EEG-30

EVALUATION OF THE WASTE ISOLATION PILOT PLANT CLASSIFICATION OF SYSTEMS, STRUCTURES AND COMPONENTS

Tenera Corporation
A Division of TERA Corporation

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Environmental Evaluation Group
Environmental Improvement Division
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FOREWORD

The purpose of the Environmental Evaluation Group (EEG) is to conduct an independent technical evaluation of the potential radiation exposure to people from the proposed Federal radioactive Waste Isolation Pilot Plant (WIPP) near Carlsbad, in order to protect the public health and safety and ensure that there is a minimal environmental degradation. The EEG is part of the Environmental Improvement Division, a component of the New Mexico Health and Environment Department -- the agency charged with the primary responsibility for protecting the health of the citizens of New Mexico.

The Group is neither a proponent nor an opponent of WIPP.

Analyses are conducted of available data concerning the proposed site, the design of the repository, its planned operation, and its long-term stability. These analyses include assessments of reports issued by the U. S. Department of Energy (DOE) and its contractors, other Federal agencies and organizations, as they relate to the potential health, safety and environmental impacts from WIPP.

The project is funded entirely by the U. S. Department of Energy through Contract DE-AC04-79AL10752 with the New Mexico Health and Environment Department.

During the period 1979 to the present, the EEG has been involved in the review of the WIPP Safety Analysis Report (SAR) and its numerous amendments. Sections of this report have described the classification system for WIPP, a system for classifying the structures, systems and components to be used in the construction of the WIPP nuclear waste repository. Specific design requirements, standards and quality control are to be assigned to these structures, systems and components depending upon their designated class. The EEG has noted several apparent deficiencies in the information reported in the SAR on this classification system and has called these matters to the attention of the DOE. These deficiencies are summarized in EEG-29. To further assess these apparent deficiencies and their potential impact on the public health and safety, the EEG contracted with the Tenera Corporation to carry out an independent evaluation of the WIPP Classification System. This report represents the result of that contract.

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1. EXECUTIVE SUMMARY

A review of the classification system for systems, structures, and components at the Waste Isolation Pilot Plant (WIPP) was performed using the WIPP Safety Analysis Report (SAR) and Bechtel document D-76-D-03 as primary source documents. The regulations of the U. S. Nuclear Regulatory Commission (NRC) covering "Disposal of High Level Radioactive Wastes in Geologic Repositories," 10 CFR 60, and the regulations relevant to nuclear power plant siting and construction (10 CFR 50, 51, 100) were used as standards to evaluate the WIPP design classification system, although it is recognized that the U. S. Department of Energy (DOE) is not required to comply with these NRC regulations in the design and construction of WIPP.

The DOE General Design Criteria Manual (DOE Order 6430.1) and the Safety Analysis and Review System for AL Operation document (AL 54f81.1A) were reviewed in part. These documents contain useful information potentially applicable to facilities such as WIPP, however, system safety functions and the concept of a "graded" quality assurance program are not discussed on a basis which allows direct comparison with the WIPP quality classification system. Within the scope of this review, it was not possible to determine the extent to which the WIPP design is consistent with the DOE manual.

This report includes a discussion of the historical basis for nuclear power plant requirements, a review of WIPP and nuclear power plant classification bases, and a comparison of the codes and standards applicable to each quality level. Observations made during the review of the WIPP SAR are noted in the text of this report.

The conclusions reached by this review are:

- O WIPP classification methodology is comparable to corresponding nuclear power procedures.
- O The classification levels assigned to WIPP systems are qualitatively the same as those assigned to nuclear power plant systems.

- O The codes and standards applied to each quality level are comparable to those used in designing many operating nuclear power plants, but do not agree with those typically used in the design of newer nuclear power plants.

- O 10 CFR 60 does not specify "graded" quality assurance requirements and does not require the use of specific codes and standards; however, it does contain a basis for deciding which components are subject to a quality assurance program. This basis is consistent with comparable requirements for nuclear power plants (e.g., Regulatory Guide 1.26). Although this criterion would be a useful addition to the SAR and the Bechtel D-76-K-03 document, the procedure in use at WIPP appears to yield comparable results.

- O Because the above conclusions are based on the assumption that design conservatism and "defense in depth" at WIPP are comparable to that at nuclear power plants, further review of accident analyses and underlying assumptions should be made. In particular, accident events involving fires and the assumptions involving component failure during such accidents should receive further review.

- O A detailed evaluation should be made on a sample of the WIPP design to verify:
  - system design bases
  - accident analysis assumptions
  - accuracy of operations safety requirements (Chapter 10 of WIPP SAR).

  In particular the electric power system and the ventilation systems should receive futher review.

II. EVALUATION PROGRAM

The Environmental Evaluation Group of the State of New Mexico's Health and Environmental Department requested that TENERA Corporation, a division of TERA Corporation, review the classification system applied to systems, structures, and components of the Waste Isolation Pilot Plant (WIPP). This report is the result of that effort. In the subsections below the objectives, scope, and methodology for the assignment are discussed.

A. OBJECTIVES

The principal objectives of the review of the classification system for WIPP systems, structures, and components were determinations of:

- How well the WIPP classification system, quality groups, and design codes and standards compare with commercial nuclear power plants.
- Whether WIPP safety objectives are compatible with the requirements of 10CFR60 ("Disposal of High-Level Radioactive Wastes in Geologic Repositories").
- Whether the classification system described in Bechtel document D-76-K-03 ("Design Classification") complies with Department of Energy (DOE) criteria specified in DOE Order 6430.1 ("General Design Criteria Manual") and in Albuquerque Operations Office Order AL 5481.1A ("Safety Analysis and Review System for AL Operations").

In the process of meeting these objectives, TENERA also reviewed and commented on the accident analyses presented in WIPP SAR, the methodology described in Bechtel document D-76-K-03, and the assignment of specific components to the quality groups.

B. SCOPE

This evaluation was of necessity an overview of the WIPP classification system as presented primarily in the following documents:

- o WIPP SAR
- o Bechtel document D-76-K-03

The evaluation was performed using the method described in Section II.C below. The review did not include a detailed evaluation of the implementation of the classification system. Thus, the scope of the review did not include independent evaluation of the doses calculated in WIPP SAR, postulation of new or alternative accident scenarios, performance of independent failure modes and effects analyses, and review of Bechtel engineering documents such as drawings, calculations, and specifications. Furthermore, the degree of conservatism in the design of WIPP and the existence of defense in depth, both of which are important concepts in commercial nuclear power, were not reviewed.

C. METHODOLOGY

The evaluation methodology consisted of the following steps:

- o Definition of Objectives
- o Data Gathering

  This activity was performed in Santa Fe September 17 and 18. Copies of relevant documents were reviewed at EEG's offices and arrangements made for the shipment of some documents to TENERA's offices.

- o Review of WIPP SAR

  The following aspects of WIPP SAR were reviewed:

  - Design criteria and bases
  - Siting criteria
  - Accident analysis scenarios
  - Accident analysis assumptions

- Accident analysis results and conclusions
- Failure modes and effects
- System functions

This review was for the purpose of extracting WIPP-specific information which would be used to meet the objectives.

o Review of Other Documents

Other documents applicable to WIPP, DOE requirements, NRC regulations, and power reactor codes and standards were reviewed. These reference materials are listed in Appendix A. These reviews provided bases for evaluating the WIPP SAR.

o Development of Comparisons and Evaluations

The documents described above were compared. The results of the comparisons were evaluated in light of the objectives stated above.

o Development of Conclusions and Report

Based on the comparisons and evaluations, conclusions were reached regarding the objectives. These conclusions and our review process are documented in this report.

III. CLASSIFICATION SYSTEM OBJECTIVES AND PROCESS

This section presents a historical perspective of the classification system for systems, structures, and components (SSC) used at nuclear power plants in order to provide a point of reference for performing an evaluation of the classification system applied to WIPP. This section also documents the classification objectives and process stated in the WIPP SAR and Bechtel document D-76-K-03, and compares them with each other and with other standards such as DOE Order 6430.1, AL 5481.1A, 10CFR60, and nuclear power plant requirements.

A. CLASSIFICATION SYSTEMS FOR POWER REACTORS

1. History

The commercial nuclear power industry concepts that certain systems, structures, and components (SSC) are important to safety and that there should be a means of classifying the quality assurance requirements for these SSC have developed in parallel with the evolution of the methods for nuclear reactor safety evaluations. The earliest reactors were constructed from available commercial materials with special processing applied when necessary for the material to meet its operational function. For example, it is well known that the graphite used in CP-1, the first reactor, required several passes through a purification process before the reactor could operate. The early reactors had few (if any) safety systems added to them. The safety of these reactors depended upon inherent safety features (e.g., negative temperature coefficients of reactivity), design conservatism, and remote siting. Commercial materials were used whenever possible. Materials were uniquely specified when necessary.

