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SEISMIC RE-EVALUATION OF US NUCLEAR POWER PLANTS

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ABSTRACT

1. INTRODUCTION

1.1 Background

The United States Nuclear Regulatory Commission (NRC) undertook a seismic re-evaluation of all operating US nuclear power plants as part of a major safety initiative. The initiative was an outgrowth of a Commission policy statement issued in August, 1985, on severe accidents in nuclear power plant. In the policy statement, the Commission concluded, based upon existing information, that existing nuclear power plants pose no undue risk to the public health and safety and that there is no present basis for immediate action on generic rulemaking or other regulatory requirement for these plants. However, the Commission recognized, based on NRC and industry experience with plant-specific probabilistic risk assessments (PRAs), that systematic examinations are valuable for identifying plant-specific vulnerabilities to severe accidents that could be fixed with low-cost improvements. Therefore, the Commission decided that each existing nuclear power plant should perform a systematic examination to identify any plant-specific vulnerabilities to severe accidents and report the results to the Commission. To implement this decision, the Commission issued Generic Letter 88-20 in November 1988 requesting that each licensee conduct an Individual Plant Examination (IPE) for internally initiated events including internal flooding. In some instances, risk from external events could be a significant contributor to core damage. In June 1991, the Commission requested that each licensee perform an evaluation of external events.

1.2 Objectives

The stated objectives of the IPE were for each licensee:

1. To develop an appreciation of the severe accident response of each facility,

2. To understand the most likely severe accident sequences that could occur at the plant,
3. To gain a quantitative understanding of the overall probabilities of core damage and fission product releases, and
4. To reduce, if necessary, the overall probabilities of core damage and fission product releases, by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

The Commission expected that the achievement of these goals would help verify that, at US nuclear power plants, severe core damage and large radioactive release probabilities are consistent with the Commission=s Safety Goal Policy Statement.

1.3 Individual Plant Examination of External Events

Examination of externally initiated events (e.g., external floods, high winds, and earthquakes) proceeded separately on a later schedule:

5. To permit the identification of which external events need a systematic examination,
6. To permit development of simplified examination procedures, and
7. To integrate other ongoing Commission programs that deal with various external event evaluations (e.g., the Seismic Design Margins Program.

In December 1987, the NRC formed an External Events Steering Group to provide recommendations to the Commission on the scope, methods, and implementation of an Individual Plant Examination of External Events (IPEEE). The Steering Group prepared recommendations covering seismic events, internal fires, high winds, tornadoes, and eight other potential external initiators. (Only earthquake will receive further consideration in this paper.) Based on Steering Group recommendations, the staff published NUREG-1407, entitled AProcedural and Submittal Guidance for the Individual Plant examination of External Events (IPEEE) for Severe Accident Vulnerabilities@ in June 1991 following publication as a draft for comment and holding a public workshop to explain the content and methodologies of an IPEEE, as well as to obtain comment.

The NRC staff recognized at the outset of the IPEEE effort that the external initiators could not necessarily be treated in exactly the same way as the internal initiators in the implementation of the Severe Accident Policy because the sources and treatment of uncertainties in the estimates of core damage frequencies for external and internal initiators can be quite different. In addition, some methods endorsed by the staff for evaluating external hazards and identifying vulnerabilities do not produce estimates of core damage frequency. For example, seismic margins methods produce estimates of seismic hazard levels of high-confidence, low-probability of failure (HCLPF) for a plant rather than estimates of core damage probability.

Therefore, the staff determined that an explicit estimate of the core damage frequency was not needed to meet the intent of the Severe Accident Policy and would not be a condition for the IPEEE. Thus, the third objective above would be addressed only indirectly by some method acceptable for use in the IPEEE. Not withstanding, this change from the IPE the key objective of gaining an understanding of plant response through a systematic examination process could be accomplished.

