Where do we discuss buildings.

Glenn 4:27
7/24/00

Appendix 2b Structural Integrity of Spent Fuel Pools Subject to Seismic Loads

1. Introduction

The staff's concern regarding seismic issues at spent fuel pools involves very large earthquakes that can structurally fail the pool. Under this scenario, the pool will suffer a significant breach, it will drain rapidly, and it will be incapable of being refilled. This would lead to rapid cladding heat up followed by a zirconium cladding fire. The staff evaluated how large an earthquake would be required to cause such damage and what would be the ~~return~~ frequency of such large earthquakes. Attachment 1 to this appendix provides the checklist proposed by NEI and enhanced by the staff to assure adequate seismic capacity at SFPs for decommissioning sites that wish to be granted exemptions to ~~EP~~. Attachment 2 to this appendix provides the analysis of earthquake return periods from Dr. Robert Kennedy for nuclear power plant sites based on a 1.2 g spectral acceleration high confidence, with low probability of failure (HCLPF) value for spent fuel pools. to NRC requirements

Spent fuel pool structures at nuclear power plants are seismically robust. They are constructed with thick reinforced concrete walls and slabs lined with stainless steel liners 1/8 to 1/4 inch thick[1]. Pool walls vary from 4.5 to 5 feet in thickness, and the pool floor slabs are around 4 feet thick. The overall pool dimensions are typically about 50 feet long by 40 feet wide and 55 to 60 feet high. In boiling water reactor (BWR) plants, the pool structures are located in the reactor building at an elevation several stories above the ground. In pressurized water reactor (PWR) plants, the spent fuel pool structures are located outside the containment structure supported on the ground or partially embedded in the ground. The location and supporting arrangement of the pool structures determine their capacity to withstand seismic ground motion beyond their design basis. The dimensions of the pool structure are generally derived from radiation shielding considerations rather than structural needs. Spent fuel structures at operating nuclear power plants are able to withstand loads substantially beyond those for which they were designed.

~~The Commission asked the staff to determine if there were a risk-informed basis for providing exemptions for decommissioning plants and to provide a technical basis for potential rule making~~. After this, the staff began to investigate the capacity of spent fuel pools to withstand large earthquakes beyond the site's normal design bases. While performing the evaluation, it became apparent that the staff ~~does~~ did not have detailed information on how all the spent fuel pools were designed and constructed. Detailed fragility analyses of spent fuel pools were only available for a few plants. The staff originally performed a simplified bounding seismic risk analysis in its June 1999 draft assessment of decommissioning plant risks to help determine if there might be a seismic concern. The analysis indicated that seismic events could not be dismissed on the basis of a simplified bounding approach. In addition after further evaluation and discussions with stakeholders, it was determined that it would not be cost effective to perform a detailed plant-specific seismic evaluation for each spent fuel pool. Working with its stakeholders, the staff developed other tools that help assure the pools are sufficiently robust.

2. Return Period of SFP-Failing Earthquakes

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[1]Except for Dresden Unit 1 and Indian Point Unit 1, whose spent fuel pools do not have any liner plates. They were permanently shutdown more than 20 years ago, and no safety significant degradation of the concrete pool structure has been reported.

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Based on existing spent fuel pool fragility analyses and engineering judgement, the staff determined that a high confidence, low probability of failure (HCLPF)[2] value of 1.2 g peak spectral acceleration (or in terms of peak ground acceleration, which is not as good an estimator, 0.5 g PGA)[3] probably existed for most SFPs. Given this assumption, with the assistance of Dr. Robert P. Kennedy (See Appendix 2b, Attachment 2), it was determined that the annual frequency of seismically induced failures of spent fuel pool structures varies from less than $1.0X10^{-6}$ to $13.6X10^{-6}$ per year.

The staff used a measure of $3x10^{-6}$ per year for the adequacy of seismic return period in its earlier versions of the report. However, comments from the Advisory Committee on Reactor Safeguards and other stake holders indicated that the proposed measure and the approach the staff was using were too conservative. Also, the proposed approach contained different assessments for the Eastern and the Western United States and was complicated by the fact that seismic fragility information for ground motion levels beyond 0.5 g is not readily available from a peer reviewed data base.