The development of commercial nuclear reactors has brought about a recognition that what had been acceptable for remotely-sited test reactors would not be appropriate for power reactors. By their very nature, power reactors had to be located relatively near populations. Thus, distance alone could not be used to mitigate the consequences of an accident. In order for power companies to invest in reactors, they had to believe that the reactors were safe and reliable.

Furthermore, for regulations to be effective there had to be standards for evaluating reactor designs.

In 1962, TID-14844 ("Calculation of Distance Factors for Power and Test Reactor Sites") was published. This document provided the fundamental bases for nuclear plant siting. In fact, TID-14844 is still referenced in 10CFR100 today. TID-14844 used the following methodology for determining the suitability of a site:

- A source term of 100 percent of noble gases, 50 percent of halogens, and 1 percent of solids in the fission product inventory is assumed to be released by a non-mechanistic event.

- A source reduction factor of 50% is applied to iodines to account for plate out within the reactor building, thus reducing the source term for iodine release to the atmosphere.

- Assumptions are made regarding the containment leak rate, meteorology, radioactive material transport, and other factors.

- An exclusion area boundary dose is calculated assuming a two hour exposure and compared against limits of 25 rem (whole body) and 300 rem (thyroid).

- A low population zone boundary dose is calculated assuming an infinite exposure time and compared with limits of 25 rem (whole body) and 300 rem (thyroid).

- A population center distance is calculated by multiplying the low population zone radius by 1-1/3. The nearest boundary of a population of 25,000 or more must be further from the reactor than the population center distance.

Subsequent regulatory guidance, such as Regulatory Guide 1.4, modified the assumptions and methodology presented in TID-14844. This approach does not assume the existence of active safety systems that reduce the source term and, consequently, the resulting doses. However, the addition of systems such as containment spray systems and standby gas treatment systems allowed the designer to reduce the radioactive release to the environment by capturing significant portions of the source term before it could escape. Whereas TID-14844 was prepared as a methodology for calculating the required distance between a reactor and a population center, it was frequently used the other way around. Power reactors were sited using normal utility practices and the distance between the reactor and the nearest population center (as defined in TID-14844) determined. From the population center distance, the maximum distance to the low population zone boundary could be calculated. A dose at the LPZ boundary was calculated using the TID-14844 model. The ratio of the LPZ dose calculated in accordance with TID-14844 to the allowable dose determined the dose reduction factor required of the engineered safety features which mitigate the consequences of an accident.

The accident evaluated using TID-14844 is non-mechanistic. That is, no specific failure mechanism is assumed, although some plant parameters (e.g., containment pressure) are determined assuming a double-ended break in the largest pipe in the reactor coolant pressure boundary. The design requirements for emergency core cooling systems (ECCS) are such that even if the double-ended pipe break were to occur, damage to the fuel is limited because of the ECCS acceptance criteria (10CFR50 Appendix K).

Since 1962, the NRC has issued regulations, regulatory guides and other documentation that provide design engineers and analysts with guidance regarding classifications, quality requirements, and safety analyses. A number of principals and assumptions have evolved from this guidance:

- Components whose failure could result in doses to the general public in excess of 0.5 rem whole body should be subject to the QA program. (See Regulatory Guide 1.26)
- A component that could fail due to a seismic event and cause (or contribute to) a dose of 0.5 rem being received by people off-site should be designated as seismic Category I. (See Regulatory Guide 1.29.)

- o Postulated accidents are subject to a limit of 25 rem (whole body) and 300 rem (thyroid) at the site boundary; however, these limits have been modified such that for practical purposes the dose limits are 20 rem (whole body) and 150 rem (thyroid). (See Regulatory Guides 1.3 and 1.4)

- o High probability events have lower acceptable limits than low probability events, but very low probability events are deemed not credible and are not part of the safety evaluation of a reactor. Thus, although one can postulate low probability accident sequences in power reactors that result in doses in excess of 25 rem (whole body) or 300 rem (thyroid), these accidents are not part of the plant design basis and are not used to determine site suitability.

- o The strictest design and quality requirements apply to those components whose failure would initiate an accident sequence (i.e., the reactor coolant pressure boundary in power reactors).

- o Lower design and quality requirements apply to systems which are required to mitigate the consequences of an accident.

- o Still lower requirements apply to the systems which support (e.g, electric power and cooling water) the accident mitigation systems.

- o Commercial grade components are used for all other systems unless their failure prevents a higher grade system from operating.

It is also important to note the following footnote in 10CFR100:

> "The whole body dose of 25 rem referred to above corresponds numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations may be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, neither its use nor that of the 300 rem value for thyroid exposure as set forth in these site criteria guides are intended to imply that these numbers constitute acceptable limits for emergency doses to the public under accident conditions. Rather, this 25 rem whole body value and the 300 rem thyroid value have been set forth in these guides as reference values, which can be used in the evaluation of reactor sites with respect to potential reactor accidents of exceedingly low probability of occurrence, and low risk of public exposure to radiation."

The following regulations and regulatory guides affect the dose criteria and classification of SSC:

- o 10CFR50 Appendix A and Appendix B
- o 10CFR100
- o Regulatory Guides 1.3 and 1.4
- o Regulatory Guide 1.26
- o Regulatory Guide 1.29

Regulatory Guides 1.26 and 1.29 have the most direct applicability to the classification of SSC at WIPP. Regulatory Guide 1.26 defines quality groups A, B, C, and D whereas Regulatory Guide 1.29 covers the seismic classification (i.e., seismic Category I or non-Category I). These classifications, as they are used in power reactors and how they compare with the systems described in WIPP SAR and in Bechtel document D-76-K-03, are discussed in detail in Sections IV and V of this report.

Thus, for nuclear power plants two off-site dose-related criteria exist:

- o Site suitability is based on the 25 rem/300 rem limits
- o QA program and seismic Category I applicability are based on the 0.5 rem limit.

Nuclear power plant systems are designed so that there is a low probability of an event which could cause doses comparable to those in Part 100. The quality groups defined in Regulatory Guide 1.26 use a graded approach based on whether a system directly prevents or mitigates the consequences of an accident or provides support to such systems. In Regulatory Guide 1.26 the reactor coolant pressure boundary is placed in Quality Group A. Systems such as emergency core cooling systems (ECCS) are in Quality Group B, and support systems such as service water are in Quality Group C. Non-safety related components are Quality Group D.

2. Quality Assurance Classification System Objectives

Economic theory and quality assurance have a very close relationship. If one assumes that all costs can be reduced to dollars, then one would expect to apply additional quality assurance requirements (e.g., inspections, tests, audits) until the marginal cost of doing the extra work exceeded the marginal reduction in costs associated with defects. Because of the difficulty of making this determination, significant amounts of judgment are applied in setting quality assurance requirements.

As discussed in Section III.A, the classification system at nuclear power plants has evolved to three classification groups (A, B, C) for safety-related components and a fourth group (D) for non-safety-related components. The objectives of such a systematic approach to quality assurance include:

- Consistency of QA requirements for components of a comparable level of importance to safety
- Reduction in the level of effort required to implement a QA program by introducing standardization
- Enhancement of the ability to make comparisons between reactors of different designs.

Although SSC are grouped into A, B, C, and D classes, the application of a quality assurance program in a nuclear plant design is essentially a "yes/no" situation much like the seismic/non-seismic classification. That is, either the quality assurance program applies ("Q component") or it does not apply ("non-Q component"). Table III-1 presents this situation.

TABLE III-1

TYPICAL SEISMIC AND QUALITY PROGRAM REQUIREMENTS

Quality Class	QA Program	Seismic Category
A	Q	Cat I
B	Q	Cat I
C	Q	Cat I
D	non-Q	non-Cat I

Although historically other combinations such as "quality group C, but non-Category I" and "D + QA" existed, Table III-1 represents the typical classification system for current power reactors. Thus, one may fairly ask how groups A, B, and C differ since all three are "Q" and seismic Category I. The answer is that the differences are in the detailed design, inspection, and testing requirements. In particular, mechanical components are subject to design requirements which differ depending upon their class. On the other hand, electrical components are in one of two classes (Class I-E or non-class I-E) and structures (other than the containment) are all quality group C.

B. WIPP SAR CLASSIFICATION OF SYSTEMS, STRUCTURES, AND COMPONENTS

The principal design criteria for WIPP are discussed in Chapter 3 of the WIPP SAR. Specifically, Subsection 3.2 discusses structural and mechanical design criteria, Subsection 3.3 discusses safety protection criteria and Subsection 3.4 discusses the classification of structure systems and components. Subsection 3.4 briefly defines three design classes. Design Class I is defined as those components whose functions are essential to the prevention or mitigation of the consequences of an accident which could result in a 50-year dose commitment to the whole body, bone or gonads of 25 rem, or 75 rem to other organs. This dose is to be calculated for people beyond the protective area boundary. The WIPP SAR goes on to note that Design Class I items are regarded as basic components within the meaning of 10CFR21. Furthermore, the WIPP SAR states that no Design Class I items have been identified for WIPP.

