January 4, 2000

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
	)	
CAROLINA POWER & LIGHT COMPANY	)	Docket No. 50-400-LA
	)	
(Shearon Harris Nuclear Power Plant)	)	ASLBP No. 99-762-02-LA

EXHIBITS SUPPORTING THE SUMMARY OF FACTS, DATA, AND ARGUMENTS ON WHICH APPLICANT PROPOSES TO RELY AT THE SUBPART K ORAL ARGUMENT

VOLUME 4

EXHIBITS 2 - 8

EXHIBIT 2

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AFFIDAVIT OF STANLEY E. TURNER, Ph.D., PE

COUNTY OF PINELLAS	)	
	) ss:	
STATE OF FLORIDA	)	

I, Stanley E. Turner, being duly sworn, do on oath state as follows:

EXPERIENCE AND QUALIFICATIONS

1. I am the Senior Vice President and Chief Nuclear Scientist of Holtec International ("Holtec"). I have been employed by Holtec since 1987, shortly after the formation of Holtec. I have also supplied the nuclear analyses used by Holtec's principal and founder, Dr. Krishna P. Singh, before the formation of Holtec, beginning about 1981. My business address is 138 Alt. 19 South, Palm Harbor, Florida, 34683.

2. Holtec is a diversified energy technology company working for the electric power industry both in the United States and in many countries around the world. Holtec performs the majority of its work for nuclear power plants. Holtec develops and markets turnkey equipment for the nuclear power industry. Holtec performs all of the design and engineering, obtains necessary governmental regulatory approvals, effectuates

manufacturing, and performs on-site installation, testing, and commissioning into service of the products it sells. Holtec currently employs over 40 professional employees. A large number of Holtec's employees hold graduate degrees from prestigious national and international universities, with approximately 30 percent holding Ph.D.'s in science and engineering.

3. Holtec designs and markets both wet storage and dry storage systems for spent fuel storage and transport. Holtec's expertise in spent fuel storage system development and supply includes expertise in solid mechanics, heat transfer, nuclear physics, and nuclear components fabrication. One of Holtec's principal business areas is the design and installation of spent fuel storage racks for the expansion of wet storage capacity at nuclear power plants. Holtec's capability in these projects includes all of the design, analysis, and licensing reports required to obtain approval and implementation of the spent fuel storage rack capacity expansions. Holtec has a practically 100% market share in wet storage expansion. Holtec has completed turnkey projects for wet pool spent fuel storage capacity expansion in over 50 spent fuel pools in nuclear plants around the world.

4. I am Holtec's Chief Nuclear Scientist responsible for all nuclear analyses performed by Holtec. Included in my role as Senior Vice President and Chief Nuclear Scientist is responsibility for all nuclear criticality analyses for spent fuel storage systems.

5. I received my Ph.D. in Nuclear Chemistry from the University of Texas in 1951. I have been elected to the academic honor societies of Sigma Pi Sigma, Phi

Lambda Epsilon, Blue Key, and Sigma Xi. I have been a registered Professional Engineer in the field of Nuclear Science for over 25 years. I am, and have been, a member of several Standards Committees in the American Nuclear Society ("ANS"). I have been a member of the ANS Standards Committee on Nuclear Criticality Safety since 1975. I am an Elected Fellow of the American Institute of Chemists. A copy of my resume is included as Attachment A to this affidavit.

6. I have been performing nuclear criticality analyses since 1957. Since 1987, I have been the Chief Nuclear Scientist for Holtec. Prior to that, from 1977 to 1987, I was the Senior Consultant for the Southern Science Office of Black & Veatch Engineers-Architects. Prior to that, from 1973 to 1977, I was a Senior Consultant for NUS Corporation. Prior to that, from 1964 to 1973, I was the Vice President for Physics for Southern Nuclear Engineering, Inc. Prior to that, from 1957 to 1964, I was a Senior Reactor Physicist for General Nuclear Engineering. Every one of these positions has included, among other things responsibility for nuclear criticality safety for reactor core operations as well as for new and spent fuel storage.

7. In my four decades of work on nuclear criticality safety, I have both developed methods for assessing nuclear criticality safety and performed the analyses to demonstrate criticality safety. I have developed nuclear analysis techniques used in criticality safety analyses. I have performed the detailed calculations to benchmark the KENO5a and MCNP4a computer codes that are widely used for criticality safety analyses. I have developed and written computer codes to generate input for nuclear criticality safety analyses. I have also performed numerous nuclear criticality safety

analyses. I have performed numerous calculations of spent fuel fission product inventories using the CASMO2E, CASMO3, CASMO4, ORIGEN, ORIGEN-II, and ORIGEN-S codes. I have performed numerous criticality safety analyses for wet spent fuel storage rack installations, dry cask storage, and transportation casks. I have personally performed criticality safety analyses, and authored the related reports, to support approximately 60 to 70 license amendment requests for spent fuel pool storage.

8. I make this affidavit to explain the physical systems or processes available as criticality control methods for spent fuel storage, and the administrative measures used to implement each method. I also discuss, and provide my understanding of, the NRC's regulations governing criticality control for spent fuel pools, including General Design Criterion 62 (10 C.F.R. Part 50, Appendix A) and 10 C.F.R. § 50.68. I address specific aspects of the NRC Staff's regulatory guidance concerning spent fuel pool criticality control, including the Double Contingency Principle and the implementation of burnup credit. I also provide information concerning the prevalence of the use of burnup credit for spent fuel pool criticality control in numerous sites across the country and overseas. Finally, I provide my review of the nuclear criticality analysis performed by the NRC Staff for this proceeding.

PHYSICAL SYSTEMS OR PROCESSES AVAILABLE FOR CRITICALITY CONTROL IN SPENT FUEL POOLS

9. Every criticality control method involves, by necessity, some physical system or process. Criticality control can only be achieved through physical measures that affect the neutron multiplication factor ("k-effective"). This is achieved through controlling the production, absorption, and leakage of neutrons. All of these are physical

measures. Neutrons will not recognize, much less obey, procedures and other administrative measures alone. Some physical measure is required to achieve criticality control.

10. There are a limited number of means available to control criticality of fuel assemblies stored in spent fuel pools. In practice, the four methods available are: 1) geometric separation; 2) solid neutron absorbers; 3) soluble neutron absorbers; and 4) fuel reactivity. These methods are physical systems or processes which have a physical effect on the neutron multiplication factor, or "k-effective," in the spent fuel pool.

11. Geometric separation is a physical system or process. Geometric separation physically affects neutron coupling between assemblies in storage. Wider spacing of the individual fuel assemblies neutronically decouples the fuel assemblies and thus decreases reactivity of the system. Geometric separation takes the form of steel racks installed in the spent fuel storage pool with fixed locations and fixed separation between the fuel assemblies in storage.

12. Solid neutron absorbers are a physical system or process. Solid neutron absorbers physically affect neutron absorption. Absorption of neutrons in the solid neutron absorbers, also referred to as neutron "poisons," remove neutrons from the system, which eliminates neutrons that could cause fission and thus decreases reactivity of the system. Boron, and specifically the isotope Boron-10, is the standard absorbing element used in solid neutron absorbers. Solid neutron absorbers take the form of fixed panels with solid boron that are installed in the spent fuel storage racks during their manufacture.

13. Soluble neutron absorbers are a physical system or process. Just like solid neutron absorbers, soluble neutron absorbers physically affect neutron absorption. Absorption of neutrons in the soluble neutron absorbers, also referred to as neutron "poisons," remove neutrons from the system, which eliminates neutrons that could cause fission and thus decreases reactivity of the system. Boron, and specifically the isotope Boron-10, is the standard absorbing element used in soluble neutron absorbers. Soluble neutron absorbers take the form of soluble boric acid dissolved in the spent fuel pool water.

14. Fuel reactivity is a physical system or process. Fuel reactivity physically affects the production, absorption, and leakage of neutrons. Fuel reactivity is determined by three factors: 1) fuel assembly structure; 2) initial (or "fresh") fuel enrichment; and 3) fuel depletion (or "burnup"). All three of these factors must be taken into account to determine fuel reactivity.

15. Fuel assembly structure, part of fuel reactivity, is a physical system or process. Fuel assembly structure physically affects the reactivity of the assemblies. The spacing of fuel rods within the fuel assembly structure determines neutron interactions, which physically affect reactivity of the system. The materials in the fuel assembly structure also act as neutron absorbers, which physically affect the reactivity of the system. Fuel-assembly structure takes the form of fuel (usually uranium dioxide) in metal cladding, as well as grid spacers, tie rods, and end fittings.

16. Fresh fuel enrichment, part of fuel reactivity, is a physical system or process. Fresh fuel enrichment physically affects neutron production. Higher fresh fuel

enrichment results in greater production of neutrons, which increases reactivity of the system. Fresh fuel enrichment is usually described in terms of weight percent of the fissile isotope Uranium-235, out of the total Uranium in the fuel, prior to loading into the reactor core and undergoing power operations.

17. Fuel burnup, part of fuel reactivity, is a physical system or process. Like fresh fuel enrichment, fuel burnup physically affects neutron production. In the burnup process, uranium initially loaded in the fresh fuel is converted, through the nuclear fission and absorption processes, into fission product nuclides and transuranic nuclides. Higher fuel burnup inherently results in lower production of neutrons, which decreases reactivity of the system. The fuel burnup process depletes the amount of fissile Uranium-235 in the fuel, while at the same time replacing the Uranium with fission products and transuranics that are, in many cases, strong neutron absorbers. While some fissile Plutonium-239 and Plutonium-241 are generated during fuel burnup, the combined quantity of fissile Uranium and fissile Plutonium decreases with increasing burnup. Fuel burnup, including the depletion of Uranium and thus the decrease in reactivity, is a well understood physical process. Fuel burnup takes into account the actual physical contents of the nuclear "fuel" material, which includes unburned fissile Uranium-235, non-fissile Uranium isotopes, fission products, and transuranics (including fissile Plutonium-239).

EVERY PHYSICAL SYSTEM OR PROCESS FOR CRITICALITY CONTROL IS IMPLEMENTED USING SOME ADMINISTRATIVE CONTROLS

18. Each of the physical systems or processes, identified above as physical measures for criticality control, requires some administrative controls for implementation. I know of no criticality control measure for fuel storage pools that can

be implemented without some degree of administrative control.

19. Spent fuel storage racks used for geometric separation are designed, constructed, and inspected according to procedural controls. The effect of the spent fuel storage racks on criticality is verified using validated computer codes. Administrative controls are used to ensure that the storage racks are constructed to match the approved design. Fabrication quality, including items such as manufacturing tolerances, is assured through the use of quality control inspections required by administrative controls. The storage racks are installed in the spent fuel pool pursuant to administrative controls, such as inspections, to ensure the racks are properly assembled and positioned.

20. Solid neutron absorber panels installed in the storage racks are likewise designed, constructed, and inspected according to procedural controls. The effect of the solid neutron absorber panels on criticality safety in the design phase is verified using computer codes validated under approved QA procedures. Administrative inspections are used to ensure that the proper amount of boron neutron absorber is loaded into each panel, and that the boron is uniformly distributed within the panel. Administrative controls, including fabrication inspections, are used to ensure that the storage racks are constructed to conform to the approved design. The solid neutron absorber panels are installed in the storage racks pursuant to administrative controls, such as inspections, to ensure the panels are properly located.

21. Soluble boron used in the spent fuel pool water is manufactured, added, and inspected according to procedural controls. The effect of the soluble boron neutron absorber on criticality safety is verified using computer codes validated under approved

QA procedures. The soluble boron is initially installed in the spent fuel pool water pursuant to administrative controls, such as tests and inspections, to ensure that the proper amount of soluble boron is installed. Once installed, it is very difficult to effectively dilute the soluble boron. The soluble boron control system is very slow and any operator error would quickly be detected and corrected weeks before dilution reached a significant level. Massive accident conditions postulate flooding the pool with many thousands of gallons of water. Such large quantities of water flowing over the storage pool floor, into and down stairwells, would be readily detectable long before the soluble boron concentration would be reduced to an undesirable level. Following initial installation, administrative controls, such as regular periodic testing, are used to verify that the level of soluble boron remains consistent with the approved design and that any credible dilution accidents would be detected and corrected on a timely basis.

22. Fuel assembly structure is also designed, constructed, and inspected according to procedural controls. The effect of the fuel assembly structure on criticality is verified using validated computer codes. Administrative controls are used to ensure that the fuel assembly structure is constructed to conform to the approved design. Fabrication quality, such as manufacturing tolerances, is assured through the use of quality control inspections according to administrative controls. The loading of the fuel pellets into the fuel assembly structure is monitored and inspected pursuant to administrative controls. Proper fuel assembly design and manufacture are also important to in-core power operation.

23. Fresh fuel enrichment is designed, produced, inspected, and tracked

according to procedural controls. The effect of the fresh fuel enrichment on criticality is verified using validated computer codes. Administrative controls are used to ensure that fresh fuel enrichment is produced to no more than the level permitted in the approved design. Enrichment quality, such as production tolerances, is assured through the use of quality control inspections required by administrative controls. The fresh fuel enrichment in different fuel assemblies is tracked using administrative controls such as material control and accounting (MC&A) procedures and related databases for control of special nuclear material. Administrative controls for MC&A track the movements, location, and fuel characteristics, including fresh fuel enrichment, of all fuel assemblies throughout their entire history at the reactor sites.

24. Fuel burnup is an inherent consequence of power operation in the reactor core. It is designed, produced, monitored, and tracked according to procedural controls. The effect of the fuel burnup on criticality is verified using validated computer codes. Administrative controls are used to ensure that fuel burnup is produced to no less than the level permitted in the approved design with conservative allowances for tolerances. Fuel burnup is verified through the use of in-core reactor power monitors used to measure the rate of fission, and therefore fuel burnup, in the reactor core. These records are developed and retained according to administrative controls. The fuel burnup is used to determine the fuel contents using verified and validated computer codes. The fuel burnup in different fuel assemblies is tracked using the material control and accounting (MC&A) procedures and related databases for control of special nuclear material. Administrative controls for MC&A track the movements, location, and fuel characteristics, including

fuel burnup, of all fuel assemblies throughout their entire history at the reactor sites.

25. While the type, degree, and timing of administrative controls vary for each of the physical systems or processes, it is a fact that every one of these physical measures for criticality control is implemented using some administrative controls.

NRC's Regulations Governing Spent Fuel Pool Criticality Control

General Design Criterion 62

26. The first NRC regulatory requirement governing spent fuel pool criticality control is General Design Criterion 62, "Prevention of criticality in fuel storage and handling" ("GDC 62"). This regulation was added to Appendix A of 10 C.F.R. Part 50 in 1971. A copy of GDC 62 is included as Attachment B to this affidavit. GDC 62 is one of the 64 general design criteria for nuclear power plants in Appendix A to 10 C.F.R. Part 50. GDC 62 reads as follows:

> *Criterion 62 – Prevention of criticality in fuel storage and handling.* Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

27. I have read, and am familiar with, the provisions of GDC 62. I have implemented the provisions of GDC 62 for over 28 years, since it was initially promulgated in 1971. I have also worked with the NRC Staff, during this same time period, to implement GDC 62 in light water spent fuel storage technologies developed to meet the requirements for expanded spent fuel storage since the mid-1970's.

28. GDC 62 requires that all spent fuel pool criticality control measures must be physical systems or processes. As I stated above, the four methods available in

practice for criticality control in spent fuel pool storage - - 1) geometric separation; 2) solid neutron absorbers; 3) soluble neutron absorbers; and 4) fuel reactivity, including 4.1) fuel assembly structure, 4.2) fresh fuel enrichment, and 4.3) fuel burnup - - are all physical systems or processes.

29. Also as I stated above, every one of these physical measures for criticality control requires some type of administrative controls to implement. In my 28 years of experience with GDC 62, I have always understood GDC 62 to encompass criticality control by physical measures that are implemented with the use of some administrative controls. As a practical matter, there can be no other way to interpret GDC 62. An interpretation that GDC 62 prohibits administrative measures used to implement the physical systems or processes would render GDC 62 a nullity, because none of the available criticality control methods could comply with such an interpretation. If this were the interpretation, GDC 62 would prohibit any method of criticality control.

30. The four different physical measures available for spent fuel pool criticality control do require different types, degrees, and timing of administrative controls for implementation. For example, the administrative controls required to implement geometric separation and solid neutron absorbers all occur before the storage racks are initially loaded with fuel, while the administrative controls attendant to soluble neutron absorbers and fuel reactivity occur both before the racks are initially loaded as well as after. However, this is a difference only in timing and duration of the administrative measures. Nothing in GDC 62 differentiates between physical systems or processes for criticality control based on the timing and duration of the administrative

measures required to implement the physical measures.

31. Specifically, fuel enrichment limits and fuel burnup limits are physical systems or processes consistent with the requirements of GDC 62. These two measures are aspects of fuel reactivity, which is clearly a physical measure.

10 C.F.R. § 50.68

32. The other NRC regulatory requirement governing spent fuel pool criticality control is 10 C.F.R. § 50.68, "Criticality Accident Requirements." This regulation supplements GDC 62 and defines the accident condition that is not specifically addressed in GDC 62. 10 C.F.R. § 50.68 was added to Part 50 in 1998 (a copy is included as Attachment C). 10 C.F.R. § 50.68 requires that the storage pool be evaluated for the accident condition which assumes the loss of all soluble boron. Though 10 C.F.R. § 50.68 does not address every postulated accident, it does address the most serious accident (loss of soluble boron), without describing conditions that might cause such an accident. The requirement is relevant to this proceeding, and requires that the storage racks remain subcritical should all soluble boron be lost.

33. 10 C.F.R. § 50.68 acknowledges and permits partial credit for soluble boron as a criticality control method for fuel stored in pools. 10 C.F.R. § 50.68(b)(4) specifically permits partial credit for soluble boron to establish an acceptable safety margin below criticality. Thus, 10 C.F.R. § 50.68 confirms the use of soluble boron as a criticality control method for spent fuel storage racks. The use of soluble boron for criticality control is just like the use of fuel burnup limits for criticality control. Both are physical measures that are implemented through administrative controls that apply prior

to initial use of the storage racks, and continue to apply during spent fuel pool operations.

34. 10 C.F.R. § 50.68 supplements and provides a practical interpretation of GDC 62 with regard to accident conditions. 10 C.F.R. § 50.68 has the effect of endorsing the single failure criterion (defined as loss of soluble boron) and does not require the evaluation of other unlikely, independent, and concurrent accidents (Double Contingency Principle).

35. 10 C.F.R. § 50.68 implicitly acknowledges and permits the use of limits on spent fuel assembly reactivity as a criticality control method for fuel stored in pools. 10 C.F.R. § 50.68(b)(4) specifically directs that "spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity" be considered for criticality control purposes. As discussed above, spent fuel assembly reactivity includes the effects of fuel burnup (as well as fuel structure and initial fuel enrichment). 10 C.F.R. § 50.68(b)(4) does not restrict the assessment of fuel reactivity to only fresh fuel enrichment.

36. 10 C.F.R. § 50.68 acknowledges and permits the use of fresh fuel enrichment limits as a criticality control method for fuel storage in pools. 10 C.F.R. § 50.68(b)(7) specifically permits the use of a limit on fresh fuel reactivity which includes burnup and enrichment as a criticality control method for fuel storage.

37. 10 C.F.R. § 50.68(b) acknowledges and permits the use of administrative controls, including plant procedures, to implement criticality control methods for fuel stored in pools. 10 C.F.R. § 50.68(b)(1) specifically endorses the use of plant procedures to implement geometric separation of fuel assemblies. 10 C.F.R. § 50.68(b)(4) specifically permits the use of soluble boron for criticality control, which requires

administrative controls to implement. 10 C.F.R. § 50.68(b)(4) specifically permits spent fuel assembly reactivity to be used in criticality control. Fuel reactivity includes the effects of fuel burnup, which requires administrative controls to implement. 10 C.F.R. § 50.68(b)(7) specifically permits the use of enrichment limits for criticality control, which requires administrative controls to implement.

NRC STAFF'S REGULATORY GUIDANCE CONCERNING CRITICALITY CONTROL

DOUBLE CONTINGENCY PRINCIPLE

38. The NRC Staff's regulatory guidance for implementing criticality control methods specifically endorse the Double Contingency Principle. The Double Contingency Principle (sometimes called the Single Failure Criterion) was originally issued in the ANSI Standard ANSI N 16.1-1975. It was later endorsed by the NRC Staff in the Staff's 1978 guidance letter to all power reactor operators, in Reg. Guide 1.13, and in the Staff's 1998 guidance memorandum (as discussed below).

39. The Double Contingency Principle is defined in Section 1.4 of Appendix A to Draft Revision 2 to Regulatory Guide 1.13, issued in 1981 ("Reg. Guide 1.13," included as Attachment D to this affidavit). While Reg. Guide 1.13 was never formally issued in final form, its provisions concerning criticality control, and specifically credit for burnup, have been implemented in the nuclear industry and by the Staff over the past 18 years in approving spent fuel storage rack license amendment requests for dozens of nuclear power plants across the country (I discuss these later in this affidavit). In this sense, though not formally issued in final form, the Staff's actions using Reg. Guide 1.13 as a basis in approving license amendments made it, through practice, final regulatory

guidance.

40. Reg. Guide 1.13, Appendix A, Section 1.4 defines the Double Contingency Principle as follows:

> At all locations in the LWR spent fuel storage facility where spent fuel is handled or stored, the nuclear criticality safety analysis should demonstrate that criticality could not occur without at least two unlikely, independent, and concurrent failures or operating limit violations.

The Double Contingency Principle, as defined in Reg. Guide 1.13, is a Staff term established in Staff guidance. It's definition can be determined through a review of Staff statements regarding the term and Staff actions implementing it. One significance of the Double Contingency Principle in the present proceeding is that, where the loss of soluble boron is evaluated as the principal accident condition (as specified in 10 C.F.R. § 50.68), it is not necessary to consider the simultaneous occurrence of other unlikely and independent accidents.

