early failures include human error, and equipment failure. These factors could lead to, among other pitfalls, the generation of flaws during manufacturing and welding, improper use of weld material or heat treatment, and contamination that could result in early failures. These failure modes could have significant consequences. The DOE should provide a technical basis to justify the use of surrogate material data in lieu of Alloy 22 and incorporate potential occurrence of unknown degradation mechanisms such as those observed in nuclear power plant operations in its model uncertainties. The use of event trees by DOE to develop probabilities is acceptable provided relevant data with sound technical bases is used to develop these event trees. The review also indicates that most analyses contained in the DOE report require significant revision and additional analyses that may result in significantly larger probabilities for early failures.

Reference

CRWMS M&O. "Analysis of Mechanisms for Early Waste Package Failure." ANL–EBS–MD–000023. Rev. 02. Las Vegas, Nevada: CRWMS M&O. 2000.

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ACKNOWLEDGMENTS

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QUALITY OF DATA: Sources of data are referenced in each chapter. CNWRA-generated data contained in this report meet quality assurance requirements described in the CNWRA quality assurance manual. Data from other sources, however, are freely used. The respective sources of non-CNWRA data should be consulted for determining levels of quality assurance.

ANALYSES AND CODES: None used.

EXECUTIVE SUMMARY

Early failure of some waste packages may lead to an early release of radionuclides. Early failure also provides a mechanism for water ingress into the failed waste packages, which may increase the potential for criticality. The U.S. Department of Energy (DOE) report (CRWMS M&O, 2000) evaluated the types of defects or imperfections that could occur in a waste package and potentially lead to its early failure. The intended use of the DOE report (CRWMS M&O, 2000) is to provide information and inputs to the DOE Total System Performance Assessment. In the report, DOE identified the types of defects applicable to waste packages, estimated the probability of their occurrence and provided a general discussion of the potential affect on the long-term performance of the waste package if a defect is present. This literature review, including the review of the DOE report, indicates that although the qualitative aspects of waste package degradation may be obtainable from historical data, any quantification of degradation rates or waste package failures will necessarily contain large uncertainties. Furthermore, special care may be required to adequately quantify uncertainties related to potential degradation mechanisms with long incubation times.

Chapter 2 provides a review of studies conducted to estimate failure probabilities of industrial components such as boiler and pressure vessels, spent nuclear fuel rods, underground storage tanks, radioactive cesium capsules, dry storage casks, and steam generator tubes. Also included in the review are a high-consequence, low-probability event from the aerospace industry and the use of nickel-base alloys in the chemical process industry, marine components, and flue-gas desulfurization plants. A discussion is provided about the use of surrogate materials and the need to incorporate potential occurrence of unknown future degradation mechanisms based on the nuclear reactor experience. DOE has not developed an adequate technical basis to justify the use of surrogate material data in lieu of Alloy 22, and incorporated possibility of future unknown degradation mechanisms in its model uncertainties.

Chapter 3 provides a review of key factors used in the assessment of initial failures, and relevant parameters are discussed. These parameters include human error probabilities, equipment failure rates, and reliability parameters associated with inspections. The DOE report also provided insight into the types of failures that led to breakdowns in these components. To achieve a lower failure rate, careful control and analysis of these failure mechanisms are required. A review of the DOE report indicates the human error probabilities in many analyses are in error because of incorrect use of data in NUREG/CR–1278 (Swain and Guttmann, 1983). Although human error probability tables in NUREG/CR–1278 (Swain and Guttmann, 1983) are routinely used in nuclear power plant probabilistic risk analysis, human error probabilities are generic values based on evaluating a wide range of experimental data and are not intended to be used in isolation. Also, some of the activities especially important to repository operations are not well covered in the technique for human error rate prediction methodology, which was developed primarily to analyze operating reactors. Additional studies are needed to expand the technique for human error rate prediction tables with values obtained from fuel handling facilities and operations. The systematic methodology described in NUREG/CR–1278 (Swain and Guttmann, 1983) should be followed. Also, equipment failure rates must address all equipment to be used in the waste package fabrication process, and additional information regarding the ultrasonic technique is needed to adequately assess the reliability of the inspection.

A review of the accumulated historical data about similar types of containers indicates that potential defects arising from the generation of flaws during manufacturing and welding, the use

of improper weld materials, improper heat treatments, inadequate weld design, handling damage, and potential contamination are important. Chapter 4 provides a review of the DOE information and the Center for Nuclear Waste Regulatory Analyses (CNWRA) assessment and analysis of applicable defects associated with early failures. While the use of event tree approach by DOE to estimating early failures is acceptable, additional analyses should be performed in areas that could significantly affect the estimated number of early waste package failures. For example, because the welding and heat treatment of the outer lid of the waste package are remote operations, it is highly unlikely the sequence of operations used by DOE for developing an event tree to estimate the probability of improper heat treatment is applicable to outer-lid closure welds. Also, the DOE report did not provide an event-tree sequence for improper weld material. Either improper weld material or improper heat treatment could result in affecting waste package performance. The DOE report lists major assumptions that would affect estimates of early waste package failures. Several of these assumptions need to be justified. For example, the assumption that the frequency of occurrence of weld flaws could be based on data collected using the expert system-based simulation RR-PRODIGAL code reflects a lack of understanding of the key components of operation and of the sensitivities of the code. Applying the results of the RR-PRODIGAL code simulation analysis for nuclear piping published by Khaleel, et al. (1999) to the early failure analysis of Alloy 22 container material requires additional technical justification. Furthermore, DOE assumed the simulation results are bounding and conservative by a factor as large as 10 based on the analysis presented by Simonen and Chapman (1999). Simonen and Chapman (1999) based their analysis on the measurement of weld flaws greater than 4 mm [0.16 in] in depth in pipes and vessels installed in U.S. nuclear power plants and showed that the RR-PRODIGAL code simulations are conservative compared to observed weld flaw frequencies. Analysis provided by DOE, however, ignores the inclusion of the small flaws that could result in a nonconservative estimate. The CNWRA review of the validation data for the RR-PRODIGAL code (Chapman and Simonen, 1998) indicates that RR-PRODIGAL code significantly underestimates the number of small flaws. The validation data showed that, for the simulation data to match the experimental data, small cracks less than 3 mm [0.12 in] in size, which account for more than 90 percent of the cracks, have to be ignored. Therefore, the DOE assumption results in a nonconservative flaw frequency distribution of one to two orders of magnitude. The DOE use of a simulation scenario with no experimental verification is inappropriate for estimating the flaw size distribution in Alloy 22. The review also indicates that most analyses contained in the DOE report require significant revision and additional analyses that may result in significantly larger possibilities for early failures.

References

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CRWMS M&O. "Analysis of Mechanisms for Early Waste Package Failure." ANL–EBS–MD–000023. Rev. 02. Las Vegas, Nevada: CRWMS M&O. 2000.

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Swain, A.D. and H.E. Guttmann. NUREG/CR-1278, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications: Final Report." Washington, DC: NRC. August 1983.

1 INTRODUCTION

The current waste package design has a corrosion resistant Alloy 22 outer container surrounding a Type 316 nuclear grade stainless steel inner container providing structural integrity (CRWMS M&O, 2001). A dual lid design is proposed for closure of the outer Alloy 22 container. The inner lid will be welded and laser peened to provide compressive stresses in the weld region. The outer lid will be secured with a deep U-groove weld joint with several passes and a cover pass, which is induction annealed to provide compressive stresses in the weld region. While the U.S. Department of Energy (DOE) is contemplating several changes in the design of the waste package that includes use of a flat final closure lid and use of inner low-plasticity burnishing as a stress mitigation method,[1] the DOE report about early failures (CRWMS M&O, 2000a) and the review documented in this report do not account for these changes. The review is based on a waste package design discussed in Civilian Radioactive Waste Management System Management and Operating Contractor (CRWMS M&O) 2001. Prior to emplacement in the repository, a remote nondestructive inspection will be performed (CRWMS M&O, 2001).

Early failure of some waste packages may lead to an early release of radionuclides. Early failure also provides a mechanism for water ingress into the failed the waste package, which is essential for the occurrence of criticality (CRWMS M&O, 2001). Early waste package failure is defined as a failure of a waste package caused by manufacturing or handling induced defects at a time earlier than would be predicted by degradation models for a defect-free waste package (CRWMS M&O, 2000a).

For the undisturbed repository, corrosion is considered the primary degradation process of the engineered barriers (NRC, 2002). Engineered barriers can degrade as a result of disruptive events, however, as presented in Figure 1-1. With the exception of igneous activity, DOE has screened out all potential disruptive events from consideration of the repository total system performance assessment based on either low-probability or low-consequence arguments (NRC, 2002). Center for Nuclear Waste Regulatory Analyses (CNWRA) studies on Alloy 22 indicate that breach of a waste package by passive or localized corrosion in the anticipated repository environments is highly unlikely within the 10,000-year performance period.[2] In the heat-affected zone around welds, however, resistance to localized corrosion may be reduced significantly leading to early waste package failure. Furthermore, the combination of residual tensile stresses, defects, and corrosive environment could lead to stress corrosion cracking or degradation of mechanical properties of welds.

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[1]Cogar, J.A. "Overview of the Design." *Presentation at the Nickel Development Workshop #5 on the Fabrication, Welding, and Corrosion of Nickel Alloys and Other Materials for Radioactive Waste Containers October 16–17, 2002.* Las Vegas, Nevada. 2002.

[2]Dunn, D.S., O. Pensado, C.S. Brossia, G.A. Cragnolino, N. Sridhar, and T.M. Ahn. "Modeling Corrosion of Alloy 22 as a High-Level Radioactive Waste Container Material." Proceedings of the Prediction of Long-Term Corrosion Behavior in Nuclear Waste Systems International Workshop, European Federation of Corrosion Event No. 256, Cadarache, France, November 26–29, 2001. In press. 2002.

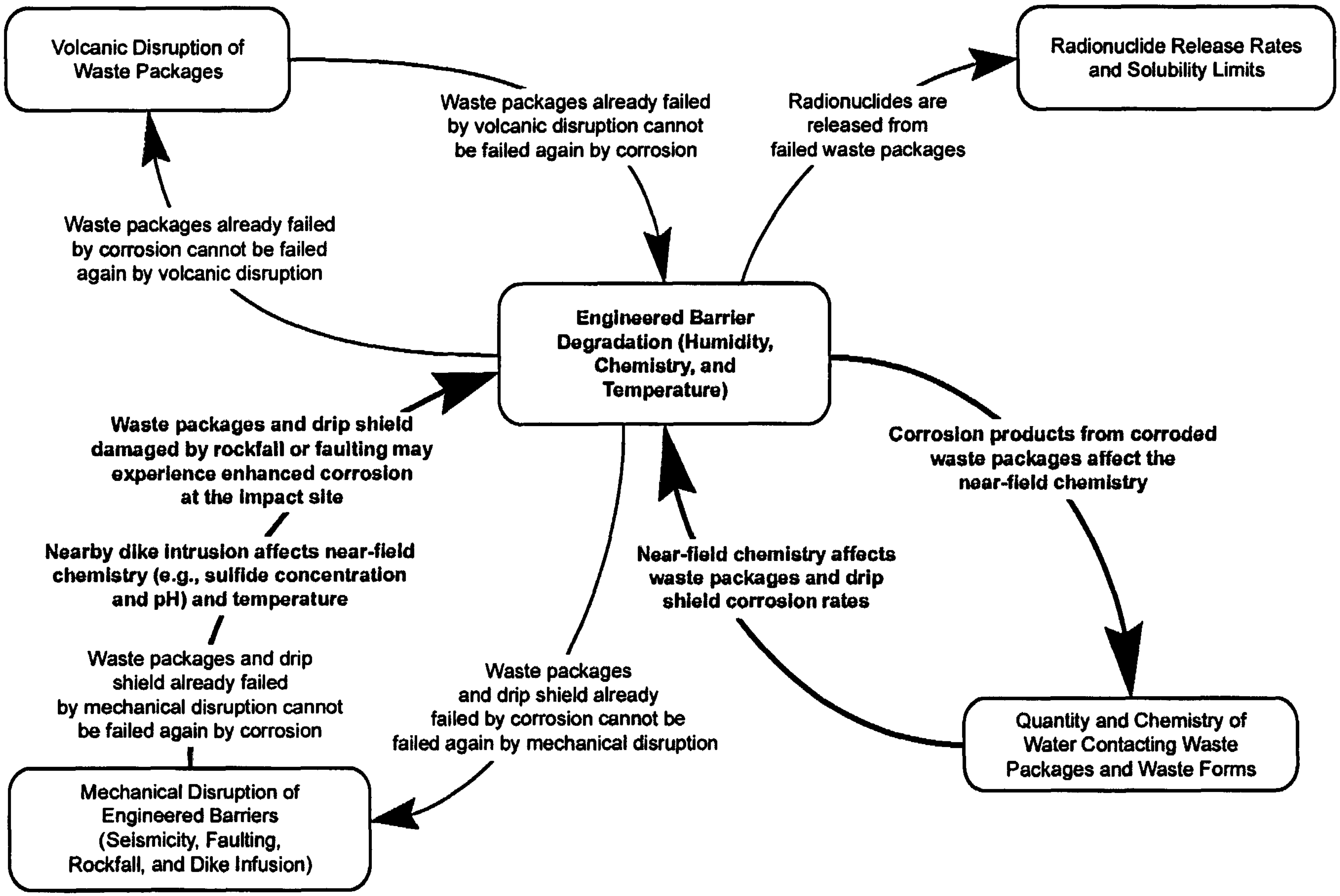

Figure 1-1. Diagram Illustrating the Relationship Between Engineered Barrier Degradation and Other Integrated Subissues

DOE evaluated the types of defects or imperfections that could occur in a waste package and potentially lead to its early failure. The intended use of the DOE report is to provide information and inputs to the DOE Total System Performance Assessment. In the report, DOE identified the types of defects applicable to waste packages, estimated the probability of occurrence and provided a general discussion of the effect on the long-term performance of the waste package if the defect is present. In the outer-lid closure weld of a waste package, weld flaws originate from defective material, inadequate welding process and technology, improper welding skills for remote welding, and unreliable equipment. This report reviews the DOE methodology for estimating early waste package failures using simulation data from the expert-system based simulation RR-PRODIGAL code and event-tree analysis to quantify probabilities of failure for various manufacturing defects (CRWMS M&O, 2000a). In addition, this report provides a review of other available information, and an analysis of weld flaws for the outer-lid closure weld.

1.1 Relevant DOE and NRC Agreement

Early waste package failures are considered in Subissue 5 of the Container Life and Source Tem Key Technical Issue (NRC, 2001) as shown in Table 1-1. This issue is considered closed-pending according to NRC and DOE agreements.

Table 1-1. DOE and NRC Agreement Related to Initial Failure

Agreement	Agreement Statement
CLST 5.03	DOE will provide an updated technical basis for screening criticality from the post-closure performance assessment. The technical basis will include (i) a determination of whether the formation of condensed water could allow liquid water to enter the waste package without the failure of the drip shield, and (ii) an assessment of improper heat treatment, if it is shown to result in early failure of waste packages, considering potential failure modes. The documentation of the technical basis is comprised of (i) Analysis of Mechanisms for Early Waste Package Failure AMR, (ii) Probability of Criticality Before 10,000 years calculation, and (iii) Features, Event, and Process System Level and Criticality AMR.

