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10 CFR Part 72

RIN 3150-AE15

List of Approved Spent Fuel Storage Casks: Additions

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) is amending its list of approved spent fuel storage casks to add one spent fuel storage cask to the list of approved casks. This amendment will allow holders of power reactor operating licenses to store spent fuel in this approved cask under a general license.

EFFECTIVE DATE: (30 days from date of publication in the Federal Register).

ADDRESSES: Copies of the environmental assessment and finding of no significant impact are available for inspection and/or copying for a fee at the NRC Public Document Room, 2120 L Street, NW. (Lower Level), Washington, DC. Single copies of the environmental assessment and the finding of no significant impact are available from the individuals listed under the next heading below.

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FOR FURTHER INFORMATION CONTACT: Mr. Gordon E. Gundersen, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 492-3803, or Mr. James F. Schneider, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 504-2692.

SUPPLEMENTARY INFORMATION:

Background

The NRC published a notice of proposed rulemaking in the Federal Register on June 26, 1992 (57 FR 28645). The comment period closed on September 9, 1992, but was subsequently reopened, as discussed below. The proposed rule would have amended 10 CFR 72.214 to include two additional spent fuel storage casks (i.e., the Transnuclear, Inc., TN-24 cask and the Pacific Sierra Nuclear Associates, VSC-24 cask) on the list of approved spent fuel storage casks that power reactor licensees may use under the provisions of a general license.

Subsequent to the expiration of the September 9, 1992 public comment period, the NRC took steps to implement the provision of Section 2.790 (c) of its regulations (41 FR 11808 (1976)) that provides that information submitted to NRC in a rulemaking proceeding which subsequently forms the basis for a final rule will not be withheld from public disclosure by NRC. Accordingly, on January 21, 1993, additional information, which was previously categorized as vendor proprietary information, was placed in the Public Document Room (PDR) and all Local Public Document Rooms. The additional information made available in the PDR related only to the VSC-24 cask. The second cask (TN-24) will be covered separately in a subsequent notice. In addition, the comment period for the June 26, 1992, proposed rule on the VSC-24 cask was reopened to provide opportunity for public comment on the additional information (January 21, 1993; 58 FR 5301). This comment period expired on February 22, 1993. Further NRC rulemaking activities are planned for the TN-24 cask which is, therefore, not covered in this notice of final rule.

Section 218(a) of the Nuclear Waste Policy Act of 1982 (NWPA) includes the following directive: "The Secretary [of DOE] shall establish a demonstration program in cooperation with the private sector, for the dry storage of spent nuclear fuel at civilian nuclear power reactor sites, with the objective of establishing one or more technologies that the [Nuclear Regulatory] Commission may, by rule, approve for use at the sites of civilian nuclear power reactors without, to the maximum extent practicable, the need for additional site-specific approvals by the Commission." After subsequent DOE technical evaluations and based on a full review of all available data, the Commission approved dry storage of spent nuclear fuel in a final rule published in the Federal Register on July 18, 1990 (55 FR 29181). The final rule established a new Subpart K within 10 CFR Part 72 entitled "General License for Storage of Spent Fuel at Power Reactor Sites."

Irradiated reactor fuel has been handled under dry conditions since the mid-1940's when irradiated fuel examinations began in hot cells. Light water reactor fuel has been examined dry, in hot cells since approximately 1960. Some of these fuels have been stored continuously at hot cells under dry conditions for approximately two decades. Experience with storage of spent fuel in dry casks is extensive. (54 FR 19379 (1990)). Further, as discussed below, the United States has extensive experience in the licensing and safe operation of independent spent fuel storage installations (ISFSI's). At the beginning of 1993 five site specific licenses for dry cask storage had been issued. They are: Virginia Power's Surry Station, issued July 2, 1986; Carolina Power and Light's (CP&L) HB Robinson Station, issued August 13, 1986; Duke Power's Oconee Station, issued January 29, 1990; Public Service of Colorado's Fort St. Vrain facility, issued November 4, 1991; and Baltimore Gas and Electric's (BG&E) Calvert Cliffs Station, issued November 25, 1992. All have commenced operation and loaded fuel with the exception of BG&E. Two hundred and fifty-two assemblies are in storage at Virginia Power, 56 assemblies are in storage at CP&L, 96 assemblies are in storage at Duke Power, and 1482 fuel elements are in storage at Public Service of Colorado; BG&E anticipates loading fuel later in 1993.[1]

As a result of the growing use of dry storage technology experience, NRC has gained over 25 staff years of experience in the review and licensing of dry spent fuel storage systems. To further support the NRC technical staff, the agency draws upon the knowledge and experience of outside scientists and engineers recognized as experts within their respective fields in the performance of the independent safety analysis of the systems and components submitted by applicants for dry cask licenses or certification. Reviews of numerous applications, seeking either site-specific ISFSIs, certificates of compliance or approval of a topical report, have been conducted over the past 7 years.

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[1] EIA Service Report SR/CNEAF/92-01 Spent Fuel Discharges from U.S. Reactors 1990, March 1992.

Section 133 of the NWPA states, in part, that "the Commission shall, by rule, establish procedures for the licensing of any technology approved by the Commission under Section 218(a) for use at the site of any civilian nuclear power reactor." This directive was implemented on July 18, 1990, (55 FR 29181) by the publication in the Federal Register of a final rule establishing a new Subpart L within 10 CFR Part 72 entitled "Approval of Spent Fuel Storage Casks." As a result of that 1990 rulemaking, four dry casks were listed in § 72.214 of Subpart K as approved by the NRC for storage of spent fuel at power reactor sites under a general license.

The final rule adds one additional spent fuel storage cask, the VSC-24 cask, to the list of approved casks in § 72.214. The cask being approved, the VSC-24 cask, is discussed in further detail below. In addition, based on public comments, the Safety Evaluation Report (SER) and Certificate of Compliance for the VSC-24 were modified. Each modification is discussed below as part of the "Analysis of Public Comments" section of this Federal Register notice.

Pacific Sierra Nuclear Associates (PSNA) submitted a "Topical Report on the Ventilated Storage Cask System for Irradiated Fuel" for their VSC-24 cask in February 1989. (VSC means "ventilated storage cask." Twenty-four (24) refers to the number of individual spent fuel assemblies which the VSC-24 is designed to hold.) The NRC completed its review and issued its Safety Evaluation Report (SER) in April 1991 approving the Topical Report for referencing in a site-specific license application. PSNA later submitted its approved Topical Report in the form of a "Safety Analysis Report for the Ventilated Storage Cask System" in November 1991 requesting certification for use under a general license. The NRC conducted additional evaluations and issued a draft Certificate of Compliance and draft SER, dated April 1992, in support of the notice of Proposed Rulemaking published in the Federal Register on June 26, 1992. Based on further staff review and analysis of public comments, with this final rulemaking, NRC is approving the VSC-24 cask for use under a general license and is simultaneously issuing a final Certificate of Compliance and SER.

The paramount objective of 10 CFR Part 72 is protecting the public health and safety, by providing for the safe confinement of the fuel and preventing the degradation of the fuel cladding. The review criteria used by the NRC for review and approval of dry cask storage under 10 CFR Part 72, consider the following: siting, design, quality assurance, emergency planning, training, and physical protection of the fuel. Included in the review of a specific system, either for a certificate of compliance or a site-specific license, are the following: earthquakes, high winds, tornados, tornado driven missiles, lightning, and floods. In addition, applicants must demonstrate to NRC's satisfaction that their proposed dry cask system will resist man-made events such as explosions, fires and drop or tipover accidents[2].

The VSC-24 cask, when used in accordance with the conditions specified in its Certificate of Compliance, meets the requirements of 10 CFR Part 72. This conclusion is reached after a detailed evaluation of the VSC-24 cask by the NRC as documented in the NRC staff's SER. Thus, use of the VSC-24 cask, as approved by the NRC, provides adequate protection of the public health and

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[2] The design bases for these events and accidents are contained within 10 CFR Part 72.

safety and the environment. Holders of power reactor operating licenses under 10 CFR Part 50 will be permitted to store spent fuel in this cask under a general license. A copy of the Certificate of Compliance is available for public inspection and copying for a fee at the NRC Public Document Room, 2120 L Street, NW. (Lower Level), Washington, DC.

Public Responses

In response to the June 26, 1992, and January 21, 1993, Federal Register notices, 232 comments were received from individuals, public interest groups, environmental groups, associations, industry representatives, Congressional representatives, and States. Although a number of the comments were received after the respective September 9, 1992 and February 22, 1993 comment closure dates for the two notices, NRC has considered comments received including those received after the comment closure dates.

As a part of this rulemaking action, NRC received requests for further opportunity to comment and in particular, for NRC to hold a public hearing to review the merits of this action. One request was from Frank J. Kelley, Attorney General of the State of Michigan, dated December 30, 1992, which requested a public hearing. Chairman Selin responded by letter of January 25, 1993, and proposed a transcribed public meeting with the Attorney General to discuss the dry spent fuel cask approval process, to answer questions, and to provide opportunity for interested members of the public to present comments. That public meeting was held on February 23, 1993, from 9:30 a.m. until 12:00 noon in Lansing, Michigan. The Attorney General, his staff, representatives of the NRC staff, and approximately one hundred interested citizens attended the meeting. The meeting was transcribed and the transcript of that meeting, including questions and comments of the Attorney General and citizens attending and participating in the meeting, has been considered by the NRC and is included in the analysis of comments. Additional written comments received within five working days subsequent to the meeting have also been considered by NRC and are included in the analysis of comments below. (See comment response number 57 for information on NRC's response to request for a hearing).

A number of comments were related to disposal of high-level waste, use of dry cask storage technology in general, or use of the VSC-24 cask specifically by Consumers Power Corporation at the Palisades Nuclear Generating Station. Examples of each include:

-Consumers Power Company knew years in advance that the day would come when their spent fuel pool would be full. They should have planned ahead of time for this day. Consumers Power should be required to build a new spent fuel pool, store their waste elsewhere, or to shut down the plant at Palisades;

-Concern was expressed that the review process might become unreasonably delayed and without approval for additional storage capacity, the Palisades plant ultimately will be forced to shut down, a result that would have serious economic consequences for southwestern Michigan.

-The Federal government's failure to resolve questions about the permanent storage of nuclear wastes leaves both the plant and public with limited options: additional storage in pools, additional storage in dry casks or plant shutdown. The federal government has an obligation to resolve the issue of permanent or interim storage. It would be difficult to overstate the need for dispatch in doing so, as hundreds of American communities will eventually face this problem.

-Ten years ago, there was an erroneous assumption that the search for and construction of a final resting place for high-level waste would be much swifter than it has been. A "demonstration" program required by law was supposed to have been for temporary storage. Because of the societal and technical obstacles which radioactive waste disposal presents, even a temporary "demonstration" program is likely to have much longer-term implications. Temporary dry cask storage in Michigan should not become de facto permanent disposal.

-It is not fair to the public of Michigan to link Consumers Power Company's attempts to continue the safe storage of its nuclear fuel with the insistence by others that we shut down Palisades and every other nuclear plant in the country.

These comments deal with broad policy and program issues relating to the storage and disposal of high-level radioactive waste including the Department of Energy's repository program. However, commenters will find a summary of relevant information on many of these broad issues in the responses to comments set out in response numbers 41, 52, 61, and 69 in the following analysis of comments.

Many of the comment letters contained comments that were similar in nature. These comments have been grouped as appropriate and addressed as single issues. The NRC has identified and responded to 75 separate issues that include the significant points raised by each commenter.

Many commenters discussed topics that were not the subject of this rulemaking and thus were not specifically addressed by the staff as a part of this final rulemaking action. These comments expressed opposition to the use of dry cask storage and included suggestions such as the following:

(1) nuclear plants generating radioactive waste should be shut down;

(2) the production of radioactive waste should be stopped when the existing spent fuel pool (and off-load-reactor capacity) is full;

(3) a formal hearing should be required at each site using dry storage casks;

(4) the Palisades Nuclear Plant should be shut down;

(5) the embrittlement of the reactor pressure vessel at Palisades dictates that the plant be shut down and no additional spent fuel generated;

(6) the use of nuclear power should be stopped and existing sites cleaned up;

(7) the use of storage only casks at Palisades is a violation of public trust; and

(8) a research and development program should be conducted on productive uses of spent fuel and on alternative energy sources.

Finally, many commenters expressed concern over the ability of dry cask storage designs to safely store spent fuel. The following responses to these comments reflect a small but important portion of NRC's review of health, safety, and environmental aspects of the VSC-24 cask, to ensure that the cask is designed to provide protection of the public health and safety and environment under both normal conditions and severe, unlikely, but credible accident conditions. Dry cask storage systems are massive devices, designed and analyzed to provide shielding from direct exposure to radiation, confine the spent fuel in a safe storage condition, and prevent releases to the environment. They are designed to perform these tasks relying on passive heat removal and confinement systems without moving parts and with minimal reliance on human intervention to safely fulfill their function for the term of storage. The designs include margins of safety under both normal and accident conditions to provide additional assurance of protection for the public health and safety, the common defense and security and the environment.

Analyses of Public Comments

A. A number of commenters raised issues relating to cask handling and the ability of the cask to withstand drop and tipover accidents.

1. Comment. Some commenters expressed concern about the operational safety of the VSC-24 cask relating to loading the multi-assembly sealed basket (MSB) into the ventilated concrete cask (VCC) and retrieving it. Particularly, the commenters contended that the loading procedure of placing the MSB transfer cask (MTC) on top of the VCC is precarious and the procedure for retrieving the MSB from the VCC is not clearly explained. One commenter indicated that there are unreviewed safety issues associated with handling

equipment including the lifting cables, lifting yoke, lugs, and transfer vehicle, that need further review. Another commenter asked about the training and oversight of personnel performing these activities. Another asked, that if the transfer cask is on top of the VCC in the fuel handling building and a seismic event occurs causing tipover, would this type of event be considered in a Section 50.59 evaluation?