41. The Double Contingency Principle is also stated in other relevant NRC Staff guidance documents. The Double Contingency Principle was first formally adopted by the Staff in the 1978 generic letter from Brian K. Grimes of the Staff's Division of Operating Reactors to all power reactor licenses ("1978 Fuel Storage Guidance," included as Attachment E to this affidavit). In Section 1.2 of the 1978 Fuel Storage Guidance, the Staff adopts the Double Contingency Principle by reference to an industry ANSI standard, stating:

> The double contingency principle of ANSI N 16.1-1975 shall be applied. It shall require two unlikely, independent, concurrent events to produce a criticality accident.

The ANSI standard, ANSI N 16.1-1975, referenced by the NRC Staff provides the original definition of the Double Contingency Principle. A copy of ANSI N 16.1-1975 is included as Attachment F to this affidavit. Section 4.2.2 of ANSI N 16.1-1975 defines the Double Contingency Principle as follows:

> Double Contingency Principle. Process designs should, in general, incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible.

The definition of Double Contingency Principle in Section 4.2.2 remained unchanged when ANSI N 16.1-1975 was revised into ANSI/ANS-8.1-1983 in 1983. A copy of ANSI/ANS-8.1-1983 is included as Attachment G to this affidavit.

The Staff provided further elucidation of its Double Contingency Principle in the Staff guidance on fuel storage criticality control issued in the 1998 memorandum from Laurence I. Kopp of the Staff's Reactor Systems Branch ("1998 Criticality Guidance," included as Attachment H to this affidavit). The 1998 Criticality Guidance has been approved by the Staff and made available to all licensees as guidance on implementing criticality control for fuel storage. Section 3 of the 1998 Criticality Guidance defines the Double Contingency Principle as follows:

> ABNORMAL CONDITIONS AND THE DOUBLE-CONTINGENCY PRINCIPLE
>
> The criticality safety analysis should consider all credible incidents and postulated accidents. However, by virtue of the double-contingency principle, two unlikely independent and concurrent incidents or postulated accidents are beyond the scope of the required analysis. The double-contingency principle means that a realistic condition may be assumed for the criticality analysis in calculating the effects of incidents or postulated accidents. For example, if soluble boron is normally present in the spent fuel pool water, the

> loss of soluble boron is considered as one accident condition and a second concurrent accident need not be assumed. Therefore, credit for the presence of the soluble boron may be assumed in evaluating other accident conditions.

The 1998 Criticality Guidance is the Staff's most recent, and most thorough, statement of the definition of the Staff's Double Contingency Principle.

42. I have been employing the Double Contingency Principle in performing criticality analyses for spent fuel storage racks for over 20 years. I have implemented the Double Contingency Principle for dozens of license applications since it was first developed. I have always understood the Double Contingency Principle to have the same meaning regardless of the document in which it appears. While the wording used in each of the documents above is slightly different, the meaning of the Double Contingency Principle in each is the same. The most recent Staff guidance on this issue, the 1998 Criticality Guidance, is the most simple, unambiguous, and easy to understand explanation of the Double Contingency Principle. It's meaning, however, is the same as that in the prior Staff guidance documents and in ANSI N 16.1-1975.

43. In all cases, the Double Contingency Principle is implemented by evaluating criticality for the expected, realistic conditions in the spent fuel storage pool, plus one unlikely, independent incident or postulated accident. The plethora of unlikely, independent accidents are not required to be analyzed concurrently. Instead, accident conditions are analyzed one at a time to develop a series of criticality results, one for each separate credible unlikely, independent accident condition. Under the Double Contingency Principle, an evaluation assuming two or more unlikely, independent, and

concurrent postulated accidents is beyond the scope of the required analysis. Only credible incidents or postulated accidents are required to be considered.

44. I have been involved with dozens of licensing applications involving the Double Contingency Principle before the NRC Staff. In my experience, the Staff has always interpreted the Double Contingency Principle this way.

45. There is no requirement in the Double Contingency Principle for applicants to demonstrate that criticality will occur with two or more unlikely, independent and concurrent incidents or accident conditions. The purpose of the Commission's criticality control regulations is to prevent criticality from occurring. It would be contrary to the Commission's purpose, and would serve no useful regulatory purpose, to define and evaluate the universe of possible scenarios of multiple concurrent accident conditions in which criticality might occur. The Double Contingency Principle clearly does not require this to be done.

46. In this case, the universe of scenarios to be evaluated under the Double Contingency Principle is the set of unlikely, but credible, independent incidents or postulated accidents that could have an adverse effect on criticality control. Of the four physical measures used for criticality control at Harris, two - - loss of the storage racks and loss of the solid neutron absorbers in the storage racks - - are not credible and need not be analyzed. The loss of control over fuel reactivity, including fuel enrichment and fuel burnup limits, through misplacement of a fuel assembly is highly unlikely, but hypothetically possible. The loss of soluble boron is so unlikely that it is probably not credible (particularly a total loss of soluble boron), but can be analyzed for completeness.

Thus, in addition to the expected conditions, two scenarios should be evaluated for Harris spent fuel pools C and D under the Double Contingency Principle: 1) expected conditions with misplacement of a fuel assembly; and 2) expected conditions with loss of soluble boron. Both of these scenarios have been analyzed by the Applicant, and both have been demonstrated to be subcritical within the regulatory limits.

47. My understanding of the Double Contingency Principle is based on over 20 years of actual experience implementing the Double Contingency Principle in NRC licensing actions, and working with the NRC Staff in implementing criticality safety and employing the Double Contingency Principle.

48. The Applicant's criticality control analysis in this case, with the addition of a supplemental analysis of two independent and concurrent accidents (the fuel assembly misplacement analysis, included as Attachment B to Exhibit 3, the Affidavit of Everett L. Redmond II, Ph.D.), confirms that, even for multiple accident conditions, the storage racks remain subcritical.

IMPLEMENTATION OF LIMITS ON FUEL BURNUP

49. The NRC Staff's guidance governing spent fuel pool criticality control permits the use of fuel burnup limits as a method for criticality control, and outlines the administrative measures required to implement fuel burnup limits. Fuel burnup was initially addressed by Staff regulatory guidance in the Reg. Guide 1.13 (Rev. 2), which is included as Attachment D to this affidavit. Appendix A of Reg. Guide 1.13 provides instructions on how to implement credit for burnup as a method for criticality control. Specifically, sections 4 and 6 address the administrative measures used to implement and

verify fuel burnup limits as a criticality control method.

50. The NRC Staff issued a new guidance memorandum in 1998 on criticality control for fuel storage that effectively replaces Reg. Guide 1.13 ("1998 Criticality Guidance," included as Attachment H to this affidavit). Like Reg. Guide 1.13, the 1998 Criticality Guidance permits the use of fuel burnup limits as a method for criticality control, and outlines the administrative measures required to implement fuel burnup limits. Sections 1, 2, and specifically 5.A.5 address the administrative measures used to implement fuel burnup limits as a criticality control method.

PREVALENCE OF BURNUP CREDIT

51. The use of burnup credit as a criticality control method for spent fuel pool storage is prevalent throughout the nuclear industry in this country and abroad. License amendments using burnup credit for spent fuel storage were approved beginning in the early 1980's. The need for burnup credit as a method for criticality control has become even more acute following the Department of Energy's failure to meet its obligation to begin accepting spent nuclear fuel beginning in 1998. I am aware of at least 20 nuclear power plants that currently use burnup credit as a criticality control method for their spent fuel pool storage. The following list identifies these 20 plants where burnup credit is used, along with the approximate year of license approval:

	Plant	Year
1.	V.C. Summer	1983
2.	Braidwood	1983
3.	Diablo Canyon	1986
4.	St. Lucie 1	1987
5.	Byron	1987

6.	Indian Point 2	1989
7.	San Onofre	1989
8.	TMI 1	1991
9.	D.C. Cook	1991
10.	Zion	1991
11.	Maine Yankee	1992
12.	Sequoyah	1993
13.	Fort Calhoun	1993
14.	ANO 1 & 2	1994
15.	Salem	1994
16.	Beaver Valley	1994
17.	Comanche Peak	1994
18.	Haddam Neck	1996
19.	Vogtle	1998
20.	Waterford	1998

NRC STAFF'S CRITICALITY ANALYSIS

52. In November, 1999, the NRC Staff performed for this proceeding an independent nuclear criticality analysis of multiple accidents involving fuel assembly misplacements. The Staff's criticality analysis was performed by Tony P. Ulses, a nuclear engineer in the NRC Staff's Reactor Systems Branch. This analysis is documented in the NRC Staff's November 5, 1999 memorandum and report, which is included as Attachment C to Exhibit 3, the Affidavit of Everett L. Redmond II, Ph.D. The Staff's analysis assumes the concurrent misplacement of an infinite number of fresh fuel assemblies of the maximum permissible reactivity. The Staff's analysis utilized boundary conditions that are reflective in the x, y, and z directions, which models and infinite array of fresh fuel assemblies. The analysis includes the effects of the soluble boron required to be present in the spent fuel pools pursuant to plant operating procedures. This analysis is not required under the Double Contingency Principle in the

Staff's regulatory guidance, since even <u>two</u> fresh fuel assembly misplacements are two independent, unlikely, concurrent events. The NRC Staff's analysis of an <u>infinite</u> number of concurrently misplaced fresh fuel assemblies of the maximum possible reactivity is far beyond what is considered a credible event for analysis purposes.

53. I have reviewed the NRC Staff's November 5, 1999 memorandum and report on its misplacement criticality analysis. I am familiar with the analysis methodology, assumptions, and computer codes used in the Staff's analysis. Based on my review, I have determined that the Staff modeled the most reactive fresh fuel assemblies permissible at Harris and the spent fuel storage racks to be used for those assemblies in Harris spent fuel pools C and D. The Staff's analysis concluded that the spent fuel storage racks will remain subcritical, with a calculated k-effective of 0.98. The Staff's analysis assumed that the k-effective bias from manufacturing tolerances is not larger than 1%. I am familiar with the manufacturing tolerances applicable to these spent fuel storage racks, and I confirm that the bias from these manufacturing tolerances is less than 1%.

54. I have performed an analysis similar to the Staff's analysis, using computer codes that I would normally use in storage rack design and analysis. My result is in complete agreement with that obtained in the Staff's analysis. Based on my independent analysis, my familiarity with the Staff's analysis, and my four decades of experience performing nuclear criticality analyses, I confirm the results of the nuclear criticality analysis performed by the NRC Staff. The results of the analysis are consistent with my expectations based on my knowledge of the spent fuel storage rack designs,

fresh fuel assembly characteristics, analytical methods, and calculations.

55. The NRC Staff's analysis (and my own confirming calculation) demonstrates that the spent fuel storage racks for Harris spent fuel pools C and D will remain subcritical, even if every location in the spent fuel storage rack is assumed to be concurrently loaded with a misplaced fresh fuel assembly of the maximum possible reactivity at the Harris Nuclear Plant. While this analysis is not required under the Staff's Double Contingency Principle, the NRC Staff's criticality analysis of an infinite number of fresh fuel assembly misplacements demonstrates that the issue of multiple fuel assembly misplacements is moot with respect to the spent fuel storage racks in Harris pools C and D.

I declare under penalty of perjury that the foregoing statements and my statements in the attached report are true and correct.

Stanley E. Turner
January 3, 2000

State of Florida
County of Pinellas

Subscribed and sworn to before me this

3rd day of January, 2000

FLDL T656-785-26-323-0
exp. 9/3/04

Vilma T. Brower
Vilma T. Brower

My commission expires

VILMA T. BROWER
MY COMMISSION # CC 589563
EXPIRES: October 7, 2000
Bonded Thru Notary Public Underwriters

STANLEY E. TURNER, Ph.D., P.E.

SENIOR VICE PRESIDENT AND CHIEF NUCLEAR SCIENTIST
HOLTEC INTERNATIONAL

EDUCATION

University of Texas
Ph.D. in Nuclear Chemistry (1951)

University of South Carolina
B.S. in Chemistry (1945)

Georgia Institute of Technology (1943-44) (1946-47)

PROFESSIONAL EXPERIENCE

HOLTEC INTERNATIONAL
Palm Harbor, Florida

1987-1997	Chief Nuclear Scientist
1997-Present	Senior Vice President and Chief Nuclear Scientist

SOUTHERN SCIENCE OFFICE OF BLACK & VEATCH ENGINEERS - ARCHITECTS
Dunedin, Florida

1977-1987	Project Manager/Senior Consultant

NUS CORPORATION
Dunedin, Florida

1973-1977	Senior Consultant

SOUTHERN NUCLEAR ENGINEERING, INC.
Dunedin, Florida

1964-1973	Vice President, Physics

GENERAL NUCLEAR ENGINEERING
Dunedin, Florida

1957-1964	Senior Reactor Physicist/Project Manager

SOCONY-MOBIL RESEARCH LABORATORY
Dallas, Texas

1952-1957	Research Scientist

U.S. NAVY RADIOLOGICAL DEFENSE LABORATORY
San Francisco, California

1951-1952	Physicist

PROFESSIONAL CERTIFICATIONS

Registered Professional Engineer (Nuclear)– Florida (1974-Present)

PROFESSIONAL SOCIETY MEMBERSHIPS/ACTIVITIES

Elected Fellow, American Institute of Chemists
Member, ANS Standards Committee 8.17 on Nuclear Criticality Safety (1975-Present)
Chairman of ANS 5.3 (Failed Fuel Consequences (1981-1995)) and 5.4 (Fission Product Release (1978-Present))
Formerly a member of the ANS 5 Committee with oversight on ANS 5.1, Decay Heat.

ACADEMIC HONORS

Sigma Pi Sigma, Phi Lambda Epsilon,
Blue Key, Sigma Xi

CONTINUING EDUCATION COURSES OFFERED TO PRACTICING GRADUATE ENGINEERS

1. Union Electric Company, St. Louis, Missouri: Use of CASMO and KENO Codes in criticality safety analysis.

2. Southern California Edison Company, San Clemente, California: Use of CASMO and KENO Codes in criticality safety analysis.

DRY AND WET SPENT FUEL STORAGE TECHNOLOGY

- Developed nuclear analysis techniques for criticality safety analyses.
- Performed criticality safety analyses for numerous wet spent fuel storage rack installations.
- Performed criticality analyses of numerous fuel designs under normal and accident conditions for the HI-STAR 100 shipping cask and HI-STORM storage cask.
- Performed detailed benchmark calculations for KENO5a and MCNP4a computer codes.
- Developed and wrote CELLDAN Computer Code to prepare input for NITAWL-KENO5a calculations.
- Supervised calculations with the QAD Point Kernal Code for gamma ray shielding.
- Performed numerous calculations of fission product inventories using ORIGEN, ORIGEN-II, and ORIGEN-S (ORIGEN-ARP) Codes.
- Participated in the development of Holtec's thermal evaluation methodologies for wet storage systems.
- Author of numerous reports on dry and wet storage facilities.
- Designed equipment for and supervised Blackness Testing at numerous power plants and performed measurements on Boraflex and Boral surveillance coupons.
- Performed R&D programs on Holtite-A neutron absorber materials and on HI-COAT coatings.
- Performed wet chemical analyses of Boral samples.

containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

Criterion 62—Prevention of criticality in fuel storage and handling. Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Criterion 63—Monitoring fuel and waste storage. Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

Criterion 64—Monitoring radioactivity releases. Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

[36 FR 3256, Feb. 20, 1971, as amended at 36 FR 12733, July 7, 1971; 41 FR 6258, Feb. 12, 1976; 43 FR 50163, Oct. 27, 1978; 51 FR 12505, Apr. 11, 1986; 52 FR 41294, Oct. 27, 1987]

APPENDIX B TO PART 50—QUALITY ASSURANCE CRITERIA FOR NUCLEAR POWER PLANTS AND FUEL REPROCESSING PLANTS

Introduction. Every applicant for a construction permit is required by the provisions of §50.34 to include in its preliminary safety analysis report a description of the quality assurance program to be applied to the design, fabrication, construction, and testing of the structures, systems, and components of the facility. Every applicant for an operating license is required to include, in its final safety analysis report, information pertaining to the managerial and administrative controls to be used to assure safe operation. Nuclear power plants and fuel reprocessing plants include structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. This appendix establishes quality assurance requirements for the design, construction, and operation of those structures, systems, and components. The pertinent requirements of this appendix apply to all activities affecting the safety-related functions of those structures, systems, and components; these activities include designing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying.

As used in this appendix, "quality assurance" comprises all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service. Quality assurance includes quality control, which comprises those quality assurance actions related to the physical characteristics of a material, structure, component, or system which provide a means to control the quality of the material, structure, component, or system to predetermined requirements.

I. ORGANIZATION

The applicant[1] shall be responsible for the establishment and execution of the quality assurance program. The applicant may delegate to others, such as contractors, agents, or consultants, the work of establishing and executing the quality assurance program, or any part thereof, but shall retain responsibility therefor. The authority and duties of persons and organizations performing activities affecting the safety-related functions of structures, systems, and components shall be clearly established and delineated in writing. These activities include both the performing functions of attaining quality objectives and the quality assurance functions. The quality assurance functions are those of (a) assuring that an appropriate quality assurance program is established and effectively executed and (b) verifying, such as by checking, auditing, and inspection, that activities affecting the safety-related functions have been correctly performed. The persons and organizations performing quality assurance functions shall have sufficient authority and organizational freedom to identify quality problems; to initiate, recommend, or provide solutions; and to verify implementation of solutions. Such persons and organizations performing quality assurance functions shall report to a management level such that this required authority and organizational freedom, including sufficient independence from cost and schedule when opposed to safety considerations, are provided. Because of the many variables involved, such as the number of

[1] While the term "applicant" is used in these criteria, the requirements are, of course, applicable after such a person has received a license to construct and operate a nuclear power plant or a fuel reprocessing plant. These criteria will also be used for guidance in evaluating the adequacy of quality assurance programs in use by holders of construction permits and operating licenses.

temperature conditions as those given the beltline material.

(2) Alternatively, the percent recovery due to thermal annealing (R_t and R_u) may be determined from the results of a verification test program employing materials removed from the beltline region of the reactor vessel[6] and that have been annealed under the same time and temperature conditions as those given the beltline material.

(3) Generic computational methods may be used to determine recovery if adequate justification is provided.

(f) *Public information and participation.* (1) Upon receipt of a Thermal Annealing Report, and a minimum of 30 days before the licensee starts thermal annealing, the Commission shall:

(i) Notify and solicit comments from local and State governments in the vicinity of the site where the thermal annealing will take place and any Indian Nation or other indigenous people that have treaty or statutory rights that could be affected by the thermal annealing,

(ii) Publish a notice of a public meeting in the FEDERAL REGISTER and in a forum, such as local newspapers, which is readily accessible to individuals in the vicinity of the site, to solicit comments from the public, and

(iii) Hold a public meeting on the licensee's Thermal Annealing Report.

(2) Within 15 days after the NRC's receipt of the licensee submissions required by paragraphs (c)(1), (c)(2) and (c)(3)(i)–(iii) of this section, the NRC staff shall place in the NRC Public Document Room a summary of its inspection of the licensee's thermal annealing, and the Commission shall hold a public meeting:

(i) For the licensee to explain to NRC and the public the results of the reactor pressure vessel annealing,

(ii) for the NRC to discuss its inspection of the reactor vessel annealing, and

(iii) for the NRC to receive public comments on the annealing.

[6] For those cases where materials are removed from the beltline of the pressure vessel, the stress limits of the applicable portions of the ASME Code Section III must be satisfied, including consideration of fatigue and corrosion, regardless of the Code of record for the vessel design.

(3) Within 45 days of NRC's receipt of the licensee submissions required by paragraphs (c)(1), (c)(2) and (c)(3) (i)–(iii) of this section, the NRC staff shall complete full documentation of its inspection of the licensee's annealing process and place this documentation in the NRC Public Document Room.

[60 FR 65472, Dec. 19, 1995]

§ 50.68 Criticality accident requirements.

(a) Each holder of a construction permit or operating license for a nuclear power reactor issued under this part or a combined license for a nuclear power reactor issued under part 52 of this chapter, shall comply with either 10 CFR 70.24 of this chapter or the requirements in paragraph (b) of this section.

(b) Each licensee shall comply with the following requirements in lieu of maintaining a monitoring system capable of detecting a criticality as described in 10 CFR 70.24:

(1) Plant procedures shall prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water.

(2) The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used.

(3) If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly reactivity and filled with low-density hydrogenous fluid, the k-effective corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent

such moderation or if fresh fuel storage racks are not used.

(4) If no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

(5) The quantity of SNM, other than nuclear fuel stored onsite, is less than the quantity necessary for a critical mass.

(6) Radiation monitors are provided in storage and associated handling areas when fuel is present to detect excessive radiation levels and to initiate appropriate safety actions.

(7) The maximum nominal U-235 enrichment of the fresh fuel assemblies is limited to five (5.0) percent by weight.

(8) The FSAR is amended no later than the next update which § 50.71(e) of this part requires, indicating that the licensee has chosen to comply with § 50.68(b).

[63 FR 63130, Nov. 12, 1998]

Inspections, Records, Reports, Notifications

§ 50.70 Inspections.

(a) Each licensee and each holder of a construction permit shall permit inspection, by duly authorized representatives of the Commission, of his records, premises, activities, and of licensed materials in possession or use, related to the license or construction permit as may be necessary to effectuate the purposes of the Act, including section 105 of the Act.

(b)(1) Each licensee and each holder of a construction permit shall upon request by the Director, Office of Nuclear Reactor Regulation, provide rent-free office space for the exclusive use of the Commission inspection personnel. Heat, air conditioning, light, electrical outlets and janitorial services shall be furnished by each licensee and each holder of a construction permit. The office shall be convenient to and have full access to the facility and shall provide the inspector both visual and acoustic privacy.