The staff reexamined the results of Table 3, Appendix 2b, Attachment 2, which estimates the return frequencies of large earthquakes that could fail spent fuel pools. It was decided that the HCLPF value of 1.2 g peak spectral acceleration was a good measure of seismic adequacy for decommissioning plant SFPs that need only be tied to the return period of the earthquake and not to the safe shutdown earthquake magnitude for the site. The staff's review indicates that only three operating eastern plant sites have frequencies greater than $4.5X10^{-6}$ per year of having an earthquake with a peak spectral acceleration greater than 1.2 g. The staff finds $4.5X10^{-6}$ per year to be an acceptable criterion for seismic return period for earthquakes that could fail the spent fuel pools since it is a factor of 2 less than the $1x10^{-5}$ per year PPG and the estimated frequency of zirconium cladding fires from other initiators is an order of magnitude lower. Such a margin is warranted due to the uncertainties of the seismic hazard and spent fuel pool fragilities at each site.

3. Seismic Checklist

The staff determined that absent specific information about SFP seismic capacities, that some plant-specific evaluation of spent fuel pool capacity was warranted. During stakeholder interactions with the staff, the staff proposed the use of a seismic checklist that built on the work done for .... and could provide assurance of the capacity of spent fuel pools. In a letter dated August 18, 1999, NEI proposed a checklist that could be used to show robustness for a seismic ground motion with a peak ground acceleration (PGA) of approximately 0.5g. This checklist was reviewed and enhanced by the staff (See Appendix 2b, Attachment 1). Dr. Kennedy reviewed the enhanced checklist and concluded that the screening criteria are

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[2]The HCLPF value is defined as the peak seismic acceleration at which there is 95% confidence that less than 5% of the time the structure, system, or component will fail.

[3]Damage to critical structures, systems, and components (SSCs) does not correlate very well to peak ground acceleration (PGA) of the ground motion. However, damage correlates much better with the spectral acceleration of the ground motion over the natural frequency range of interest, which is generally between 10 and 25 Hertz for nuclear power plant SSCs. The spectral acceleration of 1.2 g corresponds to the screening level recommended in the reference document cited in the NEI checklist, and this special ordinate is approximately equivalent to a ground motion of 0.5 g PGA.

adequate for the vast majority of central and eastern U.S. sites. The seismic checklist was developed to provide a simplified method for demonstrating a HCLPF at an acceptably low value of seismic risk. The checklist includes elements to assure there are no weaknesses in the design or construction nor any service induced degradation of the pools that would make them vulnerable to failure under earthquake ground motions that exceed their design basis ground motion. Spent fuel pools that satisfy the seismic checklist, as written, would have a high confidence in a low probability of failure for seismic ground motions up to 0.5 g peak ground acceleration (1.2g peak spectral acceleration).

4. Seismic Risk - Support System Failure

In its preliminary draft report published in June 1999, the staff assumed that a ground motion three times the SSE was the HCLPF of the spent fuel pool. This meant that 95% of the time the pool would remain intact (i.e., would not leak significantly). The staff evaluated what would happen to spent fuel pool support systems (i.e., the pool cooling and inventory make-up systems) in the event of an earthquake three times the SSE. The staff modeled some recovery as possible (although there would be considerable damage to the area's infrastructure at such earthquake accelerations). The estimate in the preliminary report for the contribution from this scenario was $1x10^{-6}$ per year. In this report, this estimate has been refined based on looking at a broader range of seismic accelerations and further evaluation of the conditional probability of recovery under such circumstances. The staff estimates that for an average site in the northeast United States the return period of an earthquake that would damage a decommissioning plant's spent fuel pool cooling system equipment (assuming it had at least minimal anchoring) is about once in 4,000 years. The staff quantified a human error probability of $1x10^{-4}$ that represents the failure of the fuel handlers to obtain off-site resources. The event was quantified using the SPAR HRA technique. The probability shaping factors chosen were as follows: expansive time (> 50 times the required time), high stress, complex task because of the earthquake and its non-routine nature, quality procedures, poor ergonomics due to the earthquake, and finally a crew who had executed these tasks before, conversant with the procedures and one another. In combination we now estimate the risk from support failure due to seismic events to be on the order of $1x10^{-8}$ per year. The risk from support system failure due to seismic events is bounded by other more likely initiators.