Response. Use of the VSC-24 cask system inside the fuel handling building (including use of the MTC to load and retrieve the MSB from the VCC) would be conducted in accordance with the 10 CFR Part 50 reactor operator's license. These cask handling operations, including loading, retrieval and training, must be evaluated by the general licensee, as required by 10 CFR 72.212(b)(4), to ensure that the procedures are clear and can be conducted safely. The MTC and MSB have been evaluated against the criteria for controlling heavy loads found in NRC publication NUREG-0612 ("Control of Heavy Loads at Nuclear Power Plants") and American National Standards Institute (ANSI) N14.6, "Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More." The lifting yoke associated with the MTC is a special purpose device designed to ANSI N14.6 criteria to ensure that the yoke can safely lift the wet MTC containing the MSB out of the spent fuel pool and can safely lift the dry MTC and MSB to the top of the VCC.

Specific requirements for lifting yokes, cables, and lugs have been identified in the Certificate of Compliance and SER and are not unreviewed safety issues. Part 72 requires that, prior to the use of a cask under the general license, the licensee determine whether activities related to storage of spent fuel under the general license involve any unreviewed safety questions or change to the facility technical specifications, as provided under 10 CFR 50.59. Load handling activities and possible load drop events and structural and radiological consequences are necessary evaluations under 10 CFR 50.59.

For example, the utility's specific analyses for load handling activities at the Palisades plant illustrate the type of mandatory evaluation by the cask user that NRC requires before the VSC-24 cask can be used under 10 CFR Part 72, Subpart K. Among others, one specific event analyzed is the evaluation of the drop of a loaded MTC onto the VCC with tipover of the MTC onto the load distribution system in the track alley area. This analysis would encompass the tipover scenario described above by the commenter who questioned whether it would be part of a utility's Section 50.59 evaluation. The result of this analysis shows that the MSB would not fail and that, while local yielding of the transfer cask may occur, the transfer cask would not fail and could be lifted back to the pool for recovery of all spent fuel in the cask.

2. Comment. One commenter questioned whether, if the MTC were lifted up by the MSB, the weight of the loaded MSB and the MTC would bear on the MSB welds. Another commenter questioned whether the MSB lifting rings could support the weight of the MSB and MTC.

Response. The weight of the MSB and the MTC could be supported by the MSB structural weld and the rings. The weld has been analyzed for this situation and was found to meet the design criteria of paragraphs 4.2.1.1 and 4.2.1.2 of ANSI N14.6, 1986. This standard, which is considered conservative, is specifically written for special lifting devices for shipping containers of radioactive materials. This situation of lifting both the MSB and MTC will not occur under normal operating conditions. However, if it does occur, as discussed above, the weld and the rings can support the weight of the MSB and MTC.

3. Comment. One commenter noted that tiles at the bottom of the VCC could break when the MSB is lowered onto them.

Response. There are numerous ceramic tiles arranged on the base of the VCC which serve as a separator between the flat bottom surface of the MSB and the parallel surface of the VCC liner to prevent the possibility of localized corrosion. Although these tiles could break, there is a substantial margin of safety to prevent breakage. However, if some breakage occurs, the tiles will still perform their function of providing a slight gap between the MSB and the VCC. Although it is not necessary, the Certificate of Compliance has been revised to include a statement that the operating procedures for handling the MSB over the VCC should include the consideration for reducing the likelihood of fracturing the ceramic tiles by impact load.

4. Comment. One commenter questioned why the NRC allows an 80 inch lift height when a drop of over 18 inches may cause enough damage to compromise shielding. Another commenter indicated that the operation of moving the VSC-24 cask from the heavy haul trailer across a piece of "bridge steel" to the storage pad sounded dangerous. One commenter also stated that if the MSB is not centered inside the VCC, possible damage could occur to the coating of the VCC liner or the ceramic tiles on the bottom of the VCC.

Response. The NRC evaluated a possible drop of the cask and has established conditions limiting the lift height for the VSC-24 cask. These conditions include a requirement to inspect the cask after any tipover or drop from a height greater than 18 inches, and the prohibition against lifting the VSC-24 cask to a height greater than 80 inches. The purpose of the 80 inch lift condition is to ensure that the MSB maintains its confinement capability even in the event of a drop of the VSC-24 cask. The MSB has been designed to meet the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) code under Service Level D conditions and a drop of 80 inches should only result, at most, in denting of the MSB shell. The purpose of the inspection for any drop from a height greater than 18 inches is to ensure that the shielding is not compromised and that any damage is immediately identified and repaired. On-site transport procedures with auxiliary equipment such as the "bridge steel" described in the Safety Analysis Report (SAR) have been reviewed and are considered to be appropriate to the design, suitable for use and to meet safety requirements which are not part of the regulations in 10 CFR Part 72. Possible damage to the ceramic tiles was discussed in the response to Comment Number 3. Finally, damage to the coating of the VCC liner would not have safety significance because the liner is not a confinement boundary and does not contribute significantly to shielding. The principal purpose of the VCC liner is to provide an inner form for the concrete during fabrication.

5. Comment. One commenter indicated that if there were a problem with a VSC-24 cask, it could not be removed to the fuel handling building because that is not allowed when the temperature is below 0 °F, and that the temperature in Michigan and Wisconsin is often below 0 °F.

Response. The purpose of restricting VSC-24 cask movement to ambient temperatures above 0 °F is to prevent the possibility of brittle fracture of the MSB in the event of a drop accident. There is a 50 °F margin of safety because the MSB material maintains ductile properties at a test temperature of -50 °F. If a situation for return to the fuel handling building arises while the ambient temperature is below 0 °F, a key option would be for the licensee to determine that the actual MSB material temperature is above 0 °F. In that event movement of the MSB could be accomplished safely without concern for brittle fracture. The MSB would most likely be above 0 °F because of the heat produced by the stored spent fuel. Another option available to a licensee would be not to move the MSB until an ambient temperature above 0 °F is reached.

6. Comment. Some commenters stated that a cask tipover accident while the VSC is on the pad was not considered, even though this type of accident was considered for other casks. Some commenters also noted that drop evaluations of the MSB were performed for only one orientation, although the NRC requires multiple drop orientations for other designs.

Response. A cask tipover accident was not specifically performed for the VSC-24 cask. However, PSNA performed an engineering analysis of cask drops from both vertical and horizontal positions which represent more severe accidents than a tipover. Therefore, NRC concluded it was not necessary to perform a tipover analysis. With respect to drop orientation, the MSB was analyzed for both vertical and horizontal drop orientations.

7. Comment. One commenter asserted that the design of the MSB is such that it is susceptible to buckling under certain off-normal and accident conditions. The commenter further indicated that this is a departure from previous spent fuel cask design and licensing criteria which allow no buckling of the basket structure.

Response. The NRC believes that this commenter refers to the fuel basket and not the MSB shell. The MSB basket structure was analyzed and the NRC concluded that buckling would not be a safety concern as discussed below.

The critical load for buckling was calculated for a single storage tube and compared to the actual load under a vertical deceleration of 124 g that would result from a drop of 80 inches. The results of the analysis indicate that there is a safety factor of 5 for a tube against buckling. Because of the conservative approach in analyzing a single fuel storage tube rather than the entire basket, the NRC believes that a higher safety factor would exist for the basket assembly. Thus, the NRC is not departing from previous design and licensing criteria.

8. Comment. Some commenters noted that the NRC allowed PSNA to use Electric Power Research Institute (EPRI) report NP-4830 in their VSC-24 cask SAR, but did not allow vendors of metal casks to reference this report in their SAR's.

Response. The concept set forth in EPRI Report No. NP-4830 is to provide for consideration of the cask reinforced concrete bearing pad behaving as a pad on an elastic foundation. In previous structural reviews of cask systems, the bearing pad has been very conservatively assumed to be infinitely rigid. The response of the pad to a dropped or overturned cask has an influence on the magnitude of the force the spent fuel support system and confinement envelope must resist. The NRC identified various issues related to the details of the concept and its application by the applicant.

Rather than relying on the EPRI report, NRC independently calculated the stresses experienced by the MSB during a drop accident. Based on these independent calculations, NRC confirmed that the design of the MSB will provide an ample margin of safety during a drop accident. Therefore, NRC concluded that the design of the MSB was acceptable and that there was reasonable assurance that the confinement integrity will be maintained even if the postulated drop accident does occur.

In order to provide additional information on the application of the concept of an elastic bearing pad to spent-fuel casks, the NRC has initiated a contract to conduct drop tests of casks from heights in the 18 to 80 inch range. This should provide test data that would be used to assess the capability of the specific computational techniques contained in EPRI NP-4830 to predict the behavior of dropped casks. Following this testing, the NRC will consider the issue of the applicability of the EPRI report, including its applicability to a postulated drop of a steel cask on concrete pads.

9. Comment. The effect of a dynamic load factor (DLF) on the MSB was not considered nor was it shown to be insignificant.

Response. The effect of a DLF was considered and found to be significant. The applicant applied a maximum possible DLF of 2.0 to the average decelerations acting on the MSB. As a result of using a DLF of 2.0, the decelerations were increased from 62 g to 124 g and 22 g to 44 g respectively, for the vertical and horizontal orientations. As noted above in comment response number 8, although NRC staff did not endorse the methods used by the vendor to determine these loads, the NRC independently concluded that these design loadings are acceptable.

10. Comment. One commenter provided a calculation of the results of a hypothetical accident involving a VSC-24 cask. The conditions of the hypothetical accident were a cask tipover while the cask was under maximum internal pressure. The results indicated that the welds of the MSB would be overstressed.

Response. The NRC reviewed this calculation and based on that review, concluded the calculation did not state the consequences of the hypothetical accident. Most importantly, the size and configuration of the welds assumed in the calculation understated the strength of the welds and their ability to withstand the hypothetical event. The strength of these welds, which meet ASME Boiler and Pressure Vessel Code criteria, has been thoroughly analyzed by the applicant and the NRC. Although a cask tipover was not specifically performed for the VSC-24 cask, a horizontal drop accident, more severe than a tipover, was analyzed as a bounding case. This analysis demonstrated that, under the conditions of a horizontal drop while the MSB is under maximum internal pressure, the welds would not be overstressed.

B. A number of commenters raised issues relating to releases of radioactivity from surface contamination and leakage from the casks under normal and accident conditions.

11. Comment. Some commenters expressed concern that there would be a small release of radioactive particulates from the MSB exterior surface during off-normal conditions and that the radioactive releases from storage casks, when combined with other releases from the reactor, would exceed dose limits at the reactor site boundary.

Response. The NRC interprets this comment to mean that during off-normal conditions there is the potential for release of radioactive contamination from the exterior surface of the MSB. The consequences of any release of contamination from the MSB exterior surface (whether normal or off-normal) is evaluated in the SAR. However, the Certificate of Compliance, in

Section 1.2.5. "Maximum MSB Removable Surface Contamination" contains specifications for limiting the amount of radioactive contamination permitted on the external surface of the MSB. These specifications are conservative, and are based, in part, on equivalent criteria used for the safe transportation of radioactive material [see 10 CFR 71.87(i)]. Hence, compliance with them will ensure that off-site dose limits of the NRC's regulations will be met for normal and off-normal conditions alike. The general licensee must also use the cask in accordance with the reactor operating license and the Certificate of Compliance. The general licensee is also responsible for complying with other Commission regulations regarding radioactivity release limits. Therefore, potential releases from the MSB when combined with routine releases from the reactor should not exceed dose limits at the site boundary.

12. Comment. Commenters indicated that casks placed close to the shore of Lake Michigan represent a serious threat to the environment, especially to the Great Lakes which have 20 percent of the world's surface fresh water.

Response. A utility's use of the VSC-24, for the storage of spent fuel in casks at a reactor site, would not have a significant impact on the environment. This finding is supported by the NRC safety and environmental evaluations for the VSC-24 cask, including the applicant's demonstration of compliance of the cask with NRC requirements, as well as by the 1990 rulemaking on dry cask storage and the 1984 and 1989 waste confidence proceedings. While the VSC-24 cask is being approved for use under a general license, it can only be used by a licensee provided the reactor site parameters (e.g., average ambient temperature, seismic accelerations, flood water velocity, fires and explosions, etc.), are enveloped by the cask design basis, as specified in the SAR and SER. Proper use of a certified storage cask at any site (whether near Lake Michigan, a river, a bay, or an ocean) with site parameters that are bounded by the cask design, would not have a significant impact on the environment.

13. Comment. Some commenters expressed concern that extremes in temperatures and humidity would cause dry casks to leak.

Response. The VSC-24 cask design was analyzed for possible effects of extremes in temperature and humidity. These analyses showed no leakage will occur as a result of temperature or humidity extremes. The thermal analysis presented in the SAR and the NRC evaluation documented in Section 4.0 of the SER considered temperature extremes for both hot and cold conditions. Based on this analysis, the NRC concludes no breach of the MSB confinement barrier or leakage from the MSB will occur.

14. Comment. Some commenters speculated that a catastrophic release of radiation may occur from a possible explosion caused by spontaneously flammable uranium hydride in the presence of oxygen. It is postulated that the temperature inside the cask will be hot enough to rupture fuel rods which will, in turn, cause the presence of hydrogen to create uranium hydride.

