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MEMORANDUM FOR: Raymond F. Fraley, Executive Director
Advisory Committee on Reactor Safety

FROM: Guy A. Arlotto, Director
Division of Engineering Technology
Office of Nuclear Regulatory Research

SUBJECT: DRAFT 1, REGULATORY GUIDE 1.13, REVISION 2,
"SPENT FUEL STORAGE FACILITY DESIGN BASIS"

Enclosed for initial review of the ACRS Regulatory Activities Subcommittee are 20 copies of Revision 2 to Regulatory Guide 1.13 (Enclosure 1) and 20 copies of the Draft Value/Impact Assessment (Enclosure 2).

The draft regulatory guide is a proposed revision to Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," which is being revised to endorse ANSI N210-1976/ANS 57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations."

The draft regulatory guide, which was originally scheduled for review at the September 9th meeting, was withdrawn to insure the incorporation of all necessary input from Division Offices.

Since this draft is preliminary, additional staff efforts, including review and resolution of public comments, will be necessary prior to implementation of a regulatory position. ACRS Regulatory Activities Subcommittee comments and recommendations are requested on the proposed regulatory position.

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RG 1.13 (R-2)

1 DRAFT 1 OF REVISION 2 TO REGULATORY GUIDE 1.13

2 SPENT FUEL STORAGE FACILITY DESIGN BASIS

3 A. INTRODUCTION

4 General Design Criterion 61, "Fuel Storage and Handling and Radioactivity
5 Control," of Appendix A, "General Design Criteria for Nuclear Power Plants,"
6 to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities,"
7 requires that fuel storage and handling systems be designed to assure adequate
8 safety under normal and postulated accident conditions. It also requires that
9 these systems be designed (1) with a capability to permit appropriate periodic
10 inspection and testing of components important to safety, (2) with suitable
11 shielding for radiation protection, (3) with appropriate containment, confine-
12 ment, and filtering systems, (4) with a residual heat removal capability having
13 reliability and testability that reflects the importance to safety of decay
14 heat and other residual heat removal, and (5) to prevent significant reduction
15 in fuel storage coolant inventory under accident conditions. This guide
16 describes a method acceptable to the NRC staff for implementing this criterion.

17 B. DISCUSSION

18 Working Group ANS-57.2 of the American Nuclear Society Subcommittee ANS-50
19 has developed a standard which details minimum design requirements for 10 CFR
20 Part 50 light water reactor spent fuel storage facilities at nuclear power
21 stations. This standard was approved by the American National Standards
22 Committee N18, Nuclear Design Criteria. It was subsequently approved and
23 designated ANSI N210-1976/ANS-57.2, "Design Objectives for Light Water Reactor

1 Spent Fuel Storage Facilities at Nuclear Power Stations" by the American National
2 Standards Institute on April 12, 1976.

3 These facilities must be designed to:

- 4 a. Prevent loss of water from the fuel pool that would uncover fuel.
- 5 b. Protect the spent fuel from mechanical damage.
- 6 c. Provide the capability for limiting the potential offsite exposures
7 in the event of significant release of radioactivity from the fuel.

8 If spent fuel storage facilities are not provided with adequate protective
9 features, radioactive materials could be released to the environment as a result
10 of either loss of water from the storage pool or mechanical damage to fuel within
11 the pool.

12 1. Loss of Water from Storage Pool

13 Unless protective measures are taken, loss of water from a fuel storage
14 pool could cause overheating of the spent fuel, resultant damage to fuel clad-
15 ding integrity, and could result in a release of radioactive materials to the
16 environment. Natural events, such as earthquakes or high winds, could damage
17 the fuel pool either directly or by the generation of missiles. Earthquakes or
18 high winds could also cause structures or cranes to fall into the pool. Design-
19 ing the facility to withstand these occurrences without significant loss of
20 watertight integrity would alleviate these concerns.

21 Dropping of heavy loads, such as a 100-ton fuel cask, although of low
22 probability, should be considered in plant arrangements where such loads are
23 positioned or moved in or over the spent fuel pool. Cranes which are capable
24 of carrying heavy loads should be prevented, preferably by design rather than
25 interlocks, from moving into the vicinity of the pool.

