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SPENT NUCLEAR FUEL PROJECT PATH FORWARD

NUCLEAR SAFETY EQUIVALENCY TO COMPARABLE NRC-LICENSED FACILITIES

November 1995

Prepared for the U.S. Department of Energy
Office of Environmental Restoration and Waste Management

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Spent Nuclear Fuel Project Path Forward

Nuclear Safety Equivalency
to
Comparable NRC-Licensed Facilities

CONTENTS

1.0	PURPOSE AND SCOPE	1
2.0	CONCLUSIONS	3
3.0	TITLE 10 CODE OF FEDERAL REGULATIONS NON-APPLICABLE PARTS/ SECTIONS/APPENDICES	5
4.0	TITLE 10 CODE OF FEDERAL REGULATIONS ACCOMMODATED BY WHC SITE-WIDE PROGRAMS	7
5.0	DISCUSSION ON ADMINISTRATIVE PROCEEDINGS	9
6.0	TITLE 10 CODE OF FEDERAL REGULATIONS TECHNICAL/SAFETY COMPARISONS . .	11
7.0	TECHNICAL/SAFETY COMPARISONS FOR POTENTIALLY APPLICABLE NRC GUIDANCE	13
8.0	GLOSSARY	113
8.1	Acronyms	113
8.2	Definitions	115
9.0	REFERENCES	121
Attachment A	Detailed Evaluations	A-1
	Quality Assurance Criteria	A-3
	10CFR50 Appendix A, General Design Criteria	A-9
	Environmental Qualification of Electrical Equipment	A-15

Attachment B Comparison of Nuclear Regulatory Commission and
Department of Energy Requirements for Independent
Spent Nuclear Fuel Storage Facilities (Sciencetech, Inc)
(enclosed separately)

LIST OF TABLES

1	Cross-Reference to Applicable Information	14
2	Actions for Consideration	16
3	Title 10 CFRs Non-Applicable Parts/Sections/Appendices	21
4	Title 10 CFRs Accommodated by WHC Site-wide Programs	28
5.a	Title 10 CFR Summary-Level Comparisons of Technical/Safety Issues	34
5.b	Title 10 CFR Summary-Level Comparisons of Part 50 Technical/Safety Issues	44
5.c	Title 10 CFR Summary-Level Comparisons of Part 72 Technical/Safety Issues	67
6.a	Technical/Safety Comparisons for Potentially Applicable NRC Regulatory Guides	91
6.b	Technical/Safety Comparisons for Other Potentially Applicable NRC Documentation	100
A-1	Comparison of the Quality Assurance Requirements of 10CFR72 and 10CFR 830.120	A-4
A-2	Potential Application of the General Design Criteria of 10CFR50, Appendix A to New SNFP Path Forward Facilities other than the CSB	A-10

Spent Nuclear Fuel Project Path Forward

Nuclear Safety Equivalency to Comparable NRC-Licensed Facilities

1.0 PURPOSE AND SCOPE

The U.S. Department of Energy (DOE), established in the K Basin Spent Nuclear Fuel Project - Regulatory Policy, dated August 4, 1995 (hereafter referred to as the Policy), the requirement for new Spent Nuclear Fuel Project (SNFP) facilities to achieve "nuclear safety equivalency" to comparable U.S. Nuclear Regulatory Commission (NRC)-licensed facilities (DOE 1995a). For that Policy, nuclear safety equivalency was defined as:

- Technical requirements which meet the nuclear safety objectives of the NRC regulations for fuel treatment and storage facilities. These include requirements regarding radiation exposure limits, safety analysis, design and construction.
- Administrative requirements which meet the objectives of the major elements of the NRC licensing process. These include formally documented design and safety analyses, independent technical review, and opportunity for public involvement.

Technical requirements, in the context of the Policy and as interpreted by the Regulatory Requirements Team, are the design and construction measures (as opposed to preoperational or operational measures) that are mandated by the NRC regulations. In addition, the Policy specifically excludes those requirements that only address environmental, Occupational Safety and Health Administration (OSHA), chemical accident safety, and other non-nuclear safety issues.

Further, NRC guidance that provides acceptable means to implement the regulations, was defined as optional rather than mandatory in the Policy. However, NRC guidance that may have a direct relevance to SNFP Path Forward activities (e.g., NRC Regulatory Guide 3.60, *Design of an Independent Spent Fuel Storage Installation [Dry Storage]*) was reviewed and evaluated for incorporation of additional requirements believed warranted to achieve nuclear safety equivalency as a prudent step in implementing the Policy's objectives.

For new SNFP Path Forward facilities, which includes the Canister Storage Building [CSB], Conditioning Facility, and K Basins Path Forward Projects, NRC regulations throughout Title 10 and a variety of NRC guidance were deemed appropriate for consideration in evaluating nuclear safety

equivalency. For the purposes of this report, K Basins Path Forward Projects includes: desludging and cleaning of the K Basins fuel, reracking, and cold vacuum drying. The processes (e.g., cold vacuum drying and hot vacuum conditioning) and associated safety functions of the Conditioning Facility and K Basins Path Forward activities are not directly reflected within the Title 10 CFRs or specific NRC guidance. These processes may also result in Safety Class 1 accident prevention and/or mitigating structures, systems, and components (SSCs). The body of NRC regulations and guidance that may apply in view of potential analogous conditions and similar hazards and safety functions was therefore evaluate for application to the Conditioning Facility and the K Basins Path Forward Projects.

Similarly, the CSB will present a substantially different situation than envisioned by the NRC for a Part 72 facility. Approximately 49 percent of the K Basins fuel has been characterized as being in less than good condition, with about 6 percent defined as being severely degraded. In addition, there are considerations for accelerated corrosion and associated hydrogen generation and the potential for chemical reaction with the metallic fuel. Moreover, Safety Class 1 accident prevention and/or mitigating SSCs cannot be ruled out at this time, and maintaining Multi-Canister Overpack (MCO) cooling and integrity is somewhat analogous to preventing radioactive release for some commercial nuclear power plant conditions. It was therefore believed appropriate to review for the CSB NRC regulations beyond Part 72 and a broader base of NRC guidance.

The purposes of this report are: 1) to document the review and evaluation of the Title 10 CFRs, Parts 0-199, revised as of January 1, 1995, and relevant NRC guidance, located in NRC public document rooms as of September 18, 1995, against the existing DOE requirements for new SNFP facilities, and 2) as a result of this review and evaluation and the conclusions drawn, identify any additional actions for consideration that may be necessary to demonstrate nuclear safety equivalence. It should be noted that in some cases specific Westinghouse Hanford Company (WHC) requirements were used to establish nuclear safety equivalence.

The information provided is presented in four principal subgroups: 1) parts, sections and appendices determined to be not applicable; 2) NRC regulations accommodated by WHC site-wide programs; 3) NRC regulations potentially applicable to new SNFP facilities and K Basins Path Forward Projects; and 4) NRC guidance potentially applicable to new SNFP facilities and K Basins Path Forward Projects.

Table 1 provides a cross-reference between the individual parts, sections and appendices, as appropriate, of Title 10 of the CFRs and NRC guidance, and where they are addressed within the tables of this report.

2.0 CONCLUSIONS

In the comparison of DOE and NRC requirements relevant to the design and construction of new SNFP facilities and K Basins Path Forward Projects, a number of differences were identified. Where differences exist, the NRC requirements for the most part were either more conservative or require the preparation of additional documentation (e.g., as part of a safety analysis report).

In regards to establishing nuclear safety equivalency, Actions for Consideration were identified for meaningful differences where existing SNFP Path Forward requirements need to be enhanced. For ease of identification, all differences, where Actions for Consideration were identified, were consolidated into Table 2. Amplifying information on each of these items is presented in Tables 4 through 6.b, which contain the results of the evaluation. The more important of these items, i.e., may have substantive project impacts, are identified below with their corresponding regulation or guidance number and a brief caption of the item. This list was compiled based on the current assumption that no Safety Class 1 equipment will be required for the CSB, Conditioning Facility, or K Basins Path Forward Projects.

- Section 72.3, *Definitions* (definition of structures, systems and components [SSCs] important to safety embraces additional Safety Class 1 SSCs).
- Section 72.24, *Contents of application: Technical information* (additional safety analysis report [SAR] information).
- Section 72.92, *Design basis external natural events*, and §72.102, *Geologic and seismological characteristics* (more conservative seismic design criteria).
- Section 72.122, *Overall requirements* (inclusion of tornado and tornado missiles).
- Regulatory Guide 3.60, *Design of an Independent Spent Fuel Storage Installation [Dry Storage]* (additional design requirements via ANSI/ANS-57.9).
- SRP 9.1.2, *Spent Fuel Storage* (use of a k_{eff} criticality safety limit of 0.95).

In addition, it should be reemphasized that the Actions for Consideration presented in Table 2 were developed in consideration that: 1) the design of the CSB at this time is preliminary and changes in the future to include Safety Class 1 accident prevention and/or mitigating SSCs could not be ruled out, and 2) the design of the Conditioning Facility and K Basins Path Forward Projects remain to be performed and may well include Safety Class 1 accident prevention and/or mitigating SSCs. For these reasons, Actions for Consideration, such as those associated with NRC Regulatory Guide 1.128 (consider incorporating the requirements of IEEE Std 484-1987, *IEEE Recommended Practices for Installation Design and Installation of Large Lead Storage Batteries for Generation Stations and Substations*, into the design and installation of Safety Class 1 batteries) and Regulatory Guide 1.153 (consider incorporating into the design for Safety Class 1 instrumentation and control systems the requirements of IEEE Std 603-1991, *IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations*) were included. It should be noted that both of these Actions for Consideration were qualified to be only applicable to Safety Class 1 equipment, and until such equipment is identified, there is no impact on design.

It should also be remembered that equivalency was established for the most part by DOE orders and in some instances by WHC procedures and instructions. If the SNFP takes steps to acquire waivers for some of the DOE orders used to establish nuclear safety equivalency, or if relevant DOE orders or WHC procedures and instructions are revised or canceled, a number of the conclusions regarding nuclear safety equivalency in this report may be nullified.

3.0 TITLE 10 CODE OF FEDERAL REGULATIONS NON-APPLICABLE PARTS/SECTIONS/APPENDICES

There are 53 existing parts to Title 10 of the CFRs. Based on review and evaluation of the individual parts, 33 were determined without detailed review to not be relevant to design or construction of new SNFP facilities or K Basins Path Forward Projects in establishing nuclear safety equivalency.

A list of these non-applicable parts is provided in Table 3, with a brief explanation of the purpose and/or scope of each. The explanations are provided to allow the reader to understand the basis for each item being determined non-applicable. It should also be noted that after a detailed review of Parts 50 and 72, some sections were determined to be not applicable to nuclear safety equivalency as applied to design and construction of new SNFP facilities or K Basins Path Forward Projects. However, to make this determination, a detailed review was required. The discussions of these sections were left in Tables 5.b and 5.c.

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4.0 TITLE 10 CODE OF FEDERAL REGULATIONS ACCOMMODATED BY WHC SITE-WIDE PROGRAMS

There are several programmatic areas, also addressed in Title 10 of the CFRs, that by their nature are applied by WHC on a site-wide basis and have no unique features for SNFP facilities or activities. In addition, for the most part these areas have limited or no direct bearing on the design or construction of new SNFP facilities or K Basins Path Forward Projects. Examples of these areas are: Part 26, *Fitness-for-Duty Program*; Section 50.7, *Employee Protection*; and Part 160, *Trespassing on Commission Property*.

Each of the parts and sections addressed in Table 4 include a brief discussion of the NRC safety issue(s) and the applicable DOE requirements implemented by WHC. Conclusions and any Actions for Consideration are also provided regarding the adequacy of the site-wide program.

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5.0 DISCUSSION ON ADMINISTRATIVE PROCEEDINGS

As stated in the Policy and in Section 1.0, Purpose and Scope, of this report, nuclear safety equivalency includes conformance with administrative requirements which meet the objectives of the major elements of the NRC licensing process. These major elements were further defined in the Policy to be formally documented design and safety analyses, an independent technical review of the design and safety analyses, and providing opportunity for public involvement.

The design for new SNFP facilities and K Basins Path Forward Projects will be documented, reviewed and approved in accordance with established quality assurance and design control procedures that reflect the requirements of 10CFR830.120, *Quality Assurance Requirements*, and 10CFR72, *Licensing Requirements For the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste*, Subpart G, *Quality Assurance*. Safety analyses for these facilities and projects will be documented, reviewed and approved in accordance with similarly established quality assurance procedures and the requirements of the WHC *Safety Analysis Manual*, WHC-CM-4-46, Section 2.0, *Safety Analysis Process*, Revision 3. Independent technical reviews of the design will be performed by WHC SNFP staff among others (i.e., the original design is being performed by WHC subcontractors). The regulatory requirements and safety analysis report (SAR) for new SNFP facilities (and their updates for K Basins Path Forward Projects), as discussed in the Policy, will receive multiple independent technical reviews. Further, the design and safety issues for new SNFP facilities and K Basins Path Forward Projects will be discussed in ongoing public forums with the various stakeholders, with opportunity for public comment.

It should be noted, however, that a number of the administrative proceedings in Part 2 of Title 10 of the CFRs (*Rules of Practice for Domestic Licensing Proceedings and Issuance of Orders*) will not be implemented as part of the SNFP public involvement process. These proceedings include formal public notification and hearings, the opportunity for written petition for leave to intervene, and formal participation in licensing proceedings if intervention is allowed.

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6.0 TITLE 10 CODE OF FEDERAL REGULATIONS TECHNICAL/SAFETY COMPARISONS

This section addresses the Title 10 CFR technical/safety issues that may be applicable to the SNFP Path Forward and are not accommodated by programmatic site-wide programs. The comparison information is presented in three tables: 1) relevant parts other than Parts 50 and 72 (Table 5.a); 2) all Part 50 sections and appendices (Table 5.b); and 3) all Part 72 sections (Table 5.c). The information for Parts 50 and 72 is provided separately due to their size and complexity and specific interest in Part 72 due to its direct applicability to the CSB.

Table 5.a is formatted to include: the Title 10 Part number and title; a discussion of the existing SNFP Path Forward requirements, including Conclusions regarding their adequacy for nuclear safety equivalency, and any Actions for Consideration identified to demonstrate nuclear safety equivalency where existing SNFP Path Forward requirements need to be enhanced; and a discussion of the associated NRC safety issue(s).

In addition, based on recommendations on September 19 and 20, 1995, from the Independent Review Panel (IRP), subsequently directed by DOE, a review was performed of the Part 70 Siemens Power Corporation application for license renewal, titled *Application for Renewal of Special Material License No. SNM-1227 (NRC Docket No. 70-1257)*. This review also included verbal discussions with Siemens' staff on the particular areas of interest pursued by the NRC. The intent of this review, in that Part 70 lacks definitive requirements, was to identify what are called "revealed regulations" (i.e., NRC requirements imposed during the licensing process that are not called out in the regulations). The IRP believed that such a review may identify additional NRC requirements needed to establish nuclear safety equivalency for the Conditioning Facility and K Basins Path Forward Projects. The subject renewal document emphasized criticality safety and heating, ventilating and air-conditioning systems (HVAC), and was in the process of expanding its discussions on safety analyses. The areas of interest expressed by the NRC included: chemical safety as it may impact nuclear safety, quality assurance, criticality safety, safety analyses, fire safety issues and fire rating standards (National Fire Protection Association [NFPA]). Based on review of the renewal document and discussions with Siemens' staff, no Actions for Consideration were identified for Part 70. Additional details of this review are included in Table 5.a under Part 70.

Table 5.b is formatted to provide the same information as Table 5.a, except that the Part 50 section or appendix, and associated title, are provided. Part 50 to a great extent was developed to address the regulatory needs of commercial nuclear power plants, which under postulated conditions pose significantly greater risk than a spent nuclear fuel storage or treatment

facility. This is due to a nuclear reactor's available energy and significant quantity of nuclear fuel in a critical condition during operations. Recognizing, however, that the CSB, Conditioning Facility, and K Basins Path Forward Projects are not representative of a typical Part 72 facility (i.e., passive with non-degraded commercial spent fuel), Part 50 was reviewed to identify nuclear safety concerns or functions, including analogous conditions, that may have relevance to the new SNFP activities. One example is control of hydrogen. Section 50.44 applies only to light water reactors. However, that section was used as a convenient location to discuss hydrogen concerns as they apply to the CSB, Conditioning Facility, and K Basins Path Forward Projects. Moreover, several Part 50 nuclear safety issues were raised as part of the evaluation in consideration that future design changes to the CSB, Conditioning Facility, or K Basins Path Forward Projects could incorporate Safety Class 1 accident prevention and/or mitigation SSCs.

Table 5.c is based substantially on a report provided by Sciencetech, Inc. (Sciencetech), under DOE contract DE-AC01-92EW50504, titled *Comparison of Nuclear Regulatory Commission and Department of Energy Requirements for Independent Spent Nuclear Fuel Storage Facilities*. The conclusions drawn by Sciencetech that were related to design or construction measures, for the Subparts of Part 72 that they reviewed, were extracted and placed in Table 5.c in the Conclusions column. The basis for the Sciencetech conclusions remain in their report, mentioned above and enclosed as Attachment B. Where the Sciencetech report was judged to adequately address the NRC safety issue(s) within a Part 72 section, only the Sciencetech conclusions provided input to the Actions for Consideration, also in the Conclusions column in Table 5.c. Where additional information, beyond the Sciencetech report, was believed to be relevant to the NRC safety issue(s), e.g., Westinghouse Hanford Company specific programs/practices, such information was also placed in Table 5.c under the Additional Information column and the Actions for Consideration were supplemented and/or modified accordingly. As for Tables 5.a and 5.b, any identified Actions for Consideration are associated with demonstrating nuclear safety equivalency.

In addition to the information provided in the above mentioned tables, the evaluation of select Title 10 sections and appendices were presented in detail in Attachment A due to their complexity and size of presentation. They are: Part 50, Appendix A, *General Design Criteria for Nuclear Power Plants*; Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants*; Section 50.49, *Environmental qualification of electrical equipment*; and Part 72, Subpart G, *Quality Assurance*. Part 50, Appendix B, and Part 72, Subpart G, were combined into a discussion titled *Quality Assurance Criteria*.

7.0 TECHNICAL/SAFETY COMPARISONS FOR POTENTIALLY APPLICABLE NRC GUIDANCE

As stated in Section 1.0, Purpose and Scope, of this report, NRC guidance that may have direct relevance to the SNFP Path Forward (the CSB, Conditioning Facility, and K Basins Path Forward Projects) was reviewed to identify any additional requirements believed warranted to achieve nuclear safety equivalency. As such, NRC Regulatory Guides (Divisions 1 through 10), NUREG, NUREG/CR and SECY documents, Standard Review Plans (SRPs), Branch Technical Positions (BTPs), Inspection and Enforcement (I&E) Notices and Bulletins, Inspection Manual Procedures, and Generic Letters were reviewed for this purpose. The results of this review are presented in Tables 6.a (Regulatory Guides) and 6.b (Other Potentially Applicable NRC Documentation). These tables provide the number of the NRC guidance document, the document's title, and applicable comments, including Conclusions and any Actions for Consideration identified to demonstrate nuclear safety equivalency.

The Sciencetech report included as part of Section 72.24, *Contents of application: Technical information*, a review of Regulatory Guide 3.48, *Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation (Dry Storage)*. The results of this review are provided in Table 5.c, which addresses Part 72, rather than in Table 6.a.

Table 1 Cross-Reference to Applicable Information

Part/Document	Title/Discussion	Table
0	Conduct of Employees	3
1	Statement of Organization and General Information	3
2	Rules of Practice for Domestic Licensing Proceedings and Issuance of Orders	Refer to Section 5.0
4	Nondiscrimination in Federally Assisted Commission Programs	3
7	Advisory Committees	3
8	Interpretations	3
9	Public Records	3
10	Criteria and Procedures for Determining Eligibility for Access to Restricted Data or National Security Information or an Employment Clearance	3
11	Criteria and Procedures for Determining Eligibility for Access to or Control Over Special Nuclear Material	5 a
12	Implementation of the Equal Access to Justice Act in Agency Proceedings	3
13	Program Fraud Civil Remedies	3
14	Administrative Claims under Federal Tort Claims Act	3
15	Debt Collection Procedures	3
16	Salary Offset Procedures for Collecting Debts Owed by Federal Employees to the Federal Government	3
19	Notices, Instructions, and Reports to Workers: Inspections	4
20	Standards for Protection Against Radiation	5 a
21	Reporting of Defects and Noncompliance	4
25	Access Authorization for Licensee Personnel	3
26	Fitness for Duty Programs	4
30	Rules of General Application to Domestic Licensing of Byproduct Material	5 a
31	General Domestic Licenses for Byproduct Material	3
32	Specific Domestic Licenses to Manufacture or Transfer Certain Items Containing Byproduct Material	3
33	Specific Domestic Licenses of Broad Scope for Byproduct Material	3
34	Licenses for Radiography and Radiation Safety Requirements for Radiographic Operations	3
35	Medical Use of Byproduct Material	3
36	Licenses and Radiation Safety Requirements for Irradiators	5 a
39	Licenses and Radiation Safety Requirements for Well Logging	3
40	Domestic Licensing of Source Material	5 a
50	Domestic Licensing of Production and Utilization Facilities All sections of 10CFR50 are addressed in Table 5 b with the exception of the following	5 b
§50.5	Deliberate misconduct	4
§50.7	Employee protection	4
§50.9	Completeness and accuracy of information	4

Table 1 Cross-Reference to Applicable Information

Part/Document	Title/Discussion	Table
\$50.12	<i>Specific exemptions</i>	4
\$50.72	<i>Immediate notification requirements for operating nuclear power reactors</i>	4
\$50.73	<i>Licensee event report (LER) system</i>	4
51	<i>Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions</i>	5 a
52	<i>Early Site Permits, Standard Design Certifications, and Combined Licenses for Nuclear Power Plants</i>	3
53	<i>Criteria and Procedures for Determining the Adequacy of Available Spent Nuclear Fuel Storage Capacity</i>	3
54	<i>Requirements for Renewal of Operating Licenses for Nuclear Power Plants</i>	3
55	<i>Operators' Licenses</i>	3
60	<i>Disposal of High-Level Radioactive Wastes in Geologic Repositories</i>	5 a
61	<i>Licensing Requirements for Land Disposal of Radioactive Waste</i>	3
62	<i>Criteria and Procedures for Emergency Access to Non-Federal and Regional Low-Level Disposal Facilities</i>	3
70	<i>Domestic Licensing of Special Nuclear Material</i>	5 a
71	<i>Packaging and Transportation of Radioactive Material</i>	5 a
72	<i>Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste</i>	5 c
73	<i>Physical Protection of Plants and Materials</i>	5 a
74	<i>Material Control and Accounting of Special Nuclear Material</i>	5 a
75	<i>Safeguards on Nuclear Material-Implementation of US/IAEA agreement</i>	5 a
76	<i>Certification of Gaseous Diffusion Plants</i>	3
81	<i>Standard Specifications for the Granting of Patent Licenses</i>	3
95	<i>Security Facility Approval and Safeguarding of National Security Information and Restricted Data</i>	3
100	<i>Reactor Siting Criteria</i>	5 a
110	<i>Export and Import of Nuclear Equipment and Material</i>	3
140	<i>Financial Protection Requirements and Indemnity Agreements</i>	3
150	<i>Exemptions and Continued Regulatory Authority in Agreement States and in Offshore Waters under Section 274</i>	3
160	<i>Trespassing on Commission Property</i>	4
170	<i>Fees for Facilities, Materials, Import and Export Licenses, and other Regulatory Services under the Atomic Energy Act of 1954, as Amended</i>	3
171	<i>Annual Fees for Reactor Operating Licenses, and Fuel Cycle Licenses and material Licenses, including Holders of Certificates of Compliance, Registrations, and Quality Assurance Program Approvals and Government Agencies Licensed by NRC</i>	3
172 through 199	<i>Reserved as of January 1, 1995</i>	3
Regulatory Guides	<i>Selected NRC Regulatory Guides that may be appropriate for SNFP Path Forward activities are addressed</i>	6 a
Other NRC guidance documents	<i>Standard Review Plans, Branch Technical Positions, NUREG and NUREG/CR reports, IE Notices and Bulletins, Generic Letters and SECY papers are addressed</i>	6 b

Table 2 Actions for Consideration

Regulations/Guidance	Actions for Consideration
Part 20. Standards for protection against radiation	The new SNFP Path Forward facilities should consider incorporating control devices for access to High Radiation Areas that conform to the requirements of §20.1601. Consideration should also be given to factoring into the design of new SNFP Path Forward facilities the §20.1301 hourly limit of 0.002 rem to the public. This can be accommodated by the location of their restricted area boundaries.
Part 21. Reporting of defects and noncompliance	Procurement clause E13 of WHC-CH-4-2, <i>Quality Assurance Manual</i> , Appendix A, Revision 1, should be considered for addition to Safety Class 1 procurement specifications to require the reporting by suppliers of defects in items or services.
Section 50.34. Contents of an application; technical information, and NRC Regulatory Guide 3.26, Revision 0. Standard Format and Content of Safety Analysis Reports for Fuel Reprocessing Plants	Consider including in the SNFP Path Forward Safety Management Plan the requirement to augment the SAR or other safety documentation for new SNFP Path Forward facilities other than the CSB with the information required by Regulatory Guide 3.26, Revision 0 that is unique to the conditioning processes.
Section 50.34. Contents of an application; technical information, and NRC Regulatory Guide 3.48, Revision 1, Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation (Dry Storage)	Consider including in the SNFP Path Forward Safety Management Plan the requirement to augment the CSB SAR with the information called for in §72.24 and Regulatory Guide 3.48, Revision 1, that is unique to storage of spent nuclear fuel.
Section 50.36a. Technical specifications on effluents from power reactors	Ensure that the AEs for new SNFP Path Forward facilities review the effluent monitoring requirements of, for example 10CFR20.100FR70.59, and 10CFR835, to provide the necessary monitoring instrumentation.
Section 50.48. Fire protection, and Appendix R, Fire Protection for Nuclear Power Facilities Operating Prior to January 1, 1979	Consideration should be given to following the evolving designs of the new SNFP Path Forward facilities to reconfirm that DOE Orders 5480.7A, <i>Fire Protection</i> , and 6430.1A, <i>General Design Criteria</i> , provide adequate fire protection requirements to achieve nuclear safety equivalency. Aspects of the designs to reconfirm are the use of a passive cooling system for MCO cooling in the CSB and the lack of Safety Class 1 accident prevention and/or mitigation systems in the CSB. The Safety Class 1 systems required for the other new SNFP Path Forward facilities remains to be determined. Further, fire protection requirements considered for incorporation into the design of the new SNFP Path Forward facilities should take into account the implementation of §72.122(c) to date for licensed ISFSIs.
Section 50.49. Environmental qualification of electric equipment important to safety for nuclear power plants	In order to achieve equivalence, consideration should be given to ensuring that the electrical equipment qualification program includes the elements that are missing from the DOE requirements for: 1) Safety Class 1 equipment; 2) non-Safety Class equipment that could, upon failure, adversely impact Safety Class 1 equipment in performance of their safety function; and, 3) certain post-accident monitoring equipment as described in Regulatory Guide 1.97, Revision 3, <i>Instrumentation for Light-Water-Cooled Nuclear Power Plants to Access Plant and Environs Conditions During and Following an Accident</i> . These elements include the following: <ul style="list-style-type: none"> • The AEs for new SNFP Path Forward facilities need to review §50.49(e)(5) and Regulatory Guide 1.89, <i>Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants</i>, during the design processes to determine the aging requirements for electrical equipment. • Testing requirements as outlined in §50.49(f)(1-4).
Section 50.54(a). Conditions of licenses (Quality Assurance Provisions)	Before implementation, consider having the DOE-RL review and approve changes to the WHC (SNFP) quality assurance program that could be interpreted as decreasing initial commitments.
Section 50.55(e). Conditions of construction permits	Consideration should be given to employing the Occurrence Reporting System during design and construction of new SNFP Path Forward facilities.

Table 2 Actions for Consideration

Regulations/Guidance	Actions for Consideration
Section 50.55a, <i>Codes and standards</i> , and NRC Regulatory Guide 1.153, Revision 0, <i>Criteria for Power, Instrumentation, and Control Portions of Safety Systems</i>	Consideration should be given to ensuring the AEs for the new SNFP Path Forward facilities apply IEEE Std 603-1991, <i>IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations</i> , to Safety Class 1 instrumentation and control systems.
Section 50.63, <i>Loss of all alternating current power</i>	Consider including in the SNFP Path Forward Safety Management Plan or the SAR Preparation Plans for new SNFP Path Forward facilities the need to evaluate the response of the facilities to a loss of ac power.
Part 50, Appendix A, <i>General Design Criteria for Nuclear Power Plants</i>	During the final design of the new SNFP Path Forward facilities other than the CSB, consideration should be given to having the AEs review 1) the conclusions of the general design criteria discussion of Attachment A to this report, and the general design criteria of 10CFR50, Appendix A to determine if NRC nuclear safety equivalency is achieved without application of any of the criteria of 10CFR50, Appendix A.
Part 50, Appendix B, <i>Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants</i> , and 10CFR72, Subpart G, <i>Quality Assurance</i>	Section 830.120, <i>Quality Assurance Requirements</i> , contains the majority of the NRC quality assurance requirements that are in 10CFR50, Appendix B and 10CFR72, Subpart G. The differences in the sets of requirements, as discussed in detail in Attachment A of this report, are included in DOE Order 4700.1, <i>Project Management System</i> , and WMC quality assurance procedures and instructions. For nuclear safety equivalency, ensure the appropriate requirements of the referenced WMC procedures and instructions remain in effect (e.g., through use of SNFP-specific documents).
Section 72.3, <i>Definitions</i>	<p>Consider identifying SSCs "important to safety", in accordance with §72.3, as follows:</p> <p>"Those SSCs that assure there is not a credible breach of the spent nuclear fuel containment/confinement such that a radioactive release could be postulated to reach the controlled area boundary defined as 100 meters from the facility release point."</p> <p>Once SSCs "important to safety" have been identified, impose the requirements for Safety Class 1 SSCs as defined in WMC-CM-4-46, <i>Safety Analysis Manual</i>, Section 9.0, Revision 1, <i>Safety Classification of Structures, Systems, and Components</i>.</p>
Section 72.24, <i>Contents of application: Technical information</i>	To have equivalent technical information in the application, the SAR(s) or other safety documentation for new SNFP facilities would need to incorporate additional information (representative details provided in Table 5.c). In terms of whether such information would be relevant to the design and construction of new SNFP facilities and to the health and safety of the public or the worker, the NRC views the submittal of such information, and the opportunity for the NRC to review the information called for in the applicable Regulatory Guides (i.e., 3.48 and 3.50), as enhancing the nuclear safety of the nuclear plant. This is due to the NRC's viewpoint that they have a broad awareness and thorough knowledge of the nuclear safety issues, more so than individual applicants, through their review of numerous applications. If one accepts this premise, the inclusion of such information is relevant to NRC nuclear safety equivalency. In addition, related to concerns for future populations, their locations, and issues such as Hanford privatization, as the Hanford Site evolves, public health and safety must be revisited and probable use of Hanford lands in the future should be considered in the design bases of new SNFP Path Forward facilities at this time.
Section 72.26, <i>Contents of application: Technical specifications</i>	Refer to §50.36a.
Section 72.44, <i>License conditions</i>	Refer to §50.36a.
Section 72.90, <i>General considerations</i>	Refer to §72.92.

Table 2 Actions for Consideration

Regulations/Guidance	Actions for Consideration
Section 72.92, Design basis external natural events	To develop, in a strict regulatory context, equivalent and the most conservative seismic criteria for Safety Class 1 SSCs, consider performing the analyses necessary to define the design earthquakes by both the DOE requirements (DOE Order 5480.28, <i>Natural Phenomena Hazards Mitigation</i> , and DOE STD-1020-94, <i>Natural Phenomena Hazards Design and Evaluation Criteria</i>) and the NRC requirements (10CFR100, Appendix A) and use the limiting design earthquakes. Or consider, 1) developing a justification for the use of a 0.35g design basis earthquake for Safety Class 1 SSCs in the design of the CSB, and 2) justify the use of DOE 5480.28, and companion standards, to determine the appropriate design earthquakes for the Conditioning Facility and other new SNFP Path Forward facilities
Section 72.98, Identifying regions around an ISFSI or MRS site	Refer to §50.34
Section 72.100, Defining potential effects of the ISFSI or MRS on the region	Refer to §§72.104 and 72.106
Section 72.102, Geological and seismological characteristics	Refer to §72.92
Section 72.104, Criteria for radioactive materials, in effluents and direct radiation from an ISFSI or MRS	To have equivalent radiological exposure criteria for the public, §72.104 exposure limits would need to be applied to the new SNFP Path Forward facilities
Section 72.106, Controlled areas of an ISFSI or MRS	Safety analysis used to determine safety classification should use a deterministic value of 5 rem for the onsite worker at the boundary of the controlled area (defined as 100 meters from the release point of the new SNFP Path Forward Facility).
Section 72.122, Overall requirements	To have equivalent requirements: <ul style="list-style-type: none"> • Refer to §72.92. • Since the CSB and the Conditioning Facility are expected to be located on essentially the same site and may be joined in some manner, design considerations need to include the NRC safety concerns regarding the sharing of common utilities and services and physical interaction between the two facilities. This consideration for the sharing of systems and facility interfaces should also be considered for other new SNFP Path Forward facilities, such as the cold vacuum drying facility. • The ability for ready retrieval of WCOs should be provided for in the design of the CSB. • The design for Safety Class 1 SSCs of the CSB and the other new SNFP Path Forward facilities (with the exception to activities internal to k Basins) should give consideration to the most severe credible tornado and tornado missile taking into consideration the most recent revision of Regulatory Guide 1.76, <i>Design Basis Tornado for Nuclear Power Plants</i>, SECY-93-087, <i>Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs</i>, and NUREG/CR-4461, <i>Tornado Climatology of the Contiguous United States</i>, (revisions under consideration for SRP 3.5.1.4, Revision 2.)
Section 72.126, Criteria for radiological protection	Refer to 10CFR20 and §§72.104 and 72.106.
Section 72.212, Conditions of general license issued	Refer to §672.92 and 72.122.

Table 2 Actions for Consideration

Regulations/Guidance	Actions for Consideration
NRC Regulatory Guide 1.26, Revision 3, <i>Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants</i>	Consider having the AEs for the new SNFP Path Forward facilities use Regulatory Guide 1.26 to assist in assigning the appropriate code class to ASME Section III components and systems.
NRC Regulatory Guide 1.60, Revision 1, <i>Design Response Spectra for Seismic Design of Nuclear Power Plants</i>	Consider justifying seismic response spectra used for Safety Class 1 SSCs for new SNFP Path Forward facilities if other than those included in Regulatory Guide 1.60 are to be used. This is in accordance with Regulatory Guide 3.48, Revision 1, <i>Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation (Dry Storage)</i> .
NRC Regulatory Guide 1.61, Revision 0, <i>Damping Values for Seismic Design of Nuclear Power Plants</i>	Consider justifying damping values for seismic design for Safety Class 1 SSCs for new SNFP Path Forward facilities if other than those included in Regulatory Guide 1.61 are to be used.
NRC Regulatory Guides 1.84 and 1.85, both Revision 30, <i>Design and Fabrication Code Case Acceptability ASME Section III, Division 1 and Materials Code Case Acceptability ASME Section III, Division 1</i> , respectively	Give consideration to having the AEs for the SNFP Path Forward facilities review the NRC positions in Regulatory Guides 1.84 and 1.85 on ASME code cases before using such cases for Safety Class 1 application. Reflect the NRC positions on the applicable code cases in the design of the facilities. Where no NRC position is stated in regards to the acceptance of a code case, that code case may be used as approved.
NRC Regulatory Guide 1.128, Revision 1, <i>Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants</i>	Consideration should be given to having the AEs for new SNFP Path Forward facilities incorporate the requirements of IEEE Std 484-1987, <i>IEEE Recommended Practice for Installation, Design and Installation of Large Lead Storage Batteries for Generating Stations and Substations</i> , into the design and installation of Safety Class 1 batteries.
NRC Regulatory Guide 1.140, Revision 1, <i>Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants</i>	Give consideration to having the AEs for new SNFP Path Forward facilities include ANSI/ANS N509-1989, <i>Nuclear Power Plant Air-Cleanup Units and Components</i> , and ANSI/ANS N510-1989, <i>Testing of Nuclear Air Treatment Systems</i> , for those HVAC systems needed to achieve onsite radiological limits or to implement the principals of ALARA (i.e., this applies to WMC Safety Class 2 and 3 systems).
NRC Regulatory Guide 1.158, Revision 0, <i>Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants</i>	Consideration should be given to having the AEs for new SNFP Path Forward facilities incorporate the requirements of IEEE Std 535-1986, <i>IEEE Standard for Qualification of Class 1E Lead Storage Batteries for Nuclear Power Generating Stations</i> , for the qualification of Safety Class 1 lead storage batteries.
NRC Regulatory Guide 3.60, Revision 0, <i>Design of an Independent Spent Fuel Storage Installation (Dry Storage)</i>	Give consideration to having the AE for the CSB perform a review of the CSB design against ANSI/ANS-57.9-1992, <i>Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type)</i> , to identify any additional requirements that would be needed to demonstrate nuclear safety equivalency.
NRC Regulatory Guide 8.5, Revision 1, <i>Criticality and Other Interior Evacuation Signals</i>	Refer to Regulatory Guide 8.12, Revision 2, <i>Criticality Accident Alarm Systems</i> .
NRC Regulatory Guide 8.8, Revision 3, <i>Information Relative to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Reasonably Achievable</i>	Consideration should be given to ensuring that the AEs for the new SNFP Path Forward facilities incorporate the requirements of Regulatory Guide 8.8, Revision 3 into the design of these facilities.

Table 2 Actions for Consideration

Regulations/Guidance	Actions for Consideration
NRC Regulatory Guide 8.12, Revision 2, <i>Criticality Accident Alarm Systems</i> and Section 70.24, <i>Criticality accident requirements</i>	For new SNFP Path Forward facilities, consider incorporating the requirements of ANSI/ANS-8.3-1986, <i>Criticality Alarm System</i> , into the design. Also consider using the MCO design, which includes the basket configuration, and the quantities/form of the K Basin fuel (special nuclear material) as a basis for not incorporating criticality accident alarm systems in new SNFP Path Forward facilities, based on demonstrating through safety analyses documented in the associated SARs that criticality is not possible. In conjunction with this activity, demonstrate compliance with the criteria in DOE Order 5480.24, <i>Nuclear Criticality Safety</i> , and ANSI/ANS-8.3-1986.
NUREG-0700, <i>Guidelines for Control Room Design Reviews</i> , and SRP 18.1, Revision 0, <i>Control Room</i>	Consideration should be given to having the AEs for the new SNFP Path Forward facilities review the NRC guidance of NUREG-0700 and SRP 18.1 against DOE Order 6430.1A, Section 1300-12.4 and the DOE draft standard, <i>Human Factors Engineering Design Criteria: Volume I, General Criteria</i> , to identify appropriate additional NRC guidance for design of these facilities. These reviews should give consideration to the differences in complexity between power reactor control rooms and those of the new SNFP Path Forward facilities.
SRP 9.1.2, Revision 3, <i>Spent Fuel Storage</i> , and NUREG-0612, <i>Control of Heavy Loads at Nuclear Power Plants, Resolution of Generic Technical Activity A-36</i>	Consideration should be given to invoking the NRC k_{eff} criticality safety limit of 0.95.
SRP 9.1.5, Revision 0, <i>Overhead Heavy Load Handling Systems</i>	Consideration should be given to requiring that the AEs for new SNFP Path Forward facilities incorporate the design requirements of ANSI/ANS-57.1, <i>Design Requirements for Light Water Reactor Fuel Handling System</i> , and ANSI/ANS-57.2, <i>Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants</i> . This would apply only to those facilities that will include the capability to lift a cask or MCO.
GLs 88-14, <i>Instrument Air Supply System Problems Affecting Safety-Related Equipment</i> ; 89-10, <i>Safety-Related Motor-Operated Valve Testing and Surveillance</i> 10CFR50.54(f), w/supplements, and 89-13, <i>Service Water System Problems Affecting Safety-Related Equipment</i> .	The applicable portions of Generic Letters 88-14, 89-10, and 89-13 should be considered by the AEs for new SNFP Path Forward facilities for application to Safety Class 1 systems and components.
IEN 95-29, <i>Oversight of Design and Fabrication Activities for Metal Components used in Spent Fuel Dry Storage Systems</i>	Consideration should be given to ensuring the areas of vendor and subcontractor quality assurance records and control of Safety Class 1 purchased material, equipment, and services receive emphasis during SNFP Path Forward audits, surveillances, and assessments.
IE Bulletins and Notices, several documents addressing procurement of potentially defective equipment	Consideration should be given to instituting a process to identify Safety Class 1 equipment that has been identified in the commercial nuclear power industry, via IE Bulletins and Notices, as being potentially defective.