Response. The NRC does not believe that an explosion inside a storage cask caused by flammable uranium hydride in the presence of oxygen is credible for the following reasons. Oxygen gas is not expected to be present because all casks are designed to have an inert atmosphere. Further, the formation of uranium hydride is not credible due to the lack of a significant source of hydrogen. Finally, all casks are designed so that the internal temperature will not cause the fuel rods to rupture. Therefore, the conditions necessary for this scenario to occur would not exist.

15. Comment. The SER states that there is no credible chain of events that could spread contamination from the MSB. Only air-coolant loss due to blockage was considered. Commenters indicated that the SER should also consider the effect of flooding of the hot cask and steam explosion. A concern was also expressed regarding the structural integrity of the pads which may, in the case of Palisades, be built on a sand dune area that shifts.

Response. The SER for the VSC-24 cask did consider the effects of flooding as well as air-coolant loss due to blockage of the vents. The analysis showed the release of contamination from the exterior surface of the MSB due to flooding is possible but the resultant contamination would not be significant. Steam explosions involving water contacting molten metal are not credible under dry spent fuel storage conditions. In addition, explosions due to steam forming under flooding conditions are not considered credible due to the fact that if steam were to be formed, it would be released non-violently through the vents.

With respect to the comment on structural integrity of the pads, the certificate of compliance requires, per 10 CFR 72.212(b), that written evaluations be performed by the licensee prior to cask use to establish that cask storage pads and areas have been designed to adequately support the static load of the stored casks. Consequently, the structural integrity of the pads would have to be evaluated and verified before the licensee could use the VSC-24 at the Palisades site or at any site.

16. Comment. A number of comments related to gaseous releases from dry storage casks. Commenters asked the following questions. What happens to gaseous components of the decay chain? Are they released to the environment? If not, is pressure buildup over time being considered? A commenter expressed

the opinion that casks should have individual radionuclide emission monitoring. An issue was raised about the effects of release of krypton-85 (Kr-85) gas on electric conditions in the atmosphere.

Response. The gaseous components of the decay chain are expected to be retained within the matrix of the spent fuel or within the fuel rod. In the case of pinhole leaks in the fuel rod cladding, the MSB is designed as a secondary confinement barrier to retain gaseous products. Therefore, because no gaseous components are released to the environment, no routine monitoring of effluent from the outlet vents is required. The primary reason for requiring the use of ASME Section III instead of other standards is to ensure the confinement of fission products. Pressure build-up of gaseous components in the MSB is not significant due to the age of the fuel and integrity of the fuel rod cladding; however, the MSB has been analyzed for a hypothetical condition in which all the fuel rods rupture. The resulting pressure within the MSB is negligible. The purpose of maintaining an inert atmosphere in the MSB cavity is to ensure that fuel rod cladding degradation does not occur, thereby preventing gross fuel rod cladding rupture. In addition to ensuring that new pin hole leaks do not develop in the fuel clad during the storage period, the licensee is responsible for monitoring the environment within the MSB prior to its opening to ensure that no unplanned release of radioactive material takes place. The amount of Kr-85 that could be potentially released from dry cask storage is so small that it would not significantly affect the physics or chemistry of the atmosphere.

C. A number of comments were received that focused on monitoring, surveillance, and inspection activities associated with dry cask storage of spent fuel, particularly as they relate to the VSC-24 cask.

17. Comment. Some commenters suggested that, with respect to the VSC-24 cask, the NRC did not enforce 10 CFR 72.122(h)(4) which reads, "Storage confinement systems must have the capability for continuous monitoring in a manner such that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions," and 10 CFR 72.122(i) and 10 CFR 72.128(a)(1) which require monitoring of systems and components that are important to safety over anticipated ranges of normal and off-normal operation. Also, one commenter suggested that because the VSC-24 cask requires surveillance to ensure that the vents are not blocked, the requirement that the cooling system must be a passive system (10 CFR 72.236(f)) is violated.

Response. NRC approval of the VSC-24 cask system is not inconsistent with 10 CFR 72.122(h)(4), 72.122(i) or 72.128(a)(1). Although the cited sections of 10 CFR Part 72, Subpart F, refer to "monitoring" or "continuous monitoring," they do not specify the details for particular monitoring programs to allow the NRC to require monitoring programs that are appropriate for the particular storage system design. The NRC has and will consider continuous monitoring where it believes continuous monitoring is needed to determine when corrective action needs to be taken. To date, under the general license, NRC has accepted continuous pressure monitoring of the inert helium atmosphere as an indicator of acceptable performance of mechanical closure seals for dry spent fuel storage casks.

The NRC does not consider such continuous monitoring for the VSC-24 cask double weld seals to be necessary because: (1) there are no known long-term degradation mechanisms which would cause the seal to fail within the design life of the MSB and (2) the possibility of corrosion has been included in the design (See SER Section 5.3.1). These conditions ensure that the internal helium atmosphere will remain. Therefore, an individual continuous monitoring device for each MSB is not necessary. However, the NRC considers that other forms of monitoring casks including periodic surveillance, inspection and survey requirements, and application of preexisting radiological environmental monitoring programs of Part 50 licensees during the period of use of the MSB canisters with seal weld closures can adequately satisfy the requirements of 10 CFR 72.122 (h)(4).

With respect to the issue of instrumentation and control systems to monitor systems which are important to safety (10 CFR 72.122 (i)), the user of the VSC-24 cask will, as provided in Chapter 14 of the SER and in Section 1.3.1 of the Certificate of Compliance, be required to verify by a temperature measurement, the cask thermal performance on a daily basis to identify conditions which threaten to approach cask design temperature criteria. The cask user will also be required to conduct a daily visual surveillance of the cask air inlets and outlets as required by Chapter 14 of the SER and Section 1.3.1 of the Certificate of Compliance.

While the MSB and VCC are considered components important to safety that comprise the VSC-24 cask design, they are not considered operating systems in the same sense as spent fuel pool cooling water systems or ventilation systems which may require other instrumentation and control systems to ensure proper functioning. Hence, due to this passive design, temperature monitoring and surveillance activities are appropriate and sufficient for this design, they assure adequate protection of the public health and safety, and meet the requirements of 72.122 (i).

18. Comment. Several commenters expressed concern related to the inlet and outlet vents, on the VSC-24 cask, which are necessary to allow cooling of the storage container by natural circulation. Some commenters also questioned the adequacy of the surveillance requirements for the VSC-24 cask and suggested that electronic continuous monitoring and recording of air outlet temperature should be required on each cask. Specific concerns include:

(a) vent blockage by bugs, webs, snow, and ice;

(b) frequency of vent outlet surveillance for blockage;

(c) drive-by or walk-through inspection is inadequate to observe outlet blockage; and

(d) critical temperatures associated with the VSC should be monitored.

Response. The NRC is requiring, as part of the VSC-24 Certificate of Compliance, that surveillance and measurement of the thermal performance of the cask be conducted by the licensee on a daily basis. The licensee is responsible for establishing the specific method of measurement; the licensee can measure the inlet and outlet air annulus temperatures, or it could also measure the MSB surface temperature, the VCC inner wall temperature or perform other appropriate measurements. The method selected by the licensee must provide a positive indication of the approach of materials to cask design temperature criteria.

In addition, analyses of safety margins of components important to safety show that even assuming surveillance were not conducted at the required daily frequency, and both the inlet and outlet vents were blocked for a 30 hour period, there would still be no loss of safety function or any immediate threat to the health and safety of the public. This conclusion is based on the adiabatic heatup thermal analysis of the VSC-24 cask, which assumes that all vents are blocked, and no heat is rejected by the cask. The concrete and cladding temperature criteria that could be exceeded under this conservative analysis, assuming complete blockage, signify the onset of very slow degradation mechanisms, not an imminent loss of safety function.

The NRC also agrees with the comment that visual surveillance of exterior air inlets and outlets may be inadequate and may not lead to a positive determination of blockage because the design includes screens placed over the vents to prevent wildlife from entering the VCC. Consequently, the NRC has revised the Certificate of Compliance surveillance requirement to make the integrity of the screens be part of the visual surveillance. A physical examination of the vent is required if its associated screen shows any evidence of breach.

19. Comment. One commenter suggested that approval of the VSC-24 cask should be denied because the snow shield was eliminated and that the analysis of air flow of the VSC took it into consideration.

Response. The snow shield was eliminated because it was not considered effective in resolving the problem of vent blockage by snow. A visual surveillance requirement is considered more effective in addressing the issue of vent blockage by snow. The Certificate of Compliance has been revised to add a daily surveillance requirement, as discussed in Comment 18, which would include checking for snow blockage during periods of snow accumulation. In addition the inclusion of a snow shield in the original design actually decreased air flow and therefore, its removal increases the thermal efficiency of the cask.

20. Comment. One commenter questioned how the condition of the inlet vents is checked for damage after the lifting arms are inserted into the air inlets for transfer.

Response. Lifting the VSC-24 cask using the hydraulic roller skid, which involves insertion of lifting arms into the air inlets, has been analyzed. The results indicate that the shear and bearing capacities of the concrete surrounding the air inlet vents [per American Concrete Institute (ACI) criteria 349-85] are not exceeded and no damage is expected. Therefore, there is no need to inspect vents for damage following use of the hydraulic roller skid.

21. Comment. The general licensee must have specific plans for the constant and careful monitoring of the casks and for the safeguarding of the waste to prevent catastrophic accidents or terrorism.

Response. In accordance with 10 CFR 72.212(b)(5), each reactor licensee must have a physical security organization and program to detect intrusion into the protected area including acts of terrorism, and to take any corrective action. The physical security program, as well as environmental monitoring and radiation protection programs for each reactor facility, provide the necessary monitoring for the casks and safeguarding of the spent fuel. Thus, the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions to protect the public health and safety. (Also see response to Comment Number 33 below).

D. A number of commenters raised technical issues related to the thermal analysis of the VSC-24 cask and thermal performance of the cask under normal, off-normal, and accident conditions.

22. Comment. One commenter questioned whether NRC intends to establish 75° F as a standard ambient temperature criteria for all storage casks and expressed concern that this temperature may not be applicable for the majority of power reactor sites.

Response. The NRC does not intend to establish 75° F, or other standard ambient condition criterion, for all cask designs. The cask vendor establishes ambient temperature criteria on which the cask is designed. In the case of the VSC-24 cask, PSNA chose 75° F. Each reactor licensee can then only use those casks which have design bases that envelop the reactor site ambient temperatures. For example, if a power reactor site has an average annual ambient temperature greater than 75° F, then that reactor licensee cannot use a cask with a 75° F ambient design temperature.

23. Comment. One commenter questioned how heat transfer for the VSC-24 cask is affected by the fact that there are no provisions for centering the MSB inside the VCC.

Response. Heat transfer for the VSC-24 cask is not significantly affected by lack of centering of the MSB inside the VCC. Therefore, no precise centering of the MSB inside the VCC is needed. However, the physical arrangement of the system restricts lateral movement and does not allow the MSB to be far from center as it is lowered into the VCC.

24. Comment. One commenter raised the concern that the VCC concrete temperatures do not comply with the ACI-349 temperature criteria.

Response. The NRC has accepted deviations from the ACI-349 Code, Appendix A.4 for the concrete temperature criteria. However, while accepting the deviation, the NRC has identified a specified maximum thermal expansion coefficient for fine and coarse aggregates in the concrete which allows operation at higher temperatures. The selection of specific fine and coarse aggregates in the concrete prevents microcracking between the cement and aggregates in the anticipated temperature range of the VCC. Thus, deviation from the ACI-349 temperature criteria is not a cause for concern and does not compromise safety.

25. *Comment*. One commenter claimed that NRC has used the unsupported assumption that 48 hours is sufficient time to reach thermal equilibrium for the irradiated fuel assemblies (high level radioactive waste) that have been removed from water storage and sealed in the metal canister.

Response. The commenter refers to the time period allowed for a loaded VSC-24 cask system to reach thermal equilibrium conditions. For the purpose of thermal equilibrium, the VSC-24 cask system is considered to be placed in service when the concrete cask cover plate is installed.

It should be noted that the Certificate of Compliance has been changed to require that the inlet and outlet air temperatures, for all VSCs placed in service, be measured until the cask reaches initial thermal equilibrium. Furthermore, a daily measurement of the thermal performance of the VSC-24 cask is required. Therefore, any reference to assumed 48 hour thermal equilibrium is covered by the enhanced surveillance requirements. The 48 hour period was selected to provide a basis for baseline measurements. There is no safety significance if thermal equilibrium is achieved in a shorter or longer time.

26. Comment. One commenter noted that in Chapter 9 of the SER, the NRC staff found it necessary to impose a pre-operational test to verify the heat removal capacity of the VSC-24 cask system. The commenter claimed that this was required because predicted fuel clad temperatures are a "mere" 4° F below their design criteria on a 75° F ambient day. It was further asserted that with a predicted fuel clad temperature of 4° F below design criteria for the off-normal condition limit, even a successful pre-operational test would not assure that the design criteria is met within the bounds of statistical uncertainty, particularly since the calibration of their temperature sensing equipment has a tolerance of plus or minus 1° F.