2 REVIEW OF INDUSTRIAL FAILURES AND FAILURE MECHANISMS

This section provides a review of studies conducted to estimate failure probabilities of industrial components in the aerospace industry, chemical process industry, marine, and flue-gas desulfurization facilities. In addition, a discussion is provided about the use of surrogate materials and the need for incorporating possibilities of unknown future degradation mechanisms based on the nuclear reactor operating experience.

2.1 Failure of Industrial Components

In 1994, the Center for Nuclear Waste Regulatory Analyses (CNWRA) conducted an in-depth review that focused on field engineering experience with structural materials (Tschoepe, et al., 1994). The summary provided in this section is extracted from Tschoepe, et al. (1994) and the U.S. Department of Energy (DOE) report about initial waste package failures (CRWMS M&O, 2000a). The information presented in this section can be used as a guide for determining the probability of failures for various errors that could occur during the fabrication and qualification of waste packages. Failure histories of components with similar welding and qualification backgrounds are available for boilers and pressure vessels, nuclear fuel rods, underground storage tanks, radioactive cesium capsules, and dry storage casks for spent nuclear fuel.

2.1.1 Boilers and Pressure Vessels

The construction of boilers and pressure vessels is similar to the construction of waste packages because they are welded, metallic components of similar thickness typically fabricated according to accepted standards and inspected prior to entering into service. In addition, there have been several statistically significant studies about the number and types of failures that occurred in a fairly large population. Information about estimated failure rates is provided in Table 2-1. Boiler and pressure vessel defects, such as weld flaws, base metal flaws, use of improper material in welds, improper heat treatment of welded or cold-worked areas, improper weld flux materials, poor joint design, and contaminants are similar in nature and may be applicable to the waste packages. The failure rate data for boilers and pressure vessels, however, cannot be directly applied to the waste packages because of significant differences in operational conditions and the degree of inspection performed prior to service.

2.1.2 Nuclear Fuel Rods

Nuclear fuel rods are conceptually similar to waste packages because they are manufactured in large numbers, subjected to rigorous quality control and inspection, and have radionuclide containment as one of their primary functions. There are significant differences, however, because nuclear fuel rods (i) are simple, single-barrier components with small wall thicknesses, (ii) have significantly different operating conditions, and (iii) have a much shorter period of operation. The failure rate information presented for nuclear fuel rods, therefore, cannot be directly used to develop a probability for early waste package failures, but the information can be used to provide some guidance. It is necessary, however, to establish a distinction between operational failures and failures caused by undetected manufacturing defects. The low frequency of fuel rod failures caused by undetected manufacturing defects is attributed 100 percent to nondestructive testing. In recent years, most of the fuel rod failures for which the

Table 2-1. Estimated Failure Rates Determined from Experience with Boilers and Pressure Vessels				
Reference of Study	**Number of Failures**	**Sample Population**	**Comments**	**Estimated Failure Rate**
Doubt*	229	20,000	Vessels—all welded or forged unfired pressure vessels with wall thicknesses greater than 9.5 mm [0.37 in] and working pressure in excess of 724 kPa [105 psi], all less than 40 years old. There were 17 cases of external leakage or rupture in service identified as caused by preexisting defects in the weld or base metal or caused by incorrect material. Most failures occurred in regions where nondestructive examination was not performed. It is assumed that if nondestructive examination was performed, a large number of the critical defects would have been detected.	8.5×10^{-4} per vessel year
National Board of Boiler and Pressure Vessel Inspectors†	6,400	27,000,000	Failures were listed as caused by faulty design or fabrication.	2.4×10^{-4} per vessel year
German Databases‡	—	—	—	Between 2×10^{-6} and 8×10^{-5} per vessel year

*Doubt, G. "Assessing Reliability and Useful Life of Containers for Disposal of Irradiated Fuel Waste." AECL–8328. Chalk River, Ontario, Canada: Atomic Energy of Canada Limited. 1984.
†National Board of Boiler and Pressure Vessel Inspectors. "Incident Reports." Columbus, Ohio: National Board of Boiler and Pressure Vessel Inspectors. 1999.
‡Tschoepe, III, E., D.M. Dancer, Jr., C.G. Interrante, and P.K. Nair. "Field Engineering Experience with Structural Materials." San Antonio, Texas: CNWRA. 1994.

failure cause was known have been attributed to operational factors (Tschoepe, et al., 1994). Estimated failure rates for nuclear fuel rods are given in Table 2-2. The failure rate data for nuclear fuel rods cannot be directly applied to waste packages because of significant differences in construction and operational conditions. Manufacturing defects, such as weld flaws, base metal flaws, mislocated welds, missing welds, material out-of-specification, handling damage, and contaminants, however, also may be encountered in waste packages.

2.1.3 Underground Storage Tanks

Extensive information is available about the causes of early failure for underground storage tanks. The U.S. Environmental Protection Agency (EPA) provided data through 1987 for bare steel, clad or coated steel, and fiberglass reinforced plastic tank systems (EPA, 1987a,b). The

Table 2-2. Estimated Failure Rates from Experience with Nuclear Fuel Rods

Reference of Study	Number of Failures	Sample Population	Comments	Estimated Failure Rate
Electric Power Research Institute*	Not reported	Not reported	Utilities, vendors, and the U.S. Nuclear Regulatory Commission monitor failure of nuclear fuel rods. A large database exists, which provided nuclear fuel rod failure rates through 1985 for both pressurized water reactors and boiling water reactors.	3×10^{-4} to 6×10^{-5} per rod
Yang†	Not reported	Not reported	Improvements were made in the design and fabrication to decrease potential for failure. Failures were caused by handling damage.	4.6×10^{-4} per assembly 1.7×10^{-6} per rod
Potts and Proestle‡	47	4,734,412	Failures were caused by manufacturing defect.	9.9×10^{-6} per rod
Tschoepe, et al.§	16	570,200	Failures were caused by manufacturing defect.	Between 1.2×10^{-5} and 8.2×10^{-6} per rod

*Electric Power Research Institute. "The Technical Basis for the Classification of Failed Fuel in the Back End of the Fuel Cycle." EPRI TR–108237. Palo Alto, California: Electric Power Research Institute. 1997.
†Yang, R.L. "Meeting the Challenge of Managing Nuclear Fuel in a Competitive Environment." Proceedings of the 1997 International Topical Meeting on LWR Fuel Performance, Portland, Oregon, March 2–6, 1997. LaGrange Park, Illinois: American Nuclear Society. pp. 3–10. 1997.
‡Potts, G.A. and R.A. Proestle. "Recent GE BWR Fuel Experience." Proceedings of the 1994 International Topical Meeting on Light Water Reactor Fuel Performance, West Palm Beach, Florida, April 17–21, 1994. LaGrange Park, Illinois: American Nuclear Society. pp. 87–95. 1994.
§Tschoepe, III, E., D.M. Dancer, Jr., C.G. Interrante, and P.K. Nair. "Field Engineering Experience with Structural Materials." San Antonio, Texas: CNWRA. 1994.

dominant factor in underground storage tank leakage is overfilling and leaking at attached piping. A significant number of cases of tank failures also have been reported. Approximately 95 percent of the failures have occurred in bare steel tanks as a result of corrosion. Many tanks evaluated in the studies showed through-wall corrosion holes that were plugged by corrosion byproducts and, consequently, did not leak. Data from EPA suggest an upper bound of 0.04 percent of the fraction of the population of underground storage tanks initially failed because of unidentified manufacturing defects. The failure rate data for underground storage tanks cannot be directly applied to waste packages because underground storage tanks are single shell, less robust, noncorrosion-resistant barriers. Commercial grade quality controls, however, can produce components with a relatively low rate of unidentified defect entering service. Various types of manufacturing defects that may be found in waste packages are weld flaws and handling damage.

Historical data show the containers were routinely subjected to varying degrees of inspections. These inspections allow mitigative actions to be taken if unanticipated causes of failure arise. The discussion in the DOE report (CRWMS M&O, 2000a) about the underground storage tanks illustrated the complexities in transferring error rates from one industry to other. Most underground storage tanks contained petrochemicals and generally leaked prematurely because of failure of the cathodic protection systems. In the mid-1980s, more stringent EPA regulations regarding underground storage tanks were enacted, and as a result, most underground storage tank owners replaced steel underground storage tanks with fiberglass tanks. The DOE report states that for one study, "5 to 7 percent of bare steel tanks leaked when they were tested for the first time due to manufacturing or installation defects." The DOE report then concluded that most of these initial defects would be discovered and repaired so the fraction of the population failed by unidentified defects would be much lower, and the report identified a different study in which manufacturing defects were closer to 0.0003 percent. These data reflect the need of

- Identifying repair criteria, repair processes, and applicable quality assurance.

- Using only relevant data from industries that operate under differing conditions with differing quality assurance criteria. For example, in some industries, containers may be able to tolerate larger manufacturing defects than would be permissible for a waste package and, therefore, may not recognize certain flaws as defects.

- Recognizing that a number of manufacturing defects may reduce with time as waste package manufacturing and other waste package handling operations incorporate lessons learned from startup operations.

2.1.4 Radioactive Cesium Capsules

At the DOE Hanford facility, 1,600 radioactive cesium capsules were fabricated between 1974 and 1983 for use as sources by commercial companies (Tschoepe, et al., 1994). There is only one known failure partly attributed to storage conditions that were drastically difficult from those for which capsules were designed; therefore, the failure rate was 6.3×10^{-4} per capsule. The west encapculation and storage facility fact sheet (http://www.hanford.gov/wastemgt/doe/files/WESF_Fact_Sheet_Final.pdf) (DOE, 2002) indicates that 23 capsules have been overpacked to ensure their integrity. This more recent information may increase the failure rate cited by a factor of 23.

2.1.5 Dry Storage Casks for Spent Nuclear Fuel

The concept of a dry storage cask was initiated in the late 1970s and early 1980s to meet the growing need for spent nuclear fuel storage. The dry storage casks allows spent nuclear fuel that has already been cooled in the spent nuclear fuel pool for at least 1 year to be surrounded by inert gas inside a container called a cask. The casks are typically steel cylinders that are either welded or bolted closed. The steel cylinder provides a leak-tight containment of the spent nuclear fuel. Each cylinder is surrounded by additional steel, concrete, or other material to provide radiation shielding to workers and members of the public. Some of the cask designs can be used for both storage and transportation. The various dry storage cask system designs approved by the U.S. Nuclear Regulatory Commission (NRC) are listed in Table 2-3. With some designs, the steel cylinders containing the fuel are placed vertically in a concrete vault; other designs orient the cylinders horizontally. Other cask designs orient the steel cylinder vertically on a concrete pad at a dry cask storage site and use both metal and concrete outer cylinders for radiation shielding. The first dry storage installation was licensed by NRC in 1986 at the Surry Nuclear Power Plant in Virginia. Dry cask storage represents the closest analog to waste package. There have been no recorded cases of closure weld failure after casks were

Table 2-3. Dry Spent Nuclear Fuel Storage Designs Approved by NRC for General Use
(http://www.nrc.gov/waste/spent-fuel-storage/designs.html)

Vendor	Storage Design Model	Certificate of Compliance Issue Date	Docket
General Nuclear Systems, Incorporated	CASTOR V/21	08/17/1990	72.1000
Westinghouse Electric	MC–10	08/17/1990	72.1001
NAC International, Inc.	NAC S/T	08/17/1990	72.1002
NAC International, Inc.	NAC–C28 S/T	08/17/1990	72.1003
Transnuclear, Inc.	TN–24	11/04/1993	72.1005
BNFL Fuel Solutions Corp.	VSC–24	05/03/1993	72.1007
Transnuclear West, Inc.	Standardized NUHOMS–24P NUHOMS–52B	01/18/1995	72.1004
Holtec International	HI-STAR 100	10/04/1999	72.1008
Holtec International	HI-STORM 100	05/31/2000	72.1014
Transnuclear Inc.	TN–32	04/19/2000	72.1021
NAC International, Inc.	NAC-UMS	11/20/2000	72.1015
NAC International, Inc.	NAC-MPC	04/10/2000	72.1025
BNFL Fuel Solutions	FuelSolutions	02/15/2001	72.1026
Transnuclear, Inc.	TN–68	05/30/2000	72.1027

placed into service (Hodges, 1998). Four cases have been reported, however, where cracks in the closure welds were identified during a post-weld inspection of the cask (Hodges, 1998). This information is presented in Table 2-4.

Table 2-4. Information About Cracks in Dry Storage Casks		
Location of Failure	**Description of Crack**	**Cause of Crack**
Shield lid-to-shell weld defect	Approximately 152.4 mm [6 in] long by 3.2 mm [1/8 in] deep	Preexisting condition in the rolling plane of the shell material that was opened by constructing the shield lid weld. The defect may have resulted from an improper repair or incomplete removal of temporary low-quality welds used to facilitate the fabrication process (e.g., an attachment weld for a strong back used to assist in the rolling of plate material)
Structural lid-to-shield weld	Three cracks, each less than 25.4 mm [1 in] long, located along the center of the root pass	Poor welding technique and moisture contamination

2.1.6 Steam Generator Tubes

Steam generator tubes have a number of characteristics in common with waste packages. Both are and will be produced in large quantities under stringent quality control and surveillance programs. Steam generator tubes, however, have much shorter lives, and operating experience with steam generator tubes has been very troubling for many pressurized water reactors.[1] Steam generator tubes degrade with service time because of physical mechanisms such as corrosion and phosphate wastage, pitting, denting, wear, stress corrosion cracking, and intergranular attack. Industry efforts have been largely successful in managing degradation caused by wastage and denting, but stress corrosion cracking and intergranular attack remain as more difficult problems. The modes of steam generator tubes have evolved from phosphate wastage in the early 1970s to denting in the late 1970s and stress corrosion cracking in 1980s. The changes in the degradation modes have led to the changes in the inspection technologies. The interpretation of nondestructive examination inspection data, however, is somewhat subjective and depends strongly on the experience of the analyst. Improvements are needed in flaw sizing capability and in the probability of detecting flaws in areas of high background noise.

While steam generator tubes have significant operational failures caused by known mechanisms as discussed above, stringent quality control and surveillance have significantly reduced manufacturing defects. Tschoepe, et al. (1994) cited steam generator tube failure data collected by the Atomic Energy of Canada Limited in 1981 in which a survey of 1,549,816 tubes

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[1]Chokshi, N.C. "Aging Effects on Plant Safety—Now and in the Future." *Presentation to the IAEA Technical Committee Meeting on Maximizing Aging in Nuclear Power Plants June 26–28, 2001*. Vienna, Austria. 2001.

showed defects in 4,692 tubes. Furthermore, only one tube had a defect possibly attributable to manufacturing.

2.2 Aircraft Jet Engine Components

Failure of aircraft jet engine compressor rotors and disks made of titanium alloy is provided in this section to show importance of minor defects that could result in catastrophic failures. While this is not directly applicable to early waste package failure, it provides an example of inadequate material performance leading to a high-consequence, low-probability event.