Table 3 Title 10 CFRs Non-Applicable Parts/Sections/Appendices

1. *Part 0 - Conduct of Employees*

The provisions of this part apply to current and former NRC employees and special Government employees (an officer or employee of the NRC performing temporary duties). This part incorporates the statutes, instructions and specific procedures pertaining to an employee's conduct.
2. *Part 1 - Statement of Organization and General Information*

This part addresses the creation, authority, and organization of the NRC and panels, boards, etc. associated with the NRC licensing process (e.g., the Atomic Safety and Licensing Board and Advisory Committee on Reactor Safeguards) among other NRC-specific information.
3. *Part 4 - Nondiscrimination in Federally Assisted Commission Programs*

The part implements the provisions of Title VI of the Civil Rights Act of 1964, which relates to nondiscrimination with respect to race, color, national origin or sex and to nondiscrimination with respect to the handicapped in any program or activity receiving Federal financial assistance from the NRC.
4. *Part 7 - Advisory Committees*

This part provides general policy and requirements governing NRC establishment and implementation of advisory committees in support of NRC activities, including topics such as charter, conduct of meetings, public notice, minutes, transcripts and reports among other committee-specific information.
5. *Part 8 - Interpretations*

This part provides interpretations by the NRC General Counsel of specific sections of the Atomic Energy Act of 1954 and other select NRC-related activities, including section 152 on inventions and discoveries, the Price-Anderson Act (section 170), AEC [NRC] jurisdiction over nuclear facilities and materials under the Atomic Energy Act, and clarification on the illumination and physical search requirements of Part 73.55(c)(5) and Part 73.55(d)(1) respectively. Part 73.55 is titled "Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage. "The physical protection requirements of Part 73.55 will be addressed in conjunction with the Part 73 discussion. The associated clarifications

Table 3 Title 10 CFRs Non-Applicable Parts/Sections/Appendices

(interpretations) provided in Part 8.5 will be included in the Part 73 discussion as appropriate as opposed to being addressed at this point.

6. Part 9 - *Public Records*

This part implements the provisions of the Freedom of Information Act, the Privacy Act, the Government in the Sunshine Act (i.e., opening NRC Commission meetings to public observation), and describes the procedures for production of agency records, information, or testimony in response to subpoenas or demands of courts or other judicial or quasi-judicial authorities in State and Federal proceedings.

7. Part 10 - *Criteria and Procedures for Determining Eligibility for Access to Restricted Data or National Security Information or an Employment Clearance*

This part establishes the criteria, procedures, and methods for 1) resolving questions concerning the eligibility of individuals employed by or applicants for employment with the NRC, or NRC contractors, agents and licensees of the NRC for access to Restricted Data or National Security Information, and 2) the eligibility of NRC employees or applicants for employment with the NRC for employment clearance.

8. Part 12 - *Implementation of the Equal Access to Justice Act in Agency Proceedings*

This part provides the regulatory requirements for award of attorney fees to eligible individuals and entities in certain administrative proceedings before the NRC.

9. Part 13 - *Program Fraud Civil Remedies*

This part implements the Program Fraud Civil Remedies Act of 1986 which establishes administrative procedures for imposing civil penalties and assessments against persons who make, submit, or present, or cause to be made, submitted, or presented, false, fictitious, or fraudulent claims or written statements to authorities or their agents.

10. Part 14 - *Administrative Claims under Federal Tort Claims Act*

This part administrative claims for money damages against the United States for damage to or loss of property or personal injury or death caused by the negligent or wrongful act or omission of any employee of the NRC where the United States would be liable to the claimant per the law of the place where the act or omission occurred.

Table 3 Title 10 CFRs Non-Applicable Parts/Sections/Appendices

11. Part 15 - *Debt Collection Procedures*

This part provides procedures for claims for the payment of debts owed to the United States unless a different procedure is specified in a statute, regulation or contract.

12. Part 16 - *Salary Offset Procedures for Collecting Debts Owed by Federal Employees to the Federal Government*

This part provides procedures for the collection by administrative offset of a Federal employee's salary without their consent to satisfy certain debts owed to the Federal Government.

13. Part 25 - *Access Authorization for Licensee Personnel*

This part establishes procedures for granting, reinstating, extending, transferring and terminating access authorizations of licensee personnel, licensee contractors or agents and other persons that may require access to information classified as Secret and Confidential National Security Information and/or Restricted Data.

14. Part 31 - *General Domestic Licenses for Byproduct Material*

This part establishes general licenses for the possession and use of byproduct material (any radioactive material, except special nuclear material, yielded in or made radioactive by exposure to the radiation incident to the process of producing or utilizing special nuclear material) contained in certain items and a general license for ownership of byproduct material. Examples include: static elimination devices; ion generating tubes; ion detection devices; certain measuring, gauging or controlling devices; luminous safety devices; and for use in in-vitro clinical or laboratory testing.

15. Part 32 - *Specific Domestic Licenses to Manufacture or Transfer Certain Items Containing Byproduct Material*

This part provides requirements for the issuance of specific licenses to persons that are involved with the manufacture, or initial transfer, of items containing byproduct material for sale or distribution.

Table 3 Title 10 CFRs Non-Applicable Parts/Sections/Appendices

16. Part 33 - *Specific Domestic Licenses of Broad Scope for Byproduct Material*

This part provides requirements for the issuance of specific licenses of broad scope for byproduct material receipt, acquisition, ownership, possession, use and transfer.

17. Part 34 - *Licenses for Radiography and Radiation Safety Requirements for Radiographic Operations*

This part provides requirements for the issuance of licenses for the use of sealed sources containing byproduct material and radiation safety requirements for persons using such sealed sources in radiography.

18. Part 35 - *Medical Use of Byproduct Material*

This part provides requirements for the licensing of byproduct material for medical uses.

19. Part 39 - *Licenses and Radiation Safety Requirements for Well Logging*

This part provides requirements for the issuance of a license authorizing the use of licensed materials including sealed sources, radioactive tracers, radioactive markers, and uranium sinker bars in well logging.

20. Part 52 - *Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants*

This part governs the issuance of early site permits, standard design certifications, and combined licenses for nuclear power facilities licensed under the Atomic Energy Act of 1954. Early site permit means an NRC approval, issued pursuant to this part, for a site or sites for one or more nuclear power facilities. A standard design certification means an NRC approval, issued pursuant to this part, of a standard design for a nuclear power facility. A combined license means a combined construction permit and operating license with conditions for a nuclear power facility issued pursuant to this part. The majority of requirements of this part are administrative in nature. Those requirements associated with the technical information in applications do not raise any new or additional requirements beyond those addressed in §§50.34, 72.24 and 72.26, which are discussed on their own merits in Tables 5.b and 5.c.

Table 3 Title 10 CFRs Non-Applicable Parts/Sections/Appendices

21. Part 53 - *Criteria and Procedures for Determining the Adequacy of Available Spent Nuclear Fuel Storage Capacity*

This part establishes procedures and criteria for making the determination required by the Nuclear Waste Policy Act of 1982 that a person owning or operating a civilian nuclear power reactor cannot reasonably provide adequate spent nuclear fuel storage capacity at the reactor site, or at the site of any other reactor operated by that person, when needed to ensure the continued orderly operation of the reactor, and that such person is diligently pursuing licensed alternatives to the use of Federal storage capacity for the storage of spent nuclear fuel expected to be generated by such person in the future.

22. Part 54 - *Requirements for Renewal of Operating Licenses for Nuclear Power Plants*

This part governs the issuance of renewed operating licenses for nuclear power plants licensed pursuant to the Atomic Energy Act of 1954. It applies to the additional amount of time beyond the expiration of the existing operating license that is requested in the renewal application.

23. Part 55 - *Operators' Licenses*

This part establishes procedures and criteria for the issuance of licenses to operators and senior operators of utilization facilities that manipulate the controls that directly affect reactivity or power level of any utilization facility licensed pursuant to Part 50.

24. Part 61 - *Licensing Requirements for Land Disposal of Radioactive Waste*

This part establishes for land disposal of radioactive waste the procedures, criteria, and terms and conditions upon which the NRC issues licenses for the disposal of radioactive wastes containing byproduct, source and special nuclear material received from other persons.

25. Part 62 - *Criteria and Procedures for Emergency Access to Non-Federal and Regional Low-Level Disposal Facilities*

This part establishes for specific low-level radioactive waste the criteria and procedures for granting emergency access under the Low-Level Radioactive Waste Policy Amendments Act of 1985 to any non-Federal or regional low-level radioactive waste disposal facility. Emergency access means access, for waste disposal purposes, to an operating non-Federal or regional low-level radioactive waste disposal facility for a

Table 3 Title 10 CFRs Non-Applicable Parts/Sections/Appendices

period not to exceed 180 days, which is granted by the NRC to a generator of low-level radioactive waste who has been denied the use of those facilities.

26. Part 76 - *Certification of Gaseous Diffusion Plants*

This part establishes requirements that will govern the operation of those portions of the Portsmouth and Paducah Gaseous Diffusion Plants that are leased by the United States Enrichment Corporation for the purpose of uranium enrichment.

27. Part 81 - *Standard Specifications for the Granting of Patent Licenses*

This part establishes the standard specification for the issuance of licenses to rights in inventions covered by patents or patent applications, as represented by or in the custody of the NRC and other patents in which the NRC has the right to accord or require the grant of licenses.

28. Part 95 - *Security Facility Approval and Safeguarding of National Security Information and Restricted Data*

This part establishes procedures for obtaining security facility approval (NRC determination that a facility is eligible to use, process, store, reproduce, transmit or handle classified matter) and for safeguarding Secret and Confidential National Security Information (information determined pursuant to Executive Order 12356 or any predecessor order to require protection against unauthorized disclosure) and Restricted Data (concerns design, manufacture or use of atomic weapons, the production of special nuclear material [SNM], or the use of SNM in the production of energy) received or developed in conjunction with activities licensed or regulated by the NRC. In addition, Appendix A to Part 95 provides guidance for classifying safeguards information.

29. Part 110 - *Export and Import of Nuclear Equipment and Material*

This part provides licensing, enforcement, and rulemaking procedures and criteria, under the Atomic Energy Act, for the export and import of nuclear equipment and material.

30. Part 140 - *Financial Protection Requirements and Indemnity Agreements*

This part provides procedures and requirements for determining the financial protection required of licenses and for the indemnification and limitation of liability of certain licensees and other persons

Table 3 Title 10 CFRs Non-Applicable Parts/Sections/Appendices

pursuant to section 170 of the Atomic Energy Act of 1954 (the Price-Anderson Act).

31. Part 150 - *Exemptions and Continued Regulatory Authority in Agreement States and in Offshore Waters under Section 274*

This part provides certain exemptions to persons in Agreement States from the licensing requirements contained in Chapters 6, 7 and 8 of the Atomic Energy Act and from the regulations of the NRC imposing requirements upon persons who receive, possess, use or transfer byproduct material, source or special nuclear material, in quantities not sufficient to form a critical mass; and to define activities in Agreement States and in offshore waters over which the regulatory authority of the NRC continues.

32. Part 170 - *Fees for Facilities, Materials, Import and Export Licenses, and other Regulatory Services under the Atomic Energy Act of 1954, as Amended*

This part provides fees charged for licensing services rendered by the NRC.

33. Part 171 - *Annual Fees for Reactor Operating Licenses, and Fuel Cycle Licenses and Materials Licenses, including Holders of Certificates of Compliance, Registrations, and Quality Assurance Program Approvals and Government Agencies Licensed by NRC*

This part identifies the annual fees charged to persons who hold licenses, Certificates of Compliance, sealed source and device registrations, and quality assurance program approvals issued by the NRC.

34. Parts 172 through 199 - Reserved as of January 1, 1995

Table 4 Title 10 CFRs Accommodated by WMC Site-Wide Programs

Part	Title	SNFP Path Forward Discussion	NRC safety issue(s)
19	<i>Notices, Instructions, and Reports to Workers; Inspection and Investigations</i>	<p><u>Conclusions:</u> This is a WMC site-wide activity with no unique K Basins SNFP Path Forward implications and the activity has no impact on design or construction of new SNFP Path Forward facilities. As such, no SNFP Path Forward action is required and no additional evaluation of 10CFR19 is provided for this equivalency comparison.</p> <p><u>Actions for Consideration:</u> None.</p>	<p>This part establishes posting requirements relating to: 10CFR20 posting requirements; the facility license, including license conditions; operating procedures applicable to the licensed activity; Notices of Violation involving radiological activities or a proposed civil penalty; and: NRC-3 "Notice to Employees" (which generally informs workers of their right to contact the NRC).</p> <p>This part also requires instructions be provided to individuals working in the restricted area regarding: the risks of radiation exposure and the need to minimize such exposure; the proper response to radiation warnings; the need to report promptly potential violations of NRC requirements or unnecessary radiation exposure; and the individuals right to request radiation exposure records.</p> <p>Requirements are also established for: informing workers of exposure received; of supporting the NRC during inspections by access to the facility, materials, records, and personnel (including private conversations); and employee rights to report conditions to the NRC.</p> <p>With the exception of the need to provide space for postings, implementation of 10CFR19 has no impact on design or construction of the licensee's facilities.</p>

Table 4 Title 10 CFRs Accommodated by WMC Site-Wide Programs

21	Reporting of Defects and Noncompliance	<p>Defect reporting will be addressed by proposed §830.122, <i>Defect Identification and Reporting</i>, if, and when, this is approved. This rule would require DOE contractors, subcontractors, and suppliers to identify, evaluate, and report defective items and services. The rule would ensure that all DOE facilities known or suspected to have used defective items or services would be notified so that appropriate corrective action could be taken.</p> <p>Title 10, CFR 820 indicates that DOE will conduct audits and assessments of its contractors and may audit or assess subcontractors and suppliers for adherence to this program requirement.</p> <p>DOE Order 5000.3B, <i>Occurrence Reporting and Processing of Operating Information</i>, provides a significant amount of information equivalent to 10CFR21. Refer to §§50.55(e) and 50.72 of Table 5 b for this information.</p> <p>Conclusions: As this is a WMC site-wide activity with no unique SNFP Path Forward implications, no SNFP Path Forward specific evaluation is required. However, one issue to be resolved WMC site-wide is that there is currently no DOE order that requires reporting of defects and noncompliance equivalent to 10CFR21 by vendors (e.g., during CSB procurement and construction).</p> <p>WMC has developed a Procurement Clause E13, <i>Reporting of Safety Defects and Noncompliance</i>, which is contained in Appendix A, Revision 1, of WMC-CM-4-2, <i>Quality Assurance Manual</i>. This clause was developed to incorporate 10CFR21 provisions into procurement documents, if needed. Additional terms and conditions provided to suppliers with Purchase Orders contain provisions relevant to 10CFR21 (e.g., 50-6.6, <i>Warranty</i>, and 50-46, <i>Counterfeit Fasteners and Components</i>). Adding Procurement Clause E13 to SNFP Path Forward procurement specifications will achieve NRC equivalency for 10CFR21.</p> <p>Actions for Consideration: Procurement clause E13 of WMC-CM-4-2, <i>Quality Assurance Manual</i>, Appendix A, Revision 1, should be considered for addition to Safety Class 1 procurement specifications to require the reporting by suppliers of defects in items or services.</p>	<p>Title 10, CFR 21 implements Section 206 of the Atomic Energy Act which imposes the obligation on directors and responsible officers of licensed and non-licensed facilities and firms to report defects in a facility, activity or basic component to the NRC. The terms "defects" and "basic component" are defined in the rule. Prior to 10CFR21, there was no clearly established rule requiring a non-NRC licensed or permitted firm to report defects to the NRC. This included vendors and utilities prior to receipt of a construction permit. Title 10, CFR 21 corrected this.</p> <p>By revision to 10CFR21, effective October 29, 1991, construction permitted and licensed facilities are, in most cases, only required to report defects under §§50.55(e) and 50.72, respectively. See the response provided for these sections in Table 5 b for additional detail.</p> <p>Vendors will continue to report defects under 10CFR21.</p>
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Table 4 Title 10 CFRs Accommodated by WMC Site-Wide Programs

26	<i>Fitness for Duty Programs</i>	<p>Fitness for duty requirements for DOE contractors are established by 10CFR707, <i>Workplace Substance Abuse Programs at DOE Sites</i> and DOE Order 3792-3, <i>Drug-Free Federal Workplace Testing Implementation Program</i>. To satisfy the requirements of these documents, a fitness for duty program is managed on a WMC site-wide basis with requirements and programs implemented in a common manner for all WMC operated nuclear facilities on the Hanford Site. There are no unique elements of this program for the SNFP Path Forward.</p> <p><u>Conclusions:</u> With a WMC site-wide implemented fitness for duty program and with no SNFP Path Forward unique requirements for this program, no SNFP Path Forward action is required and no additional evaluation of 10CFR26 is provided as part of this equivalency comparison.</p> <p><u>Actions for Consideration:</u> None.</p>	This part ensures that individuals authorized for unescorted access to the protected area are fit for duty.
50 §50.5	<i>Deliberate misconduct</i>	<p>The requirements for dealing with deliberate misconduct for DOE contractor employees are established by 10CFR820, Subpart F, <i>Criminal Penalties</i>. To satisfy the requirements of this rule, informing employees of the risk of deliberate misconduct and identification of the situation, should it occur, is managed on a WMC site-wide basis with requirements and programs implemented in a common manner for all WMC operated nuclear facilities on the Hanford Site. In addition, there are no unique elements of this program for the SNFP Path Forward.</p> <p><u>Conclusions:</u> With a WMC site-wide implemented program for dealing with deliberate misconduct and with no SNFP Path Forward unique requirements for this program, no SNFP Path Forward action is required and no additional evaluation of §50.5 is provided as part of this equivalency comparison.</p> <p><u>Actions for Consideration:</u> None.</p>	This section extended the NRC's ability to subject personnel to enforcement action for willful misconduct to those not holding a 10CFR55 operating license. This includes deliberate violation of NRC requirements, falsification of records, false statements to the NRC, and interference with NRC investigations. For ISFSIs licensed under 10CFR72, the considerations for deliberate misconduct are addressed in §72.12, which establishes the same requirements as §50.5.

Table 4 Title 10 CFRs Accommodated by WMC Site-Wide Programs

50 §50.7	Employee protection	<p>The requirements for employee protection for DOE contractor employees is established by 10CFR708, <i>DOE Contractor Employee Protection Program</i> and RLID 5480.29, <i>RI Employee Concerns Program</i>. WMC has been granted a waiver from DOE Order 5480.29, <i>Employee Concerns Management System</i> in favor of implementing RLIP 5480 SCP, <i>Safety Concerns Program</i>. This implementing procedure has since be replaced by RLID 5480.29. To satisfy the requirements of the rule and order, employee protection against discrimination for disclosure of information relating to such issues as health and safety or mismanagement is managed on a WMC site-wide basis with requirements and programs implemented in a common manner for all WMC operated nuclear facilities on the Hanford Site. In addition, there are no unique elements of this program for the SNFP Path Forward.</p> <p><u>Conclusions.</u> With a WMC site-wide implemented employee protection program and with no SNFP Path Forward unique requirements for this program, no SNFP Path Forward action is required and no additional evaluation of §50.7 is provided as part of this equivalency comparison.</p> <p><u>Actions for Consideration.</u> None.</p>	<p>Section 50.7 implements Section 210 of the Energy Reorganization Act which states that an employee may not be discharged or otherwise discriminated against for initiating or taking part in proceedings of the Act. For ISFSIs licensed under 10CFR72, the considerations for employee protection are addressed in §72.10, which establishes the same requirements as §50.7.</p>
50 §50.9	Completeness and accuracy of information	<p>The requirements for providing DOE complete and accurate information are established in §820.11, <i>Information Requirements</i>. The importance of informing DOE of information that has been found to be incomplete or inaccurate is addressed in Appendix A, Section X, <i>Inaccurate and Incomplete Information</i>, of 10CFR820. The need to provide DOE complete and accurate information is managed on a WMC site-wide basis with requirements and programs implemented in a common manner for all WMC operated nuclear facilities on the Hanford Site. In addition, there are no unique elements of this program for the SNFP Path Forward.</p> <p><u>Conclusions.</u> With a WMC side-wide program for ensuring that DOE is provided complete and accurate information and with no SNFP Path Forward unique requirements for this program, no SNFP Path Forward action is required and no additional evaluation of §50.9 is provided as part of this equivalency comparison.</p> <p><u>Actions for Consideration.</u> None.</p>	<p>This section provides assurance that information the NRC may act upon is complete and accurate and requires that the NRC be informed when inaccurate information has been provided that may have significant implication for public health and safety, common defense, and security. For ISFSIs licensed under 10CFR72, the considerations for providing the NRC complete and accurate information are addressed in §72.11, which establishes the same requirements as §50.9.</p>

Table 4 Title 10 CFRs Accommodated by WMC Site-Wide Programs

50 §50.12	<i>Specific exemptions</i>	<p>The process for requesting exemptions is specific for each Part of the Code of Federal Regulations. For example, 10CFR820, Subpart E, <i>Exemption Relief</i>, provides for relief of DOE nuclear safety requirements of that part.</p> <p>WMC-CM-4.52, <i>Nuclear Safety Rule Compliance</i>, Section 4.0, <i>Exemption Requests</i>, is being prepared to facilitate exemptions requests associated with 10CFR820. Exemption requests are not required for DOE directives as they are not regulations.</p> <p><u>Conclusions:</u> No SNFP Path Forward action is required and no specific evaluation is provided for this equivalency comparison as the process of seeking exemption relief is specific to the agency involved. In addition, the appropriateness of considering an exemption is specific to the part of Title 10 in question. For example, some parts of Title 10 do not provide for granting of exemptions.</p> <p><u>Actions for Consideration:</u> None.</p>	<p>This section provides for resolution of conflicting 10CFR50 regulations, allows for relief from a regulation that would provide little or no benefit, and provides short-term relief from specific regulations. Section 50.12 deals only with relief from NRC regulations as established in 10CFR50. It does not apply to exemption requests for other parts of Title 10. For example, 10 CFR Parts 30, 50, 55 and 70 have their own processes for requesting exemptions. For ISFSIs licensed under 10CFR72, the considerations for exemption requests are addressed in §72.7.</p>
50 §50.72	<i>Immediate notification requirements for operating nuclear power reactors</i>	<p>For the SNFP Path Forward, this prompt notification is addressed by DOE Order 5000.3B, <i>Occurrence Reporting and Processing of Operations Information</i>. The capability to inform the DOE of an occurrence (i.e., the parallel to the NRC's Emergency Notification System) is provided by the Occurrence Notification Center located in the Federal Building. Proposed §830.350, <i>Categorization, Notification, Reporting, and Processing of Operational Occurrences at DOE Nuclear Facilities</i>, if issued, would codify the requirements of DOE Order 5000.3B and draft DOE Order 5000.3C.</p> <p><u>Conclusions:</u> No SNFP Path Forward action is required and no specific evaluation is provided for this equivalency comparison as occurrence reporting is a WMC site-wide issue with no unique SNFP Path Forward considerations. The WMC site-wide design provisions (dedicated phone) are provided by the Occurrence Notification System as discussed above.</p> <p><u>Actions for Consideration:</u> None.</p>	<p>This section provides for telephone notification to the NRC of the declaration of an emergency condition or the discovery of non-emergency events. With the exception of the requirement to provide for a dedicated phone at the nuclear power station that connects to the NRC's Emergency Notification System, implementation of §50.72 has no impact on the design or construction of the licensee's facilities.</p> <p>While §50.72 applies to holders of operating licenses only, the basic requirement to inform the regulating agency of significant events applies to all aspects of design, construction, and operation. See the discussion provided for §50.55(e) in Table 5 b.</p>

Table 4 Title 10 CFRs Accommodated by WMC Site-Wide Programs

50 550.73	Licensee event report (LER) system	<p>For the SNFP Path Forward, this written notification is addressed by DOE Order 5000.3B, <i>Occurrence Reporting and Processing of Operations Information</i>. Proposed rule 10CFR830, §830.350, <i>Categorization, Notification, Reporting, and Processing of Operational Occurrences at DOE Nuclear Facilities</i>, if issued, would codify the requirements of the DOE order.</p> <p><u>Conclusions:</u> No SNFP Path Forward action is required and no specific evaluation is provided for this equivalency comparison as occurrence reporting is a WMC site-wide issue with no unique SNFP Path Forward considerations.</p> <p><u>Actions for Consideration:</u> None.</p>	<p>This section provides for written notification to the NRC of off-normal occurrences. In most cases, this reporting is follow up to the telephone notification required by 550.72. The §§50.72 and 50.73 reports are reviewed by the NRC's Office for Analysis and Evaluation of Operational Data (AEOD) to determine the need for NRC and licensee actions. The 550.73 reports (Licensee Event Reports) are also used by AEOD as input to satisfy Section 208 of the Atomic Energy Act which requires that the NRC report to Congress on a quarterly basis abnormal occurrences at any licensed facility. This is done through the issuance of NUREG-0090. The NRC regional offices use the LERs to confirm that the correct root cause(s) and corrective actions have been identified and to determine if enforcement action might be appropriate.</p> <p>While 550.73 applies to holders of operating licenses only, as is the case for 550.72 discussed above, the basic requirement to inform the regulating agency of significant events applies to all aspects of design, construction, and operation.</p> <p>Implementation of 550.73 has no design or construction impact of the licensee's facilities.</p>
160	Trespassing on Commission Property	<p>For DOE facilities this issue is addressed by 10CFR860, <i>Trespassing on Administration Property</i>. The stated purpose of the DOE and NRC rules read almost exactly the same, except for the reference to the two different agencies.</p> <p><u>Conclusions:</u> DOE and NRC requirements are equivalent. In addition, no SNFP Path Forward action is required and no specific evaluation is provided for this equivalency comparison as control of trespassing on DOE Hanford Site property for which WMC has responsibility is addressed on a WMC site-wide basis with no SNFP Path Forward unique requirements. In addition, with the exception of requirements for posting, control of trespassing does not establish design or construction requirements for new facilities.</p> <p><u>Actions for Consideration:</u> None.</p>	<p>This part provides for the protection and security of facilities, installations and real property subject to the proprietary jurisdiction or administration, or in the custody of, the NRC.</p>

Table 5.a Title 10 CFR Summary-Level Comparisons of Technical/Safety Issues

Part	Title	SNFP Path Forward Discussion	NRC safety issue(s)
11	<i>Criteria and Procedures for Determining Eligibility for Access to or Control Over Special Nuclear Material</i>	<p>For DOE facilities, this access authorization is provided by 10CFR710, Subpart A, <i>General Criteria and Procedures for Determining Eligibility for Access to Classified Matter or Significant Quantities of Special Nuclear Materials</i>.</p> <p><u>Conclusions:</u> As 10CFR710 requires that controls be established for access authorization and this issue has no impact on design or construction of new SNFP Path Forward facilities, no SNFP Path Forward action is required and no additional evaluation of 10CFR11 is provided as part of this equivalency comparison.</p> <p><u>Actions for Consideration:</u> None.</p>	Title 10, CFR 11 establishes a process by which individuals could gain special nuclear material (see <i>Definitions</i> , Section 8.2) access authorization. It has no impact on the design or construction of the licensee's facilities.
20	<i>Standards for Protection Against Radiation</i>	<p>Title 10, CFR 835, <i>Occupational Radiation Protection</i>, and DOE Order 5480.11, <i>Radiation Protection for Occupational Workers</i>, designate exposure limits and ALARA principles equivalent to NRC 10CFR20 requirements. Section 20.1101(b) states that the licensee shall use, to the extent practicable, procedures and engineering controls based on sound radiation protection principles to achieve occupational doses and doses to members of the public that are ALARA. Section 20.1701 states that process or other engineering controls shall be used, to the extent practicable, to control the concentrations of radioactive material in the air. Section 835.1001, <i>Design and control</i>, has a stronger message relative to ALARA for workers. It states that the primary methods used to maintain exposures as low as is reasonably possible shall be physical design features and that administrative methods will only be used as supplemental methods.</p> <p>Section 20.1601 requires control devices to all High Radiation Areas (defined in §20.1003 as 0.1 rem in one hour at 30 cm) whereas 10CFR835 does not require incorporation of control devices until the dose rate in an accessible area reaches 1 rem per hour at 30 cm. Control devices are hardware features, such as alarms or locked entryways, as opposed to administrative controls.</p> <p>Offsite exposure limits from normal operations and anticipated occurrences for management and storage of spent fuel are covered for both DOE and NRC operations by 40CFR191, <i>Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-level and Transuranic Radioactive Waste</i>. Title 40, CFR 191 provides the protective limits for the public but does not discuss specific design or control methods. The 40CFR191 limits are more restrictive than those contained in §20.1301, draft 10CFR834, <i>Radiation Protection for the Public and the Environment</i>, or DOE Order 5400.5, <i>Radiation Protection for the Public and the Environment</i>, except that §20.1301 contains an hourly dose limit from external sources of 0.002 rem at the restricted area boundary. Title 40, CFR 191 limits are equivalent to the general public limits for ISFSIs contained in §72.104.</p>	Title 10, CFR 20 provides radiological programmatic requirements and protective limits for workers within the plant and for the public outside the restricted area during plant operation. General ALARA and waste disposal requirements are included. Hardware requirements for control of access to high and very high radiation areas are delineated.

Table 5.a Title 10 CFR Summary-Level Comparisons of Technical/Safety Issues

Part	Title	SNFP Path Forward Discussion	NRC safety issue(s)
20 cont'd	<i>Standards for Protection Against Radiation</i>	<p><u>Conclusions.</u> In summary, two design aspects of 10CFR20 that are more restrictive than DOE requirements exist. One is the criterion for access control to High Radiation Areas by control devices. Title 10, CFR 20 requires such control to all High Radiation Areas (defined in §20.1003 as 0.1 rem in one hour at 30 cm) whereas 10CFR835 does not require incorporation of control devices until the dose rate in an accessible area reaches 1 rem per hour at 30 cm. The second design aspect is the hourly limit to the public of 0.002 rem per hour. This limit may affect shielding design, or more likely, the location of the restricted area boundaries.</p> <p><u>Actions for Consideration:</u> The new SNFP Path Forward facilities should consider incorporating control devices for access to High Radiation Areas that conform to the requirements of §20.1601. Consideration should also be given to factoring into the design of the new SNFP Path Forward facilities the §20.1301 hourly limit of 0.002 rem to the public. This can be accommodated by the location of their restricted area boundaries.</p>	
30	<i>Rules of General Application to Domestic Licensing of Byproduct Material</i>	<p>For DOE facilities, the applicable 10CFR30 requirements and the related DOE rules and orders are: employee protection (10CFR708 and RLID 5480.29); completeness and accuracy of information (10CFR820, §820.11); deliberate misconduct (10CFR820, Subpart F); exemption requests (10CFR820, Subpart E); notification (5000.38); record keeping of material transfers and reporting (5633.38); support for inspections and tests (5482.18); and enforcement response (10CFR820, Subpart B).</p> <p><u>Conclusions.</u> In Table 4, the conclusion is reached that with WMC site-wide programs in place, no additional review is required for employee protection (§50.7 of Table 4), completeness and accuracy of information (§50.9), deliberate misconduct (§50.5), and exemption requests (§50.12).</p> <p>As programs are also in place, as listed above, for record keeping of material transfers, support for inspections and requested tests, and enforcement response, no SNFP Path Forward action is required and no additional evaluation of 10CFR30 is required as a result of this equivalency comparison. It should also be noted that these latter activities have no impact on design or construction of new SNFP Path Forward facilities.</p> <p><u>Actions for Consideration:</u> None</p>	<p>Utilities need a specific 10CFR30 license prior to operation to allow for receipt and use of byproduct material (e.g., check sources in radiation monitoring equipment and the startup neutron source, also see <i>Definitions</i>, Section 8.2). When the operating license is issued, it usually incorporates the 10CFR30 license.</p> <p>Power reactors with operating licenses do not usually develop special programs to deal with byproduct material they may possess as the programs required for plant operation usually provide adequate controls for this material.</p> <p>Title 10, CFR 30 requires programs for such issues as employee protection (§30.7); completeness and accuracy of information (§30.9); deliberate misconduct (§30.10); exemption requests (§30.11); notification (§30.50); record keeping of material transfers and reporting (§§30.51 and 30.55); support on NRC inspections (§50.52) and requested tests (§50.53); and enforcement (§30.61).</p>

Table 5.a Title 10 CFR Summary-Level Comparisons of Technical/Safety Issues

Part	Title	SNFP Path Forward Discussion	NRC safety issue(s)
36	<i>Licenses and Radiation Safety Requirements for Irradiators</i>	<p>Title 10, Part 36 is not directly applicable to SNFP Path Forward activities, however, it does have some analogous conditions with the new SNFP Path Forward facilities. For example, both have radioactive material that requires cooling and control of releases (irradiators with sealed sources and the new SNFP Path Forward facilities with irradiated nuclear fuel). A number of the design requirements associated with an irradiator are due to the very high direct radiation levels within the radiation chamber, and as such are not applicable to new SNFP Path Forward facilities. Such requirements include heating of the shield walls by the radiation sources and a number of detailed items associated with access control systems, including interlocks and radiation monitor placement. Similarly, the 10CFR36 requirements specific to sealed sources and pool-type irradiators do not apply.</p> <p><u>Conclusions:</u> The requirements, in comparison to 10CFR36, for radiation monitoring, radiation exposure, and fire protection are adequately addressed by: DOE Orders 5480.7A, <i>Fire Protection</i>, 5480.11, <i>Radiation Protection for Occupational Workers</i>, and 6430.1A, <i>General Design Criteria</i>; 10CFR835, <i>Occupational Radiation Protection: the Hanford Site Radiological Control Manual</i>, HSRCM-1; and the discussions in this table for 10CFR20, Table 5.b for §50.48 and Appendix R, and Table 5.c for §§72.104, 72.106 and 72.126. Note that DOE Orders 5480.7A and 6430.1A both reference NFPA for detailed fire protection requirements. In addition, the design requirements for earthquakes are discussed in Table 5.c, §§72.92 and 72.102, and American Concrete Institute Standard ACI 318, <i>Building Code Requirements for Reinforced Concrete</i>, is referenced (without mention of an issue date or revision number) for shielding and structural design in DOE Order 6430.1A.</p> <p><u>Actions for Consideration:</u> Refer to this table, 10CFR20, Table 5.b, §50.48 and Appendix R, and Table 5.c, §§72.92, 72.104, and 72.106.</p>	<p>This part contains requirements for the issuance of a license authorizing the use of sealed sources containing radioactive materials in irradiators used to irradiate objects or materials using gamma radiation. It also contains radiation safety requirements for operating irradiators.</p>
40	<i>Domestic Licensing of Source Material</i>	<p>For DOE facilities, the applicable 10CFR40 requirements and the related DOE rules and orders are: employee protection (10CFR70B and RLID 5480.29); completeness and accuracy of information (§820.11); deliberate misconduct (10CFR820, Subpart F); exemption requests (10CFR820, Subpart E); notification (§5000.3B); record keeping of material transfers and reporting (§633.3B); support for inspections and requested tests (§482.1B); and enforcement (10CFR820, Subpart B).</p> <p><u>Conclusions:</u> In Table 4, the conclusion is reached that with WMC site-wide programs in place, no additional review is required for employee protection (§50.7); completeness and accuracy of information (§50.9); deliberate misconduct (§50.5); and exemption requests (§50.12).</p> <p>As programs are also in place, as listed above, for record keeping of material transfers, support for inspections and requested tests, and enforcement response, no SNFP Path Forward action is required and no additional evaluation of 10CFR40 is required as a result of this equivalency comparison. It should also be noted that these latter activities have no impact on design or construction of new SNFP Path Forward facilities.</p> <p><u>Actions for Consideration:</u> None</p>	<p>Title 10, CFR 40 establishes requirements similar to 10CFR30 but applies to source and byproduct material (see <i>Definitions</i>, Section 8.2).</p> <p>Application of 10CFR40 for power reactors with operating licenses is similar to that discussed above for 10CFR30. That is, they do not usually develop special programs to deal with source material they may possess as the programs required for plant operation provide adequate controls for this material.</p> <p>Title 10, CFR 40 requires programs for such issues as employee protection (§40.7); completeness and accuracy of information (§40.9); deliberate misconduct (§40.10); exemption requests (§40.14); notification (§40.60); record keeping and reporting of material transfers (§§40.61 and 40.64); support on NRC inspections (§40.62) and requested tests (§40.63); and enforcement (§40.81).</p>

Table 5.a Title 10 CFR Summary-level Comparisons of Technical/Safety Issues

Part	Title	SNFP Path Forward Discussion	NRC safety issue(s)
50	<i>Domestic Licensing of Production and Utilization Facilities</i>	See detailed evaluation for 10CFR50 provided in Table 5.b.	See Table 5.b.
51	<i>Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions</i>	<p>For DOE activities, the need to comply with the National Environmental Policy Act (NEPA) is implemented by 10CFR1021, <i>National Environmental Policy Act Implementation Procedures</i>, and DOE Order 5440 1E, <i>National Environmental Policy Act Compliance Program</i>.</p> <p><u>Conclusions:</u> As stated in Section 1.0, review of environmental issues is not part of the equivalency review process. As such, no additional analysis of 10CFR51 is provided as part of this equivalency comparison.</p> <p><u>Actions for Consideration:</u> None.</p>	This part provides for implementation of NEPA relating to NRC domestic licensing and related regulatory functions.
60	<i>Disposal of High-Level Radioactive Wastes in Geologic Repositories</i>	<p>Section 60.135 provides criteria for the waste package design, including constraints on the contents of the package.</p> <p><u>Conclusions:</u> It is not required within the scope of the SNFP Path Forward to process and store the spent nuclear fuel in a manner suitable for 10CFR60 storage. As such, the design requirements of §60.135 are not applicable to new SNFP Path Forward activities.</p> <p><u>Actions for Consideration:</u> None.</p>	This part ensures that chemical, physical, and nuclear properties of the waste package and its interactions with the emplacement environment do not compromise the function of the waste package or the repository.