5. Conclusion

The staff recommends that those plants that exceed $4.5X10^{-6}$ per year frequency for exceeding 1.2 g peak spectral acceleration in their spent fuel pool should be required to conduct plant-specific analysis beyond the confirmation of the checklist if they desire to obtain exemptions from EP, indemnification, or security at decommissioning sites. This process results in identification of four sites in the eastern U.S., only three of which are operating reactor sites - Pilgrim, H. B. Robinson, and Vogtle sites, with Maine Yankee the decommissioning site. In the western U.S., the Diablo Canyon and San Onofre sites are also beyond the scope of a simple screening evaluation. Based on the NRC sponsored study, "Seismic Failure and Cask Drop Analyses of the Spent Fuel Pools at Two Representative Nuclear power Plants," NUREG/CR 5176, January 1989, the seismic HCLPF capacity of the H. B. Robinson spent fuel pool has been estimated to be 0.65 g. For the Vogtle, Pilgrim, Diablo Canyon, and San Onofre sites, it may be necessary for the utilities to conduct a detailed site-specific seismic risk evaluation if they desire an exemption from EP when the site is in decommissioning.

To summarize the staff recommendation for seismic vulnerability of spent fuel pools, (1) all sites must conduct an assessment of the spent fuel pool structures using the revised seismic check list in order to identify any structural degradation, potential for seismic interaction from superstructures and over head cranes, and to verify that they have a seismic HCLPF value of 0.5 g or higher, (2) those sites that cannot demonstrate that a seismic HCLPF value exists, may either under take appropriate remedial action or conduct site-specific seismic risk assessment and (3) Pilgrim, H. B. Robinson, Vogtle, Diablo Canyon and San Onofre sites would have to use the seismic check list to identify any structural degradation or other anomalies and then conduct a site specific seismic risk assessment if they desire an exemption from EP when their sites are in decommissioning.

Appendix 2b Structural Integrity of Spent Fuel Pools Subject to Seismic Loads

1. Introduction

As a part of the Generic Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools," NRC has studied the hypothetical event of an instantaneous loss of spent fuel pool water. The recommendation from a study in support of this generic issue indicates that a key part of a plant specific evaluation for the effect of such an event, is the need to obtain a realistic seismic fragility of the spent fuel pool. The failure or the end state of concern in the context of this generic issue is a catastrophic failure of the spent fuel pool which leads to an almost instantaneous loss of all pool water and the pool having no capacity to retain any water even if it were to be reflooded.

Spent fuel pool structures at nuclear power plants are constructed with thick reinforced concrete walls and slabs lined with stainless steel liners 1/8 to 1/4 inch thick. Dresden Unit 1 and Indian Point Unit 1 are exceptions to this in that these two plants do not have any liner plates. They were decommissioned more than 20 years ago and no safety significant degradation of the concrete pool structure has been reported. The spent fuel pool walls vary from 4.5 to 5 feet in thickness and the pool floor slabs are approximately 4 feet thick. The overall pool dimensions are typically about 50 feet long by 40 feet wide and 55 to 60 feet high. In boiling water reactor (BWR) plants, the pool structures are located in the reactor building at an elevation several stories above the ground. In pressurized water reactor (PWR) plants, the spent fuel pool structures are located outside the containment structure and are supported on the ground or partially embedded in the ground. The location and supporting arrangement of the pool structures help determine their capacity to withstand seismic ground motion beyond their design basis. The dimensions of the pool structure are generally derived from radiation shielding considerations rather than structural needs. Spent fuel structures at operating nuclear power plants are inherently rugged in terms of being able to withstand loads substantially beyond those for which they were designed. Consequently, they have significant seismic capacity.