Titanium alloys, formerly processed by double-vacuum arc remelting and now by triple-vacuum arc remelting, are used for fans, compressor rotors and disks in aircraft jet engines. Occasional upsets during processing can result in the formation of metallurgical anomalies referred to as hard alpha. Although rare, these anomalies have led to engine failures that resulted in fatal consequences such as the accident in Sioux City, Iowa, in 1989. In 1991, as a result of this accident, the Federal Aviation Agency requested that industry review the available techniques to determine if a damage tolerance approach can be introduced to produce a reduction in the rate of uncontained rotor events. This enhancement was intended to supplement, not replace, the current safe-life design methodology. Southwest Research Institute®, in collaboration with four major gas turbine manufacturers, developed a probabilistically based damage tolerance design code called Design Assessment of Reliability With Inspection (DARWIN) to determine the risk of fracture of turbine engine rotor disks containing undetected material anomalies caused by hard alpha defects in titanium structures (Office of Aviation Research, 2000). DARWIN integrates finite element stress analysis results, fracture mechanics based life assessment of low-cycle fatigue, material anomaly data, probability of anomaly detection, and inspection schedules to determine the probability of rotor disk fracture as a function of applied operating cycles. The evaluation of anomaly data and probability of anomaly detection were based on 220 million engine flight cycles in a 6-year reporting period in which 3 titanium melt-related events were experienced. A design target risk of 5.0×10^{-9} events per cycle was selected for engines and 1.0×10^{-9} events per cycle was selected for components (Office of Aviation Research, 2000).

2.3 Nickel-Base Alloys

This section provides a brief review of available data on nickel base alloys and identifies the need for more information on the performance of Alloy 22 waste package material.

CNWRA conducted a review of the industrial experience with nickel-chromium-molybdenum alloys, which can be used as metal analogs for Alloy 22 (Sridhar and Cragnolino, 2002). These nickel-base alloys are widely used in marine components, geothermal energy conversion processes, the paper and pulp industry, flue-gas desulfurization plants, and waste processing applications because of their resistance to corrosion. A historical examination of the development of nickel-base alloys, indicates commercial production and use of these alloys extend to approximately 30 to 75 years if stainless steels are also included. An examination of the limited information available about the behavior of these alloys in more severe environments, such as those encountered in flue-gas desulfurization systems or the chemical processing industry, provides confidence in the preservation of the passivity of Alloy 22 for wide ranges of temperatures, potentials, and concentration of aggressive species, such as chloride, fluoride, and sulfur oxyanions. Agarwal, et al. (2000) discussed several case histories involving performance of nickel-based alloys in the chemical process industry. Service life for

components depends upon the chemical environment and the functional demand on the material and could range from a few days to several decades. Service life can be improved by a proper selection of material and design that may accommodate corrosive environments. Probability data about the failure of nickel-based alloy components, however, are not available because these components are subjected to varying degrees of corrosive environments and are designed for limited service life.

2.4 Use of Surrogate Materials

Providing technical bases for justifying the use of a surrogate material is a complex issue. NRC staff have been addressing the surrogate materials issue for reactor pressure vessel materials for several years. Commercial reactors have ongoing surveillance programs (Strosnider, et al., 1994), in which specimens of the pressure vessel steel used in the reactor are placed in surveillance capsules for testing, after exposure to the reactor core. After samples are withdrawn from the surveillance capsule, they are analyzed, and correlations are developed to determine the level of embrittlement in the reactor pressure vessel. The NRC established a regulation that effectively limits the allowable level of pressure vessel embrittlement to protect against failure because of pressurized thermal shock events.[2] Only very few plants, however, have their limiting materials in their surveillance capsules. For this reason, the issue of credible surrogate materials that could be tested in lieu of testing actual limiting materials for a reactor pressure vessel has been and continues to be investigated. The definition of a same material for plates and forgings could be a full-thickness section of material removed from the parent base metal of the same class and heat and given exactly the same post-weld heat treatment. For submerged arc welds, surrogate material could be defined as a full-thickness weld fabricated with exactly the same welding environment and given the same post-weld heat treatment. For example, a section from a vessel drop out or prolongation could be the same material if given the same post-weld heat treatment. Anything other than that, some materials including even minor variations in post-weld heat treatment and such, is not considered as the same material, and it may not qualify as surrogate material. For example, even the above description for same materials requires evaluation because it is well known that one specific location in a plate or weld may not give identical properties as those from other locations. The present approach involves determining how well surveillance specimens represent materials in a vessel as a function of the degree of matching between the important characteristics of the specimen and the vessel materials. As an example, in the case of weld metals, characteristics may include

- Weld wire
- Welding flux
- Thickness and weld design
- Base metal
- Chemical composition
- Welding parameters
- Simultaneous or separate fabrication
- Post-weld heat treatment

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[2]Chokshi, N.C. "Aging Effects on Plant Safety—Now and in the Future." *Presentation to the IAEA Technical Committee Meeting on Maximizing Aging in Nuclear Power Plants June 26–28, 2001.* Vienna, Austria. 2001.

Even for the base metal material for reactor pressure vessels, the plant manufacturers have been identified as having statistically significant effects on embrittlement behavior. The problems associated with surrogate materials demonstrate that addressing the inherent lack of uniformity between different manufacturers should be discussed since waste packages may have different manufacturers.

The history of reactor pressure vessel development demonstrates that wide variations in material properties can exist under narrowly defined material specifications, such as materials with identical weld wire numbers but having different fluxes, post-weld heat treatment, and such. Considering this experience and the level of significance to the calculation of the waste package failure rates, the assumption in the DOE report (CRWMS M&O, 2000a) that stainless steel data would be representative or bounding for Alloy 22 requires thorough technical bases for justification. The effect of the post-weld treatment, such as induction annealing, on the closure weld must be included as well. This could be significant depending on the time lapsed in the 900 and 700 °C [1,562 and 1,292 °F] temperature range while cooling from the annealing temperature {~1,125 °C [~2,057 °F]}. Alloy 22 is highly susceptible to the precipitation of topologically closed packed phases within 5 minutes in this temperature range and, hence, to localized corrosion (Pan, et al., 2002). Expected times for a waste package to cool from 900 °C to below 700 °C is significantly more than 5 minutes. At temperatures below 700 °C, time to precipitate topologically closed packed phases is 100 or more hours.

2.5 Unanticipated Degradation Mechanisms

It is well documented (Scott, 2000; Staehle, 2000; Marston and Jones, 1992) that the materials in nuclear power plants exhibit degradation mechanisms with long incubation periods. Some degradation mechanisms that can cause failure may not be evident at the time of design and manufacture. One example of such behavior is the experience with piping at nuclear power plants. In spite of thorough inspections, it has been shown that unanticipated degradation mechanisms were identified, as presented in Table 2-5. The most recent example of initially unanticipated degradation mechanism has been the primary water stress corrosion cracking of reactor vessel head penetration nozzles. NRC efforts to redefine the large break loss-of-coolant accident will include future unknown degradation mechanisms in its analyses. Another example involves embrittlement of reactor pressure vessel materials.[3] The recently revised embrittlement trend curve model now includes a long-term, time-effect factor and has added manganese and phosphorous chemistries as input. The recent incident at Davis-Besse nuclear power plant, characterized as degradation of the reactor pressure vessel head, illustrates the significant risk associated with the coupling of two corrosion processes that are separately well defined and investigated such as intergranular stress corrosion cracking of Alloy 600 (Scott, 2000) and boric acid corrosion of carbon steel (NRC, 1988). In this case, the intergranular stress corrosion cracking of an Alloy 600 control rod drive mechanism nozzle resulted in leakage of primary water through an axial crack. Boric acid, a component of the pressurized water reactor primary water added as a chemical shim to control reactivity, promoted severe corrosion of the low-alloy pressure vessel steel leaving a deformed portion of stainless steel clad {0.61 to 0.79 cm [0.24 to 0.31 in] thick} as the only reactor pressure boundary in the 212.9-cm^2 [33-in^2] corroded area. This last process was not anticipated despite

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[3]Chokshi, N.C. "Aging Effects on Plant Safety—Now and in the Future." *Presentation to the IAEA Technical Committee Meeting on Maximizing Aging in Nuclear Power Plants June 26–28, 2001.* Vienna, Austria. 2001.

Table 2-5. Corrosion Failure Modes of Boiling Water Reactor and Pressurized Water Reactor Components Discovered During Operations			
Component	**Material**	**Corrosion Failure Mode**	**Year of First Occurrence**
Boiling water reactor and pressurized water reactor cladding	Zircaloy-2 and Zircaloy-4	Stress corrosion cracking (pellet cladding interaction failure)	1973
Boiling water reactor recirculation piping	Type 304 SS	Intergranular stress corrosion cracking	1965
Secondary side pressurized water reactor recirculating steam generator tubing	Alloy 600	Intergranular stress corrosion cracking and intergranular corrosion	1970
Pressurized water reactor recirculating steam generator tube support plate/tubing	Carbon steel/Alloy 600	Accelerated corrosion of support plate or denting of tubing	1974
Primary side pressurized water reactor steam generator tubing	Alloy 600	Intergranular stress corrosion cracking	1977
Boiling water reactor recirculation pump shafts and pressurized water reactor coolant pump shafts	Various materials	Corrosion fatigue	1970
Secondary side pressurized water reactor recirculating steam generator tubing	Alloy 600	Pitting	1980
Reactor core structural components (control rod guide tubes, core shroud, control blade, and such)	Type 304 SS	Irradiation assisted stress corrosion cracking	1984
Reactor auxiliary systems (storage, tanks, spray pond piping, makeup water tank, and such)	Carbon steel Types 304, 304L, and 316 SS	Microbially influenced corrosion	1984
Pressurized water reactor feedwater piping and wet steam lines	Carbon steel	One- and two-phase flow-assisted corrosion (erosion corrosion)	1986
Pressurized water reactor vessel head penetration nozzles	Alloy 600	Intergranular stress corrosion cracking	1991

that the volatility of boric acid is well known, and NRC alerted the utilities in the 1980s about boric acid corrosion of carbon steel reactor pressure boundary components. As with the cases involving nuclear reactor components, it should not be assumed that all future degradation mechanisms relevant to the waste packages are known at this time. DOE should incorporate potential occurrence of such unknowns in its uncertainty estimates.

3 REVIEW OF FACTORS RELEVANT TO INITIAL FAILURE

Key factors for assessing initial failures and related parameters are discussed in this section. The parameters include human error probabilities, equipment failure rates, and reliability parameters associated with inspections.

3.1 Human Error Probabilities

3.1.1 DOE Approach

Selected human error categories and their probabilities of occurrence based on the handbook by Swain and Guttmann (1983) are presented in Table 3-1. Swain and Guttmann (1983) define human error as any member of a set of human actions that exceeds some limit of acceptability. Therefore, an error is an out-of-tolerance action where the limits of tolerable performance are defined by the system. Errors are regarded as the natural outgrowth of some unfavorable combination of people and the work situation. Either the person making an error lacks sufficient skill or motivation for consistently acceptable performance or aspects of the work situation are not in accordance with what can be done reliably, or both. It is important to note that malevolent behavior is excluded from this definition of human error. Human errors include intentional and unintentional errors. Intentional errors occur when the operator intends to perform some act that is incorrect but believes it to be the correct act. In everyday language, the operator has good intentions, but the effect on the system caused by the performance may be undesirable. Unintentional error is defined as an error that simply happens; it was not intended.

In human reliability analysis, it is important to consider not only the human error but also the consequence of the human error. Human error probability is defined as the probability that when a given task is performed, an error will occur. There are many ways to estimate the human error probability; some are statistical and some are judgmental. The reliability is then given by 1-human error probability. However, this relationship becomes more complicated if task or time sequencing is considered.

Tabulated human error probabilities are described as nominal human error probabilities. A nominal human error probabilities is the probability of a given human error when the effects of plant-specific performance shaping factors have not been considered. These nominal human error probabilities are single point estimates and represent the median of a lognormal distribution. The amount of variation or uncertainty in the estimated human error probability is described as an error factor. Uncertainty in the human error probability arises from three main sources. The first source is variability in people and conditions. A second source is the uncertainty in assessing human error probability. The third source is the modeling uncertainty (i.e., How well can the human performance be modeled in a human reliability analysis application?).

According to Swain and Guttman (1983), the error factor as defined in the technique for human error rate prediction methodology results from the choice of a lognormal distribution to describe human error probabilities, the choice of the median of the distribution as the nominal human error probabilities, and designation of the 95th and 5th percentiles respectively as the upper and lower ranges of uncertainty in the human error probabilities. The symmetry displayed by the lognormal distribution is multiplicative or geometric, which is different from the additive

Table 3-1. Selected Human Error Probabilities and Error Factors			
Action	**Human Error Probability**	**Error Factor**	**Report by Swain and Guttmann* Chapter/Page**
Failure to follow written procedure during normal conditions	0.010	3	20/22
Failure to use a checklist properly	0.500	5	20/22
Error of commission by reading and recording quantitative data from an unannunciated digital display	0.001	3	20/26
Failure of checker using written procedures to find an error made by another checker	0.100	5	20/38
Failure of operator to detect a stuck manual valve with no means of position indication	0.010	3	20/30
Failure to perform rule-based action correctly when written procedure is available (no recovery factor considered)	0.050	10	20/18
Error of commission by selecting wrong control on a panel from an array of similar-appearing controls arranged in well-delineated functional group	0.001	3	20/28
Error of commission by improperly mating a connector	0.003	3	20/28

*Swain, A.D. and H.E. Guttmann. NUREG/CR-1278, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications Final Report." Washington, DC: NRC. August 1983.

symmetry displayed by the normal distribution. As an example, the expression 0.003 (0.001 to 0.01) means that the nominal human error probability is 0.003. It is believed that the true human error probability unlikely would be lower than 0.001 in more that 5 percent of the cases, nor would it be higher than 0.01 in more than 5 percent of the cases. The expression for human error probability could be restated as 0.003 (EF = 3). The lower uncertainty bound is calculated by dividing the nominal human error probability by the error factor or 0.003/3 = 0.001. The upper uncertainty bound is calculated by multiplying the human error probability by the error factor or 0.003 × 3 = 0.009 (for convenience, this can be rounded up to 0.01). The spread between the lower and upper uncertainty bounds in the handbook by Swain and Guttmann (1983) varies according to task conditions.

In general, the spread increases with small (less than 0.001) and large (more than 0.01) human error probabilities. Error factors ranging from 3 to 10 are based on judgment and should not be confused with statistical confidence levels. An example of how to use Table 3-1 for a given human error probability of misloading an assembly into a waste package is provided in Table 3-2. A value of 0.006 was calculated by a combination of errors.