Table 5.a Title 10 CFR Summary-Level Comparisons of Technical/Safety Issues

Part	Title	SNMP Path Forward Discussion	NRC safety issue(s)
70	<i>Domestic Licensing of Special Nuclear Material</i>	<p>For DOE facilities, the applicable 10CFR70 requirement and the related DOE rules and orders are: employee protection (10CFR70.8 and RLID 5480.29); completeness and accuracy of information (10CFR820.11); deliberate misconduct (10CFR820, Subpart F); exemption requests (10CFR820, Subpart E); criticality control (5480.24); material control and accountability (5633.3B); support for inspections and requested tests (5482.1B); response to enforcement action (10CFR820, Subpart B); notification (5000.3B); training (5480.18B, 5480.20A, and 5630.15); procedures (5633.3B); protection of public health and safety (5480.1B); physical protection against threats, theft and sabotage (5633.3B); emergency planning (5500.1B, 5500.2B, and 5500.3A); and, analysis and documentation necessary for the Safety Analysis Report (5480.23).</p> <p>Conclusions: In Table 4 the conclusion is reached that with WHC site-wide programs in place, no additional review is required for employee protection (5500.1B), completeness and accuracy of information (5500.2B), deliberate misconduct (5500.3A), and exemption requests (5500.12).</p> <p>Programs are also in place, as listed above, for other requirements contained in 10CFR70 including criticality control, special nuclear material measurement control, support for inspections, enforcement response, notification, training, procedures, protection of public health and safety, security planning, transportation security, emergency planning, and safety analysis reports.</p> <p>Consideration of special hardware requirements during the design and construction of new ISFSI facilities is also necessary to meet 10CFR70 requirements. Design criteria for a number of SNFP Path Forward structures, systems and components are provided in Division 13, <i>Special Facilities</i>, of DOE Order 6430.1A, <i>General Design Criteria</i>. Design criteria related to 10CFR70 requirements include:</p> <ul style="list-style-type: none"> • Criticality control, monitoring and alarm systems - when criticality potential exists, nuclear criticality controls, including equipment and procedures, must be established. DOE design criteria are contained in DOE Order 6430.1A in Section 1300.4, <i>Nuclear Criticality Safety</i>, for exercising control over both the quantity and distribution of all fissile material and other materials capable of sustaining a chain reaction, including monitoring and alarm systems. <p>DOE Order 5480.24, <i>Nuclear Criticality Safety</i>, and ANSI/ANS-8.3-1986, <i>Criticality Accident Alarm System</i>, provide additional detail and requirements. DOE Order 5480.24 invokes ANSI/ANS-8.3, except paragraph 4.2. Coverage (refer to Table 6 a, Regulatory Guide 8.12, <i>Criticality Accident Alarm Systems</i>, for additional information). The requirements of 570.24(a)(1) are addressed in ANSI/ANS-8.3, paragraph 5.6 and WHC-CH-4-29, <i>Nuclear Criticality Safety Manual</i>, Section 11.0, <i>Criticality Alarm System</i>, Revision 0, Change 2, (May 23 1994), paragraph 5.3.1, which addresses redundant detectors.</p> <ul style="list-style-type: none"> • Special nuclear material (SNM) accountability - provides the capability to detect, deter, and assist in the prevention of authorized use or removal of SNM. DOE Order 6430.1A in Section 1300.10.3, <i>Material Control and Accountability (MC&A) Safeguards</i>, provides design criteria to provide such control including boundaries within facilities, storage areas and vaults, tamper material, monitoring access to boundaries, storage areas and vaults, tamper 	<p>Title 10, CFR 70 provides regulations for procedures and criteria to receive title to, own, acquire, deliver, receive, possess, use, and initially transfer special nuclear fuel. Commercial nuclear utilities need a specific 10CFR70 license prior to operation to allow receipt of the initial core. When the operating license is issued, it usually incorporates the 10CFR70 license.</p> <p>Title 10, CFR 70 requires programs for holders of special nuclear material for such issues as: employee protection (570.7); completeness and accuracy of information (570.9); deliberate misconduct (570.10); exemption requests (570.14); criticality monitoring (570.24); material control and accountability (570.41, 70.58 and 70.51); support on NRC inspections (570.55) and requested tests (570.56); enforcement response (570.71); and, notification (570.50 and 70.52).</p> <p>In addition, a request for a specific special nuclear material license must address: training, protection of public health and safety; physical protection against threats, theft, and sabotage; and, emergency planning (570.22).</p> <p>It should also be noted that 10CFR70 requires licensing information for items unique to special nuclear fuel such as: handling devices; working areas; shields; measuring and monitoring instruments; devices for the disposal of radioactive effluent and wastes; storage facility control; criticality accident alarm systems; criticality control; criticality monitoring; special nuclear material safeguards; and other similar items. The software items include: procedures to avoid accidental criticality; procedures for personnel monitoring and waste disposal; and post-criticality accident emergency procedures.</p> <p>Separate from the requirements of 10CFR70, the NRC has published guidance in the Federal Register (57FR35607) on fire protection for fuel cycle facilities in the form of a Technical Position. The Technical Position serves as a source of information for applicants and licensees on the features of equipment, facilities, and procedures that can be used to provide reasonable assurance of fire safety for fuel cycle facilities.</p> <p>In addition, as discussed in Section 6.0 of this report, a review was performed of a 10CFR70 application for license renewal. The intent of this review, in that 10CFR70 lacks definitive requirements, was to identify requirements imposed during the licensing process that are not called out in the regulations. The review identified a number of areas of NRC interest, including chemical safety as it may impact nuclear safety, quality assurance, criticality safety, safety analyses, fire safety issues and fire rating standards. The review found no additional Actions for Consideration.</p>

Table 5.a Title 10 CFR Summary-Level Comparisons of Technical/Safety Issues

Part	Title	SNFP Path Forward Discussion	NRC safety issue(s)
70 cont'd	<i>Domestic Licensing of Special Nuclear Material</i>	<p>indicating devices, measuring and monitoring systems, and other items relevant to SNM accountability. DOE Order 5633.3B, <i>Control and Accountability of Nuclear Materials</i>, also provides basic requirements for designs to provide the necessary protection of SNM. Physical protection against threats, theft and sabotage is also covered in DOE Order 6430.1A, Section 1300.10, <i>Physical Protection, Material Safeguards, and Storage of Special Nuclear Material</i>. This includes use of barriers and detection and communications systems.</p> <ul style="list-style-type: none"> • Radiation Protection - provides for minimizing personnel exposures to external and internal radiological hazards, provides adequate radiation monitoring and alarm systems and provides space for health physics activities. Primary radiation protection is provided through use of engineered controls such as confinement, ventilation, remote handling, equipment layout, personnel decontamination facilities and shielding. Design criteria are provided by Section 1300-6, <i>Radiation Protection</i>, of DOE Order 6430.1A. • Devices for disposal of radioactive effluent and wastes - wastes may include both radioactive and nonradioactive materials and may be in the form of liquid or airborne effluents or solids. DOE Order 6430.1A, Sections 1300-8, <i>Waste Management</i>, and 1300-9, <i>Effluent Control and Monitoring</i>, provide design criteria for reduction of wastes and control of reducing effluents to ALARA levels. This includes minimizing production of wastes, minimizing the mixing of wastes, volume reduction equipment, waste segregation, effluent monitoring, ground water monitoring and other systems. Specific DOE design requirements are contained in DOE Orders 5820.2A, <i>Radioactive Waste Management</i>, 5480.1B, <i>Environment, Safety, and Health Program for Department of Energy Operations</i>, 5480.11, <i>Radiation Protection for Occupational Workers</i>, and 5400.1, <i>General Environmental Protection Program</i>. <p>◁ Effluent monitoring - §70.59 requires the semiannual submittal of a report to the NRC that specifies the quantity of each principal radionuclides released to unrestricted areas in liquid and gaseous effluents. Collecting and recording the data for this reporting requires that the appropriate monitoring equipment be installed and operating during those times that the potential for release of radionuclides may occur.</p> <p>Regarding the NRC Technical Position on fire protection for fuel cycle facilities (see next column, previous page), relative to design and construction features that might be applicable to new SNFP Path Forward facilities, this Technical Position provides guidance in the areas of:</p> <ul style="list-style-type: none"> • Fire prevention (e.g., use of noncombustible or limited combustible materials, lightning protection, electrical installations to the appropriate NFPA codes) ◁ Fire detection (with consideration given to infrequent personnel presence and moderate or high fire hazards) ◁ Containment and suppression of fires (e.g., damper actuation, provisions for fire areas by structural barriers with appropriate fire ratings and barrier penetration protection, and water supply) 	

Table 5.a Title 10 CFR Summary-Level Comparisons of Technical/Safety Issues

Part	Title	SNFP Path Forward Discussion	NRC safety issue(s)
70 cont'd	<i>Domestic Licensing of Special Nuclear Material</i>	<p>◦ Coping with a fire (e.g., personnel egress and drainage provisions).</p> <p>The Technical Position also suggests the performance and periodic updating of a fire hazard analysis.</p> <p>These fire protection guidance in the Technical Position are provided in DOE Orders 5480.7A, <i>Fire Protection</i>, and 6430.1A, <i>General Design Criteria</i>, as follows:</p> <p>◦ Fire prevention (noncombustible construction, 5480.7A, Sections 9 and 9 b(6) and 6430.1A, 0110-6.2; lightning protection, 6430.1A, 1630-5; and use of NFPA codes for electrical installations, 5480.7A, Section 9 and 6430.1A, 1605-1)</p> <p>◦ Fire detection (5480.7A, Section 9 b(3) and 6430.1A, 1530-8.3.1)</p> <p>◦ Containment and suppression (damper actuation, 6430.1A, 1530-99.0; fire areas, 5480.7A, Section 9 b(1) and 6430.1A, 0110-6.3; water supply 5480.7A, Section 9 b(8) and 6430.1A, 1530-3 and 1530-9)</p> <p>◦ Coping with a fire (personnel egress, 5480.7A, Section 9 b(2) and drainage provisions, Section 9 b(10)).</p> <p>The need for a fire hazard analysis is established in DOE Order 5480.7A.</p> <p>Based on review of the above regulations, NRC guidance and DOE orders, it is concluded that, with the exception of the need to provide for effluent monitoring as required by §70.59, the design and construction provisions of 10CFR70 are sufficiently covered in the referenced DOE orders and no additional requirements were identified as being needed for equivalency.</p> <p><u>Actions for Consideration:</u> Ensure that the AEs for the new SNFP Path Forward facilities review the effluent monitoring requirements of, for example 10CFR20, 10CFR70.59, and 10CFR835, to provide the necessary monitoring instrumentation.</p>	
71	<i>Packaging and Transportation of Radioactive Material</i>	<p>Specific equivalency for transport of the MCOs in casks from K Basins to the new SNFP Path Forward facilities and between these facilities will be documented in the Package Design Criteria which is used as input for the Safety Analysis Report for Packaging (SARP).</p> <p><u>Conclusions:</u> Shipping casks are not considered as "new SNFP Path Forward facilities" and, as such, are not within the scope of this equivalency review. For these reasons, no additional evaluation of 10CFR71 is provided as part of this equivalency comparison.</p> <p><u>Actions for Consideration:</u> None.</p>	This part provides for high integrity packaging and offsite shipping of radioactive materials.
72	<i>Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste</i>	See detailed evaluation for 10CFR72 provided in Table 5.c.	This part ensures that for an ISFSI, the operating procedures, facility and equipment, and the licensee's capability will allow for safe operation of the facility.

Table 5.a Title 10 CFR Summary-Level Comparisons of Technical/Safety Issues

Part	Title	SNFP Path Forward Discussion	NRC safety issue(s)
73	<i>Physical Protection of Plants and Materials</i>	<p>For DOE facilities, the requirements for physical protection of plants and materials are provided by DOE Orders 5633 3B, <i>Control and Accountability of Nuclear Materials</i>, 5632 1C, <i>Protection and Control of Safeguards and Security Interests</i>, 5632 7A, <i>Protective Force Program; Responsibilities and Authorities</i>, and 6430 1A, <i>General Design Criteria</i>.</p> <p>Conclusions. Sections 73.26, 73.37, 73.72, and 73.74 contain provisions relevant to transportation of special nuclear fuel which are not applicable to SNFP Path forward activities. Several other sections of 10CFR73 are also not applicable to SNFP Path Forward design and construction activities.</p> <p>Key provisions which remain in 10CFR73 after exempt provisions are eliminated include: protection of safeguards information (§73.21); prohibitions (§73.24); physical protection at fixed sites (§73.40); requirements for physical protection of licensed activities (§73.50); Appendix B, <i>General Criteria for Security Personnel</i>; and, Appendix C, <i>Licensee Safeguards Contingency Plans</i>.</p> <p>Design and construction related items from §73.50 for new SNFP Path Forward facilities to provide physical protection of plants and materials include use of physical barriers, access systems, detection systems, and communication systems.</p> <p>Design criteria to provide physical protection of special nuclear material are provided in Division 1300-10, <i>Physical Protection, Material Safeguards, and Storage of Special Nuclear Material</i>, of DOE Order 6430 1A. These include:</p> <ul style="list-style-type: none"> • A system of barriers or other impediments to delay, channel personnel, or deny access to SNM or vital areas. • A detection system to provide the capability to detect an adversary action or anomalous behavior. • An assessment system to provide the capability to assess the nature of the adversary action. • A communications system to provide the capability to communicate to response forces and other personnel. <p>In addition, Section 0283, <i>Physical Protection</i>, of DOE Order 6430 1A contains design criteria for other physical protection features for plants and materials which are relevant to new SNFP Path Forward activities. These include security fencing, patrol roads and walkways, gates, entry control points, vehicle barriers, security inspector posts and guard towers, and lighting.</p>	<p>This part provides for physical protection against acts of radiological sabotage and to prevent the theft or diversion of special nuclear material at fixed sites, in transit, and in plants where special nuclear material is used. Title 10, CFR 73 provides the physical protection requirements for ISFSIs and is referenced in §72.180.</p> <p>Design and construction provisions are contained in §73.50.</p>

Table 5.a Title 10 CFR Summary-Level Comparisons of Technical/Safety Issues

Part	Title	SNFP Path Forward Discussion	NRC safety issue(s)
73 cont'd	<i>Physical Protection of Plants and Materials</i>	<p>The requirements of DOE Order 6430.1A and the other DOE orders discussed above are equivalent to 10CFR73 requirements except that 10CFR73 requires isolation zones (see <i>Definitions</i>, Section 8.2) and clear areas between barriers be illuminated to a minimum of 0.2 foot candles, whereas DOE Order 6430.1A requires adequate illumination to detect adversaries and reveal unauthorized persons. SECY-95-104, <i>Safeguards for Spent Nuclear Fuel or High-level Radioactive Waste</i>, (See Table 6.b) which, upon approval, is intended to provide NRC safeguards requirements for ISFSIs, requires adequate illumination between access barriers and within isolation zones to permit assessment of unauthorized penetration of or activities within." DOE Order 6430.1A, therefore meets the NRC equivalency with credit taken for the proposed revision to 10CFR73 as described in the SECY document.</p> <p><u>Actions for Consideration:</u> None</p>	
74	<i>Material Control and Accounting of Special Nuclear Material</i>	<p>For DOE facilities, the requirements for control and accountability of special nuclear material are provided for by DOE Order 5633.3B, <i>Control and Accountability of Nuclear Material</i>.</p> <p><u>Conclusions:</u> DOE has established equivalent requirements for special nuclear material control and accountability. This issue has no impact on design or construction of new SNFP Path Forward facilities. However, in regards to design impacts, this needs to be re-evaluated upon final resolution of accountability in the event canister sludge remains with the fuel (Wiborg 1995), which would affect the Integrated Process Strategy.</p> <p><u>Actions for Consideration:</u> None</p>	This part provides for maintaining control and accounting of special nuclear material. Requirements specific to ISFSIs are provided in 10CFR72, Subpart D.
75	<i>Safeguards on Nuclear Material-Implementation of US/IAEA Agreement</i>	<p>DOE Orders 1270.2B, <i>Safeguards Agreement with the International Atomic Energy Agency (IAEA)</i> and 5633.3B, <i>Control and Accountability of Nuclear Materials</i>, prescribe policies and responsibilities for compliance with this agreement.</p> <p><u>Conclusions:</u> DOE Order 1270.2B, <i>Safeguards Agreement with the International Atomic Energy Agency</i>, provides adequate control to implement the IAEA agreement and 10CFR75 establishes no requirements that would impact design or construction of new SNFP Path Forward facilities. As such, no SNFP Path Forward action is required and no additional evaluation or 10CFR75 is provided as part of this equivalency comparison.</p> <p><u>Actions for Consideration:</u> None</p>	This part establishes a system of nuclear material accounting and nuclear material control to implement the "U.S. Agency Procedures for the Implementation of the Agreement Between the United States of America and the International Atomic Energy Agency for the Application of Safeguards in the United States of America with Attached Protocol". Although 10CFR75 requires the submittal of design information to the NRC upon request, the part does not specify design requirements.

Table 5.a Title 10 CFR Summary-Level Comparisons of Technical/Safety Issues

Part	Title	SNFP Path Forward Discussion	NRC safety issue(s)
100	<i>Reactor Siting Criteria</i>	<p>Title 10, CFR 100 provides site selection guidance relative to</p> <ul style="list-style-type: none"> • Determining the exclusion area boundary, low population zone and the population center distance • The two-hour accident dose limit at the exclusion area boundary and the 30-day dose at the low population zone. • Seismology and geology <p>Conclusions: For 10CFR72 siting, these issues are addressed in an analogous manner as follows:</p> <ul style="list-style-type: none"> • Site boundary - The distance to the controlled area boundary must be at least 100 meters (§72.106(b)). • Accident dose limit - The accident dose limit is 5 rem to the whole body, or any organ, at the controlled area boundary. This boundary must be at least 100 meters (§72.106(b)). • Seismology and geology - For ISFSI sites west of the Rocky Mountain Front seismicity must be evaluated by the techniques of 10CFR100, Appendix A (10CFR72 102) and the design earthquake shall be the safe shutdown earthquake (SSE) for a nuclear power plant. <p>The three issues are addressed in more detail in Table 5.c, in response to §§72.92, 72.102, and 72.106.</p> <p>As the Conditioning Facility and the other new SNFP Path Forward facilities represent offsite radiological risks that are comparable to those of the CSB, the siting criteria of 10CFR72 are also appropriate for these other facilities.</p> <p><u>Actions for Consideration:</u> None. (See response provided for §§72.92 and 72.106 in Table 5.c.)</p>	<p>This part establishes guidance to the NRC staff relative to the site suitability issues of potential radiological risk to the nearby population due to an accident and seismic risk to the facility.</p> <p>NRC radiological risk criteria for ISFSIs are provided in §72.106(b) where the limit is 5 rem WBD, or for any organ, and the minimum distance to the controlled boundary is 100 meters.</p>

Table 5.b Title 10 CFR Summary-Level Comparisons of Part 50 Technical/Safety Issues

Sect/Appendix	Title	SNFP Path Forward Discussion	NRC Safety Issue(s)
50.1	<i>Basis, purpose, and procedures applicable</i>	Title 10, CFR50 provides for the licensing of production and utilization facilities. It also establishes requirements for individuals and contractors who supply materials and services for such facilities.	This section does not contain information relevant to NRC safety equivalency for the design and construction of new SNFP Path Forward facilities. Such sections are noted as NA elsewhere in Table 5.b.
50.2	<i>Definitions</i>	Provides definitions for terms used in 10CFR50.	This section does not contain any definition of particular importance to the SNFP Path Forward.
50.3	<i>Interpretations</i>	None.	NA
50.4	<i>Written communications</i>	None.	NA
50.5	<i>Deliberate misconduct</i>	Refer to discussion provided in Table 4 for §50.5.	NA (Also refer to Table 4, §50.5 discussion.)
50.6	Reserved by the NRC	None.	NA
50.7	<i>Employee protection</i>	Refer to discussion provided in Table 4 for §50.7.	NA (Also refer to Table 4, §50.7 discussion.)
50.8	<i>Information collection requirements OMB approval</i>	None.	NA
50.9	<i>Completeness and accuracy of information</i>	Refer to discussion provided in Table 4 for §50.9.	NA (Also refer to Table 4, §50.9 discussion.)
50.10	<i>License required</i>	None.	NA
50.11	<i>Exceptions and exemptions for licensing requirements</i>	This section provides exceptions and exemptions from the requirements of 10CFR50 for specific facilities and functions.	NA
50.12	<i>Specific exemptions</i>	Refer to discussion provided in Table 4 for §50.12.	NA (Also refer to Table 4, §50.12 discussion.)
50.13	<i>Attacks and destructive acts by enemies of the United States; and defense activities</i>	None.	NA
50.14-50.19	Reserved by the NRC	None.	NA
50.20	<i>Two classes of licenses</i>	None.	NA
50.21	<i>Class 104 licensees: for medical therapy and research and development facilities</i>	None.	NA
50.22	<i>Class 103 licensees: for commercial and industrial facilities</i>	None.	NA

Table 5.b Title 10 CFR Summary-Level Comparisons of Part 50 Technical/Safety Issues

Sect/Appendix	Title	SNFP Path Forward Discussion	NRC Safety Issue(s)
50.23	<i>Construction permits</i>	None	NA
50.24-50.29	Reserved by the NRC	None	NA
50.30	<i>Filing of application for licenses; oath or affirmation</i>	None	NA
50.31	<i>Combining applications</i>	None	NA
50.32	<i>Elimination of repetition</i>	None	NA
50.33	<i>Contents of applications: general information</i>	The required information does not deal with the details of the facility construction and design other than financial qualifications. Also see response provided in this Table for 10CFR50, Appendix C.	NA
50.33a	<i>Information requested by the Attorney General for antitrust review</i>	See response provided in this Table for 10CFR50, Appendix L.	NA

Table 5.b Title 10 CFR Summary-Level Comparisons of Part 50 Technical/Safety Issues

Sect/Appendix	Title	SNFP Path Forward Discussion	NRC Safety Issue(s)
50.34	<i>Contents of an application, technical information</i>	<p>The SARs for the CSB, Conditioning Facility, and possibly other new SNFP Path Forward facilities are to be written to the format and content requirements of DOE Order 5480.23, <i>Nuclear Safety Analysis Reports</i>, and DOE-STD-3009-94, <i>Preparation Guide for U. S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports</i>. However, Regulatory Guide 3.48, Revision 1, <i>Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation (Dry Storage)</i>, has been reviewed to determine additional material that would be appropriate for the CSB SAR in the context of NRC requirements. The results of this review are documented in Table 5.c, in response to §72.24.</p> <p><u>Conclusions:</u> The results of the review of Regulatory Guide 3.48 should be considered for the SAR for the CSB. In addition, the latest revision of Regulatory Guide 3.26, <i>Standard Format and Content of Safety Analysis for Fuel Reprocessing Plants</i>, should be reviewed for the new SNFP Path Forward facilities other than the CSB to the extent that any conditioning process associated with these other facilities might resemble those of a reprocessing plant. This review, however, should be delayed until the conditioning processes are better defined.</p> <p><u>Actions for Consideration:</u> Consider including in the SNFP Path Forward Safety Management Plan the requirement to augment the CSB SAR with the information identified in Table 5.c, in response to §72.24 and Regulatory Guide 3.48, Revision 1, that is unique to storage of spent nuclear fuel. In addition, also consider including in the Safety Management Plan the requirement to augment the SAR or other safety documentation for new SNFP Path Forward facilities other than the CSB with the information required by Regulatory Guide 3.26, Revision 0 that is unique to the conditioning processes.</p>	This section defines information required to support a timely NRC staff review of the application.
50.34a	<i>Design objectives to control releases of radionuclide material</i>	<p>The NRC requirement to submit a report does not in itself impact design and construction activities on new SNFP Path Forward facilities. The issues of effluent control and monitoring, and environmental surveillance are addressed for DOE facilities in DOE Order 6430.1A, <i>General Design Criteria</i>, Section 1300-9, <i>Effluent Control and Monitoring</i>. The design objectives for controlling public exposure as low as reasonably achievable are contained in DOE Order 5400.5, <i>Radiation Protection of the Public and the Environment</i>. Requirements for the management of solid waste are contained in DOE Order 5820.2A, <i>Radioactive Waste Management</i>.</p> <p><u>Conclusions:</u> The requirements of DOE and NRC are equivalent to §50.34a.</p> <p><u>Actions for Consideration:</u> None.</p>	<p>Section 50.34a requires that the operator of a nuclear power plant submit to the NRC:</p> <ul style="list-style-type: none"> • A description of and procedures for equipment to control gaseous and liquid effluents. • An estimate of each principal radionuclide expected to be released to unrestricted areas during operation. <p>Included in this submittal the utility must also identify design objectives and other means to be employed to achieve ALARA relative to radioactive material releases.</p> <p>In addition, the submittal must discuss provisions for the handling and shipping offsite of solid waste.</p>

Table 5.b Title 10 CFR Summary-Level Comparisons of Part 50 Technical/Safety Issues

Sect/Appendix	Title	SNFP Path Forward Discussion	NRC Safety Issue(s)
50.35	<i>Issuance of construction permits</i>	This section allows for the issuance of a construction permit if not all of the required technical information is initially supplied. It also defines the limitations of a construction permit.	NA
50.36	<i>Technical specifications</i>	<p>For DOE contractor operated nuclear facilities, the requirement to provide for control over plant features whose condition may degrade or become nonconforming is established by DOE Order 5480.22, <i>Technical Safety Requirements</i>. This order requires, in Section 9, that the TSRs include safety limits, limiting control settings, limiting conditions for operation, surveillance requirements, administrative controls, design features, and bases. DOE Order 5480.22, requires TSRs in the same format and content as the technical specifications required of §50.36 of NRC licensees. If issued, §830.320, <i>Technical Safety Requirements</i>, will codify these requirements.</p> <p>As §50.36 does not direct that any particular technical specification be included in the technical specifications, this section, by itself, does not establish design or construction requirements.</p> <p><u>Conclusions:</u> Section 50.36 does not establish requirements that would impact the design or construction of new SNFP Path Forward facilities. As such, no SNFP Path Forward action is required and no additional evaluation of §50.36 is provided for this equivalency evaluation.</p> <p><u>Actions for Consideration:</u> None.</p>	<p>This part provides increased control over plant features whose degradation or nonconforming condition could result in immediate risk to public health and safety. The technical specifications must include safety limits, limiting safety system settings (for nuclear reactors), limiting control system settings (for fuel reprocessing plants), limiting conditions for operation, surveillance requirements, administrative controls, bases, and design features.</p> <p>The requirements for technical specifications for TSFSIs licensed under 10CFR72 are provided in §72.44.</p>
50.36a	<i>Technical specifications on effluents from power reactors</i>	<p>DOE Order 5480.22, <i>Technical Safety Requirements</i>, does not require TSRs related to effluent releases.</p> <p><u>Conclusions:</u> Specific administrative controls should be considered to address the NRC requirements relative to effluent release technical specifications. The need to establish TSRs for effluent releases does not appear necessary due to current NRC/Industry improved technical specifications.</p> <p><u>Actions for Consideration:</u> Ensure that the AEs for the new SNFP Path Forward facilities review the effluent monitoring requirements of, for example 10CFR20, 10CFR70.59, and 835, to provide the necessary monitoring instrumentation.</p>	<p>Section 50.36a establishes requirements for effluent technical specifications for nuclear power plants for periods of normal operation. These technical specifications are required to contain:</p> <ul style="list-style-type: none"> c The radiological limits outlined in §20.1301. • The requirement for the maintenance and use of procedures developed to satisfy §50.34a requirements. c The requirement for annual reporting of the quantity of each principal radionuclide released to the unrestricted area. <p>As part of the NRC/Industry technical specification improvement program, it is planned to relocate the requirements of §50.36a to another document, such as the Off Site Dose Calculation Manual. Rulemaking is being pursued by the NRC to allow for this change on a generic basis.</p> <p>Section 72.44, <i>License conditions</i>, contains the same three requirements as §50.36a.</p>

Table 5.b Title 10 CFR Summary-Level Comparisons of Part 50 Technical/Safety Issues

Sect./Appendix	Title	SNFP Path Forward Discussion	NRC Safety Issue(s)
50.36b	<i>Environmental conditions</i>	This section provides for a license to contain requirements to protect the environment. It does not specify design or construction features.	NA
50.37	<i>Agreement limiting access to Restricted Data</i>	None.	NA
50.38	<i>Ineligibility of certain applicants</i>	None.	NA
50.39	<i>Public inspection of application</i>	None.	NA
50.40	<i>Common standards</i>	This section defines considerations for the content and issuance of a license. It does not make reference to specific standards relative to design or construction.	NA
50.41	<i>Additional standards for class 104 licenses</i>	This section expands on §50.40 for class 104 licenses (see <i>Definitions</i> , Section 8.2).	NA
50.42	<i>Additional standards for class 103 licenses</i>	This section expands on §50.40 for class 103 licenses (see <i>Definitions</i> , Section 8.2).	NA
50.43	<i>Additional standards and provisions affecting class 103 licenses for commercial power</i>	This section further expands on §50.40 for commercial power reactors relative to antitrust considerations.	NA
50.44	<i>Standards for combustible gas control system in light-water-cooled power reactors</i>	<p>The criteria of §50.44 are very specific to LWRs and, as such, are not directly applicable to the new SNFP Path Forward activities. The criteria of §50.44 concern the need for containment, inerting, and/or hydrogen recombiner capability for BWRs and PMRs. The NRC has not applied any aspect of post-LOCA hydrogen control to in-plant or independent spent fuel storage.</p> <p>However, the need for hydrogen control during fuel retrieval, transportation, wet-fuel staging, and dry-fuel storage is important to the SNFP Path Forward. The understanding of hydrogen generation during fuel retrieval, transportation, and wet-fuel staging is being addressed by the SNFP Path Forward Technology Acquisition Plan (TAP). Also, DOE Order 6430.1A, <i>General Design Criteria</i>, Section 1550-99.0.1, <i>General Ventilation and Off-Gas Criteria</i>, establishes a limit for maximum hydrogen concentration of 4 percent by volume for gas storage and process areas.</p> <p>Conclusions: By the TAP programs, the risks of hydrogen production during the various aspects of the SNFP Path Forward are being addressed.</p> <p><u>Actions for Consideration:</u> None</p>	This section provides for post-LOCA containment hydrogen control.

Table 5.b Title 10 CFR Summary-Level Comparisons of Part 50 Technical/Safety Issues

Sect/Appendix	Title	SNFP Path Forward Discussion	NRC Safety Issue(s)
50.45	Standards for construction permits	This section allows for the issuance of a construction permit if certain requirements of 10CFR50 have been satisfied.	NA
50.46	Acceptance criteria for emergency core cooling systems for light-water reactors	<p>The specific criteria of §50.46 and the requirements for Emergency Core Cooling System (ECCS) models of Appendix K are not directly applicable to the SNFP Path Forward. The NRC has not applied any aspect of ECCS acceptance criteria to 10CFR72 or to in-plant fuel storage (e.g., Regulatory Guide 1.13, <i>Spent Fuel Storage Facility Design Basis</i>, or SRP 9.1.3, <i>Spent Fuel Pool Cooling and Cleanup System</i>).</p> <p>Conclusions: The core cooling aspects of §50.46 have a SNFP Path Forward parallel in the need to maintain cooling of SNF. However, with implementing the SNFP Integrated Process Strategy which dries and conditions fuel (Wiborg 1995), the cooling of the spent fuel during staging and storage is not a meaningful concern in the context of §50.46. The need to establish the MCO cooling requirements is being addressed by the TAP.</p> <p><u>Actions for Consideration:</u> None.</p>	This section provides ECCS design criteria to limit the LOCA peak cladding temperature, maximum cladding oxidation, and maximum hydrogen generation and control into the design to ensure a post-LOCA coolable core geometry and long-term core cooling capability.
50.47	Emergency plans	<p>Emergency planning is implemented by a number of DOE orders which include: §500.18, <i>Emergency Management System</i>; §500.28, <i>Emergency Categories, Classes, and Notification and Reporting Requirements</i>; §500.3A, <i>Planning and Preparedness for Operational Emergencies</i>; and §500.4A, <i>Public Affairs Policy and Planning Requirements for Emergencies</i>.</p> <p>Conclusions: Review of the NRC requirements from §50.47 indicated that the above DOE orders collectively contain equivalent requirements except for the following: 1) DOE does not currently require an equivalent independent review of the Emergency Plan; 2) exposure guidelines consistent with EPA Emergency Worker and Lifesaving Activity Protection Action Guides are not specified in DOE orders; and, 3) no equivalent determinations of both a plume EPZ and an ingestion EPZ are required by DOE orders.</p> <p>However, none of these three items relate to design or construction of new SNFP Path Forward facilities. No SNFP Path Forward action is therefore required for this equivalency evaluation.</p> <p><u>Actions for Consideration:</u> None.</p>	Section 50.47 contains requirements to provide reasonable assurance that adequate protection can and will be taken in the event of a radiological emergency.

Table 5.b Title 10 CFR Summary-Level Comparisons of Part 50 Technical/Safety Issues

Sect/Appendix	Title	SNFP Path Forward Discussion	NRC Safety Issue(s)
50.48 and Appendix R	<i>Fire protection</i>	<p>For SNF storage, there is not an equivalent to the power reactor requirement of achieving hot and cold shutdown. However, there are analogous situations to maintaining cold shutdown and minimizing radioactive releases to the environment. They are maintaining MCO cooling and integrity.</p> <p>The Path Forward Redefined Technical Baseline effort has resulted in recommendations that are important to the fire protection aspect of the equivalency comparison as applied to the maintenance of MCO cooling (Wiborg 1995). These recommendations, some of which have been accepted by DOE-RL (DOE 1995b), would result in the following features important to fire protection:</p> <ul style="list-style-type: none"> c. Fuel and canister sludge will be separated at K Basins. c. Water will be drained from the MCOs and the fuel will be vacuum dried at low temperatures (~50° C) at K Basins. This will allow for shipment and staging of fuel with little free water present. c. The fuel will be hot vacuum dried (at ~300 °C) in the Conditioning Facility prior to interim dry storage with the purpose of removing the remaining free water and most of the chemically bound water. While this process may involve introduction of oxygen during the cool-down phase to oxidize exposed fuel, this process will not be a forced combustion process that would introduce a risk of complete burning of the MCO contents. 	<p>This section and Appendix R provide requirements to 1) decrease the risk of a fire, 2) provide assurance that in the event of a fire, safe shutdown of the plant can be achieved and maintained, and 3) minimize the radioactive release to the environment in the event of a fire.</p> <p>Branch Technical Position (BTP) CMEB 9.5-1, <i>Fire Protection for Nuclear Power Plants</i>, provides additional guidance in the areas of fire hazards analysis, control of combustibles, fire detection and suppression, fire brigades, and safe shutdown capabilities.</p>

Table 5.b Title 10 CFR Summary-Level Comparisons of Part 50 Technical/Safety Issues

Sect/Appendix	Title	SNFP Path Forward Discussion	NRC Safety Issue(s)
50.48 and Appendix R cont'd	<i>Fire protection</i>	<p>• With dry fuel and the subsequent high fuel temperatures allowed, cooling during staging and interim storage can be achieved by a natural circulation system.</p> <p>The importance of these recommendations to fire protection equivalency is that they lead to a CSB that will not be sensitive to loss of MCO cooling from fire. This situation may exist for other new SNFP Path Forward facilities. It is currently expected that natural circulation cooling of the MCOs, using outside air, will provide adequate heat removal during staging and interim storage in the CSB (Flour Daniel 1995).</p> <p>Differences do exist between the fire protection requirements of 10CFR50, Appendix R and DOE Orders 5480.7A, <i>Fire Protection</i>, and 6430.1A, <i>General Design Criteria</i>. They are:</p> <p>• Appendix R requires a three-hour barrier between safe shutdown paths and DOE Orders 5480.7A and 6430.1A, require a two-hour barrier between redundant safety class equipment.</p> <p>• Appendix R requires emergency lighting of at least 8-hour duration in all areas needed for operation of safe shutdown equipment and in access and egress routes. DOE Order 5480.7A relies on the NFPA <i>Life Safety Code</i>, which requires 1½-hour lighting to facilitate egress.</p> <p>• Appendix R establishes more specific requirements for the testing of fire penetration seals than DOE Orders 5480.7A and 6430.1A.</p> <p><u>Conclusions:</u> For the new SNFP Path Forward facilities as currently defined and discussed above, the fire protection requirements of DOE Orders 5480.7A and 6430.1A should provide adequate fire protection in consideration of §§50.48 and 72.122(c).</p> <p><u>Actions for Consideration:</u> Consideration should be given to following the evolving designs of the new SNFP Path Forward facilities to reconfirm that DOE Orders 5480.7A and 6430.1A provide adequate fire protection requirements to achieve nuclear safety equivalency. Aspects of the designs to reconfirm are the use of a passive cooling system for MCO cooling in the CSB and the lack of Safety Class 1 prevention and/or mitigation systems in the CSB. The need for Safety Class 1 systems for the other new SNFP Path Forward facilities remains to be determined. Further, fire protection requirements considered for incorporation into the design of the new SNFP Path Forward facilities should take into account the implementation of §72.122(c) to date for licensed ISFSIs.</p>	

Table 5.b Title 10 CFR Summary-Level Comparisons of Part 50 Technical/Safety Issues

Sect./Appendix	Title	SNRP Path Forward Discussion	NRC Safety Issue(s)
50.49	<i>Environmental qualification of electrical equipment</i>	<p>See attached discussion Environmental Qualification of Electrical Equipment, in Attachment A. A review for relevant or analogous safety concerns was performed of 50.49, Regulatory Guide 1.89, <i>Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants</i>, and Regulatory Guide 1.97, Revision 3, <i>Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident</i>, as compared to related DOE requirements. This included DOE Order 6430.14, <i>General Design Criteria</i>, DOE STD-3009-94, <i>Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports</i>, and NRC-GH-4-46, <i>Safety Classification of Structures, Systems, and Components</i>.</p> <p>Conclusions: Consideration of additional actions are necessary for new SNRP Path Forward facilities to achieve equivalence with the requirements of 50.49 for environmental qualification of Safety Class 1 electrical equipment if such equipment is incorporated into the designs.</p> <p>Actions for Consideration: In order to achieve equivalence, consideration should be given to ensuring that the electrical equipment qualification program includes the elements that are missing from DOE requirements for: 1) Safety Class 1 equipment; 2) non-safety class equipment that could, upon failure, adversely impact Safety Class 1 equipment in performance of their safety function; and 3) certain post-accident monitoring equipment as described in Regulatory Guide 1.97. These elements include the following:</p> <ul style="list-style-type: none"> • The AES for the new SNRP Path Forward facilities need to review 50.49(e)(5) and Regulatory Guide 1.89 during the design processes to determine the aging requirements for electrical equipment. • The testing requirements as they are outlined in 50.49(f)(1-4). 	<p>This section provides for qualification of safety-related electrical equipment that is relied upon to remain functional during and following design bases events for which its mitigating capability is required. Design basis events are defined as conditions of normal operation, including anticipated operational occurrences, design basis accidents, external events, and natural phenomena for which the plant must be designed. The requirements also apply to non-safety equipment that can adversely impact safety functions and certain post-accident monitoring instrumentation. This is to ensure that the reactor coolant pressure boundary is maintained, the capability to achieve safe shutdown is preserved, and that accidents that have consequences comparable to 10CFR100 limits are adequately prevented or mitigated.</p>
50.50	<i>Issuance of licenses and construction permits</i>	This is similar to 50.45	NA
50.51	<i>Duration of license, renewal</i>	None	NA
50.52	<i>Combining licenses</i>	None	NA
50.53	<i>Jurisdictional limitations</i>	None	NA

Table 5.b Title 10 CFR Summary-Level Comparisons of Part 50 Technical/Safety Issues

Sect./Appendix	Title	SNFP Path Forward Discussion	NRC Safety Issue(s)
50.54(a)	<i>Conditions of Licenses (Quality Assurance Provisions)</i>	<p>Section 830.120(b)(3) allows for the contractor to make changes to the approved quality assurance program (QAP) on an annual basis. No explicit statement is made regarding prohibition of implementing a change that could be interpreted as decreasing commitments made in the QAP prior to DOE approval.</p> <p><u>Conclusions:</u> In regards to agency approval, equivalency is not achieved for changes to the quality assurance program which could be interpreted as decreasing commitments previously made in the QAP.</p> <p><u>Actions for Consideration:</u> Before implementation, consider having the DOE-RL review and approve changes to the WAC (SNFP) quality assurance program that could be interpreted as decreasing initial commitments.</p>	<p>Section 50.54(a) contains a prohibition against changes to the quality assurance program as described in the Safety Analysis Report that may reduce the commitments made in the program without NRC approval in advance of implementation of the change. Changes that do not decrease quality assurance commitments are to be submitted for review on an annual basis. Section 50.54(a)(3) applies to operating licenses and §50.55(f)(3) applies to construction permits.</p>
50.54(p)	<i>Conditions of Licenses (Safeguards and Security)</i>	<p>DOE Order 5630.11B, <i>Safeguards and Security Program</i>, provides for an independent assessment performed by the Assistant Secretary for Environment, Safety and Health through the Deputy Assistant Secretary for Security Evaluations. DOE Order 5630.12A, <i>Safeguards and Security Inspection and Assessment Programs</i>, establishes the policies, responsibilities and scope of independent oversight of the safeguards and security program. No frequency of the assessment is stated.</p> <p>DOE Order 5630.11B and associated DOE orders do not make reference to changes to the Safeguards and Security Plan.</p> <p><u>Conclusions:</u> Equivalency between the NRC and DOE is achieved for independent assessments of safeguards and security programs except for the frequency of the assessments and the need to obtain prior approval to program changes that could be interpreted as decreasing commitments previously made. However, as these two aspects of §50.54(p) do not relate to design or construction of new SNFP Path Forward facilities, no SNFP Path Forward action is required and no additional evaluation is provided for this equivalency comparison.</p> <p><u>Actions for Consideration:</u> None.</p>	<p>Section 50.54(p) contains a requirement that the licensee shall provide for a review at least every 12 months of the Safeguards Contingency Plan by individuals independent of both security program management and personnel who have direct responsibility for implementation of the security program. Section 50.54(p)(3) contains a listing of the topics that must be covered by the annual review.</p> <p>Section 50.54(p) contains a prohibition against changes to the Safeguards Contingency Plan that may reduce the commitments made in the program without NRC approval in advance of implementation of the change. Changes which do not decrease the Safeguards Contingency Plan commitments are to be submitted for review every three years.</p> <p>For ISFSIs licensed under 10CFR72, the considerations for changes to the safeguards plan are addressed in §72.44(e) which establishes the same requirements as §50.54(p).</p>

Table 5.b Title 10 CFR Summary-Level Comparisons of Part 50 Technical/Safety Issues