(2) For a site with a single power reactor or fuel facility licensed pursuant to part 50, the space provided shall be adequate to accommodate a full-time inspector, a part-time secretary and transient NRC personnel and will be generally commensurate with other office facilities at the site. A space of 250 square feet either within the site's office complex or in an office trailer or other on site space is suggested as a guide. For sites containing multiple power reactor units or fuel facilities, additional space may be requested to accommodate additional full-time inspector(s). The office space that is provided shall be subject to the approval of the Director, Office of Nuclear Reactor Regulation. All furniture, supplies and communication equipment will be furnished by the Commission.

(3) The licensee or construction permit holder shall afford any NRC resident inspector assigned to that site, or other NRC inspectors identified by the Regional Administrator as likely to inspect the facility, immediate unfettered access, equivalent to access provided regular plant employees, following proper identification and compliance with applicable access control meaures for security, radiological protection and personal safety.

(4) The licensee or construction permit holder (nuclear power reactor only) shall ensure that the arrival and presence of an NRC inspector, who has been properly authorized facility access as described in paragraph (b)(3) of this section, is not announced or otherwise communicated by its employees or contractors to other persons at the facility unless specifically requested by the NRC inspector.

[21 FR 355, Jan. 19, 1956; 44 FR 47919, Aug. 16, 1979, as amended at 52 FR 31612, Aug. 21, 1987; 53 FR 42942, Oct. 25, 1988]

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REGULATORY RESEARCH

DRAFT REGULATORY GUIDE AND VALUE/IMPACT STATEMENT

December 1981
Division 1
Task CE 913-5

Contact: C. Schulten (301)443-5910

PROPOSED REVISION 2* TO REGULATORY GUIDE 1.13

SPENT FUEL STORAGE FACILITY DESIGN BASIS

A. INTRODUCTION

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, "Domestic Licensing of Production and Utilization Facilities," requires that fuel storage and handling systems be designed to ensure adequate safety under normal and postulated accident conditions. It also requires that these systems be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions. This guide describes a method acceptable to the NRC staff for implementing Criterion 61.

B. DISCUSSION

Working Group ANS-57.2 of the American Nuclear Society Subcommittee ANS-50 has developed a standard that details minimum design requirements for

*The substantial number of changes in this proposed revision has made it impractical to indicate the changes with lines in the margin.

spent fuel storage facilities at nuclear power stations. This standard was approved by the American National Standards Committee N18, Nuclear Design Criteria. It was subsequently approved and designated ANSI N210-1976/ANS-57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations," by the American National Standards Institute on April 12, 1976.

Primary facility design objectives are:

a. To prevent loss of water from the fuel pool that would uncover fuel,

b. To protect the spent fuel from mechanical damage, and

c. To provide the capability for limiting the potential offsite exposures in the event of significant release of radioactivity from the fuel.

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

1. LOSS OF WATER FROM STORAGE POOL

Unless protective measures are taken to prevent the loss of water from a fuel storage pool, the spent fuel could overheat and cause damage to fuel cladding integrity, which could result in the release of radioactive materials to the environment. Equipment failures in systems connected to the pool could also result in the loss of pool water. A permanent coolant makeup system designed with suitable redundancy or backup would prevent the fuel from being uncovered should pool leaks occur. Further, early detection of pool leakage and fuel damage can be made using pool-water-level monitors and pool radiation monitors that alarm locally and also at a continuously manned location to ensure timely operation of building filtration systems. Natural events such as earthquakes or high winds can damage the fuel pool either directly or by the generation of missiles. Earthquakes or high winds could also cause structures or cranes to fall into the pool. Designing the facility to withstand these occurrences without significant loss of watertight integrity will alleviate these concerns.

2. MECHANICAL DAMAGE TO FUEL

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

Plant arrangements consider low-probability accidents such as the dropping of heavy loads (e.g., a 100-ton fuel cask) where such loads are positioned or moved in or over the spent fuel pool. It is desirable that cranes capable of carrying heavy loads be prevented from moving into the vicinity of the stored fuel.

Missiles generated by high winds also are a potential cause of mechanical damage to fuel. This concern can be eliminated by designing the fuel storage facility to preclude the possibility of the fuel being struck by missiles generated by high winds.

3. LIMITING POTENTIAL OFFSITE EXPOSURES

Mechanical damage to the fuel might cause significant offsite doses unless dose reduction features are provided. Dose reduction designs such as negative pressure in the fuel handling building during movement of spent fuel would prevent exfiltration and ensure that any activity released to the fuel handling building will be treated by an engineered safety feature (ESF) grade filtration system before release to the environment. Even if measures not described are used to maintain the desired negative pressure, small leaks from the building may still occur as a result of structural failure or other unforeseen events.

The staff considers Seismic Category I design assumptions acceptable for the spent fuel pool cooling, makeup, and cleanup systems. Tornado protection requirements are acceptable for the water makeup source and its delivery system, the pool structure, the building housing the pool, and the filtration-ventilation system. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," and Regulatory Guide 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear

Power Plants," provide guidelines to limit potential offsite exposures through the filtration-ventilation system of the pool building.

Occupational radiation exposure is kept as low as is reasonably achievable (ALARA) in all activities involving personnel, and efforts toward maintaining exposures ALARA are considered in the design, construction, and operational phases. Guidance on maintaining exposures ALARA is provided in Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable."

C. REGULATORY POSITION

The requirements in ANSI N210-1976/ANS-57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations,"* are generally acceptable to the NRC staff as a means 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 (LWRs), subject to the following clarifications and modifications:

1. In lieu of the example inventory in Section 4.2.4.3(1), the example inventory should be that inventory of radioactive materials that are predicted to leak under the postulated maximum damage conditions resulting from the dropping of a single spent fuel assembly onto a fully loaded spent fuel pool storage rack. Other assumptions in the analysis should be consistent with those given in Regulatory Guide 1.25 (Safety Guide 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. In addition to meeting the requirements of Section 5.1.3, boiling of the pool water may be permitted only when the resulting thermal loads are properly accounted for in the design of the pool structure, the storage racks, and other safety-related structures, equipment, and systems.

*Copies may be obtained from the American Nuclear Society, 555 North Kensington Avenue, La Grange Park, Illinois 60525

3. In addition to meeting the requirements of Section 5.1.3, the fuel storage pool should be designed (a) to prevent tornado winds and missiles generated by these winds from causing significant loss of watertight integrity of the fuel storage pool and (b) to prevent missiles generated by tornado winds from striking the fuel. These requirements are discussed in Regulatory Guide 1.117, "Tornado Design Classification." The fuel storage building, including walls and roof, should be designed to prevent penetration by tornado-generated missiles or from seismic damage to ensure that nothing bypasses the ESF-grade filtration system in the containment building.

4. In addition to meeting the requirements of Section 5.1.5.1, provisions should be made to ensure that nonfuel components in fuel pools are handled below the minimum water shielding depth. A system should be provided that, either through the design of the system or through administrative procedures, would prohibit unknowing retrieval of these components.

5. In addition to meeting the requirements of Section 5.1.12.10, the maximum potential kinetic energy capable of being developed by any object handled above stored spent fuel, if dropped, should not exceed the kinetic energy of one fuel assembly and its associated handling tool when dropped from the height at which it is normally handled above the spent fuel pool storage racks.

6. In addition to meeting the requirements of Section 5.2.3.1, an interface should be provided between the cask venting system and the building ventilation system to minimize personnel exposure to the "vent-gas" generated from filling a dry loaded cask with water.

7. In addition to meeting the requirements of Section 5.3.3, radioactivity released during a Condition IV fuel handling accident should be either contained or removed by filtration so that the dose to an individual is less than the guidelines of 10 CFR Part 100. The calculated offsite dose to an individual from such an event should be well within the exposure guidelines of 10 CFR Part 100 using appropriately conservative analytical methods and assumptions. In order to ensure that released activity does not bypass the

filtration system, the ESF fuel storage building ventilation should provide and maintain a negative pressure of at least 3.2 mm (0.125 in.) water gauge within the fuel storage building.

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

9. In addition to the references listed in Section 6.4.4, Safety Class 3, Seismic Category I, and safety-related structures and equipment should be subjected to quality assurance programs that meet the applicable provisions of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. Further, these programs should obtain guidance from Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)," endorsing ANSI N45.2, and from the applicable provisions of the ANSI N45.2-series standards endorsed by the following regulatory guides:

1.30 (Safety Guide 30) "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment" (N45.2.4).

1.38 "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants" (N45.2.2).

1.58 "Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel" (N45.2.6).

1.64 "Quality Assurance Requirements for the Design of Nuclear Power Plants" (N45.2.11).

1.74 "Quality Assurance Terms and Definitions" (N45.2.10).

1.88 "Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records" (N45.2.9).

1.94 "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants" (N45.2.5).

1.116 "Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems" (N45.2.8).

1.123 "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants" (N45.2.13).

10. The spent fuel pool water temperatures stated in Section 6.6.1(2) exceed the limits recommended by the NRC staff. For the maximum heat load during Condition I occurrences with normal cooling systems in operation and assuming a single active failure, the pool water temperature should be kept at or below 60°C (140°F). Under abnormal maximum heat load conditions (full core unload) and also for Condition IV occurrences, the pool water temperature should be kept below boiling.

11. A nuclear criticality safety analysis should be performed in accordance with Appendix A to this guide for each system that involves the handling, transfer, or storage of spent fuel assemblies at LWR spent fuel storage facilities.

12. The spent fuel storage facility should be equipped with both electrical interlocks and mechanical stops to keep casks from being transported over the spent fuel pool.

13. Sections 6.4 and 9 of ANS-57.2 list those codes and standards referenced in ANS-57.2. Although this regulatory guide endorses with clarifications and modifications ANS-57.2, a blanket endorsement of those referenced codes and

standards is not intended. (Other regulatory guides may contain some such endorsements.)

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants regarding the NRC staff's plans for using this regulatory guide.

This proposed revision has been released to encourage public participation in its development. Except in those cases in which an applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method to be described in the active guide reflecting public comments will be used in the evaluation of applications for construction permits and operating licenses docketed after the implementation date to be specified in the active guide. Implementation by the staff will in no case be earlier than June 30, 1982.

APPENDIX A

NUCLEAR CRITICALITY SAFETY

1. SCOPE OF NUCLEAR CRITICALITY SAFETY ANALYSIS

1.1 A nuclear criticality safety analysis should be performed for each system that involves the handling, transfer, or storage of spent fuel assemblies at light-water reactor (LWR) spent fuel storage facilities.

1.2 The nuclear criticality safety analysis should demonstrate that each LWR spent fuel storage facility system is subcritical (k_{eff} not to exceed 0.95).

1.3 The nuclear criticality safety analysis should include consideration of all credible normal and abnormal operating occurrences, including:

a. Accidental tipping or falling of a spent fuel assembly,
b. Accidental tipping or falling of a storage rack during transfer,
c. Misplacement of a spent fuel assembly,
d. Accumulation of solids containing fissile materials on the pool floor or at locations in the cooling water system,
e. Fuel drop accidents,
f. Stuck fuel assembly/crane uplifting forces,
g. Horizontal motion of fuel before complete removal from rack,
h. Placing a fuel assembly along the outside of rack, and
i. Objects that may fall onto the stored spent fuel assemblies.

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

1.5 The nuclear criticality safety analysis should explicitly identify spent fuel assembly characteristics upon which subcriticality in the LWR spent fuel storage facility depends.

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

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

2. CALCULATION METHODS AND CODES

Methods used to calculate subcriticality should be validated in accordance with Regulatory Guide 3.41, "Validation of Calculational Methods for Nuclear Criticality Safety," which endorses ANSI N16.9-1975.

3. METHOD TO ESTABLISH SUBCRITICALITY

3.1 The evaluated multiplication factor of fuel in the spent fuel storage racks, k_s, under normal and credible abnormal conditions should be equal to or less than an established maximum allowable multiplication factor, k_a; i.e.,

$$k_s \leq k_a$$

The factor, k_s, should be evaluated from the expression:

$$k_s = k_{sn} + \Delta k_{sb} + \Delta k_u + \Delta k_{sc}$$

where

k_{sn} = the computed effective multiplication factor; k_{sn} is calculated by the same methods used for benchmark experiments for design storage parameters when the racks are loaded with the most reactive fuel to be stored,

Δk_{sb} = the bias in the calculation procedure as obtained from the comparisons with experiments and including any extrapolation to storage pool conditions,

Δk_u = the uncertainty in the benchmark experiments, and

Δk_{sc} = the combined uncertainties in the parameters listed in paragraph 3.2 below.

3.2 The combined uncertainties, Δk_{sc}, include:

a. Statistical uncertainty in the calculated result if a Monte Carlo calculation is used,

b. Uncertainty resulting from comparison with calculational and experimental results,

c. Uncertainty in the extrapolation from experiment to storage rack conditions, and

d. Uncertainties introduced by the considerations enumerated in paragraphs 4.3 and 4.4 below.

3.3 The various uncertainties may be combined statistically if they are independent. Correlated uncertainties should be combined additively.

3.4 All uncertainty values should be at the 95 percent probability level with a 95 percent confidence value.

3.5 For spent fuel storage pool, the value of k_a should be no greater than 0.95.

4. STORAGE RACK ANALYSIS ASSUMPTIONS

4.1 The spent fuel storage rack module design should be based on one of the following assumptions for the fuel:

a. The most reactive fuel assembly to be stored at the most reactive point in the assembly life, or

b. The most reactive fuel assembly to be stored based on a minimum confirmed burnup (see Section 6 of this appendix).

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

4.2 Determination of the most reactive spent fuel assembly includes consideration of the following parameters:

a. Maximum fissile fuel loading,
b. Fuel rod diameter,
c. Fuel rod cladding material and thickness,
d. Fuel pellet density,
e. Fuel rod pitch and total number of fuel rods within assembly,
f. Absence of fuel rods in certain locations, and
g. Burnable poison content.

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

a. Spacing between assemblies,
b. Moderation between assemblies, and
c. Fixed neutron absorbers between assemblies.

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

a. Eccentricity of fuel bundle location within the racks and variations in spacing among adjacent bundles,
b. Dimensional tolerances,
c. Construction materials,
d. Fuel and moderator density (allowance for void formations and temperature of water between and within assemblies),

e. Presence of the remaining amount of fixed neutron absorbers in fuel assembly, and

f. Presence of structural material and fixed neutron absorber in cell walls between assemblies.

4.5 Fuel burnup determination should be made for fuel stored in racks where credit is taken for burnup. The following methods are acceptable:

a. A minimum allowable fuel assembly reactivity should be established, and a reactivity measurement should be performed to ensure that each assembly meets this criterion; or

b. A minimum fuel assembly burnup value should be established as determined by initial fuel assembly enrichment or other correlative parameters, and a measurement should be performed to ensure that each fuel assembly meets the established criterion; or

c. A minimum fuel assembly burnup value should be established as determined by initial fuel assembly enrichment or other correlative parameters, and an analysis of each fuel assembly's exposure history should be performed to determine its burnup. The analyses should be performed under strict administrative control using approved written procedures. These procedures should provide for independent checks of each step of the analysis by a second qualified person using nuclear criticality safety assessment criteria described in paragraph 1.4 above.

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

Consideration should be given to the axial distribution of burnup in the fuel assembly, and a limit should be set on the length of the fuel assembly that is permitted to have a lower average burnup than the fuel assembly average.

5. USE OF NEUTRON ABSORBERS IN STORAGE RACK DESIGN

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

a. The effect of neutron-absorbing materials of construction or added fixed neutron-absorbers may be included in the evaluation if they are designed and fabricated so as to preclude inadvertent removal by mechanical or chemical action.

b. Fixed neutron absorbers should be an integral, nonremovable part of the storage rack.

c. When a fixed neutron absorber is used as the primary nuclear criticality safety control, there should be provision to:

(1) Initially confirm absorber presence in the storage rack, and

(2) Periodically verify continued presence of absorber.

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

6. CREDIT FOR BURNUP IN STORAGE RACK DESIGN

6.1 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 buildup and fission product burnable poison depletion, and the fact that the rates of depletion and plutonium and fission product buildup are not necessarily the same.

6.2 Consideration should be given to the practical implementation of the spent fuel screening process. Factors to be considered in choosing the screening method should include:

a. Accuracy of the method used to determine storage rack reactivity;
b. Reproducibility of the result, i.e., what is the uncertainty in the result?
c. Simplicity of the procedure; i.e., how much disturbance to other operations is involved?
d. Accountability, i.e., ease and completeness of recordkeeping; and
e. Auditability.

DRAFT VALUE/IMPACT STATEMENT

1. PROPOSED ACTION

1.1 Description

Each nuclear power plant has a spent fuel storage facility. 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, "Domestic Licensing of Production and Utilization Facilities," requires that fuel storage and handling systems be designed to ensure adequate safety under normal and postulated accident conditions. The proposed action would provide an acceptable method for implementing this criterion. This action would be an update of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis."

1.2 Need for Proposed Action

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

1.3 Value/Impact of Proposed Action

1.3.1 NRC

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

1.3.2 Government Agencies

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

1.3.3 Industry

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

1.3.4 Public

No major impact on the public can be foreseen.

1.4 Decision on Proposed Action

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

2. TECHNICAL APPROACH

The American Nuclear Society published ANS-57.2 (ANSI N210), "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations." Part of the update of Regulatory Guide 1.13 would be an evaluation of this standard and possible endorsement by the NRC. Also, recommendations made by Task A-36, which were published in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," would be included.

3. PROCEDURAL APPROACH

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

4. STATUTORY CONSIDERATIONS

4.1 NRC AUTHORITY

Authority for this regulatory guide is derived from the safety requirements of the Atomic Energy Act of 1954, as amended, through the Commission's regulations, in particular, General Design Criterion 61 of Appendix A to 10 CFR Part 50.

4.2 Need for NEPA Assessment

The proposed action is not a major action as defined by paragraph 51.5(a)(10) of 10 CFR Part 51 and does not require an environmental impact statement.

5. CONCLUSION

Regulatory Guide 1.13 should be updated.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 14, 1978

To All Power Reactor Licensees

Gentlemen:

Enclosed for your information and possible future use is the NRC guidance on spent fuel pool modifications, entitled "Review and Acceptance of Spent Fuel Storage and Handling Applications". This document provides (1) additional guidance for the type and extent of information needed by the NRC Staff to perform the review of licensee proposed modifications of an operating reactor spent fuel storage pool and (2) the acceptance criteria to be used by the NRC Staff in authorizing such modifications. This includes the information needed to make the findings called for by the Commission in the Federal Register Notice dated September 16, 1975 (copy enclosed) with regard to authorization of fuel pool modifications prior to the completion of the Generic Environmental Impact Statement, "Handling and Storage of Spent Fuel from Light Water Nuclear Power Reactors".

The overall design objectives of a fuel storage facility at a reactor complex are governed by various Regulatory Guides, the Standard Review Plan (NUREG-75/087), and various industry standards. This guidance provides a compilation in a single document of the pertinent portions of these applicable references that are needed in addressing spent fuel pool modifications. No additional regulatory requirements are imposed or implied by this document.

Based on a review of license applications to date requesting authorization to increase spent fuel storage capacity, the staff has had to request additional information that could have been included in an adequately documented initial submittal. If in the future you find it necessary to apply for authorization to modify onsite spent fuel storage capacity, the enclosed guidance provides the necessary information and acceptance criteria utilized by the NRC staff in evaluating these applications. Providing the information needed to evaluate the matters covered by this document would likely avoid the necessity for NRC questions and thus significantly shorten the time required to process a fuel pool modification amendment.

Sincerely,

Brian Grimes

Brian K. Grimes, Assistant Director
for Engineering and Projects
Division of Operating Reactors

Enclosures:
1. NRC Guidance
2. Notice

ENCLOSURE NO. 1

OT POSITION FOR REVIEW AND ACCEPTANCE OF SPENT FUEL STORAGE AND HANDLING APPLICATIONS

I. BACKGROUND

Prior to 1975, low density spent fuel storage racks were designed with a large pitch, to prevent fuel pool criticality even if the pool contained the highest enrichment uranium in the light water reactor fuel assemblies. Due to an increased demand on storage space for spent fuel assemblies, the more recent approach is to use high density storage racks and to better utilize available space. In the case of operating plants the new rack system interfaces with the old fuel pool structure. A proposal for installation of high density storage racks may involve a plant in the licensing stage or an operating plant. The requirements of this position do not apply to spent fuel storage and handling facilities away from the nuclear reactor complex.

On September 16, 1975, the Commission announced (40 F. R. 42801) its intent to prepare a generic environmental impact statement on handling and storage of spent fuel from light water power reactors. In this notice, the Commission also announced its conclusion that it would not be in the public interest to defer all licensing actions intended to ameliorate a possible shortage of spent fuel storage capacity pending completion of the generic environmental impact statement.

The Commission directed that in the consideration of any such proposed licensing action, an environmental impact statement or environmental impact appraisal shall be prepared in which five specific factors in addition to the normal cost/benefit balance and environmental stresses should be applied, balanced and weighed.

The overall design objectives of a fuel storage facility at the reactor complex are governed by various Regulatory Guides, the Standard Review Plan, and industry standards which are listed in the reference section. Based on the reviews of such applications to date it is obvious that the staff had to request additional information that could be easily included in an adequately documented initial submittal. It is the intent of this document to provide guidance for the type and extent of information needed to perform the review, and to indicate the acceptance criteria where applicable.

II. REVIEW DISCIPLINES

The objective of the staff review is to prepare (1) Safety Evaluation Report, and (2) Environmental Impact Appraisal. The broad staff disciplines involved are nuclear, mechanical, material, structural, and environmental.