3.1.2 CNWRA Review

The treatment of human error was based on NUREG/CR–1278 (Swain and Guttmann, 1983), a standard reference for human reliability analysis. NUREG/CR–1278 provides a methodology for users to develop human error probabilities and human reliability analyses. NUREG/CR–1278 (Swain and Guttmann, 1983) provides basic principles, guidelines, and numerous examples of how human behavior and performance are estimated for various situations in nuclear power plants to assist the user in performing human reliability analyses. Although human error probability tables in NUREG/CR–1278 (Swain and Guttmann, 1983) are routinely used in nuclear power plant probabilistic risk analysis, human error probabilities are generic values based on evaluating a wide range of experimental data and are not intended to be used in isolation. Also, some of the activities especially important to repository operations are not well covered in the technique for human error rate prediction methodology, which was developed primarily to analyze operating reactors. Additional studies are needed to expand the technique for human error rate prediction tables with values obtained from fuel handling facilities and operations. To develop human error probabilities, the process includes identification of task,

Table 3-2. An Example Calculation for Logical Combinations of Human Error Probabilities for Misloading an Assembly

Action	Human Error Probability	Error Factor	Probability of Error	Report by Swain and Guttmann* Chapter/Page
Misloading an assembly into a container designed to receive the assembly (operator fails to determine the adequate disposal container designed to receive the assembly)	0.05	10	0.005	20/18
Selection error (operator has determined which disposal container to use, but selects the wrong one)	0.001	1	0.001	20/28
Total probability of error for misloading an assembly	—	—	0.005 + 0.001 = 0.006	—

*Swain, A.D. and H.E. Guttmann. NUREG/CR–1278, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications Final Report." Washington, DC: NRC. August 1983.

analyses, process steps, conditions, underperformance, time for performance, human-machine interfaces, qualifications, dependencies among actions, and performance shaping factors, as well as other inputs. Direct reference to selected error categories from the example tables of NUREG/CR–1278 (Swain and Guttmann, 1983) is insufficient for identifying human error probabilities.

Development of human error probabilities for the waste package fabrication process should follow the systematic methodology of NUREG/CR–1278 (Swain and Guttmann, 1983), or the technical basis for an alternative methodology must be provided. The waste package fabrication process is sufficiently analogous to the nuclear power plant situations for NUREG/CR–1278 (Swain and Guttmann, 1983) to be applicable. The methodology of NUREG/CR–1278 (Swain and Guttmann, 1983) should be applied to specific steps of the waste fabrication process and should be described to a sufficient level of detail in the report to allow for adequate review. A stringent quality control and surveillance programs should be placed to reduce the number of manufacturing defects expected during the startup of the fabrication operations.

In summary, human error probabilities should be calculated utilizing the methodology of NUREG/CR–1278 (Swain and Guttmann, 1983), and the event trees discussed in the U.S. Department of Energy (DOE) report (CRWMS M&O, 2000a) should be revised to reflect appropriate human error probability inputs specific to the waste package fabrication process.

3.2 Equipment Failure Rates

3.2.1 DOE Approach

Failures of waste packages can be caused by equipment failures. The DOE report uses failure rates obtained from the Institute of Electrical and Electronics Engineers, Inc. (1984) as listed in Table 3-3.

Table 3-3. Selected Component Failure Rates per Hour

Component	Low	Mean	High	Institute of Electrical and Electronics Engineers, Inc.* Page Number
Heater, catastrophic, all modes	6×10^{-8}	1.3×10^{-6}	2.5×10^{-5}	283
Thermostat, all modes	1.2×10^{-6}	5.8×10^{-6}	1.7×10^{-5}	543

*Institute of Electrical and Electronics Engineers, Inc. *IEEE Std 500 Reliability Data—IEEE Guide to the Collection and Presentation of Electrical, Electronic, Sensing Component, and Mechanical Equipment Reliability Data for Nuclear-Power Generating Stations.* New York City, New York: Institute of Electrical and Electronics Engineers, Inc. 1984.

3.2.2 CNWRA Review

DOE selected two specific component failure rates from the Institute of Electrical and Electronics Engineers (1984). All equipment components involved in the waste package fabrication, however, should have been identified and assessed for predicted failure rates. Justification for excluding any specific equipment component should have been provided. In summary, insufficient information was provided to evaluate if the specified failure rates are adequate to reflect a systematic assessment of the potential for equipment failure and if the two failure rates selected reflect a systematic and adequate use of the reference data.

3.3 Reliability of Ultrasonic Examination

3.3.1 DOE Approach

Because the waste package outer-lid closure weld inspection will have to be performed in a hot cell, the final ultrasonic inspection of the closure weld will also have to be conducted remotely in the hot cell. Presently, only an ultrasonic inspection of the closure weld is scheduled. Information about the probability that an ultrasonic examination would fail to detect a given size flaw was obtained by DOE from Bush (1983) and Heasler and Doctor (1996). This information refers to the reliability of ultrasonic testing for detecting various types of weld defects. The DOE report summarizes the results of previous studies about reliability and provides parameters for a modified log normal function giving the probability of nondetection as a function of flaw depth as

$$P_{ND} = \varepsilon + 0.5 \times (1-\varepsilon) \times \mathrm{erfc}\left[\mathrm{v} \times \ln\left(\frac{a}{a^{*}}\right)\right] \tag{3-1}$$

where

P_{ND}	—	probability of nondetection
ϵ	—	the lower limit of P_{ND} [0.005, based on Bush (1983)]
erfc	—	the complementary error function
a	—	the flaw depth in centimeters
a^*	—	the characteristic flaw depth in centimeters
v	—	a nondimensional shape factor (Bush, 1983)

A more recent study about ultrasonic testing detection of cracks produced by intergranular stress corrosion cracking in stainless steel shows an improved reliability (Heasler and Doctor, 1996). The nondetection distribution for a flaw size, a, has the form

$$\mathrm{P_{ND}}(a) = 1 - [1 + \exp(-\beta_1 - \beta_2 \times a)]^{-1} \tag{3-2}$$

where

β_1 = −2.67 (Heasler and Doctor, 1996)
β_2 = 16.709 $\mathrm{cm^{-1}}$ (Heasler and Doctor, 1996)
a = the flaw depth in centimeters

The DOE report (CRWMS M&O, 2000a) indicates the probability of nondetection for flaws of various sizes is dependent on a number of variables such as the type of material, operator skill, access to the weld, and type of defect. In addition, the flaw size itself can be characterized in several ways based on length, depth, or area. Because all these variables could not be specified, the DOE report used a log normal complementary cumulative distribution function showing a 50-percent probability of nondetection for a 2.5-mm [0.1-in] flaw depth and a higher probability of nondetection for smaller flaws.

3.3.2 CNWRA Review

Concerning the reliability of inspection, the report should have discussed the three inspection technologies that will most likely be used during the waste package fabrication process (i.e., ultrasonic, radiographic, and dye-penetrant testing). The treatment of ultrasonic inspection is thorough, presumably because ultrasonic techniques have been widely used in remote applications. Ultrasonic technology is identified as the technology chosen for the inspection of the waste package closure welds. Remotely applied ultrasonic inspection systems with associated mechanical scanners and manipulators have been developed and used for more than 20 years to inspect nuclear pressure vessels and piping. Additional information regarding the ultrasonic examination process for the closure weld (i.e., sensitivity of equipment, experience and qualification of operators, presence or lack of remote visual equipment, number of intermediate ultrasonic examinations per closure weld) is needed to adequately assess reliability of the inspection. Reliability of ultrasonic detection, however, is highly dependent on geometry. For the waste package closure weld geometry, the probability of detection for large flaws would be expected to be very high.

Other methods, however, likely will be used during the waste package fabrication process. Clearly, the use of dye-penetrant testing will be limited to inspecting nonclosure welds, and inspecting the surfaces of the waste packages before loading. If surface preparation procedures are followed, dye-penetrant testing should be effective for defects greater than approximately 1.27 mm [0.05 in] long. However, DOE should determine the probability of leaving dye penetrant residue on weld surface that could give rise to defects such as porosity. Radiography is effective for volumetric defects such as voids, porosity, incomplete fusion, and inclusions. Radiography can also be effective for detecting crack and planar defects as long as the radiographic source beam is parallel to the planar defect and the film is normal to the planar defect. These conditions can be met prior to waste package closure. For the closure welds, however, these conditions cannot be met because the film would have to be on the inside of the waste package.

Several inspection methods were not discussed in the report. For example, eddy current inspection could be useful in detecting near-surface defects in the base metal, the nonclosure welds, and the closure welds. Eddy current inspection might also be useful for detecting variations in conductivity that could be caused by material contamination or residual stress (Schoening, et al., 1995; Chang, et al., 1999). Determining tensile residual stress on the surface of the waste package could be important. X-ray diffraction is a well-developed method for ascertaining the presence of residual stress. Recent developments make x-ray diffraction equipment applicable to on-site weld inspection (Physique and Industrie, 2001). To date, the emphasis for developing eddy current and x-ray diffraction techniques for measuring residual stress has been for jet engines. X-ray diffraction techniques have been used in nuclear power components. These techniques have been applied to titanium- and nickel-based alloys. The

DOE report should include information regarding effectiveness of the methods employed for relieving residual stresses from all waste package welds.

The DOE report cites detection of intergranular stress corrosion cracks by ultrasonic techniques in stainless steels. Detecting intergranular stress corrosion cracks is more difficult than detecting other weld defects because intergranular stress corrosion cracks follow predominantly the grain boundaries in the materials, and grain boundaries often have crack width much smaller than the wavelength of the ultrasonic energy used to detect the defects. Most defects that might be expected in Alloy 22 would be related to weld-type defects that should be transgranular and much easier to detect ultrasonically. The probability of missing intergranular stress corrosion cracks is higher than the probability of missing a fatigue crack or other welding defect. Therefore, the probability of not detecting a fatigue crack or welding defect will be overestimated by application of Eq. (3-2) which is applicable to intergranular stress corrosion crack detection.

The DOE report provided information about nondestructive evaluation techniques, initial flaw distributions, and flaw densities for the closure weld only. The DOE report should address these issues in detail for nonclosure welds and base metal also.

4 ASSESSMENT OF MANUFACTURING DEFECTS IN WASTE PACKAGES

Civilian Radioactive Waste Management System Management and Operating Contractor [CRWMS M&O (2001)] identified the following types of defects in its report on initial failures of waste packages.

- Weld flaws
- Base metal flaws
- Use of improper material in welds
- Improper heat treatment of welded or cold-worked areas
- Improper weld-flux material
- Poor joint design
- Contaminants
- Mislocated welds
- Missing welds
- Handling/installation damage
- Administrative/operational error

The following defects were excluded from further consideration for reasons given.

- Improper weld-flux material has been excluded from further consideration because the U.S. Department of Energy (DOE) will employ a welding method that does not use weld-flux material.

- Poor joint design has been excluded because DOE believes that a significant effort will go into the design of the final closure joint to ensure that weld designs are acceptable.

- Missing welds are expected in the spent nuclear fuel rods at a rate of 5.0×10^{-6} per rod. The missing weld in a waste package is easier to identify than in a spent nuclear fuel rod, and it would have a noticeable effect on the configuration of the waste package. Therefore, it is expected the occurrence rate of this defect will be below the threshold probability of 10^{-8} per waste package (CRWMS M&O, 2000a).

- Mislocated welds is only applicable to very small, single-pass welds. For large multipass welds, any significant mislocation would cause the weld arc not to strike, which would be evident to the operator and the control system of the automatic welder. Hence, this type of defect is not applicable to waste packages.

Review of the DOE information and the Center for Nuclear Waste Regulatory Analyses (CNWRA) assessment and analysis for the applicable defects is provided in the following sections. It should be noted that the DOE report did not include out-of-specification material as a source of defect in the waste packages.

4.1 Weld Flaws

In a welding process, various types of weld flaws, as shown in Figure 4-1, can originate because of defective material, inadequate welding process and technology, poor remote welding skills, unreliable equipment, and poor inspections (Chapman and Simonen, 1998). As

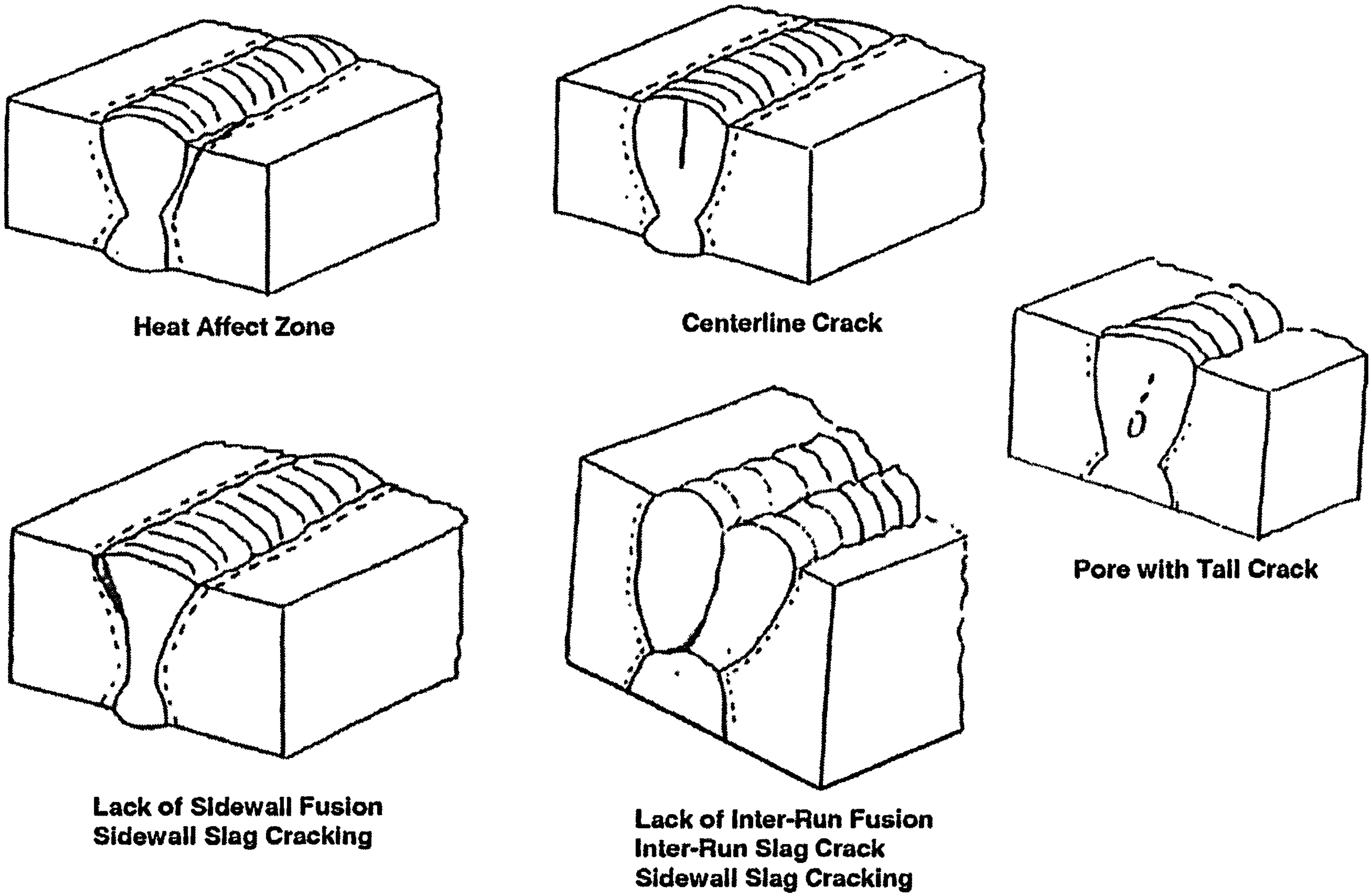

Figure 4-1. Types of Crack-Like Defects (Chapman and Simonen, 1998)

discussed by Chapman and Simonen (1998) and summarized next, these weld flaws can be described as follows. Centerline cracking results from the formation of low strength or low melting point phases caused by the collection of impurities at the top of the weld bead during solidification. The stresses present on the surface then cause a centerline crack along the weld bead. Automatic high-speed welding techniques, such as submerged arc welding, show a greater tendency to centerline cracking than do manual techniques. The fill runs of large multipass welds are less susceptible than root welds.