Sect/Appendix	Title	SNFP Path Forward Discussion	NRC Safety Issue(s)
50.54(q)	Conditions of Licenses (Emergency Plans)	<p>DOE Order 5500.1B, <i>Emergency Management System</i>, provides for the contractor to review and make changes to the approved Emergency Plans on an annual basis. No explicit statement is made regarding prohibition of implementing a change that could be interpreted as decreasing commitments made in the Emergency Plan prior to DOE approval.</p> <p>Conclusions: Equivalency is not achieved for changes to the Emergency Plan which could be interpreted as decreasing commitments previously made. However, as this aspect of 50.54(q) does not relate to design or construction of new SNFP Path Forward facilities, no SNFP Path Forward action is required and no additional evaluation is provided for this equivalency comparison.</p> <p>Actions for Consideration: None.</p>	Section 50.54(q) contains a prohibition against changes to the Emergency Plan that may reduce the commitments made in the program without NRC approval in advance of implementation of the change.
50.54(s)	Conditions of Licenses (Emergency Plans)	<p>DOE Order 5500.1B, <i>Emergency Management System</i>, states that when the EPZ extends beyond site boundaries, the DOE will assist state, tribal and local governments in development of emergency plans. This will be coordinated with FEMA, other Federal agencies, and state, tribal and local governments.</p> <p>DOE performs an assessment, not less than annually, of emergency preparedness.</p> <p>Conclusions: Emergency plans are developed by state and local governments with participation by DOE and FEMA. A completely independent assessment is not performed. However, as this aspect of 50.54(s) does not relate to design or construction of new SNFP Path Forward facilities, no SNFP Path Forward action is required and no additional evaluation is provided for this equivalency comparison.</p> <p>Actions for Consideration: None.</p>	<p>Section 50.54(s) states that the radiological emergency response plans of state and local government entities that are wholly or partially within a plume exposure pathway EPZ and plans of state governments wholly or partially within an ingestion pathway EPZ must be submitted to the NRC.</p> <p>The NRC will have a review performed by FEMA as to whether state and local emergency plans are adequate and capable of being implemented and whether the licensee's emergency plans are adequate and capable of being implemented.</p> <p>If the emergency plans do not provide reasonable assurance that adequate protective measures can and will be taken and deficiencies are not corrected within four months, the NRC will determine whether the facility shall be shut down until the deficiencies are remedied.</p>
50.54(t)	Conditions of Licenses (Emergency Plans)	<p>DOE Order 5500.1B, <i>Emergency Management System</i>, provides for an annual assessment of the emergency preparedness of the facility.</p> <p>Conclusions: It is not apparent: 1) that the scope of the annual emergency preparedness assessment encompasses that required of the NRC; and, 2) that the assessments are performed by persons having no direct responsibility for implementation of the program. However, as these aspects of 50.54(t) do not relate to design or construction of new SNFP Path Forward facilities, no SNFP Path Forward action is required and no additional evaluation is provided for this equivalency comparison.</p> <p>Actions for Consideration: None.</p>	<p>Section 50.54(t) states that a licensee shall provide for the development, revision, implementation and maintenance of its emergency preparedness program. The licensee shall provide for a review of its emergency preparedness program at least every 12 months by persons having no direct responsibility for implementation of the emergency preparedness program.</p> <p>The review shall include an evaluation for adequacy of interfaces with state and local governments and of licensee drills, exercises, capabilities, and procedures.</p> <p>The results of the review are documented, reported to the licensee's corporate and plant management and retained for five years. Results involving the adequacy of the interface with state and local governments shall be available to the appropriate state and local governments.</p>

Table 5.b Title 10 CFR Summary-level Comparisons of Part 50 Technical/Safety Issues

Sect/Appendix	Title	SNFP Path Forward Discussion	NRC Safety Issue(s)
50.54(x)	Conditions of licenses (emergency actions)	<p>DOE Order 5480.22, <i>Technical Safety Requirements</i>, contains provisions consistent with 50.54(x). It provides that a contractor may take emergency actions that depart from the approved TSRs when no actions consistent with the TSRs are immediately apparent, and when these actions are needed to protect the public health and safety. Such contractor actions are to be approved, as a minimum, by a certified operator or supervisor certified on that system through an accredited training program.</p> <p>Provisions similar to 50.54(x) for emergency actions are also provided in proposed §830.320(h).</p> <p><u>Conclusions:</u> Equivalency is achieved for emergency actions that are outside of license conditions that must be taken to protect public health and safety. It should also be noted that 50.54(x) does not contain requirements that relate to design or construction of new SNFP Path Forward facilities. As such, no SNFP Path Forward action is required and no additional evaluation is provided for this equivalency comparison.</p> <p><u>Actions for Consideration:</u> None</p>	<p>This section provides for control such that departure from license conditions and technical specifications can only be taken without prior NRC approval if such immediate action is required to protect public health and safety. Approval by a licensed senior operator is required prior to taking the action.</p>
50.55(e)	Conditions of construction permits	<p>DOE Order 5000.38, <i>Occurrence Reporting and Processing of Operations Information</i>, presents requirements for reporting and processing of occurrences related to the operation of DOE owned or operated facilities for occurrences related to safety, health, security, property, operations, or environment up to and including emergencies.</p> <p>The purpose of this program is to keep DOE contractor line management and DOE informed of all events which could: affect the health and safety of the public; seriously impact the intended purpose of DOE facilities; have a noticeable adverse effect on the environment; or, adversely effect national security or the security interests of the DOE. The purpose is also to have a system for determining appropriate corrective action and for ensuring that such action is taken to prevent recurrence.</p> <p>DOE maintains an Occurrence Reporting and Processing System (ORPS) data base containing all unclassified Occurrence Reports to provide for dissemination of the problems to other facilities to allow for any needed corrective action or lessons learned.</p> <p>DOE Order 5000.38 meets the reporting schedules and evaluation results of 50.55(e). The order does not provide the same emphasis on whether a breakdown of the quality assurance program caused or contributed to the occurrence as 50.55(e). However, the root cause evaluation for the occurrence, includes an evaluation of programmatic problems, including administrative controls, which may have caused or contributed to the occurrence. Any breakdown of the quality assurance program would be identified by this evaluation.</p>	<p>In accordance with 50.55(e), each facility holding a construction permit must adopt procedures to evaluate defects and noncompliances during design and construction associated with substantial safety hazards. The evaluation must be submitted as soon as practicable, but within 60 days of discovery, in order to identify a reportable defect or noncompliance that could create a substantial safety hazard, were it to remain uncorrected. Evaluation and reporting of potential defects and noncompliances satisfies the construction permit holders evaluation and notification obligations under 10CFR21.</p> <p>Section 50.55(e) contains a listing of the information that must be evaluated and submitted in a written notification to the NRC. The evaluation must include information related to any information reasonably indicating that the quality assurance program has undergone any significant breakdown.</p> <p>Defects for licensed facilities are covered by 50.72 in Table 4.</p>

Table 5.b Title 10 CFR Summary-Level Comparisons of Part 50 Technical/Safety Issues

Sect/Appendix	Title	SNFP Path Forward Discussion	NRC Safety Issue(s)
50.55(e) cont'd	<i>Conditions of construction permits</i>	<p>Conclusions: Although the Occurrence Reporting System of DOE Order 5000.3B is a different system than 50.55(e), it contains the necessary elements to be considered equivalent.</p> <p>However, it is not clear from review of DOE Order 5000.3B if the system is to be utilized during design and construction. From a historical perspective, it is understood that occurrence reporting was used during construction of the Hanford Waste Vitrification Plant.</p> <p>As such, use of the Occurrence Reporting System would be appropriate regarding nuclear safety equivalency during design and construction. One of the purposes of the 50.55(e) reporting is to allow the NRC to review the corrective actions taken for the occurrence and request additional actions, as appropriate.</p> <p><u>Actions for Consideration:</u> Consideration should be given to employing the Occurrence Reporting System during design and construction of new SNFP Path Forward facilities.</p>	

Table 5.b Title 10 CFR Summary-Level Comparisons of Part 50 Technical/Safety Issues

Sect/Appendix	Title	SNRP Path Forward Discussion	MRC Safety Issue(s)
50.55a	<i>Codes and standards</i>	<p>DOE Order 6430-1A, <i>General Design Criteria</i>, in Section 1540-99.0.1, <i>General Cooling System Criteria</i>, provides cooling system design criteria for safety class items in nonreactor nuclear facilities. It requires that the water system shall be designed to a national piping code (i.e., ASME Boiler and Pressure Vessel Code, Section III or ASME B31.3). Section 1300-3.2, <i>Safety Class Items</i>, of the DOE order specifically requires that safety class items be designed to ASME Section III. It also requires in-service testing or a combination of testing and analysis to demonstrate operability of fluid system components, mechanical equipment, instrumentation, and electrical equipment that are required to operate during and following a Design Basis Earthquake. It also requires performance of initial and periodic hydrostatic tests to verify system integrity, performance of periodic operational tests of system performance, and, in-service inspection of system components.</p> <p>Section 1660-99.0.2, <i>Protection System and Instrumentation and Controls</i>, of DOE Order 6430-1A provides design criteria for electrical safety class items in nonreactor nuclear facilities. It does not commit to IEEE Std 279, <i>Criteria for Protection Systems for Nuclear Power Generating Stations</i> or to its replacement IEEE Std 603, <i>IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations</i>. It does commit to IEEE Std 379, <i>IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems</i>, and IEEE Std 384, <i>IEEE Standard Criteria for Independence of Class II Equipment and Circuits</i>, to provide redundancy and separation for safety class electric systems. It requires a periodic test of safety class electric systems to verify system performance. It also requires testing or a combination of testing and analysis to demonstrate the operability of instrumentation and electrical equipment that are required to operate during and following a Design Basis Earthquake.</p> <p>The applicability of codes and standards described in 50.55a to new SNRP Path Forward facilities will be dependent on the design of the MCO and the new facilities. Section 72.24(c)(4) requires an assessment of applicable codes and standards for inclusion in the Safety Analysis Report. Also, DOE-STD-3009-94, <i>Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports</i>, requires a listing of design codes, standards, regulations, and DOE orders for inclusion in the Safety Analysis Report.</p> <p>A list of mandatory and reference standards is also provided in DOE Order 5480.4, <i>Environmental Protection, Safety, and Health Protection Standards</i>.</p>	<p>Section 50.55a ensures the integrity of ASME Class 1, 2, and 3 systems and components by establishing design, in-service inspection, and pump and valve testing requirements. Section 50.55a also ensures the reliability of plant protective systems by requiring application of IEEE Std 279, <i>Criteria for Protection Systems for Nuclear Power Generating Stations</i>. (IEEE Std 279 has been replaced by IEEE Std 603-1991, <i>IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations</i>.)</p> <p>Section 50.55a contains requirements relative to the following areas:</p> <ul style="list-style-type: none"> • ASME Code Class applicability including Addenda • ASME requirements for the reactor coolant pressure boundary • Requirements for Quality Group B components • Requirements for Quality Group C components • Inservice testing requirements • Inservice inspection requirements • Application of IEEE Std 279 for reactor protection systems <p>Section 50.55a applies to boiling or pressurized water-cooled nuclear power facilities or to a utilization facility (see <i>Definitions</i>, Section 8.2).</p>

Table 5.b Title 10 CFR Summary-Level Comparisons of Part 50 Technical/Safety Issues

Sect/Appendix	Title	SNFP Path Forward Discussion	NRC Safety Issue(s)
50.55a cont'd	Codes and standards	<p>Conclusions: Section 50.55a is applicable to nuclear reactor facilities. Applicability of the codes and standards contained in 50.55a to new SNFP Path Forward facilities is design dependent.</p> <p>Applicable codes and standards, including those described in 50.55a, need to be determined and provided in the Safety Analysis Reports for the new SNFP Path Forward facilities. This is a requirement of DOE-STD-3009-94, DOE Order 6430.1A requires use of ASME Section III for safety class items but is silent on use of IEEE Std 603, IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations, for safety class instrumentation and control systems.</p> <p>Actions for Consideration: Consideration should be given to ensuring the AES for the new SNFP Path Forward facilities apply IEEE Std 603-1991, IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations, to Safety Class I instrumentation and control systems.</p>	
50.56	Conversion of construction permit to license, or amendment of license.	None	NA
50.57	Issuance of operating license	None	NA
50.58	Hearings and report of the Advisory Committee on Reactor Safeguards	None	NA
50.59	Changes, tests and experiments	This section contains no requirements that are applicable to design or construction.	This section applies to a holder of a license authorizing the operation of a production or utilization facility. It provides protective measures against changes, without prior NRC approval, that may increase the probability or consequences of postulated accidents, result in a postulated accident not previously evaluated by the NRC, or decrease the margin of safety as defined in the technical specifications.
50.60 Appendices G and H	Acceptance criteria for fracture prevention measures for light water nuclear power reactors for normal operation	See statements for Appendices G and H provided below.	This section and the appendices provide for protection against brittle fracture of the reactor coolant pressure boundary by the establishment of fracture toughness and material surveillance requirements.
50.61	Fracture toughness requirements for protection against pressurized thermal shock events.	See statement for Appendix G provided below.	

Table 5.b Title 10 CFR Summary-Level Comparisons of Part 50 Technical/Safety Issues

Sect./Appendix	Title	SNFP Path Forward Discussion	NRC Safety Issue(s)
50 62	<i>Anticipated Transients Without Scram</i>	<p>Anticipated Transients without Scram (ATWS) is not related to the SNFP Path Forward as no aspect of the SNFP Path Forward operates in an intentional critical condition from which shutdown must be achieved. The NRC has not applied any aspect of ATWS to in-plant fuel storage (e.g., SRP 9.1.2, Spent Fuel Storage) or independent spent fuel storage (i.e., 10CFR72). Broader considerations for reactivity control (criticality) for the SNFP Path Forward are addressed relative to SRP 9.1.2, in Table 6.b and 10CFR72 equivalency in Table 5.c, in response to §72.124.</p> <p><u>Conclusions:</u> As the prevention and mitigation of ATWS events do not relate to the SNFP Path Forward and issues of criticality control are addressed relative to 10CFR72 equivalency, no additional equivalency evaluation is required.</p> <p><u>Actions for Consideration:</u> None.</p>	<p>This section establishes requirements to decrease the risk of an ATWS event for Light Water Reactors (LWRs) and provides for mitigation features should such an event occur.</p>
50 63	<i>Loss of all alternating current power</i>	<p><u>Conclusions:</u> Station Blackout (SBO) considerations are not directly applicable to the SNFP Path Forward. However, the SNFP Path Forward does have an analogous safety concern to the loss of core cooling for a power plant, the loss of cooling to the MCOs. The discussion provided above for §50.48 explains why the Path Forward Redefined Technical Baseline, the maintenance of MCO cooling within the CSB will not be a critical function. Therefore, the concerns for SBO are not applicable to this facility as currently envisioned. The importance of MCO cooling while the MCOs are located in other SNFP Path Forward facilities has not been established.</p> <p>To confirm the expected design of the CSB and to address the importance of MCO cooling in other facilities, it would be appropriate to ensure that the response of the new SNFP Path Forward facilities be evaluated for a loss of ac power.</p> <p><u>Actions for Consideration:</u> Consider including in the SNFP Path Forward Safety Management Plan or the SAR Preparation Plans for the new SNFP Path Forward facilities the need to evaluate the response of the facilities to a loss of ac power.</p> <p>None.</p>	<p>The SBO Rule requires that each light water reactor be able to withstand a SBO for a specified duration and be able to recover from the event. A SBO is defined as complete loss of ac power to the essential and nonessential switchgear buses. It does not include the loss of ac power to buses fed by station batteries through inverters. The loss of ac may be 1) plant centered; 2) a result of a loss of connection to the grid; 3) or a result of a loss of the grid (e.g., due to severe weather).</p> <p>An important aspect of SBO resolution is that, upon onset of the SBO, a plant is assumed to be operating under normal conditions and at full power. Therefore, loss of power to accident prevention and mitigating features need not be considered unless they are required to cope with conditions arising from the SBO experienced during normal operation.</p>
50 64	<i>Limitation on the use of highly enriched uranium (HEU) in domestic non-power reactors</i>	None.	NA
50 65	<i>Requirements for monitoring the effectiveness of maintenance at nuclear power reactors</i>	<p><u>Conclusions:</u> Section 50.65 does not have impact on the design or construction of new SNFP Path Forward facilities. As such, no SNFP Path Forward action is required and no additional evaluation is provided for this equivalency comparison.</p> <p><u>Actions for Consideration:</u> None.</p>	<p>This section requires each licensee to monitor the performance or condition of safety-related and nonsafety-related structures, systems, and components against licensee-established performance goals. Adjustments are to be made to ensure that the objective of preventing failures through maintenance is balanced against the objective of minimizing unavailability due to monitoring or preventive maintenance. Industry wide operating experience is to be a part of the program.</p>

Table 5.b Title 10 CFR Summary-Level Comparisons of Part 50 Technical/Safety Issues

Sect./Appendix	Title	SNFP Path Forward Discussion	NRC Safety Issue(s)
50.66-50.69	Reserved by the NRC	None.	NA
50.70	Inspections	This section does not establish design or construction requirements. Operational issues are not within the scope of the NRC nuclear safety equivalency activity.	NA
50.71	Maintenance of records, making reports	This section does not establish design or construction requirements. Operational issues are not within the scope of the NRC nuclear safety equivalency activity.	NA
50.72	Immediate notification requirements for operating nuclear power reactors	Refer to discussion provided in Table 4 for §50.72.	NA
50.73	Licensee event report system	Refer to discussion provided in Table 4 for §50.73.	NA
50.74	Notification of change in operator or senior operator status	This section does not establish design or construction requirements. Operational issues are not within the scope of the NRC nuclear safety equivalency activity.	NA
50.75	Reporting and recordkeeping for decommissioning planning	This section does not establish design or construction requirements. Operational issues are not within the scope of the NRC nuclear safety equivalency activity.	NA
50.76-50.77	Reserved by the NRC	None.	NA
50.78	Installation information and verification	This section allows the NRC to request of construction permit holders installation information to implement the US/IAEA agreement. This does not establish design or construction requirements.	NA
50.80	Transfer of license	None.	NA
50.81	Creditor regulations	None.	NA
50.82	Application for termination of license	None.	NA
50.83-50.89	Reserved by the NRC	None.	NA
50.90	Application for amendment of license or construction permit	None.	NA
50.91	Notice for public comment; State consultation	None.	NA
50.92	Issuance of amendment	None.	NA

Table 5.b Title 10 CFR Summary-Level Comparisons of Part 50 Technical/Safety Issues

Sect/Appendix	Title	SNFP Path Forward Discussion	NRC Safety Issue(s)
50.100	<i>Revocation, suspension, modification of licenses and construction permits for cause</i>	None.	NA
50.101	<i>Retaking possession of special nuclear material</i>	None.	NA
50.102	<i>Commission order for operation after revocation</i>	None.	NA
50.103	<i>Suspension and operation in war or national emergency</i>	None.	NA
50.104-50.108	Reserved by the NRC	None.	NA
50.109	<i>Backfitting</i>	This section does not establish design or construction requirements. Operational issues are not within the scope of the NRC nuclear safety equivalency activity.	NA
50.110	<i>Violations</i>	This section addresses the right of the NRC to issue an injunction or court order to prohibit a violation of the Atomic Energy Act and of the right of the NRC to issue civil penalties. For DOE facilities, this is addressed in 10CFR820, Department of Energy Procedural Rules for Nuclear Activities.	NA
50.111	<i>Criminal penalties</i>	See response provided in Table 4 for §50.5.	NA
50.112-50.119	Reserved by the NRC	None.	NA
50.120	<i>Training and qualification of nuclear power plant personnel</i>	This section does not establish design or construction requirements. Operational issues are not within the scope of the NRC nuclear safety equivalency activity.	NA

Table 5.b Title 10 CFR Summary-Level Comparisons of Part 50 Technical/Safety Issues

Sect/Appendix	Title	SNFP Path Forward Discussion	NRC Safety Issue(s)
Part 50, Appendix A	General Design Criteria for Nuclear Power Plants	<p><u>Conclusions:</u> The General Design Criteria (GDC) applicable to the CSB are provided in 10CFR72, Subpart F, <i>General Design Criteria</i> as addressed in Table 5 c. An evaluation of the application of Appendix A to the Conditioning Facility and other new SNFP Path Forward facilities is provided in Attachment A to this report.</p> <p>In the attachment the conclusion is reached that at this stage in the accident analyses and design of the new SNFP Path Forward facilities other than the CSB, conclusions cannot be reached as to whether or not existing DOE and WMC design criteria are adequate for the other facilities such that NRC nuclear safety equivalency is achieved without application of the criteria of 10CFR50, Appendix A.</p> <p><u>Actions for Consideration:</u> During the final design of the new SNFP Path Forward facilities other than the CSB, consideration should be given to having the AEs review 1) the conclusions of the general design criteria discussion of Attachment A to this report, and the general design criteria of 10CFR50, Appendix A to determine if NRC nuclear safety equivalency is achieved without application of any of the criteria of 10CFR50, Appendix A.</p>	<p>The General Design Criteria were added to 10CFR50 in 1971 to establish the minimum requirements for the principal design criteria for LWRs similar in design and location to plants for which construction permits had been issued.</p> <p>The GDC consists of the following six major categories:</p> <ol style="list-style-type: none"> I. Overall Requirements. II. Protection by Multiple Fission Product Barriers. III. Protection and Reactivity Control Systems. IV. Fluid Systems. V. Reactor Containment. VI. Fuel and Radioactivity Control.
Part 50, Appendix B	Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants	<p>See the attached Quality Assurance evaluation in Attachment A. A review of 10CFR50, Appendix B, and 10CFR72, Subpart G, <i>Quality Assurance</i>, found these two regulations equivalent, and identical in wording in many cases. The nuclear safety equivalency comparison was therefore only performed against 10CFR72, Subpart G, due to its direct application to spent nuclear fuel storage installations.</p> <p>Section 830.120, <i>Quality Assurance Requirements</i>, provides DOE requirements for ensuring that structures, systems, and components in nuclear facilities will perform satisfactorily in service.</p> <p><u>Conclusions:</u> Section 830.120 provides quality assurance requirements that are nearly equivalent to those provided in Subpart G of 10CFR72 for ISFSIs.</p> <p><u>Actions for Consideration:</u> Section 830.120 contains the majority of the NRC quality assurance requirements that are in 10CFR50, Appendix B, and 10CFR72, Subpart G. The differences in the sets of requirements, as discussed in detail in Attachment A of this report, are included in DOE Order 4700.1 and WMC quality assurance procedures and instructions. For nuclear safety equivalency, ensure the appropriate requirements of the referenced WMC procedures and instructions remain in effect (e.g., through use of SNFP-specific documents).</p>	<p>See the Quality Assurance evaluation included in Attachment A to this report.</p> <p>Subpart G of 10CFR72 interprets the quality assurance requirements of NRC's 10CFR50, Appendix B, and applies these requirements to design, purchase, fabrication, handling, shipping, maintenance, repair, modification of structures, systems, and components, and decommissioning activities that are important to safety for ISFSI facilities.</p>

Table 5.b Title 10 CFR Summary-Level Comparisons of Part 50 Technical/Safety Issues

Sect/Appendix	Title	SNFP Path Forward Discussion	NRC Safety Issue(s)
Part 50, Appendix C	<i>A Guide for the Financial Data and Related Information Required to Establish Financial Qualifications for Facility Construction Permits</i>	This appendix establishes whether the utility requesting the construction permit has adequate financial resources to complete construction	NA
Part 50, Appendix D	Reserved by the NRC	None.	NA
Part 50, Appendix E	<i>Emergency Planning and Preparedness for Production and Utilization Facilities</i>	The discussion provided above in this table for §50.47 is applicable to Appendix E.	The discussion provided above in this table for §50.47 is applicable to Appendix E.
Part 50, Appendix F	<i>Policy Related to Siting of Fuel Reprocessing Plants and Related Waste Management Facilities</i>	This appendix establishes siting criteria for fuel reprocessing plants to be located on land not owned by the Federal Government.	NA

Table 5.b Title 10 CFR Summary-Level Comparisons of Part 50 Technical/Safety Issues

Sect./Appendix	Title	SNFP Path Forward Discussion	NRC Safety Issue(s)
Part 50, Appendix G	Fracture Toughness Requirements	<p>Appendix G is not directly applicable to the SNFP Path Forward as the MCOs will receive a very low neutron fluence over their life. (Some neutrons will be present from spontaneous fissions and alpha-neutron reactions; this will be less than that of LWR spent fuel because of fewer oxygen atoms with the metal fuel). Also, the MCOs are to be made of austenitic stainless steel. Austenitic stainless steels do not have a similar concern for fracture toughness as the material is more ductile.</p> <p>Low-temperature fracture toughness of ferritic material used for the shipping casks is important. For NRC licensees, this issue is addressed by Regulatory Guide 7.11, <i>Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 Inches (0.1 m)</i> and Regulatory Guide 7.12, <i>Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Wall Thickness Greater than 4 Inches (0.1 m) But Not Exceeding 12 Inches (0.3 m)</i>, which applies to vessels with a wall thickness of greater than four inches. However, shipping casks are not considered to be "new SNFP Path Forward facilities" and, as such, are not within the scope of this equivalency comparison.</p> <p>Hydrogen induced embrittlement may be important in the selection of the material used for the MCOs. This concern is being considered by the SNFP in selection of the material to be used for the MCOs. The concern is, given the admittedly low hydrogen pressure, might hydrogen embrittlement still be a problem considering the required long life (i.e., 75 years) of the MCO. The potential for hydrogen generation during the various aspects of the SNFP Path Forward is being addressed by the Technology Acquisition Plan (TAP) discussed in this table, in response to §50.44.</p> <p>Conclusions: Irradiation induced embrittlement is not a concern for the new SNFP Path Forward facilities. Hydrogen induced embrittlement is being addressed as part of the TAP and the SNFP by considerations being given to hydrogen generation and MCO material selection.</p> <p>Actions for Consideration: None.</p>	<p>This appendix ensures that a margin of safety exists for ferritic materials of the pressure-retaining portion of the reactor coolant pressure boundary for light water reactors. Of particular concern is neutron-induced embrittlement of the belt line region of the reactor pressure vessel.</p>

Table 5.b Title 10 CFR Summary-Level Comparisons of Part 50 Technical/Safety Issues

Sect./Appendix	Title	SNFP Path Forward Discussion	NRC Safety Issue(s)
Part 50, Appendix H	<i>Reactor Vessel Material Surveillance Program Requirements</i>	<p>Neutron-induced embrittlement of ferritic materials is not applicable to the SNFP Path Forward as discussed above for Appendix G. Also, Appendix G does not require a monitoring program if the peak neutron fluence will not exceed 10^{19} n/cm² during the design life of the vessel. The NRC has not applied any aspect of Appendix H to in-plant or independent spent fuel storage.</p> <p><u>Conclusions:</u> As SNFP Path Forward components will not obtain significant neutron fluence and as the potential for low-temperature induced embrittlement discussed above in response to Appendix G does not require a surveillance program, Appendix H has no application to the SNFP Path Forward. As such, no additional equivalency analysis is required.</p> <p><u>Actions for Consideration:</u> None.</p>	This appendix ensures that licensees establish a surveillance specimen capsule withdrawal schedule to provide for the testing required of Appendix G.
Part 50, Appendix I	<i>Numerical Guidelines for Achieving ALARA</i>	<p><u>Conclusions:</u> DOE Order 5400.5, <i>Radiation Protection of the Public and the Environment</i>, provides applicable numerical guides for ALARA objectives at DOE nuclear facilities.</p> <p>utilize DOE Order 5400.5 for ALARA guidance in the design and construction of the new SNFP Path Forward facilities as the Appendix I criteria are specifically called out as not applicable.</p> <p><u>Actions for Consideration:</u> None.</p>	This appendix provides numerical guides for design objectives for effluents released to unrestricted areas to achieve ALARA principles. It states: "The guides presented in this appendix are appropriate only for light-water-cooled nuclear power reactors and not for other types of nuclear facilities."
Part 50, Appendix J	<i>Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors</i>	<p>Appendix J deals with the potential for leakage out of a pressurized containment. This concern is not applicable to the SNFP Path Forward as, for the credible accidents identified, no potential for significant building pressurization has been identified. For buildings that provide containment or confinement for the SNFP Path Forward, the maintenance of a negative air pressure during accident conditions will ensure that any leakage is into the building where it then would be exhausted through an elevated release and/or HEPA filters, if required for accident mitigation, and the stack. The NRC has not applied any aspect of Appendix J to in-plant or independent spent fuel storage.</p> <p><u>Conclusions:</u> With no potential identified for pressurization of the CSB or other new SNFP Path Forward facilities, Appendix J is not applicable to the SNFP Path Forward. As such, no additional analysis is provided for this equivalency review.</p> <p><u>Actions for Consideration:</u> None.</p>	This appendix provides for limitation of primary containment leakage to the environment that would occur during accident conditions that pressurize containment.
Part 50, Appendix K	<i>ECCS Evaluation Models</i>	<p><u>Conclusions:</u> Appendix K is not applicable to the SNFP Path Forward for the reasons provided above in this table for §50.46.</p> <p><u>Actions for Consideration:</u> None.</p>	This appendix ensures that models used to confirm the ECCS acceptance criteria of §50.46 are satisfied by conservative calculational methods.

Table 5.b Title 10 CFR Summary-Level Comparisons of Part 50 Technical/Safety Issues

Sect/Appendix	Title	SNEP Path Forward Discussion	NRC Safety Issue(s)
Part 50, Appendix L	<i>Information for Anti-Trust Review</i>	This appendix defines the information required by the U.S. Attorney General to complete the anti-trust review as required by Section 105c of the Atomic Energy Act. The main issue is that large utilities shall not use nuclear power in a manner that is unfair to smaller utilities that may not have the resources to construct and operate a nuclear power plant.	NA
Part 50, Appendix M	<i>Standardization of Design: Manufacture of Nuclear Power Reactors; Construction and Operation of Nuclear Power Reactors Manufactured Pursuant to Commission License</i>	This appendix establishes the requirements and provisions to allow for the manufacture of a production or utilization facility (see <i>Definitions</i> , Section 8.2) independent of its subsequent construction and installation at the site of future operation for which a future construction would be required.	NA
Part 50, Appendix N	<i>Standardization of Nuclear Power Plant Designs: Licenses to Construct and Operate Nuclear Power Reactors of Duplicate Design at Multiple Sites</i>	This appendix provides for construction and operation of one plant design by one or more applicants with the plants to be located at different sites.	NA
Part 50, Appendix O	<i>Standardization of Design: Staff Review of Standard Designs</i>	This appendix provides for the filing, NRC staff review and referral to the Advisory Committee on Reactor Safeguards of standard designs for a nuclear power plant.	NA
Part 50, Appendix P	Reserved by the NRC	NA	NA
Part 50, Appendix Q	<i>Pre-Application Early Site Review of Site Suitability Issues</i>	This appendix provides for the early resolution of site suitability issues, primarily administrative in nature, that do not impact design or construction.	NA
Part 50, Appendix R	<i>Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979.</i>	See discussion provided above in this table for §50.48.	See discussion provided above for §50.48. Appendix R added fire protection requirements beyond those of Branch Technical Position (BTP) APCS 9.5-1 (the BTP in place at that time) and other NRC documents gave them the force of regulation. The title for Appendix R makes reference to a specific date for applicability, because at the time the rule was issued, it was planned to issue a second rule for plants licensed after January 1, 1979 (SECY-BI-114). This was not accomplished and the NRC invoked Appendix R on these newer plants by issuing docket-specific letters requiring the applicant to include as part of its fire protection program, a comparison of its fire protection program to the requirements of Appendix R.

Table 5.c Title 10 CFR Summary-Level Comparisons of Part 72 Technical/Safety Issues

Sect.	Title	Additional Information	Conclusions
72.1	Purpose	Subpart A, General Provisions, to 10CFR72, which includes §§2.1 through 72.12, was not addressed in the Scientech document prepared for DOE HQ in June 1995, titled <i>Comparison of MRC and DRI Requirements for Independent Spent Nuclear Fuel Storage Facilities</i> , because this Subpart contains NRC administrative licensing procedures and was judged by Scientech to not be important for the purposes of the SNFP Path forward. However, some of the sections of Subpart A have relevance in regards to potential impact on nuclear safety and are addressed herein to the appropriate degree.	This section does not contain information relevant to MRC nuclear safety equivalency. It would not affect the design or construction of new SNFP facilities or public or worker health and safety from design or construction related activities. Such sections are noted as NA elsewhere in Table 5.c, where an "NA" exists under the Conclusions column of this table, the NA designation was not established by Scientech.
72.2	Scope	None.	NA
72.3	Definitions	This section provides three criteria for defining SSCs important to safety: <ul style="list-style-type: none"> to maintain the conditions required to store spent fuel or high-level radioactive waste safely; to prevent damage to the spent fuel or the high-level radioactive waste container during handling and storage; or to provide reasonable assurance that spent fuel or high-level radioactive waste can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public. It is understood that the MRC applies these "important to safety" criteria in a manner to assure that there is not a credible breach of the spent nuclear fuel containment/confinement such that a radioactive release could be postulated to reach the controlled area boundary defined as 100 meters from the facility release point. Therefore, SSCs whose failure or maloperation would cause or allow such a release, would be defined as "important to safety." This implementation of the "important to safety" concept is prescriptive rather than consequence or risk based like the DOE and/or MRC criteria used to establish safety classification. Hence, the §72.3 criteria, when implemented, may identify SSCs "important to safety" under 10CFR72 that would not be "important to safety" under a consequence or risk-based approach.	<p>Actions for Consideration: Consider identifying SSCs "important to safety" in accordance with §72.3, as follows:</p> <p>"Those SSCs that assure there is not a credible breach of the spent nuclear fuel containment/confinement such that a radioactive release could be postulated to reach the controlled area boundary defined as 100 meters from the facility release point."</p> <p>Once SSCs "important to safety" have been identified, impose the requirements for Safety Class 1 SSCs as defined in <i>WHC-CH-4-46, Safety Analysis Manual</i>, Section 9.0, Revision 1, <i>Safety Classification of Structures, Systems, and Components</i>.</p>
72.4	Communications	None.	NA
72.5	Interpretations	None.	NA
72.6	License required: Types of licenses	None.	NA
72.7	Specific Exemptions	Refer to Table 4, §50.12 discussion.	None. (Also refer to Table 4, §50.12 discussion.)
72.8	Denial of licensing by Agreement States	None.	NA

Table 5.c Title 10 CFR Summary-Level Comparisons of Part 72 Technical/Safety Issues

Sect.	Title	Additional Information	Conclusions
72.9	<i>Information collection requirements: OMB approval</i>	None.	NA
72.10	<i>Employee protection</i>	Refer to Table 4, §50.7 discussion.	None. (Also refer to Table 4, §50.7 discussion.)
72.11	<i>Completeness and accuracy of information</i>	Refer to Table 4, §50.9 discussion.	None. (Also refer to Table 4, §50.9 discussion.)
72.12	<i>Deliberate misconduct</i>	Refer to Table 4, §50.5 discussion.	None. (Also refer to Table 4, §50.5 discussion.)
72.16	<i>Filing of application for specific license</i>	Subpart B, <i>License Application, Form, and Contents</i> , which includes §§72.16 through 72.34, was addressed in the Sciencetech document prepared for DOE HQ in June 1995, titled <i>Comparison of NRC and DOE Requirements for Independent Spent Nuclear Fuel Storage Facilities</i> .	NA
72.18	<i>Elimination of repetition</i>	None.	NA
72.20	<i>Public inspection of application</i>	None. Refer to the 10CFR2 discussion in Section 5.0 of this report.	Refer to the 10CFR2 discussion in Section 5.0 of this report.
72.22	<i>Contents of application: (General and financial) information</i>	None.	NA

Table 5.c Title 10 CFR Summary-Level Comparisons of Part 72 Technical/Safety Issues

Sect.	Title	Additional Information	Conclusions
72.24	<i>Contents of application: technical information</i>	<p>Refer to the June 1995 Sciencetech document, titled <i>Comparison of NRC and DOE Requirements for Independent Spent Nuclear Fuel Storage Facilities</i>, for details that support the Sciencetech conclusions. [Note: this statement applies to all Sciencetech conclusions provided in this table.]</p> <p>The Sciencetech document also included NRC Regulatory Guides 3.48, Revision 1, <i>Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation (Dry Storage)</i>, and 3.50, Revision 1, <i>Standard Format and Content for a License Application to Store Spent Fuel and High-Level Radioactive Waste</i>, as part of the comparison between DOE and NRC requirements. The review provided as part of this report (SAIC) included DOE-STD-3009-94, <i>Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports</i>.</p> <p>In regards to aircraft impacts, DOE Order 6430.1A, <i>General Design Criteria</i>, Sections 0111.99.0.5 and 0200.99.0.2 do require consideration of offsite hazards such as proximity to airports. In addition, NRC Regulatory Guide 1.70, Revision 3 (<i>Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants</i>), paragraphs 2.2.2.5 Airports, 2.2.3.1 <i>Determination of Design Basis Events</i> and 3.5.1.6, <i>Aircraft Hazards</i>, have criteria for determining design basis events external to the nuclear plants such as those that may be associated with postulated aircraft impacts. Specifically, such events must have a probability of occurrence on the order of about 10^{-6} per year or greater, and for aircraft risk the local airport(s) must first meet certain requirements in regards to distance, flight paths, and magnitude of operations. Based on evaluations associated with the WNP-1/4 and WNP-2 plants, aircraft hazards were determined to not be a concern for those locations and as a result did not become a design basis event. Since the new SNFP Path Forward facilities sites are located further from the local airports than WNP-1/4 or WNP-2, aircraft impacts are not considered a concern for new SNFP Path Forward facilities. It should also be noted that transportation risks (including airport considerations) are evaluated using a risk-based approach ($<10^{-6}$ per year) in more recent NRC guidance as well, draft Regulatory Guide DG-4004, <i>General Site Suitability Criteria for Nuclear Power Stations</i>, dated February 1995. In light of referencing SECY-93-087, <i>Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Designs</i>, in regards to considering a reduction of the tornado design basis wind speed (refer to §72.122), it should be noted that SECY-93-087 also states by relaxing the tornado design basis wind speed, light aircraft aviation crashes would need to be evaluated on a site-specific basis. Should SECY-93-087 be used as a basis for reducing the tornado wind speeds, such an evaluation would need to be performed.</p> <p>Related to concerns for future populations, their locations, and issues such as Hanford privatization, as the Hanford Site evolves, public health and safety must be continually revisited and probable use of Hanford lands in the future should be considered in the design basis of new SNFP facilities at this time.</p>	<p>Sciencetech:</p> <p>DOE and NRC require the same types of information to be provided in the SAR. However, NRC requirements are more specific regarding the details about what is to be provided.</p> <p>DOE does not require evaluation of trends and future population projections on the suitability of the facility siting. Considering changes in missions, this is an important deficiency.</p> <p>DOE facilities have not traditionally evaluated impacts from offsite hazards such as toxic gases and aircraft impacts.</p> <p>NRC requires significant detail on meteorology, climatology, hydrology and natural phenomena hazards to determine suitability of sites. DOE sites were selected prior to such detailed evaluations primarily on the basis of large land areas and low population densities, thus less emphasis was placed upon site characteristics.</p> <p>The required description of safety considerations for the design is similar for NRC and DOE. NRC also focuses on off-site facilities near the regulated facility, which might impact safety or emergency operations. DOE does not require such facilities to be addressed unless the accident analyses show that such facilities represent a hazard. The large DOE sites generally make such evaluations unnecessary, but diversity of missions and future privatizations make this a potentially serious deficiency.</p> <p>DOE requires a description and evaluation of engineered safety features, but does not require a similar description of auxiliary systems which are important to the safety of the installation. Safety considerations and control of such systems can impact safety. NRC is very specific as to which auxiliary systems should be evaluated.</p> <p>DOE does not require evaluation of margins of safety during normal and expected operational occurrences.</p> <p>DOE does not require that a listing of radiation sources or estimates of radiation exposures from operations be provided in the SAR.</p> <p>DOE does not require a discussion of ALARA design and operational considerations.</p> <p>NRC requires person-rem estimates for each radionuclide that could escape the controlled area during normal operation.</p> <p>DOE does not require a discussion of plans to resolve new or unproven structures, systems, or components.</p> <p>DOE does not require in the SAR a description of the provisions for packaging, storage, and disposal of solid wastes, nor does DOE require a discussion of dose estimates for each type of effluent and waste.</p>

Table 5.c Title 10 CFR Summary-Level Comparisons of Part 72 Technical/Safety Issues

Sect.	Title	Additional Information	Conclusions
72.24 cont'd	<i>Contents of application technical information</i>	<p>The Sciencetech conclusion regarding ALARA is in reference to NRC Regulatory Guide 8.8, Revision 3, <i>Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As Is Reasonably Achievable</i>, which is referred to from Regulatory Guide 3.48, Revision 1. It should be noted that DOE Order 6430.1A, Section 1540.99.0.6, <i>System Installation</i>, references Regulatory Guide 8.8, Revision 3, for piping design considerations for systems that carry radioactive material.</p> <p>Section 72.24(i) requires that SSCs important to safety whose functional adequacy or reliability have not been demonstrated by prior use for that purpose or cannot be demonstrated by reference to performance data in related applications or to widely accepted engineering principles, should be identified along with a schedule showing how safety questions will be resolved prior to the initial receipt of spent fuel.</p> <p>Refer to 72.26 for related information.</p> <p>Additional items required by 72.24 and/or Regulatory Guide 3.48, Revision 1, that are not required by DOE Order 5480.23 or DOE STD-3009-94 and are not listed in the Sciencetech conclusions, are:</p> <ul style="list-style-type: none"> • Compliance to the 10CFR20 radiological limits (refer to Table 5.a, 10CFR20 discussion, for additional information). • Description of the Security Plan including design features; 10CFR72, Subpart H, plans; and a physical security plan for radiological sabotage protection and safeguards to promote the common defense and security. (Note: this information is provided as a separate document to the DOE, as opposed to being part of the SAR.) • Tornados and tornado missiles for nonreactor nuclear facilities (refer to 72.122 for additional information). • Justification for not using Regulatory Guide 1.60, Revision 1, <i>Design Response Spectra for Seismic Design of Nuclear Power Plants</i>, criteria (refer to Table 6.a, Regulatory Guide 1.60 for additional information). 	<p>NRC requires the following, which are not mentioned in DOE 5480.23: methods of event detection, description of models and computer programs used in the analyses, time-dependent plots of releases, and margins of protection.</p> <p><u>Actions for Consideration:</u> To have equivalent technical information in the application, the SAR(s) for new SNF facilities would need to incorporate additional information (representative details provided herein). In terms of whether such information would be relevant to the design and construction of new SNF Path Forward facilities and to the health and safety of the public or the worker, the NRC views the submittal of such information, and the opportunity for the NRC to review the information called for in the applicable Regulatory Guides (i.e., 3.48 and 3.50), as enhancing the nuclear safety of the nuclear plant. This is due to the NRC's viewpoint that they have a broad awareness and thorough knowledge of the nuclear safety issues, more so than individual applicants, through their review of numerous applications. If one accepts this premise, the inclusion of such information is relevant to NRC nuclear safety equivalency. In addition, related to concerns for future populations, their locations, and issues such as Hanford privatization, as the Hanford Site evolves, public health and safety must be continually revisited and probable use of Hanford lands in the future should be considered in the design bases of new SNF Path Forward facilities at this time.</p>
72.26	<i>Contents of application technical specifications</i>	<p>The WHC selection criteria for Technical Safety Requirements (TSRs) include radiological and criticality considerations, plus hazardous material (chemical) considerations which may act as an initiating event for a subsequent radiological event. This approach to TSRs varies to some degree from some interpretations of the DOE Order, 5480.22, <i>Technical Safety Requirements</i>, selection criteria. Refer to the response provided for 50.36a in Table 5.b for additional information.</p>	<p><u>Sciencetech:</u></p> <p>The required format for NRC Technical Specifications (TSs) and DOE Technical Safety Requirements (TSRs) are comparable. However, in addition to similar DOE and NRC requirements to monitor and report environmental and effluent releases, NRC also requires TS control of effluents to prevent releases in excess of 10CFR20 and ALARA limits.</p> <p><u>Actions for Consideration:</u> Refer to the response provided for 50.36a in Table 5.b.</p>

Table 5.c Title 10 CFR Summary-Level Comparisons of Part 72 Technical/Safety Issues

Sect.	Title	Additional Information	Conclusions
72.28	<i>Contents of application: Applicant's technical qualifications</i>	This section does not establish design or construction requirements. It focuses on operational activities. In addition, WMC (as the M&O Contractor) and design and construction subcontractors are qualified through the procurement process.	NA
72.30	<i>Financial assurance and recordkeeping for decommissioning</i>	None	NA
72.32	<i>Emergency plan</i>	Refer to Table 5.b, §50.47 discussion, and Table 6.b, SECY-93-006 discussion.	None. (Also refer to Table 5.b, §50.47 discussion, and Table 6.b, SECY-93-006 discussion.)
72.34	<i>Environmental report</i>	This section does not establish design or construction requirements. Environmental issues are not within the scope of this NRC nuclear safety equivalency activity.	NA
72.40	<i>Issuance of license</i>	Subpart C, <i>Issuance and Conditions of License</i> , to 10CFR72, which includes §§72.40 through 72.62, was not addressed in the Sciencetech document prepared for DOE-HQ in June 1995, titled <i>Comparison of NRC and DOE Requirements for Independent Spent Nuclear Fuel Storage Facilities</i> , because this subpart contains NRC administrative licensing procedures and was judged by Sciencetech to not be important for the purposes of the SNFP Path Forward. However, some of the sections of Subpart C have relevance to potential impact on nuclear safety and will be addressed herein to the appropriate degree. This section does not establish design or construction requirements.	NA
72.42	<i>Duration of license: renewal</i>	None	NA
72.44	<i>License conditions</i>	This section was addressed as part of the Sciencetech report, associated with §72.26, due to its direct relationship to NRC Technical Specifications/DOE Technical Safety Requirements. Refer to the response provided for §50.36a in Table 5.b for additional information.	Sciencetech: The required format for NRC Technical Specifications (TSs) and DOE Technical Safety Requirements (TSRs) are comparable. In addition to similar DOE and NRC requirements to monitor and report environmental and effluent releases, NRC also requires TS control of effluents to prevent releases in excess of 10CFR20 and ALARA limits. Actions for Consideration: Refer to the response provided for §50.36a in Table 5.b.
72.46	<i>Public hearings</i>	This section was addressed as part of the Sciencetech report, associated with §72.16, due to its being part of Subpart B. The overall issue of NRC administrative proceedings, addressed in 10CFR2, is the subject of a separate discussion in Section 5.0 of this report.	Sciencetech: DOE 5480-23 does not provide for docketing, notice of proposed action, or formal public hearings as part of the SAR review and approval process. Actions for Consideration: None. Refer to the 10CFR2 discussion in Section 5.0 of this report.

