UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

May 27, 1999

NRC INFORMATION NOTICE 99-15: MISAPPLICATION OF 10 CFR PART 71 TRANSPORTATION SHIPPING CASK LICENSING BASIS TO 10 CFR PART 50 DESIGN BASIS

Addressees

All holders of operating licenses or construction permits for nuclear power reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to alert addressees to a potential problem with use of the vendor's consolidated safety analysis report for the IF-300 spent fuel shipping cask, which could place plants outside their design basis during the loading or unloading of spent fuel. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Background

The IF-300 spent fuel shipping cask is licensed under Title 10 of the Code of Federal Regulations Part 71 (10 CFR Part 71) as a type B package and, therefore, must be designed to withstand the tests for hypothetical accident conditions required by 10 CFR 71.73. The cask must be able to withstand a 9-meter [30-foot] drop through air onto a flat, horizontal, essentially unyielding surface. This test is performed in the "transportation ready" position with the head in place and fully tensioned, and the valve box covers installed.

In 1998, Chem-Nuclear Systems, Inc. became the license holder for the IF-300 shipping cask. Vectra Technologies was the license holder from 1988 to 1998.

Description of Circumstances

NRC Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment," dated April 11, 1996 (Accession Number 9604080259) asked licensees to review their programs for handling heavy loads and to send a response to the NRC. A request for additional information (RAI) was sent to a number of plants that did not have single-failure-proof cranes asking them to evaluate an accident scenario for in-plant cask movement where cask integrity would not be achieved should the lid not be fully secured.

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Since Carolina Power and Light Company (CP&L), the licensee for the Shearon Harris Nuclear Power Plant (Harris plant), does not have a single-failure-proof crane, it was asked to evaluate the scenario described in the RAI. At the Harris plant, the valve box covers are removed in the rail bay before the cask is lifted. With the valve box covers removed, the cask is no longer in the transportation configuration. The cask is then moved from the rail car to the cask preparation area. The IF-300 shipping cask Consolidated Safety Analysis Report (CSAR) and the IF-300 shipping cask Operating Manual describe detensioning all cask head sleeve nuts and removing all but four sleeve nuts in the cask preparation area. Again, these changes put the cask outside its transportation ready configuration. The cask is then moved from the preparation area to the loading basin. With the top of the cask suspended 30.5 centimeters [1-foot] above the surface of the unloading basin, the final four sleeve nuts are removed. The cask is then placed in the unloading basin and the head removed. CP&L had implemented the cask vendor loading/unloading procedures at the Harris plant since it was licensed in 1987. Since these steps were described in the vendor's CSAR and operating manual, CP&L assumed the vendor had an analysis demonstrating that full cask integrity was maintained even when the four-sleeve-nut configuration was used. This assumption was undocumented, but it formed the basis for the statements in Section 15.7.5, "Spent Fuel Cask Drop Accident," of the Harris plant's Final Safety Analysis Report (FSAR), which states that the potential drop of a spent fuel cask is limited to less than an equivalent 9-meter [30-foot] drop onto a flat, essentially unyielding, horizontal surface. Since the spent fuel cask is designed to withstand such loadings, the radiological consequences of these accidents are not evaluated.

In evaluating the RAI, CP&L requested that the vendor provide the analysis that supported the CSAR statements related to the four-sleeve-nut configuration. CP&L was told by the vendor that no analysis existed. Such an analysis is not required by 10 CFR Part 71. The CP&L stopped any further cask unloading until this issue could be resolved. The vendor was asked to perform an analysis to support the CSAR statements. Upon completion of the analysis, the vendor concluded that the lid would not become dislodged, and that fuel elements would remain in the cask. However, it could not show that the cask would remain gas tight. CP&L used this information in an analysis to determine the consequences of releasing the radioactive nuclides contained in the fuel gap, due to a loss of cask head sealing, into the fuel handling building. CP&L found that the consequences were a small fraction of the NRC acceptance criteria for Section 15.7.5 of the Standard Review Plan (NUREG-0800). Since no release had been postulated in the FSAR, this was considered an increase in consequences and was submitted to the NRC as an unreviewed safety question on March 14, 1997. In addition, this item was reported to the NRC in accordance with 10 CFR 50.73 in licensee event report (LER) 50-400/97-004-00 on March 31, 1997.