10CFR21 was added to Title 10 Code of Federal Regulations as a result of Section 206 of the Energy Reorganization Act of 1974. Part 21 requires reporting defects in basic components of the facility to the NRC. Basic components are those whose failure would create a substantial safety hazard. In turn, substantial safety hazard is defined in Part 21 as "a loss of safety function to the extent that there is a major reduction in the degree of protection provided to public health and safety for any facility or activity licensed . . . pursuant to Parts . . . 60 . . ." For nuclear power plants the definition of basic component is such that equipment used to prevent or mitigate the consequences of accidents could result in potential offsite exposures comparable to those in Part 100. Because "major reduction" is not further defined, it is reasonable to assume that a substantial safety hazard for a facility licensed under Part 60 would be one which has the potential for resulting in offsite exposures comparable to the Part 100 limits. In this regard, the definition of Design Class I for WIPP and basic component within the meaning of Part 21 appear to be consistent. (Note: ANSI/ANS 51.1 states " A single item of SC-1, SC-2, or SC-3 equipment and a 'Basic Component' as defined in Code of Federal Regulations, Part 21, . . . are equivalent.")

Design Class II is defined in WIPP SAR to be those systems, structures and components which are not Design Class I, but which perform certain specified safety functions including providing, shielding, and monitoring important variables. The applicability of 10CFR21 to Design Class II is not discussed.

Design Class III applies to all items which are neither Design Class I nor Design Class II.

The WIPP SAR provides the information regarding Design Class interfaces which are consistent with comparable interface requirements in nuclear power plants.

The WIPP SAR also states that quality assurance requirements contained in Chapter II are applied to Design Class I items and that quality requirements may be applied to selected items in Design Classes II and III depending on the importance of the item to the reliable operation of WIPP and on the complexity and uniqueness of the item. This classification of components for which the quality assurance program is applicable is discussed in subsequent portions of this report.

C. BECHTEL DOCUMENT D-76-K-03 CLASSIFICATION OF SYSTEMS, STRUCTURES AND COMPONENTS

Bechtel document D-76-K-03 provides the design classification for WIPP. Our review used revision 9 of this document, which is dated November 30, 1983. Revision 0 which was issued for review was dated January 18, 1979. Revision 7 of the Bechtel document was issued November 30, 1982 and thus preceded the January 1983 version of the WIPP SAR which was discussed in the previous section. The most immediate reaction to the Bechtel document after reviewing WIPP SAR is that there is a degree of inconsistency in basic terminology between the two documents.

The Bechtel document discusses four classes. Design Class I is defined essentially in the same manner as Design Class I in WIPP SAR. Design Class II is also defined in much the same way as Design Class II in WIPP SAR. However, Design Class III has been divided into two classes: Design Class IIIA and Design Class IIIB. Design Class IIIA is defined as those components for which a higher level of quality is desired beyond that expected in commercial industrial practice. The Bechtel document provides bases to determine which components must meet this requirement. Design Class IIIB is comparable to Design Class III in WIPP SAR. One may take the position that Design Class IIIA and Design Class IIIB are simply a means of implementing the SAR statement that Design Class II and III items may have additional quality assurance requirements applied to them. In fact, Design Classes I, II, and IIIA are shown in Table 2 of the Bechtel document to be subject to the quality assurance program. This would indicate that the approach being used by Bechtel, in fact, meets the requirements stated in WIPP SAR.

The Bechtel document contains a flow chart procedure (Figure 1) for design classification. This procedure is consistent with the definitions contained elsewhere in the Bechtel document. A copy of this figure is provided as Appendix C to this report.

D. COMPARISON AND EVALUATION

1. Other Classification Methods

For the purposes of the following discussion, the other classification methods considered are:

- o 10CFR60
- o DOE Order 6430.1
- o ANSI/ANS 51.1 and ANS/ANSI 52.1
- o ONWI 390 and ONWI 496

These methods are discussed in the following three subsections. Subsection III.D.2 compares these methods.

a. 10CFR60

Title 10, Code of Federal Regulations, Part 60 (10CFR60) contains the Nuclear Regulatory Commission (NRC) requirements for the disposal of high-level radioactive wastes in geological repositories.

The NRC published 10CFR60 in final form in 1983. Because of the extensive comments received on proposed Part 60 published in 1981, the NRC published a report, NUREG-0804, containing the text of the comments, an analysis of the comments, and the NRC's response. The scope of 10CFR60 is such that it may be compared to 10CFR50 and 10CFR100, which are applicable to nuclear power plants. A comparison of Part 60 against Parts 50 and 100 yields the following observations:

- o Although Part 100 contains numerical siting criteria through the specification of a radioactive source term, dose model, and allowable dose limits, Part 60 does not contain such detailed information.

- o Although Part 50, including Appendices A and B, uses the terms "important to safety" and "safety related," neither are defined. Both appendices have been in the regulations for over twelve years but the definitions of these terms are still being resolved. At a July 31, 1984 meeting, the NRC Commissioners called for a debate on the question of whether the two terms ("important to safety" and "safety related") are equivalent.

- o Part 60 uses the term "important to safety" and defines it as:

  "Those engineered structures, systems, and components essential to the prevention or mitigation of an accident that could result in a radiation dose to the whole body, or any organ, of 0.5 rem or greater at or beyond the nearest boundary of the unrestricted area at any time until the completion of permanent closure."

  Part 60 consistently uses "important to safety" when setting design requirements. See, for example, 60.131(b)(1) through (6). Paragraph 60.131(b)(5)(i) states, "Each utility service system that is important to safety shall be designed so that essential safety functions can be performed under both normal and accident conditions."

  One could make the interpretation that the NRC intended to use the 0.5 rem dose in Part 60 as a basis for determining site suitability. NUREG-0804, however, clarifies the NRC's intent and states, "Structures, systems, and components are important to safety if, in the event they fail to perform their intended function, an accident could result which causes a dose commitment greater than 0.5 rem to the whole body or any organ of an individual in an unrestricted area. The value of 0.5 rem is equal to the annual dose to the whole body of an individual in an unrestricted area that would be permitted under 10CFR Part 20 for normal operations, the same as permitted for normal operations of certain other activities licensed by NRC. Such systems, structures, and components would be subject to additional design requirements and to a quality assurance program to ensure that they performed their intended functions . . . In the final rule, the term 'important to safety' applies solely to the functioning of structures, systems, and components during the period of operations prior to repository closure."

Thus, the 0.5 rem is a definition of important to safety. That is, a component whose failure results in a dose greater than 0.5 rem must be considered important to safety and subject to a quality assurance program. Furthermore, because the 0.5 rem dose is related to 10CFR20, it is also reasonable to believe that the dose should be calculated as a dose to an individual located offsite. No dose limit for an onsite individual is specified in Part 60 except for a reference to Part 20 being applicable in the restricted area [60.131(a)]. Parts 50 and 100 do not specify a dose limit for onsite personnel under accident situations except that the control room of a nuclear power plant must be designed to limit doses to operators to 5 rem whole body (see 10CFR50, Appendix A, Criterion 19). In a nuclear power plant safety evaluation report (SAR), no evaluation is generally made of doses to other plant personnel immediately following the accident. Information comparable to the SAR evaluation of dose to facility personnel in accident situations is typically not contained in nuclear power plant SARs.

- The original version of 10CFR60 defined important to safety as those components "that provide reasonable assurance that radioactive waste can be received, handled, and stored without undue risk to the health and safety of the public." Essentially, the identical definition of important to safety is contained in Section 3.4 of WIPP SAR.

The following conclusions are drawn about the use of 10CFR60 in evaluating classification of systems, structures, and components:

- Components whose failure results in an offsite dose of 0.5 rem, should be classified as "Q" and "seismic Category I" components (to use nuclear power plant terminology).

- Because no doses for siting criteria are specified, it is reasonable to use 10CFR100 as an upper limit, although notice should be taken of Regulatory Guides 1.3 and 1.4 which specify maximum doses of 20 rem (whole body) and 150 rem (thyroid) for evaluation of sites prior to issuance of a construction permit.

- o Only low probability events should be allowed to result in doses comparable to the Part 100 limits. Systems should be provided to mitigate the consequences of higher probability events that result in doses approaching the Part 100 limits.

- o Because no accident source term and model is specified for accident evaluations, care must be applied to define adequately the source term (e.g., total quality of radioactivity involved, the fraction of the total that is released, nuclide mix), the evaluation model (e.g., leakage paths, meteorology, dose conversion factors), and facility system availability (e.g., "Q" systems only, offsite power availability, application of single failure criterion).