2. Seismic Checklist

In the preliminary draft report published in June 1999, the staff assumed that the spent fuel pools were robust for seismic events less than about three times the safe shutdown earthquake (SSE). It was assumed that the high confidence, low probability of failure (HCLPF)[1] value for pool integrity is 3 times SSE. For most Central and Eastern U.S. (CEUS) sites, 3 X SSE is in the peak ground acceleration (PGA) range of 0.35 to 0.5 g (where g is the acceleration of gravity). Seismic hazard estimates developed by the Lawrence Livermore National Laboratory (NUREG-1488) show that, for most CEUS plants, the mean frequency for a PGA equal to 3 X SSE is less than 2E-5 per year. For western plants, the mean frequency for PGA equal to 2 X SSE is equivalently small.

These low probabilistic frequency-of-occurrence estimates are supported by deterministic considerations. The design basis earthquake ground motion, or the SSE ground motion, for

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[1]A HCLPF is the peak acceleration value at which there is 95% confidence that less than 5% of the time the structure, system or component will fail.

nuclear power plant sites were based on the assumption of the largest event geophysically ascribable to a tectonic province or a capable structure at the closest proximity of the province or fault to the site. In the case of the tectonic province in which the site is located, the event is assumed to occur at the site. For the eastern seaboard, the Charleston event is the largest magnitude earthquake and current research has established that such large events are confined to the Charleston region. The New Madrid zone is another zone in the central US where very large events have occurred. Recent research has identified the source structures of these large New Madrid earthquakes. Both of these earthquake sources are fully accounted for in the assessment of the SSE for currently licensed plants. The SSE ground motions for nuclear power plants are based on conservative estimates of the ground motion from the largest earthquake estimate to be generated under the current tectonic regime. The seismic hazards at the west coast sites are generally governed by known active fault sources, consequently, the hazard curves, which are plots of ground acceleration versus frequency of occurrence, have a much steeper slope near the higher ground motion end. In other words, as the amplitude of the seismic acceleration increases, the probability of its occurrence decreases rapidly. Therefore it is reasonable to conclude that the frequency of ground motion exceeding 3 X SSE for CEUS plants and
2 X SSE for western plants is less then 2E-5 per year.

Several public meetings were held from April to July 1999 to discuss the staff's draft report. At the July public workshop, the NRC proposed, and the industry group agreed to develop a seismic checklist, which could be used to examine the seismic vulnerability of any given plant. In a letter dated August 18, 1999, the Nuclear Energy Institute (NEI) proposed a checklist which is based on assuring a robustness for a seismic ground motion with a PGA of approximately 0.5g. A copy of this submittal is included in Appendix 5a.

The NRC contracted with Dr. Robert P. Kennedy to perform an independent review of the seismic portion of the June draft report, as well as the August 18, 1999, submittal from NEI. Dr. Kennedy's comments and recommendations were contained in an October 1999 report entitled "Comments Concerning Seismic Screening and Seismic Risk of Spent Fuel Pools for Decommissioning Plants," which is included as Appendix 5b of this report. Dr. Kennedy raised three significant concerns about the completeness of the NEI checklist.

The results of Dr. Kennedy's review, as well as staff comments on the seismic checklist, were forwarded to NEI and other stakeholders in a December 3, 1999, memorandum from Mr. William Huffman (Appendix 5c). In a letter from Mr. Alan Nelson, dated December 13, 1999 (Appendix 5d), NEI submitted a revised checklist, which addressed the comments from Dr. Kennedy and the NRC staff. Dr. Kennedy reviewed the revised checklist, and concluded in a letter dated December 28, 1999 (Appendix 5f), that the industry seismic screening criteria are adequate for the vast majority of CEUS sites.

3. Seismic Risk - Catastrophic Failure

The preliminary risk assessment report published in June 1999 used an approximate method for estimating the risk of spent pool failure. It was assumed that the HCLPF value for the pool integrity is 3 times SSE. For most CEUS sites, 3 X SSE has a ground motion with a PGA range of 0.35 to 0.5 g. Seismic hazard curves from the Lawrence Livermore National Laboratory (NUREG-1488) show that, for most CEUS sites, the mean frequency for PGA equal to 3 X SSE is less than 2E-5. For western plants, the mean frequency of ground motion exceeding 2 X SSE is comparably small. In the June report, the working group used the approximation that the frequency of a seismic event that will challenge the spent fuel pool integrity is 5% of 2E-5, or a value of 1E-6.

