Technical Issues Related to Siting Criteria and Seismic Design of ISFSI Using DCSS

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ABSTRACT

Currently 10 CFR Part 72.102 requires that for Independent Spent Fuel Storage Installations (ISFSIs) for storing spent fuel in a dry cask storage system (DCSS), located west of the Rocky Mountain Front (approximately 104° west longitude), seismicity must be evaluated using deterministic techniques of 10 CFR Part 100, Appendix A, and that the design earthquake (DE) must be equivalent to a safe shutdown earthquake (SSE) for a nuclear power plant. In June 1998, the Nuclear Regulatory Commission (NRC) approved a Rulemaking Plan to amend Part 72 to allow the new ISFSI licensees to use probabilistic seismic hazard analysis methods for evaluating geological and seismological criteria, and to recognize uncertainties in geoscience parameters.

These criteria are proposed to make 10CFR Part 72 regulations conform to the recent changes in 10CFR Part 50 regulations in nuclear power plants seismic criteria. The Rulemaking Plan also recommended that the new ISFSI licensees be allowed an option to use a graded approach to seismic design of "Important to Safety" ISFSI structures, systems, and components (SSCs). This paper examines technical issues related to the proposed 10CFR Part 72 amendments in the Rulemaking Plan for the seismic siting criteria and design of ISFSIs. The issues examined in the paper include level of site investigations required, selection of a design earthquake and associated probability of exceedance, and structural stability/structural integrity of casks during a seismic event. The suggested graded approach has been examined in details from the perspectives of the risk to public health and safety, and regulatory burden on the stakeholders.

BACKGROUND AND EXISTING REGULATIONS

In 1980, the Nuclear Regulatory Commission (NRC) codified regulations in Code of Federal Regulations, Title 10, Part 72 (10 CFR Part 72) establishing licensing requirements for storing spent nuclear fuel on an interim basis in an independent spent fuel storage installation (ISFSI) facility. Subpart E of Part 72 contains siting evaluation factors that must be investigated and assessed with respect to the siting of an ISFSI, including a requirement for evaluation of geological and seismological characteristics.

Part 72 was amended in 1988 to include monitored retrievable storage facilities (MRSs). The 1988 amendments also relocated the provisions governing evaluation of geological and seismological characteristics to section 72.102. Section 72.102 has separate requirements for ISFSI facilities located east and west of the Rocky Mountain Front (approximately 104° West longitude). For any ISFSI located west of the Rocky Mountain Front or in any areas of known potential seismic activity, seismicity is required to be evaluated by the techniques of Appendix A of Part 100, and that, for sites evaluated under the Appendix A criteria, the Design Earthquake (DE) must be equivalent to the safe shutdown earthquake (SSE) for a nuclear power plant (NPP). For ISFSIs located east of the Rocky Mountain Front and not in areas of known seismic activity, a standardized DE described by an appropriate response spectrum anchored at a peak ground acceleration of 0.25 g may be used. Alternatively, a site-specific DE may be determined by using the criteria and level of investigations required by Appendix A of 10 CFR Part 100. The minimum horizontal ground motion acceleration for a DE is 0.1g with the appropriate response spectrum.

The procedures in Appendix A of Part 100 for determining the quantitative vibratory ground motion design basis at a proposed site requires the use of "deterministic" approaches in the development of a single set of earthquake sources. The applicant develops for each source a postulated earthquake to be used as the source of ground motion that can affect the site, locates the postulated earthquake maximum magnitude according to prescribed rules, and then calculates ground motions at the site, using appropriate attenuation characteristics of soil/rock medium. Because this approach has not explicitly recognized uncertainties in geoscience parameters, Probabilistic Seismic Hazard Analysis (PSHA) methods have been developed [1] that allow explicit expressions for the uncertainty in ground motion estimates and provide a means for assessing sensitivity to various parameters.

This alternative probabilistic approach takes into account the developments in the field over the past two decades. Regulations for Part 50 for new nuclear power plants have been changed (10CFR100.23 and R.G. 1.165[2]) to require the use of probabilistic methods for determining design earthquakes.