Response. The NRC has imposed a test to benchmark the heat removal capacity for the first VSC-24 cask placed in-service. However, the 4° F margin stated on page 9-4 of Chapter 9 of the SER cited by the commenter, is a typographical error. The correct margin is 24° F, as stated on page 4-7 of the SER. This 24° F margin is the difference between the maximum allowable fuel clad temperature and the calculated fuel clad temperature, assuming an average annual ambient temperature of 75° F for normal continuous conditions. For off-normal conditions involving higher ambient temperatures, a maximum fuel clad temperature of 708° F was calculated assuming an ambient temperature of 100° F. This temperature is 4° F below an acceptable fuel clad temperature criterion of 712° F. The NRC accepted this margin on the basis of the following conservative factors applied in the off-normal case analyzed in the SAR:

a. The calculation assumes steady state conditions. It would take several days of sustained 100° F ambient temperature to approach the calculated fuel clad temperature value of 708° F.

b. The fuel temperature criterion is based on prevention of fuel failures due to long-term degradation mechanisms. Short term variations in the average temperature, such as when the daily summer average temperatures exceed 85° F, have no effect on the long term degradation mechanisms that affect the fuel cladding. Therefore, the annual average 75° F temperature would be a more realistic condition to use in the calculation than the 100° F temperature actually used in the calculation.

c. Heat conduction in the axial direction is treated conservatively because little credit is taken for heat transfer out of the ends of the MSB canister.

d. Fuel clad temperature is treated conservatively because a peak heat generation rate rather than an average was used in the calculation.

These conservative factors used in the calculation of fuel clad temperatures provide reasonable assurance that the actual temperature will be lower than the calculated temperature, considering uncertainties, and therefore this 4° F margin below the fuel clad temperature criterion is acceptable.

27. Comment. One commenter questioned whether cladding failures would affect the temperature of the MSB or the VCC and the heat removal capacity of the VSC-24 cask. Another asked why helium was used to fill the cask. The only helium cooled reactor in the country, Ft. St. Vrain, was operational merely 15% of the time.

Response. Fuel cladding failure is not expected to occur because the VSC-24 cask is designed to maintain an inert helium atmosphere inside the MSB to prevent fuel cladding failure. However, fuel cladding failure would neither affect the temperature of the MSB or VCC nor affect the heat removal capacity of the VSC-24 cask. The temperature of the MSB and the VCC depends on the heat generated by the fuel in the MSB, which is not affected by a fuel cladding failure. In addition, heat removal capacity of the VSC-24 cask depends on the airflow on the outside of the MSB which also is unaffected by fuel conditions inside the MSB. Helium was chosen because it is inert and it has good heat transfer characteristics. The fact that the Ft. St. Vrain reactor used helium as a coolant did not contribute to its operational problems.

28. Comment. One commenter wanted clarification of "approximately 24 kW," when referring to the heat source loaded into the first MSB for tests conducted by the licensee to verify heat removal capacity of the VSC system. The commenter also indicated that the Certificate of Compliance is overly restrictive in requiring a 24 kW heat load for the first cask because some reactors do not have spent fuel assemblies which could make up the 24 kW heat load. The commenter recommended that the requirement be changed to require that the first cask be loaded with a heat load as high as practicable (but not to exceed 24 kW) to verify the calculated heat removal capability. Another commenter asked why not test the cask with artificial thermal loads rather than with spent fuel.

Response. The intent of the language, "approximately 24 kW" was to provide some flexibility to a potential user because there is no way to ensure that the first fuel placed in the cask will have a heat load of exactly 24 kW that was used in the thermal analysis. The purpose of the test is to measure the cask performance and establish baseline data. Following loading and temperature testing of the cask with a 24 kW loading, the licensee would be able to load fuel at lower thermal ratings without the need to provide NRC with separate temperature test data and additional analysis since the 24 kW heat loading is a bounding analysis. However, because the cask vendor has not provided thermal analyses at lower heat loadings, the NRC believes that if a licensee's first fuel loading has a heat load less than 24 kW, the licensee should conduct both a temperature measurement and a thermal analysis. The purpose of conducting both the analysis and the measurement is to measure system performance and to establish baseline data for the expected inlet and outlet temperature difference. The Certificate of Compliance has been revised to this effect and the word "approximately" has been deleted. With respect to the issue of artificial thermal loads, the NRC will accept alternate heat loads other than spent fuel and the Certificate of Compliance has been revised accordingly. A licensee could use such an artificial heat source to test an initial cask at a bounding heat load of 24 kW prior to loading fuel.

29. Comment. One commenter noted that Page 4-1 of the SER for the VSC-24 cask states that the applicant will remove any cask from service which has inlets and outlets blocked. It should say "or" instead of "and."

Response. The statement refers to a proposal made by the applicant and is correct as quoted on page 4-1. However, the NRC did not accept this proposal because the applicant did not provide acceptable evidence that the cask will be adequately cooled in the event of a full blockage of either all inlets or outlets. Sections 1.3.1 and 1.3.4 of the Certificate of Compliance require that a VCC be removed from service whenever either all inlets or all outlets are found to have blockage for 24 hours and the concrete temperature criterion of 350° F has been exceeded. This conclusion is also stated on page 4-1 of the SER.

30. Comment. One commenter noted that Table 4.1-1 of the November 1991 SAR for the VSC-24 cask fails to state what the temperature difference would be if all inlets were blocked over a long-term.

Response. The commenter is correct. However, a temperature criterion of 350° F has been established for the concrete cask. Calculations indicate that a temperature of 350° F could be reached after 30 hours if either all inlets or all outlets are blocked. If this situation is identified, the licensee must demonstrate that accident temperature criteria have not been exceeded or is required to take the cask out of service.

NRC notes that reaching 350° F is not an unsafe condition with respect to the containment integrity of the MSB or the stored fuel. Rather it is a criterion for deciding whether to take the VCC out of service. This action is highly conservative, since only the onset of very slow degradation occurs if the concrete temperature reaches 350° F. As discussed below, in response to Comment Number 31, a conservative adiabatic heatup analysis determined that it would take 7 days to reach unacceptable fuel clad temperatures. The NRC considers that within this time frame, the licensee's enhanced daily surveillance program, which must include a component that verifies the thermal performance of the cask, would identify the blockage and allow sufficient time for necessary corrective actions to be taken.

31. Comment. One commenter indicated that the safety evaluation for the tipover of the VCC only considered the structural aspects of the accident and ignored the thermal consequences. The issue raised was that the VSC-24 cask uniquely requires a vertical orientation to adequately remove heat and that heat removal in the horizontal configuration is degraded even if all vents are unblocked which should not be assumed.

Response. Thermal consequences of a VSC-24 cask tipover were considered and are bounded by the adiabatic heat-up analysis performed for the cask. Adiabatic heat-up is not affected by orientation, either horizontal or vertical. The adiabatic analysis determined that it would take approximately seven days to reach unacceptable fuel clad temperatures. The NRC considers that within this timeframe the licensee would take necessary corrective actions to return the cask to an upright position.

32. Comment. One commenter stated that an analysis based on Diffusion Controlled Cavity Growth (DCCG) has been the only method accepted by the NRC to determine the maximum allowable fuel cladding temperature. The commenter further stated that it was not apparent that an analysis based on DCCG had been performed in evaluating maximum cladding temperature for the VSC-24 cask.

Response. The NRC agrees that DCCG is the only current method acceptable to the NRC to determine maximum allowable fuel clad temperature. The VSC-24 cask was evaluated by this method. See Section 5.3.3 of the SER.

E. A number of commenters expressed concern about emergency planning and response to contingencies.

33. Comment. Some commenters expressed concern that no evacuation plan was required. They also stated that there is a lack of contingency planning for catastrophic events. They noted these events could include but would not be limited to:

a. Direct or indirect lightning strikes on the casks;

b. Plane crash into the casks;

c. Sabotage;

d. Earthquakes;

e. Fire; and

f. Emergency planning for cask malfunctions.

A commenter wanted the utility to notify either state or local government before loading casks to make sure local services were aware and would know how to respond if necessary under the emergency plan.

Response. The Code of Federal Regulation, 10 CFR Parts 50 and 72 requires that nuclear plant structures, systems and components important to safety shall be designed and appropriately protected against dynamic effects, including the effects of tornado-driven missiles, that may result from events and conditions outside the nuclear power unit. This includes the effects of possible airplane crashes.

The licensee's site evaluation for a nuclear plant also considers the effect of nearby transportation and military activities. A licensee proposing to use the VSC-24 cask is required to evaluate and verify that the SER for the facility encompasses the design basis analysis performed for the VSC-24 or any certified cask. Generally, a cask's inherent design will withstand tornado missiles and other design loads and thus, also provides protection against the collision forces imposed by light general aviation aircraft (i.e. 1500-2000 pounds) which constitute the majority of aircraft in operation today. NUREG-0800, Section 3.5.1.6 "Standard Review Plan for Light Water Reactors", contains methods and acceptance criteria for determining if the probability of an accident involving larger aircraft (both military and civilian) exceeds the acceptable criterion. It is incumbent upon the licensee to determine whether or not the reactor site parameters are enveloped by the cask design basis as required by 10 CFR 72.212(b)(3). This would include an evaluation demonstrating that the requirements of Section 72.106 have been met.

NRC reviewed potential issues related to possible radiological sabotage of storage casks at reactor site independent spent fuel storage installations (ISFSIs) in the 1990 rulemaking that added Subparts K and L to 10 CFR Part 72 (55 FR 29181). NRC regulations in 10 CFR Part 72 establish physical protection and security requirements for an ISFSI located within the owner controlled area of a licensed power reactor site. Section 72.212(b)(5) requires that the spent fuel in the ISFSI be protected against the design basis threat for radiological sabotage using provisions and requirements comparable to those applicable for other spent fuel at the associated reactor subject to certain additional conditions and exceptions described in 10 CFR 72.212. Each utility licensed to have an ISFSI at its reactor site is required to develop security plans and install a security system that provides high assurance against unauthorized activities which could constitute an unreasonable risk to the public health and safety. The security systems at an ISFSI and its associated reactor are similar in design features to ensure the detection and assessment of unauthorized activities. All alarm annunciations at the ISFSI are monitored by the security alarm stations at the reactor site. Response to intrusion is required. Each ISFSI is periodically inspected by NRC and annually audited by the licensee to ensure that the security systems are operating within their design limits. The validity of the threat is continually reviewed, with a formal evaluation every six months by the NRC.

An adequate evacuation plan exists for the use of certified casks because of the fact that the existing reactor emergency plan covers the entire site. In addition, contingency planning for the events described above exists because these events are covered within the emergency plans of the reactor facilities which will use the cask. In accordance with 10 CFR 72.212(b), the reactor licensee must review the emergency plan to ensure it provides adequate protection. The licensee's emergency plan provides for responsive action if an event has happened which has the possibility of creating an emergency or after an actual emergency has occurred. Through communications between the utility and governments, the contents of the emergency plan and the actions to be executed by each entity for various situations are understood. In addition, the utility is required to conduct a periodic emergency exercise involving the utility and government agency staff.

34. Comment. One commenter stated that there was no contingency for accidents except to reload the spent fuel back into the cooling pool which may not be possible due to lack of pool storage space or impact on the spent fuel due to the accident.

Response. Because of the design features, as well as the procedures and requirements discussed elsewhere in this response and the associated safety analysis, the likelihood of an accident occurring which will require removal of the spent fuel from the cask is very small. However, even if such an unlikely accident occurs, the cask design is required to have capability to permit retrieval. (10 CFR 72.122(l)). NRC does not require a licensee to maintain a reserve capability in the spent fuel pool. Many licensees may do so, however, and they would, therefore, have the option of returning the fuel to the pool in the unlikely event of an accident requiring removal of fuel from the cask. In addition, licensees will have other options available to cover this unlikely contingency including temporary storage in a spare storage cask or use of an existing certified transportation cask. Licensees would have to consider these, and other available options, in the unlikely event an accident occurs requiring removal of the fuel.

F. Other comments which do not specifically fit those categories above follow below. These comments deal with a broad range of other technical and procedural issues.

35. Comment. There are outstanding safety issues that the NRC expects to resolve in the first test.

Response. The NRC SER addresses all significant safety issues, and there are no outstanding safety issues about the VSC-24 cask that remain unresolved. Accordingly, the first test does not involve any safety issue. Its purpose, rather, is to benchmark the heat removal capability of the VSC-24 cask.

36. Comment. One commenter asked that a requirement to submit a report to the NRC within 15 days of the test and evaluation of the first cask and prior to construction of the second cask be added to the VSC-24 cask Certificate of Compliance. Also the report and subsequent NRC review should be placed in NRC's Public Document Room.

Response. A letter report summarizing the results of the thermal test and evaluation of the first cask placed in service will be submitted to the NRC and placed in the Public Document Room. The licensee may, at their own financial risk, fabricate additional casks prior to using the first cask. If the first cask does not perform as specified, the NRC would prevent use of the other casks or modify conditions on how they could be used.

37. Comment. It is unacceptable from a public health and safety standpoint to conduct the first full scale test of a VSC-24 cask at a reactor site because it places the power plant workers, the public, and the

environment at risk. Two commenters stated that the VSC-24 had not been tested to the full range of climatic conditions.

<u>Response</u>. Although the volume of data that is available to support certification of the VSC-24 cask does not include results of full scale tests, the available data is more than sufficient to show that the use of the VSC-24 cask by a licensee will not place power plant workers, the public, or the environment at any undue risk. Also the conditions of use for the VSC-24 cask in the Certificate of Compliance ensure adequate protection of the workers, the public, and the environment. Further, the VSC-24 cask has been designed and will be fabricated to well established criteria of the ASME B&PV and ACI codes. In addition, it uses construction materials which have well known and documented properties to provide the necessary structural strength and radiation shielding to meet regulatory requirements. While the NRC has not relied on testing of the VSC-17 cask (a smaller version of the VSC-24 cask design) for approval of the VSC-24 cask, the VSC-17 cask has been tested by DOE at its Idaho National Engineering Laboratory. The report "Performance Testing and Analysis of the VSC-17 Ventilated Concrete Cask," EPRI TR-100305, dated May 1992, concluded that the VSC-17 Cask can be safely used at reactor sites. While the VSC-24 cask approval does not rely on the VSC-17 cask, the designs are similar and many parallels in design and function can be drawn. DOE testing of the VSC-17 demonstrates that ventilated storage cask technology can provide safe storage of spent fuel. Thus, in view of the above, although the commenter's observation that the VSC-24 had not been fully tested under climatic conditions is technically correct, the cask has been designed for ambient temperature extremes from -40°F to +100°F and meets the ASME and ACI requirements.