Dr. Kennedy, in his October 1999 report, pointed out that this approximation is nonconservative for CEUS hazard curves with shallow slopes; i.e., where an increase of more than a factor of two in ground motion is required to achieve a 10-fold reduction in annual frequency of exceedance. Dr. Kennedy proposed a calculation method, which had previously been shown to give risk estimates that were 5 to 20% conservative when compared to more rigorous methods, such as convolution of the hazard and fragility estimates. Using this approximation, Dr. Kennedy estimated the spent fuel pool failure frequency for a site with HCLPF of 1.2[2] peak spectral acceleration if sited at each of the 69 CEUS sites. A total of 35 sites had frequencies exceeding 1E-6 per year, and eight had frequencies in excess of 3E-6 per year. The remaining sites had frequencies below 1E-6[3]. Dr. Kennedy's report notes that spent fuel pools that pass the appropriately defined screening criteria are likely to have capacities higher than the screening level capacity. Thus, the frequencies quoted above are upper bounds.

For those CEUS plants where the ground motion of 3 X SSE is less than or equal to the NEI screening criterion of 0.5g PGA, the staff concludes that the risk is acceptably low. A similar conclusion can be drawn for western plants where the ground motion at 2 X SSE is within the screening criterion. For CEUS plants where 3 X SSE exceeds the screening criterion, a detailed assessment will be required to demonstrate that the pool HCLPF equals 3 X SSE. A similar conclusion can be drawn for western plants where the ground motion at 2 X SSE exceeds the screening criterion.

The staff has no estimate of the seismic risk for decommissioning plants at sites west of the Rockies. However, based on considerations described above, the staff estimates that western plants which can demonstrate a HCLPF greater than 2 X SSE will have an acceptably low estimate of risk.

In his letter of December 28, 1999, Dr. Kennedy concurred that this performance goal assures an adequately low seismic risk for the spent fuel pool.

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[2]Damage to critical SSCs does not correlate very well to PGA of the ground motion. However, damage correlates much better with the spectral acceleration of the ground motion over the natural frequency range of interest, which is generally between 10 and 25 hertz for nuclear power plants SSCs. The spectral acceleration of 1.2g corresponds to the screening level recommended in the reference document cited in the NEI checklist, and this spectral ordinate is approximately equivalent to a ground motion with 0.5g PGA.

[3]These estimates are based on the Lawrence Livermore National Laboratory 1993 (LLNL 93) seismic hazard curves. Recently, the Senior Seismic Hazard Analysis Committee (SSHAC) published NUREG-CR-6372, "Recommendation for Probabilistic Seismic Hazard Analysis: Guidance On Uncertainty and Use of Experts." The report gives guidance on future application of seismic hazards. However, site specific hazard estimates have not been performed for all sites with the new method.

4. Seismic Risk - Support System Failure

In its preliminary draft report published in June 1999, the staff assumed that a ground motion three times the SSE was the HCLPF of the spent fuel pool. This meant that 95% of the time the pool would remain intact (i.e., would not leak significantly). The staff evaluated what would happen to spent fuel pool support systems (i.e., the pool cooling and inventory make-up systems) in the event of an earthquake three times the SSE. We modeled some recovery as possible (although there would be considerable damage to the area's infrastructure at such earthquake accelerations). The estimate in the preliminary report for the contribution from this scenario was 1×10^{-6} per year. In this report, this estimate has been refined based on looking at a broader range of seismic accelerations and further evaluation of the conditional probability of recovery under such circumstances. The staff estimates that for an average site in the northeast United States the return period of an earthquake that would damage a decommissioning plant's spent fuel pool cooling system equipment (assuming it had at least minimal anchoring) is about once in 4,000 years. The staff quantified a human error probability of 1×10^{-4} that represents the failure of the fuel handlers to obtain off-site resources. The event was quantified using the SPAR HRA technique. The probability shaping factors chosen were as follows: expansive time (> 50 times the required time), high stress, complex task because of the earthquake and its non-routine nature, quality procedures, poor ergonomics due to the earthquake, and finally a crew who had executed these tasks before, conversant with the procedures and one another. In combination we now estimate the risk from support failure due to seismic events to be on the order of 1×10^{-8} per year. The risk from support system failure due to seismic events is bounded by other more likely initiators.