- o In considering the use of a 0.5 rem criterion for classification of a component as "important to safety," it should be understood that this criterion applies to unexpected failures and not to planned activities. Thus, this criterion applies in situations fundamentally different than for normal activities for which other criteria are appropriate. For example, 10CFR50 (Appendix I), 40CFR190, and proposed 40CFR191 all contain dose values that are significantly lower than 0.5 rem. These apply to routine releases of radioactive waste, and are a reflection of the "as low as reasonably achievable" concept. The maximum "normal" release limit is governed by 10CFR20 which is the basis for the 0.5 rem criterion. 40CFR190 and proposed 40CFR191 contain specific exemptions for abnormal operating conditions. Thus, there is no conflict if a 0.5 rem criterion is used to evaluate whether a particular component must be subject to a quality assurance program.

b. DOE Order 6430.1

DOE Order 6430.1 is a general design criteria document issued by the Department of Energy and is applicable to DOE facilities. It provides a wide range of requirements in electrical, mechanical, structural and architectural areas and contains specific requirements for systems such as fire protection. It is applicable to DOE facilities ranging from reactors to plutonium processing and handling facilities. Because of the wide range of applications to which the DOE Order may be applied, it is of necessity very general. However, it does provide some general guidelines which can be used in the evaluation of WIPP. For example, the DOE Order (page IV-8) states that a 10,000-year mean recurrence level for natural phenomena should be used for design of the structures which are high hazard and high value. Specifically, in-process plutonium containing facilities fall within the requirement that 10,000-year mean recurrence level events be considered in the design. Such events include earthquakes, tornados, and similiar hazards. (However, on page XXI-6 of the DOE Order a 1,000,000 year recurrence interval is stated as being required for tornado design of plutonium facilities.) The DOE design criteria document also contains requirements for power system reliability, emergency power requirements, structural load combinations and fire protection.

Thus, while the DOE General Design Criteria Manual provides useful information and guidance which may be used to supplement information available from other sources, it does not contain a specific classification system nor does it reference on a consistent basis applicable codes and standards which can be directly related to safety system classifications.

c. ANSI/ANS 51.1 and 52.1

ANS 51.1 and 52.1 are American National Standards which provide a classification system applicable to commercial pressurized and boiling water reactors. These documents are consensus standards developed over a period of years by the Nuclear Power Plants Standards Committee (NUPPSCO) of the American Nuclear Society. The working groups responsible for the standards, Subcommittees 51 and 52, and NUPPSCO have broad representation reflecting all aspects of the commercial nuclear industry. These standards reflect the evaluation of several years of development and comments on previous standards and drafts.

These standards contain three important concepts:

- o A relationship between an acceptable dose level and the probability of occurrence of an event resulting in a dose that large.

- o A procedure for determining which combinations of independent events are appropriate for evaluation.

- o A classification system with three safety-related classes and one non-safety class.

Table III-2 provides the correlation between acceptable dose and event probability. Table III-3 defines three ANSI/ANS safety classes. Appendix B contains Tables 3-5 through 3-8 of ANSI/ANS 51.1 which provide further details concerning design requirements.

TABLE III-2

OFFSITE RADIOLOGICAL DOSE CRITERIA
FOR PLANT CONDITIONS

Best-Estimate Frequency of Occurrence (F) Per Reactor Year	Plant Condition (PC)	Offsite Radiological Dose Criterion
Normal Operations	PC-1	10 CFR 50, App. I
F 10^{-1}	PC-2	10 CFR 50, App. I
10^{-1} F 10^{-2}	PC-3	10% 10 CFR 100
10^{-2} F 10^{-4}	PC-4	25% 10 CFR 100
10^{-4} F 10^{-6}	PC-5	100% 10 CFR 100

Source: ANSI/ANS 51.1

TABLE III-3

SAFETY CLASS DEFINITIONS
FROM ANSI/ANS 51.1

Class	Definition
SC-1	Safety Class 1 (SC-1) shall apply to pressure-retaining portions and supports of mechanical equipment that form part of the RCPB whose failure could cause a loss of reactor coolant in excess of the reactor coolant normal makeup capability and whose requirements are within the scope of the ASME Boiler and Pressure Vessel Code, Section III.
SC-2	Safety Class 2 (SC-2) shall apply to pressure-retaining portions and supports of primary containment and other mechanical equipment, requirements for which are within the scope of the ASME Boiler and Pressure Vessel Code, Section III, that are not included in SC-1 and are designed and relied upon to accomplish the following nuclear safety functions:
	a. Provide fission product barrier or primary containment radioactive material holdup or isolation.
	b. Provide emergency heat removal for the primary containment atmosphere to an intermediate heat sink, or emergency removal of radioactive material from the primary containment atmosphere (e.g., containment spray).
	c. Introduce emergency negative reactivity to make the reactor subcritical (e.g., boron injection system), or restrict the addition of positive reactivity via pressure boundary equipment.
	d. Ensure emergency core cooling where the equipment provides coolant directly to the core (e.g., residual heat removal and emergency core cooling).
	e. Provide or maintain sufficient reactor coolant inventory for emergency core cooling (e.g., refueling water storage tank).
SC-3	Safety Class 3 (SC-3) shall apply to equipment, not included in SC-1 or -2, that is designed and relied upon to accomplish the following nuclear safety functions:
	a. Provide for functions defined in SC-2 where equipment, or portions thereof, is not within the scope of the ASME Boiler and Pressure Vessel Code, Section III.

TABLE III-3

SAFETY CLASS DEFINITIONS
FROM ANSI/ANS 51.1
(CONTINUED)

Class	Definition
SC-3 (cont.)	b. Provide secondary containment radioactive material holdup, isolation, or heat removal.
	c. Except for primary containment boundary extension function, ensure hydrogen concentration control of the primary containment atmosphere to acceptable limits.
	d. Remove radioactive material from the atmosphere of confined spaces outside primary containment (e.g., control room or fuel building) containing SC-1, -2, or -3 equipment.
	e. Introduce negative reactivity to achieve or maintain subcritical reactor conditions (e.g., boron makeup).
	f. Provide or maintain sufficient reactor coolant inventory for core cooling (e.g., reactor coolant normal makeup system).
	g. Maintain geometry within the reactor to ensure core reactivity control or core cooling capability (e.g., core support structures).
	h. Structurally load-bear or protect SC-1, -2, or -3 equipment.
	i. Provide radiation shielding for the control room or offsite personnel.
	j. Ensure required cooling for liquid-cooled stored fuel (e.g., spent fuel storage pool and cooling system).
	k. Ensure nuclear safety functions provided by SC-1, -2, or -3 equipment (e.g., provide heat removal for SC-1, -2, or -3 heat exchangers, provide lubrication of SC-2 or -3 pumps, provide fuel oil to the emergency diesel engine).
	l. Provide actuation or motive power for SC-1, -2, or -3 equipment.
	m. Provide information or controls to ensure capability for manual or automatic actuation of nuclear safety functions required of SC-1, -2, or -3 equipment.

TABLE III-3

SAFETY CLASS DEFINITIONS
FROM ANSI/ANS 51.1
(CONTINUED)

Class	Definition
SC-3 (cont.)	n. Supply or process signals or supply power required for SC-1, -2, or -3 equipment to perform their required nuclear safety functions.
	o. Provide a manual or automatic interlock function to ensure or maintain proper performance of nuclear safety functions required of SC-1, -2, or -3 equipment.
	p. Provide an acceptable environment for SC-1, -2, or -3 equipment and operating personnel.
NNS	Non-Nuclear Safety (NNS) shall apply to equipment that is not included in SC-1, -2, or -3. This equipment is not relied upon to perform a nuclear safety function.

The advantages of using the ANSI/ANS standards as the basis against which other classification systems are evaluated include:

- o A logical framework underlies the classification system
- o The system reflects the experience of many years of development
- o The system is generally consistent with Regulatory Guides 1.26 and 1.29.

The primary disadvantage is that the NRC has not formally adopted these standards by reference in regulatory guides. Because of the similarity of the standards with Regulatory Guides 1.26 and 1.29, this is a slight disadvantage and is more than offset by the advantages.

d. ONWI 390 and ONWI 496

Two reports from the Office of Nuclear Waste Isolation (ONWI) contain information relating to the classification of systems, structures, and components. These reports (ONWI 390 and ONWI 496) are a conceptual design report and a preliminary design report for the Exploratory Shaft Facility -- Paradox Basin. Although these reports contain a three-level classification system, the bases seem to be different than those in use at WIPP. In particular, the definition of Class I in the ONWI reports seems to cover all systems, structures, and components containing or potentially containing radioactive material. Thus, confinement structures are Class I under this system, but are Design Class II at WIPP. The ONWI reports do not contain a detailed listing of codes and standards which makes direct comparison difficult. Therefore, it was concluded that the ONWI reports are not an appropriate standard against which to compare the WIPP classification system.

2. Comparison of Classification Methods

It is obvious that differences exist among all of the classification methods discussed above. However, it is also obvious that there is a great deal of similarity among them. The differences between the Bechtel document and the WIPP SAR appear to be primarily at the level of defining the components to which additional quality assurance requirements should be applied. As noted above, the Bechtel document appears to envelope the requirements of WIPP SAR. Because no specific methodology is described in WIPP SAR one cannot readily determine the procedure by which the classification system would be implemented. However, given the consistency between these documents and in the definitions of Classes I and II, it appears that the methodology of the Bechtel document is consistent with the intent of WIPP SAR. Furthermore, the Bechtel document and WIPP SAR make use of the graded approach to quality assurance. That is, factors such as importance to safety and reliability are considered in selecting the applicable quality assurance requirements.