Nuclear and thermal-hydraulic aspects of the review include the potential for inadvertant criticality in the normal storage and handling of the spent fuel, and the consequences of credible accidents with respect to criticality and the ability of the heat removal system to maintain sufficient cooling.

Mechanical, material and structural aspects of the review concern the capability of the fuel assembly, storage racks, and spent fuel pool system to withstand the effects of natural phenomena such as earthquakes, tornadoes, flood, effects of external and internal missiles, thermal loading, and also other service loading conditions.

The environmental aspects of the review concern the increased thermal and radiological releases from the facility under normal as well as accident conditions, the occupational radiation exposures, the generation of radioactive waste, the need for expansion, the commitment of material and nonmaterial resources, realistic accidents, alternatives to the proposed action and the cost-benefit balance.

The information related to nuclear and thermal-hydraulic type of analyses is discussed in Section III.

The mechanical, material, and structural related aspects of information are discussed in Section IV.

The information required to complete an environmental impact assessment, including the five factors specified by the Commission, is provided in Section V.

III. NUCLEAR AND THERMAL-HYDRAULIC CONSIDERATIONS

1. Neutron Multiplication Factor

To include all credible conditions, the licensee shall calculate the effective neutron multiplication factor, k_{eff}, in the fuel storage pool under the following sets of assumed conditions:

1.1 Normal Storage

a. The racks shall be designed to contain the most reactive fuel authorized to be stored in the facility without any control rods or any noncontained* burnable poison and the fuel shall be assumed to be at the most reactive point in its life.

b. The moderator shall be assumed to be pure water at the temperature within the fuel pool limits which yields the largest reactivity.

c. The array shall be assumed to be infinite in lateral extent or to be surrounded by an infinitely thick water reflector and thick concrete,** as appropriate to the design.

d. Mechanical uncertainties may be treated by assuming "worst case" conditions or by performing sensitivity studies and obtaining appropriate uncertainties.

e. Credit may be taken for the neutron absorption in structural materials and in solid materials added specifically for neutron absorption, provided a means of inspection is established (refer to Section 1.5).

1.2 Postulated Accidents

The double contingency principle of ANSI N 16.1-1975 shall be applied. It shall require two unlikely, independent, concurrent events to produce a criticality accident.

Realistic initial conditions (e.g., the presence of soluble boron) may be assumed for the fuel pool and fuel assemblies. The

*"Noncontained" burnable poison is that which is not an integral part of the fuel assembly.

**It should be noted that under certain conditions concrete may be a more effective reflector than water.

postulated accidents shall include: (1) dropping of a fuel element on top of the racks and any other achievable abnormal location of a fuel assembly in the pool; (2) a dropping or tipping of the fuel cask or other heavy objects into the fuel pool; (3) effect of tornado or earthquake on the deformation and relative position of the fuel racks; and (4) loss of all cooling systems or flow under the accident conditions, unless the cooling system is single failure proof.

1.3 Calculation Methods

The calculation method and cross-section values shall be verified by comparison with critical experiment data for assemblies similar to those for which the racks are designed. Sufficiently diverse configurations shall be calculated to render improbable the "cancellation of error" in the calculations. So far as practicable the ability to correctly account for heterogeneities (e.g., thin slabs of absorber between storage locations) shall be demonstrated.

A calculational bias, including the effect of wide spacing between assemblies shall be determined from the comparison between calculation and experiment. A calculation uncertainity shall be determined such that the true multiplication factor will be less than the calculated value with a 95 percent probability at a 95 percent confidence level. The total uncertainity factor on k_{eff} shall be obtained by a statistical combination of the calculational and mechanical uncertainties. The k_{eff} value for the racks shall be obtained by summing the calculated value, the calculational bias, and the total uncertainty.

1.4 Rack Modification

For modification to existing racks in operating reactors, the following information should be provided in order to expedite the review:

(a) The overall size of the fuel assembly which is to be stored in the racks and the fraction of the total cell area which represents the overall fuel assembly in the model of the nominal storage lattice cell;

(b) For H_2O + stainless steel flux trap lattices; the nominal thickness and type of stainless steel used in the storage racks and the thermal (.025 ev) macroscopic neutron absorption cross section that is used in the calculation method for this stainless steel;

(c) Also, for the H_2O + stainless steel flux trap lattices, the change of the calculated neutron multiplication factor of infinitely long fuel assemblies in infinitely large arrays in the storage rack (i.e., the k_∞ of the nominal fuel storage lattice cell and the changed k_∞) for:

(1) A change in fuel loading in grams of U^{235}, or equivalent, per axial centimeter of fuel assembly where it is assumed that this change is made by increasing the enrichment of the U^{235}; and,

(2) A change in the thickness of stainless steel in the storage racks assuming that a decrease in stainless steel thickness is taken up by an increase in water thickness and vice versa;

(d) For lattices which use boron or other strong neutron absorbers provide:

(1) The effective areal density of the boron-ten atoms (i.e., B^{10} atoms/cm^2 or the equivalent number of boron-teh atoms for other neutron absorbers) between fuel assemblies.

(2) Similar to Item C, above, provide the sensitivity of the storage lattice cell k_∞ to:

(a) The fuel loading in grams of U^{235}, or equivalent, per axial centimeter of fuel assembly,

(b) The storage lattice pitch; and,

(c) The areal density of the boron-ten atoms between fuel assemblies.

1.5 Acceptance Criteria for Criticality

The neutron multiplication factor in spent fuel pools shall be less than or equal to 0.95, <u>including all uncertainties</u>, under all conditions

(1) For those facilities which employ a strong neutron absorbing material to reduce the neutron multiplication factor for the storage pool, the licensee shall provide the description of onsite tests which will be performed to confirm the presence and retention of the strong absorber in the racks. The results of an initial, onsite verification test shall show within 95 percent confidence limits that there is a sufficient amount of neutron absorber in the racks to maintain the neutron multiplication factor at or below 0.95. In addition, coupon or other type of surveillance testing shall be performed on a statistically acceptable sample size on a

periodic basis throughout the life of the racks to verify the continued presence of a sufficient amount of neutron absorber in the racks to maintain the neutron multiplication factor at or below 0.95.

(2) Decay Heat Calculations for the Spent Fuel

The calculations for the amount of thermal energy that will have to be removed by the spent fuel pool cooling system shall be made in accordance with Branch Technical Position APCSB 9-2 entitled, "Residual Decay Energy for Light Water Reactors for Long Term Cooling." This Branch Technical Position is part of the Standard Review Plan (NUREG 75/087).

(3) Thermal-Hydraulic Analyses for Spent Fuel Cooling

Conservative methods should be used to calculate the maximum fuel temperature and the increase in temperature of the water in the pool. The maximum void fraction in the fuel assembly and between fuel assemblies should also be calculated.

Ordinarily, in order not to exceed the design heat load for the spent fuel cooling system it will be necessary to do a certain amount of cooling in the reactor vessel after reactor shutdown prior to moving fuel assemblies into the spent fuel pool. The bases for the analyses should include the established cooling times for both the usual refueling case and the full core off load case.

A potential for a large increase in the reactivity in an H_2O flux trap storage lattice exists if, somehow, the water is kept out or forced out of the space between the fuel assemblies, conceivably by trapped air or steam. For this reason, it is necessary to show that the design of the storage rack is such that this will not occur and that these spaces will always have water in them. Also, in some cases, direct gamma heating of the fuel storage cell walls and of the intercell water may be significant. It is necessary to consider direct gamma heating of the fuel storage cell walls and of the intercell water to show that boiling will not occur in the water channels between the fuel assemblies. Under postulated accident conditions where all non-Category I spent fuel pool cooling systems become inoperative, it is necessary to show that there is an alternate method for cooling the spent pool water. When this alternative method requires the installation of alternate components or significant physical alteration of the cooling system, the detailed steps shall be described, along with the time required for each. Also, the average amount of water in the fuel pool and the expected heat up rate of this water assuming loss of all cooling systems shall be specified. *

(4) Potential Fuel and Rack Handling Accidents

The method for moving the racks to and from and into and out of the fuel pool, should be described. Also, for plants where the spent fuel pool modification requires different fuel handling procedures than that described in the Final Safety Analysis Report, the differences should be discussed. If potential fuel and rack handling accidents occur, the neutron multiplication factor in the fuel pool shall not exceed 0.95. These postulated accidents shall not be the cause of the loss of cooling for either the spent fuel or the reactor.

(5) Technical Specifications

To insure against criticality, the following technical specifications are needed on fuel storage in high density racks:

1. The neutron multiplication factor in the fuel pool shall be less than or equal to 0.95 at all times.

2. The fuel loading (i.e., grams of uranium-235, or equivalent, per axial centimeter of assembly) in fuel assemblies that are to be loaded into the high density racks should be limited. The number of grams of uranium-235, or equivalent, put in the plant's technical specifications shall preclude criticality in the fuel pool.

Excessive pool water temperatures may lead to excessive loss of water due to evaporation and/or cause fogging. Analyses of thermal load should consider loss of all pool cooling systems. To avoid exceeding the specified spent fuel pool temperatures, consideration shall be given to incorporating a technical specification limit on the pool water temperature that would resolve the concerns described above. For limiting values of pool water temperatures refer to ANSI-N210-1976 entitled, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations," except that the requirements of the Section 9.1.3.III.1.d of the Standard Review Plan is applicable for the maximum heat load with normal cooling systems in operation.

IV. MECHANICAL, MATERIAL, AND STRUCTURAL CONSIDERATIONS

(1) Description of the Spent Fuel Pool and Racks

Descriptive information including plans and sections showing the spent fuel pool in relation to other plant structures shall be provided in order to define the primary structural aspects and elements relied upon to perform the safety-related functions of the pool and the racks. The main safety function of the spent fuel pool and the racks is to maintain the spent fuel assemblies in a safe configuration through all environmental and abnormal loadings, such as earthquake, and impact due to spent fuel cask drop, drop of a spent fuel assembly, or drop of any other heavy object during routine spent fuel handling.

The major structural elements reviewed and the extent of the descriptive information required are indicated below.

(a) Support of the Spent Fuel Racks: The general arrangements and principal features of the horizontal and the vertical supports to the spent fuel racks should be provided indicating the methods of transferring the loads on the racks to the fuel pool wall and the foundation slab. All gaps (clearance or expansion allowance) and sliding contacts should be indicated. The extent of interfacing between the new rack system and the old fuel pool walls and base slab should be discussed, i.e., interface loads, response spectra, etc.

If connections of the racks are made to the base and to the side walls of the pool such that the pool liner may be perforated, the provisions for avoiding leakage of radioactive water of the pool should be indicated.

(b) Fuel Handling: Postulation of a drop accident, and quantification of the drop parameters are reviewed under the environmental discipline. Postulated drop accidents must include a straight drop on the top of a rack, a straight drop through an individual cell all the way to the bottom of the rack, and an inclined drop on the top of a rack. Integrity of the racks and the fuel pool due to a postulated fuel handling accident is reviewed under the mechanical, material, and structural disciplines. Sketches and sufficient details of the fuel handling system should be provided to facilitate this review.

(2) Applicable Codes, Standards and Specifications

Construction materials should conform to Section III, Subsection NF of the ASME* Code. All Materials should be selected to be compatible with the fuel pool environment to minimize corrosion and galvanic effects.

Design, fabrication, and installation of spent fuel racks of stainless steel material may be performed based upon the AISC** specification or Subsection NF requirements of Section III of the ASME B&PV Code for Class 3 component supports. Once a code is chosen its provisions must be followed in entirety. When the AISC specification procedures are adopted, the yield stress values for stainless steel base metal may be obtained from the Section III of the ASME B&PV Code, and the design stresses defined in the AISC specifications as percentages of the yield stress may be used. Permissible stresses for stainless steel welds used in accordance with the AISC Code may be obtained from Table NF-3292.1-1 of ASME Section III Code.

Other materials, design procedures, and fabrication techniques will be reviewed on a case by case basis.

(3) Seismic and Impact Loads

For plants where dynamic input data such as floor response spectra or ground response spectra are not available, necessary dynamic analyses may be performed using the criteria described in Section 3.7 of the Standard Review Plan. The ground response spectra and damping values should correspond to Regulatory Guide 1.60 and 1.61 respectively. For plants where dynamic data are available, e.g., ground response spectra for a fuel pool supported by the ground, floor response spectra for fuel pools supported on soil where soil-structure interaction was considered in the pool design or a floor response spectra for a fuel pool supported by the reactor building, the design and analysis of the new rack system may be performed by using either the existing input parameters including the old damping values or new parameters in accordance with Regulatory Guide 1.60 and 1.61. The use of existing input with new damping values in Regulatory Guide 1.61 is not acceptable.

Seismic excitation along three orthogonal directions should be imposed simultaneously for the design of the new rack system.

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*American Society of Mechanical Engineers Boiler and Pressure Vessel Codes, Latest Edition.

**American Institute of Steel Construction, Latest Edition.

The peak response from each direction should be combined by square root of the sum of the squares. If response spectra are available for a vertical and horizontal directions only, the same horizontal response spectra may be applied along the other horizontal direction.

The effect of submergence of the rack system on the damping and the mass of the fuel racks has been under study by the NRC. Submergence in water may introduce damping from two sources, (a) viscous drag, and (b) radiation of energy away from the submerged body in those cases where the confining boundaries are far enough away to prevent reflection of waves at the boundaries. Viscous damping is generally negligible. Based upon the findings of this current study for a typical high density rack configuration, wave reflections occur at the boundaries so that no additional damping should be taken into account.

A report on the NRC study is to be published shortly under the title "Effective Mass and Damping of Submerged Structures (UCRL-52342)," by R. G. Dong. The recommendations provided in this report on the added mass effect provide an acceptable basis for the staff review. Increased damping due to submergence in water is not acceptable without applicable test data and/or detailed analytical results.

Due to gaps between fuel assemblies and the walls of the guide tubes, additional loads will be generated by the impact of fuel assemblies during a postulated seismic excitation. Additional loads due to this impact effect may be determined by estimating the kinetic energy of the fuel assembly. The maximum velocity of the fuel assembly may be estimated to be the spectral velocity associated with the natural frequency of the submerged fuel assembly. Loads thus generated should be considered for local as well as overall effects on the walls of the rack and the supporting framework. It should be demonstrated that the consequent loads on the fuel assembly do not lead to a damage of the fuel.

Loads generated from other postulated impact events may be acceptable, if the following parameters are described in the report: the total mass of the impacting missile, the maximum velocity at the time of impact, and the ductility ratio of the target material utilized to absorb the kinetic energy.

(4) Loads and Load Combinations:

Any change in the temperature distribution due to the proposed modification should be identified. Information pertaining to the applicable design loads and various combinations thereof should be provided indicating the thermal load due to the effect of the maximum temperature distribution through the pool walls and base slab. Temperature gradient across the rack structure due to differential heating effect between a full and an empty cell should be indicated and incorporated in the design of the rack structure. Maximum uplift forces available from the crane should be indicated including the consideration of these forces in the design of the racks and the analysis of the existing pool floor, if applicable.

The specific loads and load combinations are acceptable if they are in conformity with the applicable portions of Section 3.8.4-II.3 of the Standard Review Plan.

(5) Design and Analysis Procedures

Details of the mathematical model including a description of how the important parameters are obtained should be provided including the following: the methods used to incorporate any gaps between the support systems and gaps between the fuel bundles and the guide tubes; the methods used to lump the masses of the fuel bundles and the guide tubes; the methods used to account for the effect of sloshing water on the pool walls; and, the effect of submergence on the mass, the mass distribution and the effective damping of the fuel bundle and the fuel racks.

The design and analysis procedures in accordance with Section 3.8.4-II.4 of the Standard Review Plan are acceptable. The effect on gaps, sloshing water, and increase of effective mass and damping due to submergence in water should be quantified.

When pool walls are utilized to provide lateral restraint at higher elevations, a determination of the flexibility of the pool walls and the capability of the walls to sustain such loads should be provided. If the pool walls are flexible (having a fundamental frequency less than 33 Hertz), the floor response spectra corresponding to the lateral restraint point at the higher elevation are likely to be greater than those at the base of the pool. In such a case using the response spectrum approach, two separate analyses should be performed as indicated below:

(a) A spectrum analysis of the rack system using response spectra corresponding to the highest support elevation provided that there is not significant peak frequency shift between the response spectra at the lower and higher elevations; and,

(b) A static analysis of the rack system by subjecting it to the maximum relative support displacement.

The resulting stresses from the two analyses above should be combined by the absolute sum method.

In order to determine the flexibility of the pool wall it is acceptable for the licensee to use equivalent mass and stiffness properties obtained from calculations similar to those described "Introduction to Structural Dynamics" by J. M. Biggs published by McGraw Hill Book Company. Should the fundamental frequency of the pool wall model be higher than or equal to 33 Hertz, it may be assumed that the response of the pool wall and the corresponding lateral support to the new rack system are identical to those of the base slab, for which appropriate floor response spectra or ground response spectra may already exist.

(6) Structural Acceptance Criteria

When AISC Code procedures are adopted, the structural acceptance criteria are those given in Section 3.8.4.II.5 of the Standard Review Plan for steel and concrete structures. For stainless steel the acceptance criteria expressed as a percentage of yield stress should satisfy Section 3.8.4.II.5 of the Standard Review Plan. When subsection NF, Section III, of the ASME B&PV Code is used for the racks, the structural acceptance criteria are those given in the Table below.

For impact loading the ductility ratios utilized to absorb kinetic energy in the tensile, flexural, compressive, and shearing modes should be quantified. When considering the effects of seismic loads, factors of safety against gross sliding and overturning of racks and rack modules under all probable service conditions shall be in accordance with the Section 3.8.5.II-5 of the Standard Review Plan. This position on factors of safety against sliding and tilting need not be met provided any one of the following conditions is met:

(a) it can be shown by detailed nonlinear dynamic analyses that the amplitudes of sliding motion are minimal, and impact between adjacent rack modules or between a rack module and the pool walls is prevented provided that the factors of safety against tilting are within the values permitted by Section 3.8.5.II.5 of the Standard Review Plan.

(b) it can be shown that any sliding and tilting motion will be contained within suitable geometric constraints such as thermal clearances, and that any impact due to the clearances is incorporated.

(7) Materials, Quality Control, and Special Construction Techniques:

The materials, quality control procedures, and any special construction techniques should be described. The sequence of installation of the new fuel racks, and a description of the precautions to be taken to prevent damage to the stored fuel during

TABLE

Load Combination	
Elastic Analysis	**Acceptance Limit**
D + L	Normal limits of NF 3231.1a
D + L + E	Normal limits of NF 3231.1a
D + L + To	1.5 times normal limits or the lesser of 2 Sy and Su
D + L + To + E	1.5 times normal limits or the leser of 2 Sy and Su
D + L + Ta + E	1.6 times normal limits or the lesser of 2 Sy or Su
D + L + Ta + E[1]	Faulted condition limits of NF 3231.1c
Limit Analysis	
1.7 (D + L)	Limits of XVII-4000 of Appendix XVII of ASME Code Section III
1.7 (D + L + E)	
1.3 (D + L + To)	
1.3 (D + L + E + To)	
1.1 (D + L + Ta + E)	

Notes:

1. The abbreviations in the table above are those used in Section 3.8.4 of the Standard Review Plan where each term is defined except for Ta which is defined as the highest temperature associated with the postulated abnormal design conditions.

2. Deformation limits specified by the Design Specification limits shall be satisfied, and such deformation limits should preclude damage to the fuel assemblies.

3. The provisions of NF 3231.1 shall be ammended by the requirements of the paragraphs c.2, 3, and 4 of the Regulatory Guide 1.124 entitled "Design Limits and Load Combinations for Class 1 Linear-Type Component Supports."

the construction phase should be provided. Methods for structural qualification of special poison materials utilized to absorb neutron radiation should be described. The material for the fuel rack is reviewed for compatibility inside the fuel pool environment. The quality of the fuel pool water in terms of the pH value and the available chlorides, fluorides, boron, heavy metals should be indicated so that the long-term integrity of the rack structure, fuel assembly, and the pool liner can be evaluated.

Acceptance criteria for special materials such as poison materials should be based upon the results of the qualification program supported by test data and/or analytical procedures.

If connections between the rack and the pool liner are made by welding, the welder as well as the welding procedure for the welding assembly shall be qualified in accordance with the applicable code.

If precipitation hardened stainless steel material is used for the construction of the spent fuel pool racks, hardness testing should be performed on each rack component of the subject material to verify that each part is heat treated properly. In addition, the surface film resulting from the heat treatment should be removed from each piece to assure adequate corrosion resistance.

(8) Testing and Inservice Surveillance

Methods for verification of long-term material stability and mechanical integrity of special poison material utilized for neutron absorption should include actual tests.

Inservice surveillance requirements for the fuel racks and the poison material, if applicable, are dependent on specific design features. These features will be reviewed on a case by case basis to determine the type and the extent of inservice surveillance necessary to assure long-term safety and integrity of the pool and the fuel rack system.

V. COST/BENEFIT ASSESSMENT

1. Following is a list of information needed for the environmental Cost/Benefit Assessment:

1.1 What are the specific needs that require increased storage capacity in the spent fuel pool (SFP)? Include in the response:

(a) status of contractual arrangements, if any, with fuel-storage or fuel-reprocessing facilities,

(b) proposed refueling schedule, including the expected number of fuel assemblies that will be transferred into the SFP at each refueling until the total existing capacity is reached,

(c) number of spent fuel assemblies presently stored in the SFP,

(d) control rod assemblies or other components stored in the SFP, and

(e) the additional time period that spent fuel assemblies would be stored onsite as a result of the proposed expansion, and

(f) the estimated date that the SFP will be filled with the proposed increase in storage capacity.

1.2 Discuss the total construction associated with the proposed modification, including engineering, capital costs (direct and indirect) and allowances for funds used during construction.