Heat-affected zone cracking is caused by the combination of absorption of hydrogen during cooling and formation of hardened structure in the heat-affected zone. The risk of heat-affected zone cracking is higher for the low heat input method and thicker joints because of the low level of hydrogen diffusivity. There are higher chances of the heat-affected zone cracking in thicker weld joints and in multipass welds. Also, higher restraint may result in a higher number of heat-affected zone cracks. Heat-affected zone cracking is more likely in high-carbon ferritic steels.

Lack of fusion flaws result from lack of union between weld metal and parent plate or between successive weld runs. Chances of lack of fusion are higher in narrower or deeper weld grooves. In addition, thicker sections or limited accessibility of electrodes result in a greater chance of lack of fusion flaws.

Nonmetallic inclusions occur because of incomplete slag removal between weld runs or slag laminations within the parent plate. Mill scale, rust, or damaged electrode coatings could also cause formation of such inclusions. Slag inclusions are common in submerged arc welding and manual metal arc welding, while oxide inclusions are common in tungsten inert gas welding deposits. Nonmetallic inclusions are more likely in tighter and thicker weld joints. A thorough removal of slag between the runs of a multipass weld reduces the amount of nonmetallic inclusions.

Porosity is caused by the gas-forming elements present in the welded joint. These gaseous phases evolve as the weld is cooled and form cavities. Porosity is caused by moisture, rust, grease on the plate surface, and oxygen or nitrogen from the atmosphere or shielding gas. Isolated porosity areas could be attributed to an unstable arc. In addition, interdendritic or shrinkage porosity could occur at stop–start positions. Fluxless processes such as tungsten inert gas welding are more susceptible to porosity compared with fluxed processes such as submerged arc and manual metal arc welding. Also, lack of clean surfaces is a frequent cause of porosity.

Although weld defects are common, and, if detected, they are often mitigated by rewelding, there has been a limited number of research projects focused on the systematic analysis of defects in a weld. Available information is limited to studies about size and distribution of weld flaws that were conducted using the RR-PRODIGAL crack simulation code (Chapman and Simonen, 1998) for reactor pressure vessels, and nuclear piping.

The RR-PRODIGAL code is a crack simulation software that reproduces the initiation and interaction of defects observed in piping and vessels. The software couples Monte Carlo simulations with a knowledge base developed by interactions with welding experts. Figure 4-2 shows a schematic drawing of a weld buildup and the positioning of different types of defects presented in Figure 4-1. A predefined set of distributions is used to select depth and

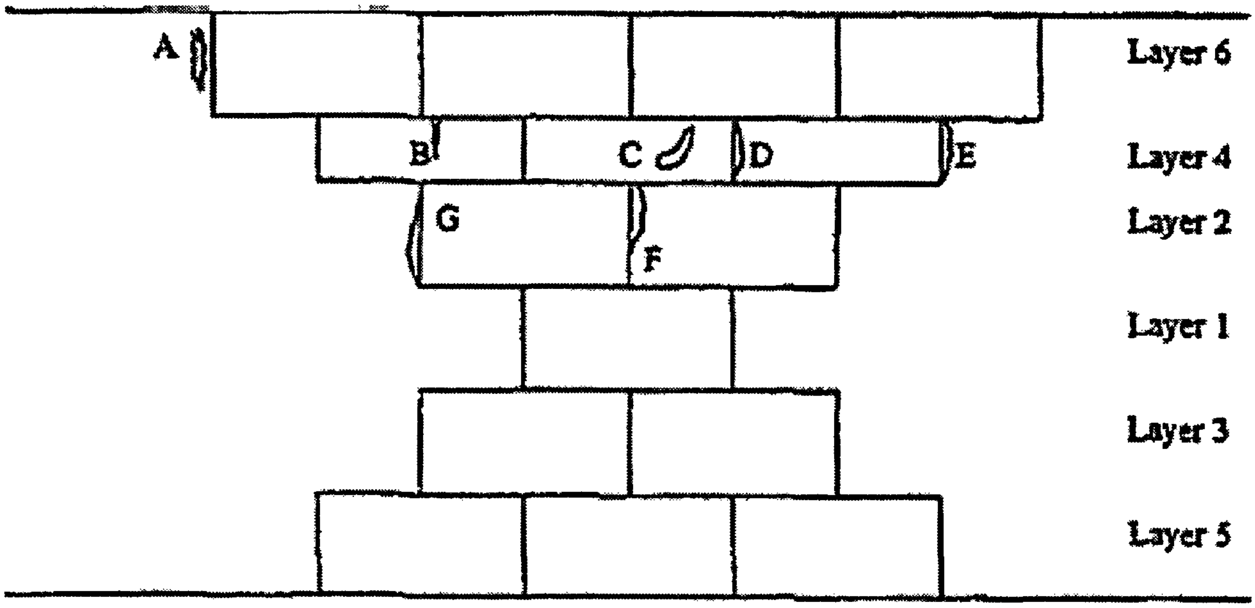

Figure 4-2. Schematic Representation of Different Types of Crack-Like Defects (A—Heat Affected Zone Crack, B—Centerline Crack, C—Pore with Tail, D—Lack of Sidewall Fusion, E—Inter-Run Slag, and G—Sidewall Slag) (Chapman and Simonen, 1998)

dimensions for a defect, and a decision is made if the defect can propagate. If a defect can propagate, it is taken to the next weld layer. If a second defect initiates within the vicinity of the first defect with an overlapping influence zone, the two defects are combined. If a defect fails to propagate, it is assumed to be left behind by the welding process after simulating its depth and length. Defect width and angle are not simulated. These are, however, used in radiographic and dye-penetrant inspection simulations.

The RR-PRODIGAL code was developed to predict the frequency and distribution of flaws that occur during multipass welding for a given type or family of welds. The code modeled welds for steel piping and vessels of less than 10.16 cm [4 in] in wall thickness and was used by the U.S. Nuclear Regulatory Commission (NRC) licensees for leak-before-break submittals to the NRC. The code was modified in a collaborative program between the Pacific Northwest National Laboratory and Rolls-Royce Associates funded by NRC to address vessels with wall thicknesses of 20.32 cm [8 in] or more. The RR-PRODIGAL code has been applied to estimating the probabilities of defects in reactor pressure vessel and in HI-STORM cask welds (Santos, et al., 2001).

A A533B steel pressure vessel research user facility {4.39 m [173 in] in diameter and 13.34 m [525 in] in height} was used for validation of the RR-PRODIGAL code (Doctor, et al., 1999). The wall thickness of the vessel varied from one region to the other but within 25 cm [10 in] of the belt line welds, the thickness was 22 cm [8.6 in]. Approximately 2,500 flaws were detected using the synthetic aperture focusing technique for ultrasonic testing. Most of the flaws had a through-wall dimension of greater than 3 mm [0.12 in]. Of 2,500 flaws, 884 were detected in the welded region that included the heat affected zone. A total weld volume of 0.214 m^3 [7.6 ft^3] was examined. This gave an average flaw density of 4,131 flaws/m^3 [117 flaws/ft^3] of weld. This number is significantly higher than weld flaw density obtained by Simonen and Chapman (1999), who conducted a systematic study of welding defects in pipes and vessels

installed in U.S. nuclear power plants. The study found 23 flaws with through-wall depths less than 4 mm [0.16 in]. In examination of 1.9 m^3 [67.1 ft^3] welds and using the Gumbel distribution, 597 weld flaws {greater than 0 mm [0 in]} were estimated. A flaw density of 314.2 flaws/m^3 [8.9 flaws/ft^3] was calculated, which is more than one order of magnitude lower than the experimentally determined flaw density in the pressure vessel research user facility vessel. The data were dominated by welds of approximately 2.54 cm [1 in] thickness.

Weld flaws were also examined in the Shoreham reactor pressure vessel (Schuster, et al., 1999). Table 4-1 provides the number of flaws observed in the weld region in the inner and outer 25 mm [0.98 in] depths of the vessel. Ninety-eight percent of the flaws were less than 4 mm [0.16 in] in through-wall thickness (Schuster, et al., 1999). These small flaws were primarily located in the fusion area slightly {1 mm [0.025 in]} inside the weld. The through-wall extent and the length of small flaws were estimated to be below the resolution of the synthetic aperture focusing technique for ultrasonic testing inspection. The predominant shape was round, indicating these flaws were less than 3.5 mm [0.138 in] in length.

The observed data for pressure vessel research user facility vessels showed a much larger number of flaws were smaller in size than predicted by the RR-PRODIGAL code. Although this may not be a significant concern for structural integrity, small flaws are of interest for estimating initial waste package failures. These small flaws can act as initiating sites for stress corrosion cracking. Chapman and Simonen (1998) attribute this inconsistency to

- The RR-PRODIGAL code may systematically underestimate flaw frequencies

- The RR-PRODIGAL code was developed to address only crack-like flaws and excludes volumetric types of flaws. The inclusion of volumetric flaws would roughly double the predicted flaw frequencies.

- The RR-PRODIGAL code was developed to predict the expected number of flaws for a large population of vessel welds with given attributes and does not address the differences for individual welds.

For the flaw depths greater than 5 mm [0.20 in], the observed flaw rates are consistent with the results predicted by the RR-PRODIGAL code without x-ray examination, while for flaw depths of 10 mm [0.39 in], the observed flaw rates are consistent with predicted flaw rates that include x-ray examination. These observations point out serious deficiencies in predicting flaw size and densities in vessels and warrant us to limit applicability of the RR-PRODIGAL code in the presence of small flaws.

A selection of pressure vessel research user facility samples were characterized using the radiographic method to determine the shape of the flaws (Schuster, et al., 2000). Results of radiographic testing confirmed that the fusion surfaces between weld and base metal contained an elevated concentration of flaws and that 30 percent of the flaws were rounded. For larger flaws, however, the rounded flaws were 60 percent.

A comparison of weld flaw data for pressure vessel research user facility and Shoreham vessel indicates the majority (more than 98 percent) of the flaws are small in size {less than 3 mm [0.12 in]}, with characterization limited by the resolution of the instrument. A large variability in the flaw density for small flaws, as shown in Table 4-1, was observed between vessels, however.

Table 4-1. Observed Weld Flaw Frequencies*†

Vessel	Location	Size of Cracks	Weld Volume	Number of Flaws	Flaws/m³ [flaws/ft³]
Pressure Vessel Research User Facility	Near Surface Zone {25 mm [0.98 in]}	< 3 mm [< 0.12 in]	0.014 m^3 [0.49 ft^3]	191	13,571 [384.3]
	Near Surface Zone {25 mm [0.98 in]}	> 3 mm [> 0.12 in]	0.014 m^3 [0.49 ft^3]	13	929 [26.3]
	Remaining thickness	< 5 mm [< 0.20 in]	0.20 m^3 [0.7 ft^3]	653	3,625 [102.6]
	Remaining thickness	> 5 mm [> 0.20 in]	0.20 m^3 [0.7 ft^3]	27	135 [3.8]
Shoreham Reactor Pressure Vessel	Inner 25 mm [0.98 in] surface	< 4 mm [< 0.16 in]	0.0226 m^3 [0.8 ft^3]	459	20,309 [574.5]
	Inner 25 mm [0.98 in] surface	> 4 mm [> 0.16 in]	0.0226 m^3 [0.8 ft^3]	9	398 [11.3]
	Outer 25 mm [0.98 in] surface	< 4 mm [< 0.16 in]	0.0241 m^3 [0.85 ft^3]	639	26,515 [750.8]
	Outer 25 mm [0.98 in] surface	> 4 mm [> 0.16 in]	0.0241 m^3 [0.85 ft^3]	19	788 [22.3]

*Doctor, S.R., G.J. Schuster, and F.A. Simonen. NUREG/CP–0166, Vol. 1, "Fabrication Flaws in Reactor Pressure Vessels." Proceedings of the Twenty-Sixth Water Reactor Safety Information Meeting, Bethesda, Maryland, October 26–28, 1998. Washington, DC: NRC. pp. 85–103. June 1999.

†Schuster, G.J., S.R. Doctor, S.L. Crawford, and A.F. Pardini. NUREG/CR–6471, Vol. 3, "Characterization of Flaws in U.S. Reactor Vessels—Density and Distribution of Flaw Indications in Shoreham Vessel." Washington, DC: NRC. November 1999.

The literature review indicates no studies have been published about Alloy 22, which is the material of choice for the waste package outer container.

4.1.1 DOE Approach

To estimate the frequency of weld flaws in the closure weld of the Alloy 22 waste package outer lid, DOE used the results of a simulation scenario used by Khaleel, et al. (1999) as an input to

the RR-PRODIGAL code (CRWMS M&O, 2000a). As discussed in the previous section, the RR-PRODIGAL code has several limitations including underestimation of small flaws that may act as initiating sites for stress corrosion cracking. DOE used the following assumptions to develop the probability of having various size weld flaws in the waste package shell and lid welds.

- Weld flaw density and size distribution information for tungsten-inert-gas welded stainless steel can be applied to Alloy 22. The basis for this assumption is that welding Alloy 22 has been identified as a process similar to welding austenitic stainless steel (ASM International, 1993).

- Information on the frequency of occurrence of weld flaws, their locations, and their depth distributions was obtained from results of the RR-PRODIGAL code.

- Information on the median flaw size and shape parameters for tungsten-inert-gas welded stainless steel is given as a function of wall thickness {6.35- to 63.5-mm- [0.25- to 2.5-in-] thick welds}.

- Information on density of flaws is based on the simulation scenario that used 25.4-mm- [1-in-] thick stainless steel manual metal arc welds performed in the shop and subjected to inspections by radiographic and dye-penetrant testing.

- All flaws detected by post-weld inspections are perfectly repaired.

- Information on the reliability of radiographic, ultrasonic, and dye-penetrant tests is applicable to the materials and inspection methods that will be used for waste packages. Khaleel, et al. (1999) consider that flaw density increases by a factor of 12.8 for welds with no radiography for stainless steel manual metal arc, and this value is similar for a tungsten-inert-gas weld. An increase in flaw density that uses only radiographic and no dye-penetrant test is 31.4.

- Possible defects present in the welds are either rounded and have no direction (e.g., tungsten inclusion, silicon, or porosity) or planar (e.g., lack of fusion). It is assumed that 1 percent of the defects are planar and in a direction normal to the direction of the weld centerline.