What needs to be established is the appropriateness and the details of the use of probabilistic methods for determining a design earthquake for 10CFR Part 72. The Rulemaking Plan (SECY-98-126[3]) approved by the Commission in 1998, proposed conforming changes in Part 72 to promote consistency with the Part 50 regulations. This Rulemaking Plan will alleviate the need to request exemptions from the present part 72 regulations (72.102(f)), for use of the probabilistic methods, as submitted by the US Dept. of Energy (DOE), and approved[4] by the Commission. The DOE request was for storing TMI-2 debris at the Idaho National and Environmental and Engineering Laboratory (INEEL) facility in Idaho.

SUGGESTED CHANGES TO PART 72

Following changes in 10CFR Part 72 are suggested:

1. Require new Part 72 specific licensees to use the probabilistic methods for determining seismic vibratory ground motions (similar to 10 CFR 100.23), in lieu of 10 CFR 72.102(f), which requires the use of deterministic methods of 10 CFR Part 100 Appendix A. The general licensee applying for a specific license may be allowed an option to use either the suggested changes or the existing geological and seismologic design criteria for the NPP.
2. Maintain the present Part 72 requirement of using a single level of design earthquake, however, with a mean annual probability of exceedance of 5.0E-04.
3. Require that the design of cask storage pads and areas must adequately account for dynamic loads in addition to static loads.

The suggested approach would simplify the graded approach proposed in the Rulemaking Plan (SECY-98-126), for application to ISFSIs. This approach would continue to classify all SSCs important to safety in a single category because failure probabilities of SSCs and potential release of radioactivity for ISFSI facilities during a seismic event are low. This approach would be consistent with the Commission's strategic goals of risk-informing the regulations, and reducing unnecessary regulatory burden on the licensees, while maintaining the health and safety of the public. The rationale for the changes suggested above, including the technical basis for a single category of SSCs important to safety designed for a single Design Earthquake, presently contained in the Part 72 regulations, are discussed below.

RATIONALE FOR SUGGESTED CHANGES

The use of PSHA method or suitable sensitivity analyses will enable the applicants to account for the uncertainties in estimation of earthquake vibratory ground motion, and to determine the DE for design of an ISFSI facility. While the use of deterministic approaches (as currently required in Part 72) for siting ISFSIs has worked reasonably well in the past, in the sense that ISFSIs sited with this approach are judged to be suitably conservative, the approach has not explicitly addressed risks associated with uncertainties in geoscience parameters.

Probabilistic methods were developed in the past 15 to 20 years specifically for evaluation of seismic safety of nuclear facilities. These methods have been designed to allow explicit incorporation of different models for zonation, earthquake size, ground motion and other parameters. Significant experience has been gained by applying this methodology at nuclear facility sites throughout the United States. The WUS (west of approximately Longitude 104° West) and the EUS have fundamentally different tectonic environments and histories of tectonic deformation. The results of these applications identified the need to vary the fundamental PSHA methodology application depending on the tectonic environment of a site. The experience also served as the basis for developing guidelines[5] that are generally applicable for conducting a PSHA for nuclear facilities.

By accepting the use of a PSHA methodology in section 100.23, the Commission recognized that the uncertainties in seismological and geological information must be formally evaluated and appropriately accommodated in the determination of the SSE for seismic design of NPPs. The Commission, in promulgating Section 100.23, further recognized that the nature of uncertainty and the appropriate approach to account for it, depends on the tectonic environment of the site and on properly characterizing parameters input to the PSHA such as: seismic sources, the recurrence of earthquakes within a seismic source, the maximum magnitude of earthquakes within a seismic source, engineering estimation of earthquake ground motion, and the level of understanding of the tectonics. Therefore, methods other than probabilistic methods such as sensitivity analyses (examples of which are contained in Regulatory Guide 1.165) may be adequate for some sites to account for uncertainties.