1.3 Discuss the alternative to increasing the storage capacity of the SFP. The alternatives considered should include:

(a) shipment to a fuel reprocessing facility (if available),

(b) shipment to an independent spent fuel storage facility,

(c) shipment to another reactor site,

(d) shutting down the reactor.

The discussion of options (a), (b) and (c) should include a cost comparison in terms of dollars per KgU stored or cost per assembly. The discussion of (d) should include the cost for providing replacement power either from within or outside the licensee's generating system.

1.4 Discuss whether the commitment of material resources (e.g., stainless steel, boral, B_4C, etc.) would tend to significantly foreclose the alternatives available with respect to any other licensing actions designed to ameliorate a possible shortage of spent fuel storage capacity. Describe the material resources that would be consumed by the proposed modification.

1.5 Discuss the additional heat load and the anticipated maximum temperature of water in the SFP which would result from the proposed expansion, the resulting increase in evaporation rates, the additional heat load on component and/or plant cooling water systems and whether there will be any significant increase in the amount of heat released to the environment.

V.2. RADIOLOGICAL EVALUATION

2. Following is a list of information needed for radiological evaluation:

2.1 The present annual quantity of solid radioactive wastes generated by the SFP purification system. Discuss the expected increase in solid wastes which will result from the expansion of the capacity of the SFP.

2.2. Data regarding krypton-85 measured from the fuel building ventilation system by year for the last two years. If data are not available from the fuel building ventilation system, provide this data for the ventilation release which includes this system.

2.3 The increases in the doses to personnel from radionuclide concentrations in the SFP due to the expansion of the capacity of the SFP, including the following:

(a) Provide a table showing the most recent gamma isotopic analysis of SFP water identifying the principal radionuclides and their respective concentrations.

(b) The models used to determine the external dose equivalent rate from these radionuclides. Consider the dose equivalent rate at some distance above the center and edge of the pool respectively. (Use relevant experience if necessary).

(c) A table of recent analysis performed to determine the principal airborne radionuclides and their respective concentrations in the SFP area.

(d) The model and assumptions used to determine the increase, if any, in dose rate from the radionuclides identified in (c) above in the SFP area and at the site boundary.

(e) An estimate of the increase in the annual man-rem burden from more frequent changing of the demineralizer resin and filter media.

(f) The buildup of crud (e.g., ^{58}Co, ^{60}Co) along the sides of the pool and the removal methods that will be used to reduce radiation levels at the pool edge to as low as reasonably achievable.

(g) The expected total man-rem to be received by personnel occupying the fuel pool area based on all operations in that area including the doses resulting from (e) and (f) above.

A discussion of the radiation protection program as it affects (a) through (g) should be provided.

2.4 Indicate the weight of the present spent fuel racks that will be removed from the SFP due to the modification and discuss what will be done with these racks.

V.3 ACCIDENT EVALUATION

3.1 The accident review shall consider:

(a) cask drop/tip analysis, and

(b) evaluation of the overhead handling system with respect to Regulatory Guide 1.104.

3.2 If the accident aspects of review do not establish acceptability with respect to either (a) or (b) above, then technical specifications may be required that prohibit cask movement in the spent fuel building.

3.3 If the accident review does not establish acceptability with respect to (b) above, then technical specifications may be required that:

(1) define cask transfer path including control of

(a) cask height during transfer, and

(b) cask lateral position during transfer

(2) indicate the minimum age of fuel in pool sections during movement of heavy loads near the pool. In special cases evaluation of consequences-limiting engineered safety features such as isolation systems and filter systems may be required.

3.4 If the cask drop/tip analysis as in 3.1(a) above is promised for future submittal, the staff evaluation will include a conclusion on the feasibility of a specification of minimum age of fuel based on previous evaluations.

3.5 The maximum weight of loads which may be transported over spent fuel may not be substantially in excess of that of a single fuel assembly. A technical specification will be required to this effect.

3.6 Conclusions that determination of previous Safety Evaluation Reports and Final Environmental Statements have not changed significantly or impacts are not significant are made so that a negative declaration with an Environmental Impact Appraisal (rather than a Draft and Final Environmental Statement) can be issued. This will involve checking realistic as well as conservative accident analyses.

VI. REFERENCES

1. Regulatory Guides

   1.13 - Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations

   1.29 - Seismic Design Classification

   1.60 - Design Response Spectra for Seismic Design of Nuclear Power Plants

   1.61 - Damping Values for Seismic Design of Nuclear Power Plants

   1.76 - Design Basis Tornado for Nuclear Power Plants

   1.92 - Combining Modal Responses and Spatial Components in Seismic Response Analysis

   1.104 - Overhead Crane Handling Systems for Nuclear Power Plants

   1.124 - Design Limits and Loading Combinations for Class 1 Linear-Type Components Supports

2. Standard Review Plan

   3.7 - Seismic Design

   3.8.4 - Other Category I Structures

   9.1 - Fuel Storage and Handling

   9.5.1 - Fire Protection System

3. Industry Codes and Standards

   1. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code Section III, Division 1

   2. American Institute of Steel Construction Specifications

   3. American National Standards Institute, N210-76

   4. American Society of Civil Engineers, Suggested Specification for Structures of Aluminium Alloys 6061-T6 and 6067-T6

5. The Aluminium Association, Specification for Aluminium Structures

SPENT FUEL STORAGE

Intent To Prepare Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel

From the early days of the nuclear power industry in this country, electric utilities planning to construct and operate light water nuclear power reactors contemplated that the used or spent fuel discharged from the reactors would be chemically reprocessed to recover the remaining quantities of fissile and fertile materials (uranium and plutonium), and that the materials so recovered would be recycled back into fresh reactor fuel. It was contemplated by the nuclear industry that spent fuel would be discharged periodically from operating reactors, stored in onsite fuel storage pools for a period of time to permit decay of radioactive materials contained within the fuel and to cool, and periodically shipped offsite for reprocessing. Typically, space was provided in onsite storage pools for about one and one-third nuclear reactor cores. Assuming a four-year reactor fuel reload cycle, such onsite storage pools were planned to hold an average of one year's discharge with sufficient remaining capacity to hold a complete core should unloading of all of the fuel from the reactor be necessary or desirable because of operational difficulties. Under normal operating conditions, an average of five years' discharge could be accommodated before the pools were filled.

Persons planning to conduct commercial reprocessing of spent reactor fuels provided sufficient storage capacity for the spent fuels at their facilities to allow some operational flexibility. Typically, space has been provided or planned for several spent fuel core reloads. Three commercial reprocessing plants have been planned for operation in the United States. The only such plant that has actually operated, Nuclear Fuel Services (NFS) plant at West Valley, New York, was shut down in 1972 for extensive alterations and expansion. There is a pending proceeding before the Nuclear Regulatory Commission (Commission) on NFS's application for a permit to construct these alterations and expansion (docket no. 50-201). The second plant, General Electric Company's Midwest Fuel Recovery Plant at Morris, Illinois, has never operated and is in a decommissioned condition. The third plant, Allied General Nuclear Services' (AGNS) proposed plant in Barnwell, South Carolina, is under construction and is the subject of pending proceedings before the Commission regarding the continuation, modification or suspension of the construction permit from an environmental protection standpoint, and the possible issuance of an operating license (docket no. 50-332), as well as a related matter (docket no. 70-1729).

On May 8, 1975, the Nuclear Regulatory Commission published a notice in the FEDERAL REGISTER setting forth its provisional views that, subject to consideration of comments, (1) a cost-benefit analysis of alternative safeguards programs should be prepared and set forth in draft and final environmental impact statements before a Commission decision is reached on wide-scale use of mixed oxide (recycle plutonium) fuels in light water nuclear power reactors, (2) there should be no additional licenses granted for use of mixed oxide fuel in light water nuclear power reactors except for experimental purposes, (3) with respect to light water nuclear power reactor fuel cycle activities which depend for their justification on wide-scale use of mixed oxide fuel in light water nuclear power reactors, there should be no additional licenses granted which would foreclose future safeguards options or result in unnecessary "grandfathering", and (4) the granting of licenses would not be precluded for fuel cycle activities for experimental and/or technical feasibility purposes.

In light of the status of the three planned commercial reprocessing plants in the United States, as outlined above, the earliest that spent fuel reprocessing could begin on a commercial basis, if authorized, would be late 1976. This assumes that the pending licensing proceedings are completed and licenses issued by this date. However, the spent fuel pools at a number of reactors may soon be filled, and still other reactors will have their pools filled before the end of 1978. Accordingly, even if limited reprocessing should begin in late 1976, there would still be a shortage in spent fuel storage capacity.

The existing pools at the GE and NFS reprocessing plants have some remaining marginal licensed storage capacity which may be able to accommodate the fuel discharges from some reactors; any increases planned at these plants may not be sufficient for industry in the future. Consequently, there is the possibility of a future shortage in licensed spent fuel capacity regardless of the outcome of the proceedings on the May 8th notice.

The Commission has not promulgated any regulation which specifies a given size for on-site reactor spent fuel pools; however, proposals by reactor licensees to significantly change the manner of spent fuel storage or spent fuel pool size would be subject to licensing review by the Commission. In the event that a particular on-site spent fuel pool should become filled, and no alternative form of spent fuel storage could be found the reactor would be eventually forced to shut down and "store" the last spent reactor fuel in the reactor pressure vessel. While no serious adverse consequences to the public health and safety, common defense and security, or environment would likely result, the reactor shutdown would, of course, remove the plant from service and this in turn could adversely affect the electric utility's ability to meet electrical energy needs, or force the utility to operate other plants that are less economical to operate or which have greater environmental impact, and thereby adversely affect the public interest.

There appear to be a number of possible alternatives for increasing spent fuel storage capacity including, among other things, increasing the storage capacity at present reactor sites, and construction of independent spent fuel storage facilities. The shortage of spent fuel storage capacity will occur at individual reactors, and the Commission could adequately address the issues involved on a case-by-case basis within the context of individual licensing reviews. Indeed, the Commission has not, to date, found it necessary, in the discharge of its licensing and related regulatory functions, to develop any overall program of action to deal with the problem. The Commission does, however, have the discretion to deal with issues of this type on a generic basis through the exercise of its rulemaking authority and/or the issuance of a "generic" environmental impact statement. Rulemaking proceedings and/or the issuance of a generic environmental impact statement can, as appropriate, serve as the content for the promulgation of more definitive criteria regarding size and design of spent fuel pools and/or the licensing of independent spent fuel storage facilities, and for consideration of possible revision of the fuel cycle environmental impacts set forth in 10 CFR § 51.20(e) in light of additional spent fuel storage and attendant transportation. Also, the possible implications of increased spent fuel storage on the options available for intermediate and long-term storage of nuclear waste materials could profitably be examined within this context.

One group of interested organizations (Natural Resources Defense Council, Sierra Club, and Businessmen for the Public Interest) has requested the Commission to prepare a generic environmental impact statement on the handling and storage of spent reactor fuel and related matters. (letter to L. V. Gossick from Anthony Z. Roisman, dated May 20, 1975, copy on file at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C.)

While the Commission believes, as earlier indicated, that the matter of spent fuel storage capacity can adequately be addressed on a case-by-case basis within the context of individual licensing reviews, it also believes that, from the standpoint of longer range policy, this matter can profitably be examined in a broader context. It views the preparation of a generic environmental impact statement as a suitable vehicle for such an examination. Notice is hereby given that a generic environmental impact statement on the handling and storage of spent light water power reactor fuel will be prepared by the Commission. The statement will focus on the time period between now and the mid 1980's and will address:

(1) The magnitude of the possible shortage of spent fuel storage capacity;

(2) The alternatives for dealing with the problem including, but not necessarily limited to:

(a) Permitting the expansion of spent fuel storage capacity at power reactors;

(b) Permitting the expansion of spent fuel storage capacity at reprocessing plants;

(c) Licensing of independent spent fuel storage facilities;

(d) Storage of spent fuel from one or more reactors at the storage pools of other reactors;

(e) Ordering that generation of spent fuel (reactor operation) be stopped or restricted;

(3) A cost-benefit analysis of the alternatives listed in (2), along with any other reasonably feasible alternatives, including:

(a) Impacts on public health and safety and the common defense and security;

(b) Environmental, social, and economic costs and benefits;

(c) Commitments of resources;

(d) Implications regarding options available for the intermediate and long-term storage of nuclear waste materials;

(e) Relationship between local short-term uses of the environment and long-term productivity;

(4) The impacts of possible additional transportation of spent fuel that may be required should one or more of the alternatives be adopted;

(5) More definitive standards and criteria to govern the licensing of one or more of the alternatives for dealing with the problem; and

(6) Possible amendments to 10 CFR § 51.20(e).

If appropriate, rulemaking proceedings on items (5) and (6) listed above, or on other issues related to the handling and storage of spent reactor fuel, will be initiated on or about that time of issuance of the draft generic environmental impact statement.

The Commission has also given careful consideration to the question whether licensing actions intended to ameliorate a possible shortage of spent fuel storage capacity, including such actions as the issuance of operating license amendments to permit increases in the storage capacity of reactor spent fuel pools or reprocessing plant spent-fuel storage pools, or the licensing of independent spent fuel storage facilities, should be deferred pending completion of the generic environmental impact statement. Such a deferral was requested in the letter on behalf of Natural Resources Defense Council, Sierra Club, and Businessmen for the Public Interest noted above. In considering this matter, the Commission had two basic objectives in mind: on the one hand, the generic impact statement should not serve as a justification for a full embargo; on the other hand, the public interest considerations associated with such a deferral should be carefully weighed. The Commission has concluded that there should be no such general deferral, and that these related licensing actions may continue during the period required for preparation of the generic statement, subject to certain conditions. In reaching this conclusion, the Commission has considered the following specific factors:

(1) It is likely that each individual licensing action of this type would have a utility that is independent of the utility of other licensing actions of this type;

(2) It is not likely that the taking of any particular licensing action of this type during the time frame under consideration would constitute a commitment of resources that would tend to significantly foreclose the alternatives available with respect to any other individual licensing action of this type;

(3) It is likely that any environmental impacts associated with any individual licensing action of this type would be such that they could adequately be addressed within the context of the individual license application without overlooking any cumulative environmental impacts;

(4) It is likely that any technical issues that may arise in the course of a review of an individual license application can be resolved within that context; and

(5) A deferral or severe restriction on licensing actions of this type would result in substantial harm to the public interest. As indicated, such a restriction or deferral could result in reactor shutdowns as existing spent fuel pools become filled. It now appears that the spent fuel pools of as many as ten reactors could be filled by mid-1978. These ten reactors represent a total of about 6 million kilowatts of electrical energy generating capacity. The removal of these reactors from service could reduce the utilities' service margins to a point where reliable service would be in jeopardy, or force the utilities to rely more heavily on less economical or more polluting forms of generation that would impose economic penalties on consumers and increase environmental impacts.

The Commission expects that any licensing action intended to ameliorate a possible shortage of spent fuel storage capacity during this interim period would be accompanied by an environmental impact statement (10 CFR § 51.5(a)) or impact appraisal (10 CFR § 51.5(c)) tailored to the facts of the case. Since the Commission's general conclusions with respect to the five factors, as set forth above, may not fit the factual circumstances of particular licensing actions, the five factors will be applied, weighed and balanced within the context of these statements or appraisals, in reaching licensing determinations.

Dated at Washington, D.C. this 10th day of September 1975

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

January 18, 1979

To All Power Reactor Licensees

Gentlemen:

Our letter of April 14, 1978, provided NRC Guidance entitled, "Review and Acceptance of Spent Fuel Storage and Handling Applications." Enclosed are modifications to this document for your information and use. These involve pages IV-5 and IV-6 of the document and comprise modified rationale and corrections.

Sincerely,

Brian Grimes

Brian K. Grimes, Assistant Director
for Engineering and Projects
Division of Operating Reactors

Enclosure:
Pages IV-5 and IV-6

cc w/enclosure:
Service List

In order to determine the flexibility of the pool wall it is acceptable for the licensee to use equivalent mass and stiffness properties obtained from calculations similar to those described in "Introduction to Structural Dynamics" by J. M. Biggs published by McGraw Hill Book Company. Should the fundamental frequency of the pool wall model be higher t' n or equal to 33 Hertz, it may be assumed that the response o. the pool wall and the corresponding lateral support to the new rack system are identical to those of the base slab, for which appropriate floor response spectra or ground response spectra may already exist.

(6) Structural Acceptance Criteria

When AISC Code procedures are adopted, the structural acceptance criteria are those given in Section 3.8.4.II.5 of the Standard Review Plan for steel and concrete structures. For stainless steel the acceptance criteria expressed as a percentage of yield stress should satisfy Section 3.8.4.II.5 of the Standard Review Plan. When subsection NF, Section III, of the ASME B&PV Code is used for the racks, the structural acceptance criteria are those given in the Table below. When buckling loads are considered in the design, the structural acceptance criteria shall be limited by the requirements of Appendix XVII-2110(b) of the ASME Boiler and Pressure Vessel Code.

For impact loading the ductility ratios utilized to absorb kinetic energy in the tensile, flexural, compressive, and shearing modes should be quantified. When considering the effects of seismic loads, factors of safety against gross sliding and overturning of racks and rack modules under all probable service conditions shall be in accordance with the Section 3.8.5.II-5 of the Standard Review Plan. This position on factors of safety against sliding and tilting need not be met provided any one of the following conditions is met:

(a) it can be shown by detailed nonlinear dynamic analyses that the amplitudes of sliding motion are minimal, and impact between adjacent rack modules or between a rack module and the pool walls is prevented provided that the factors of safety against tilting are within the values permitted by Section 3.8.5.II.5 of the Standard Review Plan.

(b) it can be shown that any sliding and tilting motion will be contained within suitable geometric constraints such as thermal clearances, and that any impact due to the clearances is incorporated.

(7) Materials, Quality Control, and Special Construction Techniques:

The materials, quality control procedures, and any special construction techniques should be described. The sequence of installation of the new fuel racks, and a description of the precautions to be taken to prevent damage to the stored fuel during

TABLE

Load Combination	
Elastic Analysis	**Acceptance Limit**
D + L	Normal limits of NF 3231.1a
D + L + E	Normal limits of NF 3231.1a
D + L + To	Lesser of 2Sy or Su stress range
D + L + To + E	Lesser of 2Sy or Su stress range
D + L + Ta + E	Lesser of 2Sy or Su stress range
D + L + Ta + E[1]	Faulted condition limits of NF 3231.1c
Limit Analysis	
1.7 (D + L)	Limits of XVII-4000 of Appendix XVII of ASME Code Section III
1.7 (D + L + E)	
1.3 (D + L + To)	
1.3 (D + L + E + To)	
1.1 (D + L + Ta + E)	

Notes:

1. The abbreviations in the table above are those used in Section 3.8.4 of the Standard Review Plan where each term is defined except for Ta which is defined as the highest temperature associated with the postulated abnormal design conditions.

2. Deformation limits specified by the Design Specification limits shall be satisfied, and such deformation limits should preclude damage to the fuel assemblies.

3. The provisions of NF 3231.1 shall be amended by the requirements of the paragraphs c.2, 3, and 4 of the Regulatory Guide 1.124 entitled "Design Limits and Load Combinations for Class 1 Linear-Type Component Supports."

Library

ANS-8.1
ANSI N16.1-1975

American National Standard

nuclear criticality safety in operations with fissionable materials outside reactors

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Revision of
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American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors

Secretariat
American Nuclear Society

Prepared by the
American Nuclear Society
Standards Committee
Subcommittee ANS-8

Published by the
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Approved April 14, 1975
by the
[illegible]

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Foreword

(This Foreword is not a part of American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors, N16.1-1975/ANS-8.1.)

This Standard provides guidance for the prevention of criticality accidents in the handling, storing, processing, and transporting of fissionable materials. It was first drafted in 1958 by a subcommittee of both the Standards Committee of the American Nuclear Society and Sectional Committee N6 of the American Standards Association and was designated American Standard N6.1-1964. (In 1966, the USA Standards Institute was constituted as successor to the ASA; in 1969 the name of the Institute was changed to American National Standards Institute.) Increased basic knowledge and operating experience made desirable the revision designated American National Standard N16.1-1969 which extended American National Standard N6.1-1964 and included the specification of limits applicable to process variables for the prevention of criticality accidents. The wide acceptance of the revision has made desirable its reaffirmation, with some changes to improve clarity and to modify the concentration limit for ^{239}Pu in accordance with recent re-examinations of the minimum critical concentration of ^{239}Pu in water.

The prescribed five-year review of American National Standard N16.1-1969 was performed by Subcommittee 8 of the Standards Committee of the American Nuclear Society, the originating body. The review was managed by H. K. Clark, *E.I. du Pont de Nemours and Co., Savannah River Laboratory.*

This resulting revision was prepared under the guidance of Subcommittee 8, Fissionable Material Outside Reactors, having the following membership:

J. D. McLendon, Chairman, *Union Carbide Corporation, Nuclear Division*
E. B. Johnson, Secretary, *Union Carbide Corporation Nuclear Division*
F. M. Alcorn, *Babcock and Wilcox Company*
H. K. Clark, *Savannah River Laboratory*
E. D. Clayton, *Battelle Pacific Northwest Laboratories*
D. M. Dawson, *General Electric Company*
W. A. Johnson, *U.S. Energy Research and Development Administration*
Norman Ketzlach, *U.S. Nuclear Regulatory Commission*
W. G. Morrison, *Allied Chemical Corporation*
David R. Smith, *Los Alamos Scientific Laboratory*
J. T. Thomas, *Oak Ridge National Laboratory*
Frank E. Woltz, *Goodyear Atomic Corporation*

The American National Standard Committee N16, Nuclear Criticality Safety, which reviewed and approved this revision in 1975, had the following membership:

Dixon Callihan, Chairman
E. B. Johnson, Secretary

Organization Represented	*Name of Representative*
American Institute of Chemical Engineers	Alex Perge
American Nuclear Society	Dixon Callihan
American Society for Testing and Materials	A. N. Tschaeche
	John H. Bystrom (Alt)
Atomic Industrial Forum, Inc.	D. F. Cronin
Health Physics Society	Fred Sanders
	F. F. Haywood (Alt)
Institute of Nuclear Materials Management	C. Leslie Brown
	George Wuller (Alt)
U.S. Energy Research and Development Administration	Wade C. McCluggage
U.S. Nuclear Regulatory Commission	R. L. Stevenson
U.S. Public Health Service	Robert Jacobs
Individual Members	E. B. Johnson
	H. C. Paxton

Contents

Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors

1. Introduction

Operations with fissionable materials introduce risks of a criticality accident resulting in the release of radiation that may be lethal to nearby personnel. However, experience has shown that extensive operations can be performed safely and economically when proper precautions are exercised. The few criticality accidents that have occurred show frequency and severity rates far below those typical of nonnuclear accidents. This favorable record can be maintained only by continued adherence to good operating practices. In seeking a balance between operating cost and nuclear criticality safety, the protection of operating personnel[1] and the public must be the dominant consideration; however, it is not possible to establish safe processes in an absolute sense by this or any standard.