Table III-1 indicated that nuclear power plant classification systems could be reduced in their simplest forms to two classes, one class consisting of "Q" and seismic Category I components and a second class consisting of all other components. This approach is consistent with the requirement of 10CFR60 in that components whose failure could result in exceeding a dose limit of 0.5 rem whole body must be subject to the quality assurance program. The systems are also consistent in that doses in excess of the Part 100 limits are not permitted for a credible event. A credible event must, of course, be defined. The ANSI/ANS standards provide that events with a frequency occurrence of less than 10^{-6} need not be considered in the design basis of a nuclear power plant. Furthermore, events of low probability (10^{-4} to 10^{-6} events per year) are allowed to approach the Part 100 limits. Part 60, on the other hand, provides only a criterion for deciding which components must be subject to the quality assurance program. It does not provide a numerical site suitability criterion such as that contained in Part 100 for nuclear power plants. Reviews of other nuclear waste documents such as DOE Order 6430.1 and ORNL-TM-4219 indicate general agreement that the Part 100 limits should apply to any credible accident involving facilities other than nuclear power plants.

Of the documents reviewed, the Bechtel document contains the most specific information applicable to WIPP. Therefore, to accomplish the comparison between nuclear power plant requirements and the WIPP design basis, the Bechtel D-76-K-03 document will be used as the definition of WIPP requirements and the ANSI/ANS document will be used to define nuclear power plant requirements.

3. Comparison of Bechtel Document D-76-K-03 with ANSI/ANS 51.1

Both the Bechtel document D-76-K-03 and ANSI/ANS 51.1 use a system of classification based on four discrete equipment classes. The lowest levels in both systems (NNS in the ANSI document and Class IIIB in the Bechtel document) refer to commercial grade items to which the quality assurance program does not apply. The ANSI document places components in Safety Class I if their failure could result in the direct release of potentially radioactive materials. This class consists of the reactor coolant pressure boundary, including the reactor vessel and attach piping as further defined in 10CFR50. ANSI Safety Class 2 and Safety Class 3 are defined on somewhat different bases than Design Classes II and IIIA in the Bechtel document. Many of these differences are due to inherent differences between a nuclear power plant and a waste repository, but similarities also exist. For example containment of radioactivity released after an accident is considered a Safety Class 2 function in ANSI and Design Class II in the Bechtel document. ANSI Safety Class 3 includes those components which enable higher class components to achieve their safety function. For example, cooling water and electric power supplies are Safety Class 3 even when they are supplying services to Safety Class 2 components. The Bechtel document on the other hand is not as specific. The procedure for design classification designation in the Bechtel document states as a basis for determining whether an item should be classified as Design Class IIIA the following question: "Are special design requirements necessary to assure that failure of this structure, system, or component will not result in a significant shutdown of the facility or inhibit the accessibility or maintainability of required equipment or have special significance to health and safety of operations personnel?"

In summary, the concept discussed in the Bechtel document is consistent with the general approach contained in the ANSI/ANS document.

Table III-4 is a comparison of the terminology used in the various documents reviewed.

TABLE III-4

CLASSIFICATION SYSTEM COMPARISON

Bechtel Document D-76-K-03	ANSI/ANS 51.1 Standard	Regulatory Guide 1.26	"Q"/Seismic
I	SC I	A	Q/Seismic
II	SC 2	B	Q/Seismic*
IIIA	SC 3	C	Q/**
IIIB	NNS	D	Non-Q/Non Seismic

* Design Class II components for confinement and control of radioactivity required to be seismic, but other components are evaluated on a case-by-case basis.

** The ANSI/ANS standard requires SC-3 to meet seismic requirements. Regulatory Guide 1.26 does not address seismic requirements. Design Class IIIA is evaluated on a case-by-case basis.

IV. CLASSIFICATION SYSTEM APPLICATION

This section discusses the application of the classification system methodologies discussed in the previous section. This discussion is in three parts. The first describes the factors that affect classification. The second discusses the failure modes and effects evaluations that are contained in WIPP SAR and provides information about system safety functions. The third part is a comparison of the results of the application of the classification methodologies.

A. FACTORS AFFECTING CLASSIFICATION

The factors affecting classification may be generally grouped into two categories. The first category may be labeled "acceptance criteria." By this we mean criteria which are used to determine the acceptability of a component being classified within one category as opposed to a higher category. The second grouping of factors affecting classification are the assumptions which are used to determine whether an acceptance criterion is met.

One of the acceptance criteria used in the implementation of classification systems is the minimum threshold for inclusion in the "Q" or seismic groups. At the other extreme is the upper limit for site suitability. For WIPP (and for Part 60 facilities) no site suitability dose criterion is stated, but the Part 100 limits used for reactor site evaluation appear to be generally accepted.

For WIPP the upper limit criterion is comparable to the Part 100 limit. Items whose failure could result in 50-year dose commitment in excess of 25 rem at the protected area boundary are Design Class I. Although not stated in WIPP documents, the expected approach would be to apply the highest quality standards to Design Class I components (to minimize the probability of an accident) and to provide safety systems (e.g., atmospheric cleanup systems) to ensure that even if the failure were to occur, the resultant dose would not exceed the site suitability limits. In power reactors the highest quality standards are applied to the reactor coolant pressure boundary.

The acceptance criteria for Design Classes II and IIIA for WIPP (and Safety Classes 2 and 3 for nuclear power plants) are not directly related to dose except to the extent that Safety Class 3 is affected by the application of Regulatory Guides 1.26 and 1.29 for nuclear power plants. Both power reactors and WIPP make use of a concept of importance to safety. In the power reactor case, Safety Class 2 applies to those systems which directly mitigate the consequences of an accident and Safety Class 3 applies to support systems. For WIPP the confinement systems used to respond to an accident are classified as Design Class II. The criteria stated in the Bechtel document and WIPP SAR do not directly address support systems such as electric power and cooling water. However, a review of the implementation as shown in Attachment I to Bechtel document D-76-K-03 shows that the emergency power system* is classified as Design Class IIIA. For power reactors the emergency generator is classified as Safety Class 3 (or quality group C as defined in Regulatory Guide 1.26). Thus we can reach a reasonable conclusion that the acceptance criteria used for both WIPP and power reactor classification systems are comparable, with due consideration given to the apparent differences between these types of facilities. Table IV-I provides a comparison between the terminology used in the Bechtel document and that used in the ANSI document. The comparison also contains the classification system provided in Regulatory Guide 1.26.

To determine compliance with the acceptance criteria discussed above, it is necessary to make numerous assumptions regarding plant operation, system reliability, and combinations of events. For nuclear power plants the general

* Although D-76-K-03 refers to the diesel-generator as the "emergency diesel-generator," the SAR currently uses terms such as "onsite engine generator," rather than "emergency diesel-generator" which was used prior to Amendment 8. It is not clear if any significance should be attached to the change in terminology. The SAR does not describe the standards which are applied to the diesel-generator. In nuclear power plants various terms, including "emergency diesel-generator," "standby diesel-generator," and "auxiliary diesel-generator" are used interchangeably. For consistency, the term "emergency diesel-generator" is used in this report.

TABLE IV-1

CLASSIFICATION TERMINOLOGY

Qualitative Description	WIPP Classification	ANSI/ANS 51.1 Classification	Regulatory Guide 1.26
Highest Quality	Design Class I	Safety Class 1	A
	Design Class II	Safety Class 2	B
	Design Class IIIA	Safety Class 3	C
Commercial Grade Items	Design Class IIIB	Non-nuclear Safety (NNS)	D

Note: This table provides only a comparison of classification systems against qualitative descriptions of quality requirements. This table does not imply that at each level specific requirements are equivalent.

design criteria of 10CFR50, Appendix A provide the basic requirements for conducting evaluations of systems, structures, and components that are important to safety. Although specific design criteria exist for WIPP, there is no set of applicable general design criteria to which the plant design may be compared. That is, no set of criteria directly comparable to 10CFR50, Appendix A or 10CFR60 exists for WIPP. Existing SAR and other criteria reflect implementation of design considerations, but do not present general concepts such as the single failure criterion in nuclear power plants. Table IV-2 is a comparison between general design criteria contained in 10CFR60 and comparable nuclear power plant criteria.

The following points are noted:

- Except in the area of criticality safety, the single failure criterion is not directly addressed in Part 60.
- No source term or design basis accident is defined in Part 60, thus the designer must define his own source term and design basis accident.
- Release fractions from the potential source term are not defined in Part 60, whereas nuclear power plants have guidance regarding fractions of core inventory which must be assumed to be released. The design of WIPP is based on assumptions regarding the percentage of material released and the percentage of that material which is 10 microns or less. Clearly these are important assumptions.
- Power plant regulatory requirements specify applicable meteorology for accident assessment.
- The basic safety criteria in Part 60 are comparable to those in Part 50, Appendix A.
- Part 50 requires that consideration be given to combinations of events. The sets of these combinations that must be considered have evolved for power reactors and are specified in regulatory guides. Part 60 is less explicit.