- Flaws are divided into five depth regions: outer surface flaws, flaws within the outer quarter of the weld, embedded center flaws, flaws within the inner quarter of the weld, and inner surface flaws. Based on Khaleel, et al. (1999), and assuming that weld flaws are uniformly distributed throughout the depth of the weld, it was estimated that 1 percent of the flaws are

on the inner surface, 11 percent are in the inner quarter of the weld, 49 percent are embedded in the center, 35 percent are in the outer quarter of the weld, and 4 percent are on the outer surface.

- The probability of a flaw exceeding a depth ranging 1–11.5 mm [0.04–0.45 in] is 1.06×10^{-9}.

- The most likely location for base metal flaws is along the edge of the plate material.

- Embedded flaws are not a concern for postclosure performance, which is based on the assumption that waste packages will not be subjected to cyclic fatigue (the primary mechanism for causing embedded flaws to grow through-wall in similar components).

DOE used the following methodology for estimating the probability of having various size weld flaws in the waste package shell and lid welds.

- The total flaws per type of waste package weld were calculated by multiplying the weld length by the linear flaw density {given as 0.6839 flaws/m [0.0174 flaws/in] of weld for a 25.4-mm- [1-in-] thick stainless steel tungsten-inert-gas weld performed in shop conditions} and by an adjustment factor for weld thickness from Figure 4-3.

- The flaw size distribution was used to determine the probability that a flaw would have a size within a given range.

- The probability of each range was multiplied by the total number of flaws per weld to determine the expected number of flaws within that size range. For welds subjected to ultrasonic inspection, the expected number of flaws within each range was then reduced by multiplying by the probability of nondetection for the lower end of the size range. Because the ultrasonic test probability of nondetection is based on a single-angle ultrasonic examination, and a multiangle examination is planned for the lid welds, the square of the probability of nondetection was used for the lid welds.

- For all cases, each range was then multiplied by 0.34 percent (Khaleel, et al., 1999) to yield the expected number of outer surface-breaking flaws within that range. On average, 0.34 percent of flaws are surface breaking, as shown from Khaleel, et al. (1999).

- Finally, the expected numbers of outer surface-breaking flaws in each size range are summed to determine a new value for total flaws per weld, which accounts for the ultrasonic inspection. The data for the Alloy 22 shell welds are shown in Figure 4-4 and results for lid welds in Figure 4-5.

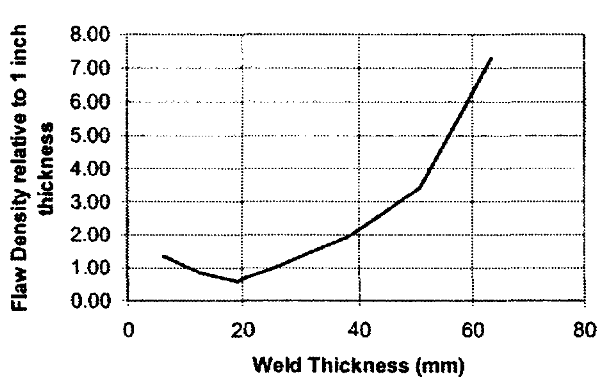

Figure 4-3. Effect of Weld Thickness on Flaw Density Normalized to a Thickness of 25.4 mm [1 in] (CRWMS M&O, 2000a).
Note: Information Is Presented in mm, use 1 mm = 0.039 in For Equivalent Conversion.

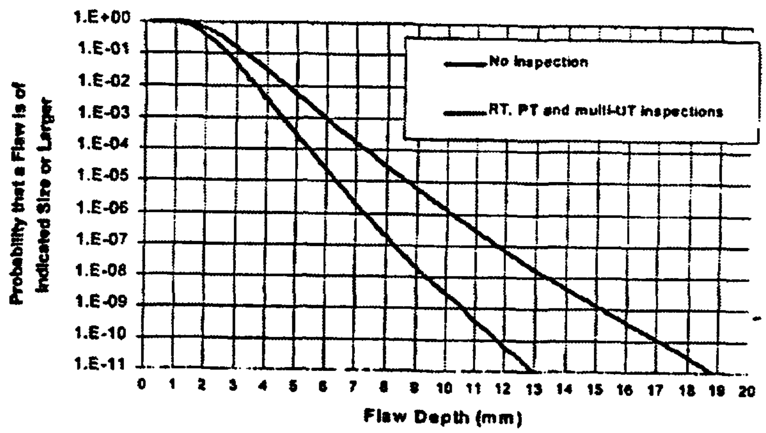

Figure 4-4. Size Distribution for Indicated Frequency of Occurrence for Outer Surface Breaking Flaws in Waste Package Alloy 22 Shell Welds (CRWMS M&O, 2000a). Note: Information is Presented in mm, use 1 mm = 0.039 in For Equivalent Conversion.

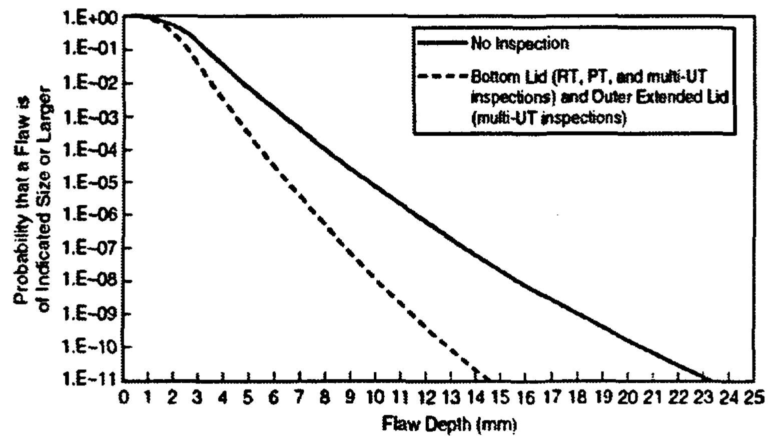

Figure 4-5. Size Distribution for Indicated Frequency of Occurrence for Outer Surface Breaking Flaws in Waste Package Alloy 22 Lid Weld (CRWMS M&O, 2000a). Note: Information is Presented in mm, use 1 mm = 0.039 in For Equivalent Conversion.

The consequence of an outer surface-breaking flaw in combination with the presence of an aggressive environment and high (near-yield) residual stress from the weld could potentially lead to stress corrosion cracking of the Alloy 22 container. Another consequence of surface flaws of any size is the potential for growth of these flaws into deeper pits or crevices. This is unlikely, however, in view of the high resistance of materials, such as Alloy 22, to pitting or crevice corrosion under the expected repository conditions.

4.1.2 CNWRA Review

Section 4.1.1 lists major assumptions that would affect the calculations for estimating the potential of early failures. Some assumptions are reasonable, however, some are erroneous or pose serious concerns because of their significant potential to affect analysis of the waste package failures nonconservatively. These assumptions include

- Published stainless steel data using the RR-PRODIGAL code are bounding for predicting features of flaws in the welds for Alloy 22. DOE assumed the simulation results obtained by Khaleel, et al. (1999) are bounding and the results provided by the RR-PRODIGAL code are conservative by a factor as large as 10 (Simonen and Chapman, 1999). Simonen and Chapman (1999) based their analysis on the measurement of weld flaws greater than 4 mm [0.16 in] depth in pipes and vessels installed in U.S. nuclear power plants and showed that the RR-PRODIGAL code simulations are conservative compared with observed weld flaw frequencies in reactor pressure vessels. However, they completely ignored observed small flaws as reported by Chapman, et al. (1996), Chapman and Simonen (1998), Doctor, et al. (1999), and Schuster, et al., (2000, 1999, and 1998). Validation data showed that for the simulation to match the experimental data, small cracks less than 3 mm [0.12 in] in size have to be ignored. Therefore, the DOE assumption results in a nonconservative flaw frequency distribution of one to two orders of magnitude. The DOE use of simulation results with no verification is inappropriate for estimating the flaw size distribution in Alloy 22.

- Weld flaw density and depth distributions for tungsten-inert-gas welded stainless steel can be applied to tungsten-inert-gas welded Alloy 22. Despite differences in the flow characteristics of Alloy 22 and stainless steel caused by the sluggish flow of Alloy 22, the DOE report assumed that welding of Alloy 22 is similar to the welding process of austenitic stainless steel. Even though DOE has welded Alloy 22, no systematic analysis has been conducted to show that the observed weld flaw frequencies are similar to austenitic stainless steel.

- Flaws detected by postweld inspections are perfectly repaired. In reality, this is known not to be the case (Chapman and Simonen, 1998). Experience with nuclear materials has shown that a large flaw may be repaired, but weld repair tends to leave many small flaws instead, and repairs to flaws can result in flaws greater than the original flaw. The DOE report does not include any information regarding requirements for repairs, repair processes, or criteria for approval of repair completion.

- In the analysis, the DOE report assumed that 1 percent of the defects are planar and in a direction normal to the direction of the weld centerline. In the Uncertainty Analyses and Strategy (Bechtel SAIC Company LLC, 2001), however, DOE has modified this assumption to state that all flaws are oriented in such a way that they can grow in the radial direction in the presence of hoop stress. Similar approach has been taken by the NRC for the analysis

of the HI-STORM cask that assumed 100 percent of the defects would be axially oriented (Santos, et al., 2001). CNWRA agrees with DOE that this revised assumption is conservative.

- In the analysis, DOE assumed that embedded flaws are of no concern. DOE needs to evaluate if there exists a critical flaw size that can cause a through-wall crack during a seismic event.

The DOE report used frequency of occurrence of weld flaws, their location and depth distribution obtained from the results of the RR-PRODIGAL code simulations by Khaleel, et al. (1999). No new calculations were performed using the RR-PRODIGAL code. Various errors by DOE were noted in the assumptions and use of the data calculated from the results of the RR-PRODIGAL code. The paper by Khaleel, et al. (1999) identifies data for piping materials derived from the original RR-PRODIGAL code. This paper cites only the flaws near the inner surface of a pipe that would affect fatigue crack growth, and does not address embedded flaws. The waste package outer container material, Alloy 22, may have potentially different cracking mechanisms and different probabilities of occurrence than stainless steel. Flaws buried within the welds and flaws breaking the outer surface are of concern to waste package performance. In the RR-PRODIGAL code, the heat-affected zone for stainless steel is assigned a zero probability for cracking. It is likely that the probability of the heat-affected zone cracking in Alloy 22 is not zero and verification is needed.

Although the probabilities of various defects that may potentially lead to early failures in waste package are discussed, there is no discussion on acceptable range of predicted flaw sizes. DOE proposed using the ASME Boiler and Pressure Vessel code as a guide and indicated that unacceptable weld defects, as defined in the applicable portions of the ASME code, Section III, Division 1, Subsection NB, shall be repaired in accordance with subsection NB (CRWMS M&O, 2000b). The discussion on weld flaws developed in the report should be extended to include acceptable range of weld flaw size.

4.2 Base Metal Flaws in Waste Package

Base metal flaws could result from improper waste package fabrication processes, in particular from weld repairs made to the base metal.

4.2.1 DOE Approach

The following main assumptions were used to develop the probability of having various sizes of base metal flaws in the waste package.

- All base metal flaws occur as a result of weld repairs made to the base metal.

- The most likely location for base metal flaws will be along the edge of the plate material.

- Fabrication procedures will restrict the use of welded attachments to base metal that will be removed prior to completion of the waste package fabrication.

- A quality control check of the fabrication process will be performed and documented to identify base metal weld repairs not executed according to the fabrication procedure.

The flaw distribution for base metal is based on information from Doctor, et al. (1999) obtained from unused reactor pressure vessels. The flaw densities were found to be one order of magnitude lower than those for weld metal. If base-metal flaws occur only as a result of weld repairs in regions near the welds, and those weld repairs are strictly controlled (CRWMS M&O, 2000b), then base-metal flaws can only occur as a result of failure to follow the fabrication procedure relating to base-metal forming and weld repair. The human error probability for failing to follow a written operating procedure is estimated to be 0.01, and the probability that the quality control check of the fabrication process will fail to find a violation in the fabrication procedure is estimated to be 0.1. It is further estimated, that there is a nominal flaw density reduction factor of 0.1 for a base-metal flaw compared with weld flaws. Therefore, the frequency of occurrence of base-metal flaws is estimated to have a probability of occurrence 10,000 times less than the flaws in the uninspected welds.

Any outer-surface-breaking flaws, in combination with the presence of an aggressive environment and sufficiently high residual stresses, could potentially lead to stress corrosion cracking. Another possible consequence is the growth of surface flaws into deeper pits or crevices. Alloy 22, however, is highly resistant to pitting corrosion, and therefore, surface flaws are not expected to grow by pitting corrosion, according to the DOE report.

4.2.2 CNWRA Review

Although the number of base metal flaws is expected to be lower when compared with weld flaws, the DOE report does not provide basis for initiating repairs that could result in formation of base-metal flaws. DOE proposed using the ASME Boiler and Pressure Vessel code as a guide and indicated that unacceptable weld defects, as defined in the applicable portions of the ASME Code, Section III, Division 1, Subsection NB, shall be repaired in accordance with Subsection NB (CRWMS M&O, 2000b). The discussion on weld flaws developed in the report should be extended to include acceptable range of base-metal weld flaw size.

4.3 Improper Material in Alloy 22 Welds

DOE plans to fabricate, weld, and inspect the waste packages in accordance with the portions of the ASME Boiler and Pressure Vessel code, Section III, Division 1, Subsection NB (Class 1 Components) (ASME International, 1995) that will ensure the waste package will perform in accordance with the design basis. In addition to two closure-lid welds, the Alloy 22 outer container of the waste package will undergo five welds during fabrication (two longitudinal and one circumferential for welding the outer barrier, one for welding the support ring, and one for welding the bottom lid).

4.3.1 DOE Approach

The following two main assumptions were used to develop the probability of improper material in the waste package Alloy 22 shell or lid welds.

- Field verification of the chemical composition of each weld wire will be performed prior to its use in fabricating any weld on the waste package. It is also assumed that such field verification will use new instrumentation such as portable x-ray spectroscopy equipment, which is assumed to work perfectly.

- The Alloy 22 outer container contains approximately 249 kg [550 lb] of weld material (including closure weld).

Inspection of 47 reactor vessels from 1966 to 1978, indicated 29.5 to 158.8 kg [65 to 350 lb] of weld wire was out of specification of the 1,935,202 kg [4,266,390 lb] of weld wire used. Probability of use of improper weld material ranges from 6.8×10^{-4} to 3.7×10^{-5} per unit mass of weld material. DOE used an adjusted mean of 5.0×10^{-5} per unit mass for this analysis. In addition, the DOE report indicates that the Babcock and Wilcox response to NRC Bulletin 78-12 concluded that the evolution of shop practices had virtually eliminated the possibility of using improper weld material in the fabrication of a reactor vessel. There is still a 1/1,000 probability (probability of improperly checking the digital readout), however, that an operator performing such verification would fail to perform the operation correctly. The probability of using improper weld material is estimated to be approximately 5.0×10^{-8} per unit mass of weld material. Using an assumed 249 kg [550 lb] mass of weld material in the Alloy 22 barrier yields an estimated probability of 2.7×10^{-5} per waste package for improper weld material.