2. Scope

This Standard is applicable to operations with ^{235}U, ^{233}U, ^{239}Pu, and other fissionable materials outside of nuclear reactors, except the assembly of these materials under controlled conditions, such as in critical experiments. Generalized basic criteria are presented and limits are specified for some simple single fissionable units, but not for multiunit arrays.[2] This Standard does not include the details of administrative controls, the design of processes or equipment, the description of instrumentation for process control, or detailed criteria to be met in transporting fissionable materials.

3. Definitions

3.1 Limitations. The definitions given below are of a restricted nature for the purposes of this Standard. Other specialized terms are defined in American National Standard Glossary of Terms in Nuclear Science and Technology, N1.1-1967.

[1] Guidance for establishing an alarm system is contained in American National Standard Criticality Accident Alarm System, N16.2-1969.

[2] Limits for certain multiunit arrays are contained in American National Standard Guide for Nuclear Criticality Safety in the Storage of Fissile Materials, N16.5-1975.

3.2 Glossary of Terms

shall, should, and may. The word "shall" is used to denote a requirement, the word "should" to denote a recommendation, and the word "may" to denote permission, neither a requirement nor a recommendation. In order to conform with this Standard, all operations shall be performed in accordance with its requirements, but not necessarily with its recommendations.

areal density. The product of the thickness of an infinite, uniform slab and the concentration of fissionable material within the slab; hence, the mass of fissionable material per unit area projected onto a plane parallel to the slab surfaces.

criticality accident. The release of energy as a result of accidentally producing a self-sustaining or divergent neutron chain reaction.

nuclear criticality safety. The prevention or termination of inadvertent nuclear chain reactions in nonreactor environments.

slurry. A mixture or suspension of water and insoluble particulate fissionable material such as metal shavings, oxide particles, salts, or precipitates.

subcritical limit (limit). The limiting value assigned to a controlled parameter that results in a system known to be subcritical provided the limiting value of no other controlled parameter of the system is violated. The subcritical limit allows for uncertainties in the calculations and experimental data used in its derivation but not for contingencies, e.g., double batching or failure of analytical techniques to yield accurate values.

4. Nuclear Criticality Safety Practices

4.1 Administrative Practices

4.1.1 Responsibilities. Management shall clearly establish responsibility for nuclear criticality safety. Supervision should be made as responsible for nuclear criticality safety as for production, development, research, or other functions. Nuclear criticality safety differs in no intrinsic way from industrial safety, and good managerial practices apply to both.

Management shall provide personnel skilled in the interpretation of data pertinent to nuclear criticality safety and familiar with operations to serve as advisors to supervision. These specialists should be, to the extent practicable administratively independent of process supervision.

Management shall establish the criteria to be satisfied by nuclear criticality safety controls. Distinction may be made between shielded and unshielded facilities, and the criteria may be less stringent when adequate shielding assures the protection of personnel.

4.1.2 Process Analysis. Before a new operation with fissionable materials is begun or before an existing operation is changed, it shall be determined that the entire process will be subcritical under both normal and credible abnormal conditions.[3]

4.1.3 Written Procedures. Operations to which nuclear criticality safety is pertinent shall be governed by written procedures. All persons participating in these operations shall be familiar with the procedures.

4.1.4 Materials Control. The movement of fissionable materials shall be controlled. Appropriate materials labeling and area posting shall be maintained specifying material identification and all limits on parameters that are subjected to procedural control.

4.1.5 Operational Control. Deviations from procedures and unforeseen alterations in process conditions that affect nuclear criticality safety shall be investigated promptly and action shall be taken to prevent a recurrence.

4.1.6 Operational Reviews. Operations shall be reviewed frequently to ascertain that procedures are being properly followed and that process conditions have not been altered so as to affect the nuclear criticality safety evaluation. These reviews shall be conducted, in consultation with operating personnel, by individuals who shall be knowledgeable in nuclear criticality safety and who, to the extent practicable, should not be immediately responsible for the operation.

4.1.7 Emergency Procedures. Emergency procedures shall be prepared and approved by management. Organizations, local and offsite, that are expected to respond to emergencies shall be made aware of conditions that might be encountered, and they should be assisted in preparing suitable procedures governing their responses.

4.2 Technical Practices

4.2.1 Controlling Factors. Nuclear criticality safety is achieved by exercising control over:

(1) The mass, and distribution of all fissionable materials, and

(2) The mass, distribution, and nuclear properties of all other materials with which the fissionable materials are associated.

4.2.2 Double Contingency Principle. Process designs should, in general, incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible.

4.2.3 Geometry Control. Where practicable, reliance should be placed on equipment design in which dimensions are limited, rather than on administrative controls. Full advantage may be taken of any nuclear characteristics of the process materials and equipment. Control shall be exercised to maintain all dimensions and nuclear properties on which reliance is placed.

4.2.4 Neutron Absorbers. Reliance may be placed on neutron-absorbing materials, such as cadmium and boron, that are incorporated in process materials or equipment or both.[4] Control shall be exercised to maintain their continued presence with the intended distributions and concentrations. Care should be taken with solutions of absorbers because of the difficulty of exercising such control.

4.2.5 Subcritical Limits. Where applicable data are available, subcritical limits shall be established on bases derived from experiments, with adequate allowance for uncertainties in the data. In the absence of directly applicable experimental measurements, the limits may be derived from calculations made by a method shown to be valid by comparison with experimental data, provided sufficient allowances are made for uncertainties in the data and in the calculations.

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[3] In some cases it may be necessary to resort to *in situ* neutron multiplication measurements to confirm the subcriticality of proposed configurations. Guidance for safety in performing such measurements is contained in American National Standard for Safety in Conducting Subcritical Neutron-Multiplication Measurements In Situ, N16.3-1975.

[4] Guidance for the use of a particular absorber is contained in American National Standard Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material, N16.4-1971.

5. Single-Parameter Limits for Fissile Nuclides

Operations with fissile materials may be performed safely by complying with any one of the subcritical limits given in 5.1, 5.2, and 5.3 provided the conditions under which it applies are maintained. A limit shall be applied only when the effects of neutron reflectors and of other nearby fissionable materials are no greater than reflection by an unlimited thickness of water. The limits shall not be applied to mixtures of ^{235}U, and ^{233}U, and ^{239}Pu.

Process specifications shall incorporate margins to protect against uncertainties in process variables and against a limit being accidentally exceeded.

5.1 Uniform Aqueous Solutions. Any one of the limits of Table 1 is applicable provided a uniform aqueous solution is maintained and provided, for ^{239}Pu, at least four nitrate ions are present for each plutonium ion. The ^{239}Pu limits apply to mixtures of plutonium isotopes provided the concentration of ^{240}Pu exceeds that of ^{241}Pu and provided ^{241}Pu is considered to be ^{239}Pu in computing mass or concentration.

5.2 Slurries

5.2.1 Uniform Slurries. The limits of 5.1 may be used for macroscopically uniform slurries, provided:

(1) There are at least four nitrate ions intimately associated with each plutonium atom, and

(2) For the dimensional and volume limits, the ratio of hydrogen-to-fissionable material does not exceed that in an aqueous solution having the same concentration of fissionable material.

The limit on the enrichment of uranium in 5.1 is valid [8][5] only for slurries in which the ratio of surface-to-volume of the particles is at least 80 cm^{-1}.

5.2.2 Nonuniform Slurries. The limits on cylinder diameter and slab thickness in Table 1 may be used for nonuniform slurries [5] provided:

(1) Four nitrate ions are intimately associated with each plutonium atom,

[5] Numbers in brackets refer to corresponding numbers in Section 8, References.

Table 1
Single-Parameter Limits for Uniform Aqueous Solutions Containing Fissile Nuclides

Parameter	Subcritical Limit for ^{235}U	Subcritical Limit for ^{233}U	Subcritical Limit for ^{239}Pu Provided N:Pu≥4
Mass of fissile nuclide, kg	0.76[a]	0.55[a]	0.51[b]
Solution cylinder diameter, cm	13.9[a]	11.5[a]	15.7[b]
Solution slab thickness, cm	4.6[a]	3.0[a]	5.8[b]
Solution volume, liters	5.8[a]	3.5[a]	7.7[b]
Concentration of fissile nuclide, g/liter	11.5[a]	10.8[a]	7.0[c]
Areal density of fissile nuclide, g/cm^2	0.40[a]	0.35[a]	0.25[d]
Uranium enrichment, wt % ^{235}U	1.00[e]	—	—
Uranium enrichment in presence of two nitrate ions per uranium atom, wt % ^{235}U	2.07[f]	—	—

[a] Data from Ref. 1 (The references are listed in Section 8)
[b] Data from Ref. 2
[c] Data from Refs. 3 and 4
[d] Data from Ref. 5
[e] Data from Ref. 6
[f] Data from Ref. 7

(2) The restriction on the ratio of hydrogen-to-fissionable atoms, specified in Condition 2 of 5.2.1, is met everywhere throughout the system,

(3) For cylinders, the concentration gradient is only along the length, and

(4) For slabs, the concentration gradient is only parallel to the faces.

For ^{239}Pu in the absence of nitrate ions, but with the proviso that no localized regions of density greater than 0.25 g of ^{239}Pu/cm^3 are permitted, limits of 15.1 and 5.4 cm on cylinder diameter and slab thickness, respectively, are applicable [2] under Conditions 2, 3, and 4 above (this section).

The areal densities given in 5.1 are valid for nonuniform slurries provided these densities are uniform.

The subcritical mass limits for ^{235}U, ^{233}U, and ^{239}Pu in nonuniform slurries are 0.70, 0.52, and 0.45 kg, respectively [2, 5]. Nitrate ions need not be present.

5.3 Metallic Units. The enrichment limit for uranium and the mass limits given in Table 2 apply to a single piece having no concave surfaces. They may be extended to an assembly of smaller units provided there is no inter-unit moderation.

The ^{235}U and ^{233}U limits apply to mixtures of either isotope with ^{234}U, ^{236}U, or ^{238}U provided all isotopes except ^{238}U are considered to be ^{235}U or ^{233}U, respectively, in computing mass. The ^{239}Pu limits apply to isotopic mixtures of plutonium provided the concentration of ^{240}Pu exceeds that of ^{241}Pu, all plutonium isotopes are considered to be ^{239}Pu in computing mass, and no more than 1% ^{238}Pu is present.

Table 2
Single-Parameter Limits for Metal Units

Parameter	Subcritical Limit[a] for		
	^{235}U	^{233}U	^{239}Pu
Mass of fissile nuclide, kg	20.1	6.7	4.9
Cylinder diameter, cm	7.3	4.6	4.4
Slab thickness, cm	1.3	0.54	0.65
Uranium enrichment, wt % ^{235}U	5.0	—	—

[a]Data from Ref. 9.

6. Multiparameter Control

Although the single-parameter limits are adequate for many purposes, they are inconveniently and uneconomically small for many others. In many cases, simultaneous limitation of two or more parameters may allow more flexible operational control. General guidance for multiparameter control and for the extension of certain single parameter limits may be found in the technical literature [10-13].

6.1 Uranium Enriched to No More Than 5% ^{235}U. An application of multiparameter control is control of both the enrichment of uranium and one of the parameters of 5. Subcritical limits [14, 15] applicable to aqueous systems containing uranium metal or uranium oxide (UO_2), regardless of the size and shape of metal or oxide pieces, are specified as functions of enrichment in Figs. 1 through 5 which give, respectively, the mass of ^{235}U, the cylinder diameter, the slab thickness, the volume, and the areal density.[6] These limits shall be applied only when the effects of neutron reflectors and other nearby fissionable materials are no greater than reflection by an unlimited thickness of water.

Process specifications shall incorporate margins to protect against uncertainties in process variables and against a limit being accidentally exceeded.

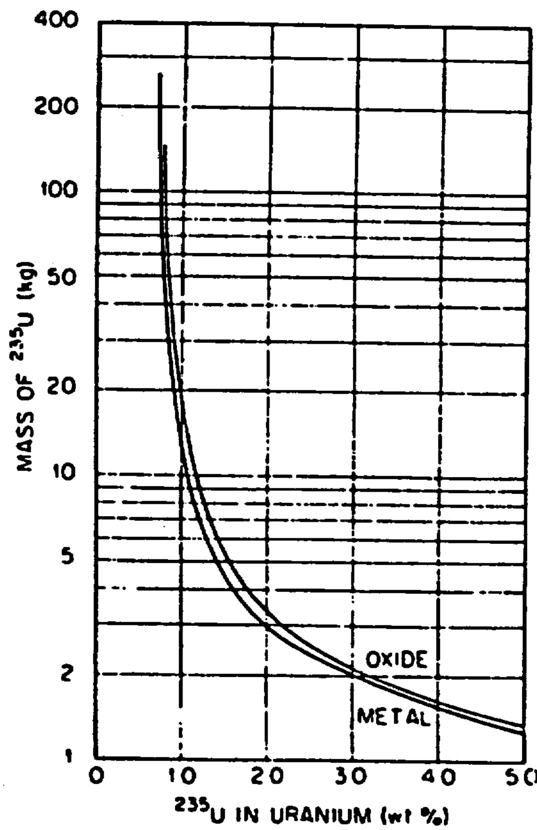

Fig. 1 Mass limit for uranium-water lattices.

[6]The data points through which the curves in Figs. 1-5 were drawn are the subcritical values listed in tables VI-VIII of Reference 15.

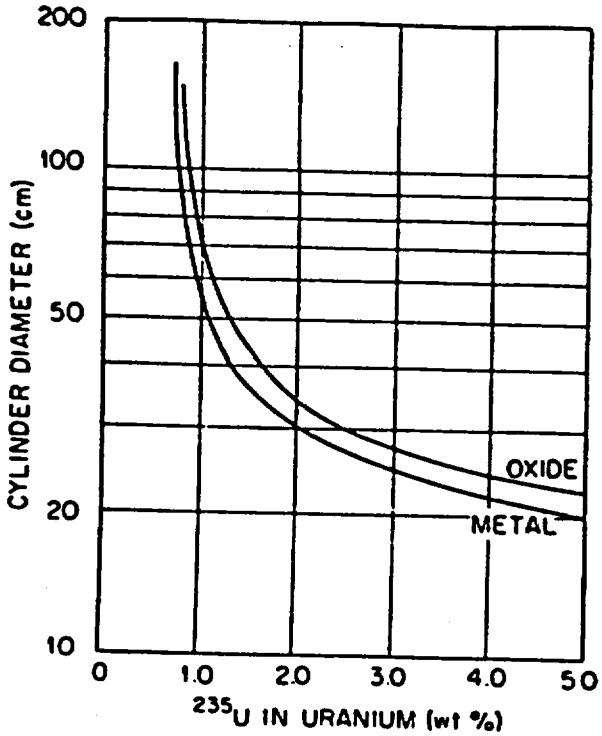

Fig. 2 Cylinder diameter limit for uranium-water lattices.

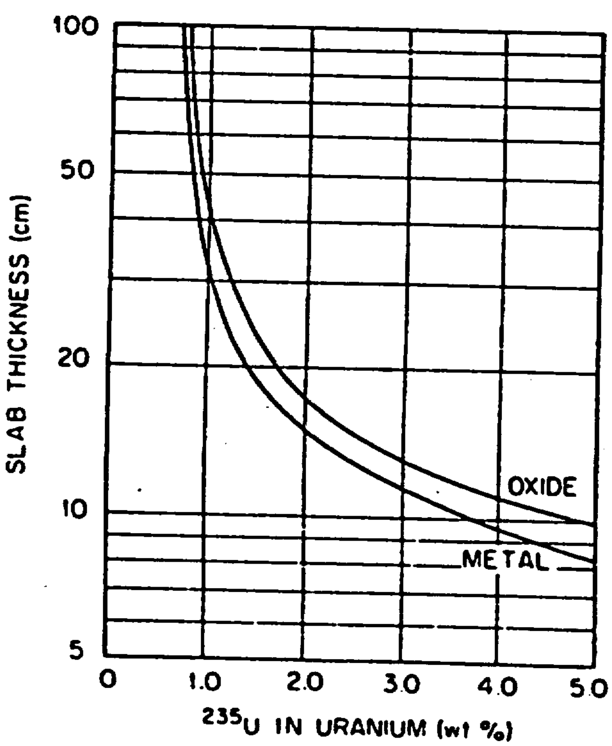

Fig. 3 Slab thickness limit for uranium-water lattices.

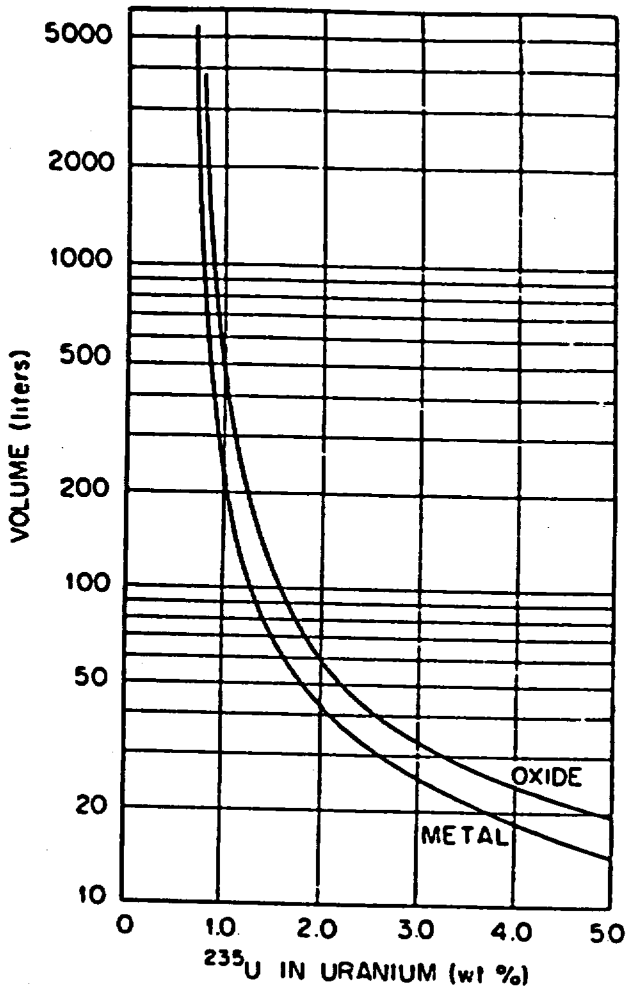

Fig. 4 Volume limit for uranium-water lattices.

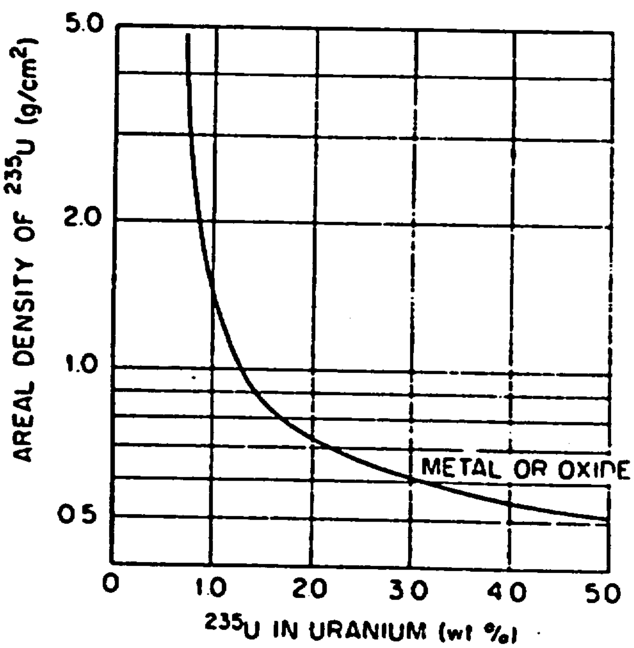

Fig. 5 Areal density limit for uranium-water lattices.

7. Revision of American National Standards Referred to in This Document

When the following American National Standards referred to in this document are superseded by a revision approved by the American National Standards Institute, Inc., the revision shall apply:

1. *Glossary of Terms in Nuclear Science and Technology*, N1.1-1967.
2. *Criticality Accident Alarm System*, N16.2-1969.
3. *Safety in Conducting Subcritical Neutron-Multiplication Measurements In Situ*, N16.3-1975.
4. *Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material*, N16.4-1971.
5. *Guide for Nuclear Criticality Safety in the Storage of Fissile Materials*, N16.5-1975.