38. Comment. One commenter noted that Consumers Power does not have a plan to remove spent fuel stored under general license from the reactor site as required by 10 CFR 72.218.

Response. The licensee is not required to have a plan to remove spent fuel stored on site under the general license until an application to terminate the reactor operating license is submitted to the NRC. This requirement is found in 10 CFR 72.218(b) and 10 CFR 50.54(bb).

39. Comment. One commenter noted that the NRC does not specifically require inspections against 10 CFR 72.236(j)-(m). Questions were raised regarding quality assurance problems encountered during the inspection of systems currently in operation, and during the construction of the first five casks, that are expected to be placed in service. Another question was raised pointing out that the vendor did not use weld inspectors qualified/certified to American Weld Society D.1.1.

Response. The NRC ensures compliance with 10 CFR 72.236(j) and (k) through inspections, and ensures compliance with 10 CFR 72.236(l) and (m) through the cask approval process. This process will identify different areas that may need correction, but that is the purpose of an inspection program. If a violation of the requirements is detected, the NRC can impose penalties, or even stop work. The NRC takes note of the fact that problems noted by the commenters were identified as a result of NRC's inspection program during the construction of specific casks. This experience reemphasizes the need for close and continuing quality surveillance under vendor and user QA programs during all VSC-24 and other cask construction activities. The NRC will continue to conduct the inspections of construction activities in accordance with NRC's Inspection Procedures in conjunction with vendor's quality assurance (QA) program, specifications, drawings, etc. to ensure quality work. As to the specific point of the qualification of welds and inspectors, the NRC notes that the welds referenced were not structural welds and, as allowed by the vendor's fabrication specifications, do not have to be qualified to the same extent as a structural weld.

40. Comment. Concern was expressed that the measurement of actual effectiveness of a technology in delivering stated requirements must be demonstrated empirically, and that the NRC has not demonstrated the goal of this technology, defined acceptance criteria, or specified how compliance is demonstrated. Some commenters also expressed concern that the review of the concrete cask was not done at the same level as that performed for metal casks and that no independent computer analyses were performed for the design event review. Some commenters noted that the review requires more than limited computer models.

Response. For the issue of acceptance criteria, the NRC has established specific requirements in 10 CFR Part 72 that must be met in order to obtain a Certificate of Compliance for a cask. The details of the review and bases for the NRC concluding that the cask meets the requirements of 10 CFR Part 72 are provided in the SER. The goal of dry cask storage technology is to store spent fuel safely. That goal, and the effectiveness of the technology, previously has been demonstrated empirically and experimentally. Different cask designs may require different types of analysis to demonstrate their safety, and therefore different review methods may be appropriate to reach that conclusion. In each case the level of review performed is that needed to provide assurance of adequate protection of the public health and safety.

41. Comment. Some commenters claimed that Part 72, Subpart K was originally intended to apply to metal casks only. Concrete cask systems were not addressed in the original rulemaking.

Response. As discussed below, both the language and history of Subpart K show that it applies to any NRC-approved dry cask storage system including concrete casks systems, and commenters are therefore mistaken in their view that it was intended for metal casks only.

Subpart K applies "to casks approved under the provisions of this part" which includes casks approved by NRC under 10 CFR Part 72, Subpart L. Subpart L contains NRC's approval conditions "for NRC spent fuel storage casks designs" which would include concrete casks. None of the approval conditions in Subpart L requires that the cask must use a metal cask design.

Additionally, there is information on concrete storage technologies in the Subpart K rulemaking record that would not support limiting it only to metal casks. Specifically, the Commission's notice of proposed rulemaking (NPRM) for Subpart K referenced the Canadians' use of "concrete casks called silos" in describing "the knowledge and experience of dry spent fuel storage in concrete casks." 54 FR 19379-80 (May 5, 1989). The proposed rule also referenced DOE's demonstration of dry storage in sealed storage casks (SSC) which it described as "an above-ground, steel-lined, reinforced concrete cylinder or cask." Id. Further, it cited experience gained from spent fuel storage "in stainless steel canisters stored inside concrete modules at the H.B. Robinson 2 site. . . " Id. If the Commission had intended to limit Subpart K to metal casks, it would not have included data from other dry storage technologies in the record supporting its action.

Although the Commission has not previously approved concrete storage systems (or casks) under Subpart L, it expressly noted such systems might be approved (and thereby included in Subpart K) in the future. In particular, the Commission gave the following explanation for not approving certain concrete module designs in the final Subpart K rule:

> A major reason that these spent fuel storage systems [e.g., NUHOMS; Modular Vault Dry Store], which are being considered by the Commission for use under a general license, are not being approved at this time is that they have components that are dependent on site-specific parameters and, thus, require site-specific approvals. 55 FR 29181 (July 18, 1990).

Moreover, the NPRM included the statement that "[t]he Commission has evaluated and approved, in specific licenses issued under 10 CFR Part 72, other types of dry storage modules [and t]hese methods may be approved in the future for use under a general license." 54 FR 19382. It also noted that "[s]torage casks certified in the future will be routinely added to the listing in § 72.214 through rulemaking procedures." 54 FR 19380.

These statements collectively show the Commission specifically envisioned the possibility of future rulemaking (i.e., the procedure NRC is now using) to add concrete storage systems to the list of approved spent fuel storage casks in Subpart K. Consequently, concrete storage systems _can_ be "casks approved under the provisions of this part" for purposes of Part 72, Subpart K if, for example, they are _not_ dependent on site-specific parameters and therefore do _not_ require site-specific approvals and if they conform to the approval conditions of Subpart L.

Finally, it is noteworthy that the Commission adopted Subparts K and L for the express purpose of implementing certain interim storage provisions of the Nuclear Waste Policy Act of 1982 that, significantly, are not limited to metal casks. 54 FR 19379 (May 5, 1989). In particular, the Act authorized the Commission to approve by rule "one or more [storage] technologies" for use at reactor sites. Sec. 218(a)(42 USC § 10198(a)). The Act also directed the Commission to establish procedures for the licensing of "any technology" approved by the Commission under Section 218. (Sec. 133 (42 USC § 10153)). Therefore, because the Act's provisions are not limited only to metal storage cask designs, it would be inconsistent with the Commission's purpose to limit the application of Subparts K and L to such designs.

42. Comment. One commenter requested the proceeding be stopped until the NRC revises all regulatory requirements pertaining to the storage of high-level waste and spent fuel to require testing procedures which include testing to destruction.

Response. The NRC does not require testing to destruction or other tests if we have confidence in the analyses which are done or if the design relies on nationally recognized codes and standards. Testing to destruction is an option that can be used to confirm design adequacy. However, destructive tests of an entire cask are not necessary to evaluate a design when other non-destructive tests or destructive testing of the components will provide the necessary information to evaluate a design.

43. Comment. Some commenters expressed concerns that fuel handling could be under less than ideal conditions and that storage could be under harsh environmental conditions. Sites where the VSC-24 cask is proposed for use would experience low winter temperatures, freeze-thaw cycles, high

humidity, and marine conditions. Concern was also expressed that harsh environmental conditions and damage to the MSB protective coating will degrade the containers as a result of corrosion, embrittlement, cracks, fatigue and other aging effects which would affect the ability of the cask to survive over extended periods.

Response. Handling of fuel and loading of the cask is performed under well controlled conditions in the reactor's fuel handling building using written procedures developed in accordance with the reactor operating license. The VSC-24 system has been evaluated for the possible effects of harsh environmental conditions and the MSB has been evaluated for the possible effects of corrosion due to humid and marine environmental conditions. As a result of the corrosion analysis of the MSB, the NRC found the design acceptable with the consideration of localized corrosion mechanisms (i.e., pitting, stress corrosion cracking, crevice corrosion and galvanic corrosion) as well as general corrosion. Localized corrosive attack on the MSB surfaces is minimized by choice of materials and design features such as the ceramic tiles between the VCC liner and the bottom surface of the MSB. Furthermore, the NRC allows no credit for the attributes of the paint.

Aging issues attributed to fatigue for the MSB were evaluated according to the ASME B&PV Code, Section III, and it met acceptable standards.

Temperature extremes, such as freeze-thaw cycles which exist in the Great Lakes region, were considered in the evaluation of the VSC-24 cask. According to the conditions for cask use, the user of the VSC-24 system will perform site-specific analyses to verify that the temperature conditions assumed in the analysis bound the conditions existing at the site.

The possibility of MTC and MSB cracks was addressed as a part of ferritic material considerations. Based on guidance provided in ANSI N14.6 and NUREG CR-1815 the NRC established test and operating limits for the MTC and the MSB to preclude the possibility of brittle fracture.

Finally, the VCC is designed and fabricated to American Concrete Institute Code requirements which consider durability under extreme conditions for extended periods. The cask is also subject to annual visual surface inspections for chipping, spalling, or other surface defects. Any surface defects found can be easily corrected. The fluence of the neutron flux within the spent fuel is five orders of magnitude less than the fluence encountered within an operating reactor, and therefore embrittlement of the MSB is not of concern.

44. Comment. A commenter asked how the NRC will correct the problem when something goes wrong with the VSC-24 cask. In the event of a tipover or drop of a loaded VCC, the commenter believes the licensee should be required to report the incident to the NRC within 4 hours and the NRC, rather than the licensee, should determine whether the MSB and/or the VCC should be reloaded for spent fuel storage.

Response. The licensee is responsible for correcting problems when they occur. The NRC is responsible for ensuring that the licensee takes appropriate corrective action. These rules reflect existing regulatory practice and procedure. The regulations and Certificate of Compliance identify specific events and conditions where the licensee would have to notify the NRC.

In accordance with 10 CFR 72.216(a) the licensee is required to report cases involving any defect as a result of a tipover or a drop to the NRC within 4 hours. The licensee would also have to inspect and evaluate the MSB after any tipover or drop of 18 inches or higher. Based on that evaluation, the licensee, not the NRC, would be responsible for determining continued use of that cask. NRC's responsibility is to monitor and oversee the licensee's activities. NRC has, however, the authority to order the licensee to cease use of a cask, if that were determined to be necessary.

45. Comment. One commenter stated that the double seal welds at the top of the MSB do not comply with the ASME Code, Section III, Subsection NC.

Response. The double seal welds at the top of the MSB meet all of the ASME requirements except the volumetric inspection requirement. This inspection is not possible due to the presence of the radioactive fuel loaded into the MSB. However, an additional margin of safety is provided because: (1) the welded joint is a double weld; (2) the weld joint has been analyzed according to ASME Section III criteria for all load conditions including accidental drop; (3) the pressure inside the canister during normal storage operations is approximately atmospheric, resulting in very low stress intensities; and (4) the confinement integrity is established by ASME code test procedures, which include dye penetrant testing of the root and cover welds of both the inner and outer welds. In addition, the NRC is requiring testing for helium leaks prior to the placing of the MSB in storage.

46. Comment. A number of commenters questioned the lack of transportability of casks and the apparent noncompliance with the requirement of 10 CFR 72.236(m). Several commenters expressed concern that the VSC-24 cask is not compatible with transportation requirements. Several commenters questioned how the spent fuel will be transported to a Federal Repository and what will be the additional handling cost.

Response. These casks are currently approved for storage of spent fuel, not off-site transportation. Therefore, there is no need for the VSC-24 cask to be compatible with transportation requirements. These casks are only moved between the fuel handling building and the storage pad at the site where the fuel will be stored. Although 10 CFR 72.236(m) states, "To the extent practicable in the design of storage casks, consideration should be given to compatibility with removal of the stored spent fuel from a reactor site, transportation, and ultimate disposition by the Department of Energy," there is no requirement that the storage cask itself be transportable off site. If the cask vendor wants to have its cask used for the transportation of spent fuel, it would have to obtain a transportation Certificate of Compliance issued by the NRC under 10 CFR Part 71.

The mechanism for transporting the spent fuel from a reactor site to a Federal Repository is unknown at this time. However, it could be by truck, rail, barge, or some combination. Also, the handling costs are unknown since DOE compatibility requirements are not known and regulatory requirements at the time of transfer could be different.

47. Comment. One commenter pointed out the NRC indicates that the analyses presented in the SAR are "based on non-consolidated, zircaloy-clad fuel with no cladding failures." Please clarify whether there exists an inconsistency between "no cladding failures" and the language which the NRC uses in Table 1-1, Characteristics of Spent Fuel to be Stored in the VSC-24 System, referring to Fuel Cladding as: "Zircaloy clad fuel with no known or suspected gross cladding failures."

Response. The NRC agrees that there is an inconsistency. Acceptability is based on zircaloy clad fuel with no known or suspected gross cladding failures. Section 1.2.1 of the Certificate of Compliance has been revised to "specify no known or suspected gross cladding failures." The intent of this specification is to rely on the cladding to safely confine the UO_2 fuel material within the rods to preclude operational safety problems during its removal from storage. Fuel cladding with pin hole leaks is still capable of confining the fuel and therefore is acceptable for storage. In addition the inert atmosphere and fuel clad initial temperatures provide assurance that the cladding will be protected during storage against degradation that leads to gross rupture.

48. Comment. Commenters stated that there is no evidence that PSN considered the effects of worst case tolerance combinations in the structural analysis.