1 The negative pressure in the fuel handling building during movement of
2 spent fuel should be at least minus 3.2 mm (-0.125 inches) water gauge to pre-
3 vent exfiltration and to assure that any activity released to the fuel handling
4 building will be treated by an engineered safety feature (ESF) grade filtration
5 system before release to the environment.

6 Even if the measures described above which are used to maintain the desired
7 negative pressure are followed, small leaks from the building may still occur as
8 a result of structural failure or other unforeseen events. For example, equip-
9 ment failures in systems connected to the pool could result in loss of water
10 from the pool if this loss is not prevented by design. A permanent fuel-pool-
11 coolant makeup system with a moderate capability, and with suitable redundancy
12 or backup, could prevent the fuel from being uncovered if these leaks should
13 occur. Early detection of pool leakage and fuel damage could be provided by
14 both pool-water-level monitors and radiation monitors. Both types of monitors
15 should be designed to alarm both locally and in a continuously manned location.
16 Timely operation of building filtration systems can be assured if these systems
17 are actuated by a signal from local radiation monitors.

18 2. Mechanical Damage to Fuel

19 The release of radioactive material from fuel may occur during the refueling
20 process, and at other times, as a result of fuel-cladding failures or mechanical
21 damage caused by the dropping of fuel elements or the dropping of objects onto
22 fuel elements.

23 Missiles generated by high winds are also a potential cause of mechanical
24 damage to fuel. This concern could be eliminated by designing the fuel storage

1 facility to preclude the possibility of the fuel being struck by missiles
2 generated by high winds.

3 3. Limiting Potential Offsite Exposures

4 A relatively small amount of mechanical damage to the fuel or fuel over-
5 heating might cause significant offsite doses of radiation if no dose reduction
6 features are provided. Use of a controlled leakage building surrounding the
7 fuel storage pool, with associated capability to limit releases of radioactive
8 material resulting from a refueling accident, would appear feasible and do much
9 to eliminate this concern.

10 For the spent fuel pool cooling, makeup and cleanup systems, the staff
11 will consider the design acceptable if it includes seismic Category 1 and
12 tornado protection for the water makeup source and its delivery system, the
13 pool structure, the building housing the pool, and the storage building's
14 filtration-ventilation systems. The pool building's filtration-ventilation
15 systems should be designed to meet the guidelines of Regulatory Guide 1.52,
16 "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety-
17 Feature Atmosphere Cleanup System Air Filtration and Absorption Units of Light-
18 Water-Cooled Nuclear Power Plants."

19 In all activities involving personnel exposure to radiation, attention
20 should be directed toward keeping occupational radiation as low as reasonably
21 achievable (ALARA). Efforts toward maintaining exposures ALARA should be
22 included in the design, construction, and operational phases. Guidance on
23 maintaining exposures ALARA is provided in Regulatory Guide 8.8, "Information
24 Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power
25 Stations Will Be As Low As Is Reasonably Achievable."

C. REGULATORY POSITION

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The requirements that are included in ANSI N210-1976/ANS-57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations"¹ are generally acceptable to the NRC staff. The staff has determined that this standard provides an adequate basis for complying with the requirements of General Design Criterion 61 "Fuel Storage and Handling and Radioactivity Control" of Appendix A "General Design Criteria for Nuclear Power Plants" to 10 CFR Part 50 as related to light water reactors and subject to the following clarifications and modifications:

1. The example in Section 4.2.4.3(1) should be modified. The inventory of radioactive materials that could possibly leak from the spent fuel building should correspond to the amount predicted to leak under the postulated maximum damage conditions resulting from the dropping of a spent fuel assembly in the spent fuel building. However, in any event, the inventory should not be less than the amount available due to rupture of all fuel rods of a spent fuel assembly. Other assumptions in the analysis should be consistent with those given in Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."²

2. In addition to meeting the requirements of Section 5.1.12 the maximum potential kinetic energy capable of being developed by those objects handled

¹Copies may be obtained from the American Nuclear Society, 555 North Kensington Avenue, La Grange Park, Illinois 60525
²Copies of Regulatory Guides may be obtained from the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

1 above stored spent fuel, if dropped, is not to exceed the kinetic energy of
2 one fuel assembly and its associated handling tool when dropped from the height
3 at which it is normally handled above the spent fuel pool storage racks.