8. References

1. J. W. WEBSTER, "Calculated Neutron Multiplication Factors of Uniform Aqueous Solutions of ^{233}U and ^{235}U," ORNL-CDC-2, Criticality Data Center, Oak Ridge National Laboratory (1967).
2. C. R. RICHEY, "Theoretical Analyses of Homogeneous Plutonium Critical Experiments," *Nuclear Science and Engineering*, 31, 32 (1968).
3. H. K. CLARK, "Minimum Critical Concentration of ^{239}Pu," *Transactions of American Nuclear Society*, 17, 278 (1973).
4. C. R. RICHEY, "Re-Examination of the Value for the Minimum Critical Concentration of ^{239}Pu in Water," *Nuclear Science and Engineering*, 55, 244 (1974).
5. H. K. CLARK, "Effect of Distribution of Fissile Material on Critical Mass," *Nuclear Science and Engineering*, 24, 133 (1966).
6. V. I. NEELY and H. E. HANDLER, "Measurements of Multiplication Constant for Slightly Enriched Homogeneous UO_3-Water Mixtures and Minimum Enrichment for Criticality," HW-70310, Hanford Atomic Products Operation (1961).
7. S. R. BIERMAN and G. M. HESS, "Minimum Critical ^{235}U Enrichment for $UO_2(NO_3)_2$ Hydrogenous Systems," ORNL-CDC-5, Criticality Data Center, Oak Ridge National Laboratory (1968). (Report prepared at the Pacific Northwest Laboratory.)
8. C. E. NEWLON, "The Effect of Uranium Density on the Safe ^{235}U Enrichment Criterion," K-1550, Oak Ridge Gaseous Diffusion Plant, Union Carbide Corporation (1962).
9. W. H. ROACH and D. R. SMITH, "Estimates of Maximum Subcritical Dimensions of Single Fissile Metal Units," ORNL-CDC-3, Criticality Data Center, Oak Ridge National Laboratory (1967). (Report prepared at the Los Alamos Scientific Laboratory.)
10. H. C. PAXTON, J. T. THOMAS, D. CALLIHAN, and E. B. JOHNSON, "Critical Dimensions of Systems Containing ^{235}U, ^{239}Pu, and ^{233}U," TID-7028, U.S. Atomic Energy Commission (1964).
11. L. J. TEMPLIN, "Reactor Physics Constants," ANL-5800, Argonne National Laboratory (1963).
12. Subcommittee 8 of the American Standards Association Sectional Committee N6 and Project 8 of the American Nuclear Society Standards Committee, "Nuclear Safety Guide," TID-7016, Rev. 1, U.S. Atomic Energy Commission (1961).
13. H. K. CLARK, "Handbook of Nuclear Safety," DP-532, Savannah River Laboratory (1961).
14. H. K. CLARK, "Maximum Safe Limits for Slightly Enriched Uranium and Uranium Oxide," in *Proceedings of a Symposium on Criticality Control of Fissile Materials, Stockholm*, p. 35, International Atomic Energy Agency, Vienna (1966).
15. H. K. CLARK, "Critical and Safe Masses and Dimensions of Lattices of U and UO_2 Rods in Water," DP-1014, Savannah River Laboratory (1966).

Appendix

(This Appendix is not a part of American National Standard N16.1-1975/ANS-8.1, but is included for information purposes only.)

The determination that a process will be subcritical under credible abnormal conditions requires careful study. The following are typical examples of changes in process conditions that should be considered:

(1) A change in intended shape or dimensions resulting from bulging, corrosion, or bursting of a container, or failure to meet specifications in fabrication.

(2) An increase in the intended mass of fissionable material as the result of operational error.

(3) A change in the intended ratio of moderator to fissionable material resulting from:

(a) Inaccuracies in instruments or chemical analyses.

(b) Flooding, spraying, or otherwise supplying units or groups of units with water, oil, snow (i.e., low-density water), cardboard, wood, or other moderating material.

(c) Evaporating or displacing moderator.

(d) Precipitating fissionable material from solutions.

(e) Diluting concentrated solutions with additional moderator.

(4) A change in the effectiveness of an absorber resulting from:

(a) Loss of solid absorber by corrosion.

(b) Loss of moderator.

(c) Redistribution of absorber and fissionable material by precipitation of one but not the other from a solution.

(d) Redistribution of solid absorber within a matrix of moderator or solution by clumping.

(e) Failure to add the intended amount of absorber to a solution or failure to add it with the intended distribution.

(5) A change in the effectiveness of a reflector resulting from:

(a) An increase in reflector thickness by adding additional material (e.g., water or personnel).

(b) A change in reflector composition such as loss of absorber (e.g., by corrosion of an outer casing of absorber).

(6) A change in the interaction between units and reflectors resulting from:

(a) The introduction of additional units or reflectors (e.g., personnel).

(b) Improper placing of units.

(c) Loss of moderator and absorber between units.

(d) Collapse of a framework used to space units.

(7) An increase in the intended density of fissionable material.

(8) The substitution of units containing more fissionable material than intended as a result of operational error or improper labeling.

ANSI/ANS-8.1-1983
Revision of
ANSI N16.1-1975

American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors

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Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors

1. Introduction

Operations with some fissionable materials introduce risks of a criticality accident resulting in a release of radiation that may be lethal to nearby personnel. However, experience has shown that extensive operations can be performed safely and economically when proper precautions are exercised. The few criticality accidents that have occurred show frequency and severity rates far below those typical of nonnuclear accidents. This favorable record can be maintained only by continued adherence to good operating practices such as are embodied in this standard; however, the standard, by itself, cannot establish safe processes in an absolute sense. Good safety practices must recognize economic considerations, but the protection of operating personnel[1] and the public must be the dominant consideration.

2. Scope

This standard is applicable to operations with fissionable materials outside nuclear reactors, except the assembly of these materials under controlled conditions, such as in critical experiments. Generalized basic criteria are presented and limits are specified for some single fissionable units of simple shape containing ^{233}U, ^{235}U, or ^{239}Pu, but not for multiunit arrays.[2] Requirements are stated for establishing the validity and areas of applicability of any calculational method used in assessing nuclear criticality safety. This standard does not include the details of administrative controls, the design of processes or equipment, the description of instrumentation for process control, or detailed criteria to be met in transporting fissionable materials.

[1]Guidance for establishing an alarm system is contained in American National Standard Criticality Accident Alarm System, ANSI/ANS-8.3-1979.

[2]Limits for certain multiunit arrays are contained in American National Standard Guide for Nuclear Criticality Safety in the Storage of Fissile Materials, ANSI/ANS-8.7-1982.

3. Definitions

3.1 Limitations. The definitions given below are of a restricted nature for the purposes of this standard. Other specialized terms are defined in American National Standard Glossary of Terms in Nuclear Science and Technology, ANSI N1.1-1976/ANS-9 [1].[3]

3.2 Shall, Should, and May. The word "shall" is used to denote a requirement, the word "should" to denote a recommendation, and the word "may" to denote permission, neither a requirement nor a recommendation. In order to conform with this standard, all operations shall be performed in accordance with its requirements, but not necessarily with its recommendations.

3.3 Glossary of Terms

area(s) of applicability. The ranges of material compositions and geometric arrangements within which the bias of a calculational method is established.

areal density. The total mass of fissionable material per unit area projected perpendicularly onto a plane. (For an infinite, uniform slab, it is the product of the slab thickness and the concentration of fissionable material within the slab.)

bias. A measure of the systematic disagreement between the results calculated by a method and experimental data. The uncertainty in the bias is a measure of both the precision of the calculations and the accuracy of the experimental data.

calculational method (method). The mathematical equations, approximations, assumptions, associated numerical parameters (e.g., cross sections), and calculational procedures which yield the calculated results.

controlled parameter. A parameter that is kept within specified limits.

criticality accident. The release of energy as a result of accidentally producing a self-sustaining or divergent neutron chain reaction.

[3]Numbers in brackets refer to corresponding numbers in Section 7, References.

effective multiplication factor (k_{eff}). The ratio of the total number of neutrons produced during a time interval (excluding neutrons produced by sources whose strengths are not a function of fission rate) to the total number of neutrons lost by absorption and leakage during the same interval.

nuclear criticality safety. Protection against the consequences of an inadvertent nuclear chain reaction, preferably by prevention of the reaction.

subcritical limit (limit). The limiting value assigned to a controlled parameter that results in a subcritical system under specified conditions. The subcritical limit allows for uncertainties in the calculations and experimental data used in its derivation but not for contingencies; e.g., double batching or failure of analytical techniques to yield accurate values.

4. Nuclear Criticality Safety Practices

4.1 Administrative Practices

4.1.1 Responsibilities. Management shall clearly establish responsibility for nuclear criticality safety. Supervision should be made as responsible for nuclear criticality safety as for production, development, research, or other functions. Each individual, regardless of position, shall be made aware that nuclear criticality safety in his work area is ultimately his responsibility. This may be accomplished through training and periodic retraining of all operating and maintenance personnel. Nuclear criticality safety differs in no intrinsic way from industrial safety, and good managerial practices apply to both.

Management shall provide personnel skilled in the interpretation of data pertinent to nuclear criticality safety and familiar with operations to serve as advisors to supervision. These specialists should be, to the extent practicable, administratively independent of process supervision.

Management shall establish the criteria to be satisfied by nuclear criticality safety controls. Distinction may be made between shielded and unshielded facilities, and the criteria may be less stringent when adequate shielding and confinement assure the protection of personnel.[4]

4.1.2 Process Analysis. Before a new operation with fissionable materials is begun or before an existing operation is changed, it shall be determined that the entire process will be subcritical under both normal and credible abnormal conditions.[5] Care shall be exercised to determine those conditions which result in the maximum effective multiplication factor (k_{eff}).

4.1.3 Written Procedures. Operations to which nuclear criticality safety is pertinent shall be governed by written procedures. All persons participating in these operations shall understand and be familiar with the procedures. The procedures shall specify all parameters they are intended to control. They should be such that no single, inadvertent departure from a procedure can cause a criticality accident.

4.1.4 Materials Control. The movement of fissionable materials shall be controlled. Appropriate materials labeling and area posting shall be maintained specifying material identification and all limits on parameters that are subjected to procedural control.

4.1.5 Operational Control. Deviations from procedures and unforeseen alterations in process conditions that affect nuclear criticality safety shall be reported to management and shall be investigated promptly. Action shall be taken to prevent a recurrence.

4.1.6 Operational Reviews. Operations shall be reviewed frequently (at least annually) to ascertain that procedures are being followed and that process conditions have not been altered so as to affect the nuclear criticality safety evaluation. These reviews shall be conducted, in consultation with operating personnel, by individuals who are knowledgeable in nuclear criticality safety and who, to the extent practicable, are not immediately responsible for the operation.

4.1.7 Emergency Procedures. Emergency procedures shall be prepared and approved by management. Organizations, local and offsite, that are expected to respond to emergencies shall be made aware of conditions that might be encountered, and they should be assisted in preparing suitable procedures governing their responses.

[4]Guidance is provided in American National Standard Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement, ANSI/ANS-8.10-1983.

[5]In some cases it may be necessary to resort to *in situ* neutron multiplication measurements to confirm the subcriticality of proposed configurations. Guidance for safety in performing such measurements is contained in American National Standard for Safety in Conducting Subcritical Neutron-Multiplication Measurements In Situ, ANSI/ANS-8.6-1983.

4.2 Technical Practices

4.2.1 Controlling Factors. The effective multiplication factor (k_{eff}) of a system containing fissionable material depends on:

(1) The mass and distribution of all fissionable materials and

(2) The mass, distribution, and nuclear properties of all other materials with which the fissionable materials are associated.

Nuclear criticality safety is achieved by controlling one or more parameters of the system within subcritical limits. Control may be exercised administratively through procedures (e.g., by requiring that a mass not exceed a posted limit), by physical restraints (e.g., by confining a solution to a cylindrical vessel with diameter no greater than the subcritical limit), through the use of instrumentation (e.g., by keeping a fissile concentration below a specific limit by devices that measure concentration and prevent its buildup through reflux in a chemical system), by chemical means (e.g., by prevention of conditions that allow precipitation, thereby maintaining concentration characteristic of an aqueous solution), by relying on the natural or credible course of events (e.g., by relying on the nature of a process to keep the density of uranium oxide less than a specified fraction of theoretical), or by other means. All controlled parameters and their limits shall be specified.

4.2.2 Double Contingency Principle. Process designs should, in general, incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible.

4.2.3 Geometry Control. Where practicable, reliance should be placed on equipment design in which dimensions are limited[6] rather than on administrative controls. Full advantage may be taken of any nuclear characteristics of the process materials and equipment. All dimensions and nuclear properties on which reliance is placed shall be verified prior to beginning operations, and control shall be exercised to maintain them.

4.2.4 Neutron Absorbers. Reliance may be placed on neutron-absorbing materials, such as cadmium and boron, that are incorporated in process materials or equipment, or both.[7] Control shall be exercised to maintain their continued presence with the intended distributions and concentrations. Extraordinary care should be taken with solutions of absorbers because of the difficulty of exercising such control.

4.2.5 Subcritical Limits. Where applicable data are available, subcritical limits shall be established on bases derived from experiments, with adequate allowance for uncertainties in the data. In the absence of directly applicable experimental measurements, the limits may be derived from calculations made by a method shown by comparison with experimental data to be valid in accordance with 4.3.

4.3 Validation of a Calculational Method. There are many calculational methods suitable for determining the effective multiplication factor (k_{eff}) of a system or for deriving subcritical limits. The methods vary widely in basis and form, and each has its place in the broad spectrum of problems encountered in the nuclear criticality safety field. However, the general procedure to be followed in establishing validity is common to all.

4.3.1 Bias shall be established by correlating the results of criticality experiments with results obtained for these same systems by the method being validated. Commonly the correlation is expressed in terms of the values of k_{eff} calculated for the experimental systems, in which case the bias is the deviation of the calculated values of k_{eff} from unity. However, other parameters may be used. The bias serves to normalize a method over its area(s) of applicability so that it will predict critical conditions within the limits of the uncertainty in the bias. Generally neither the bias nor its uncertainty is constant; both should be expected to be functions of composition and other variables.

4.3.2 The area(s) of applicability of a calculational method may be extended beyond the range of experimental conditions over which the bias is established by making use of the trends in the bias. Where the extension is large, the method should be supplemented by other calculational methods to provide a better estimate of the bias in the extended area(s).

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[6] Guidance for assessing the safety of piping systems for uranyl nitrate solutions is contained in American National Standard Nuclear Criticality Safety Guide for Pipe Intersections Containing Aqueous Solutions of Enriched Uranyl Nitrate, ANSI/ANS-8.9-1978.

[7] Guidance for the use of a particular absorber is contained in American National Standard Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material, ANSI/ANS-8.5-1979.

4.3.3 A margin in the correlating parameter, which margin may be a function of composition and other variables, shall be prescribed that is sufficient to ensure subcriticality. This margin of subcriticality shall include allowances for the uncertainty in the bias and for uncertainties due to any extensions of the area(s) of applicability.

4.3.4 If the method involves a computer program, checks shall be performed to confirm that the mathematical operations are performed as intended. Any changes in the computer program shall be followed by reconfirmation that the mathematical operations are performed as intended.

4.3.5 Nuclear properties such as cross sections should be consistent with experimental measurements of these properties.

4.3.6 A written report of the validation shall be prepared.[8] This report shall:

(1) Describe the method with sufficient detail, clarity, and lack of ambiguity to allow independent duplication of results.

(2) State computer programs used, the options, recipes for choosing mesh points where applicable, the cross section sets, and any numerical parameters necessary to describe the input.

(3) Identify experimental data and list parameters derived therefrom for use in the validation of the method.

(4) State the area(s) of applicability.

(5) State the bias and the prescribed margin of subcriticality over the area(s) of applicability. State the basis for the margin.

5. Single-Parameter Limits for Fissile Nuclides

Operations with fissile materials may be performed safely by complying with any one of the limits given in 5.1, 5.2, 5.3, and 5.4 for single units provided the conditions under which the limit applies are maintained; these limits were calculated by methods satisfying the requirements of 4.3. A limit shall be applied only when surrounding materials, including other nearby fissionable materials, can be shown to increase the effective multiplication factor (k_{eff}) no more than does enclosing the unit by a contiguous layer of water of unlimited thickness. A limit may be applied to a mixture of fissile nuclides by considering all components of the mixture to be the one with the most restrictive limit.

Process specifications shall incorporate margins to protect against uncertainties in process variables and against a limit being accidentally exceeded.

5.1 Uniform Aqueous Solutions. Any one of the limits of Table 1 is applicable provided a uniform aqueous solution is maintained. It is therefore implied that the concentrations of the saturated solutions are not exceeded. The ^{239}Pu limits apply to mixtures of plutonium isotopes provided the concentration of ^{240}Pu exceeds that of ^{241}Pu and provided ^{241}Pu is considered to be ^{239}Pu in computing mass or concentration. (Less restrictive limits are provided in 6.3 for plutonium isotopic compositions containing appreciable concentrations of ^{240}Pu.) The limit on atomic ratio is equivalent to the limit on solution concentration, but the ratio limit may also be applied to non-aqueous solutions regardless of the chemical form of the fissile nuclide.

5.2 Aqueous Mixtures. The areal densities of Table 1 are independent of chemical compound and are valid for mixtures which may have density gradients provided the areal densities are uniform. The subcritical mass limits for ^{233}U, ^{235}U, and ^{239}Pu in mixtures that may not be uniform are 0.50, 0.70, and 0.45 kg, respectively, and are likewise independent of compound [2-4].

5.2.1 Enrichment Limits. Table 2 contains ^{235}U enrichment limits for uranium compounds mixed homogeneously[9] with water with no limitations on mass or concentration.

[8]Management may limit the distribution of the report to protect proprietary information.

[9]In the "homogeneous" mixtures to which calculations of these limits were normalized the average particle size of dry UO_3 was 60 microns [V. I. NEELEY and H. E. HANDLER, "Measurement of Multiplication Constant for Slightly Enriched Homogeneous UO_3-Water Mixtures and Minimum Enrichment for Criticality," HW-70310, Hanford Atomic Products Operations (August 1961)]. It seems likely that the average particle size of the dihydrate of $UO_2(NO_3)_2$ was approximately 100 microns [V. I. NEELEY, J. A. BERBERET and R. H. MASTERSON, "k_∞ of Three Weight Per Cent ^{235}U Enriched UO_3 and $UO_2(NO_3)_2$ Hydrogeneous Systems," HW-66882, Hanford Atomic Products Operations (September 1961)]. Various H/U ratios in the nitrate mixtures were achieved with 1/8-inch spheres of polyethylene [S. R. BIERMAN and G. M. HESS, "Minimum Critical ^{235}U Enrichment of Homogeneous Uranyl Nitrate," ORNL-CDC-5, Oak Ridge Criticality Data Center (June 1968)].

5.3 Metallic Units. The enrichment limit for uranium and the mass limits given in Table 3 apply to a single piece having no concave surfaces. They may be extended to an assembly of pieces provided there is no interspersed moderation.

The ^{233}U and ^{235}U limits apply to mixtures of either isotope with ^{234}U, ^{236}U, or ^{238}U provided ^{234}U is considered to be ^{233}U or ^{235}U, respectively, in computing mass [3]. The ^{239}Pu limits apply to isotopic mixtures of plutonium provided the concentration of ^{240}Pu exceeds that of ^{241}Pu and all isotopes are considered to be ^{239}Pu in computing mass [4]. Density limits may be adjusted for isotopic composition.

5.4 Oxides. The limits in Tables 4 and 5 apply only if the oxide contains no more than 1.5% water by weight. The mass limits apply to a single piece having no concave surfaces. They may be extended to an assembly of pieces provided there is no additional interspersed moderation.

The mass limit is given equivalently as mass of nuclide and as mass of oxide (including moisture). It is emphasized that the limits in Tables 4 and 5 are valid only under the specified bulk density restrictions.[10] With water content limited to 1.5% the enrichment limit of Table 2 for uranium oxides is increased to 3.2% ^{235}U [3].

[10]The user is cautioned that, particularly for UO_3, material densities in excess of the full densities of Table 4 may be possible and hence that the limits of Table 4 may not be valid for highly compacted oxides. However, it is expected that oxides will generally be in the form of loose powders or, in the case of UO_2, of accumulations of pellets and that the limits of Table 4 and perhaps Table 5 will be valid. Where other density limits are desired, where it is inconvenient to maintain the water content below 1.5% ($H/U \cong 0.47$), or where oxides are non-stoichiometric, the limits may be useful as points of departure in deriving more appropriate values.

The maximum bulk densities were derived from CRC Handbook values of 10.96, 8.3, 7.29, and 11.46 g/cm^3 for UO_2, U_3O_8, UO_3, and PuO_2 together with the assumption of additive volumes of oxide and water. However, x-ray densities of UO_3 as high as 8.46 g/cm^3 have been reported. Moreover, the assumption of additive volumes may be incorrect; with H_2O assigned a density of unity, an effective UO_3 density of 10.47 g/cm^3 is required to produce a reported x-ray density of 6.71 g/cm^3 for α-$UO_2(OH)_2$.

6. Multiparameter Control

Although the single-parameter limits are adequate for many purposes, they are inconveniently and uneconomically small for many others. Simultaneous limitation of two or more parameters results in a less restrictive limit for the one of interest. A few particularly useful examples are given in 6.1 through 6.4. All were calculated by methods satisfying 4.3. These limits shall be applied only when surrounding materials can be shown to increase the effective multiplication factor (k_{eff}) no more than does enclosing the system by a contiguous layer of water of unlimited thickness. General guidance for multiparameter control may be found in the technical literature.[11-14]

Process specifications shall incorporate margins to protect against uncertainties in process variables and against a limit being accidentally exceeded.