5. Conclusion

Spent fuel pools that satisfy the seismic checklist, as written, would have a high confidence in a low probability of failure for seismic ground motions up to 0.5 g peak ground acceleration (1.2g peak spectral acceleration). Thus, sites in the central and eastern part of the U.S. that have three times SSE values less than or equal to 0.5 g PGA and pass the seismic check list would have an acceptably low level of seismic risk. Similarly, West Coast sites that have two times SSE values less than 0.5 g. and pass the seismic check list would have acceptably low values of seismic risk. From a practical point of view, a limited number of sites in the central and eastern part of the U.S. have three times SSE values greater than 0.5g; the two times SSE values exceed 0.5g for two West Coast plants. In order to demonstrate acceptably low seismic risk, those central and eastern sites for which the three times SSE values exceed 0.5g and the two West Coast sites would have to perform additional plant specific analyses to demonstrate HCLPF for their spent fuel pools at three times SSE and two times SSE values of ground acceleration, respectively. For these sites the frequency of failure is bounded by 3×10^{-6} per year, and other considerations indicate the frequency may be significantly lower. Plants which cannot demonstrate HCLPF values equivalent to 3 X SSE or 2 X SSE as appropriate may perform a risk assessment to demonstrate acceptably low frequency of SFP failure.

Appendix 2c Structural Integrity of Spent Fuel Pool Structures Subject to Heavy Loads Drops

1. Introduction

A heavy load drop into the spent fuel pool (SFP) or onto the spent fuel pool wall can affect the structural integrity of the spent fuel pool. A loss-of-inventory from the spent fuel pool could occur as a result of a heavy load drop. For single failure proof systems where load drop analyses have not been performed at decommissioning plants, the mean frequency of a loss-of-inventory caused by a cask drop was estimated to be $2.0x10^{-7}$ per year (assuming 100 lifts per year). For a non-single failure proof handling system where a load drop analysis has not been performed, the mean frequency of a loss-of-inventory event caused by a cask drop was estimated to be $2.1x10^{-5}$ per year. The staff believes that performance and implementation of a load drop analysis that has been reviewed and approved by the staff will substantially reduce the expected frequency of a loss-of-inventory event from a heavy load drop for either a single failure proof or non-single failure proof system.

2. Analysis

The staff revisited NUREG-0612[1] [Ref. 1] to review the evaluation and the supporting data available at that time to determine its applicability to and usefulness for evaluation of heavy load drop concerns at decommissioning plants. In addition, three additional sources of information were identified by the staff and used to reassess the heavy load drop risk:

(1) U.S. Navy crane experiences (1990s Navy data) for the period 1996 through mid-1999,

(20) WIPP/WID-96-2196 [Ref. 2], "Waste Isolation Pilot Plant Trudock Crane System Analysis," October 1996 (WIPP)

(21) NEI data on actual spent fuel pool cask lifts at U.S. commercial nuclear power plants [Ref.3]

The staff's first area of evaluation was the frequency of heavy load drops. The number of occasions (incidents) where various types of faults occurred that potentially could lead to a load drop was investigated. Potential types of faults investigated included improper operation of equipment, improper rigging practices, poor procedures, and equipment failures. Navy data from the 1990s were compared to the data used in NUREG-0612. The data gave similar, but not identical, estimates of the various faults leading to heavy load drops (See Table A2c-1.) The NEI cask handling experience also supported the incident data used in this evaluation, and in NUREG-0612. Once the frequency of heavy load drops was estimated (i.e., load drops per lift), the staff investigated the conditional probability that such a drop would seriously damage the spent fuel pool (either the bottom or walls of the pool) to the extent that the pool would drain very rapidly and it would not be possible to refill it using onsite or offsite resources. To do this the staff used fault trees taken from NUREG-0612 (See Figure A2c-1.) By mathematically