DESIGN EARTHQUAKE

The graded approach of using two levels of earthquakes, as described above, was recommended in SECY-98-126 to make the Part 72 seismic and geological requirements consistent with the Parts 60/63 requirements for pre-closure surface facilities. This approach was evaluated in relation to the existing Part 72 regulations described above. Use of two levels of earthquakes for SSCs was evaluated: one at a lower level (return period of 1000 years) during normal operation for worker protection, and the other at a higher level (return period of 10000 years). It was concluded that this approach would not provide any benefit in the design of SSCs for ISFSI. It would only add to the design efforts without reduction in cost or risk. Instead of such an approach, it would be more efficient to maintain the present Part 72 approach of a single earthquake level (DE), but reduce the return period from 10,000 years to 2,000 years to account for a low level of risk in the design of ISFSIs SSCs. This is based on the following:

a. Use of a mean annual probability of exceedance of 5.0E-04 (return period of 2,000 years) for the DE is consistent with the Commission's approval of DOE's request for an exemption from section 72.102(f)(1) for a proposed ISFSI at the INEEL to store TMI-2 spent fuel generated at the Three Mile Island nuclear power plant. Section 72.102(f)(1) requires that for sites that have been evaluated under the criteria of Appendix A of Part 100, the DE must be equivalent to the SSE for an NPP. In its evaluation of the request, NRC staff considered the relative risk posed by the ISFSI. Considering the minor radiological consequences expected from a cask failure, and the lack of a credible mechanism to cause a failure, NRC found that the design earthquake using a mean annual probability of exceedance of 5.0E-04 for dry storage facilities at INEEL would be conservative.

b. The total probability of exceedance for a DE at an ISFSI facility with an operational period of 20 years (20 years x 5.0E-04 = 1.0E-02) is the same as the total probability of exceedance for an earthquake event at the pre-closure facility at Yucca Mountain with an operational period of 100 years (100 years x 1.0E-04 = 1.0E-02).

c. Important to safety SSCs in an ISFSI facility are few, and classifying them into one of two different categories for earthquake designs would unnecessarily increase the complexity in applications.

d. Dry cask storage system casks are designed for hypothetical accident conditions, such as drop and tip-over. These accidents create deceleration effects in the order of 40 g to 60 g compared to maximum seismic acceleration values in the order of 1 g to 2 g. Since the casks are rugged, they tend to have relatively high natural frequency; consequently damage from the drop or tip over accidents are expected to be far greater and more severe than the seismic inertia loads. Therefore, seismic inertia loads are bounded by other loads. The dry storage cask designs are very rugged and robust, and are expected to have substantial margin to withstand forces from a very severe seismic event with a mean annual probability of exceedance of 1.0E-04, or a return period of 10,000 years or greater.

e. During a seismic event, a cask may slide if lateral seismic forces are greater than friction resistance between the cask and the concrete pad. The sliding and resulting displacements due to earthquake ground motion are computed to demonstrate that the casks are spaced sufficiently apart to preclude impacts with other casks. The cask designer is also required to demonstrate that at a seismic event equal to the proposed DE, there will be no tip over. However, it follows from the previous discussion on the severity of accidental drop and tip over conditions, that there will be adequate margin for structural integrity of casks during a hypothetical seismic event greater than the proposed DE, even if the casks slide and come in contact with each other. Therefore, the structural integrity of the cask is will be maintained to meet the Part 72 exposure limits for radiological protection, even if the seismic event exceeds the proposed design earthquake with a mean annual probability of exceedance of 5.0E-04.

f. The mean annual probability of exceedance of 5.0E-04 for ISFSI facilities is consistent with the design approach used in DOE Standard DOE-STD-1020[6] for similar type facilities.

g. In comparison with a nuclear power plant, an operating ISFSI facility is a relatively simple facility in which the primary activities are waste receipt, handling, and storage. An ISFSI facility does not have the variety and complexity of active systems necessary to support an operating nuclear power plant. Therefore, the radiological risk associated with an ISFSI facility is significantly smaller than the risk associated with a nuclear power plant.