4 3. In addition to meeting the requirements of Section 5.1.3, boiling of
5 the pool water may be permitted only when the resulting thermal loads are
6 properly accounted for in the design of the pool structure, the storage racks,
7 and other safety-related structures, equipment, and systems.

8 4. In addition to meeting the requirements of Section 5.1.3, the fuel
9 storage pool should be designed (a) to keep tornado winds and missiles generated
10 by these winds from causing significant loss of watertight integrity of the
11 fuel storage pool and (b) to keep missiles generated by tornado winds from
12 striking the fuel. These requirements are discussed in Regulatory Guide 1.117,
13 "Tornado Design Classification." The fuel storage building, including walls
14 and roof, should be designed to prevent penetration by tornado missiles or from
15 seismic damage to assure that nothing bypasses the ESF grade filtration system
16 in the containment building. In the event an earthquake or a tornado missile
17 damages both the fuel pool containment and the fuel pool cooling system, no
18 credit can be given to the filtration system used to reduce the amount of
19 airborne radioactivity.

20 5. In addition to meeting the requirements of Section 5.1.5.3, provisions
21 should be made for handling highly radioactive non-fuel irradiated components
22 in fuel pools. Either the design of the retrieval system or administrative
23 controls should be included which would prohibit unknowing retrieval of
24 irradiated components.

1 6. In addition to meeting the requirements of Section 5.2.3.1, an interface
2 between the cask venting system and the installed building ventilation system
3 should be provided. This interface would provide for the proper handling of
4 the "vent-gas" generated from filling a dry, loaded cask with water and thereby
5 minimizing personnel exposure from the untreated off gas.

6 7. In order to limit the potential offsite release of radioactivity during
7 a Condition IV fuel handling accident, Section 5.3.3 should include the require-
8 ment that the released radioactivity be either contained or removed by filtration
9 so that the dose to an individual is less than 10 CFR Part 100 guidelines.
10 The calculated offsite dose to an individual from such an event should be well
11 within (approximately 25% of) the exposure guidelines of 10 CFR Part 100 using
12 appropriately conservative analytical methods and assumptions. In order to
13 assure that released activity does not bypass the filtration system, the
14 engineered safety feature fuel storage building ventilation should provide and
15 maintain a negative pressure of at least minus 3.2mm (-0.125 inches), water
16 gauge within the fuel storage building.

17 8. In addition to the requirements of Section 6.3.1, overhead handling
18 systems used to handle the spent fuel cask should be designed such that travel
19 directly over the spent fuel storage pool or safety-related equipment is not
20 possible. This should be verified by analysis to show that the physical structure
21 under all cask handling pathways will be adequately designed so that unacceptable
22 damage to the spent fuel storage facility or safety-related equipment will not
23 occur in the event of a load drop.

1 9. In addition to the references listed in Section 6.4.4, Safety Class
2 3, Seismic Category I and safety-related structures and equipment should be
3 subject to a quality assurance program which meets the applicable provisions
4 of Appendix B to 10 CFR Part 50. Further, those programs should obtain guidance
5 from Regulatory Guide 1.28 endorsing ANSI N45.2 "Quality Assurance Program
6 Requirements for Nuclear Facilities" and the applicable provisions of ANSI N45.2
7 daughter standards endorsed by Regulatory Guides.

8 The Regulatory Guides endorsing the applicable ANSI N45.2 daughter stan-
9 dards are as follows:

- 10 1.30 Quality Assurance Requirements for the Installation, Inspection,
11 and Testing of Instrumentation and Electric Equipment (N45.2.4).
- 12 1.38 Quality Assurance Requirements for Packaging, Shipping, Receiving,
13 Storage, and Handling of Items for Water-Cooled Nuclear Power
14 Plants (N45.2.2).
- 15 1.58 Qualification of Nuclear Power Plant Inspection, Examination,
16 and Testing Personnel (N45.2.6).
- 17 1.64 Quality Assurance Requirements for the Design of Nuclear Power
18 Plants (N45.2.11).
- 19 1.74 Quality Assurance Terms and Definitions (N45.2.10).
- 20 1.88 Collection, Storage, and Maintenance of Nuclear Power Plant
21 Quality Assurance Records (N45.2.9).
- 22 1.94 Quality Assurance Requirements for Installation, Inspection,
23 and Testing of Structural Concrete and Structural Steel During
24 the Construction Phase of Nuclear Power Plants (N45.2.5).