NRC requested additional information in relation to particulate contamination in the cask that could be blown out with the fuel gap fission product gas venting. Additional analysis was performed by CP&L, which showed that the release would be a small fraction of 10 CFR Part 100 limits and was well within the Standard Review Plan acceptance criteria. On June 26, 1997, the NRC issued Amendment 73 to the Harris license in relation to this issue.

Subsequently, the CP&L Brunswick site and Robinson site evaluated the results of the Harris findings. Both plants have single-failure-proof cranes. Both plants concluded that the removal of the valve covers constituted a change from the transportation configuration and, therefore, was not previously analyzed. This was based on the fact that the lift to place the cask onto the rail car was conducted with the crane not in the single-failure-proof configuration and without the cask valve covers installed. At both Brunswick and Robinson sites, the licensee concluded that an unreviewed safety question existed and submitted a license amendment to address the issue. CP&L maintained a hold on these casks at the applicable site until the unreviewed safety question was resolved.

Discussion

Where 10 CFR Part 71 licensing-basis information is being relied upon to satisfy the design basis for 10 CFR Part 50, licensees should ensure that the Part 71 information is adequately supported to satisfy the requirements of Part 50. Information provided by a vendor should be consistent but may not always be site specific. Where a site-specific analysis is needed to support FSAR statements, vendor information should be supplemented with the necessary additional analysis. It is important to review vendor information before using their equipment.

Plants with single-failure-proof cranes that move the cask in other than the transportation ready configuration may also find this problem applicable, even though cask drops or tipping accidents are not required to be considered. When a cask is moved in other than the transportation ready configuration, a plant-specific analysis would be necessary to determine that the consequences are bounded by the current design basis of the plant. For example, the removal of the valve covers before movement may make the cask susceptible to damage from bumping into the side of the pool or building or being bumped into by other equipment. In this case, the valves on the side of the cask could be damaged, causing the cask to lose its leak-tight quality.

Related Generic Communications

NRC Information Notice (IN) 97-51, "Problems Experienced With Loading and Unloading Spent Nuclear Fuel Storage and Transportation Casks," issued July 11, 1997 (Accession Number 9707080365).

NRC Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment," April 11, 1996 (Accession Number 9604080259).

This information notice requires no specific action or written response. However, recipients are reminded that they are required to consider industry wide operating experience (including NRC information notices) where practical, when setting goals and performing periodic evaluations under Section 50.65, "Requirement for monitoring the effectiveness of maintenance at nuclear power plants," to 50 of Title 10 of the Code of Federal Regulations. If you have any questions about the information in this notice, please contact the technical contact listed below or the appropriate Office of Nuclear Reactor Regulation project manager.

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LIST OF RECENTLY ISSUED NRC INFORMATION NOTICES

Information Notice No.	Subject	Date of Issuance	Issued to
99-14	Unanticipated Reactor Water Draindown at Quad Cities Unit 2, Arkansas Nuclear One Unit 2 and Fitzpatrick	5/5/99	All holders of licenses for nuclear power, test, and research reactors
99-13	Insights from NRR Inspections of Low- and Medium-Voltage Circuit Breaker Maintenance Programs	4/29/99	All holders of operating licenses for nuclear power reactors
99-12	Year 2000 Computer Systems Readiness Audits	4/28/99	All holders of operating licenses or construction permits for nuclear power plants
99-11	Incidents Involving the Use of Radioactive Iodine-131	4/23/99	All medical use licensees
97-15, Sup 1	Reporting of Errors and Changes in Large-Break/Small-Break Loss-of-Coolant Evaluation Models of Fuel Vendors and Compliance with 10 CFR 50.46(a)(3)	4/16/99	All holders of operating licenses for nuclear power reactors, except those who have permanently cease operations and have certified that fuel has been permanently removed from the reactor
99-10	Degradation of Prestressing Tendon Systems in Prestressed Concrete Containments	4/13/99	All holders of operating licenses for nuclear power reactors
99-09	Problems Encountered When Manually Editing Treatment Data on The Nucletron Microselectron-HDR (New) Model 105.999	3/24/99	All medical licensees authorized to conduct high-dose-rate (HDR) remote after loading brachytherapy treatments

OL = Operating License
CP = Construction Permit

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DATE	04 / 08 /99		04 / 05 /99		04 / 19 /99		04 / 08 /99		

OFFICE	DBC/REXB		BC/REXB			
NAME	JLyons*		TMarsh lm KAG			
DATE	05/20/99		6 / 2 /99			

* See previous concurrence

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