In general, it may be concluded that Part 60 provides less guidance than Appendix A regarding analysis assumptions and design criteria. DOE Order

TABLE IV-2

PART 60 DESIGN CRITERIA
FOR SYSTEMS, STRUCTURES AND COMPONENTS
IMPORTANT TO SAFETY

Part 60 Reference	Subject	Part 50, App A Reference	Comment
60.131(b)(1)	Protection against natural phenomena and environmental conditions	Criterion 2 Criterion 4	Part 60 requires protection against "anticipated" natural phenomena whereas Appendix A requires "most severe" historically reported conditions plus a margin.
60.131(b)(2)	Protection against dynamic effects of equipment failure	Criterion 4	Requirements are comparable
60.131(b)(3)	Protection against fires and explosions	Criterion 3	Requirements are comparable
60.131(b)(4)	Emergency capability	Criterion 16 Criterion 19 Criterion 34 Criterion 35 Criterion 38 Criterion 41 Criterion 50 Criterion 64 10CFR50, App E	Part 50 is much more detailed in its requirements than is Part 60.
60.131(b)(5)	Utility Services	Criterion 17 Criterion 44	Appendix A is much more detailed and requires independence as well as redundancy for the onsite electric system. Two circuits to the offsite transmission network are required. Single Failure assumption is required by Appendix A.
60.131(b)(6)	Inspection, testing and maintenance	Criterion 18 Criterion 21 Criterion 36 Criterion 37 Criterion 39 Criterion 40 Criterion 42 Criterion 43 Criterion 45 Criterion 46 Criterion 52 Criterion 53	Inspection and testing are addressed at the system level in Appendix A, but otherwise the requirements are comparable.
60.131(b)(7)	Criticality control	Criterion 62	Part 60 is more detailed and prescriptive.
60.131(b)(8)	Instrumentation and control	Criterion 13	Requirements are comparable.
60.131(b)(9)	Compliance with mining regulations	Not applicable	---
60.131(b)(10)	Shaft conveyances	Criterion 61 NUREG-0612	Although not directly comparable, generally comparable requirements exist.

6430.1 was reviewed to a limited extent. In some areas the DOE document is more detailed than Appendix A (in fact its level of detail is comparable to a Regulatory Guide); however, it lacks guidance on the assumptions that should be used in making evaluations.

It would be of benefit in evaluating WIPP to have a set of general design criteria that are comparable to those in 10CFR60 and 10CFR50, Appendix A.

B. FAILURE MODES AND EFFECTS

WIPP SAR contains detailed failure modes and effects analyses addressing the consequences of various failures in the facility's major systems. An independent verification of these analyses was not performed, but a general review was made. The following comments and observations resulted from this review:

- The formats of the analyses differ. For example, Table 4.4-15 (fire protection) defines the sequence of events and remedial actions for each failure mode whereas other tables, such as Table 4.3-1, do not provide this valuable information.

- Operational modes are not clearly evaluated in the failure modes analyses. It is not clear whether any of the Chapter 7 accidents have been combined with the Chapter 4 failure analyses to determine whether an accident combined with a single failure is credible and, if credible, whether the consequences are acceptable.

- Emergency mode is not defined for Table 4.4-4. It appears that emergency as used in this table is operation with the emergency diesel and not an accident event.

- A statement is made on page 4.4-22 regarding loss of power from the Potash substation. According to the Environmental Evaluation Group, power has been lost at the site at least twice since WIPP construction began. This discrepancy should be reviewed and resolved so that an estimate of the reliability of the offsite power system can be assessed.

- o Consequences to other systems from a failure in the system being evaluated are not addressed. For example, in the electric power system failure analysis, no discussion is provided of the consequences in hoist systems. No discussion is provided indicating the acceptable interval between loss of power and starting of the diesel.

Recommendations concerning this area of review are provided in Section VI of this report.

C. CLASSIFICATION IMPLEMENTATION COMPARISON

The "bottom line" of any classification system is the determination of applicable class for each type of component. For WIPP a review was made of the information contained in Attachment I to Bechtel document D-76-K-03. There are no Design Class I components at WIPP. This is because no accident scenario was identified, which is considered credible, that yields a dose in excess of limits specified for categorization as Design Class I. It is important to note, however, that this conclusion is based on specific accident scenarios and associated assumptions. The use of alternative assumptions could result in higher doses. We believe that the dose values specified for Design Class I are appropriate for that classification. We have not made an exhaustive review of the accident scenarios presented in WIPP SAR. Our limited review indicates a number of areas where changes in assumptions could result in significant dose increases. Most critical among these are the assumptions concerning percentages of material released from damaged containers and the percentage of those materials which are respirable.

A listing of Design Class II and Design Class IIIA items was prepared from the attachment to the Bechtel document. This listing was then sorted into general categories of equipment to facilitate a comparison between WIPP systems and nuclear power plant systems. Table IV-3 presents this information for Design Class II and IIIA components.

A review was also made of components classified as Design Class IIIB. In all cases components designated as Design Class IIIB are comparable in function to

TABLE IV-3

EXAMPLES OF WIPP AND NUCLEAR POWER PLANT
CLASSIFICATION OF SYSTEMS, STRUCTURES, AND COMPONENTS

	WIPP		Nuclear Power Plant	
General Description	Examples	Design Class	Examples	Safety Class
Structures	Waste Handling Building	II	Containment	2
	Cask Loading Room	II		
	Exhaust Filter Building	II		
	Hot Cell	II		
	Support Building	IIIA	Auxiliary Building	3
	Water Pump House	IIIA	Diesel Generator Building	3
			Control Building	3
			Ultimate Heat Sink	3
	Warehouse/Shop	IIIB	Radwaste Building	NNS
			Turbine Building	NNS
Isolation Features	Shield Plugs	II	Piping and Valves	2
	Shield Valves	II	Forming Containment Isolation System	
	Tornado Doors and Dampers	II		
Electric Power System	Emergency Diesel Generator and Associated Equipment	IIIA	Emergency Diesel Generator and Associated Equipment	3
	Uninterruptable Power Supply	IIIA	Emergency 125/250 V DC System	3
	13.8 kV System	IIIB	Offsite Power System	NNS
Air Cleaning HVAC	Hot Cell HEPAs/Dampers	II	Standby Gas Treatment System (BWRs)	2
	RM Waste Handling Area HEPA/Dampers	II		
	U/G Exhaust System HEPA/Dampers	II	Containment Spray System (PWRs)	2
	U/G Exhaust Fans	II		
	CMR HVAC System	IIIA	Control Room HVAC	3
Fire Protection	Electric and Diesel Fire Water Pumps	IIIA		
	Diesel Fuel Day Tank for Fire Water Pump	IIIB		
	Other Fire Protection Systems	IIIB	Fire Protection System	NNS
Radiation Monitoring	RM/CM Area Rad Monitor	IIIA	Inputs to Reactor Protection System	3
	RM/CM Continuous Rad Monitors	IIIA	Accident Monitoring	3
	Air Sampling System	IIIA	Effluent Monitoring	NNS
			Area Monitors	NNS

Note: NNS = Non-nuclear safety.

components which are designated as non-nuclear safety (NNS) or quality group D in nuclear power plant terminology.

Table IV-3 shows that systems which perform comparable functions in nuclear power plants and at WIPP are assigned to comparable quality levels on a relative scale such as shown in Table IV-1. That is, systems which are SC-2 in nuclear power plants are Design Class II at WIPP which indicates both are in a higher quality class than SC-3/Design Class IIIA systems. As noted in Table IV-1 this does not imply that absolute quality requirements are equivalent.

V. CODES AND STANDARDS

As discussed in Section IV, the qualitative levels of system classification between nuclear power plants and WIPP are comparable. This section evaluates the absolute requirements applicable to each quality level so that a direct comparison of requirements can be made. This evaluation is made through a comparison of the codes and standards applicable to each quality level.

A. WIPP Codes and Standards

Table V-1 is a compilation of codes and standards applicable to WIPP as presented in Attachment I to Bechtel Document D-76-K-03. This information was obtained by extracting relevant portions of Table 2 of the Bechtel document.

B. Nuclear Power Plant Codes and Standards

Appendix B presents an excerpt from ANSI/ANS 51.1 which provides information comparable to Table V-1 for nuclear power plants. This information is presented in a somewhat complex form in the ANSI standard; therefore, Table V-2 was prepared using the information contained in Appendix B so that a more direct comparison between WIPP and nuclear power plant components may be made. Table V-2 is therefore directly comparable to Table V-1.