Response. There are several generic areas where improper tolerance combinations could jeopardize the structural integrity of a design. These areas are: (1) Over-tolerance of weight which could result in unallowable stress levels for some components; (2) improper tolerances for dynamic parts such as in machinery which could result in interference and failure; (3) improper tolerance for fuel positioning in the basket; (4) improper tolerances of parts of an assembly which could lead to induced stresses from an interference fit or the converse situation, i.e., loose tolerances which could lead to an ill-defined load path; and (5) improper tolerances which might cause a heat conduction path to exist or not exist.

The NRC has reviewed and verified that tolerances specified in the application would prohibit a weight which is above the load used in the calculation package. The NRC also reviewed specified dimensioning, which, when followed as required, will prohibit interference and failure of dynamic parts such as machinery or fuel positioning in the basket. The NRC reviewed the vendor's calculations to assure that the loads which were analyzed and heat conduction paths account for the range of tolerances. For these reasons, the NRC has concluded that tolerance combinations are adequately addressed for the vendor's structural and thermal analysis.

49. Comment. A commenter indicated that the VSC-24 was exempted from established cladding temperature criteria for short term normal condition events, in which the maximum fuel cladding temperature limit is exceeded by as much as 170° F.

Response. The VSC-24 has not been exempted from a short term temperature limit for fuel cladding. In comparing the short-term and long-term thermal hydraulic evaluation shown in Table 4.1-1 of the SAR, the short-term temperature will exceed the long-term temperature by as much as 170° F. This higher temperature, however, is acceptable during the short-term while the fuel is dried prior to filling the MSB with an inert gas (helium), weld sealing the MSB, and final placement of the MSB in the cask for interim storage. The NRC conservatively assumed that air was present during the drain-down and dry-out periods and calculated the oxidation rate. The maximum length of fuel oxidation for defective fuel was determined. The cladding strain was estimated to be less than 1 percent. Therefore no defect extension or fuel powdering is anticipated. The short term increased temperature is desirable to ensure removal of moisture. Following dry-out and helium introduction, the temperature will drop below the long term limit.

50. Comment. Some of the commenters indicated that the SER for the VSC-24 cask allows k_{eff} of 0.98 and that this deviates from the normally accepted limit of 0.95 specified in NRC Regulatory Guide 1.13, Proposed

Revision 2, "Spent Fuel Storage Facility Design Basis." The commenter indicated that NRC should allow other vendors to modify their cask to k_{eff} of 0.98. One commenter expressed concern that the benchmark experiments that were cited in the analysis dated to the 1970's and because of their age were considered inappropriate for use, and commented that there was a difference in the geometry between the benchmark calculations and the VSC-24.

Response. The k_{eff} of 0.95 is guidance and is thus, not a requirement. As such, a licensee has flexibility and may propose an alternative limit. Based upon NRC review, NRC accepted the licensee's proposed use of a k_{eff} of 0.98 for the accident case of misloading the MSB with all fresh fuel of maximum enrichment and optimum moderation conditions. This accident condition borders on the incredible since it requires a mutually exclusive condition: that is, 24 unirradiated fuel assemblies that have heat generation rates sufficient to produce enough boiling for optimum moderation. Therefore, NRC would accept a k_{eff} of 0.98 for any cask generically for this accident case, but a k_{eff} of 0.95 would apply otherwise. The conditions of nuclear criticality, and the experiments that provide that information can be and have been measured with a high degree of accuracy, since the 1940's. The age of the data is not significant. It is desirable that the benchmark experiments represent the system under evaluation as closely as possible. The features or parameters that are important to this purpose are the fuel composition and enrichment, the geometry of the fuel assembly, i.e., rod diameter and pitch, cladding type, and any neutron absorbers in the vicinity of the fuel pins. These parameters must be properly considered in the processing of nuclear cross sections used in criticality analysis so that the benchmark experiments are used to determine a method bias, or systematic error that may result from the particular set of nuclear cross section data that are used, or from the methods used to process the cross section data. Once method bias is determined for the particular fuel parameters, the calculations are quite insensitive to the macroscopic geometry of the system.

Therefore, it is not necessary that the gross or macroscopic geometry of the benchmark experiments be similar to the VSC design as long as the method bias has been determined for the appropriate fuel parameters. The B&W critical experiments have been widely used for this purpose since they were performed using light water reactor fuel assemblies similar to those used in many light water reactors.

51. Comment. One commenter indicated that the Certificate of Compliance for the VSC-24 cask is unnecessarily restrictive in requiring that the MSB contain 2850 ppm boron solution while it is being loaded. This concentration of boron would keep k_{eff} less than 0.95 even if all 24 storage spaces in the MSB were loaded with fuel assemblies which average 4.2 weight percent (wt.%) U_{235}. Some nuclear power plants do not have 4.2 wt.% U_{235} fuel on site. Therefore, there is no possibility of fuel containing that concentration of U_{235} being loaded in a MSB. The commenter recommended that the Certificate of Compliance requirement for boron concentration in the MSB cavity water be changed to allow other concentrations to be used such that the boron concentration used would maintain k_{eff} less than 0.95 even if fuel assemblies containing the highest wt.% U_{235} in the spent fuel pool were placed in the MSB.

Response. The NRC agrees that the boron specification in the Certificate of Compliance for the VSC-24 cask may be restrictive. The boron specification is consistent with the maximum allowable uranium enrichment

(4.2 wt.%), based on the criticality analysis presented in the SAR. The Certificate of Compliance specification for boron concentration in water is a bounding condition which was chosen to limit reliance on administrative controls to determine the proper required boron concentration for each cask loading. A method like that proposed by the commenter, to determine the boron concentration required, based on the maximum initial U_{235} enrichment of fuel at each reactor site, could be considered as a future amendment to the Certificate of Compliance.

52. Comment. Some commenters suggested that the NRC should consider limiting the cask storage time and expressed concern that cask storage could become permanent if the DOE might not accept fuel as they are required to do. Commenters also noted that the NRC requirement that cask viability be evaluated for "at least" 20 years, does not, in itself, guarantee safety in the apparently likely event the casks remain years or decades beyond the original intended duration.

Response. By approval of the Certificate of Compliance, the NRC has limited the cask storage time to 20 years. After the 20-year period, the certificate can be renewed, with each renewal period not to exceed 20 years, upon demonstration of continued protection of the public health and safety and the environment. In the event that safe storage of spent fuel in a particular cask cannot be demonstrated beyond 20 years, an alternate means of storage will be required. Finally, DOE is required by the Nuclear Waste Policy Act of 1982 to accept spent fuel for ultimate disposal. As one commenter noted, DOE is proposing a new strategy in which Congress would authorize it to select a site in time to receive spent fuel for interim storage by 1998.

53. Comment. Commenters indicated that PSN made an error in calculating the dose rate at the gap between the MSB and MTC. PSN had 440 mrem/hr compared to NRC's calculated 4140 mrem/hr. Why weren't these discrepancies resolved? How would welders be protected?

Response. PSN did not make an error in their calculation. Rather, they made an error when transcribing a calculated value to an SAR table. This discrepancy is identified and resolved in the SER (pg 6-12). With respect to protection of welders, the operating procedures and radiation protection program of the licensee will include precautions so that the exposure of personnel working with the system inside the fuel handling building will be maintained within the dose limits of 10 CFR Part 20.

54. Comment. Commenters stated that the reported dose of 130 mrem/hr for the VSC-24 cask sides is still 6 times higher than the stated limit/specification of 20 mrem/hr.

Response. The limit of 20 mrem/hr stated in Section 1.2.4 of the Certificate of Compliance applies to the sides of the VCC, at the pad. The 130 mrem/hr value quoted in the comment refers to the maximum dose rate at the sides of the MTC when loaded with the MSB, inside the fuel handling building. Because the MSB has not been loaded into the VCC cask at this point, it is not subject to the 20 mrem/hr specification.

55. Comment. Commenters believed that PSN made several mistakes in calculating how much radiation might come off the surface of the VSC-24 cask. Because the VSC-24 cask has never been built, it is fair to say that no one has any definite idea of what the actual dose rates will be. In addition, some commenters noted that conclusions drawn from testing a prototype are of dubious import "when dealing with the effects of radiation."

Response. As stated in Section 6.3 of the SER, a number of errors were discovered in the vendor's shielding analysis. An adequate explanation for these errors was offered by the vendor. However, the NRC made independent confirmatory calculations to estimate the dose levels associated with the VSC-24 system. The vendor's shielding design and expected dose rates along the surface of the VCC were determined to be acceptable based on a comparison with the independent NRC calculations. NRC agrees with the commenter that the actual dose rates from specific fuel loaded into the cask cannot be exactly determined a priori. However, dose calculations can readily predict expected dose rates for the VSC-24 cask with sufficient accuracy to assure that NRC limits will not be exceeded. In addition, these calculations tend to be conservative and tend to overestimate actual dose rates that would be experienced during actual operations. Prototype testing was not used in evaluation of the adequacy of the shield design for the VSC-24 cask. Finally, the licensee will conduct surveys to ensure compliance with regulatory requirements and the Certificate of Compliance.

56. Comment. Commenters believed that PSN benchmarking of shielding codes against measured dose rates for the VSC-24 cask was grossly in error. Further, PSN did not benchmark the SKYSHINE-II calculation method. The NRC calculated direct and air-scattered dose rates, at various distances from the cask, which were many times higher than the PSN calculated dose rates.

Response. PSN's benchmarking of the ANISN and QAD computer codes for dose rate calculations was found by the NRC to be incomplete because it did not address differences in dose rates calculated by the ANISN and QAD computer codes. The NRC conducted independent confirmatory calculations to estimate the dose levels associated with the VSC-24 cask system for comparison with the vendor's calculations. Based on that comparison, the NRC concluded the design provided acceptable shielding.

Although PSN did not benchmark the SKYSHINE II calculation method, they used that method to calculate site boundary dose rates. Based on review of their calculations and independent NRC calculations, the NRC concluded that PSN had not calculated conservative neutron and gamma dose rates at the site boundary. However, even with the NRC's more conservatively calculated site boundary dose rates, the NRC concluded that general licensees using the VSC-24 cask will meet all applicable regulatory requirements.

In addition, the NRC also requires any VSC-24 user to measure the external cask surface dose rates to ensure the cask has been properly loaded and radiation monitoring to ensure compliance with regulatory requirements.

57. Comment. A number of commenters requested a public hearing on this rulemaking. Approximately half of the commenters requested that a full public hearing be held at each reactor facility site prior to the use of dry cask storage at that site.

Response. Consistent with the applicable procedure, the NRC does not intend to hold formal public hearings on the VSC-24 cask rule or separate hearings at each reactor site prior to use of the dry cask technology approved by the Commission in this rulemaking. Rulemaking procedures, used by the NRC for generic approval of the VSC-24 cask, including the underlying staff technical reviews and the opportunity for public input, are more than adequate to obtain public input and assure protection of the public health, safety and the environment. Further, in this rulemaking, NRC has taken extra steps to elicit and fully consider public comments on the VSC-24 technology.

Section 133 of the Nuclear Waste Policy Act of 1982 authorizes NRC to approve spent fuel storage technologies by rulemaking. When it adopted the generic process in 1990 for review and approval of dry cask storage technologies, the Commission stated that "casks. . . [are to] be approved by rulemaking and any safety issues that are connected with the casks are properly addressed in that rulemaking rather than in a hearing procedure." 55 FR 29181 (July 18, 1990). Rulemaking under NRC rules of practice, described in 10 CFR 2.804 and 2.805, provides full opportunity for expression of public views, but does not use formal hearings of the type requested by commenters.

In this proceeding, rulemaking clearly provided adequate avenues for members of the public to provide their views regarding NRC's proposed approval of the VSC-24 cask, including the opportunity to participate through the submission of statements, information, data, opinions and arguments. In this connection, the NRC staff prepared for public examination two separate, technical evaluations for the VSC-24 dry cask system, each time making detailed, documented findings of compliance with NRC safety, security and environmental requirements. The staff's first evaluation, prepared in March 1991, reviewed and approved the VSC-24 for reference in a site-specific application for an independent spent fuel storage installation. In May 1992, the NRC staff reviewed the VSC-24, and approved the design for purposes of initiating this rulemaking to grant a generic approval of the design. In addition, the staff conducted a third review in response to the public comments on the VSC-24 in this rulemaking, again finding compliance with NRC requirements as set forth in this notice of final rule and response to comments.

In addition to reviewing systematically and in depth the technical issues important to protecting public health, safety and the environment, the NRC has taken extra steps to obtain and fully consider public views on the VSC-24 technology, and has made every effort to respond to public concerns and questions about the VSC-24 cask's compliance with NRC safety, security and environmental requirements. The initial public comment period opened on June 26, 1992, and closed on September 9, 1992. In addition, NRC received a number of comments after the close of that period, all of which were fully considered. Subsequently, NRC extended the period for submission of public comments until February 22, 1993. Thus, the public comment period for this rule has effectively been almost nine months. In addition, the NRC staff made every effort to consider comments received after February 22, 1993. Further, the staff proposed and participated in a public meeting near one of the nuclear plants proposing to use the VSC-24 cask (i.e., Palisades), with the Attorney General of the State of Michigan, to provide further opportunity for public input on the safety, security and environmental compliance issues in this rulemaking. NRC also participated in an earlier meeting of the Van Buren County Commission near the plant site.

Under these circumstances, formal hearings would not appreciably add to NRC's efforts to ensure adequate protection of public health, safety and the environment, and are unnecessary to NRC's full understanding and consideration of public views on the VSC-24 cask.