Table 5.c Title 10 CFR Summary-Level Comparisons of Part 72 Technical/Safety Issues

Sect.	Title	Additional Information	Conclusions
72.48	<i>Changes, tests, and experiments</i>	This section does not establish design or construction requirements. Operational issues are not within the scope of the NRC nuclear safety equivalency activity.	NA
72.50	<i>Transfer of license</i>	None	NA
72.52	<i>Creditor regulations</i>	None	NA
72.54	<i>Expiration and termination of licenses and decommissioning of sites and separate buildings or outdoor areas</i>	None	NA
72.56	<i>Application for amendment of license</i>	None	NA
72.58	<i>Issuance of amendment</i>	None	NA
72.60	<i>Modification, revocation, and suspension of license</i>	None	NA
72.62	<i>Backfitting</i>	This section does not establish initial design or construction requirements and is associated with the operating phase.	NA
72.70	<i>Safety analysis report updating</i>	Subpart D, <i>Records, Reports, Inspections, and Enforcement</i> , to 10CFR72, which includes §§72.70 through 72.86, was addressed in the Sciencetech document prepared for DOE-HQ in June 1995, titled <i>Comparison of NRC and DOE Requirements for Independent Spent Nuclear Fuel Storage Facilities</i> . This section does not establish design or construction requirements.	NA
72.72	<i>Material balance, inventory, and records requirements for stored materials</i>	This section does not establish design or construction requirements. Operational issues are not within the scope of this NRC nuclear safety equivalency activity.	NA (Also refer to Table 5.a. Parts 70 and 74 discussions.)
72.74	<i>Reports of accidental criticality or loss of special nuclear material</i>	This section does not establish design or construction requirements. Operational issues are not within the scope of this NRC nuclear safety equivalency activity.	NA (Also refer to Table 5.a. Parts 70 and 74 discussions.)

Table 5.c Title 10 CFR Summary-Level Comparisons of Part 72 Technical/Safety Issues

Sect.	Title	Additional Information	Conclusions
72.75	Reporting requirements for specific events and conditions	This section does not establish design or construction requirements. Operational issues are not within the scope of this NRC nuclear safety equivalency activity.	NA
72.76	Material status reports	This section does not establish design or construction requirements. Operational issues are not within the scope of this NRC nuclear safety equivalency activity.	NA (Also refer to Table 5 a, Parts 70 and 74 discussions.)
72.78	Nuclear material transfer reports	This section does not establish design or construction requirements. Operational issues are not within the scope of this NRC nuclear safety equivalency activity.	NA (Also refer to Table 5 a, Parts 70 and 74 discussions.)
72.80	Other records and reports	This section does not establish design or construction requirements. Operational issues are not within the scope of this NRC nuclear safety equivalency activity.	NA
72.82	Inspections and tests	This section does not establish design or construction requirements. Operational issues are not within the scope of this NRC nuclear safety equivalency activity.	NA (Also refer to Table 5 a, Part 70 discussion.)
72.84	Violations	Refer to Table 5 b, §50.110 discussion.	Sciencetech: No comparison is provided since any DOE facility will be required to abide by the requirements in 10CFR820. Actions for <u>Consideration</u> : None. (Also refer to Table 5 b, §50.110 discussion.)
72.86	Criminal penalties	Refer to Table 4, §50.5 discussion.	Sciencetech: No comparison is provided since any DOE facility will be required to abide by the requirements in 10CFR820. Actions for <u>Consideration</u> : None. (Also refer to Table 4, §50.5 discussion.)

Table 5.c Title 10 CFR Summary-Level Comparisons of Part 72 Technical/Safety Issues

Sect.	Title	Additional Information	Conclusions
72 90	General considerations	<p>Subpart E, <i>Siting Evaluation Factors</i>, to 10CFR72, which includes §§72.90 through 72.108, was addressed in the Sciencetech document prepared for DOE HQ in June 1995, titled <i>Comparison of NRC and DOE Requirements for Independent Spent Nuclear Fuel Storage Facilities</i>. Refer to §72.92 and 72.102 for further information.</p>	<p>Sciencetech:</p> <p>Part 72, Subpart E, requirements and DOE 5480.28 and 6430.1A contain similar requirements for defining and evaluating characteristics for the siting of an ISFSI. NRC does not use a graded approach in the NPH analysis of SSCs. NRC practice has been to identify safety-related systems based on the functional needs to safely shutdown the facility, mitigate the effects of accidents and NPHs, and to prevent the uncontrolled release of radioactive materials; then to use a deterministic approach to determine the impact of those NPH and accidental events on safety-related SSCs. NRC would likely have difficulty in its review of an ISFSI design that utilizes a graded approach in the analysis of NPH events.</p> <p>DOE and NRC require the identification and evaluation of those site characteristics that affect safety. External natural and man-induced events described by NRC and DOE are similar and the frequency and consequences of these events dictate the design of the facility.</p> <p><u>Actions for Consideration:</u> Refer to §72.92.</p>
72 92	Design basis external natural events	<p>Safety-related SSCs are referred to as SSCs important to safety in 10CFR72, as defined in §72.3.</p> <p>Refer to draft WHC document <i>Spent Nuclear Fuel Project Seismic Design Criteria, Nuclear Regulatory Commission Equivalency Evaluation Report</i>, October 1995, for related information.</p> <p>Also, refer to Table 6 a, DG-1032, for additional information.</p>	<p>Sciencetech:</p> <p>DOE and NRC require that natural phenomena, which can occur and have an impact on public safety, be identified and assessed for the potential effects on safe facility operation. DOE applies a graded approach to placing SSCs in performance categories.</p> <p>DOE 5480.28 applies a probabilistic approach to NPH in conjunction with a graded approach. It assigns performance goals and performance categories to SSCs. NRC does not [currently] use such an approach in reviewing facility designs. NRC specifies a deterministic approach in 10CFR100, Appendix A-VI, for the seismic design of safety-related SSCs. NRC would likely not look favorably on a probabilistic approach in the analysis of NPH events, especially when so much reliance is placed on engineering judgment and a complicated methodology for determining the thresholds of each category.</p> <p><u>Actions for Consideration:</u> To develop, in a strict regulatory context, equivalent and the most conservative seismic criteria for Safety Class 1 SSCs, perform the analyses necessary to define the design earthquakes by both the DOE requirements (DOE Order 5480.28, <i>Natural Phenomena Hazards Mitigation</i>, and DOE STD-1020-94, <i>Natural Phenomena Hazards Design and Evaluation Criteria</i>) and the NRC requirements (10CFR100, Appendix A) and use the limiting design earthquakes. Or consider: 1) developing a justification for the use of a 0.35g design basis earthquake for Safety Class 1 SSCs in the design of the CSB, and 2) justify the use of DOE 5480.28, and companion standards, to determine the appropriate design earthquakes for the Conditioning Facility and other new SNF Path Forward facilities.</p>

Table 5.c Title 10 CFR Summary-Level Comparisons of Part 72 Technical/Safety Issues

Sect.	Title	Additional Information	Conclusions
72.94	<i>Design basis external man-induced events</i>	See discussion provided in Table A-2 for General Design Criterion 4.	<p>Sciencetech:</p> <p>DOE's treatment of man-induced events is consistent with the treatment of such events by the NRC.</p> <p><u>Actions for Consideration:</u> None.</p>
72.96	<i>Siting limitations</i>	This section for the most part addresses a Monitored Retrievable Storage Installation (MRS) and limitations regarding the MRS's proximity to a HLW repository. In addition, in §72.96(a) it states: "An ISFSI which is owned and operated by DOE must not be located at any site within which there is a candidate site for a HLW repository. This limitation shall apply until such time as DOE decides that such candidate site is no longer a candidate site under consideration for development as a HLW repository."	NA
72.98	<i>Identifying regions around an ISFSI or MRS site</i>	<p>Related to concerns for future populations, their locations, and issues such as Hanford privatization, as the Hanford Site evolves, public health and safety must be continually revisited and probable use of Hanford lands in the future should be considered in the design bases of new SNFP Path Forward facilities at this time.</p>	<p>Sciencetech:</p> <p>NRC and DOE requirements associated with the region around the site are similar. Both identify and use external phenomena as the basis for design, both consider the impact of construction activities, operations and decommissioning, and both consider present population, and land and water uses that might influence the potential release of radioactive material during ISFSI operations.</p> <p>DOE does not address future characteristics of population, and land and water use in the region under consideration. NRC requires that such future needs be addressed. There is a potential for DOE to construct a new facility in a high growth area, thereby increasing the potential risks to future population centers that might gradually encroach on the facility.</p> <p><u>Actions for Consideration:</u> Refer to §50.34.</p>
72.100	<i>Defining potential effects of the ISFSI or MRS on the region</i>	Refer to §§72.104 and 72.106.	<p>Sciencetech:</p> <p>The requirements for treating direct radiation effects and radioactive material in the effluent of an ISFSI are consistent for the NRC and DOE.</p> <p><u>Actions for Consideration:</u> Refer to §§72.104 and 72.106.</p>

Table 5.c Title 10 CFR Summary-Level Comparisons of Part 72 Technical/Safety Issues

Sect.	Title	Additional Information	Conclusions
72 102	<i>Geological and seismic characteristics</i>	<p>DOE Order 5480-28 (to be effective at the Hanford Site in the near term) applies a probabilistic approach to NPH in conjunction with a graded approach. It assigns performance goals and performance categories to SSCs. NRC, under the existing Title 10 of the CFRs, does not use such an approach in reviewing facility designs. NRC specifies a deterministic approach in 10CFR100, Appendix A, for the seismic design of safety-related SSCs (SSCs) important to safety, as defined in §72.3).</p> <p>NRC required that for an ISFSI west of the Rocky Mountain Front, and in other areas of known potential seismic activity, seismicity will be evaluated by the techniques of Appendix A of 10CFR100. For sites evaluated under the criteria of 10CFR100, Appendix A, the design earthquakes for use in the design of structures must be equivalent to the safe shutdown earthquake (SSE) for nuclear power plants.</p> <p>Assuming the appropriate performance category for a new SNPP Path Forward facility is 3 under the DOE Order 5480-28 approach, the peak horizontal free-field acceleration is about 0.2g. Using the existing NRC 10CFR100, Appendix A, approach, the SSE has been estimated to be 0.48g.</p>	<p>Scientech:</p> <p>NRC and DOE (DOE 6430-1A-0111-99-0-4) have similar requirements for the collection and use of seismic data and historical records, and both apply comparable methodologies in seismic analyses. However, DOE and NRC differ in specifying DOE ground acceleration requirements. NRC specifies 10CFR100 analyses for areas west of the divide and for sites that lie within range of strong near field ground motion. From a practical point-of-view, the differences are not significant, since an ISFSI sited anywhere in the U.S. would be required to be designed using dynamic analysis methods based on seismic, geologic and soil evaluations of the region being considered.</p> <p>NRC and DOE have similar requirements for site-specific investigations to ensure proposed foundation loading, and state that siting structures and facilities over active geologic faults, in areas of soil instability, and over soil subject to liquefaction during vibratory ground motion should be avoided.</p> <p><u>Actions for Consideration:</u> Refer to §72.92.</p>
72 104	<i>Criteria for radioactive materials in effluents and direct radiation from an ISFSI or NRS</i>	<p>The Scientech document refers to Subpart F of 10CFR72 for the discussion and conclusions regarding §72.104.</p> <p>Title 40, CFR 191.3, <i>Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes</i>, states: "Management and storage of spent nuclear fuel ... at all facilities for disposal of such fuel or waste that are operated by the Department [DOE] and that are not regulated by the Commission [NRC] or Agreement States shall be conducted in such a manner as to provide reasonable assurance that the combined annual dose equivalent to any member of the public in the general environment resulting from discharges of radioactive material and direct radiation from such management and storage shall not exceed 25 millirems to the whole body and 75 millirems to any critical organ."</p> <p>The radiological exposure criteria in §72.104 (which are the same as for NRC regulated spent nuclear fuel facilities in 40CFR191.3) are: 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other critical organ.</p> <p>The minimal differences in critical organ dose limits for the public are not expected to impact the design of the CSB or the other new SNPP Path Forward facilities.</p>	<p>Scientech:</p> <p>Section 72.104 exposure limits are more restrictive than DOE requirements.</p> <p><u>Actions for Consideration:</u> To have equivalent radiological exposure criteria for the public, §72.104 exposure limits would need to be applied to the new SNPP Path Forward facilities.</p>

Table 5.c Title 10 CFR Summary-Level Comparisons of Part 72 Technical/Safety Issues

Sect.	Title	Additional Information	Conclusions
72.106	Controlled areas of an ISFSI or MRS	<p>The Sciencetech document refers to Subpart F, <i>General Design Criteria</i> of 10CFR72 for the discussion and conclusions regarding §72.106.</p> <p>It should be noted that the accident dose criteria used by WMC for determining safety classification and acceptable risk are defined in WMC-CH-4-46, Sections 7, Revision 3, and 9, Revision 1. The criteria in these two sections are compatible and are event frequency based as follows:</p> <p>For the offsite public:</p> <p>$>10^{-2}$ to $\leq 10^{-1}$ 5 rem $>10^{-4}$ to $\leq 10^{-2}$ 5.0 rem $>10^{-6}$ to $\leq 10^{-4}$ 25.0 rem</p> <p>For the onsite worker:</p> <p>$>10^{-2}$ to $\leq 10^{-1}$ 5.0 rem $>10^{-4}$ to $\leq 10^{-2}$ 25.0 rem $>10^{-6}$ to $\leq 10^{-4}$ 100.0 rem</p> <p>The consequence thresholds may vary as a function of the event frequency when the initiating event is a) natural phenomenon, b) a specific equipment failure, or c) a specific failure mode, where the associated frequency (events per year) is supported by adequate data to make the safety analysis uncertainties small. The frequency to be used corresponds to that of the initiating event, where the frequency is not known or is suspect, a deterministic value of 5 rem and 5.0 rem are used respectively for the offsite public and the onsite worker.</p> <p>In regards to the §72.106 criterion (i.e., individuals at or beyond the controlled area boundary shall not receive a dose greater than 5 rem to the whole body or any organ from a design basis accident), this individual would be the onsite worker at Hanford. For the highest frequency range and for cases where the initiating event frequency is in question, the WMC and NRC criteria are the same. However, where the initiating event frequency is known and it falls into the lower two frequency ranges, the NRC criterion is more conservative.</p>	<p>Sciencetech:</p> <p>The Sciencetech conclusion in Subpart F, §72.126(d) states: the §72.106 design basis accident dose of 5 rem is more restrictive than the DOE criteria.</p> <p>Actions for Consideration: Safety analysis used to determine safety classification should use a deterministic value of 5 rem for the onsite worker at the boundary of the controlled area (defined as 100 meters from the release point of the new SNF Path Forward facility).</p>
72.108	Spent fuel or high-level radioactive waste transportation	This section does not establish design or construction requirements. Environmental issues are not within the scope of this NRC nuclear safety equivalency activity.	NA
72.120	General considerations	Subpart F, <i>General Design Criteria</i> to 10CFR72, which includes §72.120 through 72.130, was addressed in the Sciencetech document prepared for DOE-HQ in June 1995, titled <i>Comparison of WMC and DOE Requirements for Independent Spent Nuclear Fuel Storage Facilities</i> .	<p>Sciencetech:</p> <p>DOE and NRC requirements for development of project-specific design criteria are generally comparable.</p> <p>Actions for Consideration: None.</p>

Table 5.c Title 10 CFR Summary-Level Comparisons of Part 72 Technical/Safety Issues

Sect.	Title	Additional Information	Conclusions
72.122	<i>Overall requirements</i>	<p>The Sciencetech document refers to the discussion in Subpart E of 10CFR72 for further relevant details.</p> <p>The proposed hazard category for the CSB and the other new SNFP Path Forward facilities is 2.</p> <p>Refer to §§72.92 and 72.102 and 10CFR100, Appendix A, for additional details on seismic design considerations.</p> <p>Section 72.102, regarding seismic design requirements for an ISFSI in the west, requires the design earthquake to be equivalent to the safe shutdown earthquake (SSE) for a nuclear power plant. In addition, §72.122 requires the design to withstand the most severe of the natural phenomena reported for the site. There is no mention of the operating basis earthquake (OBE) for an ISFSI. This was deliberate because the conditions pose significantly greater risk due to available energy and significant quantities of nuclear fuel in a critical condition. The OBE is not applicable to the CSB or other new SNFP Path Forward facilities. Refer to DOE Order 6430.1A, Section 0111.99.0.4, <i>Earthquakes</i>, for additional information.</p> <p>The SNFP Integrated Process Strategy, which includes desludging, reracking, cold-vacuum drying, and hot-vacuum conditioning, prevents further degradation of the SNFP Path Forward nuclear fuel.</p>	<p>Sciencetech:</p> <p>NRC and DOE safety definitions and methodologies differ; however, they are consistent in principle. NRC requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance to safety of the function to be performed. DOE STD-3009-94 defines "safety class" and "safety significant" SSCs. NRC defines "safety-related" as a subset of "important-to-safety."</p> <p>DOE and NRC requirements generally agree if the ISFSI is categorized as a high hazard facility.</p> <p>Until 10CFR100, Appendix A, is revised to allow probabilistic and deterministic approaches, the DOE and NRC approaches to determining the DBE are fundamentally different.</p> <p>Current NRC requirements are deterministic. Current DOE requirements are based on probabilistic methods and assigning performance categories to SSCs. DOE's graded approach may allow for a less severe hazard (and corresponding lower ground motion for DBE). The Operating Basis Earthquake, required by NRC, is not required by DOE.</p> <p>There is no specific DOE citation concerning the sharing of SSCs important to safety. However, two citations are listed related to cooling water and electrical power that indicate a commitment to redundant and independent supplies.</p> <p>NRC is more specific in requiring continuous monitoring systems for measuring radiation levels during normal operations and accident conditions. Also, the NRC requirement that the spent fuel cladding must be protected to prevent degradation of the fuel during storage is more specific than the DOE requirement to provide a corrosion-resistant confinement.</p> <p>For emergency utility services, NRC requires the design to permit testing of the functional operability and capacity, including the full operational sequence of each system for transfer between normal and emergency supply sources. DOE does not require testing of the full operational sequence.</p>

Table 5.c Title 10 CFR Summary-Level Comparisons of Part 72 Technical/Safety Issues

Sect.	Title	Additional Information	Conclusions
72 122 (cont.)		<p>On the surface, ready retrieval of an MCO for further processing or disposal does not appear to be directly related to the health and safety of the public or the worker. However, in the event a problem arises with an MCO(s) during storage, steps may need to be taken to establish a more safe condition for the MCO(s) to preclude off-normal conditions that would contribute to a radiological release.</p> <p>Section 72 122 also requires for an ISFSI consideration of the most severe tornado. Near-term DOE NPH criteria for the Hanford Site (DOE Order 5480 28 and DOE STD 1020 94), nor established Hanford Site specific natural phenomena criteria (SDC-4 1, Revision 12, Standard Arch-Civil Design Criteria, Design Loads for Facilities), require tornado consideration for new SNFP Path Forward facilities. Historically, WASH-1300, <i>Technical Basis for Interim Regional Tornado Criteria</i>, and Regulatory Guide 1.76, <i>Design Basis Tornado for Nuclear Power Plants</i>, have governed MRC tornado design criteria. However, during NRC/industry proceedings regarding advanced light water reactors, the NRC staff concluded that it was acceptable to reduce the maximum tornado wind speeds in comparison to Regulatory Guide 1.76. For sites west of the Rocky Mountains this reduction would be from 240 to 200 mph. Discussions on this issue are found in SECY-93-087 (<i>Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs</i>, April 2, 1993) and in the Safety Evaluation Report. SECY-93-087 states that with the reduction in tornado wind speed, protection of safety related SSCs from external events such as explosions and general aviation crashes would need to be considered on a site-specific basis.</p> <p>(Note: in consideration of the CSB design, which includes a below-grade vault and an operating floor, a vertical missile should be evaluated as part of the tornado issue.)</p> <p>In addition, the NRC is considering revisions to SRP 3.5.1.4, Revision 2, <i>Missiles Generated by Natural Phenomena</i>, to incorporate data on tornado wind speeds contained in NUREG/CR-4461, <i>Tornado Climatology of the Contiguous United States</i>. The revisions to the Spectrum I and generally reduce the missile velocities for the Spectrum I missiles (Section III 4). The SRP revision would also make explicit for tornado missile protection the acceptance criterion of 10⁶ events/yr relative to the need to protect an essential system from a single event.</p>	<p>NRC requires for an ISFSI located on the site of another facility, sharing of common utilities and services and physical connection with another facility may occur only when there is no significant 1) increase in the probability or consequence of an accident or malfunction of SSCs important to safety, or 2) reduction in the margin of safety. DOE does not address this issue.</p> <p>NRC requires that storage systems must be designed to allow ready retrieval of spent fuel for further processing or disposal. DOE does not address this issue.</p> <p><u>Actions for Consideration:</u></p> <p>To have equivalent requirements:</p> <ul style="list-style-type: none"> ● Refer to §72.92. ● Since the CSB and the Conditioning Facility are expected to be located on essentially the same site and may be joined in some manner, design considerations need to include the NRC safety concerns regarding the sharing of common utilities and services and physical interaction between the two facilities. This consideration for the sharing of systems and facility interfaces should also be considered for other new SNFP Path Forward facilities, such as the cold vacuum drying facility. ● The ability for ready retrieval of MCOs should be provided for in the design of the CSB. ● The design for Safety Class 1 SSCs of the CSB and the other new SNFP Path Forward facilities (with the exception to activities internal to K Basins) should give consideration to the most severe credible tornado and tornado missile taking into consideration the most recent revision of Regulatory Guide 1.76, <i>Design Basis Tornado for Nuclear Power Plants</i>, SECY-93-087, <i>Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Designs</i>, and NUREG/CR-4461, <i>Tornado Climatology of the Contiguous United States</i>, (revisions under consideration for SRP 3.5.1.4, Revision 2.)
72 124	Criteria for nuclear criticality safety	None (see the discussion provided in Table 6 b, for SRP 9 1.2, Revision 3, relative to the K _{eff} acceptance criterion used for criticality calculations)	<p>Sciencetech:</p> <p>DOE and NRC requirements are comparable.</p> <p><u>Actions for Consideration:</u> None.</p>

Table 5.c Title 10 CFR Summary-Level Comparisons of Part 72 Technical/Safety Issues

Sect.	Title	Additional Information	Conclusions
72.126	Criteria for radiological protection	<p>For §72.126(a), Exposure control, refer to Table 5.a, the 10CFR20 discussion for additional information.</p> <p>In regards to §72.126(b), Radiological alarm systems, 10CFR835, Occupational Radiation Protection, §§835.403, 835.404 and 835.501 provide additional detail on control of accessible areas during changing radiological conditions.</p> <p>For §72.126(d), Effluent control, refer to Table 5.a, the 10CFR20 discussion and to Table 5.b, the §50.34a discussion for additional information on ALARA considerations. Refer to §§72.104 and 72.106 for further information on exposure limits.</p>	<p>Sciencetech:</p> <p>DOE and NRC requirements are generally comparable. The six specific radiological design requirements stated are generally addressed by DOE requirements. However, NRC requirements are more specific concerning the control of access to areas of potential contamination or high radiation.</p> <p>DOE and NRC requirements for providing alarms are generally comparable. However, the requirement in DOE 6430.1A-0110-99.08 is not as clear as the NRC's requirement regarding what radiological parameters should be alarm setpoints.</p> <p>NRC exposure limits in §72.104 are more restrictive than DOE requirements. The NRC design basis accident dose limit of 5 rem in §72.106 is more restrictive than the DOE criteria. DOE and NRC both cite ALARA as the philosophy for normal operational releases.</p> <p><u>Actions for Consideration:</u> Refer to Table 5.a, 10CFR20, and §§72.104 and 72.106.</p>
72.128	Criteria for spent fuel, high-level radioactive waste, and other radioactive waste storage and handling	None	<p>Sciencetech:</p> <p>DOE and NRC requirements are comparable.</p> <p><u>Actions for Consideration:</u> None.</p>
72.130	Criteria for decommissioning	None.	<p>Sciencetech:</p> <p>DOE and NRC requirements are comparable.</p> <p><u>Actions for Consideration:</u> None.</p>
72.140	Quality assurance requirements	<p>Subpart G, <i>Quality Assurance</i>, to 10CFR72, which includes §§72.140 through 72.176, was addressed in the Sciencetech document prepared for DOE-HQ in June 1995, titled <i>Comparison of NRC and DOE Requirements for Independent Spent Nuclear Fuel Storage Facilities</i>.</p> <p>In consideration of DOE Order 4700.1, <i>Project Management System</i>, the DOE requirements are equivalent to the NRC requirements. Refer to Table 5.b, Appendix B, and Attachment A of this evaluation for additional information.</p>	<p>Sciencetech:</p> <p>Although organized and worded differently, the DOE QA rule requirements are intended to be generally consistent with basic nuclear industry and NRC QA requirements.</p> <p>For new SNFP Path Forward facilities, the DOE QA rule is not specific on when QA programs are required to be submitted and approved.</p> <p><u>Actions for Consideration:</u> None. (Also refer to Table 5.b, Appendix B, and Attachment A of this evaluation for additional information.)</p>

Table 5.c Title 10 CFR Summary-Level Comparisons of Part 72 Technical/Safety Issues

Sect.	Title	Additional Information	Conclusions
72.142	Quality assurance organization	The combination of §830.120 and the WHC <i>Quality Assurance Manual</i> (WHC-QA-4-2), OR 1.0, Revision 4, Section 3.1, are equivalent to the NRC requirements. Refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.	<p>Sciencetech:</p> <p>The delegation of QA Program responsibilities to contractors is dealt with in a consistent manner by DOE and NRC. Although 10CFR830 does not clearly state that contractors include subcontractors, this is done under 10CFR820.</p> <p>The DOE and NRC requirements for documenting organizational QA responsibilities are comparable.</p> <p>The DOE and NRC requirements on individual and organizational authority and freedom are comparable.</p> <p>DOE does not specifically require that persons and organizations performing QA functions (except for personnel with authority to stop work) have sufficient independence from cost and schedule when opposed to safety considerations.</p> <p>Actions for Consideration: None. (Also refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.)</p>

Table 5.c Title 10 CFR Summary-Level Comparisons of Part 72 Technical/Safety Issues

Sect.	Title	Additional Information	Conclusions
72.144	Quality assurance program	<p>The combination of §830.120 and WHC procedures (specifically Section 9.0, Revision 1, <i>Safety Classification of Structures, Systems, and Components</i>, of WHC-CH-4-46, and QR 9.0, Revision 4, <i>Control of Processes</i>, of WHC-CH-4-2) are equivalent to the NRC requirements. Refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.</p>	<p>Scientech:</p> <p>The DOE QA rule does not incorporate the NRC sense of urgency to get a QAP in place as early as feasible in a project. However, the DOE project management order emphasizes the need for QA early in the project life cycle.</p> <p>The general intent of the DOE SAR order requirement to have a QA program defined in the SAR assists in ensuring that QA is considered early in a project. Nevertheless, a description of a QA program in the SAR is not sufficient to ensure early documentation and implementation of a QA program.</p> <p>The DOE and NRC requirements for SSC identification are comparable. DOE relies on a general definition of "items" to address what the industry refers to as SSCs. As stated in the rule, the identification requirement is somewhat ambiguous. Some relief from this shortcoming is available from DOE orders, especially DOE 5480.23 on safety analysis reports (SAR). Nevertheless, SARs are not required to be in place as early as QAPs.</p> <p>The NRC requirements emphasize that the controlling processes must be provided through the QA program and that the approved designs must be maintained. The DOE language is not as clear, allowing that the controlling processes may be implemented by the line organization rather than through the QA program.</p> <p>The DOE requirements are not as clear as the NRC requirements regarding the meaning of controlled conditions.</p> <p>Both NRC and DOE requirements allow a graded approach.</p> <p>Except for the NRC's use of the term "indoctrination" and the DOE's use of the term "qualified", the training requirements are comparable.</p> <p>The NRC and DOE requirements for assessments of the QA program are comparable.</p> <p>Actions for Consideration: None. (Also refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.)</p>

Table 5.c Title 10 CFR Summary-Level Comparisons of Part 72 Technical/Safety Issues

Section	Title	Additional Information	Conclusions
72 146	Design control	<p>The combination of §830.120 and MHC procedures (specifically QR 3.0, Revision 5, Design Control, Section 3.0, of MHC-CM-4-2; MHC-CM-6-1, [manual revision dated August 8, 1995], Standard Engineering Practices; and MHC-IP-1026, [manual revision dated July 10, 1995], Engineering Practice Guideline) are equivalent to the MRC requirements. Refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.</p> <p>DOE Order 6430.1A, Section 1300-3.4.2, Environmental Qualification of Equipment, requires that safety class items (Safety Class 1) be able to perform their safety function under DBA conditions, and that the temperature, pressure and humidity environments be based on the most severe postulated accident affecting the particular item.</p>	<p>ScienceTech:</p> <p>The DOE and MRC requirements to establish measures to convert regulatory and design requirements into specifications, drawings, procedures, and instructions are essentially the same.</p> <p>DOE does not contain the specific requirement for measures to ensure the design documents include appropriate quality standards and that deviations from standards are controlled.</p> <p>DOE and MRC requirements are comparable regarding the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of SSCs.</p> <p>The MRC requirements for managing design interfaces are comparable to the DOE requirements.</p> <p>DOE and MRC design verification requirements are comparable.</p> <p>The DOE rule does not specify qualification testing under the most adverse design conditions.</p> <p>The DOE rule does not provide the specific list of functions and phenomena listed in this section, for which design control measures are to be applied.</p> <p>Control of design changes is comparable in the DOE and MRC requirements, and Attachment A of this report for additional information.)</p> <p>Actions for Consideration: None. (Also refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.)</p>
72 148	Procurement document control	<p>The combination of §830.120 and MHC procedures (specifically QR 4.0, Revision 4, Procurement Document Control, of MHC-CM-4-2) are equivalent to the MRC requirements. Refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.</p>	<p>ScienceTech:</p> <p>DOE and MRC requirements on procurement document content are comparable.</p> <p>The DOE and MRC requirements for ensuring that contractors and subcontractors meet appropriate QA requirements are comparable. However, DOE does not specifically require that subcontractors develop quality assurance programs that meet the provisions of the DOE QA rule.</p> <p>Actions for Consideration: None. (Also refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.)</p>
72 150	Instructions, procedures, and drawings	<p>Refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.</p>	<p>ScienceTech:</p> <p>The DOE and MRC requirements on use of instructions, procedures, and drawings are comparable.</p> <p>The DOE and MRC acceptance criteria requirements are comparable.</p> <p>Actions for Consideration: None. (Also refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.)</p>

Table 5.c Title 10 CFR Summary-Level Comparisons of Part 72 Technical/Safety Issues

Sect.	Title	Additional Information	Conclusions
72.152	<i>Document control</i>	Refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.	<p>Sciencetech: DOE and NRC requirements for document control are comparable. Actions for Consideration: None. (Also refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.)</p>
72.154	<i>Control of purchased material, equipment, and services</i>	The combination of 8830.120 and WHC procedures (specifically QR 7.0, Revision 4, <i>Control of Purchased Items or Services</i> , and QI 7.2, Revision 5, <i>Supplier Evaluation</i> , of WHC-CH-4-2) are equivalent to the NRC requirements. Refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.	<p>Sciencetech: NRC and DOE procurement requirements are comparable. DOE and NRC requirements for evaluating and inspecting contractors and suppliers and their products are comparable. DOE does not specifically require documentary evidence of conformance to procurement specifications before installation or use of material and equipment, as does the NRC. DOE's requirement for assessment [re: control of quality by contractors and subcontractors] is more general and does not specifically name contractor and supplier QC programs for assessment. Actions for Consideration: None. (Also refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.)</p>
72.156	<i>Identification and control of materials, parts, and components</i>	Refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.	<p>Sciencetech: DOE and NRC material control requirements are comparable. Actions for Consideration: None. (Also refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.)</p>
72.158	<i>Control of special processes</i>	Refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.	<p>Sciencetech: DOE and NRC requirements to provide controls over special processes are comparable. Actions for Consideration: None. (Also refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.)</p>

Table 5.c Title 10 CFR Summary-Level Comparisons of Part 72 Technical/Safety Issues

Sect	Title	Additional Information	Conclusions
72.160	Licensee inspection	The combination of 10 CFR 120 and WMC procedures (specifically OR 10.0, Revision 3, <i>Inspection and Surveillance</i> , of WMC-CH-4-2) are equivalent to the NRC requirements. Refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.	<p>Sciencetech:</p> <p>DOE and NRC quality inspection requirements are comparable.</p> <p>Inspector independence is required by DOE and NRC.</p> <p>DOE and NRC test and inspection requirements are comparable.</p> <p>DOE does not require alternative or indirect approaches to inspection when direct inspection is not possible.</p> <p>Concurrent or complementary use of inspections and process monitoring to ensure quality is not addressed by DOE requirements.</p> <p><u>Actions for Consideration:</u> None. (Also refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.)</p>
72.162	Test control	Refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.	<p>Sciencetech:</p> <p>DOE and NRC have comparable testing requirements.</p> <p><u>Actions for Consideration:</u> None. (Also refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.)</p>
72.164	Control of measuring and test equipment	Refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.	<p>Sciencetech:</p> <p>DOE and NRC have comparable calibration requirements.</p> <p><u>Actions for Consideration:</u> None. (Also refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.)</p>
72.166	Handling, storage, and shipping control	Refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.	<p>Sciencetech:</p> <p>DOE and NRC have comparable handling, storage, shipping, and preservation requirements.</p> <p><u>Actions for Consideration:</u> None. (Also refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.)</p>

Table 5.c Title 10 CFR Summary-Level Comparisons of Part 72 Technical/Safety Issues

Sect.	Title	Additional Information	Conclusions
72.168	<i>Inspection, test, and operating status</i>	The combination of §830.120 and WMC procedures (specifically QR 14.0, Revision 5, <i>Inspection, Test and Operating Status</i> , of WMC-CM-4-2) are equivalent to the NRC requirements. Refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.	<p>Sciencetech:</p> <p>DOE and NRC have comparable requirements regarding the status or results of inspections and tests. However, the NRC specifically requires marking items to indicate the status of inspections and tests.</p> <p>The DOE rule does not require labeling of equipment which has satisfactorily passed inspections and tests as specified by the NRC. DOE orders allow, but do not require, identifying items that have satisfactorily passed inspections and tests.</p> <p>Although the DOE QA rule is not specific on locking and tagging requirements, the DOE and NRC requirements on identifying the operating status of SSCs are comparable.</p> <p><u>Actions for Consideration:</u> None. (Also refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.)</p>
72.170	<i>Nonconforming materials, parts, or components</i>	Refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.	<p>Sciencetech:</p> <p>DOE and NRC requirements for controlling nonconforming items are comparable.</p> <p><u>Actions for Consideration:</u> None. (Also refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.)</p>
72.172	<i>Corrective action</i>	Refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.	<p>Sciencetech:</p> <p>DOE and NRC have comparable requirements regarding corrective actions.</p> <p>DOE and NRC root-cause analysis and recurrence prevention requirements are comparable.</p> <p><u>Actions for Consideration:</u> None. (Also refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.)</p>
72.174	<i>Quality assurance records</i>	<p>The types of records required are discussed in §830.120, <i>Quality Assurance Requirements</i>, and in existing WMC procedures (WMC-CM-4-2, <i>Quality Assurance Manual</i>, QR 17.0, Revision 2, <i>Quality Assurance Records</i>). Section 1.5, Revision 4, <i>Qualifications and Certifications</i>, of WMC-CM-4-5, <i>Quality Assurance Qualifications and Instructions</i>, provides requirements for qualifying QA auditors and maintenance of records pertaining to their qualification. Section 1.3, Revision 5, <i>Training</i>, of this manual pertains to training for QA personnel and maintenance of their training records in a QA Inspection and Test Personnel Training Plan.</p> <p>Refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.</p>	<p>Sciencetech:</p> <p>The DOE rule is not as specific as the NRC requirements in indicating the type of records required.</p> <p>DOE does not specifically require including the qualifications of QA auditors with the records that it develops and maintains.</p> <p>DOE and NRC requirements for inspection and test records are comparable.</p> <p>DOE and NRC requirements for records identification and retrieval are comparable.</p> <p><u>Actions for Consideration:</u> None. (Also refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.)</p>

Table 5.c Title 10 CFR Summary-Level Comparisons of Part 72 Technical/Safety Issues