[1]NUREG-0612 documented the results of the staff's review of the handling of heavy loads at operating nuclear power plants and included the staff's recommendations on actions that should be taken to assure safe handling of heavy loads.

combining the frequency of load drops with the conditional probability of pool failure given a load drop, the staff was able to estimate the frequency of heavy load drops causing a zirconium fire at decommissioning facilities.

3. Frequency of Heavy Load Drop

The database used in this evaluation (primarily the 1990s Navy data) considered a range of values for the number of occasions where faults occurred, the frequency of heavy load drops and the availability of backup systems. The reason that there is a range of values is that while the number of equipment failures and load drops were reported, the denominator of the estimate, the actual total number of heavy load lifts, was only available based on engineering judgement. High and low estimates of the ranges were made, and it was assumed that the data had a log normal distribution with the high and low number of the range representing the 5th and 95th percentile of the distribution. From this the mean of the distribution was calculated. Data provided by NEI on actual lifts and setdowns of spent fuel pool casks at commercial U.S. nuclear power plants (light water and gas-cooled reactors) gave a similar estimated range for the incidents at the 95 percent confidence level.

Load drops were broken down into two categories: failure of lifting equipment and failure to secure the load.

Crane failures (failure of lifting equipment) were evaluated using the fault tree shown in Figure A2c-1, which comes from NUREG-0612. At the time that heavy loads were evaluated in NUREG-0612, low density storage racks were in use and after 30 to 70 days (a period of about 0.1 to 0.2 per year), no radionuclide releases were expected if the pool were drained. It was assumed in NUREG-0612 that after this period, the fuel gap noble gas inventory had decayed and no zirconium fire would have occurred. Today, most decommissioning facilities use high density storage racks. This analysis evaluates results at one year after reactor shutdown. Our engineering evaluations indicate that for today's fuel configurations, burnup, and enrichment, a zirconium cladding fire may occur if the pool were drained during a period as long as five years.

A literature search performed by the staff searching for data on failure to secure loads identified a study (WIPP report) that included a human error evaluation for improper rigging. This study was used by the staff to re-evaluate the contribution of rigging errors to the overall heavy load (cask) drop rate and to address both the common mode effect estimate and the 1990s Navy data. Failure to secure a load was evaluated in the WIPP report for the Trudock crane. The WIPP report determined that the most probable human error was associated with attaching the lifting legs to the lifting fixture. In the WIPP report, the failure to secure the load (based on a 2-out-of-3 lifting device) was estimated based on redundancy, procedures, and a checker. The report assumed that the load could be lowered without damage if no more than one of the three connections were not properly made. Using NUREG/CR-1278 [Ref. 4] information, the mean failure rate due to improper rigging was estimated in the WIPP report to be 8.7×10^{-7} per lift. Our requantification of the NUREG-0612 fault tree using the WIPP improper rigging failure rate is summarized in Table A2c-2. The WIPP evaluation for the human error probabilities is summarized in Table A2c-3.

These estimates provided a rate for failures per lift. Based on input from the nuclear industry at the July 1999 SFP workshop, we assumed in our analysis that there will be a maximum of 100 cask lifts per year at a decommissioning plant.