- 1 1.116 Quality Assurance Requirements for Installation, Inspection,
2 and Testing of Mechanical Equipment and Systems (N45.2.8).
3 1.123 Quality Assurance Requirements for Control of Procurement of
4 Items and Services for Nuclear Power Plants (N45.2.13).

5 10. The spent fuel pool water temperature of 65.6°C (150°F) stated in Sec-
6 tion 6.6.1(2)(a) exceeds the NRC staff recommended limit. With the normal
7 cooling system in operation, the pool water temperature should be kept at
8 or below 60°C (140°F) with full core offload except when the pool water
9 temperature is based on comparative analyses of the pool conditions that
10 have been found acceptable previously. The spent fuel pool water tempera-
11 ture recommended limits for normal and abnormal cases are indicated in the
12 table below.

13 NORMAL OPERATION

14	<u>Case I</u>	<u>Case II</u>
15	. both trains operational	. both trains operational
16	. normal refueling	. full core offload
17	. pool full of spent fuel	. pool full of spent fuel
18	<u>Maximum operating temperature</u>	<u>Maximum operating temperature</u>
19	< 48.9°C (120 °F)	< 60°C (140° F)
20	based on fogging criteria and	to protect the ion exchange
21	personnel comfort	resin from degradation

1 ABNORMAL OPERATION

2 Case III

Case IV

3 . one train operational

. no cooling loops operational

4 . normal refueling

. full core offload

5 . pool full of spent fuel

. pool full of spent fuel

6 Maximum operating temperature

Pool boiling permitted

7 <60°C (140°F)

8 11. A nuclear criticality safety analysis should be performed in accordance
9 with Annex A for each light water reactor spent fuel storage facility that
10 involves the handling, transfer, or storage of spent fuel assemblies.

11 12. Sections 6.4 and 9 of ANS 57.2 lists codes and standards that are referenced
12 in this standard. Endorsement of ANS 57.2 by this regulatory guide does
13 not constitute an endorsement of the referenced codes and standards.

14 D. IMPLEMENTATION

15 The purpose of this section is to provide information to applicants regard-
16 ing the NRC staff's plans for using this regulatory guide.

17 This guide reflects current NRC staff practice for construction permit
18 review. Therefore, except in those cases in which the applicant proposes an
19 acceptable alternative method for complying with specified portions of the
20 Commission regulations, the methods described herein will be used in the
21 evaluation of license applications docketed after _____.

1 ANNEX A

2 Nuclear Criticality Safety

3 1. Scope of Nuclear Criticality Safety Assessment

4 1.1 A nuclear criticality safety analysis shall be performed for each
5 light water reactor spent fuel storage facility system that involves
6 the handling, transfer, or storage of spent fuel assemblies.

7 1.2 The nuclear criticality safety analysis shall demonstrate that
8 each reactor spent fuel storage facility system is subcritical
9 (k_{eff} shall not exceed 0.95).

10 1.3 The nuclear criticality safety analysis shall include consideration
11 of all credible normal and abnormal operating occurrences, including:
12 a) Accidental tipping or falling of a spent fuel assembly
13 b) Accidental tipping or falling of a storage rack during transfer
14 c) Misplacement of a spent fuel assembly
15 d) Accumulation of solids containing fissile materials on the
16 pool floor or at locations in the cooling water system.
17 e) Fuel drop accidents
18 f) Stuck fuel assembly/crane uplifting forces
19 g) Horizontal motion of fuel before complete removal from rack
20 h) Placing a fuel assembly along the outside of rack
21 i) Objects that may fall onto the stored spent fuel assemblies

1 . 1.4 At all locations in the reactor spent fuel storage facility where
2 spent fuel is handled or stored, the nuclear criticality safety
3 analysis shall demonstrate that criticality could not occur without
4 at least two unlikely, independent, and concurrent failures or
5 operating limit violations.

6 1.5 The nuclear criticality safety analysis shall explicitly identify
7 spent fuel assembly characteristics upon which subcriticality in the
8 reactor spent fuel storage facility depends.