Sect.	Title	Additional Information	Conclusions
72 176	<i>Audits</i>	Refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.	<p>Sciencetech:</p> <p>DOE and NRC requirements to conduct periodic independent audits (i.e., assessments) are comparable.</p> <p>DOE and NRC requirements for independent audits are comparable.</p> <p>DOE and NRC requirements for reviewing audit results are comparable.</p> <p>DOE and NRC requirements for follow-up action from independent audits are comparable.</p> <p>Actions for Consideration: None. (Also refer to Table 5.b, Appendix B, and Attachment A of this report for additional information.)</p>
72 180	<i>Physical security plan</i>	<p>Subpart H, <i>Physical Protection</i>, to 10CFR72, which includes §§72.180 through 72.186, was addressed in the Sciencetech document prepared for DOE-HQ in June 1995, titled <i>Comparison of NRC and DOE Requirements for Independent Spent Nuclear Fuel Storage Facilities</i>.</p> <p>Refer to the response provided for SECY-95-104 in Table 6.b for additional information.</p>	<p>Sciencetech:</p> <p>DOE and NRC requirements are generally comparable. The NRC requirements for physical protection at fixed facilities in 10CFR73 are applicable to an ISFSI. The applicable portions of 10CFR73 relevant to spent fuel include §§73.20, 73.37, 73.40, 73.50 and Appendix C. DOE 5630.11B, Section 7, and DOE 6430.1A-110-13.1 and 283-1c, all state that levels of protection appropriate to particular security interests shall be provided in a graded manner in accordance with the potential risks to national security and the health and safety of employees or the public. This approach is commensurate with the NRC's material categories.</p>
72 180 cont'd	<i>Physical security plan</i>		<p>NRC requires a two-part plan for security measures for physical protection. Part I must address plans for compliance with the applicable requirements of 10CFR73, including the design for physical protection. Comparable DOE requirements are contained in DOE 5610.14 and 5630.15. The design for physical protection is addressed in the conclusions for §72.182. The safeguards contingency plan is addressed in the conclusions for §72.184. Part II must list tests, inspections, audits, and other means to be used to demonstrate compliance. Comparable DOE requirements are contained in DOE 5630.12A and 5630.16A.</p> <p>Actions for Consideration: None.</p>
72 182	<i>Design for physical protection</i>	Refer to the response provided for SECY-95-104 in Table 6.b for additional information.	<p>Sciencetech:</p> <p>NRC and DOE requirements are generally comparable. DOE 5632.1C addresses almost all of the §73.50 requirements.</p> <p>Actions for Consideration: None.</p>

Table 5.c Title 10 CFR Summary-Level Comparisons of Part 72 Technical/Safety Issues

Sect.	Title	Additional Information	Conclusions
72.184	<i>Safeguards contingency plan</i>	None	<p>Sciencetech:</p> <p>Safeguards contingency plans required by NRC are equivalent to DOE response plans, developed by the security contractors at DOE sites. Response plans contain similar information as required by 10CFR73, Appendix C. The requirements in 10CFR73, Appendix C, and in DOE 5630.14A, 5630.11B and 5632.1C for site safeguard and security plans, key program elements and protection strategies are comparable.</p> <p><u>Actions for Consideration:</u> None.</p>
72.186	<i>Change to physical security and safeguards contingency plans</i>	None.	NA
72.190	<i>Operator requirements</i>	<p>Subpart I, <i>Training and Certification</i>, to 10CFR72, which includes §§72.190 through 72.194, was addressed in the Sciencetech document prepared for DOE-HQ in June 1995, titled <i>Comparison of NRC and DOE Requirements for Independent Spent Nuclear Fuel Storage Facilities</i>.</p> <p>This section does not establish design or construction requirements. Operational issues are not within the scope of this NRC nuclear safety equivalency activity.</p>	NA
72.192	<i>Operator training and certification program</i>	This section does not establish design or construction requirements. Operational issues are not within the scope of this NRC nuclear safety equivalency activity.	NA
72.194	<i>Physical requirements</i>	This section does not establish design or construction requirements. Operational issues are not within the scope of this NRC nuclear safety equivalency activity.	NA
72.200	<i>Provision of MRS information</i>	Subpart J, <i>Provisions of MRS Information to State Governments and Indian Tribes</i> , to 10CFR72, which includes §§72.200 through 72.206, was not addressed in the Sciencetech document prepared for DOE-HQ in June 1995, titled <i>Comparison of NRC and DOE Requirements for Independent Spent Nuclear Fuel Storage Facilities</i> , because this subpart specifically addresses monitored retrievable storage installations and was judged by Sciencetech to not be important for the purposes of the SNFP Path Forward. In addition, there are no substantive requirements in this subpart that would affect design or construction or the health and safety of the public or the worker.	NA
72.202	<i>Participation in license reviews</i>	None	NA
72.204	<i>Notice to states</i>	None.	NA
72.206	<i>Representation</i>	None	NA

Table 5.c Title 10 CFR Summary-Level Comparisons of Part 72 Technical/Safety Issues

Sect.	Title	Additional Information	Conclusions
72.210	<i>General license issued</i>	Subpart K to 10CFR72, which includes §§72.210 through 72.220, was not addressed in the Scientech document prepared for DOE-HQ in June 1995, titled <i>Comparison of NRC and DOE Requirements for Independent Spent Nuclear Fuel Storage Facilities</i> . Because this subpart addresses a special type of license for facilities at existing commercial nuclear power plants and was judged by Scientech to not be important for the purposes of the SNFP Path Forward. In addition, this general license is limited to storage of spent fuel in casks specifically approved under the provisions of Subpart K.	NA
72.212	<i>Conditions of general license issued</i>	It should be noted that Subpart K (§72.212) does require 1) the cask storage pads and areas to be designed to adequately support the static load of the stored casks; 2) the cask design bases to envelop the analyses of the applicable earthquake intensity and tornado missiles; and 3) the protection of the spent fuel against the design basis threat of radiological sabotage in accordance with the same provisions and requirements as set forth in the licensee's physical security plan per 10CFR73.55. These latter three items are addressed as part of §§72.92, 72.120, 72.122 and 72.182. Refer to the response provided for SECT-95-104 in Table 6.b for additional information.	<u>Actions for Consideration:</u> Refer to §§72.92 and 72.122.
72.214	<i>List of approved spent fuel storage casks</i>	None	NA
72.216	<i>Reports</i>	None	NA
72.218	<i>Termination of licenses</i>	None	NA
72.220	<i>Violations</i>	Refer to Table 4, §50.110 discussion.	None. (Also refer to Table 4, §50.110 discussion.)
72.230	<i>Procedures for spent fuel storage cask submittals</i>	Subpart L to 10CFR72, which includes §§72.230 through 72.240, was addressed in the Scientech document prepared for DOE-HQ in June 1995, titled <i>Comparison of NRC and DOE Requirements for Independent Spent Nuclear Fuel Storage Facilities</i> . However, Subpart L, <i>Approval of Spent Fuel Storage Casks</i> , is outside of the NRC nuclear safety equivalency demonstration scope. DOE's K Basin Spent Nuclear Fuel Project - Regulatory Policy, in regards to nuclear safety equivalence to comparable NRC-licensed facilities only applies to new SNFP Path Forward facilities.	NA
72.232	<i>Inspection and tests</i>	None	NA
72.234	<i>Conditions of approval</i>	None	NA

Table 5.c Title 10 CFR Summary-Level Comparisons of Part 72 Technical/Safety Issues

Sect.	Title	Additional Information	Conclusions
72.236	Specific requirements for spent fuel storage cask approval	None	NA
72.238	Issuance of an NRC Certificate of Compliance	None	NA
72.240	Conditions for spent fuel storage cask reapproval	None	NA

Table 6.a Technical/Safety Comparisons for Potentially Applicable NRC Regulatory Guides

No.	Title	Comment
Power Reactors		
1.7	<i>Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident, Revision 2</i>	See the response provided for §50.44 in Table 5.b.
1.13	<i>Spent Fuel Storage Facility Design Basis, Revision 1 (for comment)</i>	<p>This guide is directed at spent fuel storage pools to be located within the reactor facility.</p> <p><u>Conclusions:</u> As this guide is directed at in plant pool storage, a detailed comparison is not provided. This position is taken as 10CFR72, Subpart F, provides more appropriate design criteria for an ISFSI, and much of Regulatory Guide 1.13 is directed at issues specific to pool storage which is not the selected concept for the CSB. General guidance of interest to the CSB provided in Regulatory Guide 1.13 and the location of this guidance in 10CFR72 for ISFSIs is as follows:</p> <p><u>RG 1.13</u></p> <p><u>10CFR72</u></p> <p>The spent fuel storage facility should be designed to Category I seismic requirements §572.102(f)(1) and 72.122(b)(2)</p> <p>The facility should be designed to keep missiles generated by tornadoic winds from contacting fuel. §72.122(b)(2)</p> <p>A controlled leakage building should enclose the fuel pool. §72.122(h)(3)</p> <p>Regulatory Guide 1.13 also suggests restrictions on the movement of heavy loads, such as a shipping cask, over the spent fuel. These restrictions are not practical for the CSB as the MCO Handling Machine (MHM) cask and MCO must be moved over MCOs already stored.</p> <p><u>Actions for Consideration:</u> None. (Refer to the responses provided for §572.102 and 72.122 in Table 5.c.)</p>
1.26	<i>Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants, Revision 3</i>	<p>The guide provides guidance in assigning ASME Section III, Class 2 and 3, and Section VIII classifications.</p> <p><u>Conclusions:</u> For the new SNFP Path Forward facilities, gas and fluid systems assigned Safety Class 1 by 10CFR-4.46, Section 9, Revision 1, <i>Safety Classification of Structures, Systems, and Components</i>, or through the use of "important to safety" by §72.3, this guide could be useful in making the appropriate ASME classification as this guidance is not provided in DOE Order 6430.1A, <i>General Design Criteria</i>.</p> <p><u>Actions for Consideration:</u> Consider having the AEs for the new SNFP Path Forward facilities use Regulatory Guide 1.26 to assist in assigning the appropriate code class to ASME Section III components and systems.</p>
1.29	<i>Seismic Design Classification, Revision 3</i>	<p>This guide provides guidance on the assignment of seismic design classification.</p> <p><u>Conclusions:</u> For the SNFP Path Forward, the existing process of identifying the safety classification of systems by WHC-CH-4-46, Section 9, Revision 1, <i>Safety Classification of Structures, Systems, and Components</i>, and then assigning the appropriate seismic design requirements by SD-4.1, Revision 12, <i>Standard Architectural/Civil Design Criteria</i>, is an equivalent process. Additional technical issues associated with seismic design, including the use of DOE Order 5480.28, <i>Natural Phenomena Hazards Mitigation</i>, and companion DOE standards, are presented in Table 5.c. In response to §572.92, 72.102 and 72.122.</p> <p><u>Actions for Consideration:</u> None.</p>

Table 6.a Technical/Safety Comparisons for Potentially Applicable NRC Regulatory Guides

No.	Title	Comment
1.32	Criteria for Safety-Related Electric Power Systems for Nuclear Plants, Revision 2	<p>This guide adopts, with comment, IEEE Std 308-1974, <i>IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations</i>, as an acceptable basis for complying with GDC 17 and 18 with respect to the design, operation, and testing of electric power systems. The current issue of this standard is IEEE Std 308-1991. Because Regulatory Guide 1.32 has not been revised subsequent to the issuance of the latest revision of IEEE Std 308, a detailed review of the NRC comments on the earlier issue of the standard has not been performed. As implementation of IEEE Std 308 is required by DOE Order 6430.1A, <i>General Design Criteria</i>, no action is required.</p> <p>Action for Consideration: None.</p>
1.52	Design, Testing, and Maintenance for Post Accident Atmosphere Cleanup Systems, Revision 2	<p>Conclusions: Air cleanup will be important to some aspects of the SNPP Path Forward. However, this guide is quite old (March 1978) and compliance to ANSI/ASME N509-1989, <i>Nuclear Power Plant Air-Cleanup Units and Components</i>, for design and ANSI/ASME N510-1989, <i>Testing of Nuclear Air Treatment Systems</i>, for testing should be adequate. As both ANSI/ASME N509 and N510 are required by DOE Order 6430.1A, Chapter 15, no action is required.</p> <p>See the response provided for Regulatory Guide 1.140 for air cleanup considerations for Safety Class 2 and 3 systems.</p> <p>Action for Consideration: None.</p>
1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants, Revision 1	<p>Conclusions: This guide provides normalized response spectra that can be used in the design of facilities underlain by either rock or soil. Regulatory Guide 3.48, <i>Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation (Dry Storage)</i>, Revision 1, states that if other spectra are to be used for ISFSIs, their use should be justified.</p> <p>Action for Consideration: Consider justifying seismic response spectra used for Safety Class 1 SSCs for new SNPP Path Forward facilities if other than those included in Regulatory Guide 1.60 are to be used. This is in accordance with Regulatory Guide 3.48, Revision 1.</p>
1.61	Damping Values for Seismic Design of Nuclear Power Plants, Revision 0	<p>This Regulatory Guide provides the damping values to be used for dynamic seismic analyses of the Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) for nuclear power plants. Values are provided for large and small diameter pipe, welded and bolted steel structures, and prestressed and reinforced concrete structures. The guide allows for higher damping values if documented test data are provided to support the higher values.</p> <p>SDC-4.1, <i>Standard Architectural/Civil Design Criteria</i>, Revision 12, requires in Section 3.1.2.2, that Safety Class 1 SSCs for non-reactor facilities use damping values that conform to UCRL-15910, <i>Design and Evaluation Guidelines for DOE Facilities Subjected to Natural Phenomena Hazards</i>. For Safety Class 2 SSCs, Section 3.2.2 references UCRL-15910, ASE 4-86, <i>Seismic Analysis of Safety-Related Nuclear Structures and Commentary on Standard for Seismic Analysis of Safety-Related Nuclear Structures</i>, and Regulatory Guide 1.61 for the damping values for structures and provides a critical damping value of 5 percent for equipment.</p> <p>Conclusions: For Safety Class 1 SSCs, the damping values required by UCRL-15910 are less restrictive than those of Regulatory Guide 1.61. The guidance provided in SDC-4.1 for Safety Class 2 structures is not clear as the three documents referenced have conflicting damping values and the SDC does not provide guidance as to which ones to use. For example, the values provided in Table 4-6 of UCRL-15910 are not in agreement with Table 1 of Regulatory Guide 1.61. The damping value of 5 percent provided in SDC-4.1 for Safety Class 2 equipment is less restrictive than the values provided in Regulatory Guide 1.61, which has SSE damping values of 2 to 3 percent for equipment and piping.</p> <p>Action for Consideration: Consider justifying damping values for seismic design for Safety Class 1 SSCs for new SNPP Path Forward facilities if other than those included in Regulatory Guide 1.61 are to be used.</p>

Table 6.a Technical/Safety Comparisons for Potentially Applicable NRC Regulatory Guides

No.	Title	Comment
1 75	<i>Physical Independence of Electrical Systems, Revision 2</i>	<p><u>Conclusions:</u> The physical separation requirements of Regulatory Guide 1.75 should be applied to Safety Class 1 redundant electrical systems. However, Revision 2 of the guide makes reference to the IEEE Std 384-1974, <i>IEEE Trial-Use Standard Criteria for Separation of Class 1E Equipment and Circuits</i>. The current issue of this guide is IEEE Std 384-1992, <i>IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits</i>. The IEEE standard is somewhat broader in scope than the direction of Regulatory Guide 1.75 in that the standard also deals with electrical isolation. Because Regulatory Guide 1.75 has not been revised subsequent to the issuance of the latest revision of IEEE Std 384, a detailed review of the NRC comments regarding the standard has not been performed. As IEEE Std 384 is required by DOE Order 6430.1A, Chapter 16, <i>Electrical</i>, no action is required.</p> <p><u>Actions for Consideration:</u> None.</p>
1 84	<i>Design and Fabrication Code Case Acceptability ASME Section III, Division 1, Revision 30</i>	<p><u>Conclusions:</u> This guide is used to document NRC approval of design and fabrication code cases for ASME Section III, Division 1. This, relieves licensees of the need to request specific code case approval as required by 850.55a(a)(3).</p> <p><u>Actions for Consideration:</u> Give consideration to having the AEs for the new SNRP Path Forward facilities review the NRC positions in Regulatory Guide 1.84 on ASME code cases before using such cases for Safety Class 1 application. Reflect the NRC positions on the applicable code cases in the design of the facilities. Where no NRC position is stated in regards to the acceptance of a code case, that code case may be used as approved.</p>
1 85	<i>Materials Code Case Acceptability ASME Section III, Division 1, Revision 30</i>	<p><u>Conclusions:</u> Same as immediately above for Regulatory Guide 1.84, except as related to ASME Section III, Division 1 material considerations.</p> <p><u>Actions for Consideration:</u> Give consideration to having the AEs for the new SNRP Path Forward facilities review the NRC positions in Regulatory Guide 1.85 on ASME code cases before using such cases for Safety Class 1 application. Reflect the NRC positions on the applicable code cases in the design of the facilities. Where no NRC position is stated in regards to the acceptance of a code case, that code case may be used as approved.</p>
1 89	<i>Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants, Revision 1</i>	See the response provided for 850.49 in Table 5 b and Attachment A.
1 120	<i>Fire Protection Guidelines for Nuclear Power Plants, Revision 1 (for comment)</i>	<p><u>Conclusions:</u> This guide does not reflect current NRC guidance and requirements relative to fire protection and implementation of 10CFR50, Appendix R, Branch Technical Position CKB 9.5-1 provides this guidance. See the response provided for 10CFR50, Appendix R in Table 5 b.</p> <p><u>Actions for Consideration:</u> None. (See the response provided for 850.48 and 10CFR50, Appendix R in Table 5 b.)</p>

Table 6.a Technical/Safety Comparisons for Potentially Applicable NRC Regulatory Guides

No.	Title	Comment
1 128	<i>Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants, Revision 1</i>	<p><u>Conclusions:</u> IEEE Std. 484-1987, <i>IEEE Recommended Practice for Installation, Design and Installation of Large Lead Storage Batteries for Generating Stations and Substations</i>, provides more current NRC requirements for station batteries. Because Regulatory Guide 1.128 has not been revised subsequent to the issuance of the latest revision of IEEE Std 484, a detailed review of the NRC comments on the earlier issue of standard has not been performed.</p> <p><u>Actions for Consideration:</u> Consideration should be given to having the AEs for new SNFP Path Forward facilities incorporate the requirements of IEEE Std 484-1987, <i>IEEE Recommended Practice for Installation, Design and Installation of Large Lead Storage Batteries for Generating Stations and Substations</i>, into the design and installation of Safety Class 1 batteries.</p>
1 140	<i>Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants, Revision 1</i>	<p><u>Conclusions:</u> This guide applies to air filtration and air adsorption units installed to collect radioactive materials during normal operation, including anticipated operational occurrences so as to maintain compliance with §50.34a, §50.36a and 10CFR 50, Appendix I. As is the case for Regulatory Guide 1.52, this guide is quite old (March 1978) and compliance to ANSI/ASME N509-1989, <i>Nuclear Power Plant Air-Cleanup Units and Components</i>, for design and ANSI/ASME N510-1989, <i>Testing of Nuclear Air Treatment Systems</i>, for testing should be adequate. ANSI/ASME N509 is referenced in the CSB Functions and Requirements document. Both ANSI/ASME N509 and N510 are mentioned in DOE Order 6430.1A, <i>General Design Criteria</i>, Chapter 15, Section 1550-2.5.5, <i>Air Cleaning Devices</i>, states: "In-place testing design requirements shall meet all the recommendations of... ASME N510..." and "Adsorbers for nuclear or toxic applications should [as opposed to shall-note added] comply with... ASME N509... and ASME N510." DOE Order 6430.1A also states in Section 1300-3, <i>Safety Class Criteria</i>, 1300-3.6, <i>Testing</i>: "The facility design shall allow for routine in-place testing of HEPA filtration systems as outlined by ASME N510." The combination of the "should" statement and only ASME N510 being referenced under safety class criteria has created some confusion on the Hanford Site regarding the proper application of ANSI/ASME N509 and N510. Hence, for Regulatory Guide 1.140 equivalency, the AEs for the new SNFP Path Forward facilities should consider applying these standards to Safety Class 2 and 3 systems.</p> <p><u>Actions for Consideration:</u> Give consideration to having the AEs for new SNFP Path Forward facilities include ANSI/ANS N509-1989, <i>Nuclear Power Plant Air-Cleanup Units and Components</i>, and ANSI/ANS N510-1989, <i>Testing of Nuclear Air Treatment Systems</i>, for those HVAC systems needed to achieve onsite radiological limits or to implement the principals of ALARA (i.e., this applies to WHC Safety Class 2 and 3 systems).</p>
1 153	<i>Criteria for Power, Instrumentation, and Control Portions of Safety Systems, Revision 0</i>	<p><u>Conclusions:</u> This guide adopts, with comment, IEEE Std 603-1980, <i>IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations</i>, as providing a method for complying with NRC regulations with regard to the design, reliability, and testability of power, instrumentation, and control portions of safety systems. The current revision of this standard is IEEE Std 603-1991. This revision of the standard acknowledges comments made by the NRC relative to the 1980 issue. Because Regulatory Guide 1.153 has not been revised subsequent to the issuance of the latest revision of IEEE Std 603, a detailed review of the NRC comments on the earlier issue of the standard has not been performed.</p> <p><u>Actions for Consideration:</u> Consideration should be given to having the AEs for the new SNFP Path Forward facilities incorporate into the design for Safety Class 1 instrumentation and control systems, the requirements of IEEE Std 603-1991, <i>IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations</i>.</p>
1 155	<i>Station Blackout, Revision 0 (corrected August 1988)</i>	See the response provided for §50.63 in Table 5.b.

Table 6.a Technical/Safety Comparisons for Potentially Applicable NRC Regulatory Guides

No.	Title	Comment
1 158	<i>Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants. Revision 0</i>	<p><u>Conclusions:</u> This guide simply adopts IEEE Std 535-1986, <i>IEEE Standard for Qualification of Class 1E Lead Storage Batteries for Nuclear Power Generating Stations</i>. The only NRC comments relate to the implementation dates for use of the standard.</p> <p><u>Actions for Consideration:</u> Consideration should be given to having the AEs for new SNFP Path Forward Facilities incorporate the requirements of IEEE Std 535-1986, <i>IEEE Standard for Qualification of Class 1E Lead Storage Batteries for Nuclear Power Generating Stations</i>, for the qualification of Safety Class 1 lead storage batteries.</p>
1 160	<i>Monitoring the Effectiveness of Maintenance at Nuclear Power Plants. Revision 0</i>	See the response for \$50.65 in Table 5.b.
Division 2 Research and Test Reactors		These guides are not considered to be applicable to the SNFP Path Forward.
Division 3 Fuels and Materials Facilities		
3 18	<i>Confinement Barriers and Systems for Fuel Reprocessing Plants. Revision 0</i>	<p>This guide addresses confinement barriers to meet the requirements of 10CFR20 rather than those required to mitigate design basis accidents. This is addressed in Table 5.b, in response to \$50.34a.</p> <p><u>Actions for Consideration:</u> None.</p>
3 26	<i>Standard Format and Content of Safety Analysis Reports for Fuel Reprocessing Plants. Revision 0</i>	See the response provided in Table 5.b for \$50.34.
3 44	<i>Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation (Water-Basin Type). Revision 2</i>	This guide is not applicable to the dry storage option selected for the SNFP Path Forward. For this option, Regulatory Guide 3.48, addressed immediately below, is the applicable guide.
3 48	<i>Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation (Dry Storage). Revision 1</i>	See the response provided in Table 5.b for \$50.34 and Table 5.c for \$72.24.
3 49	<i>Design of an Independent Spent Fuel Storage Installation (Water-Basin Type). Revision 0</i>	This guide is not applicable to the dry storage option selected for the SNFP Path Forward. For this option, Regulatory Guide 3.60, addressed below, is the applicable guide.
3 50	<i>Standard Format and Content for a License Application to Store Spent Fuel and High-Level Radioactive Waste. Revision 1</i>	<p>This guide represents a standard format that is acceptable to the NRC staff for the application for a 10CFR72 license application.</p> <p><u>Conclusions:</u> Information specific to the design and construction of dry ISFISs is provided in greater detail in Regulatory Guide 3.48. See the response provided above for Regulatory Guide 3.48.</p> <p><u>Actions for Consideration:</u> None.</p>

Table 6.a Technical/Safety Comparisons for
Potentially Applicable NRC Regulatory Guides

No.	Title	Comment
3 53	<i>Applicability of Existing Regulatory Guides to the Design and Operation of an Independent Spent Fuel Storage Installation, Revision 0</i>	<p><u>Conclusions:</u> This guide was used to help establish the Regulatory Guides addressed in this table. Being used as a guide only, there is not a one-for-one comparison between this table and the guides listed in Regulatory Guide 3.53.</p> <p><u>Actions for Consideration:</u> None.</p>
3 60	<i>Design of an Independent Spent Fuel Storage Installation (Dry Storage), Revision 0</i>	<p>This guide invokes ANSI/ANS-57.9-1984, <i>Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type)</i>. This standard was subsequently reissued in 1992 and retitled as, <i>Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type)</i>.</p> <p><u>Conclusions:</u> The adopted ANSI/ANS standard is directed at commercial LWR spent nuclear fuel that is in the form of UO₂. As the SNFP Path Forward metal fuel presents risks that are not present with oxide fuel, the requirements of the ANSI/ANS standard should be viewed as a minimum set of requirements.</p> <p><u>Actions for Consideration:</u> Give consideration to having the AE for the CSB perform a review of the CSB design against ANSI/ANS-57.9-1992, <i>Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type)</i>, to identify any additional requirements that would be needed to demonstrate nuclear safety equivalency.</p>
3 61	<i>Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask, Revision 0</i>	This guide is not applicable to this equivalency evaluation as the SNFP Path Forward shipping cask is not considered as a new facility.
3 62	<i>Standard Format and Content for the Safety Analysis Report for Onsite Storage of Spent Fuel Storage Casks, Revision 0</i>	<p><u>Conclusions:</u> This guide is similar to Regulatory Guide 3.48, but is applicable to SNF storage that is to be stored on a site already licensed. For the SNFP Path Forward, review of Regulatory Guide 3.48 is adequate. No additional SNFP Path Forward review is required.</p> <p><u>Actions for Consideration:</u> None.</p>
Division 4 Environmental and Siting		See the response provided for 10CFR51 in Table 5a.
Division 5 Materials and Plant Protection		<p>These guides were not reviewed because DOE Order 5633.3B establishes adequate protection against threats, thefts, and sabotage relating to special nuclear materials and high-level radioactive waste. See the response provided for 10CFR73 in Table 5.a.</p> <p><u>Actions for Consideration:</u> None.</p>
Division 6 Products		Products include brachytherapy and well-logging equipment. This division was not found to be applicable to the SNFP Path Forward.
Division 7 Transportation		<p>Division 7 guides provide guidance for shipping radioactive material and shipping containers. The SNFP Path Forward shipping casks are not new facilities and, as such, they are not within the scope of this equivalency comparison.</p> <p><u>Actions for Consideration:</u> None.</p>

Table 6.a Technical/Safety Comparisons for Potentially Applicable NRC Regulatory Guides

No.	Title	Comment
Division 8 Occupational Health		<p><u>Conclusions:</u> Three Division 8 Regulatory Guides with possible design and construction impact are:</p> <p>RG 8.5: <i>Criticality and Other Interior Evacuation Signals</i></p> <p>RG 8.8: <i>Information Relative to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Reasonably Achievable</i></p> <p>RG 8.12: <i>Criticality Accident Alarm Systems</i></p> <p>The remaining guides were not reviewed in detail as they do not relate to design or construction activities</p>
8.5	<i>Criticality and Other Interior Evacuation Signals, Revision 1</i>	<p><u>Conclusions:</u> Regulatory Guide 8.5 adopts ANSI/ANS-2.3-1979 for criticality and other interior evacuation signals. DOE Order 6430.1A, <i>General Design Criteria</i>, also references this standard in Section 1300.6.5.5, <i>Warning and Alarm System Features</i>, for warning and alarm systems. However, this standard has been replaced by ANSI/ANS-8.3-1986, <i>Criticality Accident Alarm System</i>.</p> <p><u>Actions for Consideration:</u> Refer to Regulatory Guide 8.12.</p>
8.8	<i>Information Relative to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Reasonably Achievable, Revision 3</i>	<p><u>Conclusions:</u> The principal design impacts of this guide are in the areas of shielding, HVAC zoning, and in-plant radiation monitoring. Regulatory Guide 3.48 requires a discussion in Chapter 7 of the SAR of how the guidance of Regulatory Guide 8.8 is implemented. DOE Order 6430.1A, <i>General Design Criteria</i>, Section 1540.99.0.6, <i>System Installation</i>, references Regulatory Guide 8.8 for piping design considerations for systems that carry radioactive material.</p> <p><u>Actions for Consideration:</u> Consideration should be given to ensuring that the AEs for the new SNRP Path Forward Facilities incorporate the requirements of Regulatory Guide 8.8, Revision 3, into the design of these facilities.</p>
8.12	<i>Criticality Accident Alarm Systems, Revision 2</i>	<p>Regulatory Guide 8.12 adopts ANSI/ANS-8.3-1986, <i>Criticality Accident Alarm System</i>, for criticality alarm systems required by 70.24. The guide points out that 70.24 requires alarm coverage in each area in which licensed special nuclear material is handled, used, or stored, whereas the ANSI/ANS standard states that the need for criticality alarms must be evaluated for such areas. Regulatory Guide 8.12 states that if a departure from this aspect of 70.24 is desired, an exemption request must be submitted. It should be understood that this NRC position does not necessarily indicate that a departure from this aspect of the rule would be difficult to justify. It is simply a reminder that absent the granting of an exemption, NRC regulations must be followed as stated.</p> <p>DOE Order 5480.24, <i>Nuclear Criticality Safety</i>, states that the requirements of ANSI/ANS-8.3 relating to the need for an alarm system are not applicable to the order. The DOE order provides alternative criteria, which include risk considerations, for requiring criticality alarms and evacuation signals.</p>

Table 6.a Technical/Safety Comparisons for Potentially Applicable NRC Regulatory Guides

No.	Title	Comment
8 12 cont'd	<i>Criticality Accident Alarm Systems, Revision 2</i>	<p>It should also be noted that §72.124, <i>Criteria for nuclear criticality safety</i>, states: "Monitoring of dry storage areas where special nuclear material (SNM) is packaged in its stored configuration under a license issued under this subpart is not required." This is in keeping with the Regulatory Position in Regulatory Guide 8.12 that states: "... If such evaluation does not determine that a potential for criticality exists, as for example where the quantities or form of SNM make criticality practically impossible or where geometric spacing is used to preclude criticality, such as in some storage spaces for unirradiated nuclear power plant fuel, it is appropriate to request an exemption from §70.24.</p> <p>Conclusions: DOE Order 5480.24 adopts a position in regards to ANSI/ANS-8.3-1986, with discussion, that is in conflict with ANSI/ANS-8.3-1986 and the requirements of §70.24. As a result, nuclear safety equivalency is not directly achieved by DOE requirements for this aspect of §70.24. However, in consideration of the §72.124 and Regulatory Guide 8.12 statements immediately above, if the safety analyses for the initial fuel conditioning, staging, final conditioning, and interim storage of K Basins fuel demonstrate that the potential for criticality does not exist, then not including criticality accident alarm systems in the new facility designs would be compatible with the NRC intent.</p> <p>Part of the basis for this approach also resides in the tightly controlled configuration of the fuel, and fuel fragments/particles, throughout initial fuel conditioning, staging, final conditioning, and interim storage. Such configuration control will be provided by the fuel being continuously constrained to a specific geometry within the MCO baskets that maintains subcriticality. (Refer to WHC-SD-SNF-FRD-017, <i>Performance Specification for the Spent Nuclear Fuel Multi-Canister Overpack</i> (Womack 1995), for additional information.)</p> <p>Actions for Consideration: For the new SNFP Path Forward facilities, consider incorporating the requirements of ANSI/ANS-8.3-1986 into the designs. Also, consider using the MCO design, which includes the basket configuration, and the quantities/form of the K Basin fuel (special nuclear material) as a basis for not incorporating criticality accident alarm systems in the new SNFP Path Forward facilities, based on demonstrating through safety analyses documented in the associated SARs that criticality is not possible. In conjunction with this activity, demonstrate compliance with the criteria in DOE Order 5480.24 and ANSI/ANS-8.3-1986.</p>
Division 9 Anti-trust and Financial Review		Division 9 guides do not relate to the design or construction of new facilities. See the response provided for 10CFR50, Appendix L, in Table 5b.
Division 10 General		<p>This division of Regulatory Guides deals with the requirements for possession of byproduct, special nuclear, and source material for such applications as academic institutions and laboratory use, industrial radiography, and gamma irradiators. The SNFP Path Forward has adequate controls for these materials, as applicable. See the discussion provided in Table 5.a for 10 CFR Parts 30, 36 and 40.</p> <p>Actions for Consideration: None.</p>

Table 6.a Technical/Safety Comparisons for Potentially Applicable NRC Regulatory Guides

No.	Title	Comment
DG-1032 (draft, February 1995)	<i>Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion</i>	<p>This draft guide was developed to support proposed revisions to 10CFR100, Appendix A relative to geologic and seismic siting factors. It provides procedures acceptable to the NRC to conduct geologic, geophysical, seismological, and geotechnical investigations; identify and characterize seismic sources; conduct probabilistic seismic hazard analyses; and determine the Safe Shutdown Earthquake (SSE) for satisfying the requirements of the proposed §100.23. The guide includes appendices that contain a list of definitions; describe the procedure that can be used to determine the reference probability for the SSE exceedance level; discuss the development of a seismic hazard information base and the determination of the probabilistic ground motion level and controlling earthquakes; discuss site-specific geological, seismological, and geophysical investigations; describe a method to confirm the adequacy of existing seismic sources and source parameters as the basis for determining the SSE for a site; and describe procedures to determine the SSE.</p> <p>The guide was previously issued as draft guide DG-1015.</p> <p><u>Conclusions:</u> This draft guide provides insight into current NRC thinking on seismic issues.</p> <p><u>Actions for Consideration:</u> None. (See response provided for §72.92 in Table 5.c.)</p>
HF 608-4 (draft, March 1982)	<i>Training and Certification of Independent Spent Fuel Storage Installation Operators</i>	<p>This guide provides guidance for implementing 10CFR72, Subsection I, <i>Training and Certification of ISFSI Personnel</i>. The draft was issued in March 1982 and has not yet been issued in its approved version.</p> <p><u>Conclusions:</u> This draft guide does not contain guidance that would impact the design or construction of new SNPP Path Forward facilities.</p> <p><u>Actions for Consideration:</u> None.</p>
EP 034-4 (draft, December 1980)	<i>Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation</i>	<p>This draft guide provides parametric curves of heat generation vs. years after shutdown that may be used to determine long-term heat generation rates for spent nuclear fuel. Curves are provided for 33, 46 and 55 Gd/MTU.</p> <p><u>Conclusions:</u> The curves of the draft guide are not useful for SNPP Path Forward activities as they provide data for high-burnup commercial fuel (i.e., 33 to 55 Gd/MTU). The SNPP Path Forward fuel is in the range of 4 to 6 Gd/MTU.</p> <p><u>Actions for Consideration:</u> None.</p>

Table 6.b Technical/Safety Comparisons for Other Potentially Applicable NRC Documentation

Number	Subject	Comment
NUREG-0554	Single Failure Proof Cranes for Nuclear Power Plants	<p><u>Conclusions:</u> This document will be useful if a decision is made to purchase overhead cranes to meet single failure requirements. By NUREG-0612, this would eliminate the need to analyze for a cask drop; however, by DOE Order 5480.23, <i>Nuclear Safety Analysis Reports</i>, all credible events must be analyzed. The reliability data provided in NUREG-0612 for a single failure proof crane would not allow this event to be discounted as not being credible (Breth 1995). It is not the intent for the SNRP Path Forward to adopt NRC requirements that are less stringent than DOE requirements; hence, use of NUREG-0612 to eliminate a cask drop from consideration is not appropriate.</p> <p>This guide applies to only overhead cranes. It does not apply to the MCO Handling Machine used to move MCOs to their storage locations.</p> <p><u>Actions for Consideration:</u> None.</p>
NUREG-0575	Environmental Impact Statement on Handling and Storage of Spent Light Water Reactor Fuel	<p>This report discusses the results of an environmental assessment of alternatives for the handling and storage of spent nuclear fuel from LWRs with emphasis on developing a long-range plan. It considers the impacts of restriction or termination of power generation.</p> <p><u>Conclusions:</u> As stated in Section 1.0 of this document, review of environmental issues is not part of the equivalency review process. As such, no additional analysis is required for this evaluation.</p> <p><u>Actions for Consideration:</u> None.</p>
NUREG-0612 Generic Letter 81-07 Generic Letter 85-11	Control of Heavy Loads at Nuclear Power Plants, Resolution of Generic Technical Activity A-36 Control of Heavy Loads at Nuclear Power Plants Completion of Phase II of "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612.	<p>In regards to relevant design and construction activities, NUREG-0612 and Generic Letter 81-07 include the following:</p> <ul style="list-style-type: none"> • The accidental dropping of a heavy load onto fuel should not result in a k_{eff} of greater than 0.95 • The accidental dropping of a heavy load should not result in an offsite dose greater than 25 percent of 10CFR100 limits (i.e., 75 rem thyroid or 6.25 rem MBD). • Mechanical or electrical stops or interlocks should be provided to prevent movement of heavy loads over irradiated fuel. <p>In Generic Letter 85-11, the NRC stated that they were satisfied with the reduction of risk that had been achieved relative to the handling of heavy loads and, as a result, they would not be performing a detailed review of each licensee's implementation of Phase II of the program.</p> <p><u>Conclusions:</u> Regarding the above three items as related to SNRP Path Forward activities:</p> <ul style="list-style-type: none"> • By DOE requirements, the k_{eff} acceptance criterion is 0.98. This issue needs to be resolved for the SNRP Path Forward as identified for SRP 9.1.2. • The NRC acceptance criterion of 5 rem EDE of MRC-CH-4-46, Section 9.0, Revision 1, <i>Safety Classification of Structures, Systems, and Components</i>, bounds the MRC limit of 6.25 rem. The MRC value of 5 rem is for $>10^{-4}$ to $<10^{-3}$ events/yr (see the discussion provided in Table 5.c for \$72.106) which includes the estimated event frequency for a load drop at the CSB (Bergsman 1995). • Travel of heavy loads over spent fuel can not be avoided at the CSB. However, the proposed MCO Handling Machine is not an overhead crane. <p><u>Actions for Consideration:</u> Consideration should be given to invoking the NRC k_{eff} criticality safety limit of 0.95.</p>

Table 6.b Technical/Safety Comparisons for Other Potentially Applicable NRC Documentation

Number	Subject	Comment
NUREG-0695	<i>Environmental Impact Appraisal Related to the Renewal of Materials License SM-1265 for the Receipt, Storage and Transfer of Spent Fuel</i>	<p>This appraisal was prepared in compliance with 10CFR51 for the renewal of the license for the General Electric Morris Operation.</p> <p><u>Conclusions:</u> As stated in Section 1.0. of this document, review of environmental issues is not part of the equivalency review process. As such, no additional analysis is required for this evaluation.</p> <p><u>Actions for Consideration:</u> None</p>
NUREG-0700	<i>Guidelines for Control Room Design Reviews</i>	<p>NUREG-0700, issued in September 1981, summarizes NRC guidance relative to human factors review of control room design that is contained in NUREG-0660, <i>NRC Action Plan Developed as a Result of the TMI-2 Accident</i>, issued in May 1980, and NUREG-0737, <i>Clarification of TMI Action Plan Requirements</i>, issued in November 1980 (thus, the report numbering is not an indication of the order in which they were published). NUREG-0700 also introduced the need to perform task and systems analyses to support human factors engineering. NUREG-0737, Supplement 1, <i>Requirements for Emergency Response Capability</i>, issued in December 1982, encompassed the full scope of NUREG-0700 with the exception that the supplement reduced the scope of the function and task analysis to consideration of only emergency operations. The NRC also issued NUREG-0801, <i>Evaluation Criteria for Detailed Control Room Design Review</i>, which identifies four NRC evaluation phases used to determine the acceptability of an applicant's control room design review and resultant corrective actions. Standard Review Plan 18.1, <i>Control Room and its Appendix, Evaluation Criteria for Detailed Control Room Design Reviews (DEROR)</i>, were issued to provide the NRC staff guidance in the implementation of NUREG-0660, NUREG-0737, NUREG-0737, Supplement 1, and NUREG-0700.</p> <p>DOE requirements for human factors engineering are established in DOE Order 6430.1A, <i>General Design Criteria, Section 1300-12.4, General Human Factors Implementation</i>. Additional DOE guidance will be provided in <i>Human Factors Engineering Design Criteria, Volume I, General Criteria</i> when issued. Note that while DOE Order 6430.1A makes frequent reference to NUREG-0700, it does not specifically require that the human factors approach of the NUREG be implemented. DOE Order 5480.23, <i>Nuclear Safety Analysis Reports</i>, requires a chapter on human factors which is to include a discussion of considerations given to 1) the work environment, 2) the layout and design of controls and instrumentation, and the provisions of labeling, and 3) the determination that the furnished instrumentation, provisions for communication, and operational aids are sufficient to support the timely, reliable performance of human operations important to safety.</p> <p><u>Conclusions:</u> The conceptual design for the CSB identifies a location for a control room (Floor 1995). The nature of the activities to be conducted in the control room have not yet been established. With regard to the Conditioning Facility and the other new SNFP Path Forward facilities, the design has not progressed to the point of defining the need for a control room. Some aspects of the above-discussed human factors guidance may be appropriate to SNFP Path Forward control rooms.</p> <p><u>Actions for Consideration:</u> Consideration should be given to having the AEs for the new SNFP Path Forward facilities review the NRC guidance of NUREG-0700 and SRP 18.1 against DOE Order 6430.1A, Section 1300-12.4 and the DOE draft standard, <i>Human Factors Engineering Design Criteria, Volume I, General Criteria</i>, to identify appropriate additional NRC guidance for design of these facilities. These reviews should give consideration to the differences in complexity between power reactor control rooms and those of the new SNFP Path Forward facilities.</p>