DOE further states that if an improper material is used, it is expected the improper material would be another nickel-based alloy for the outer barrier or another stainless steel alloy for the structural barrier. The DOE report considers that the use of the improper material could affect the corrosion performance of the barrier.

4.3.2 CNWRA Review

It is not evident from the DOE analysis that the welding material used in the reactor pressure vessels routinely undergo verification prior to welding. If the welding material in reactor vessels undergo verification similar to the planned verification for waste packages, the DOE analysis is incorrectly multiplying the probability by 0.001 which accounts for an operator performing verification of the digital readout failing to perform the operation correctly. This would increase the probability of improper material to 2.7×10^{-2} per waste package. Furthermore, use of the human error probability equal to .001, which is the probability of misreading a digital display, may not be appropriate. If the composition of the weld material is to be checked by spectrographic analysis, then more than a single reading would be involved, because multiple constituents are present. Also, each constituent would need to be within certain percentage composition ranges for the weld material to be acceptable. The activity would then appear to require comparison of a set of reading, n = number of constituents, to allowable ranges. A more appropriate human error probability to use for this activity might be from Table 20-10 [Swain and Guttmann, 1983 (NUREG/CR–1278)], line 11, detect out-of-range arithmetic calculations = 0.05. It should be noted that DOE did not develop an event tree to calculate probability of improper material in Alloy 22 welds.

The use of improper material such as stainless steel may result in a waste package susceptible to both stress corrosion cracking and localized corrosion, as well as degradation in mechanical properties. As a result, waste package may have significant consequences on the repository performance.

4.4 Improper Heat Treatment

Each disposal container will undergo heat treatment after welding to remove residual stresses. However, no heat treatment is planned after the outer-lid closure weld. Note that before

high-level waste is placed in a container, the container is called a disposal container. Once a container is loaded with high-level waste and welded, the canister is called a waste package.

4.4.1 DOE Approach

The following assumptions on the heat-treatment process were used by DOE to develop an event-sequence tree to quantify the probability that a waste package is subjected to improper heat treatment.

- There is only one heat treatment/annealing operator and the furnace is manually controlled (not computer controlled).
- There are written operating procedures for the ramp-up/hold-time phases and the quench phase for every disposal container component.
- The operator needs to initially match the waste package components to be heat treated with the appropriate written heat-treatment operating procedure(s) via some type of digital identification code.
- If the operator has a mismatch between the components and the procedure, the ramp-up and hold time will be inappropriate for the components subjected to the heat treatment.
- The ramp-up and hold-time operations can be addressed by one procedure (and, hence, modeled with a single human error probability).
- A quality assurance check of the furnace occurs following the ramp-up and hold-time operations and can identify an error in implementing the written operating procedure.
- The quality assurance check of the furnace cannot identify a noncatastrophic equipment failure.
- Components are annealed with a quench following the quality assurance check, and the quench is accomplished with nozzles and hoses (not immersion).
- An independent laboratory check can identify failures caused by noncatastrophic equipment failures, such as not following the ramp-up and hold-time operating procedures or not following the quench procedures.
- The furnace has two failure modes: (i) catastrophic, which can be immediately detected; and (ii) noncatastrophic, which can only be detected using a laboratory test.
- The annealing time is approximately 24 hours, during which a failure in the furnace could lead to an improper heat treatment.

Because heat treatment involves human interaction, the probability of improper heat treatment is evaluated using an event-sequence tree that focuses on human errors. The heat-treatment process involves several decision points and one potential hardware failure mode. The DOE report determined the probability of placing a waste package repository with improper heat treatment as 2.21×10^{-5}.

Although the likelihood of improper heat treatment is small because of the administrative (procedural) controls and multiple checks, the DOE report noted that the consequences of improper heat treatment can be significant depending on the error. The inconsistent cooling rate of alloys such as Alloy 22 may result in precipitation of carbides and inter-metallic phases in the grain boundaries, which make the material more susceptible to a localized attack along the grain boundaries and, hence, promote stress corrosion cracking.

4.4.2 CNWRA Review

The DOE report determined the probability of placing a waste package into operation with improper heat treatment was 2.21×10^{-5}. It should be noted that 95 percent of this probability is contributed by a sequence in which the furnace suffers a noncatastrophic failure and an independent laboratory incorrectly determines that the component was subjected to proper heat treatment. Improper heat treatment produces a defect in the metal not identifiable in the procedural quality assurance check (because this is not a procedure error).

The exact nature of quality assurance checks is not delineated, so the estimated human error probabilities for these actions may be optimistic. DOE assumes that a quality assurance check of the furnace occurs after the ramp-up-and-hold operating procedure. However, DOE has not provided information on the nature of this quality assurance check for example if a strip chart will record the temperature of the furnace and/or the waste package during heat treatment and the quality assurance check will be performed on this record. If this is correct, a possible malfunctioning of the thermocouple and recording instrument have not been factored into the event tree. The quality assurance checker would need to compare the actual temperature versus time record and make a judgment of whether the ramp-up and hold times and temperatures were within acceptable limits. The human error probabilities in this activity are not analogous to failure to use a written operations procedure under normal operating conditions. Strictly speaking, Table 20-6 describes errors of omission, (i.e., failure to perform the check; even if the check is performed), an error of commission could result from performing the check improperly. Since the check involves comparisons of several values (times and temperatures), there are multiple opportunities for error. Although the probability of error for incorrect check-reading an analog strip chart are substantially smaller (~0.002–0.006) than the probability of error for omitting the check, since multiple checks are involved, the total error rate for improperly checking would increase. Another issue may be the nature of the independent laboratory check. If a separate test coupon will be prepared and heat treated along with the waste package and then be sent to the laboratory for test and evaluation. Then, the potential for a number of additional errors arises regarding the testing and possible misidentification of the test coupons. DOE should provide additional information on this event tree to correctly estimate probability of improper heat treatment.

DOE approach for determining the probability for improper heat treatment is not feasible to conduct independent laboratory check for an outer-lid closure weld because waste package cannot be moved outside the remotely operated facility. DOE should develop an event tree for estimating probability of improper heat treatment for the outer-lid closure weld. As indicated by DOE, improper heat treatment could result in a waste package susceptible to both stress corrosion cracking and localized corrosion, as well as degradation of mechanical properties. These effects may have significant consequences on the repository performance.

4.5 Contamination

The waste package will undergo at least nine cleaning steps before emplacement—after fabrication of each of the five welds (two longitudinal and one circumferential for welding the outer container, one for welding the support ring, and one for welding the bottom lid), after heat treatment, prior to shipping the empty disposal container to the site, prior to loading the high-level waste, and prior to emplacement of the waste package in the drift.

4.5.1 DOE Approach

The following main assumptions were used by the DOE to develop an event-sequence tree to quantify the probability of having corrosion-enhancing surface contamination on the waste package.

- There are different operators for each cleaning occurrence.
- Each cleaning is independent of the other cleanings.
- Procedures exist to prohibit cleaners that could have a corrosion-enhancing effect on the waste package metal.
- There is a written operating procedure to perform the cleaning process.
- An incorrect cleaning process with proper cleaning agents cannot leave a residue that can have adverse effects on the metal.
- A check of the cleaning process occurs.
- The probability of contamination just before final cleaning is assumed to be 0.0163 based on data from the steam generators of Indian Point-3 and nuclear fuel rods.
- Failure to have proper cleaning agents is estimated as 0.005, based on the expectation that mislabeling of cleaning supplies is similar to the failure to follow a written procedure (probability of 0.01) and that the stocking person fails to recover by using the wrong checklist form (probability of 0.5).
- Human error probability to check the post cleaning correctly is approximated as a probability to follow a written procedure (probability of 0.1). A lower limit of 0.02 was used for a more rigorous check.
- Failure of the cleaning process is approximated by a failure to follow a written procedure with an human error probability of 0.01.

The probability of placing a waste package into service after being subjected to corrosion-enhancing surface contamination involves various human interactions with the potential for human errors. The probability of a contaminated waste package is the sum of the probability of contamination during each cleaning plus the probability of leaving a waste package contaminated during the last cleaning. In the first step, an unacceptable consequence is obtained when the proper cleaning agent is not used, an operator check fails to acknowledge

the use of the wrong material, and the checker review fails to determine that the cleaning was performed properly. This calculation step provides a probability of 5.0×10^{-5} per cleaning or 4.5×10^{-4} per waste package (assuming nine cleanings). In the second step, which is the last cleaning, an operator receives a contaminated waste package and incorrectly follows the written procedure, and the checker fails to provide a rigorous review. This calculation step provides a probability of 3.26×10^{-6}. Based on the assumption that the outer container will be subjected to nine cleanings before emplacement, the probability of a waste package with a contaminated Alloy 22 outer container is approximately 4.5×10^{-4}.

DOE states that the waste package fabrication specifications will restrict the chemical compositions of the cleaning materials and solvents. The fabrication process also calls for removing all contaminants prior to heat treatment and other operations. Human error, however, could cause either the cleaning to be insufficiently performed or not accomplished at all. Thus, the consequence of this error is not expected to be a significant factor in corrosion.

4.5.2 CNWRA Review

In the DOE report, the probability of contamination was obtained by assuming the total number of cleanings equals 10, however, there are only nine cleanings. This was obtained by multiplying by nine the probability of contamination during cleaning followed by a final cleaning. In the first event tree, probability should be multiplied by eight. This correction provides an estimated probability of 4.0×10^{-4} per waste package.

Furthermore, the probability of a contaminated component is estimated to be $1/(75 \times 5) = 0.00267$, where five instances of contaminated components were found at 75 operating reactors. The usual rule for estimating frequency is to divide the number of occurrences by the number of opportunities; in this case, this rule would yield $f = 5/75 = 0.0667$. This 25-fold increase may be significant. In addition, DOE has chosen geometric means because the geometric mean estimates the midpoint of skewed values (10^{-3} and 0.267) more adequately than the arithmetic mean. However, no justification is provided for this statement. Use of the geometric mean skews the result toward a lower error rate, which is optimistic. This correction may significantly increase the probability of contaminated waste package placed in a repository.

DOE has not provided a list of chemicals that will be present at the site that could result in a human error. Without the availability of this information, it is not possible to assign a probability or estimate consequences for this event.

4.6 Unidentified Handling Damage

Improper handling can occur during transport at the repository. Handling damage is defined as any gouging or denting of a waste package surface, significant enough to affect postclosure performance of the Alloy 22 outer container and that might have the potential to go unnoticed. This event does not consider complete penetration of the outer container or malicious damage to the container.

4.6.1 DOE Approach

The following main assumptions were used to develop an event-sequence tree for quantifying the probability of a waste package being emplaced with unidentified handling damage.

- Damage is significant enough to cause penetration of the waste package, and the damage would not go unnoticed.
- The probability that a waste package is significantly gouged or dented during transport or handling at the repository is equivalent to the rate at which the fuel assembly failures occurred because of handling damage, which is 5×10^{-4}.
- At least the outer container will be inspected for handling damage on arrival at the repository, and the waste package will be inspected before final emplacement in the drift.
- If handling damage occurs and is identified, the waste package is either completely repaired or scrapped.
- Human error probability for failure by the operator to realize that the waste package has been damaged is considered as 0.01. Failure of inspection to detect a damaged waste package is considered as 0.1.

Because this type of potential damage involves primarily human error, an event-sequence tree was developed that included (i) waste package damage during transportation to the repository, (ii) failure of the operator to realize that the waste package had been damaged during movement, and (iii) failure to detect the damage during inspection. Based on this analysis, the probability of placing a waste package in the emplacement drift with undetected damage to the outer Alloy 22 container is estimated to be 5.5×10^{-6}.

Gouges on the waste package outer surface could provide sites for the start of crevice corrosion.

4.6.2 CNWRA Review

The probability of handling damage is estimated to be 5.5×10^{-6}. Ninety-one percent of this probability is associated with the sequence in the event tree in which the handling damage to a disposal container goes undetected during inspections on arrival at the repository and prior to emplacement in the drift as a waste package. The probability of handling damage prior to arrival at the repository is estimated to be 5.0×10^{-4}, based on handling damage data for spent nuclear fuel rods for the period 1989–1995. In addition to the 10 failures (of the 21,810 assemblies) caused by handling damage, the data in Table 4-2 indicate 226 failures between 1989–1995 attributed to defective assemblies which were not inspected after fabrication. This number results in a probability of failure of 0.01 (i.e., 226/21,810). Because the latter probability is considerably larger, the analysis should be expanded to incorporate an additional event-tree sequence wherein an uninspected and damaged disposal container arriving from the fabricator goes undetected during arrival inspections at the repository and prior to emplacement. The probability of this new sequence will be two orders of magnitude greater than the currently estimated probability and will have a significant effect on the number of early waste package failures expected to occur in the repository.

Table 4-2. Causes of Fuel Failures in Pressurized Water Reactors*								
	Number of Assemblies							
Failure Cause	**1989**	**1990**	**1991**	**1992**	**1993**	**1994**	**1995**	**1996 (Partial)**
Handling Damage	—	6	2	—	—	1	1	1
Debris	146	11	67	20	13	6	10	1
Baffle jetting	—	—	—	—	—	—	—	—
Grid fretting	14	18	9	33	36	9	33	19
Primary hydriding	—	1	—	4	—	—	—	—
Crudding/corrosion	—	—	—	—	—	—	4	1
Cladding creep collapse	—	—	—	—	—	—	1	—
Other fabrication	1	15	1	5	3	1	15	3
Other hydraulic	—	—	—	—	1	—	—	—
Inspected/unknown	—	—	—	—	36	36	13	2
Uninspected	43	58	35	61	14	3	12	1
Totals	204	109	114	123	103	56	89	27
Total discharged	2,196	3,461	2,937	3,302	3,612	2,636	3,666	—

*Yang, R.L. "Meeting the Challenge of Managing Nuclear Fuel in a Competitive Environment." Proceedings of the 1997 International Topical Meeting on LWR Fuel Performance, Portland, Oregon, March 2–6, 1997. LaGrange Park, Illinois: American Nuclear Society. pp. 3–10. 1997.

It is not evident from the DOE description that the probability of damage during handling will be similar to spent nuclear fuel because the disposal container weighs significantly more than the fuel assemblies and, therefore, could have a higher occurrence of damage during regular or remote handling.

4.7 Waste Package Having Thermal Output Outside the Expected Range

According to waste package specifications, the maximum allowable heat output of a waste package is 11.8 kW [4.03×10^4 Btu/hr] per waste package. Currently, maximum heat output of 11.53 kW [3.93×10^4 Btu/hr] is from waste packages containing 21 assemblies and minimum heat output of 0.52 kW [1.77×10^3 Btu/hr] is from waste packages containing 24 assemblies of boiling water reactor fuel (DOE, 2001).