58. Comment. Commenters believed that a full democratic process is needed in this decision.

Response. Because this rulemaking was conducted pursuant to the procedures for approving dry storage casks for use under a general license, as required by Congress in the Nuclear Waste Policy Act of 1982, and pursuant to the public notice and comment procedures of the Administrative Procedures Act, the resulting final rule approving the VSC-24 cask is the product of a process prescribed by law.

59. Comment. One commenter stated that the gap between the MSB and the MTC is given as 0.5 inch in WEP-109.001.4 and as 1.0 inch in Figure 5-5 of WEP-109.w13. This commenter also stated that the dose rate was not clear.

Response. The difference in the referenced gap size is a consequence of changes made as a result of earlier reviews. The final design was based on the 0.5 inch gap as indicated in WEP-109.001.4. The calculation of WEP-009.0013, which uses a 1.0 inch gap, is therefore conservative for shielding calculations. Because the gamma dose is more than 30 times that due to neutrons, any small decrease in the neutron dose rate, due to a smaller gap, would not significantly change the calculated neutron and gamma dose rates used to assess occupational exposure. In addition, these calculations conservatively neglect the shielding ring which would further reduce dose rate.

60. Comment. Commenters expressed concern that VSC-24 casks were being built at the Palisades Nuclear Plant before approval or certification.

Response. The NRC granted Pacific Sierra Nuclear Associates' request for an exemption to fabricate a limited number of the casks before issuance of the Certificate of Compliance under its NRC approved quality assurance program, and at its financial risk. The NRC's finding, based on the SAR for the VSC-24 cask and the NRC's SER, concluded that beginning fabrication prior to the issuance of the Certificate of Compliance would pose no undue risk to public health and safety. Use of these casks is dependent on satisfactory completion of NRC's certification process.

61. Comment. Some commenters requested that the NRC prepare an environmental impact statement (EIS) and update the Generic EIS for the

handling and storage of spent fuel. The EIS should be submitted to the Environmental Protection Agency (EPA) and to the State of Michigan. Some commenters also requested that action on this rule be delayed until the Wisconsin Environmental Impact Statement is complete.

Response. The potential environmental impacts of utilities using the VSC-24 cask (or any of the other spent fuel casks approved by NRC (10 CFR 72.214)) have been fully considered and are documented in a published Environmental Assessment (EA) covering this rulemaking. Further, as described below, the EA indicates that use of the casks would not have significant environmental impacts. Specifically, the EA notes the 30-plus years of experience with dry storage of spent fuel, identifies the previous extensive NRC analyses and findings that the environmental impacts of dry storage are small, and succinctly describes what impacts there are, including the non-radiological impacts of cask fabrication (i.e., the impacts associated with the relatively small amounts of steel, concrete and plastic used in the casks are expected to be insignificant), the radiological impacts of cask operations (i.e., the incremental offsite doses are expected to be a small fraction of and well within the 25 mrem/yr limits in NRC regulations), the potential impacts of a possible dry cask accident (i.e., the impacts are expected to be no greater than the impacts of an accident involving the spent fuel storage basin), and the potential impacts due to possible sabotage (i.e., the offsite dose is calculated to be about one rem). All of the NRC analyses collectively yield the singular conclusion that the environmental impacts and risks are expected to be extremely small.

The absence of significant environmental impacts from dry cask storage at a reactor site is also the conclusion of other NRC EA's for previously approved dry casks analyzed in earlier rulemakings addressing Part 72, and in the Commission's Waste Confidence decisions in 1984 (August 31, 1984; 49 FR 34658) and 1989 (September 29, 1989; 54 FR 39765). In the 1984 Waste

Confidence decision, the Commission concluded there was reasonable assurance spent fuel can be safely stored at reactor sites without significant environmental impacts, for at least 30 years beyond expiration of NRC reactor operating licenses. The 1989 Waste Confidence decision review reaffirmed prior Commission conclusions on the absence of significant environmental impacts.

Thus, given the Commission's specific consideration of the environmental impacts of dry storage summarized above, and given the absence of any new information casting doubt on the conclusion that such impacts are expected to be extremely small and not environmentally significant, no meaningful environmental insights are likely to be gained from further preparation of either an EIS or an updated GEIS for the dry storage methodology.

The EA covering the proposed rule, as well as the finding of no significant impact (FONSI) prepared and published for this rulemaking, fully comply with the NRC environmental regulations in 10 CFR Part 51. Moreover, since the Commission's environmental regulations in Part 51 implement NEPA and give proper consideration to the guidelines of CEQ, they assure that the EA and the FONSI conform to NEPA procedural requirements, and that further analyses are therefore not legally required.

In connection with the EA and FONSI, it bears emphasizing that 10 CFR Part 72, Subpart K already authorizes dry cask storage and already approves dry casks for use by utilities to store spent fuel at reactor sites. See 10 CFR 72.214 for a listing of information on Cask Certificate Nos. 1000 through 1003. The present rulemaking is accordingly for the limited purpose of adding one more cask to the list of casks already approved by NRC. Furthermore, the cask, to be added to the NRC list by this rulemaking will comply with all applicable NRC safety requirements.

Finally, this rulemaking applies to cask use by any power reactor licensee within the United States. Therefore, it is not dependent on any one individual State's actions including preparation of a separate EIS by any State. Further, nothing in this rulemaking would preclude any State from implementing its environmental statutes and regulations as may otherwise be permitted by law.

62. Comment. Commenters believed that a cost/benefit analysis should be prepared. One commenter proposed a cost comparison formula which would estimate costs associated with dry cask storage over the next 1000 years.

Response. A regulatory analysis, which considers both benefits and impacts of adding the VSC-24 cask to the list of NRC-approved casks under 10 CFR Part 72, Subpart K, was prepared in support of this rulemaking action. It was included as a part of the notice of proposed rulemaking and is also included in this final rulemaking notice. This regulatory analysis reflects the limited economic scope of this rulemaking. The 1000 year cost comparison identified above assumes 1000-year interim storage at Palisades, an assumption the NRC is not proposing or adopting in this rulemaking. The NRC Waste Confidence decisions concluded there is reasonable assurance the Federal government will begin receiving spent fuel for disposal by 2025. Thus, the likelihood of 1000-year interim storage at Palisades is extremely small.

63. Comment. One commenter wanted letter reports to the NRC distributed to local and state government authorities and local libraries in the vicinity of facilities using the VSC-24 cask.

Response. The NRC interprets this comment as applying to letter reports required by the Certificate of Compliance. Letter reports sent to the NRC are routinely placed in the Public Document Room and Local Public Document Rooms near each facility. Local Public Document Rooms are located in public, university, and special libraries. A directory of Local Public Document Rooms is published by the NRC as NUREG BR-88. The NRC would respond to State requests for copies of such reports through NRC's State Relations Program.

64. Comment. Commenters indicated that operating procedures, evaluation reports, and training programs should be submitted to the NRC, state and local government authorities, and placed in local libraries near such facilities.

Response. These documents expand on generically approved procedures in the SAR, Certificate of Compliance, or in the case of the boron determination, on national standards. In accordance with the NRC requirements, licensees are not required to submit this information to the NRC or other government authorities. Rather, this information is evaluated by the licensee and is available for inspection by the NRC. The NRC's inspection program includes requirements to inspect these procedures.

65. Comment. Commenters stated that the VSC-24 is not a cask. The designer called it a cask system.

Response. The NRC considers it to be a cask. It is called a cask system because it consists of several components.

66. Comment. Commenters believe that there is poor management at Consumers Power Company. NRC Information Notice 91-56 says they still have a provisional license after 20 years. Consumers Power Company had serious quality control violations, below average operating capacity, and faulty construction at Midland.

Response. Although this comment is not directly related to this rulemaking, which is to provide generic approval of the VSC-24 cask design that is not dependent on site specific consideration for any one licensee, NRC notes that its Systematic Assessment of Licensee Performance (SALP) program is an integrated staff effort to collect available observations and data on a periodic basis and to evaluate licensee performance, including Consumers Power, on the basis of this information. The most recent SALP report for Palisades, covering the period January 1, 1991 through March 31, 1992, states in summary, "Overall performance at the Palisades Nuclear Power Plant was characterized by generally steady or improving results and showed a conservative and safe operating philosophy. The overall degree of management attention and effectiveness was acceptable in all areas." Finally, the Palisades Nuclear Plant was granted a full term operating license on February 21, 1991.

The SALP report for the preceding period from September 1, 1989 through December 31, 1990 provided similar conclusions and stated, "the degree of management attention and effectiveness ranged from commendable in some areas to needing attention in others. Overall the conduct of activities was appropriately directed to assurance of safety. Management appeared proactive and effective in demonstrating a conservative operating philosophy and establishing high standards of performance in operations, maintenance/surveillance, and security."

67. Comment. One commenter believed that the Certificate of Compliance should list all NRC regulations controlling the use of the VSC-24 cask for the storage of spent fuel.

Response. The Certificate of Compliance contains a general reference to the provisions of 10 CFR Part 72, which includes in Subpart K, the regulations relevant to the storage of spent fuel under a general license. A specific reference to each regulation section is, therefore, unnecessary.

68. Comment. One commenter was favorable to the VSC-24 cask stating that it was cost-effective, made in the USA, additional shielding could be added at low cost if required, the welded closure requires no monitoring, and risk is minimized by weld sealing the MSB in the reactor fuel handling building. Another commenter noted that this rulemaking is a positive action which should decrease cost and increase the safety of storing fuel. Another commenter noted the Palisades spent fuel pool is closer to Lake Michigan than the cask pad, both in terms of distance and elevation. The storage of spent

fuel in a pool requires active systems for shielding, cooling and reactivity control. The VSC is passive, requiring no pumps, valves, or heat exchangers.

Response. None required.

69. Comment. Commenters believed that it is not acceptable to increase the number of approved cask designs. The goal must be the function of the cask itself to contain radioactivity in high concentrations and prevent it from dispersing into the biosphere as well as to shield workers and others from radiation exposure. Some suggested that alternative actions to dry cask storage should be considered.

Response. The NRC, in implementing the Nuclear Waste Policy Act of 1982, has an obligation to approve the use of casks for the storage of spent fuel, provided these casks meet applicable regulatory requirements. The NRC agrees with the commenter that these casks should contain radioactivity and protect workers, the public, and the environment. The previous rulemaking of 1990 (55 FR 29181) found that spent fuel stored in dry storage casks designed to meet the NRC regulatory requirements can contain radioactivity safely. This rulemaking adds one cask design, which meets the safety requirements previously developed. The previous responses to comments, as well as the detailed safety and environmental analyses underlying this rulemaking, and described elsewhere in this notice, all reveal that the VSC-24 cask will conform to the NRC requirements, and that its use should not pose the potential for significant environmental impacts.

The principal alternatives available to the NRC would be procedural in nature, whereby dry cask spent fuel storage could be approved under other existing or new parts of Title 10, Code of Federal Regulations. Regardless of the method selected to approve such dry cask spent fuel storage, all would have similar environmental impacts.

The NWPA directed that the NRC approve one or more technologies, that have been developed and demonstrated by DOE, for the use of spent fuel storage at the sites of civilian nuclear power reactors without, to the extent practicable, the need for additional site-specific review. The NWPA also directed that the NRC, by rulemaking, set forth procedures for licensing the technology. Regulations for accomplishing this are in place. Therefore, the no action alternative is not acceptable.

Alternative spent fuel storage technologies exist. However, at this time, the NRC considers them neither sufficiently demonstrated nor practicable for use under the general license provisions of Subpart K of 10 CFR Part 72 without additional site-specific reviews. If other storage technologies become more amenable to this type of action, they could be considered at a later time.

70. Comment. Commenters expressed concern that Pacific Nuclear, Inc., the original designer and manufacturer of the VSC-24 cask system, had ended its involvement with the cask. Reasons cited included the issue of liability, negligence issues that might surface in the future with the cask, the fact that the original designers divested themselves due to concern about the cask, and who would be responsible in the event of leakage. Commenters also questioned whether NRC had attempted to ascertain the reason for the divestiture action by Pacific Nuclear to discover if the reason related to safety of the cask, liability, or any other consequences.

Response. NRC is not aware of any safety, negligence, liability or legal concerns which prompted Pacific Nuclear, Incorporated to divest itself from the VSC-24 cask. The key individual involved in the design and development of the VSC-24 was also involved in the design and development of a new modular horizontal concrete spent fuel storage system (NUHOMS design) and formed a new company, Pacific Sierra Nuclear, for the commercial manufacture and marketing of the VSC-24 storage system. NRC focuses its efforts on assuring safety and environmental protection through reviews of applications for licenses and safety analysis reports. If a new company applies for a certificate of compliance, that new company must meet all NRC requirements as would any existing company. Through NRC's review and independent evaluation of the applicant's safety evaluation report and through this rulemaking action, NRC will assure that the cask meets Part 72 requirements and can be used by individual nuclear power plant licensees with full assurance of protection of the public health and safety and the environment. The NRC has experienced no difficulty obtaining safety information or answers to its questions from either firm, either before, or after the divestiture.

Following the divestiture, Pacific Nuclear sent a letter containing comments on the VSC-24 design. The staff satisfactorily resolved and answered these comments with a letter; both the Pacific Nuclear and NRC letters are available in the Public Document Room. The issues contained in this exchange of letters and all other safety issues related to the design of the VSC-24 are described in the staff's SER.

71. Comment. A commenter noted that Consumers Power's comments to the NRC during this rulemaking indicate that they do not have the kind of fuel that was specified in the Certificate of Compliance for the casks at Palisades. They noted it is hard to believe that the NRC does not know what kind of fuel it is licensing the cask for, but noted that appeared to be the case. The commenter further noted that any approval given by the NRC would have to be site specific and not generic and therefore, this would require a hearing.