4. Evaluation of the Load Path

Just because a heavy load is dropped does not mean that it will drop on the spent fuel pool wall or on the pool floor. It may drop at other locations on its path. A load path analysis is plant-specific. In NUREG-0612 it was estimated that the heavy load was near or over the spent fuel pool for between 5% and 25% of the total path needed to lift, move, and set down the load. It was further estimated that if the load were dropped from 30 feet or higher (or in some circumstances from 36 feet and higher depending on the assumptions) when it is over the pool floor, and if a plant-specific load drop analysis had not been performed[2], then damage to the pool floor would result in loss-of-inventory. In addition we looked at the probability that the load drop occurred over the pool wall from eight to ten inches above the edge of the pool wall. In our analysis we evaluated the chances the load was raised sufficiently high to fail the pool and evaluated the likelihood that the drop happened over a vulnerable portion of the load path. Table A2c-2 presents the results for a heavy load drop on or near the spent fuel pool. Based on NUREG-0612, if the cask were dropped on the spent fuel pool floor, the likelihood of a loss-of-inventory given the drop is 1.0. Based on the evaluation presented in NUREG/CR-5176 [Ref. 5], if the load were dropped on the spent fuel pool wall, the likelihood of a loss-of-inventory given the drop is 0.1.

5. Conclusion

Our heavy load drop evaluation is based on the method and fault trees developed in NUREG-0612. New 1990s Navy data were used to quantify the failure rate of the lifting equipment. The WIPP human error evaluation was used to quantify the failure to secure the load. We estimated the mean frequency of a loss-of-inventory from a cask drop onto the pool floor or onto the pool wall from a single failure proof system to be 2.0×10^{-7} per year for 100 lifts per year.

However, only some of the plants that will be decommissioning plants in the future currently have single failure proof systems. Historically, many facilities have chosen to upgrade their crane systems to become single failure proof. However, this is not an NRC requirement. The guidance in NUREG-0612, phase 2 calls for systems to either be single failure proof or if they are non-single failure proof to perform a load drop analysis. The industry through NEI has indicated that it is willing to commit to follow the guidance of all phases of NUREG-0612.

For licensees that choose the non-single failure proof handling system option in NUREG-0612, we based the mean frequency of a loss-of-inventory event on the method used in NUREG-0612. In NUREG-0612, an alternate fault tree than that used for the single failure proof systems was used to estimate the frequency of exceeding the release guidelines (loss-of-inventory) for a non-single failure proof system. We calculated the mean frequency of

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[2] If a load drop analysis were performed, it means that the utility has evaluated the plant design and construction to pick out the safest path for the movement of the heavy load. In addition, it means that the path chosen has been evaluated to assure that if the cask were to drop at any location on the path, it would not catastrophically fail the pool or its support systems. If it is determined that a portion of the load path would fail if the load were dropped, the as-built plant must be modified (e.g., by addition of an impact limiter or enhancement of the structural capacity of that part of the building) to be able to take the load drop or a different safe load path must be identified.

catastrophic pool failure (for drops into the pool, or on or near the edge of the pool) for non-single failure proof systems to be about $2.1x10^{-5}$ per year when corrected for the 1990s Navy data and 100 lifts per year. This estimate exceeds the proposed pool performance guideline of $1x10^{-5}$ per year. The staff believes that a licensee which chooses the non-single failure proof handling system option in NUREG-0612 can reduce this estimate to the same range as that for single failure proof systems by performing a comprehensive and rigorous load drop analysis. The load drop analysis is assumed to include implementation of plant modifications or load path changes to assure the spent fuel pool would not be catastrophically damaged by a heavy load drop.

References:

(1) U.S. Nuclear Regulatory, "Control of Heavy Loads at Nuclear Power Plants, Resolution of Generic Technical Activity A-36," NUREG-0612, July 1980.

(2) Pittsburgh, Westinghouse, P.A., and Carlsbad, WID, N.M., "Waste Isolation Pilot Plant Trudock Crane System Analysis," WIPP/WID-96-2196, October 1996.

(3) Richard Dudley, NRC memorandum to Document Control Desk, "Transmittal of Information Received From the Nuclear Energy Institute (NEI) For Placement InThe Public Document Room," dated September 2, 1999.

(4) Swain, A.D., and H.E. Guttmann, "Handbook of Reliability Analysis with Emphasis on Nuclear Power Plant Applications," NUREG/CR-1278, August 1983.

(5) P.G. Prassinos, et al., "Seismic Failure and Cask Drop Analyses of Spent Fuel Pools at Two Representative Nuclear Power Plants," NUREG/CR-5176, LLNL, January 1989.