4.7.1 DOE Approach

The assumptions used by DOE to develop an event-sequence tree to quantify the probability of a waste package containing 21 assemblies of pressurized water reactor fuel having a thermal output outside the expected range as a result of a thermal misload were

- It is possible to load a waste package in such a manner that it will be outside the thermal design basis of 11.8 kW [4.03×10^4 Btu/hr] using only the population of fuel available in the pool at any given time.
- Waste package thermal misload results from assembly misload.
- A loading diagram is developed for each waste package, and any failure in developing the loading diagram will lead to a misloaded waste package if the failure is not identified by a quality assurance check or independent verification.
- Quality assurance checks are performed for the loading diagram development and for the loaded waste package.
- A thermal verification of a waste package is performed, and the operator will simply read the measured thermal output from a digital display.
- Any waste package found to be misloaded will be reloaded so it is within the allowable thermal output range.

The probability that a waste package is accidentally loaded with fuel not within its thermal design basis was estimated for a 21-pressurized water reactors waste package because this represents the waste package closest to the maximum allowable heat output of 11.8 kW [4.03×10^4 Btu/hr]. There are two types of human errors that the operator might commit when selecting the waste package, the fuel assemblies to be loaded, or both: conceptual and selection. The conceptual human error represents the intentional selection of a wrong item based on an erroneous belief that the item is actually correct. The selection human error represents the unintentional selection of the wrong item while trying to collect the correct item.

The probability of misloads may be higher, however, if the current strategy of blending fuel to achieve the desired heat output is used. Again, there are several human steps involved: (i) developing an incorrect written loading diagram that the operator follows, (ii) failing to properly check the incorrect written loading diagram, (iii) loading the fuel incorrectly, and (iv) failing to match components with the written procedures. The event-sequence tree developed in the DOE report showed the probability of storing a waste package with an improper heat load to be 1.2×10^{-5}, per waste package if the thermal output is verified prior to disposal, which can increase to 1.2×10^{-2} if the thermal output is not verified prior to disposal. This analysis clearly shows that the thermal output of the load should be verified to minimize potential misloading.

In the DOE report it is noted, based on data obtained from the viability assessment repository design, that the outer temperature of the waste packages as a function of time appears to be roughly the same after the first 200 years regardless of the initial thermal output of the waste package. Therefore, it is believed that a thermal misload would not significantly alter the performance of a waste package after closure.

4.7.2 CNWRA Review

DOE has assumed that

- Each type of waste package is loaded according to a unique and predefined loading diagram, assigning to each fuel-basket tube an allowable heat-load range. This assumption implicitly also assumes that all the fuel loading diagrams are generated without error, an optimistic point of view. Table 20-5 of NUREG/CR–1278 (Swain and Guttmann, 1983) indicates the probability of error for a set of instructions could be as high as 0.003.

- Any failure of the measurement instrumentation will be readily detectable. The rationale for this assumption is based on how other thermal measuring devices operate. It is not clear, if DOE has included calibration errors for instruments that must be periodically calibrated.

- The human error probabilities associated with the conceptual error is approximated by a rule-based action after diagnosis. This appears to be a misinterpretation of the NUREG/CR–1278 (Swain and Guttmann, 1983) tabulated human error probabilities. The first entry in Table 20-2 was used as a response to abnormal events and depicts errors in responding to a screening diagnosis. Then the probability of error is adjusted downward by the error factor because this is clearly a normal operating circumstance. There does not appear to be any justification in NUREG/CR–1278 (Swain and Guttmann, 1983) to use error factors in this fashion. As the flow chart on page 20-3 in NUREG/CR-1278 (Swain and Guttmann, 1983) clearly shows, there is a direct path for normal events to the rule based action tables, which avoids the diagnostic aspects of responding to an abnormal event. The conceptual error (the operator fails to properly determine the characteristics of an adequate fuel assembly as required by the loading diagram) appears to be a rule-based procedural task. Appropriate alternative human error probabilities for this action is shown in Table 4-3. However, this table is mainly directed toward errors of omission, rather than errors of commission. Furthermore, the error described may be a cognitive error (not recognizing a cooler assembly should be loaded), so it is largely beyond the capability of the technique for human error rate prediction methodology. Instead of a human error probability of 0.001, a value of 0.01 to 0.5 appears to be more appropriate.

- The human error probability associated with the selection error is approximated by an error of commission in selecting the wrong control on a panel of similar looking controls that are arranged in well-delineated functional groups. From the task description, it appears the operator will need to select an assembly from a cask or from a storage area (e.g., a pool). All the assemblies will have similar shape, size, and presence of tags; they may be designated by serial numbers. They all have the same function, to be loaded into a waste package. Therefore, it is not clear how the assemblies will be distinguished by well-delineated functional groups. Based on Table 20-12, entry (2) appears more appropriate with a human error probability of 0.003. Table 20-13 suggests that an human error probability as high as 0.008 might be appropriate.

Table 4-3. Approximate Alternative Human Error Probabilities for Thermal Output Outside the Range (Swain and Guttmann, 1983)*			
Table	Entry	Human Error Probability Value	Description
20-6	3	0.01	Use written operations procedure under normal operating conditions
20-6	5	0.01	Use a valve change or restoration list
20-6	8	0.50	Use a checklist properly
*Swain, A.D. and H.E. Guttmann. NUREG/CR-1278, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications: Final Report." Washington, DC: NRC. August 1983.			

If these factors are taken into consideration, probability of thermal misload in a waste package may be significantly larger compared to current DOE estimate.

Even though after 200 years the temperature of the waste package containing 21 assemblies of pressurized water reactor spent nuclear fuel rods is not significantly different compared to other waste packages, the differential between the waste packages containing 21 assemblies of pressurized water reactor fuel and 24 assemblies of boiling water reactor fuel before 200 years is almost 60 °C [104 °F] (CRWMS M&O, 2000a). This differential could create a cold trap effect that may provide a source of moisture to other waste packages in the drift. Most current information taking into account various thermal options should be used in determining the effect of thermal loading.

4.8 Gap in the Drip Shield Over the Waste Package

In the current engineered barrier design, a titanium drip shield will be placed over the waste packages at the time of repository closure, and this will be accomplished remotely by a mobile gantry.

4.8.1 DOE Approach

The assumptions used to develop an event-sequence tree for quantifying the probability of having a gap in the drip shield over the waste package were

- The operator remotely emplaces the drip shield and performs a self check of the work. The probability that the operator fails to properly place the drip shield is based on the human error probability of 0.003 estimated for improperly mating a connector. Using the maximum error factor of 3, DOE is using a probability of 0.009 per waste package for misplaced drip shield.

- A remote self-check by the operator and a quality assurance inspection of the emplaced drip shield are performed and have a human error probability of 0.1.

- Once a gap in the drip shield has been identified, it is perfectly repaired and a new gap is not introduced anywhere along the length of the drift.

DOE identified two types of emplacement errors that could occur during emplacement of the drip shields. First, an operator does not place the drip shield segments (their lengths are slightly larger than that of the waste package) so that the neighboring drip shield segments properly overlap. Second, operator self inspection and quality assurance inspection fail to verify that the drip shield segments were installed properly. An event-sequence tree was developed in the DOE report to evaluate the human errors involved with this process, and it was determined the potential for drip shield emplacement error was 9.0×10^{-5} per waste package.

An improperly placed drip shield will provide a gap that will allow water to drip directly onto the waste package. DOE further states that the current waste package degradation model takes no credit for the drip shield against dripping (CRWMS M&O, 2000c).

4.8.2 CNWRA Review

It is unlikely that human error related to the manipulation and emplacement of a 5 m long drip shield can be adequately modeled using the human error probability for mating electrical connectors [Swain and Guttmann, 1983 (NUREG/CR–1278, Table 20-12, entry 13). A further complication is that the emplacement of drip shields is likely to be accomplished either remotely or by computer directed mechanisms. Failures associated with remote operation have not been addressed. DOE has not considered and included failure rates for electronic components, such as cameras or computers and software reliability.

A drip shield emplacement error will result in additional dripping of water on the waste package surface and, depending on other factors, may provide localized corrosion and enhance radionuclide release.

5 SUMMARY

Early failure of some waste packages may lead to an early release of radionuclides. Early failure also provide a mechanism for water ingress into the failed waste packages, which may increase the potential for criticality. This review of the U.S. Department of Energy (DOE) report indicates that although the qualitative aspects of waste package degradation may be obtainable from historical data, any quantification of degradation rates or waste package failures will necessarily contain large uncertainties. Furthermore, special care may be required to adequately quantify uncertainties related to potential degradation mechanisms with long incubation times. DOE should provide a technical basis to justify the use of surrogate material data in lieu of Alloy 22, and it should incorporate future unknown degradation mechanisms such as those observed in nuclear power plant operations in its model uncertainties.

The DOE report lists major assumptions that would affect estimates of early waste package failures. Several of these assumptions are not fully supported. For example, the assumption that the frequency of occurrence of weld flaws could be based on data collected using the expert system-based simulation RR-PRODIGAL code reflects a basic lack of understanding of key components of operation and of the sensitivities of the code. Also, the use of data obtained from the RR-PRODIGAL code simulation results published by Khaleel, et al. (1999) is not applicable to Alloy 22. Furthermore, DOE assumed the simulation results obtained by Khaleel, et al. (1999) are bounding and that the results provided by the RR-PRODIGAL code are conservative by a factor as large as 10, as noted by Simonen and Chapman (1999). Simonen and Chapman (1999) based their analysis on the measurement of weld flaws greater than 4 mm [0.16 in] in depth in piping and vessels installed in U.S. nuclear power plants and showed that the RR-PRODIGAL code simulations are conservative compared to observed weld flaw frequencies. However, they ignored observed small flaws as reported by Chapman and Simonen (1998). The validation data showed that, for the simulation data to match the measurement data, small cracks less than 3 mm [0.12 in] in size have to be ignored. Therefore, the DOE assumption results in a nonconservative flaw frequency distribution of one to two orders of magnitude. The DOE use of simulation values from Khaleel, et al. (1999) with no experimental verification is inappropriate for estimating the flaw size distribution in Alloy 22.

The DOE report also provided a review of failures in various types of similar containers. To achieve a lower failure rate, careful control and analysis of failure mechanisms are required. A review of the accumulated historical data about similar types of containers indicates that potential defects arising from the generation of flaws during manufacturing and welding, the use of improper weld materials, improper heat treatments, inadequate weld design, handling damage, and potential contamination are important. Staff review is summarized in Table 5-1. Specifically, staff review indicates that the DOE report should provide additional information on the following:

- Some human error probabilities are in error due to incorrect use of data in NUREG/CR–1278 (Swain and Guttmann, 1983). Although human error probability tables in NUREG/CR–1278 (Swain and Guttmann, 1983) are routinely used in nuclear power plant probabilistic risk analysis, human error probabilities are generic values based on evaluating a wide range of experimental data and are not intended to be used in isolation. Also, some of the activities especially important to repository operations are not well covered in the technique for human error rate prediction methodology, which was developed primarily to analyze operating reactors. Additional studies are needed to expand the technique for

Table 5-1. Summary of the Review Results. This Table Combines the DOE Data from CRWMS M&O* and Includes the CNWRA Comments in the Last Column						
Waste Package Defect Type	**Probability per Waste Package**	**Possible Consequences for Postclosure Performance**				**CNWRA Comments**
	Alloy 22 Barrier	**Degraded Mechanical Properties**	**Pitting or Crevice Corrosion**	**Stress Corrosion Cracking**	**Early Water Contact**	
Weld Flaws (outer surface, breaking only, assuming inspections using radiographic penetrant, and ultrasonic testing)	Less than 10^{-4} for flaws greater than 5 mm [0.2 in] deep	—	Possible	Possible	—	Flaws < 5 mm [< 0.20 in] are ignored in the analysis, which accounts for greater than 90 percent of the flaws. Data used for estimating the number of flaws is not justified.
Buried flaw extends to the critical depth of 11.5 mm [0.45 in] or beyond	10^{6} for 5.5 mm [0.22 in] of weld corroding away	—	—	—	Possible	A critical flaw size that can result in a through-wall crack from a seismic event should be evaluated.
Base metal flaws	10^{-4} times less than the weld flaw rate	—	Possible	Possible	—	No technical basis provided for accepting base metal flaws resulting from weld repairs.
Improper weld material	2.7×10^{-5}	Possible	Possible	Possible	—	No event tree was developed. Analysis lacks justification for the human error probability.

Table 5-1. Summary of the Review Results. This Table Combines the DOE Data from CRWMS M&O* and Includes the CNWRA Comments in the Last Column (continued)						
	Probability per Waste Package	**Possible Consequences for Postclosure Performance**				
Waste Package Defect Type	**Alloy 22 Barrier**	**Degraded Mechanical Properties**	**Pitting or Crevice Corrosion**	**Stress Corrosion Cracking**	**Early Water Contact**	**CNWRA Comments**
Improper heat treatment	2.2×10^{-5}	Possible	Possible	Possible	—	This should be evaluated taking into account all fabrication steps. Human error probabilities needs to be reevaluated. No analysis is provided for closure welds.
Surface contamination	4.5×10^{-4}	—	Possible	—	—	Errors found in the analysis. Human error probabilities should be reevaluated. Consequence analysis should include evaluation of all chemicals planned for use.
Handling damage	5.5×10^{-6}	—	Possible	—	—	Analysis is incomplete. It should include all handling steps in the event tree.
Thermal misload of waste package	1.2×10^{-2} without waste package thermal verification; 1.2×10^{-5} with waste package thermal verification	—	—	—	—	Human error probabilities should be reevaluated. Analysis should include the cold trap effect that could have significant consequence.

Table 5-1. Summary of the Review Results. This Table Combines the DOE Data from CRWMS M&O* and Includes the CNWRA Comments in the Last Column (continued)						
	Probability per Waste Package	Possible Consequences for Postclosure Performance				
Waste Package Defect Type	Alloy 22 Barrier	Degraded Mechanical Properties	Pitting or Crevice Corrosion	Stress Corrosion Cracking	Early Water Contact	CNWRA Comments
Drip shield emplacement error	9.0×10^{-5}	—	—	—	Possible	Analysis does not provide an assessment of human error probability for remote operations.

*CRWMS M&O. "Analysis of Mechanisms for Early Waste Package Failure." ANL–EBS–MD–000023. Rev. 02. Las Vegas, Nevada: CRWMS M&O. 2000.

human error rate prediction tables with values obtained from fuel handling facilities and operations. The systematic methodology described in NUREG/CR–1278 (Swain and Guttmann, 1983) must be followed.

- Equipment failure rates must address all equipment to be used in the waste package fabrication process.

- Additional information regarding ultrasonic techniques is needed to assess the reliability of the inspection adequately.

Although an extensive set of event-tree sequences was provided to estimate the failure probabilities associated with some mechanisms for early failure (e.g., improper heat treatment, contamination, and improper handling), additional analyses should be performed in areas that could significantly affect the estimated number of early waste package failures. For example, because the welding and heat treatment of the outer lid of the waste package are remote operations, it is highly unlikely the sequence of operations used by DOE for developing an event tree to estimate the probability of the improper heat treatment is applicable to the outer-lid closure welds. Also, the DOE report did not provide an event-tree sequence for improper weld material. Either improper weld material or improper heat treatment could have significant consequences in the waste package performance.

The use of event trees by DOE to develop probabilities is acceptable provided relevant data with sound technical bases is used to develop these event trees. Most analyses, however, requires significant revision and additional analyses, that may result in a significant increase in the probability for early failures.

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