C. Evaluation

Five categories of design requirements (seismic/structural, piping, pressure vessels, tanks, and electrical) presented in Tables V-1 and V-2 are directly compared in Tables V-3 through V-7. Table V-8 provides a list of codes, standards, and acronyms used in these tables. From these tables three basic differences exist between the WIPP requirements and nuclear power

TABLE V-1

CODES AND STANDARDS APPLICABLE TO WIPP

	Design Class			
	I	II	IIIA	IIIB
Seismic Basis	DBE	DBE/UBC*	UBC*	UBC
Structural Requirements	ACI 318/AISC 310	ACI 318/AISC 310*	A58.1*	ANSI A58.1
Piping	B31.1	B31.1	B31.1	**
Pressure Vessels	ASME VIII	ASME VIII	**	**
Tanks	API 620/650	API 620/650	--	--
Electrical	IEEE Class 1-E	--	--	--
QA	NQA-1	NQA-1	NQA-1	--

* DBE/ACI 318/AISC requirements apply to Design Class II confinement structures. Requirements for other Design Classes II and IIIA structures are determined on a case-by-case basis. As a minimum, UBC and ANSI A58.1 requirements apply.

** Requirements are determined on a case-by-case basis.

Note: See Table V-8 for a listing of these standards and acronyms.

TABLE V-2

CODES AND STANDARDS APPLICABLE TO NUCLEAR POWER PLANTS

	Safety Class			
	1	2	3	NNS
Seismic Basis	Cat I	Cat I	Cat I	non-seismic
Structural Requirements	--	Concrete: ACI 349* Steel: AISC 326*	Concrete: ACI 349 Steel: AISC 326	Concrete: ACI 318 Steel: AISC 310
Piping	ASME III, Class 1	ASME III, Class 2	ASME III, Class 3	B31.1**
Pressure Vessels	ASME III, Class 1	ASME III, Class 2	ASME III, Class 3	ASME VIII**
Tanks	--	ASME III, Class 2	ASME III, Class 3	API 620/650**
Electrical	--	--	IEEE Class 1-E	--
QA	Appendix B/NQA-1	Appendix B/NQA-1	Appendix B/NQA-1	--

* Containment structures must comply with ASME III Class MC (steel) and Division 2 (concrete).

** Although not specified in ANSI/ANS 51.1, nuclear power plants typically use these requirements.

TABLE V-3

SEISMIC/STRUCTURAL REQUIREMENTS

Design Class	WIPP Codes and Standards	Nuclear Power Plant Codes and Standards
I	Design Basis Earthquake ACI 318/AISC 310	**
II (Confinement)	Design Basis Earthquake ACI 318/AISC 310	Safe Shutdown Earthquake ASME III Class MC or Div 2
(Other)	UBC/A58.1*	ACI 349/AISC 326
IIIA	UBC/A58.1*	ACI 349/AISC 326
IIIB	UBC/A58.1	ACI 318/AISC 310

* Other requirements determined on a case-by-case basis.

** No power plant structures are SC-1 because this classification applies only to the reactor coolant pressure boundary. The reactor coolant pressure boundary is designed to withstand the safe shutdown earthquake.

TABLE V-4

PIPING REQUIREMENTS

Design Class	WIPP Codes and Standards	Nuclear Codes and Standards
I	B31.1	ASME III, Class 1
II	B31.1	ASME III, Class 2
IIIA	B31.1	ASME III, Class 3
IIIB	*	B31.1**

* Determined on a case-by-case basis.

** Typical requirement.

TABLE V-5

PRESSURE VESSEL REQUIREMENTS

Design Class	WIPP Codes and Standards	Nuclear Power Plant Codes and Standards
I	ASME VIII	ASME III, Class 1
II	ASME VIII	ASME III, Class 2
IIIA	*	ASME III, Class 3
IIIB	*	ASME VIII**

* Determined on a case-by-case basis.

** Typical requirement.

TABLE V-6

TANK REQUIREMENTS

Design Class	WIPP Codes and Standards	Nuclear Power Plant Codes and Standards
I	API 620/650	*
II	API 620/650	ASME III, Class 2
IIIA	--	ASME III, Class 3
IIIB	--	API 620/650**

* Because tanks are by definition low pressure, no Safety Class I tanks can exist since this classification applies only to the reactor coolant pressure boundary.

** Typical requirement.

TABLE V-7

ELECTRICAL REQUIREMENTS

Design Class	WIPP Codes and Standards	Nuclear Power Plant Codes and Standards
I	IEEE 1-E	--
II	--	--
IIIA	--	IEEE 1-E
IIIB	--	--

Note: ANSI/ANS 51.1 lists a large number of IEEE standards which are applicable to Class 1-E. Bechtel document D-76-K-03 does not define Class 1-E requirements.

TABLE V-8

CODES, STANDARDS AND ACRONYMS APPLICABLE TO TABLES V-1 THROUGH V-7

ASME III	American Society of Mechanical Engineers, Boiler & Pressure Vessel Code, Section III, Nuclear Power Plant Components
ASME VIII	American Society of Mechanical Engineers, Boiler & Pressure Vessel Code, Section VIII, Unfired Pressure Vessels
API 620/650	American Petroleum Institute, Low Pressure Storage Tanks (620) and Oil Storage Atmospheric Tanks (650)
AISC 310	American Institute of Steel Construction, Specifications for Design, Fabrication, and Erection of Structural Steel for Buildings
AISC 326	American Institute of Steel Construction, Specifications for Design, Fabrication, and Erection of Structural Steel for Buildings
ACI 318	American Concrete Institute, Requirements for Reinforced Concrete
ACI 349	American Concrete Institute, Requirements for Nuclear Safety Related Concrete Structures
B31.1	American National Standards Institute, Power Piping Code
IEEE	Institute of Electrical and Electronics Engineers
NQA-1	American Society of Mechanical Engineers, Quality Assurance Program Requirements for Nuclear Power Plants
Appendix B	10CFR50, Appendix B
DBE	Design Basis Earthquake
UBC	Uniform Building Code
Cat I	Seismic Category I (must be designed to withstand the safe shutdown earthquake)

plant requirements. These differences are:

- Seismic classification

  Nuclear power plant Safety Classes 1, 2 and 3 all are considered Seismic Category I whereas for WIPP only Design Class I must meet the design basis earthquake requirements. For Design Classes II and IIIA seismic requirements are determined on a case-by-case basis; however, WIPP requirements also state that the design basis earthquake design requirement must be applied to structures and supports needed for confinement and control of radioactivity.

- Pressure retaining component requirements

  Nuclear power plant pressure retaining components are designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, which is entitled "Nuclear Power Plant Components." Table 2 of the Bechtel document references ASME Section VIII for pressure vessels, ANSI B31.1 for piping, API 610 for pumps, and API 620/650 for tanks. All of these components in nuclear power plants are within the scope of ASME Section III. The codes and standards listed for WIPP are those typically used in non-nuclear power plant design in the power industry and were the codes used on older nuclear power plants. For example, many currently operating nuclear power plants were designed in accordance with ANSI B31.1. However, it would seem to be appropriate to use ASME Section III in these areas since Table 2 of the Bechtel document requires that these components meet the intent of ANSI/ASME NQA-1 which is required for nuclear power plant components within the scope of ASME Section III. DOE should be asked to explain the reasons for this difference. It is likely that the decision to use codes other than ASME Section III was based on cost considerations and engineering judgment that no substantial differences would result if Section III had been used. The application of NQA-1 provides a basis for concluding that the quality of these components is acceptable.

- There is a difference between nuclear power plant classifications and WIPP regarding the application of IEEE Class 1-E requirements. In nuclear power plants, Safety Class 3 components are required to meet IEEE class 1-E requirements. Class 1-E requirements are not applicable to Safety Classes 1 and 2. Safety Class 1 is defined in terms of the reactor coolant pressure boundary and electrical components never form the reactor coolant pressure boundary. Therefore, in nuclear power plants there can be no Safety Class I electrical requirements. Safety Class 2 applies to those components which mitigate the consequences of an accident. Typically, these are referred to as engineered safety features and include systems such as the emergency core cooling system, containment spray systems, and the containment itself. Since electrical and instrumentation systems do not directly provide these functions they are not classified as Safety Class 2. Electrical components provide support to the Safety Class 2 components and are therefore classified as Safety Class 3. The high reliability associated with these systems is further justification for this classification. In the electrical area, for example, there are very specific requirements contained in the regulations and in IEEE standards regarding the application of the single failure criterion and availability of on-site and off-site power. For example, electric power systems for nuclear power plants are required to be designed such that safety-related components are powered from both on-site and off-site sources. Furthermore, at least two separate sources of off-site power must be provided to nuclear power plants. The off-site power must also be backed up by an on-site power system. The on-site power system, in turn, must be designed to meet the single failure criterion; therefore, power to achieve a safety function in a nuclear power plant can only be lost if two sources of off-site power and two sources of on-site power are lost. This degree of redundancy and the ability to monitor and test the availability of back-up systems allows the use of a Safety Class 3 designation.

The above three areas represent the fundamental differences between the codes and standards applicable to nuclear power plants and those applicable to WIPP. We believe that the codes and standards provided in Table 2 of Bechtel document D-76-K-03 are generally appropriate. The combination of ASME Section VIII with NQA-I and ANSI B31.I with NQA-I, for example, provide a reasonable alternative to ASME Section III. In the seismic design classification area, the exemption to Design Classes II and IIIA provided in Table 2 can be justified if an appropriate combination of seismic events and accident events is made. ANSI/ANS 51.I provides a method for combining independent events to determine the appropriate dose criteria for evaluating the event. Events which are low probability need not be combined with low probability earthquake events; therefore, the exemptions in Table 2 may be acceptable.