Table 6.b Technical/Safety Comparisons for Other Potentially Applicable NRC Documentation

Number	Subject	Comment
NUREG-0709	<i>Safety Evaluation Report Related to the Renewal of Material License SWM-1265 for the Receipt, Storage, and Transfer of Spent Fuel Pursuant to Part 72</i>	<p>This SER was completed to allow for the renewal of the Part 72 license to operate the General Electric Morris Operation. Some of the technical issues in the SER of importance to SNFPP Path Forward activities are:</p> <ul style="list-style-type: none"> • A ground motion acceleration of 0.20 g was established using the methods of 10CFR100, Appendix A. In addition, Regulatory Guide 1.60 response spectra were used although the SER acknowledges that this guide is for nuclear power plants. See the response provided in Table 5.c for §72.122. • An accident analysis was performed for a tornado missile although the SER acknowledges that (at that time) this was not a requirement of §72.72(b)(2) (the need to address tornadoes was added to 10CFR72 later). The resulting doses were bounded by a fuel basket drop event. • For fire protection the staff was satisfied with a program that: 1) committed to the use of noncombustible and heat resistant materials where possible; and 2) provided for detection, suppression and fire brigade response. The SER does not make reference to a parallel of the safe shutdown analysis required by Appendix R for operating plants. • The acceptance criterion for criticality calculations was a $k_{eff} \leq 0.95$. <p>Conclusions: The above list of items, by itself, does not establish design or construction requirements for new SNFPP Path Forward facilities. However, they are useful in understanding and addressing these issues as they have been identified elsewhere in this comparison.</p> <p><u>Actions for Consideration:</u> None.</p>
NUREG-1092	<i>Environmental Assessment for 10 CFR Part 72 "Licensing for the Independent Storage of Spent Fuel and High-Level Radioactive Waste"</i>	<p>This environmental assessment addresses the need to make changes to 10CFR72 to include consideration of Monitored Retrievable Storage.</p> <p>Conclusions: As stated in Section 1.0 of this document, review of environmental issues is not part of the equivalency review process. As such, no additional analysis is required for this evaluation. It is noteworthy, however, that one of the aspects of 10CFR72 that was identified as requiring resolution was the need to eliminate the restrictions of 10CFR72 placed on fuel cladding integrity that require the fuel cladding be protected against degradation and gross ruptures. The substitution of restrictions on radioactive releases to the public and the environment was suggested.</p> <p><u>Actions for Consideration:</u> None.</p>
NUREG-1140	<i>A Regulatory Analysis on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees</i>	<p>This document discusses a regulatory analysis that investigated if the NRC should impose additional emergency preparedness requirements on certain fuel cycle and other radioactive material licensees for dealing with accidents that might have offsite releases of radioactive material. Also see the discussion provided below for SECY-93-006.</p> <p>For spent nuclear fuel stored in dry casks or in pools, the report concludes that offsite emergency preparedness is not required.</p> <p><u>Actions for Consideration:</u> None.</p>

Table 6.b Technical/Safety Comparisons for Other Potentially Applicable NRC Documentation

Number	Subject	Comment
NUREG-1297 NUREG-1298	Peer Review for High-Level Nuclear Waste Repositories <i>Qualification of Existing Data for High-Level Nuclear Waste Repositories</i>	These two documents apply to the need for DOE to demonstrate in a license application that the applicable health, safety, and environmental regulations in 10CFR60, <i>Disposal of High-Level Radioactive Wastes in Geologic Repositories</i> , have been fulfilled. They do not establish requirements for SNF to be stored in the repository. <u>Conclusions:</u> These two documents are not applicable to the new SNFP Path Forward facilities as Path Forward facilities are not to be licensed under 10CFR60. <u>Actions for Consideration:</u> None.
NUREG-1353	<i>Regulatory Analysis for the Resolution of Generic Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools"</i>	This report presents a regulatory analysis that was conducted to determine whether the use of high density fuel storage racks pose an unacceptable risk to public health and safety. The analysis shows that no new regulatory requirements are warranted in relation to this issue. <u>Conclusions:</u> The issue addressed by the analysis was specific to the use of high density fuel racks and, therefore, is not related to SNFP Path Forward activities. <u>Actions for Consideration:</u> None.
NUREG/CR-0601	<i>Independent Spent Fuel Storage Installations (ISFSIs), Annual Report for FY 1978</i>	The primary objective of the program discussed in the report was the identification of technical aspects of the design, operation, and maintenance of a water pool ISFSI which could contribute to the technical bases for NRC regulations and Regulatory Guides. <u>Conclusions:</u> As this report summarizes work performed in 1978, it could be concluded that any issues identified by the effort of importance to 10CFR72 licensing, did in fact, become incorporated into regulations and Regulatory Guides. However, some aspects of the SNFP Path Forward deviate from what the NRC was considering during rulemaking for 10CFR72. The most significant deviation being the storage of degraded metal fuel. As such, the report was reviewed to identify any SNFP Path Forward unique considerations. No such issues were identified. This is mainly due to the fact that the 1978 effort was directed at pool storage of fuel bundles that were not to be placed in any type of vessel. <u>Actions for Consideration:</u> None.
NUREG/CR-1223	<i>Dry Storage of Spent Nuclear Fuel</i>	This report summarizes a survey of the dry fuel storage facilities in existence in 1980, including facilities at INEL, Hanford, NIS and the Whiteshell Nuclear Research Establishment. The report does not contain conclusions relative to the design or construction requirements for new facilities. <u>Conclusions:</u> The report does not provide information relative to NRC design or construction requirements for present day ISFSIs. <u>Actions for Consideration:</u> None.
NUREG/CR-3405	<i>Long-Term Performance of Materials Used for High-Level Waste Packaging, Annual Report March 1982-April 1983</i>	<u>Conclusions:</u> This reference may contain information important to the material selection for the MCOs. This information has been provided to the individuals involved in the selection of the MCO material and implementing the Technology Acquisition Plan for MCO material selection. <u>Actions for Consideration:</u> None.
NUREG/CR-3427	<i>Long-Term Performance of Materials Used for High-Level Waste Packaging, Volumes 1-4, Quarterly Reports for April 1983 through April 1984</i>	Same comment as above for NUREG/CR-3405.

Table 6.b Technical/Safety Comparisons for Other Potentially Applicable NRC Documentation

Number	Subject	Comment
NUREG/CR-3900	Long-Term Performance of Material Used for High-Level Waste Packaging, Volumes 1-4, Quarterly reports for April 1984 through 1985	Same comment as above for NUREG/CR-3405
NUREG/CR-4379	Long-Term Performance of Materials Used for High-Level Waste Packaging, Volumes 1-4, Quarterly Reports for April 1985 through March 1986	Same comment as above for NUREG/CR-3405
NUREG/CR-4477	Methodologies for Assessing Long-Term Performance of High-Level Radioactive Waste Packages	Same comment as above for NUREG/CR-3405
NUREG/CR-4954	Long-Term Performance of Spent Fuel Waste Forms	Same comment as above for NUREG/CR-3405
NUREG/CR-4955	Long-Term Performance of Container Materials for High-Level Waste	Same comment as above for NUREG/CR-3405
NUREG/CR-4956	System Performance of High-Level Waste Package Components	Same comment as above for NUREG/CR-3405
NUREG/CR-5176	Seismic Failure and Cask Drop Analyses of the Spent Fuel Pools at Two Representative Nuclear Power Plants	<p>This report discusses analyses of the probability and consequences of seismic damage to the spent fuel pool at two plants (H. B. Robinson and Vermont Yankee). The dominant failure mode was gross structural damage followed by failure of the pool liner and loss of water. Other failure modes considered but found to be of less concern were: 1) loss of functioning of the spent fuel pool cooling system; 2) damage to fuel racks leading to criticality; and 3) loss of the pool liner integrity due to a cask drop.</p> <p>Conclusions: The risks of failure of the pool liner by a seismic event or cask drop are not significant to the dry vault storage option recommended for the CSB. The risk of the loss of pool cooling is also not great for the dry vault and natural circulation cooling system proposed for the CSB (Fluor 1995).</p> <p>Criticality for SNFP Path Forward fuel is to be prevented for all credible events, including an MCO drop.</p> <p><u>Actions for Consideration:</u> None.</p>
SECY-87-298, with corrections and addenda	Final Rule for Revision to CFR Part 72 Entitled "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste" and Conforming Amendments to 10 CFR Parts 2, 19, 20, 21, 51, 70, 73, 75 and 150	<p>Conclusions: The focus of this document is the addition of Monitored Retrievable Storage to 10CFR72. The document does address the change in ISFSI design requirements relative to the need to consider tornado missiles unless it can be shown that such missiles will not have any effect on structures, systems, and components important to safety. As the results of the Commission effort have already been incorporated into the current issue of 10CFR72, and as the current issue was used to perform the comparison review for Table 5.c, no additional comparison review is required in regards to SECY-87-298.</p> <p><u>Actions for Consideration:</u> None.</p>

Table 6.b Technical/Safety Comparisons for Other Potentially Applicable NRC Documentation

Number	Subject	Comment
SECY-90-101	<i>Independent Spent Fuel Storage Installation (ISFSI) Training Requirements Evaluation</i>	<p>This document addresses the NRC and the Institute of Nuclear Power Operations (INPO) oversight of training programs for ISFSI operators.</p> <p><u>Conclusions:</u> Training issues are not considered as part of this equivalency evaluation as they do not relate to design and construction of new facilities. The response provided for 10CFR55 in Table 3 provides additional information.</p> <p><u>Actions for Consideration:</u> None.</p>
SECY-91-324	<i>Proposed License Under 10 CFR Part 72 for Dry Storage of Spent Fuel at Public Service Company of Colorado's Fort St. Vrain Nuclear Generating Station Site</i>	<p>This SECY document transmitted to the Commission the SER, proposed license, and environmental assessment for the Fort St. Vrain ISFSI.</p> <p><u>Conclusions:</u> The document identifies no significant nuclear safety issue and contains no information of importance to the SNFP Path Forward activities.</p> <p><u>Actions for Consideration:</u> None.</p>
SECY-93-006	<i>Proposed Amendments to 10 CFR Part 17 to Establish the Emergency Preparedness Licensing Regulations for Independent Spent Fuel Storage Facilities (ISFSI) and Monitored Retrievable Storage Facilities (MRS)</i>	<p>This document discusses a proposed addition of §72.32 to 10CFR72 that would provide emergency preparedness requirements specific to ISFSIs and MRSs. The proposed addition, with modifications, were added to 10CFR72 on June 22, 1995 and became effective September 20, 1995. The addition does not have any impact on the design or constructions of ISFSIs.</p> <p>The document does state that ISFSI emergency plans need not be equivalent to those for power reactors as the risks are not the same. Reference is made to NUREG-1140 that estimates the ISFSI accident dose to the public to be less than 1 rem EDE.</p> <p><u>Conclusions:</u> As the addition of §72.32 does not include any requirements that would impact design or construction of new facilities, no SNFP Path Forward action is required and no additional review of SECY-93-006 is required as part of the equivalency comparison.</p> <p><u>Actions for Consideration:</u> None.</p>

Table 6.b Technical/Safety Comparisons for Other Potentially Applicable NRC Documentation

Number	Subject	Comment
SECY-93-087	Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs	<p>SECY-93-087 summarizes recommended positions for evolutionary and advanced reactors as developed by the NRC staff and presented to the Commission. The following recommended positions contained in SECY-93-087 relate to issues discussed elsewhere in this equivalency evaluation.</p> <p><u>Station Blackout.</u> A spare (full capacity) alternate ac source of diverse design (e.g., gas turbine) that is capable of powering at least one set of normal shutdown loads must be provided.</p> <p><u>Fire Protection.</u> Three hour barriers should be provided for each fire area such that all contained equipment in a fire area can be rendered inoperable by the fire and safe shutdown can be obtained without the need to re-enter the affected area. The control room is excluded from this requirement.</p> <p><u>Operating Basis Earthquake OBE.</u> The OBE should be eliminated from being a design requirement for SSCs.</p> <p><u>Tornado Design Basis.</u> The tornado design basis wind speed for sites west of the Rocky Mountains should be reduced from 240 mph to 200 mph. See the response provided in Table 5.c for \$72.122 for additional information on this recommendation.</p> <p>The SECY document also addresses the potential for a reduction in the radionuclide source term with consideration given to plumeout and aerosol deposition. However, proposed resolution of this issue is left for a separate Commission paper.</p> <p><u>Conclusions.</u> This SECY document provides insight into the NRC plans for some issues that relate to SNFP Path Forward activities.</p> <p><u>Actions for Consideration.</u> None.</p>

Table 6.b Technical/Safety Comparisons for Other Potentially Applicable NRC Documentation

Number	Subject	Comment
SECY-95-104	Safeguards for Spent Nuclear Fuel or High-Level Radioactive Waste - 10 CFR Parts 60, 72, 73, and 75	<p>Currently, 10CFR72 refers to 10CFR73 for requirements for physical protection. However, the current 10CFR73 does not identify any physical protection requirements as specific to the storage of spent nuclear fuel. In practice, affected facilities are being licensed using selected portions of §§73.50 and 73.55 and interim licensing criteria.</p> <p>The SECY document proposes to revise §72.180, <i>Physical security plan</i>, and 72.212, <i>Conditions of general license issued under §72.210</i>, to require the applicant or licensee to either commit to a new §73.51, <i>Requirements for the physical protection of stored spent nuclear fuel or high-level radioactive waste</i>, written specifically for ISFSIs or to the existing security program of §73.55, that would be in place for an operating reactor.</p> <p>New section 73.51 would require the following relative to the design and construction of the ISFSI:</p> <p>General performance objectives:</p> <ul style="list-style-type: none"> • Store spent nuclear fuel within a protected area. • Detect and assess unauthorized penetration of or activities within the protected area. • Provide timely communication to a designated response force whenever the response is called upon to act. • The physical protection system must be designed to protect against radiological sabotage. <p>Physical protection systems:</p> <ul style="list-style-type: none"> • Access to the spent nuclear fuel stored within the protected areas must require passage through or penetration of two physical barriers, one barrier at the perimeter of the protected area and one barrier offering substantial penetration resistance. The physical barrier at the perimeter of the protected area must be as defined in §73.2 <i>Definitions</i>, for an isolation zone. Typically a zone 20 feet wide must be provided on both sides of this barrier to facilitate assessment of unauthorized entry. The barrier offering substantial resistance to penetration may be provided by an approved storage cask or building walls such as those of a reactor or fuel storage building. Other suitable measures to provide the barrier may be acceptable, if approved by the NRC on a case-by-case basis. • A means of illumination must be provided sufficient to permit assessment of unauthorized penetration of or activities within the protected area and associated isolation zones. • The perimeter of the protected area must be subject to continual surveillance and be protected by an intrusion detection system with provisions for a redundant monitoring system. The detection and surveillance systems must be monitored in a continuously-staffed alarm station, not necessarily located within the protected area, and in one remote location. • A controlled personnel identification and lock system must be established and maintained to limit access to the protected area and to individuals authorized unescorted access to the protected area. • Redundant communication capability must be provided between the security organization and designated force or the local law enforcement agency. • All detection and surveillance systems and supporting subsystems must be tamper-indicating with line supervision and be maintained in operable condition. <p>Conclusions: Nuclear safety equivalency to 10CFR73 is addressed in Table 5.a. The DOE orders referenced in the 10CFR73 discussion in Table 5.a are also equivalent to the requirements of the proposed §73.51 as presented in SECY-95-104, which are more general than existing 10CFR73 requirements.</p> <p>Actions for Consideration: None.</p>

Table 6.b Technical/Safety Comparisons for Other
Potentially Applicable NRC Documentation

Number	Subject	Comment
SRP 9.1.2, Revision 3	Spent Fuel Storage	<p><u>Conclusions:</u> Standard Review Plan 9.1.2 establishes the requirement that k_{eff} not be greater than 0.95. The WAC value for providing criticality safety is a k_{eff} of 0.98 as established in WAC-CH-4-29, <i>Nuclear Criticality Safety Manual</i>.</p> <p>Title 10, CFR 72 provides more appropriate design guidance than SRP 9.1.2 for other aspects of the CSB. The SRP is directed at storage of spent fuel in pools.</p> <p><u>Actions for Consideration:</u> Consideration should be given to invoking the NRC k_{eff} criticality safety limit of 0.95.</p>
SRP 9.1.3, Revision 1	Spent Fuel Pool Cooling and Cleanup System	<p><u>Conclusions:</u> With the recommendation that dry storage be utilized for the SNFP Path Forward and the expectation that adequate cooling will be provided by a natural circulation system (Fluor 1955), this SRP is not applicable to the CSB.</p> <p><u>Actions for Consideration:</u> None.</p>
SRP 9.1.5, Revision 0	Overhead Heavy Load Handling Systems	<p><u>Conclusions:</u> SRP 9.1.5 references NUREG-0544, NUREG-0612, ANSI 57.1/ANSI N208, <i>Design Requirements for Light Water Reactor Fuel Handling System</i>, and ANSI 57.2/ANSI N210 (now ANSI/ANS-57.2), <i>Design Objective for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants</i>, for specific criteria to be met to satisfy the requirements for resolving concerns for the handling of heavy loads. The second standard now exists as ANSI/ANS-57.2, <i>Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants</i>.</p> <p>NUREG-0554 and NUREG-0612 are discussed above in this table.</p> <p><u>Actions for Consideration:</u> Consideration should be given to requiring that the AEs for the new SNFP Path Forward facilities incorporate the design requirements of ANSI/ANS-57.1, <i>Design Requirements for Light Water Reactor Fuel Handling System</i>, and ANSI/ANS-57.2, <i>Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants</i>. This would apply only to those facilities that will include the capability to lift a cask or MCC.</p>
SRP 18.1, Revision 0	Control Room	See discussion provided in this table for NUREG-0700.
SRP Branch Technical Positions (BTPs)	NA	<p><u>Conclusions:</u> With the exception of BTP CMEB 9.5-1, <i>Fire Protection for Nuclear Power Plants</i>, discussed in Table 5.b in response to \$50.48, no other BTPs were identified that would relate to SNFP Path Forward activities.</p> <p><u>Actions for Consideration:</u> None.</p>
IEN B4-72	Clarification of Conditions for Waste Shipments Subject to Hydrogen Gas Generation	<p>This notice sets limits on the amount of hydrogen accumulation during shipment of radioactive waste to five percent by volume.</p> <p><u>Conclusions:</u> The contents of this guide are not applicable to new facilities. Table 5.b provides a discussion, in response to \$50.44, of the importance of hydrogen control for other aspects of the SNFP Path Forward.</p> <p><u>Actions for Consideration:</u> None.</p>

Table 6.b Technical/Safety Comparisons for Other Potentially Applicable NRC Documentation

Number	Subject	Comment
ICN 95-28	<i>Emplacement of Support Pads for Spent Fuel Dry Storage Installations at Reactor Sites</i>	<p>This notice identifies the need to ensure that when an NRC-certified cask is to be used by a licensee, they must ensure that the storage pad upon which the cask is to be placed is of adequate design.</p> <p><u>Conclusions:</u> This notice is a reminder to licensees that when a certified cask is to be used, there are some facility-specific considerations that must not be overlooked. That is, the process is not as simple as procuring the cask, loading it with fuel, and setting it down on the site. This should not be a concern for the SNFP Path Forward as all elements of the project will need to be closely integrated.</p> <p><u>Actions for Consideration:</u> None.</p>
ICN 95-29	<i>Oversight of Design and Fabrication Activities for Metal Components used in Spent Fuel Dry Storage Systems</i>	<p>This notice documents NRC observations of licensee shortcomings in oversight of designers and fabricators of metal components for dry storage systems during their inspections to 10CFR72, Subpart G. Inspection findings were primarily in the areas of document control, quality assurance records, and control of purchased material, equipment, and services.</p> <p>Inadequate ISFSI vendor and subcontractor quality assurance was also mentioned by the NRC at an August 22, 1995 meeting with representatives from the Nuclear Energy Institute.</p> <p><u>Actions for Consideration:</u> Consideration should be given to ensuring the areas of vendor and subcontractor quality assurance records and control of Safety Class 1 purchased material, equipment, and services receive emphasis during SNFP Path Forward audits, surveillances, and assessments.</p>
ICN 95-38	<i>Degradation of Boraflex Neutron Absorber in Spent Fuel Storage Racks</i>	<p>This notice discusses the potential for a gradual release of silica and boron carbide from Boraflex neutron absorbers following gamma irradiation and long-term exposure to the spent fuel pool environment. When Boraflex is subjected to gamma radiation in the aqueous environment of the pool, the silicon polymer matrix becomes degraded and silica filler and boron carbide are released.</p> <p><u>Conclusions:</u> This issue is not a concern for the SNFP Path Forward as the fuel is to be stored in a dry environment.</p> <p><u>Actions for Consideration:</u> None.</p>

Table 6.b Technical/Safety Comparisons for Other
Potentially Applicable NRC Documentation

Number	Subject	Comment
IE Notices and Bulletins	There are a number of Notices and Bulletins on use of fraudulent materials that warrant being addressed in the context of NRC nuclear safety equivalency	<p>The following IE Notices and Bulletins deal with counterfeit and fraudulent equipment and materials.</p> <p>IEB 85-19 <i>Alleged Falsification of Certification and Alteration of Markings on Piping, Valves and Fittings</i></p> <p>IEB 89-70 <i>Possible Indications of Misrepresented Vendor Products</i></p> <p>IEB 92-56 <i>Counterfeit Valves in the Commercial Grade Supply System</i></p> <p>IEB 87-02 <i>Fastener Testing to Determine Conformance with Applicable Material Specifications</i></p> <p>IEB 88-05 <i>Nonconforming Materials Supplied by Piping Supplies, Inc. at Folsom, New Jersey and West Jersey Manufacturing Company at Williamstown, NJ</i></p> <p>IEB 88-10 <i>Non-Conforming Molded Case Circuit Breakers</i></p> <p>Conclusions: WMC has a provision that is contained within Purchase Orders relating to counterfeit components. This provision (SO-46, <i>Counterfeit Fasteners and Components</i>), states that "... Seller warrants that all products, components, parts and assemblies furnished under this order are genuine and match the quality, test reports, markings and/or fitness for use represented or implied by the materials themselves or their use by Seller as component parts. WMC reserves the right to question Seller and require Seller to certify and/or furnish proof regarding the quality, authenticity, application or fitness for use of the materials supplied by Seller under this order. Any materials furnished as part of this order and which have been previously found by WMC, the Department of Energy, or the Department of Commerce to be counterfeit or which are listed by the Department of Commerce as suspect will be deemed, without more proof, to be subject to the above requirement of further proof or certification. WMC also reserves the right to question the circumstances and make available a report of any such review to the Government. Further references about suspected counterfeit products should be obtained directly from the Department of Commerce or US Customs Service."</p> <p>With this contract language in place, it is concluded that WMC has adequate controls in place to deal with the potential delivery of fraudulent equipment and materials.</p> <p><u>Actions for Consideration:</u> None.</p>
IE Notices and Bulletins	In addition to the number of IENs and IEBs dealing with fraudulent materials discussed above, there is a much larger set dealing with defective components manufactured by specific vendors. These include Rosemount and SOR pressure and differential sensors and switches; Anchor-Darling and Borg-Warner check valves; Limitorque MOV actuators and limit switches; GE, W. and ITE/ABB/Gould circuit breakers; and relays manufactured by several vendors.	<p><u>Actions for Consideration:</u> Consideration should be given to instituting a process to identify Safety Class I equipment that has been identified in the commercial nuclear power industry, via IE Bulletins and Notices, as being potentially defective.</p>

Table 6.b Technical/Safety Comparisons for Other Potentially Applicable NRC Documentation

Number	Subject	Comment
Generic Letters	No Generic Letters were identified that apply directly to [SFS]. Generic Letter 95-01, <i>NRC Staff Technical Position on Fire Protection for Fuel Cycle Facilities</i> , contains guidance that might be appropriate for the Conditioning Facility. It is addressed below. In addition, three Generic Letters, GL 88-14, 89-10, and 89-13, were of particular importance to the power reactor industry.	Actions for Consideration: The applicable portions of Generic Letters 88-14, 89-10, and 89-13 should be considered by the AEs for the new SNF Path Forward facilities for application to Safety Class 1 systems and components.
GL 88-14	<i>Instrument Air Supply System Problems Affecting Safety-Related Equipment</i>	Other documentation that provides information on problems with instrument air systems includes INPO SOER 88-01, IEB 86-03, and IENs 86-50, 87-28 and 88-24.
GL 89-10	<i>Safety-Related Motor-Operated Valve Testing and Surveillance 10CFR50.54(f), w/supplements</i>	Additional information on problems with motor-operated valves is provided in INPO SOER 83-09.
GL 89-13	<i>Service Water System Problems Affecting Safety-Related Equipment</i>	Other documentation that provides information on problems with cooling water systems includes AECOD/CB01, AECG/593-03, NUREG 1275, and NUREG/CB-2797.
GL 95-01	<i>NRC Staff Technical Position on Fire Protection for Fuel Cycle Facilities</i>	The Generic Letter was to make licensees and applicants for uranium conversion and fuel fabrication facilities aware of a Technical Position published in the Federal Register (57FR35607) of fire protection and to request a response stating how the Technical Position would be implemented by the addressee. Conclusions: The Technical Position is discussed in Table 5 a in response to 10CFR70. Actions for Consideration: None.
IH 60847	<i>Inspection of Dry Storage of Spent Nuclear Fuel</i>	The objective of this NRC Inspection Manual procedure is to determine whether components and operation related to the dry storage of spent nuclear fuel in concrete modules at nuclear power reactor sites are in accordance with the license. Conclusions: The inspection manual contains the following two statements that imply design requirements: 1) dry shielded canisters are to have markings that allow for identification with quality assurance certification records, and 2) the inspector is to verify that equipment and training of operators complies with NUREG-0612. The need for each MCO to have an identification number is required to satisfy DOE Order 5633.38, <i>Control and Accountability of Nuclear Materials</i> . NUREG-0612 is addressed separately in this table. Actions for Consideration: None.

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8.0 GLOSSARY

8.1 Acronyms

ac	alternating current
ACI	American Concrete Institute
AE	architect engineer
AEC	Atomic Energy Commission
AEOD	Analysis and Evaluation of Operational Data (NRC NRR Office for)
ALARA	As Low As Reasonably Achievable
ALWR	advanced light-water reactor
APCSB	Applied Power and Containment Systems Branch (of the NRC NRR)
ATWS	anticipated transients without scram
BTP	branch technical position
CFR	Code of Federal Regulations
Ci	curie
CMEB	Chemical/Mechanical Engineering Branch (of the NRC NRR)
CSB	canister storage building
DBA	design basis accident
DBE	design basis earthquake
DCRDR	detailed control room design review
DNFSB	Defense Nuclear Facilities Safety Board
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
EDE	effective dose equivalent
EIS	Environmental Impact Statement
EPZ	emergency planning zone
FEMA	Federal Emergency Management Agency
FR	Federal Register
FRD	Functions and Requirements Document
GDC	general design criteria
GE	General Electric
GL	generic letter
GWD	gigawatt-days
HEPA	high efficiency particulate air (filter)
HEU	highly enriched uranium
HLW	high level waste
HSRCM	Hanford Site Radiological Control Manual
HVAC	heating, ventilating, and air conditioning
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IE	Inspection and Enforcement (NRC Office for)
IEB	Inspection and Enforcement Bulletin
IEN	Inspection and Enforcement Notice
IM	inspection manual
INEL	Idaho National Engineering Laboratory
INPO	Institute of Nuclear Power Operations
IP	implementing procedure
ISFSI	independent spent fuel storage installation
LER	licensee event report
LWR	light water reactor
MC&A	material control and accountability
MCO	multi-canister overpack
MHM	MCO handling machine
MOV	motor operated valve

MPC	multi-purpose canister (for an MRS and geologic repository)
MRP	Management Requirements and Procedures
MRS	monitored retrievable storage
MTU	metric tons uranium
MWD	megawatt-days
NEPA	National Environmental Policy Act
NFPA	National Fire Protection Association
NRC	U.S. Nuclear Regulatory Commission
NPH	natural phenomena hazard
NRR	Nuclear Reactor Regulation (NRC Office of)
NPRDS	nuclear plant reliability data system
NTS	Nevada Test Site
OMB	Office of Management and Budget
ORPS	Occurrence Reporting and Processing System
OSHA	Occupational Safety and Health Administration
PSE	preliminary safety evaluation
QA	quality assurance
QAP	quality assurance program
QAPP	quality assurance program plan
QC	quality control
QI	quality instruction
QR	quality requirements
RCRA	Resource Conservation and Recovery Act
rem	roentgen-equivalent man
RL	Department of Energy Richland Operations Office
RLID	RL implementing document
RLIP	RL implementing procedure
SAR	safety analysis report
SARP	safety analysis report for packaging
SBO	station blackout
SDC	standard design criteria
SECY	(see definitions)
SEL	safety equipment list
SNF	spent nuclear fuel
SNFP	spent nuclear fuel project
SNM	special nuclear material
SOER	significant operating experience report
SRP	standard review plan
SSC	structures, systems, and components
SSE	safe shutdown earthquake
STD	standard (also Std)
TMI	Three Mile Island (reactor)
TS	technical specifications
TSR	technical safety requirements
UCRL	University of California Research Laboratory
WBD	whole body dose
WHC	Westinghouse Hanford Company

8.2 Definitions

As Low As Reasonably Achievable (10CFR20). ALARA means making every reasonable effort to maintain exposures to radiation as far below the dose limits in 10CFR20 as is practical consistent with the purpose for which the licensed activity is undertaken, taking into account the state of technology, the economics of improvements in relation to state of technology, the economics of improvements in relation to benefits to the public health and safety, and other societal and socioeconomic considerations, and in relation to utilization of nuclear energy and licensed materials in the public interest.

As Low As Reasonably Achievable (10CFR835). ALARA is the approach to radiation protection to manage and control exposures (both individual and collective) to the work force and to the general public to as low as is reasonably achievable, taking into account social, technical, economic, practical, and public policy considerations. As used in 10CFR835, it is not a dose limit but a process of attaining doses as far below the applicable limit of 10CFR835 as is reasonably achievable.

Branch Technical Positions (NUREG-0800). Documents included in the Standard Review Plans. These documents typically set forth the solutions and approaches determined to be acceptable in the past by the NRC staff in dealing with a specific safety problem or safety-related design area. These solutions and approaches are codified in this form so that staff reviewers can take uniform and well-understood positions as the same safety problems arise in future cases.

Byproduct material (10CFR20 and 50). Any radioactive material (except special nuclear material) yielded in, or made radioactive by, exposure to the radiation incident to the process of producing or utilizing special nuclear material.

Canister. Metal (aluminum or stainless steel) container used to store reactor fuel rods/elements. Canisters containing fuel are configured in a double-barrel shape.

Cask. Container which provides shielding, containment, and subcriticality for SNF during transfer/transport.

Class 103 license (10CFR50). A license issued under §50.22 for commercial and industrial facilities.

Class 104 license (10CFR50). A license issued under §50.21 for medical therapy and research and development facilities.

Commission (10CFR2). The NRC Commission of five members or a quorum thereof sitting as a body, as provided by section 201 of the Energy Reorganization Act of 1974 (88 Stat. 1242), or any officer to whom has been delegated authority pursuant to section 161n of the Atomic Energy Act.

Conditioning. Processing of SNF by physical, chemical or mechanical means to prepare the SNF to meet predetermined requirements for safe and compliant storage or disposal. Does not include packaging.

Controlled access area (10CFR73). Any temporarily or permanently established area which is clearly demarcated, access to which is controlled and which affords isolation of the material or persons within it.

Controlled area (10CFR20). An area, outside of a restricted area but inside the site boundary, access to which can be limited by the licensee for any reason.

Controlled area (10CFR72). That area immediately surrounding an ISFSI or MRS for which the licensee exercises authority over its use and within which ISFSI or MRS operations are performed.

Controlled area (10CFR835). Any area to which access is managed in order to protect individuals from exposure to radiation and/or radioactive material. Individuals who enter only the controlled area without entering radiological areas are not expected to receive a total effective dose equivalent of more than 100 mrem (0.001 sievert) in a year.

Design basis accidents (DOE Order 6430.1A). Postulated accidents, or natural forces, and resulting conditions for which the confinement structure, systems, components and equipment must meet their functional goals. These safety class items are those necessary to assure the capability; to safely shut down operations, maintain the plant in a safe shutdown condition, and maintain integrity of the final confinement barrier of radioactive or other hazardous materials; to prevent or mitigate the consequences of accidents; or to monitor releases that could result in potential offsite exposures.

Design basis earthquake (DOE Order 6430.1A). That earthquake that is the most severe design accident of this type and that produces the vibratory ground motion for which safety class items are designed to remain functional.

Design basis earthquake (10CFR100). By footnote one in 10CFR100, Appendix A, the design basis earthquake is the same as the safe shutdown earthquake.

Design basis fire (DOE Order 6430.1A). That fire is the most severe design basis accident of this type. In postulating such a fire, failure of automatic and manual fire suppression provisions shall be assumed except for those systems considered critical items.

Effective dose equivalent (10CFR70). The sum of the products of the dose equivalent to the body organ or tissue and the weighting factors applicable to each of the body organs or tissues that are irradiated. Weighting factors are: 0.25 for gonads, 0.15 for breast, 0.12 for red bone marrow, 0.12 for lungs, 0.03 for thyroid, 0.03 for bone surface, and 0.06 for each of the other five organs.

Effective dose equivalent (10CFR835). The summation of the products of the dose equivalent received by specified tissues of the body (H_T) and the appropriate weighting factor (w_T)—that is, $H_E = \sum w_T H_T$. It includes the dose from radiation sources internal and/or external to the body. The effective dose equivalent is expressed in units of rem (or sievert).

Exclusion area (10CFR100). That area surrounding the reactor, in which the reactor licensee has the authority to determine all activities including exclusion or removal of personnel and property from the area.

Exclusion area (DOE Order 6430.1A). A security area for the protection of classified matter where mere access to the area would result in access to classified matter.

Generic Letters. Letters prepared by the NRC primarily to inform applicants and licensees of regulatory requirements related to licensing matters and schedules for compliance. Generic letters are also used to clarify NRC policy and to request and transmit information.

High radiation area (10CFR20). An area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 0.1 rem (1 mSv) in 1 hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates.

High radiation area (10CFR835). Any area, accessible to individuals, in which radiation levels could result in an individual receiving a deep dose equivalent in excess of 0.1 rem (0.001 sievert) in 1 hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates.

Important to safety (10CFR50, Appendix A). Structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

Important to safety (10CFR72). Those features of the ISFSI or MRS whose function is:

- (1) To maintain the conditions required to store spent fuel or high-level radioactive waste safely.
- (2) To prevent damage to the spent fuel or the high-level radioactive waste container during handling and storage, or
- (3) To provide reasonable assurance that spent fuel or high-level radioactive waste can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public.

Important to safety (WHC-CM-4-46, Section 9). Structures, systems, and components identified in the safety equipment list plus those necessary to perform or maintain a required safety function as identified in the safety analysis and other authorization basis documentation.

Independent spent fuel storage installation (10CFR72). A complex designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage.

Interim Storage. Hanford storage of SNF for an extended period, until final disposition of the SNF.

Isolation Zone (10CFR73). An area adjacent to a physical barrier, clear of all objects which could conceal or shield an individual.

Licensed material (10CFR20). Source material, special nuclear material, or by-product material received, possessed, used, transferred or disposed of under a general or specific license issued by the NRC.

Licensee (10CFR2). A person, including an individual, corporation, firm or association, who is authorized to conduct activities under a license, including a renewed license, or construction permit issued by the Commission.

Multi-Canister Overpack. Item that houses SNF in a configuration which is to be determined. Designed specifically for K Basin SNF only.

Monitored Retrievable Storage Installation (10CFR72). A complex designed, constructed, and operated by DOE for the receipt, transfer, handling, packaging, possession, safeguarding, and storage of spent nuclear fuel aged for at least one year and solidified high-level radioactive waste resulting from civilian nuclear activities, pending shipment to a HLW repository or other disposal.

NRC Bulletins. Formerly known as I&E (Inspection and Enforcement) bulletins, these documents are used by the NRC to transmit information and request action and/or a written response from licensees and permit holders regarding matters of health and safety, safeguards or environmental significance.

NRC Information Notices. Documents issued by the NRC to licensees or permit holders to give preliminary information on an event or condition that the NRC believes may be relevant to health and safety, safeguards, or protection of the environment. Notices may also be issued to inform licensees and permit holders promptly of changes in NRC procedures and the implementation of new rules and regulations.

NRC Inspection Manual. A manual that provides policy, guidance and procedures for the NRC inspection program to ensure that planned and existing licensed operations can be and are being conducted without undue risk to the public.

Operating Basis Earthquake (DOE Order 6430.1A). That earthquake that produces the vibratory ground motion for which the plant structures, systems, and components are designed to either remain operable or be readily restorable to operating condition. The OBE is equivalent to at least one-half the DBE in terms of ground acceleration.

Path Forward. Integrated activities to achieve a defined strategy for final disposition of K Basin SNF.

Production facility (10CFR20 and 50). (1) Any nuclear reactor designed or used primarily for the formation of plutonium or uranium-233; or (2) Any facility designed or used for the separation of the isotopes of plutonium, except laboratory scale facilities designed or used for experimental or analytical purposes only; or (3) Any facility designed or used for the processing of irradiated materials containing special nuclear material (see 10CFR50.2 for specific exceptions).

Protected area (10CFR73). An area encompassed by physical barriers and to which access is controlled.

Protected area (DOE Order 6430.1A). An area encompassed by physical barriers (e.g., walls or fences), subject to access controls, surrounding a material access area or containing Category II special nuclear material.

Radiation area (10CFR20). An area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in

excess of 0.005 rem (0.05 mSv) in 1 hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates.

Regulatory Guides. Documents that describe methods acceptable to the NRC staff for implementing specific portions of NRC regulations. Some regulatory guides lay out steps taken by the staff in evaluating specific situations. Others provide guidance to applicants concerning information needed by staff in its review of applications for permits and licenses or refer to or endorse national standards.

Restricted area (10CFR20). An area, access to which is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. The restricted area does not include areas used as residential quarters, but separate rooms in a residential building may be set apart as a restricted area.

Safe shutdown earthquake (10CFR100). The earthquake which is based upon an evaluation of the maximum earthquake potential considering the regional and local geology and seismology and specific characteristics of local subsurface material.

Safety class item (DOE Order 6430.1A). Systems, components and structures, including portions of process systems, whose failure could adversely affect the environment or safety and health of the public.

SECY Papers. Documents issued by the Secretary of the Commission. SECY papers respond to questions raised by the Commission, the Chairman and/or the Commissioners; propose rulemaking or respond to petitions for rulemaking; and bring matters to the attention of the Commission for information or action.

Site boundary (10CFR20). That line beyond which the land or property is not owned, leased, or otherwise controlled by the licensee.

Site boundary (DOE Order 6430.1A). A well-marked boundary of the property over which the owner or operator can exercise strict control without the aid of outside authorities.

Sludge. Spent nuclear fuel pieces and degradation products in the K Basins, plus all other solid debris including concrete grit. Sludge material is finer than 1/4 in. in size.

Source material (10CFR40). Source material is: 1) uranium or thorium, or any combination thereof, in any physical or chemical form, or 2) ores which contain by weight one-twentieth of one percent (0.05%) or more of: (i) uranium, (ii) thorium, or (iii) any combination thereof. Source material does not include special nuclear material.

Source material (10CFR50). Source material as defined in subsection 11z. of the Atomic Energy Act of 1954, as amended, and in the regulations contained in 10CFR40.

Special nuclear material (10CFR20 and 50). (1) Plutonium, uranium-233, uranium enriched in the isotope 233 or in the isotope 235, and any other material that the Commission, pursuant to the provisions of section 51 of the Atomic Energy Act of 1954, as amended, determines to be special nuclear material, but does not include source material; or (2) Any material

artificially enriched by any of the foregoing but does not include source material.

Special nuclear material (DOE Order 6430.1A). Plutonium, uranium-233, uranium enriched in uranium-233 or in the uranium-235, or any material artificially enriched in any of the foregoing (but does not include source material) and any other material that, pursuant to the provisions of Section 51 of the Atomic Energy Act of 1954, as amended, has been determined to be special nuclear material.