9 1.6 The nuclear criticality safety analysis shall explicitly identify
10 design limits upon which subcriticality depends that require physical
11 verification at the completion of fabrication or construction.

12 1.7 The nuclear criticality safety analysis shall explicitly identify
13 operating limits upon which subcriticality depends that require
14 implementation in operating procedures.

15 2. Calculational Methods and Codes

16 Methods used to calculate subcriticality shall be validated in accordance
17 with Regulatory Guide 3.41, "Validation of Calculational Methods for Nuclear
18 Criticality Safety." (Endorses ANSI N16.9-1975)

1 3. Method to Establish Subcriticality

2 3.1 The evaluated multiplication factor of fuel in the spent fuel
3 storage racks under normal and credible abnormal conditions shall
4 be equal to or less than an established maximum allowable multi-
5 plication factor k_a ; i.e.,

6
$$k_s \leq k_a \quad (\text{Eq. 1})$$

7 where

8 k_s = the evaluated maximum multiplicaton factor of fuel in the
9 spent fuel storage racks, including any necessary allowance
10 for statistical uncertainties in the calculational technique
11 such as in Monte Carlo calculations.

12 The maximum allowable multiplication factor shall be calculated
13 from the expression:

14
$$k_a = k_c - \Delta k_u - \Delta k_m \quad (\text{Eq. 2})$$

15 where

16 $k_c = k_{\text{eff}}$ computed for the most reactive fuel assembly at the most
17 reactive point by the same calculational method which was used
18 for the benchmark experiments.

19 Note: k_c is the value of k_{eff} that results from the calcu-
20 lation of the benchmark experiments using a particular
21 calculational method. The value represents a combina-
22 tion of theoretical technique and numerical data. (For
23 more detail, see Regulatory Guide 3.41, "Validation of
24 Calculational Methods for Nuclear Criticality Safety.")

1 b) the most reactive fuel assembly to be stored based on a minimum
2 confirmed burn up. [~~if credit is taken for burnup, an allowable~~
3 ~~fuel assembly reactivity shall be established and it shall be~~
4 ~~shown by actual measurement that each fuel assembly meets this~~
5 ~~criterion before it is allowed to be placed in storage.~~] (See
6 Annex B.)

7 Both types of rack modules may be present in the same storage
8 pool.

9 4.2 Determination of the most reactive spent fuel assembly shall include
10 consideration of the following parameters:

- 11 . maximum fissile fuel loading,
- 12 . fuel rod diameter,
- 13 . fuel rod cladding material and thickness,
- 14 . fuel pellet density,
- 15 . fuel rod pitch and total number of fuel rods within assembly,
- 16 . absence of fuel rods in certain locations, and
- 17 . burnable poison content.

18 4.3 The fuel assembly arrangement assumed in storage rack design shall
19 be the arrangement that results in the highest value of k_s considering:

- 20 a) spacing between assemblies,
- 21 b) moderation between assemblies, and
- 22 c) fixed neutron absorbers between assemblies.

1 4.4 Determination of the spent fuel assembly arrangement with the highest
2 value of k_s shall include consideration of the following:

- 3 a) eccentricity of fuel bundle location within the racks and
4 variations in spacing among adjacent bundles,
5 b) dimensional tolerances,
6 c) construction materials,
7 d) fuel and moderator density (allowance for void formations and
8 temperature of water between and within assemblies),
9 e) presence of the remaining amount of fixed neutron absorbers in
10 fuel assembly, and
11 f) presence of structural material and fixed neutron absorber in
12 cell walls between assemblies.

13 4.5 Determination of burn up for storage shall be made in racks for which
14 credit is taken for burn up. The following methods are acceptable:

- 15 a) a minimum allowed fuel assembly reactivity shall be established and
16 a reactivity measurement shall be performed to assure that each assembly
17 meets this criterion; or
- 18 b) a minimum fuel assembly burn up value shall be established as deter-
19 mined by initial fuel assembly enrichment or other correlative param-
20 eters and a measurement shall be performed to assure each fuel assembly
21 meets the established criterion; or

1 c) a minimum fuel assembly burn up value shall be established as deter-
2 mined by initial fuel assembly enrichment or other correlative param-
3 eters and an analysis of each fuel assembly's exposure history shall
4 be performed to determine its burn up. The analyses shall be performed
5 under strict administrative control using approved written procedures.
6 The procedures shall provide for independent checks of each step of
7 the analysis by a second qualified person using nuclear criticality
8 safety assessment criteria described in Section 1.4.