With regard to the IEEE 1-E requirement stated in Table 2 of the Bechtel document, it appears that a discrepancy exists. Since there are no Design Class I items at WIPP, one would not expect to find any Class 1-E items because Class 1-E only applies to Design Class I. Systems that provide containment functions and require electric power should be powered from Class 1-E power supplies, unless it can be shown that a loss of power for a reasonable length of time would not result in off-site dose consequences in excess of the 25 rem whole body/75 rem other organ criteria used for Design Class I definition. If engineered safety features (containment, confinement, filtering) are required to ensure that doses do not exceed this limit, then the power supplies should be Class 1-E. Since the emergency diesel system is IIIA, it would appear that Table 2 of document D-76-K-03 should indicate the applicability of Class 1-E to Design Class IIIA and should indicate that Class 1-E is not applicable to Design Classes I and II. The only exception which can be envisioned is if a failure modes and effects analysis indicated that the loss of AC power would directly result in the release of radioactive material such that the dose limits would be violated. In such a case the electric power requirements for Class 1-E should be applied to Design Class 2 components.

VI. CONCLUSIONS AND RECOMMENDATIONS

Our review reaches the following conclusions regarding the classification of systems, structures and components for WIPP.

1. The basic methodology described in the Bechtel document is adequate and appropriate for its intended use.

2. With due consideration given to the differences inherent in the types of facilities involved, the classification of systems, structures and components at WIPP is comparable to that at nuclear power plants.

3. The 25-rem whole body dose and 75-rem organ dose are appropriate upper limits for accident evaluations and for determining site suitability. This conclusion is predicated upon a low probability being demonstrated for accident scenarios which could result in doses comparable to these limits.

4. The 0.5-rem dose referred to in Part 60 is an appropriate criterion for determining whether components must comply with the quality assurance program. This would be a useful addition to the information contained in the Bechtel document and WIPP SAR. This change would update the WIPP SAR definition of "important to safety" to agree with the current version of 10CFR60.

5. Although the codes and standards specified by Bechtel in document D-76-K-03 are reasonable when combined with a requirement to meet the intent of NQA-1, it would be appropriate to request DOE to explain the basis for selecting those standards rather than ASME Section III.

6. Further review of the accident scenarios and in particular the assumptions concerning release fractions should be made. A detailed review of events involving fires would be appropriate.

7. Further review should be made of the failure modes and effects analysis; in particular, attention should be applied to assumptions regarding operator action, single failure, and loss of off-site AC power. This could be performed using the ventilation and electric power systems as sample systems to be evaluated. While not entirely applicable, the assumptions used in evaluating nuclear power plant responses to the failures can be used as a point of departure for evaluating applicable failure modes.

8. WIPP SAR should be updated to be consistent with Bechtel document D-76-K-03.

9. The Environmental Evaluation Group should review the implementation of the WIPP quality assurance program. This review should include reviews of QA audits performed in accordance with the program.

10. Operational requirements can be more important than the quality assurance program applied during design and construction. Therefore, EEG should review the operations safety requirements (SAR Chapter 10) in detail.

APPENDIX A

REFERENCES AND SOURCES OF INFORMATION

APPENDIX A

REFERENCES AND SOURCES OF INFORMATION

1. Waste Isolation Pilot Plant Safety Analysis Report as amended.

2. Bechtel Document D-76-K-03, Rev. 9, November 30, 1983.

3. Code of Federal Regulations, Title 10.

4. American Nuclear Society Standard ANSI/ANS 51.1, Nuclear Safety Criteria for Stationary Pressurized Water Reactor Plants.

5. USNRC report NUREG 0804, Staff Analysis of Public Comments on Proposed Rule 10CFR Part 60, December 1983.

6. Oak Ridge National Laboratory report ORNL-TM-4219, Site Selection Factors for the Bedded Salt Pilot Plant.

7. Office of Nuclear Waste Isolation report ONWI-496, Exploratory Shaft Facility Preliminary Designs: Paradox Basin.

8. Office of Nuclear Waste Isolation report ONWI-390, Exploratory Shaft Conceptual Design Report: Paradox Basin.

9. DOE Order AL 5481.1A, Safety Analysis and Review System for AL Operations, September 15, 1982.

10. DOE Order 6430.1, General Design Criteria Manual, December 12, 1983.

11. U.S. Atomic Energy Commission Report TID-14844, Calculation of Distance Factors for Power and Test Reactor Sites.

12. Code of Federal Regulations, Title 40, Parts 190 and 191 (Draft 4, April 23, 1984).

APPENDIX B

EXTRACTED TABLES FROM ANSI/ANS 51.1

TABLE IV-2

PART 60 DESIGN CRITERIA
FOR SYSTEMS, STRUCTURES AND COMPONENTS
IMPORTANT TO SAFETY

Part 60 Reference	Subject	Part 50, App A Reference	Comment
60.131(b)(1)	Protection against natural phenomena and environmental conditions	Criterion 2 Criterion 4	Part 60 requires protection against "anticipated" natural phenomena whereas Appendix A requires "most severe" historically reported conditions plus a margin.
60.131(b)(2)	Protection against dynamic effects of equipment failure	Criterion 4	Requirements are comparable
60.131(b)(3)	Protection against fires and explosions	Criterion 3	Requirements are comparable
60.131(b)(4)	Emergency capability	Criterion 16 Criterion 19 Criterion 34 Criterion 35 Criterion 38 Criterion 41 Criterion 50 Criterion 64 10CFR50, App E	Part 50 is much more detailed in its requirements than is Part 60.
60.131(b)(5)	Utility Services	Criterion 17 Criterion 44	Appendix A is much more detailed and requires independence as well as redundancy for the onsite electric system. Two circuits to the offsite transmission network are required. Single Failure assumption is required by Appendix A.
60.131(b)(6)	Inspection, testing and maintenance	Criterion 18 Criterion 21 Criterion 36 Criterion 37 Criterion 39 Criterion 40 Criterion 42 Criterion 43 Criterion 45 Criterion 46 Criterion 52 Criterion 53	Inspection and testing are addressed at the system level in Appendix A, but otherwise the requirements are comparable.
60.131(b)(7)	Criticality control	Criterion 62	Part 60 is more detailed and prescriptive.
60.131(b)(8)	Instrumentation and control	Criterion 13	Requirements are comparable.
60.131(b)(9)	Compliance with mining regulations	Not applicable	---
60.131(b)(10)	Shaft conveyances	Criterion 61 NUREG-0612	Although not directly comparable, generally comparable requirements exist.

TABLE IV-3

EXAMPLES OF WIPP AND NUCLEAR POWER PLANT
CLASSIFICATION OF SYSTEMS, STRUCTURES, AND COMPONENTS

General Description	WIPP		Nuclear Power Plant	
	Examples	Design Class	Examples	Safety Class
Structures	Waste Handling Building	II	Containment	2
	Cask Loading Room	II		
	Exhaust Filter Building	II		
	Hot Cell	II		
	Support Building	IIIA	Auxiliary Building	3
	Water Pump House	IIIA	Diesel Generator Building	3
			Control Building	3
			Ultimate Heat Sink	3
	Warehouse/Shop	IIIB	Radwaste Building	NNS
			Turbine Building	NNS
Isolation Features	Shield Plugs	II	Piping and Valves	2
	Shield Valves	II	Forming Containment Isolation System	
	Tornado Doors and Dampers	II		
Electric Power System	Emergency Diesel Generator and Associated Equipment	IIIA	Emergency Diesel Generator and Associated Equipment	3
	Uninterruptable Power Supply	IIIA	Emergency 125/250 V DC System	3
	13.8 kV System	IIIB	Offsite Power System	NNS
Air Cleaning HVAC	Hot Cell HEPAs/Dampers	II	Standby Gas Treatment System (BWRs)	2
	RM Waste Handling Area HEPA/Dampers	II		
	U/G Exhaust System HEPA/Dampers	II	Containment Spray System (PWRs)	2
	U/G Exhaust Fans	II		
	CMR HVAC System	IIIA	Control Room HVAC	3
Fire Protection	Electric and Diesel Fire Water Pumps	IIIA		
	Diesel Fuel Day Tank for Fire Water Pump	IIIB		
	Other Fire Protection Systems	IIIB	Fire Protection System	NNS
Radiation Monitoring	RM/CM Area Rad Monitor	IIIA	Inputs to Reactor Protection System	3
	RM/CM Continuous Rad Monitors	IIIA	Accident Monitoring	3
	Air Sampling System	IIIA	Effluent Monitoring	NNS
			Area Monitors	NNS

Note: NNS = Non-nuclear safety.

APPENDIX C

FIGURE I FROM BECHTEL
DOCUMENT D-76-K-03 (REV. 9)