Spent Nuclear Fuel. Nuclear fuel that has been withdrawn from a nuclear reactor following irradiation and the constituent elements that have not been separated by reprocessing.

Staging. Holding MCOs prior to conditioning of K Basin SNF.

Standard Review Plan (NUREG-0800). A document prepared for the guidance of NRC staff reviewers in performing safety reviews of applications to construct or operate nuclear power plants. The principal purpose of the SRP is to assure the quality and uniformity of staff reviews and to present a well-defined base from which to evaluate proposed changes in the scope and requirements of the reviews. It is also a purpose of the SRP to make information about regulatory matters widely available and to improve communication and understanding of the staff review process by interested members of the public and the nuclear power industry.

Transfer. Movement of materials within the confines of a facility on the Hanford site.

Transport. Movement of materials between facilities on the Hanford site.

Unrestricted area (10CFR20). An area, access to which is neither limited nor controlled by the licensee.

Very high radiation area (10CFR20, 10CFR835). An area, accessible to individuals, in which radiation levels could result in an individual receiving an absorbed dose in excess of 500 rads (5 grays) in 1 hour at 1 meter from a radiation source or from any surface that the radiation penetrates.

9.0 REFERENCES

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ATTACHMENT A
DETAILED EVALUATIONS

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QUALITY ASSURANCE CRITERIA

Background

Title 10, CFR50, Appendix B, and 10CFR72, Subpart G, establish quality assurance requirements for the design, construction, and operation of structures, systems, and components that prevent or mitigate accidents that could cause undue risk to the health and safety of the public. As mentioned in Table 5.b, under 10CFR50, Appendix B, a review of 10CFR50, Appendix B, and 10CFR72, Subpart G, *Quality Assurance*, found these two regulations equivalent, and identical in wording in many cases. The nuclear safety equivalency comparison was therefore only performed against 10CFR72, Subpart G, due to its direct application to spent nuclear fuel storage installations.

Path Forward Discussion

The current DOE regulatory policy for the SNFP is that design and construction of new SNFP Path Forward facilities will achieve a level of nuclear safety comparable to an NRC-licensed commercial nuclear facility (DOE 1995a). DOE requirements will comprise the baseline for new SNFP Path Forward facilities. However, DOE requirements will be supplemented by specific NRC requirements, where necessary, to demonstrate nuclear safety equivalency to comparable NRC-licensed facilities.

Section 830.120, *Quality Assurance Requirements*, provides DOE requirements for ensuring that structures, systems, and components in nuclear facilities will perform satisfactorily in service. Section 830.120 is intended to supersede the existing requirements in DOE Order 5700.6C, *Quality Assurance*, for nuclear facilities. The DOE has also stated that §830.120 addresses Title 10, Appendix B requirements.

NRC's quality assurance regulations for spent nuclear fuel storage installations are contained in 10CFR72, Subpart G. Subpart G interprets the requirements of 10CFR50, Appendix B, and establishes requirements, procedures, and criteria for the issuance of licenses to receive, transfer, and possess power reactor spent fuel and other radioactive materials associated with spent fuel storage in an independent spent fuel storage installation.

Subpart G of 10CFR72 provides quality assurance requirements applying to design, purchase, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair, modification of structures, systems, and components, and decommissioning that are important to safety. Quality assurance programs developed to meet these requirements must be approved by the NRC. **Note:** It is stated in Subpart G that the quality assurance program requirements apply to a licensee and also to any design, construction, fabrication, assembly, and testing activities that are accomplished prior to issuance of a license.

A matrix, Table A-1, was developed to assist in determining equivalency of quality assurance requirements between §830.120 and the requirements of 10CFR72, Subpart G.

Table A-1 Comparison of the Quality Assurance Requirements of 10CFR72 and 10CFR830.120

Quality Affecting Activity. 10CFR72	Equivalency Comparison to 10CFR830.120
Section 72.140. <i>Quality assurance requirements</i>	<p>Section 72.140 requires approval of the Quality Assurance Plan (QAP) submitted by the licensee prior to receipt of spent fuel at the ISFSI. It states in §72.24(n) that a description of the QAP must be included in the SAR with provisions of Subpart G to be applied to design, fabrication, construction, testing, operation, modification, and decommissioning of the structures, systems and components (SSCs) of the ISFSI important to safety.</p> <p>For new SNFP Path Forward facilities, §830.120 is not specific on when the QAP is required to be submitted for approval. However, DOE Order 4700.1, <i>Project Management System</i>, in Part D, <i>Quality Assurance</i>, of Chapter III, requires that Quality Assurance begins at project conception and runs through design, development, construction, fabrication, and operation. This would require approval of the QAP prior to initiation of these activities.</p> <p>Therefore, equivalency to NRC requirements is met through a combination of §830.120 and DOE Order 4700.1.</p>
Section 72.142. <i>Quality assurance organization</i>	<p>Section 72.142 requirements are contained in Criteria 1 and 10 of §830.120 except for the following:</p> <p>~ Section §830.120 does not specifically state that persons and organizations performing QA functions must have independence from cost and schedule considerations as stated in §72.142(b). However, it does state that the organization shall have sufficient authority and freedom from the line to carry out its responsibilities. Quality Requirement (QR) 1.0, Revision 4, <i>Organization</i>, of WHC-CM-4-2, <i>Quality Assurance Manual</i>, states in Section 3.1 that the organization will have sufficient independence from cost and schedule considerations.</p> <p>Equivalency to NRC requirements is met through a combination of §830.120 and QR 1.0, Revision 4, of WHC-CM-4-2.</p>
Section 72.144. <i>Quality assurance program</i>	<p>Section 72.144 requirements are contained in Criteria 1, 2, 3, 5, 8, 9 and 10 of §830.120 except for the following:</p> <p>~ Section 72.144 requires that the licensee identify the structures, systems and components to be covered by the QAP. Criteria for this determination include: 1) impact of malfunction or failure of an item on safety; 2) design and fabrication complexity or uniqueness of the item; 3) need for special controls and surveillance over processes and equipment; 4) degree to which functional compliance can be demonstrated by inspection or test; and 5) quality history and degree of standardization of the item.</p> <p>The above criteria are not specifically stated in §830.120. However, WHC-CM-4-46, <i>Safety Analysis Manual</i>, Section 9.0, Revision 1, <i>Safety Classification of Structures, Systems, and Components</i>, provides equivalent criteria and methodology for the identification and safety classification of SSCs. This provides a graded approach applicable for design and quality assurance requirements.</p> <p>~ Section 72.144 emphasizes that control over activities affecting the quality of the identified SSCs must be provided through the QA Program. This is not specifically stated in §830.120. As stated in the above procedures, SSCs are classified commensurate with the function of each system, structure, and component in preventing or mitigating the consequences of hazards and postulated design basis accidents. Design and quality assurance requirements of SSCs are controlled commensurate with this classification by the QAP.</p> <p>~ Section 72.144 requires that activities affecting quality are accomplished under suitably controlled conditions. Controlled conditions include the use of appropriate equipment, suitable environmental conditions for accomplishing the activity, such as adequate cleanliness, and assurance that all prerequisites for the given activity have been satisfied. Quality Requirement 9.0, Revision 4, <i>Control of Processes</i>, in WHC-CM-4-2, <i>Quality Assurance Manual</i>, states that processes affecting the quality of items or services shall be controlled by instructions, procedures, drawings, checklists, travelers, or other appropriate means. These means shall ensure that process parameters are controlled within defined limits, and that specified environmental conditions are maintained.</p> <p>Based on the above, equivalency to NRC requirements is met by §830.120 in conjunction with WHC-CM-4-46, Section 9.0, Revision 1, and QR 9.0, Revision 4, of WHC-CM-4-2.</p>

Table A-1 Comparison of the Quality Assurance Requirements of 10CFR72 and 10CFR830.120

Quality Affecting Activity. 10CFR72	Equivalency Comparison to 10CFR830.120
Section 72.146 <i>Design control</i>	<p>Section 72.146 requirements are met by Criterion 6 of §830.120 except for the following</p> <p>~ Section 72.146 requires that design control measures shall be applied to items such as: criticality physics, radiation, shielding, stress, thermal, hydraulic and accident analyses, compatibility of materials, accessibility for in-service inspection, maintenance and repair, features to facilitate decontamination, and delineation of acceptance criteria for inspections and tests.</p> <p>WHC-CM-6-1, (manual revision dated August 8, 1995) <i>Standard Engineering Practices</i>, and WHC-IP-1026, (manual revision dated July 10, 1995) <i>Engineering Practice Guideline</i>, contain information related to items and services that will have design control measures applied consistent with the graded approach previously discussed above from WHC-CM-4-2, and WHC-CM-4-46.</p> <p>• Section 72.146 requires that design documents must contain quality standards and any deviations are controlled. Section 830.120 does not specifically state that design documents must contain quality standards and deviations must be controlled. Quality Requirement 3.0, Revision 5, <i>Design Control</i>, of WHC-CM-4-2, <i>Quality Assurance Manual</i>, states in Section 3.0 that design inputs, including standards, shall be identified and documented and that changes in design inputs shall be identified, documented, approved and controlled.</p> <p>Based on the above, equivalency to NRC requirements is met by §830.120 in conjunction with WHC-CM-6-1, 8/8/95, WHC-IP-1026, 7/10/95, and QR 3.0, Revision 5, of WHC-CM-4-2.</p>
Section 72.148 <i>Procurement document control</i>	<p>Section 72.148 requirements are met by Criterion 7 of §830.120 except for the following</p> <p>~ Section 72.148 requires contractors or subcontractors to provide a quality assurance plan consistent with the provisions of §72.148. Section 830.120 does not specifically state that contractors and subcontractors are required to develop such quality assurance plans.</p> <p>Quality Requirement 4.0, Revision 4, <i>Procurement Document Control</i>, of WHC-CM-4-2, <i>Quality Assurance Manual</i>, states that procurement documents shall require that the supplier have a documented quality assurance program that implements applicable, or all of the requirements, of the QAP. It also requires the supplier to incorporate quality assurance program requirements into subtier procurement documents.</p> <p>Based on the above, equivalency to NRC requirements is met by §830.120 in conjunction with QR 4.0, Revision 4, of WHC-CM-4-2.</p>
Section 72.150 <i>Instructions, procedures, and drawings</i>	Section 72.150 requirements are met by Criteria 1, 5, and 8 of §830.120
Section 72.152 <i>Document control</i>	Section 72.152 requirements are met by Criteria 4, 5, 6, and 8 of §830.120
Section 72.154 <i>Control of purchased material, equipment, and services</i>	<p>Section 72.154 requirements are met by Criteria 7 and 8 of §830.120 except for the following</p> <p>~ Section 72.154 requires documentary evidence that material and equipment conform to the procurement specifications prior to installation or use of the material and equipment. Section 830.120 does not specifically require conformance to procurement specifications prior to installation or use of material and equipment.</p> <p>Quality Requirement 7.0, Revision 4, <i>Control of Purchased Items or Services</i>, of WHC-CM-4-2, <i>Quality Assurance Manual</i>, states that acceptance of an item or related service from a supplier will be performed by a certificate of conformance, source verification, receiving inspection, or post-installation test, or a combination of these methods. In each case, acceptance will be documented.</p> <p>~ Section 72.154 requires that the licensee assess the effectiveness of the control of quality by contractors and subcontractors. Section 830.120 does not clearly require contractors and supplier quality assurance programs to be assessed in this regard.</p> <p>Quality Instruction (OI) 7.2, Revision 5, <i>Supplier Evaluation</i>, and QR 7.0, Revision 4, <i>Control of Purchased Items and Services</i> (for commercial grade items) of WHC-CM-4-2, requires offsite and onsite evaluation of suppliers' quality assurance programs in accordance with provisions of the instruction and procedure.</p> <p>Based on the above, equivalency to NRC requirements is met by §830.120 in conjunction with QR 7.0, Revision 4, and OI 7.2, Revision 5, of WHC-CM-4-2.</p>

Table A-1 Comparison of the Quality Assurance Requirements of 10CFR72 and 10CFR830.120

Quality Affecting Activity, 10CFR72	Equivalency Comparison to 10CFR830.120
Section 72.156: <i>Identification and control of materials, parts, and components</i>	Section 72.156 requirements are met by Criteria 3, 5, and 7 of §830.120.
Section 72.158: <i>Control of special processes</i>	Section 72.158 requirements are met by Criteria 2, 5 and 8 of §830.120.
Section 72.160: <i>Licensee inspection</i>	<p>Section 72.160 requirements are met by Criteria 5, 8 and 10 of §830.120 except for the following</p> <ul style="list-style-type: none"> Section 72.160 requires direct inspection of processed materials, but if this cannot be performed, indirect inspection by monitoring processing materials, equipment and personnel is performed. Section 830.120 does not require alternative or indirect approaches to inspection when direct inspection is not possible. <p>Quality Requirement 10.0, Revision 3, <i>Inspection and Surveillance</i>, of WHC-CM-4-2, <i>Quality Assurance Manual</i>, requires indirect control by surveillance of processing methods, personnel, and equipment when direct inspection cannot be performed</p> <ul style="list-style-type: none"> Section 72.160 requires both inspection and process monitoring when quality control is inadequate without both. The concurrent or complimentary use of inspection and process monitoring to ensure quality is not addressed by §830.120. Quality Requirement 10.0, Revision 3, <i>Inspection and Surveillance</i>, requires both methods be performed when quality control is inadequate without both. <p>Based on the above, equivalency to NRC requirements is met by §830.120 in conjunction with QR 10.0, Revision 3, of WHC-CM-4-2</p>
Section 72.162: <i>Test control</i>	Section 72.162 requirements are met by Criteria 6 and 8 of §830.120.
Section 72.164: <i>Control of measuring and test equipment</i>	Section 72.164 requirements are met by Criterion 8 of §830.120.
Section 72.166: <i>Handling, storage, and shipping control</i>	Section 72.166 requirements are met by Criterion 5 of §830.120.
Section 72.168: <i>Inspection, test, and operating status</i>	<p>Section 72.168 requirements are met by Criterion 8 of §830.120 except for the following</p> <ul style="list-style-type: none"> Section 72.168 requires marking items to indicate the status of inspection and tests performed upon individual items of the ISFSI. Section 72.168 also requires identification of items which have satisfactorily passed required inspections and tests. Section 830.120 allows, but does not require, marking items to indicate the status of inspections and tests. Section 830.120 also allows, but does not require, identifying items that have satisfactorily passed inspections and tests. <p>Quality Requirement 14.0, Revision 5, <i>Inspection, Test and Operating Status</i>, of WHC-CM-4-2, <i>Quality Assurance Manual</i>, provides requirements for status indicators which identify the inspections, test and operating status of items. It also requires application of a status indicator to the items upon completion of inspection and testing</p> <p>Based on the above, equivalency to NRC requirements is met by §830.120 in conjunction with QR 14.0, Revision 5, of WHC-CM-4-2.</p>
Section 72.170: <i>Nonconforming materials, parts, or components</i>	Section 72.170 requirements are met by Criteria 3, 7 and 8 of §830.120.
Section 72.172: <i>Corrective action</i>	Section 72.172 requirements are met by Criteria 3, 7, 8, 9 and 10 of §830.120.
Section 72.174: <i>Quality assurance records</i>	Section 72.174 requirements are met by Criteria 4 and 6 of §830.120.
Section 72.176: <i>Audits</i>	Section 72.176 requirements are met by Criterion 10 §830.120.

Actions for Consideration:

Section 830.120 contains the majority of the NRC quality assurance requirements that are in 10CFR72, Subpart G. The differences in the sets of requirements, as indicated above, are included in DOE Order 4700.1, *Project Management System*, and WHC quality assurance procedures and instructions. For nuclear safety equivalency, consideration should be given to ensuring that the appropriate requirements of the referenced WHC procedures and instructions remain in effect (e.g., through use of SNFP-specific documents).

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10CFR50, APPENDIX A, GENERAL DESIGN CRITERIA

Background

In May 1971, the U.S. Atomic Energy Commission (now the NRC) made effective an amendment to 10CFR50 that added Appendix A, *General Design Criteria*. The Statements of Consideration that accompanied this change stated that the appendix was added to establish:

"... the minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. They also provide guidance in establishing the principal design criteria for other types of nuclear power plants. Principal design criteria established by an applicant and accepted by the Commission will be incorporated by reference in the construction permit. In considering the issuance of an operating license under Part 50, the Commission will require assurance that these criteria have been satisfied in the detailed design and construction of the facility and that any changes in such criteria are justified."

The addition of Appendix A, as originally proposed in July 1967 (32FR10213), was to include 64 criteria. The appendix, as issued, reduced the number to 55 but those remaining were not renumbered so that gaps exist in the numbering system.

Path Forward Discussion

General design criteria have been developed for ISFSIs and are included in 10CFR72 as Subpart F, *General Design Criteria*. These criteria are considered to be more appropriate for the CSB than the criteria of 10CFR50, Appendix A. An evaluation of 10CFR72 criteria for application to the CSB is included in Table 5.c. in response to §§72.120 through 72.130.

For the new SNFP Path Forward facilities other than the CSB, (referred to in the following discussions as "other facilities") a similar set of specific criteria does not exist. Therefore, in consideration of identifying potentially relevant safety issues and associated requirements, an evaluation of the appropriateness of each of the criteria of Appendix A to these other facilities, as they are currently envisioned, has been performed. The results of the evaluation are documented in Table A-2.

Table A-2 Potential Application of the General Design Criteria of 10CFR50,
Appendix A to New SNFP Path Forward Facilities other than the CSB

General Design Criteria (GDC)	Path Forward Discussion
1 <i>Quality standards and records</i>	This GDC establishes requirements relative to the use of appropriate quality standards, quality assurance, and record keeping. The use of appropriate standards for DOE facilities is addressed in DOE Orders 6430.1A, <i>General Design Criteria</i> , and 5480.4, <i>Environmental Protection, Safety, and Health Protection Standards</i> . The quality assurance and record keeping aspects of SNFP Path Forward activities are addressed in the Quality Assurance Criteria section of this attachment.
2 <i>Design bases for protection against natural phenomena</i>	The requirements of GDC 2 are addressed by application of SDC-4.1, <i>Standard Architectural-Civil Design Criteria</i> with the exception of seismic issues which are discussed in Table 5.c in response to §72.92.
3 <i>Fire protection</i>	Fire protection for new SNFP Path Forward activities is addressed in the discussion provided for §50.48 and Appendix R in Table 5.b.
4 <i>Environmental and dynamic effects design basis</i>	This GDC requires that SSCs important to safety be designed to accommodate the effects of such events as pipe breaks, missiles, and events external to the facility. For the SNFP Path Forward, the need to consider internal events is addressed by the safety classification criteria of WHC-CM-4-46, Section 9.0, Revision 1, <i>Safety Classification of Structures, Systems, and Components</i> . This document specifically requires that physical interactions such as seismic, pipe whip, jet impingement, water damage, and environmental changes be considered in assigning safety classification. Regarding external events, DOE Order 6430.1A, <i>General Design Criteria</i> , Section 0200-99.0.2, <i>Other Facilities and Operations</i> , requires that the design of special facilities consider the potential hazards from other on-site facilities and off-site sources of hazards that could affect the safe operation of the facility. The order states that off-site hazards can be due to the facility's proximity to airports, transportation routes, private industrial facilities, and military facilities. These are the same hazards as suggested in NRC Standard Review Plan 2.2-1.2.3.2, <i>Identification of Potential Hazards in Site Vicinity</i> , with the exception that the SRP also mentions pipelines and storage facilities. There are no large oil or gas pipelines or storage facilities on the Hanford Site.
5 <i>Sharing of structures, systems, and components</i>	This issue is addressed in Table 5.c in response to §72.122.
6-9 Reserved by the NRC	NA
10 <i>Reactor design</i>	This GDC concerns protection of fuel design limits. It is not applicable to the other facilities (i.e., other than the CSB). An analogous situation is addressed in the response provided for §50.46 in Table 5.b.
11 <i>Reactor inherent protection</i>	This GDC concerns the need to maintain a negative power coefficient during operation. It is not applicable to the other facilities as the MCO contents will always be maintained in a subcritical condition.
12 <i>Suppression of power oscillations</i>	This GDC is not applicable to the other facilities as the MCO contents will always be maintained in a subcritical condition.
13 <i>Instrumentation and control</i>	The requirements of GDC 13 should be considered for Safety Class 1 instrumentation and control systems. The need for Safety Class 1 instrumentation and control systems for the other facilities has not yet been established.
14 <i>Reactor coolant pressure boundary</i> 15 <i>Reactor coolant system design</i> 16 <i>Containment design</i>	These three GDC address the importance of maintaining the integrity of multiple fission product barriers. The issue is applicable to the other facilities relative to maintaining the integrity of the MCO and any additional Safety Class 1 barriers that may be required. Regarding the MCO, the requirements of 10CFR72 and the performance specification for the MCO (Womack 1995) meet the intent of these GDC. The need for radionuclide barriers for the other facilities other than that provided by the MCO has not been established.
17 <i>Electrical power systems</i>	General Design Criterion 17 addresses the need to maintain reliable onsite and offsite electric power systems. The offsite power system is to include two physically independent sources. The sensitivities of the maintenance of electrical power for the other facilities have not yet been established.
18 <i>Inspection and testing of electrical power systems</i>	The requirements of GDC 18 should be considered for Safety Class 1 electrical systems used in the other facilities. The need for Safety Class 1 electrical systems for the other facilities has not yet been established.

Table A-2 Potential Application of the General Design Criteria of 10CFR50.
Appendix A to New SNFP Path Forward Facilities other than the CSB

General Design Criteria (GDC)	Path Forward Discussion
19 Control Room	<p>This GDC establishes the need for a control room that remains habitable during accident conditions. It also establishes the need for an alternate location outside the control room from which hot shutdown can be achieved.</p> <p>The design of the other facilities has not progressed to the point to properly address this issue. However, it is not expected that post-accident radiological releases or that required mitigating actions will be such that radiological protection of operators will be a concern. In addition, WHC-CM-4-46, Section 9.0, Revision 1, <i>Safety Classification of Structures, Systems, and Components</i>, addresses the potential need to provide radiological protection for the operators. It requires that operator exposure be limited to 5 rem if they are needed to mitigate the accident. These aspects of the other facilities and the conditioning process should be confirmed during the final design of the facility.</p>
20 Protection system functions 21 Protection system reliability and testability 22 Protection system independence 23 Protection system failure modes 24 Separation of protection and control systems	<p>These five GDC establish requirements for protective systems needed to achieve a reactor scram and initiate engineered safety features.</p> <p>In terms of preserving the safety function, relevant portions of these GDC should be considered for application to Safety Class 1 instrumentation and control systems used in the other facilities. The need for Safety Class 1 instrumentation and control for the other facilities has not yet been established.</p>
25 Protection system requirements for reactivity control malfunctions 26 Reactivity control system redundancy and capability 27 Combined reactivity control systems capability 28 Reactivity limits	<p>These four GDC deal with the need to maintain reactivity control during reactor operation and during shutdown conditions. They are not applicable to the other facilities as the spent nuclear fuel is to be maintained in a subcritical condition without reliance on control systems.</p>
29 Protection against anticipated operational occurrences	<p>This GDC requires that protective systems have a high degree of probability of performing their safety function during anticipated operational occurrences. This GDC should be applied to Safety Class 1 instrumentation and control systems of the other facilities. The need for Safety Class 1 instrumentation and control for the other facilities has not yet been established.</p>
30 Quality of the reactor coolant pressure boundary	<p>This GDC requires the components of the reactor coolant pressure boundary (RCPB) be of the highest practical quality and that a means of detecting and locating RCPB leakage be provided. These issues are applicable to the MCO for which the requirements of 10CFR72 and the performance specification for the MCO (Womack 1995) meet the intent of this GDC.</p>
31 Fracture prevention of reactor coolant pressure boundary	<p>This GDC is not applicable to the SNFP Path Forward for the reasons provided in Table 5.b for 10CFR50, Appendix G.</p>
32 Inspection of reactor coolant pressure boundary	<p>This GDC concerns the need to design the facility to allow for periodic inspection of the RCPB. It also addresses the material selections of the RCPB. The need to provide for inspection of Safety Class 1 SSCs for the other facilities is established by DOE Order 6430.1A, <i>General Design Criteria</i>, Section 1300-3.5, <i>Maintenance</i>. The requirements that will guide the material selection of the MCO are provided in the MCO performance specification (Womack 1955).</p>
33 Reactor coolant makeup	<p>This GDC requires the capability to provide primary system makeup water for small breaks that may occur in the RCPB. It is not applicable to the other facilities.</p>
34 Residual heat removal 35 Emergency core cooling 36 Inspection of emergency core cooling system 37 Testing of emergency core cooling system	<p>The requirements of these four GDC are very specific to power reactors. The need for residual heat removal and emergency core cooling is analogous to the need to maintain cooling of the MCOs. This is addressed in the response provided for §50.46 in Table 5.b. Requirements for the other facilities relative to inspection and testing are established in DOE Order 6430.1A, <i>General Design Criteria</i>, Sections 1300-3.5, <i>Maintenance</i>, and 1300-3.6, <i>Testing</i>.</p>

Table A-2 Potential Application of the General Design Criteria of 10CFR50.
Appendix A to New SNFP Path Forward Facilities other than the CSB

General Design Criteria (GDC)	Path Forward Discussion
38 Containment heat removal 39 Inspection of containment heat removal 40 Testing of containment heat removal system	These three GDC establish requirements for post-accident containment heat removal systems. They are not applicable to the other facilities as the facilities will not be sensitive to the need to remove post-accident heat from the buildings and such conditions do not result from credible accident scenarios.
41 Containment atmosphere cleanup 42 Inspection of containment atmosphere cleanup systems 43 Testing of containment atmosphere cleanup systems	These three GDC establish the requirements for the systems required for post-accident control and removal of hydrogen, oxygen, and fission products. The GDC should be considered for the design, testing, and inspection requirements of Safety Class 1 features of the other facilities used for hydrogen control and the control of radionuclide releases.
44 Cooling water 45 Inspection of cooling water systems 46 Testing of cooling water systems	These three GDC establish requirements for water systems needed to transfer decay heat to the ultimate heat sink. The importance of these GDC to the other facilities will depend upon the importance of the MCD cooling systems to accident prevention and mitigation. Equivalent single failure protection and testing requirements of these GDC, as would be applicable to a Safety Class 1 system if required, are addressed by DOE Order 6430 1A, <i>General Design Criteria</i> (Sections 1300-3.3 and 1300-3.6). Requirements for the other facilities relative to inspection and testing are established in DOE Order 6430 1A, <i>General Design Criteria</i> , Sections 1300-3.5, <i>Maintenance</i> , and 1300-3.6, <i>Testing</i> .
47-49 Reserved by the NRC	NA
50 Containment design basis 51 Fracture prevention of containment pressure boundary	These two GDC establish requirements for containment integrity in the event of a loss-of-coolant accident (LOCA) that pressurizes the containment. If the other facilities require confinement, they will be designed to maintain a negative pressure during accident conditions. As such, these GDC are not applicable to the other facilities.
52 Capability for containment leakage rate testing 53 Provisions for containment testing and inspection	These two GDC concern containment leakage that may occur while the containment is pressurized following a LOCA, and, as such, are not directly applicable to the other facilities. However, these facilities have an analogous situation in the need to confirm that leakage from the confinements will not be so great that the required negative pressure cannot be maintained. Such confirmations would be provided by the Technical Safety Requirements required to ensure that the facilities do not operate outside the bounds of the safety analyses. This requirement is established by DOE Order 5480 22, <i>Technical Safety Requirements</i> .
54 Piping systems penetrating containment 55 Reactor coolant pressure boundary penetrating containment	These two GDC concern the need to provide for containment isolation to prevent liquid and gaseous release paths that would bypass the containment. They are not applicable to the other facilities as, even under accident conditions, there would not be a significant leakage of radionuclides through such paths (e.g., building vents and drains). This design feature of the other facilities should be confirmed during the final design of the facilities.
56 Primary containment isolation	This GDC addresses the need to isolate paths that connect directly to containment. The concern is that with the post-LOCA pressurization of containment, such paths could result in a significant release of radionuclides. This is not applicable to the confinements of the other facilities as they will not be pressurized during accident conditions.
57 Closed system isolation valves	Same response as provided above for GDC 54 and 55.
58-59 Reserved by the NRC	NA
60 Control of releases of radioactive materials to the environment	This GDC contains requirements relative to controlling the release of radioactive material in liquid and gaseous effluents and to the handling of solid radwaste. These controls for special DOE facilities, such as the other facilities, are provided in DOE Order 5820 2A, <i>Radioactive Waste Management</i> , and DOE Order 6430 1A, <i>General Design Criteria</i> , Sections 1300-8, <i>Waste Management</i> , and 1300-9, <i>Effluent Control and Monitoring</i> .
61 Fuel storage and handling and radioactivity control	This GDC contains requirements for fuel handling and storage. The specific items mentioned in the GDC and the location of where each is implemented for DOE facilities is as follows: capability for inspection and testing (DOE Order 6430 1A, <i>General Design Criteria</i> , Sections 1300-3.5, <i>Maintenance</i> , and 1300-3.6, <i>Testing</i>); radiation shielding (DOE Order 6430 1A, <i>General Design Criteria</i> , Section 1300-6.2, <i>Shielding Design</i>); provisions for containment and confinement (DOE Order 6430 1A, <i>General Design Criteria</i> , Section 1300-7, <i>Confinement Systems</i>); and residual heat removal capability and the potential for a loss of fuel storage coolant inventory (neither is applicable to the other facilities).

Table A-2 Potential Application of the General Design Criteria of 10CFR50.
Appendix A to New SNFP Path Forward Facilities other than the CSB

General Design Criteria (GDC)	Path Forward Discussion
62 Prevention of criticality in fuel storage and handling	This GDC states that criticality in fuel storage and handling should be prevented by physical systems or processes, preferably by use of geometrically safe configurations. DOE Order 5480.24, <i>Nuclear Criticality Safety</i> , establishes sufficient requirements to implement GDC 62. Also see the discussion provided for SRP 9.1.2 in Table 6.b.
63 Monitoring fuel and waste storage	This GDC contains requirements to detect the loss of heat removal and excessive radiation levels. The sensitivity of loss of MCO heat removal for the other facilities has not yet been established. The need to provide for radiation monitoring is provided for by DOE Order 6430.1A, <i>General Design Criteria, Section 1300-6.5, Monitoring, Warning, and Alarm Systems</i> .
64 Monitoring radioactivity releases	This GDC establishes requirements for monitoring the containment for radioactivity during normal operations, anticipated operational occurrence, and accidents. These requirements for the other facilities are established by DOE Order 6430.1A, <i>General Design Criteria, Sections 1300-6.5, Monitoring, Warning, and Alarm Systems, 1300-1.4, Guidance on Limiting Exposure of the Public, and 1300-9, Effluent Control and Monitoring</i> .

Conclusions:

At this stage in the accident analyses and design of the new SNFP Path Forward facilities other than the CSB, conclusions cannot be reached as to whether or not existing DOE and WHC design criteria are adequate for the other facilities such that NRC nuclear safety equivalency is achieved without application of the criteria of 10CFR50, Appendix A.

Actions for Consideration:

During the final design of the new SNFP Path Forward facilities other than the CSB, consideration should be given to having the AEs review 1) the conclusions of the general design criteria discussion of this attachment, and the general design criteria of 10CFR50, Appendix A to determine if NRC nuclear safety equivalency is achieved without application of any of the criteria of 10CFR50, Appendix A.

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ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT

Nuclear Safety Issue

Section 50.49 provides for qualification of safety-related electrical equipment that is relied upon to remain functional during and following design bases events (harsh conditions). This relates to: the integrity of the reactor coolant pressure boundary; the capability to shut down the reactor and maintain it in a safe shutdown condition; and, the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the 10CFR100 guidelines. Design basis events are defined as conditions of normal operation, including anticipated operational occurrences, design basis accidents, external events, and natural phenomena for which the plant must be designed.

Background

Structures, systems, and components (SSCs) important to safety must be designed to accommodate the effects of environmental conditions (i.e. remain functional under postulated accident conditions), and for such SSCs, design control measures such as testing shall be used to check the adequacy of design. The equipment qualification program required by §50.49 must include and be based on: temperature and pressure; humidity; chemical effects; radiation; aging; submergence; synergistic effects; and margins. Section 50.49 provides various methods that are acceptable for qualifying electrical equipment. Section 50.49 does not include requirements for seismic and dynamic qualification; protection of electrical equipment against other natural phenomena and external events; and equipment located in a mild environment.

Additional details on requirements that apply to equipment qualification are contained in Regulatory Guide 1.89, *Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants*. In addition, general requirements are provided in General Design Criteria 4 and 23 of 10CFR50, Appendix A, *General Design Criteria for Nuclear Power Plants*.

In addition to qualification of safety-related electrical equipment, the environmental qualification requirements of §50.49 also apply to non-safety equipment that can adversely impact safety functions of safety-related equipment and to post-accident monitoring equipment. Additional detail on requirements that apply to post-accident monitoring equipment are defined in Regulatory Guide 1.97, *Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident*.

Section 50.49(d) requires determination of performance information as detailed in §50.49(d)(1-3). The performance information includes:

- Performance characteristics of the electrical equipment during and following design basis accidents (DBAs).
- Voltage, frequency, load and other electrical characteristics for which the performance during and following DBAs can be ensured.

- Environmental conditions, including temperature, pressure, humidity, radiation, chemicals, and submergence at the location where the equipment must perform.

Section 50.49(d) requires that this information be maintained in an auditable form for the entire period when such equipment is installed in the plant or is stored for future use.

Each item of electrical equipment must be qualified by one of the methods stated in §50.49(f)(1-4). These methods include:

- Testing an identical item of equipment under identical conditions or under similar conditions with supporting analysis.
- Testing a similar item of equipment with supporting analysis to show that the equipment to be qualified is acceptable.
- Experience with identical or similar equipment under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable.
- Analysis in combination with partial type test data that supports the analytical assumptions and conclusions.

Path Forward Resolution

DOE Order 6430.1A, *General Design Criteria*, Section 1300-3.4.2, *Environmental Qualification of Equipment*, states that "Equipment qualification shall provide assurance that safety class items will be capable of performing required safety functions under DBA (i.e., harsh environment) conditions. The qualification shall demonstrate that the equipment can at least perform for the period of time that its safety functions are required. Subsequent equipment failure, after its safety function is no longer required, may be allowable. Temperature, pressure, and humidity environments shall be based on the most severe postulated accident affecting the particular item. The postulated environment shall reflect an environment that considers both radiological composition and chemical composition of all material physical forms likely to affect the equipment. DOE Order 6430.1A does not require that aging, submergence, synergistic effects, and margins be considered for qualification of electrical equipment.

Chapter 3, *Hazard and Accident Analyses*, of DOE-STD-3009-94, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports*, identifies and assesses hazards to evaluate the potential internal, external, and natural-phenomena events that can cause identified hazards to develop into accidents. The hazards analysis is to include the complete spectrum of accidents that may occur including design basis accidents.

WHC-CM-4-46, *Safety Analysis Manual*, in Section 9.0, Revision 1, *Safety Classification of Structures, Systems, and Components*, requires that a safety equipment list (SEL) be developed containing all Safety Class 1, 2, and 3 items in each facility. Section 9.0 also contains procedural requirements and criteria for designation of SSCs into Safety Class 1, 2, or 3. Safety Class 1 and 2 SSCs are identified and evaluated against the complete spectrum of accidents that may occur resulting from the hazards analyses.

As noted above, §50.49 requires that the electrical equipment remain functional during and following design basis events. Section 1300-3.4.2 of DOE Order 6430.1A, requires only that the equipment performs during the period when its safety functions are required. This is considered acceptable from a safety consideration standpoint. Section C.4 of Regulatory Guide 1.89 presents criteria to assist in determining when the equipment safety function is required.

It states in Appendix B, *Design Criteria*, to Section 9.0 of WHC-CM-4-46 that Safety Class 1 items are equivalent to safety class items as defined in DOE Order 6430.1A. Safety Class 1 items must therefore meet the environmental qualification requirements stated above from §1300-3.4.2 (i.e., harsh environment). Safety Class 2 systems are non-safety class systems, as defined in DOE Order 6430.1A, which have been designated Safety Class 2 primarily because of their importance to onsite worker safety. Section 9.0 requires the application of criteria for non-safety class items, as defined in DOE Order 6430.1A, be applied to Safety Class 2 and 3 items. The DOE Order 6430.1A criteria state that failure of non-safety class items shall not adversely affect the environment or the safety and health of the public or prevent safety class items from performing their required functions.

Appendix B of Section 9.0 of WHC-CM-4-46, also presents requirements for Safety Class 1, 2, and 3 SSCs to: 1) "Apply design, procurement, fabrication, construction, test, maintenance, and QA requirements as necessary to preserve those SSC attributes that perform or maintain the required safety or environmental protection functions as defined in the SEL"; and, 2) "Design to perform throughout the period of time that the safety or environmental protection function is required. Consider exposure to abnormal temperature, pressure, humidity, radiation, loading and chemical interactions."

Appendix B of Section 9.0 of WHC-CM-4-46 exempts demonstration of Safety Class 2 systems from performance of aging concerns. It states that "... periodic testing for active functions and periodic surveillance for degradation shall be incorporated as part of the applicable maintenance and surveillance programs. Similarly, periodic surveillance or in-place surveillance systems shall be used when process chemical conditions are uncertain or vendor compliance to conditions such as radiation resistance cannot be demonstrated."

Demonstration of performance following aging is not included in the environmental qualification program for Safety Class 1, 2 or 3 systems except as noted above for treating aging of Safety Class 2 systems. It could be implied that similar testing and surveillance provisions would apply to Safety Class 1 systems, as a minimum, although such requirements are not found in WHC procedures. Section 50.49(e)(5) requires testing the performance of equipment that has been preconditioned to end-of-installed life conditions prior to installation in the plant. Regulatory Guide 1.89 recognizes that there are a number of uncertainties regarding the aging process. However, equivalence to §50.49 is not currently attained by the DOE requirements for aging considerations.

Although submergence is not specifically mentioned in DOE requirements, it would be evaluated based upon location of the equipment during and following DBAs. Application of margins to account for unquantified uncertainty, although not specifically mentioned, would be included in the development of information for Chapter 5.0, *Derivation of Technical Safety*

Requirements, of DOE-STD-3009-94. Suggested values for margins are contained within Section 6.3.1.5, *Margins*, of IEEE Std 323-1974, *IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations*. Section 2.0, *Safety Analysis Process*, of WHC-CM-4-46, requires consideration of synergistic effects during preparation of the SAR.

It states in §1300-3.4.2 of DOE Order 6430.1A, that failure of non-safety class items shall not adversely affect the environment or the safety and health of the public or prevent safety class items from performing their required function. Section 9.0 of WHC-CM-4-46 also states that support systems whose failure could prevent a Safety Class 1 or 2 SSC from performing its safety or environmental protection function shall be identified including the specific design attributes causing the failure. That attribute would then be included in the equipment qualification program as stated in Section 9.0 of WHC-CM-4-46.

As stated above, Section 50.49(b)(3) requires post-accident monitoring equipment to be included in the environmental qualification program. The required equipment must be capable of surviving the accident environment in which it is located for the length of time that its function is required. Regulatory Guide 1.97 provides specific guidance concerning the types of variables to be monitored. A structured task to determine post-accident monitoring equipment that needs to monitor plant variables and systems during and following an accident is provided by Section 3.2.3, *Identifying support systems for each SC-1, SC-2, and SC-3 system or structure*, of WHC-CM-4-46. This Section provides requirements to ensure that the support system can provide this function.

Section 50.49(d) requires determination of performance information that is detailed in §50.49(d)(1-3). This information will be generated during the hazards and accident analyses that will be performed to develop information for Chapter 3, *Hazard and Accident Analyses*, of DOE-STD-3009-94. Appendix E, *Qualification Documentation For Electric Equipment*, of Regulatory Guide 1.89, provides performance information that should be submitted to NRC to ensure that the environmental qualification conforms to §50.49 requirements.

Environmental testing requirements are provided by §50.49(f)(1-4). IEEE Std 323-1974 describes basic procedures for qualifying safety-related equipment including nonsafety-related equipment whose failure could adversely affect safety-related equipment. DOE documentation does not provide equivalent detail on environmental qualification testing that must be performed.

Conclusion:

Consideration of additional actions is necessary for new SNFP Path Forward facilities to achieve nuclear safety equivalence with the requirements of §50.49 for environmental qualification of electrical equipment. These are provided below.

Actions for Consideration:

In order to achieve equivalence, consideration should be given to ensuring that the electrical equipment qualification program include the elements identified above that are not included in the DOE requirements for:

- 1) Safety Class 1 equipment;
- 2) non-safety class equipment that could, upon

failure, adversely impact Safety Class 1 equipment in performance of their safety function; and, 3) certain post-accident monitoring equipment, as described in Regulatory Guide 1.97. These elements include the following:

1. The AEs for the new SNFP Path Forward facilities need to review \$50.49(e)(5) and Regulatory Guide 1.89 during the design process to determine the aging requirements for such electrical equipment.
2. Testing requirements as outlined in \$50.49(f)(1-4).

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