6.1 Uranium Metal- and Uranium Oxide-Water Mixtures at Low ^{235}U Enrichment. An application of multiparameter control is control of both the ^{235}U enrichment of uranium and one of the parameters of Section 5. Subcritical limits [5] applicable to aqueous systems containing uranium metal or uranium oxide (UO_2), regardless of the size and shape of metal or oxide pieces, are specified as functions of enrichment in Figs. 1 through 5 which give, respectively, the mass of ^{235}U, the cylinder diameter, the slab thickness, the volume, and the areal density.[15]

[11]H. C. PAXTON, J. T. THOMAS, D. CALLIHAN, and E. B. JOHNSON, "Critical Dimensions of Systems Containing ^{235}U, ^{239}Pu, and ^{233}U," TID-7028, U.S. Atomic Energy Commission (1964).

[12]J. T. THOMAS, "Nuclear Safety Guide, TID-7016, Rev. 2," NUREG/CR-0095 (ORNL/NUREG/CSD-6), Oak Ridge National Laboratory (1978).

[13]H. K. CLARK, "Handbook of Nuclear Safety," DP-532, Savannah River Laboratory (1961).

[14]R. D. CARTER, G. R. KEIL, K. R. RIDGWAY, "Criticality Handbook," ARH-600, Atlantic Richfield Hanford Company (1978).

[15]The data points through which the curves in Figs. 1-5 were drawn are the subcritical values listed in Tables VI-VIII of Ref. [5].

6.2 Aqueous Uranium Solutions at Low ^{235}U Enrichment. A similar application of multiparameter control is control of both ^{235}U enrichment and one of the parameters of Table 1, together with the maintenance of a uniform aqueous solution. Table 6 lists subcritical limits for uniform aqueous solutions of uranium where the enrichment is controlled within the stated limit. Concentrations of saturated solutions, which are here taken to be 5 molar for UO_2F_2 solutions and 2.5 molar for $UO_2(NO_3)_2$ solutions, shall not be exceeded.

6.3 Uniform Aqueous Solutions of $Pu(NO_3)_4$ Containing ^{240}Pu. Reliance on, and hence control of, the isotopic concentration of ^{240}Pu in plutonium permits greater limits for $Pu(NO_3)_4$ solutions than are listed in Table 1.[16] However, the amount of the increase is dependent on ^{241}Pu concentration. Table 7 contains limits for uniform aqueous solutions of $Pu(NO_3)_4$ as a function of isotopic composition. Any ^{238}Pu or ^{242}Pu present shall be omitted in computing the isotopic composition.

6.4 Aqueous Mixtures of Plutonium Containing ^{240}Pu. Subcritical mass limits for plutonium as PuO_2 in aqueous mixtures, which may be nonuniform, where ^{240}Pu and ^{241}Pu are subject to the three pairs of restrictions on isotopic composition of Table 7, are, in increasing order of ^{240}Pu concentration, 0.53, 0.74, and 0.99 kg, respectively [4].

[16]Where plutonium, in addition, is intimately mixed with natural uranium, limits are even greater. Limits for this case are included in American National Standard for Nuclear Criticality Control and Safety of Homogeneous Plutonium-Uranium Fuel Mixtures Outside Reactors, ANSI/ANS-8.12-1978.

7. References

[1] American National Standard Glossary of Terms in Nuclear Science and Technology, ANSI/N1.1-1976/ANS-9. American Nuclear Society, La Grange Park, Ill.

[2] H. K. CLARK, "Subcritical Limits for ^{233}U Systems," *Nucl. Sci. Eng.* 81, 379-395 (1982). American Nuclear Society, La Grange Park, Ill.

[3] H. K. CLARK, "Subcritical Limits for ^{235}U Systems," *Nucl. Sci. Eng.* 81, 351-378 (1982). American Nuclear Society, La Grange Park, Ill.

[4] H. K. CLARK, "Subcritical Limits for Pu Systems," *Nucl. Sci. Eng.* 79, 65-84 (1981). American Nuclear Society, La Grange Park, Ill.

[5] H. K. CLARK, "Critical and Safe Masses and Dimensions of Lattices of U and UO_2 Rods in Water," DP-1014, Savannah River Laboratory, Aiken, S. C., (1966).

When the preceding American National Standard referred to in this document is superseded by a revision approved by the American National Standards Institute, Inc., the revision shall apply.

August 19, 1998

MEMORANDUM TO: Timothy Collins, Chief
Reactor Systems Branch
Division of Systems Safety and Analysis

FROM: Laurence Kopp, Sr. Reactor Engineer /s/
Reactor Systems Branch
Division of Systems Safety and Analysis

SUBJECT: GUIDANCE ON THE REGULATORY REQUIREMENTS FOR CRITICALITY ANALYSIS OF FUEL STORAGE AT LIGHT-WATER REACTOR POWER PLANTS

Attached is a copy of guidance concerning regulatory requirements for criticality analysis of new and spent fuel storage at light-water reactor power plants used by the Reactor Systems Branch. The principal objective of this guidance is to clarify and document current and past NRC staff positions that may have been incompletely or ambiguously stated in safety evaluation reports or other NRC documents. It also describes and compiles, in a single document, NRC staff positions on more recently proposed storage configurations and characteristics in spent fuel rerack or enrichment upgrade requests. This guidance is not applicable to fuel storage in casks, nor does it consider the mechanical, chemical, thermal, radiological, and other aspects of the storage of new and spent fuel.

Attachment:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

GUIDANCE ON THE REGULATORY REQUIREMENTS FOR CRITICALITY ANALYSIS OF FUEL STORAGE AT LIGHT-WATER REACTOR POWER PLANTS

1. INTRODUCTION

This document defines the NRC Reactor Systems Branch guidance for the assurance of criticality safety in the storage of new (unirradiated or fresh) and spent (irradiated) fuel at light-water reactor (LWR) power stations. Safety analyses submitted in support of licensing actions should consider, among other things, normal operation, incidents, and postulated accidents that may occur in the course of handling, transferring, and storing fuel assemblies and should establish that an acceptable margin exists for the prevention of criticality under all credible conditions.

This guidance is not applicable to fuel storage in casks, nor does it consider the mechanical, chemical, thermal, radiological, and other aspects of the storage of new and spent fuel. The guidance considers only the criticality safety aspects of new and spent LWR fuel assemblies and of fuel that has been consolidated; that is, fuel with fuel rods reassembled in a more closely packed array.

The guidance stated here is based, in part, on (a) the criticality positions of Standard Review Plan (SRP) Section 9.1.1 (Ref. 1) and SRP 9.1.2 (Ref. 2), (b) a previous NRC position paper sent to all licensees (Ref. 3), and (c) past and present practices of the staff in its safety evaluation reports (SERs). The guidance also meets General Design Criterion 62 (Ref. 4), which states:

> Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

The principal objective of this guidance is to clarify and document current and past staff positions that may have been incompletely or ambiguously stated in SERs or other staff documents. A second purpose is to state staff positions on recently proposed storage configurations and characteristics in spent fuel rerack or enrichment upgrade requests (for example, multiple-region spent fuel storage racks, checkerboard loading patterns for new and spent fuel storage, credit for burnup in the spent fuel to be stored, and credit for non-removable poison inserts). Although these statements are not new staff positions, this document compiles them in a single paper. In addition, a recently approved staff position for pressurized-water reactors (PWRs) would allow partial credit for soluble boron in the pool water (Ref. 5).

The guidance stated here is applicable to both PWRs and boiling-water reactors (BWRs). The most notable difference between PWR and BWR fuel storage facilities is the larger size of the fuel assemblies and the presence of soluble boron in the spent fuel pool water of PWRs.

The determination of the effective multiplication factor, k_{eff}, for the new or spent fuel storage racks should consider and clearly identify the following:

a. fuel rod parameters, including:

   1. rod diameter

   2. cladding material and cladding thickness

   3. fuel rod pellet or stack density and initial uranium-235 (U-235) enrichment of each fuel rod in the assembly (a bounding enrichment is acceptable)

b. fuel assembly parameters, including:

   1. assembly length and planar dimensions

   2. fuel rod pitch

   3. total number of fuel rods in the assembly

   4. locations in the fuel assembly lattice that are empty or contain nonfuel material

   5. integral neutron absorber (burnable poison) content of various fuel rods and locations in fuel assembly

   6. structural materials (e.g., grids) that are an integral part of the fuel assembly

The criticality safety analysis should explicitly address the treatment of axial and planar variations of fuel assembly characteristics such as fuel enrichment and integral neutron absorber (burnable poison), if present (e.g., gadolinia in certain fuel rods of BWR and PWR assemblies or integral fuel burnable absorber (IFBA) coatings in certain fuel rods of PWR assemblies).

Whenever reactivity equivalencing (i.e., burnup credit or credit for imbedded burnable absorbers) is employed, or if a correlation with the reactivity of assemblies in a standard core geometry is used (k_∞), such as is typically done for BWR racks, the equivalent reactivities must be evaluated in the storage rack configuration. In this latter approach, sufficient uncertainty should be incorporated into the k_∞ limit to account for the reactivity effects of (1) nonuniform enrichment variation in the assembly, (2) uncertainty in the calculation of k_∞, and (3) uncertainty in average assembly enrichment.

If various locations in a storage rack are prohibited from containing any fuel, they should be physically or administratively blocked or restricted to non-fuel material. If the criticality safety of the storage racks relies on administrative procedures, these procedures should be explicitly identified and implemented in operating procedures and/or technical specification limits.

2. CRITICALITY ANALYSIS METHODS AND COMPUTER CODES

A variety of methods may be used for criticality analyses provided the cross-section data and geometric capability of the analytical model accurately represent all important neutronic and geometrical aspects of the storage racks. In general, transport methods of analysis are necessary for acceptable results. Storage rack characteristics such as boron carbide (B_4C) particle size and thin layers of structural and neutron absorbing material (poisons) need to be carefully considered and accurately described in the analytical model. Where possible, the primary method of analysis should be verified by a second, independent method of analysis. Acceptable computer codes include, but are not necessarily limited to, the following:

- CASMO - a multigroup transport theory code in two dimensions
- NITAWL-KENO5a - a multigroup transport theory code in three dimensions, using the Monte Carlo technique
- PHOENIX-P - a multigroup transport theory code in two dimensions, using discrete ordinates
- MONK6B - a multigroup transport theory code in three dimensions, using the Monte Carlo technique
- DOT - a multigroup transport theory code in two dimensions, using discrete ordinates

Similarly, a variety of cross-section libraries is available. Acceptable cross-section libraries include the 27-group, 123-group, and 218-group libraries from the SCALE system developed by the Oak Ridge National Laboratory and the 8220-group United Kingdom Nuclear Data Library (UKNDL). However, empirical cross-section compilations, such as the Hansen-Roach library, are not acceptable for criticality safety analyses (see NRC Information Notice No. 91-26). Other computer codes and cross-section libraries may be acceptable provided they conform to the requirements of this position statement and are adequately benchmarked.

The proposed analysis methods and neutron cross-section data should be benchmarked, by the analyst or organization performing the analysis, by comparison with critical experiments. This qualifies both the ability of the analyst and the computer environment. The critical experiments used for benchmarking should include, to the extent possible, configurations having neutronic and geometric characteristics as nearly comparable to those of the proposed storage facility as possible. The Babcock & Wilcox series of critical experiments (Ref. 6) provides an acceptable basis for benchmarking storage racks with thin strong absorber panels for reactivity control. Similarly, the Babcock & Wilcox critical experiments on close-packed arrays of fuel (Ref. 7) provide an acceptable experimental basis for benchmark analyses for consolidated fuel arrays. A comparison with methods of analysis of similar sophistication (e.g., transport theory) may be used to augment or extend the range of applicable critical experiment data.

The benchmarking analyses should establish both a bias (defined as the mean difference between experiment and calculation) and an uncertainty of the mean with a one-sided tolerance factor for 95-percent probability at the 95-percent confidence level (Ref. 8).

The maximum k_{eff} shall be evaluated from the following expression:

$$k_{eff} = k(calc) + \delta k(bias) + \delta k(uncert) + \delta k(burnup),$$

where

k(calc)	= calculated nominal value of k_{eff},
δk(bias)	= bias in criticality analysis methods,
δk(uncert)	= manufacturing and calculational uncertainties, and
δk(burnup)	= correction for the effect of the axial distribution in burnup, when credit for burnup is taken.

A bias that reduces the calculated value of k_{eff} should not be applied. Uncertainties should be determined for the proposed storage facilities and fuel assemblies to account for tolerances in the mechanical and material specifications. An acceptable method for determining the maximum reactivity may be either (1) a worst-case combination with mechanical and material conditions set to maximize k_{eff} or (2) a sensitivity study of the reactivity effects of tolerance variations. If used, a sensitivity study should include all possible significant variations (tolerances) in the material and mechanical specifications of the racks; the results may be combined statistically provided they are independent variations. Combinations of the two methods may also be used.

3. ABNORMAL CONDITIONS AND THE DOUBLE-CONTINGENCY PRINCIPLE

The criticality safety analysis should consider all credible incidents and postulated accidents. However, by virtue of the double-contingency principle, two unlikely independent and concurrent incidents or postulated accidents are beyond the scope of the required analysis. The double-contingency principle means that a realistic condition may be assumed for the criticality analysis in calculating the effects of incidents or postulated accidents. For example, if soluble boron is normally present in the spent fuel pool water, the loss of soluble boron is considered as one accident condition and a second concurrent accident need not be assumed. Therefore, credit for the presence of the soluble boron may be assumed in evaluating other accident conditions.

4. NEW FUEL STORAGE FACILITY (VAULT)

Normally, fresh fuel is stored temporarily in racks in a dry environment (new fuel storage vault) pending transfer into the spent fuel pool and then into the reactor core. However, moderator may be introduced into the vault under abnormal situations, such as flooding or the introduction of foam or water mist (for example, as a result of fire fighting operations). Foam or mist affects the neutron moderation in the array and can result in a peak in reactivity at low moderator density (called "optimum" moderation, Ref. 9). Therefore, criticality safety analyses must address two independent accident conditions, which should be incorporated into plant technical specifications:

a. With the new fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with pure water, the maximum k_{eff} shall be no greater than 0.95, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level.

b. With the new fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with moderator at the (low) density corresponding to optimum moderation, the maximum k_{eff} shall be no greater than than 0.98, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level.

An evaluation need not be performed for the new fuel storage facility for racks flooded with low-density or full-density water if it can be clearly demonstrated that design features and/or administrative controls prevent such flooding.

Under the double-contingency principle, the accident conditions identified above are the principle conditions that require evaluation. The simultaneous occurrence of other accident conditions need not be considered.

Usually, the storage racks in the new fuel vault are designed with large lattice spacing sufficient to maintain a low reactivity under the accident condition of flooding. Specific calculations, however, are necessary to assure the limiting k_{eff} is maintained no greater than 0.95.

At low moderator density, the presence of relatively weak absorber material (for example, stainless steel plates or angle brackets) is often sufficient to preclude neutronic coupling between assemblies, and to significantly reduce the reactivity. For this reason, the phenomenon of low-density (optimum) moderation is not significant in racks in the *spent fuel pool* under the initial conditions before the pool is flooded.

Under low-density moderator conditions, neutron leakage is a very important consideration. The new fuel storage racks should be designed to contain the highest enrichment fuel assembly to be stored without taking credit for any nonintegral neutron absorber. In the evaluation of the new fuel vaults, fuel assembly and rack characteristics upon which subcriticality depends should be explicitly identified (e.g., fuel enrichment and the presence of steel plates or braces).

5. SPENT FUEL STORAGE RACKS

A. Reference Criticality Safety Analysis

1. For BWR pools or for PWR pools where no credit for soluble boron is taken, the criticality safety analyses must address the following condition, which should be incorporated into the plant technical specifications:

   a. With the spent fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with full-density unborated water, the maximum k_{eff} shall be less than or equal to 0.95, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level.

2. If partial credit for soluble boron is taken, the criticality safety analyses for PWRs must address two independent conditions, which should be incorporated into the plant technical specifications:

   a. With the spent fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with full-density unborated water, the maximum k_{eff} shall be less than 1.0, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level.

   b. With the spent fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with full density water borated to [*] ppm, the maximum k_{eff} shall be no greater than 0.95, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level.[1]

3. The reference criticality safety analysis should also include, as a minimum, the following:

   a. If axial and planar variations of fuel assembly characteristics are present, they should be explicitly addressed, including the locations of burnable poison rods.

   b. For fuel assemblies containing burnable poison, the maximum reactivity should be the peak reactivity over burnup, usually when the burnable poison is nearly depleted.

   c. The spent fuel storage racks should be assumed to be infinite in the lateral dimension or to be surrounded by a water reflector and concrete or structural material as appropriate to the design. The fuel may be assumed to be infinite in the axial dimension, or the effect of a reflector on the top and bottom of the fuel may be evaluated.

   d. The evaluation of normal storage should be done at the temperature (water density) corresponding to the highest reactivity. In poisoned racks, the highest reactivity will usually occur at a water density of 1.0 (i.e., at 4°C). However, if the temperature coefficient of reactivity is positive, the evaluation should be done at the highest temperature expected during normal operations: i.e., equilibrium temperature under normal refueling conditions (including full-core offload), with one coolant train out of service and the pool filled with spent fuel from previous reloads.

4. The fuel assembly arrangement assumed in the criticality safety analysis of the spent fuel storage racks should also consider the following:

[1] [*] is the boron concentration required to maintain the $0.95k_{eff}$ limit without consideration of accidents.

a. the effect of eccentric positioning of fuel assemblies within the storage cells

b. the reactivity consequence of including the flow channel in BWR fuel assemblies

5. If one or more separate regions are designated for the storage of spent fuel, with credit for the reactivity depletion due to fuel burnup, the following applies.

   a. The minimum required fuel burnup should be defined as a function of the initial nominal enrichment.

   b. The spent fuel storage rack should be evaluated with spent fuel at the highest reactivity following removal from the reactor (usually after the decay of xenon-135). Operating procedures should include provision for independent confirmation of the fuel burnup, either administratively or experimentally, before the fuel is placed in storage cells of the designated region(s).

   c. Subsequent decay of longer-life nuclides, such as Pu-241, over the rack storage time may be accounted for to reduce the minimum burnup required to meet the reactivity requirements.

   d. A reactivity uncertainty due to uncertainty in the fuel depletion calculations should be developed and combined with other calculational uncertainties. In the absence of any other determination of the depletion uncertainty, an uncertainty equal to 5 percent of the reactivity decrement to the burnup of interest is an acceptable assumption.

   e. A correction for the effect of the axial distribution in burnup should be determined and, if positive, added to the reactivity calculated for uniform axial burnup distribution.

B. Additional Considerations

1. The reactivity consequences of incidents and accidents such as (1) a fuel assembly drop and (2) placement of a fuel assembly on the outside and immediately adjacent to a rack must be evaluated. Under the double-contingency principle, credit for soluble boron, if present, is acceptable for these postulated accident conditions.

2. If either credit for burnup is assumed or racks of different enrichment capability are in the same fuel pool, fuel assembly misloadings must be considered. Normally, a misloading error involving only a single assembly need be considered unless there are circumstances that make multiple loading errors credible. Under the double-contingency principle, credit for soluble boron, if present, is acceptable for these postulated accident conditions.

3. The analysis must also consider the effect on criticality of natural events (e.g., earthquakes) that may deform, and change in the relative position of, the storage racks and fuel in the spent fuel pool.

4. Abnormal temperatures (above those normally expected) and the reactivity consequences of void formation (boiling) should be evaluated to consider the effect on criticality of loss of all cooling systems or coolant flow, unless the cooling system meets the single-failure criterion. Under the double-contingency principle, credit for soluble boron, if present, is acceptable for these abnormally elevated temperature conditions.

5. Normally, credit may only be taken for neutron absorbers that are an integral (nonremovable) part of a fuel assembly or the storage racks. Credit for added absorber (rods, plates, or other configurations) will be considered on a case-by-case basis, provided it can be clearly demonstrated that design features prevent the absorbers from being removed, either inadvertently or intentionally without unusual effort such as the necessity for special equipment maintained under positive administrative control.

6. If credit for soluble boron is taken, the minimum required pool boron concentration (typically, the refueling boron concentration) should be incorporated into the plant technical specifications or operating procedures. A boron dilution analysis should be performed to ensure that sufficient time is available to detect and suppress the worst dilution event that can occur from the minimum technical specification boron concentration to the boron concentration required to maintain the 0.95k_{eff} design basis limit. The analysis should consider all possible dilution initiating events (including operator error), dilution sources, dilution flow rates, boration sources, instrumentation, administrative procedures, and piping. This analysis should justify the surveillance interval for verifying the technical specification minimum pool boron concentration.

7. Consolidated fuel assemblies usually result in low values of reactivity (undermoderated lattice). Nevertheless, criticality calculations, using an explicit geometric description (usually triangular pitch) or as near an explicit description as possible, should be performed to assure a k_{eff} less than 0.95.

6. REFERENCES

1. NRC, "Standard Review Plan," NUREG-0800, Rev.2, Section 9.1.1, "New Fuel Storage," July 1981.

2. NRC, "Standard Review Plan," NUREG-0800, Rev. 2, Section 9.1.2, "Spent Fuel Storage," July 1981.

3. Brian K. Grimes, NRC, letter to all power reactor licensees, with enclosure, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978.

4. *Code of Federal Regulations*, Title 10, Part 50, Appendix A, General Design Criterion 62, "Prevention of Criticality in Fuel Storage and Handling."

5. Westinghouse Electric Corporation, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," WCAP-14416-NP-A, November 1996.

6. Babcock & Wilcox Company, "Critical Experiments Supporting Close Proximity Water Storage of Power Reactor Fuel," BAW-1484-7, July 1979.

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8. National Bureau of Standards, *Experimental Statistics*, Handbook 91, August 1963.

9. J. M. Cano, R. Caro, and J. M. Martinez Val, "Supercriticality Through Optimum Moderation in Nuclear Fuel Storage," *Nuclear Technology*, Volume 48, May 1980.