9 The uncertainties in determining fuel assembly storage acceptance criteria
10 shall be considered in establishing storage rack reactivity, and auditable
11 records shall be kept of the method used to determine fuel assembly storage
12 acceptance criterion for as long as the fuel assemblies are stored in the
13 racks.

14 Consideration shall be given to the axial distribution of burn up in the
15 fuel assembly and a limit shall be set on the length of the fuel assembly
16 which is permitted to have a lower average burn up than the fuel assembly
17 average.

18 5. Use of Neutron Absorbers in Storage Rack Design

19 5.1 Fixed neutron absorbers may be used for criticality control under
20 the following conditions:

21 a) The effect of neutron-absorbing materials of construction or
22 added fixed neutron-absorbers may be included in the evaluation

1 if they are designed and fabricated so as to preclude inadver-
2 tent removal by mechanical or chemical action.

3 b) Fixed neutron absorbers shall be an integral, non-removable part
4 of the storage rack.

5 c) When a fixed neutron absorber is used as the primary nuclear
6 criticality safety control, there shall be provision to:

7 1) initially confirm absorber presence in the storage rack,
8 and

9 2) periodically verify continued presence of absorber.

10 5.2 The presence of a soluble neutron absorber in the pool water
11 shall not normally be used in the evaluation of k_s . However, when
12 calculating the effects of Condition IV faults, realistic initial
13 conditions (e.g., the presence of soluble boron) may be assumed for
14 the fuel pool and fuel assemblies.

1

ANNEX B

2

Most Reactive Fuel Assembly to be Stored

3

Based on a Minimum Confirmed Burnup

4

If credit is to be taken for fuel burnup in the design of spent fuel storage racks, an acceptable basis for setting and meeting the limit must be established.

5

6

The rationale for this basis will evolve from many rather complex considerations.

7

Consideration should be given to the fact that the reactivity of any given spent fuel assembly will depend on initial enrichment, ^{235}U depletion, amount of burnable poison, plutonium buildin and fission product burnable poison depletion, and the fact that the rates of depletion and plutonium and fission product buildin are not necessarily the same.

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Consideration should be given to how burnup limits are selected and specified for a particular fuel type:

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The allowable ^{235}U depletion in the spent fuels without burnable poison must not be set too high. If too much depletion is credited in the analysis compared to the range of ^{235}U depletion in spent fuel assemblies to be stored, the design could be nonconservative from the standpoint of criticality safety. On the other hand, if too little depletion is credited in the analysis compared to the spent fuel to be stored, then the design will be conservative. Thus a maximum depletion to be allowed in design

1 can be established consistent with the range of ^{235}U depletions expected
2 in the spent fuel assemblies to be stored. (This limit would then
3 correspond to the minimum depletion that would be allowed in a particular
4 fuel assembly type destined to be stored in the racks.)

5 The allowable plutonium content in the spent fuel upon which design would
6 be based must not be set too low. If design is based on too little pluto-
7 nium compared to the range of plutonium concentrations that may be in the
8 spent fuel assemblies to be stored in the racks, the design could be non-
9 conservative from the standpoint of nuclear criticality safety. On the
10 other hand, if too much plutonium is credited in the analysis of the
11 storage racks compared to the spent fuel assemblies to be stored, then
12 the design would be conservative. Thus, a minimum plutonium content to
13 be allowed in design can be established consistent with the range of
14 plutonium concentrations expected in the spent fuel assemblies to be stored.

15 (This limit would then correspond to the maximum plutonium content that
16 would be allowed in a particular fuel assembly type destined to be stored
17 in the racks.)