Based on the above, it is concluded that the present design philosophy of designing ISFSI SSCs for a single design basis accident level earthquake (called Design Earthquake) only is sufficient to ensure that the public health and safety are maintained. Use of an approach requiring further classification of SSCs into two different categories for earthquake designs, to account for the safety importance of SSCs and risk levels, will not result in benefit to the stakeholders. Instead of the graded approach, proposed in the Rulemaking Plan, the suggested change in the return period of the Design Earthquake to account for the robustness of the design of ISFSI SSCs is a realistic and a risk-informed approach.

A return period for the Design Earthquake of 2,000 years, with a mean annual probability of exceedance of 5E-04 is appropriate for ISFSI applications. The present design earthquake return period of approximately 10000 years is based on the nuclear plant requirements. In the Statements of Consideration accompanying the initial Part 72 Rulemaking, the Commission recognized that the design peak horizontal acceleration for SSCs need not be as high as for a nuclear power reactor, and should be determined on a "case-by-case" basis until "more experience is gained with licensing of these types of units." With over 10 years of experience in licensing dry cask storage, and analyses demonstrating robust behavior of casks in accident scenarios, it is appropriate to consider a different design value that is adequate for licensing dry storage ISFSIs.

RISK PERSPECTIVE AND EARTHQUAKE RISK ESTIMATE

Perspective on actual risk due to a DE event must include consideration of the frequencies (i.e., return period or a probability) as well as their consequences, as "risk" is defined as "the probability of an event multiplied by its consequences." Consequences of a radioactive dose at the controlled boundary of an ISFSI facility depend on a radioactive source term, the magnitude of which would be very small compared to the potential, however unlikely, at a nuclear power plant (e.g., from a postulated loss of coolant event). As such, the estimated consequences resulting from limited source term generation at an ISFSI facility would be correspondingly limited. This conclusion is consistent with the results of a preliminary risk assessment[7] by DOE of a conceptual repository design at Yucca Mountain, Nevada. Based on the study, DOE concluded that the highest estimated offsite dose was 2.1 rem for all external (earthquake, tornados, and flooding), and internal events (crane failures, cask drop events, transporter collisions, building or facility exhaust filter fires and exhaust filter bypass failure), with an associated probability of occurrence of 5.0E-07 per year. This is less than normally accepted fatal cancer risks of 1.0E-05 to 1.0E-06 per year from exposure to radiation would be acceptable to the individual members of the public.

Estimates provided by the National Council on Radiation Protection and Measurements[8] indicate the lifetime risk to individuals in the general population is .05 fatal cancers per S_v of exposure. Therefore, the lifetime risk of fatal cancer from 2.1 rem (.021 S_v) is 1.05E-03. Considering the probability of occurrence of these events of 5.0E-07 per year, the estimated risk of cancer fatality from these low probability events would be 5.3E-10. The International Commission on Radiological Protection notes[9] that, based on a review of information related to risks regularly accepted in everyday life for stochastic phenomena, a fatal cancer risk of 1.0E-05 to 1.0E-06 per year from exposure to radiation would be acceptable to the individual members of the public.

The study for a Private Fuel Storage Facility[10] for a hypothetical, non-mechanistic accident beyond the design basis shows that the off-site dose at 500 meters is 28.4 mrem and 75.9 mrem for a HI-STORM and TranStor cask, respectively. The HI-STORM cask analysis was based on 68 BWR fuel assemblies with 40 GWd/MTU burnup and 5 years cooling time, while the TranStor cask analysis was based on 61 BWR fuel assemblies with 40 GWd/MTU burnup and 6 years cooling time. The study assumes that the canister leaks at the maximum rate permitted by the closure helium leakage test acceptance criteria for 30 days, and that cladding of 100 percent of the fuel rods stored in the canister has ruptured. These consequences during a seismic event are not credible because the casks and its components are designed to withstand much more severe loads due to drop and tip-over events. Assuming conservatively that the dose varies inversely as the square of the distance, the estimated maximum off-site dose at 100 m would be 75.9 mrem x (500/100)2 = 1.88 rem. This would result in a lifetime risk to individuals in the general population of 9.4E-04-0fatal cancers. Considering the probability of occurrence of the earthquake event as 2,000 years or 5.0E-04 per year, the estimated risk of fatal cancer would be 4.7E-07 per year. This is less than a cancer risk of 1.0E-05 to 1.0E-06 per year from other stochastic phenomena in everyday life, as discussed above.