Response. The type of fuel that is being approved for storage in the VSC-24 cask is specified in the vendor's Safety Analysis Report as well as in the Certificate of Compliance and SER prepared by the NRC staff. NRC regulations require the vendor to specify the type of spent fuel to be stored in the cask before NRC approval, and NRC thoroughly reviewed the vendor's SAR and spent fuel specifications and made them appropriate items for public comment in this rulemaking. Commenters are therefore mistaken in saying the type of fuel to be stored in the VSC-24 cask is not known.

The kind of fuel to be loaded into and stored in the VSC-24 cask at Palisades, should Consumers Power proceed with use of the VSC-24 cask, must be acceptable fuel for storage in the cask and must meet the Certificate of Compliance specifications mentioned above for acceptable fuel which may be stored in the cask. In this regard, the Certificate of Compliance and SER have been clarified to specifically identify the fuel assembly classes acceptable for storage in the VSC-24 cask and to identify limits for physical dimensions, weight, burnup, decay power, and radiation source term for other fuels that may be stored in the VSC-24 cask. NRC regulations prohibit Consumers Power from using the VSC-24 cask in violation of the Certificate of Compliance spent fuel specifications, and Consumers Power must perform written evaluations before using the cask that verify all Certificate of Compliance conditions are met.

As is evident from this and other responses to public comments, this rulemaking provides NRC approval for storage of spent fuel in the VSC-24 at any site in accordance with the generic conditions and specifications in the Certificate of Compliance. As noted, it does not constitute a site-specific approval of the VSC-24 cask for use by Consumers Power at the Palisades plant.

72. Comment. A number of commenters requested that the comment period be extended principally citing the fact that NRC had released a large volume of highly technical material associated with the VSC-24 cask and that the 30 day reopening of the comment period which NRC had provided was not a sufficient time for review and comment on the material. Commenters also questioned why the information was not released earlier.

Response. NRC is not granting an additional extension to the comment period. First, the new information that was released is only an increment to that previously disclosed. In addition, most of the individual pages released are computer output printouts, the results of which were previously available in various documents made available at the beginning of the public comment period. In the Federal Register Notice (January 21, 1993; 58 FR 5301) announcing the comment period extension, NRC made clear the limited, incremental character of the technical information. The information of the cask vendor being disclosed at this time added detail to the information NRC previously placed in the Public Document Room at the outset of this rulemaking. It complements and supplements the design information already disclosed, providing further detail on such matters as the vendor's design calculations (often in the form of computer runs) and specific data inputs for models used by the vendor for such calculations, as well as cask design details such as reinforcing steel sizing and shield lid thickness. The information being disclosed therefore provides additional specificity for the public about the technical information that was considered by the NRC staff in preparing the principal NRC documents underlying this rulemaking. These documents include the proposed Certificate of Compliance for the cask and the associated NRC staff SER and related EA, which were previously placed in the NRC Public Document Room at the outset of this proposed rulemaking.

Second, the initial public comment period opened on June 26, 1992, and closed on September 9, 1992. The comment period was reopened on January 21, 1993 and ended on February 22, 1993. In addition, at the public meeting held with the Michigan Attorney General on February 23, 1993, NRC assured that comments received within five working days after that meeting would be considered. Although the comment periods have closed, NRC has considered all comments received. Thus, the public comment period for this rule has effectively been almost nine months which the NRC believes constitutes more than sufficient time for this type of rulemaking.

73. Comment. One commenter questioned the validity of neglecting gamma dose at the nozzles.

Response. The referenced Case 5 calculates the dose rate as the MSB is lowered into the VCC during transfer. Dose is estimated at the point of maximum exposure, that is at the outlet vent and the top of the VSC. Under these circumstances, the entire distribution of radioactive material in the spent fuel assemblies contributes to the dose in a transient fashion. The assumption that the source is directly from the active fuel which is aligned with the air exhaust is conservative, since it is the highest and is sustained for a short period of time. Other MSB/VCC relative positions during transfer would yield smaller dose rates. Calculations demonstrated that the dose rate from gamma emitting radioactive material in the nozzle is three orders of magnitude less than the dose rate from the active fuel section.

74. Comment. A commenter noted that the geometry for dose calculations was based on an earlier design and not on the latest configuration.

Response. The changes in design referred to by the commenter were slight repositionings of the inlet air duct. The reorientation involves minor changes of both the horizontal and vertical orientation of the duct but does not change the circuitous path which contributes to radiation protection. In addition, the analysis does not take credit for the 0.5 inch steel liner of the duct which would offset any small changes in dose due to reorientation of the duct. Therefore, the design changes do not result in a significant change in the radiation dose rate calculations.

75. Comment. Commenters asked who would be responsible for oversight of fuel stored in casks after decommissioning of the reactor, shipment of the fuel off-site, and for decommissioning of the casks after stored fuel was shipped off-site.

Response. In accordance with 10 CFR 50.54(bb), all operating nuclear power reactor licensees are required, no later than 5 years prior to the expiration of the operating license, to provide the NRC, for review and approval, the licensee's program to manage and provide funding for the management of all irradiated fuel. NRC's review of the licensee's fuel management program will be undertaken as part of continued licensing under the provisions of Part 50 and Part 72 of the Commission's regulations.

With respect to decommissioning, the licensee may select a decommissioning alternative that will:

1. allow storage of spent fuel in the spent fuel pool, in which case the licensee will be required to maintain its Part 50 license;
2. allow storage of fuel in a certified cask under the provisions of Part 72 as long as the Part 50 license remains in effect; or
3. allow storage in an on-site independent spent fuel storage installation under the site specific licensing provisions of Part 72.

For any of the above alternatives, the licensee will be responsible for safe storage of spent fuel during the period of storage, for later shipment off-site for further storage or disposal and for final decommissioning of the reactor spent fuel pool, dry storage cask or ISFSI to a level permitting unrestricted release of the site and facility. The requirements for decommissioning are provided in 10 CFR Part 72.30, which defines decommissioning planning, financial assurance and recordkeeping provisions.

Finding of No Significant Environmental Impact: Availability

Under the National Environmental Policy Act of 1969, as amended, and the Commission's regulations in Subpart A of 10 CFR Part 51, the Commission has determined that this rule is not a major Federal action significantly affecting the quality of the human environment and, therefore, an environmental impact statement is not required. This final rule adds an additional cask to the list of approved spent fuel storage casks that power reactor licensees can use to store spent fuel at reactor sites without additional site-specific approvals by the Commission. The environmental assessment and finding of no significant impact on which this determination is based is available for inspection at the NRC Public Document Room, 2120 L Street, NW. (Lower Level), Washington, DC. Single copies of the Environmental Assessment and the Finding of No Significant Impact are available from Mr. Gordon E. Gundersen, Office of Nuclear Regulatory Research, U. S. Nuclear Regulatory Commission, Washington, DC, 20555, telephone (301) 492-3803.

Paperwork Reduction Act Statement

This final rule does not contain a new or amended information collection requirement subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq.). Existing requirements were approved by the Office of Management and Budget approval number 3150-0132.

Regulatory Analysis

On July 18, 1990 (55 FR 29181), the Commission issued an amendment to 10 CFR Part 72, which provided for the storage of spent nuclear fuel under a general license. Any nuclear power reactor licensee can use these casks if: (1) They notify the NRC in advance; (2) The spent fuel is stored under the conditions specified in the cask's Certificate of Compliance; and (3) The other conditions of the general license are met. As part of the 1990 rulemaking, four spent fuel storage casks were approved for use at reactor sites, and were listed in 10 CFR 72.214. That rulemaking envisioned that storage casks certified in the future could be routinely added to the listing in § 72.214 through rulemaking procedures. Procedures and criteria for obtaining NRC approval of new spent fuel storage cask designs were provided in 10 CFR 72.230.

The alternative to this proposed action is to withhold certification of these new designs and to consider the granting of a site-specific license to each utility that applied for permission to use these new casks. This alternative would be more costly and time consuming because each site-specific license application would require a specific review. In addition, withholding certification would ignore the rulemaking procedures and criteria in 10 CFR Part 72, Subparts K and L, for the addition of new cask designs. Further, it is in conflict with the Congressional direction in Sections 133 and 218 of the Nuclear Waste Policy Act of 1982 to establish procedures for the licensing of technologies for the use of spent fuel storage at the sites of civilian nuclear power reactors without, to the extent practicable, the need for additional site reviews. Also, this alternative would exclude new vendor cask designs from the approved NRC list under Subpart K without cause and would arbitrarily limit choice of cask designs available to power reactor licensees under the general license.

This final rulemaking will eliminate the above problems. Further, this action will have no adverse effect on the public health and safety.

The benefit of this final rule to nuclear power reactor licensees is to make available a greater choice of spent fuel storage cask designs which can be used under a general license. However, the newer cask designs may or may not have an advantage over the existing designs in that power reactor licensees may or may not prefer to use the newer casks. The new cask vendors with casks to be listed in § 72.214 benefit by being able to obtain NRC certificates once for a cask design which can then be used by many power reactor licensees under the general license. Vendors with cask designs already listed may be adversely impacted in that power reactor licensees may choose a newly listed design over an existing one. However, the NRC is required by its regulations and NWPA requirements to establish a procedure and to consider applications to certify and list approved casks. The NRC also benefits because it will be able to certify a cask design based on one generic safety and environmental review, for use by multiple licensees. This final rulemaking has no significant identifiable impact or benefit on other government agencies.

Based on the above discussion of the benefits and impacts of the alternatives, the NRC concludes that the requirements of the final rule are commensurate with the Commission's responsibilities for protection of the public health and safety and the common defense and security. No other available alternative is believed to be as satisfactory; thus, this action is recommended.

Regulatory Flexibility Act Certification

In accordance with the Regulatory Flexibility Act, 5 U.S.C. 605(b), the Commission certifies that this rule, will not have a significant economic impact on a substantial number of small entities. This amendment affects only licensees owning and operating nuclear power reactors and cask vendors. The owners of nuclear power plants do not fall within the scope of the definition of "small entities" set forth in Section 601(3) of the Regulatory Flexibility Act, 15 U.S.C. 632, or the Small Business Size Standards set out in regulations issued by the Small Business Administration at 13 CFR Part 121.

Backfit Analysis

The NRC has determined that the backfit rule, 10 CFR 50.109, does not apply to this final rule, and, thus, a backfit analysis is not required for this final rule, because this amendment does not involve any provisions which would impose backfits as defined in § 50.109(a)(1).

List of Subjects in 10 CFR Part 72

Manpower training programs, Nuclear materials, Occupational safety and health, Reporting and recordkeeping requirements, Security measures, Spent fuel.

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and 5 U.S.C. 552 and 553, the NRC is adopting the following amendments to 10 CFR Part 72.

PART 72 - LICENSING REQUIREMENTS FOR THE INDEPENDENT STORAGE OF SPENT NUCLEAR FUEL AND HIGH-LEVEL RADIOACTIVE WASTE

1. The authority citation for Part 72 continues to read as follows:

AUTHORITY: Secs. 51, 53, 57, 62, 63, 65, 69, 81, 161, 182, 183, 184, 186, 187, 189, 68 Stat. 929, 930, 932, 933, 934, 935, 948, 953, 954, 955, as amended, sec. 234, 83 Stat. 444, as amended (42 U.S.C. 2071, 2073, 2077, 2092, 2093, 2095, 2099, 2111, 2201, 2232, 2233, 2234, 2236, 2237, 2238, 2282); sec. 274, Pub. L. 86-373, 73 Stat. 688, as amended (42 U.S.C. 2021); sec. 201, as amended, 202, 206, 88 Stat. 1242, as amended, 1244, 1246 (42 U.S.C. 5841, 5842, 5846); Pub. L. 95-601, sec. 10, 92 Stat. 2951 (42 U.S.C. 5851); sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332); secs. 131, 132, 133, 135, 137, 141, Pub. L. 97-425, 96 Stat. 2229, 2230, 2232, 2241, sec. 148, Pub. L. 100-203, 101 Stat. 1330-235 (42 U.S.C. 10151, 10152, 10153, 10155, 10157, 10161, 10168).

Section 72.44(g) also issued under secs. 142(b) and 148(c), (d), Pub. L. 100-203, 101 Stat. 1330-232, 1330-236 (42 U.S.C. 10162(b), 10168(c)(d)). Section 72.46 also issued under sec. 189, 68 Stat. 955 (42 U.S.C. 2239); sec. 134, Pub. L. 97-425, 96 Stat. 2230 (42 U.S.C. 10154). Section 72.96(d) also issued under sec. 145(g), Pub. L. 100-203, 101 Stat. 1330-235 (42 U.S.C. 10165(g)). Subpart J also issued under secs. 2(2), 2(15), 2(19), 117(a), 141(h), Pub. L. 97-425, 96 Stat. 2202, 2203, 2204, 2222, 2244 (42 U.S.C. 10101, 10137(a), 10161(h)). Subparts K and L are also issued under sec. 133, 98 Stat. 2230 (42 U.S.C. 10153) and sec. 218(a), 96 Stat. 2252 (42 U.S.C. 10198).

2. In § 72.214, Certificate of Compliance 1007 is added to read as follows:

§ 72.214 List of approved spent fuel storage casks.

*   *   *   *   *

Certificate Number: 1007

SAR Submitted by: Pacific Sierra Nuclear Associates

SAR Title: Safety Analysis Report for the Ventilated Storage Cask System

Docket Number: 72-1007

Certification Expiration Date: (20 years after final rule effective date)

Model Number: VSC-24

Dated at Rockville, Maryland, this 1st day of April, 1993.

For the Nuclear Regulatory Commission.

James H. Sniezek

James H. Sniezek
Acting Executive Director
for Operations