18 Credit for fission product content presents special problems, such as the
19 identities and quantities of the various fission products present and how
20 to evaluate the effect of decay rates on the credit taken. The allowable
21 fission product content in the spent fuel upon which design would be based
22 must not be set too high. If design is based on too high of a fission
23 product content compared to the range of fission product concentrations
24 that may be in the spent fuel assemblies to be stored in the racks, the

1 design could be non-conservative from the standpoint of criticality safety.
2 On the other hand, if too few fission products are credited in the analysis
3 of the racks compared to the spent fuel assemblies to be stored, then the
4 design would be conservative. Thus, with proper consideration a maximum
5 fission product content to be allowed in design could be established consis-
6 tent with the range of fission product concentrations expected in the spent
7 fuel to be stored.

8 (This limit would then correspond to the minimum fission product content
9 that would be allowed in a particular fuel assembly type to be stored in
10 the racks.)

11 Finally, consideration should be given to the practical implementation of
12 the spent fuel screening process. Factors to be considered in choosing the
13 screening method should include: [~~Depletion-of-²³⁵U-and-plutonium-and-fission~~
14 ~~product-buildin-cannot-be-easily-or-practically-determined-analytically--An~~
15 ~~obvious-approach-would-be-to-translate-the-allowable-burnup-to-a-net-allowable~~
16 ~~fuel-assembly-reactivity-and-then-measure-every-fuel-assembly-to-confirm-that~~
17 ~~the-minimum-criterion-is-met:]~~

- 18 - accuracy of the method in determining the storage rack reactivity;
19 - reproducibility of the result, i.e., what is the confidence in the
20 result?
21 - simplicity of the procedure; i.e., how much disturbance to other opera-
22 tions is involved?;
23 - accountability, i.e., ease and completeness of recordkeeping; and
24 - auditability.

1 VALUE/IMPACT ASSESSMENT ON NUCLEAR POWER PLANT
2 SPENT FUEL STORAGE FACILITY DESIGN

3 1. PROPOSED ACTION

4 1.1 Description

5 Each nuclear power plant has a spent fuel storage facility. General Design
6 Criteria 61, "Fuel Storage and Handling and Radioactivity Control" of Appendix A,
7 "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic
8 Licensing of Production and Utilization Facilities," requires that fuel storage
9 and handling systems be designed to assure adequate safety under normal and
10 postulated accident conditions. The proposed action would provide an acceptable
11 method for implementing this criterion. This action would be an update of
12 Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis."

13 1.2 Need for Proposed Action

14 Since Regulatory Guide 1.13 was last published in December of 1975, addi-
15 tional guidance has been provided in the form of ANSI standards and NUREG reports.
16 The Office of Nuclear Reactor Regulation has requested this guide be updated.

1 1.3 Value/Impact of Proposed Action

2 1.3.1 NRC

3 The applicants' basis for the design of the spent fuel storage facility
4 will be the same as that used by the staff in its review of a construction
5 permit application. Therefore, there should be a minimum of cases where the
6 applicant and the staff radically disagree on the design criteria.

7 1.3.2 Government Agencies

8 Applicable only if the agency, such as TVA, is an applicant.

9 1.3.3 Industry

10 The value/impact on the applicant will be the same as for the NRC staff.

11 1.3.4 Public

12 No major impact on the public can be foreseen.

13 1.4 Decision on Proposed Action

14 The guidance furnished on the design basis for the spent fuel storage
15 facility should be updated.

16 2. TECHNICAL APPROACH

17 The American Nuclear Society published ANS-57.2 (ANSI N210), "Design
18 Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear
19 Power Stations." Part of the update of Regulatory Guide 1.13 would be an

1 evaluation of this standard and possible endorsement by the NRC. Also recommenda-
2 tions made by Task A-36 which were published in NUREG-0612, "Control of Heavy
3 Loads at Nuclear Power Plants" would also be included.

4 3. PROCEDURAL APPROACH

5 Since Regulatory Guide 1.13 already deals with the proposed action, logic
6 dictates that this guide be updated.

7 4. STATUTORY CONSIDERATIONS

8 4.1 NRC AUTHORITY

9 This guide would fall under the authority and safety requirements of the
10 Atomic Energy Act of 1954, as amended. In particular under General Design
11 Criterion 61, Appendix A, 10 CFR Part 50 of the NRC's implementing regulations.

12 4.2 Need for NEPA Assessment

13 The proposed action is not a major action as defined by 10 CFR Part 51.5(a)(10)
14 and does not require an environmental impact statement.

15 5. CONCLUSION

16 Regulatory Guide 1.13 should be updated.