Based on the information provided above, it can be concluded that the risk to public health and safety from a seismic event beyond the design basis event is less than risks from other stochastic phenomena in everyday life.

The present requirements in 10 CFR 72.212(b)(2)(ii) for general licensees, regarding meeting the conditions set forth in the Certificate of Compliance, states that cask storage pads and areas must be designed to adequately support the static load of the stored cask. The cask storage pad and areas are subjected to dynamic loads during an earthquake event. Therefore, there is a need to clarify that the cask storage pads and areas are designed appropriately for static and dynamic loads. These requirements are included in NUREG-1536[11], Chapter 3, section V.2.b, and NRC Inspection Procedure 60851[12], and Inspection Procedure 60856[13], Appendix A.

In addition, depending on the properties of soil and structures, the free-field earthquake acceleration input loads may be amplified at the top of the storage pad. These amplified acceleration input values must be bounded by the design bases seismic acceleration values for the cask specified in the Certificate of Compliance. Liquefaction of the soil and instability during a vibratory motion due to an earthquake event may affect the cask stability, and thus must be addressed. Similar requirements exist in 10 CFR 72.102(c) for an ISFSI specific license.

CONCLUSIONS

Based on the above, following conclusions are made:

1. Probabilistic methods for seismic hazard analysis reflect the latest developments in the risk assessments for seismic events.
2. Considering the cask failure probabilities and radioactivity dose releases, use of a single Design Earthquake with a return period of 2000 years is a reasonable risk-informed approach.

REFERENCES

1. Lawrence Livermore National Laboratory, Investigation of Techniques for the Development of Seismic Design Basis Using the Probabilistic Seismic Hazard Analysis, NUREG/CR-6606,, April, 1998.
2. U.S. NRC, Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion, Regulatory Guide 1.165, 1996.
3. U.S. NRC, Rulemaking Plan: Geological and Seismological Characteristics for the Siting and Design of Dry Cask ISFSIs, 10 CFR Part 72 (SECY-98-126), 1998.
4. U.S. NRC, Exemption to 10 CFR 72.102(f)(1) Seismic Design Requirement for Three Mile Island Unit 2 Independent Spent Fuel Storage Installation (SECY-98-071), 1998.
5. Senior Seismic Hazard Analysis Committee, Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts, NUREG/CR- 6372, 1997.
6. U. S. Department of Energy, Natural Phenomena Hazards Design Evaluation Criteria for Department of Energy Facilities,, DOE-STD-1020-94, April 1994 (Change Notice # 1, January 1996).
7. U. S. Department of Energy, Site Characterization Plan, Yucca Mountain Site, Nevada Research and Development Area, Nevada, DOE/RW-0199, December, 1988.
9. International Commission on Radiological Protection, Recommendations of the International Commission on Radiological Protection, Publication 26, January, 1977.
8. National Council on Radiation Protection and Measurements (NCRP), Risk Estimates for Radiation Protection,, NCRP Report No. 115, December 31, 1993.
10. Private Fuel Storage L.L.C, Safety Analysis Report, Private Fuel Storage Facility, NRC Docket No. 72-22, Section 8.2.7, September 25, 2000.
11. U.S. NRC, Standard Review Plan for Dry Cask Storage Systems, NUREG-1536, Final Report, January, 1977.
12. U.S. NRC, 2000. NRC Inspection Manual, Inspection Procedure 60851, Design Control of ISFSI Components, U.S. Nuclear Regulatory Commission, May 22, 2000.
13. U.S. NRC, NRC Inspection Manual, Inspection Procedure 60856, Review of 10CFR72.212(b) Evaluations, November 30, 1999.

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