2. PROCESS

The IPEEE was not the first seismic re-evaluation of operating nuclear power plants in the U.S. The NRC recognized the seismic safety criteria had undergone a period of very rapid evolution between the 1960s and the mid-1970s during the licensing of the first nuclear power plants. The Commission initiated the Systematic Evaluation Program in 1977 to document the technical bases and criteria for these first power plants by comparing them to the design criteria for the power plants currently being reviewed and licensed. The seismic safety of these older power plant was one of the important issues evaluated. The program conducted an evaluation of 11 older power plants, which had been built between 1956 and 1967, against then current criteria. The three most relevant new criteria were Appendix A to 10 CFR Part 50 (General Design Criteria), Appendix A to 10 CFR Part 100 (Seismic and Geologic Siting Criteria), and the seismic portions of the Standard Review Plan. This review program and its results provided part of the impetus for the development of techniques to simplify the re-evaluation of seismic issues for existing nuclear power plants and by extension nuclear facilities, in general.[1]

2.1 Acceptable Examination Options

The NRC staff identified in NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination for External Events (IPEEE) for Severe Accident Vulnerabilities," two methodologies acceptable to the staff for performing an IPEEE to identify potential seismic vulnerabilities at nuclear power plants. The first acceptable methodology is a Level 1 seismic probabilistic risk assessment (SPRA) with a qualitative containment performance analysis. The second acceptable methodology is a seismic margins assessment (SMA). The staff developed the IPEEE guidance with the recognition that the SPRA approach can differ substantially from the SMA approaches in terms of procedures, objectives, results, and the format used to present the results. However, in terms of satisfying the objectives of the seismic IPEEE, both the SPRA and SMA approaches involve a systematic, comprehensive walkdown of important components and systems and an evaluation of equipment anchorage and spatial interactions. Both are capable of identifying plant vulnerabilities.

2.1.1 Seismic PRA

The objective of a Level 1 SPRA is to provide a measure of the seismic by estimating the core damage frequency (CDF) due to seismic initiators. The SPRA is initiated by estimating the seismic hazard to a plant by defining the annual frequency of exceeding a relevant ground motion parameter, such as peak acceleration, which can be correlated with damage to critical structures, systems, and components (SSC). For each SSC, a seismic fragility curve is estimates in terms of the selected ground motion parameter. Conditional failure probabilities of SSCs for various ground motion levels are then used in the Boolean logic of the accident sequence analysis to obtain conditional core damage probabilities or a core damage fragility curve. This result is convolved with the exceedance frequency of the hazard estimates to obtain a seismic core damage frequency estimate.

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[1] The development of probabilistic seismic hazard assessment received considerable momentum at this time as a quantitative tool for selecting the order of the next set of older power plants to undergo a review under the Systematic Evaluation Program.

The NRC staff has conducted in-depth reviews of previous SPRAs: from these reviews the staff has concluded that in certain areas the treatment has been inconsistent or absent. Thus, the staff requested the following enhancements for new SPRAs.

- Plant Walkdowns - performed to find as-designed, as-built, and as-operated seismic weaknesses
- Relay Chatter - includes compnents such as, electric relays, contactors, and switches that are prone to chatter
- Liquefaction - the potential for soil liquefaction and associated effects needs to be examined for sites with specific soil conditions - impact on plant operations should be assessed for both potential for and conseqiences of liquefaction
- HCLPF Calculations (Optional) - licensees can report plant-level, sequence-level, and component-level HCLPF values and use this information to support decisions to the idenification and listing of vulnerabilities

For the use of an existing SPRA, an additional enhancement is recommended.

- Seismic Hazard Selection - at sites east of the Rocky Mtns that did not opt to use either the LLNL or EPRI mean hazard estimates, sensitivity studies should be conducted to determine if the use of these results would affect the delineation or ranking of seismic sequences. Similarly for Western U.S. sites, sensitivity studies should be carried out to determine the effect of uncertainty in hazard on delineation and ranking of seismic sequences.

The principal products of an SPRA include:

- an estimate of the seismic CDF,
- a list of the dominant contributors to the seismic CDF, and
- a probabilistic plant-level capacity, i.e., a fragility curve.

2.1.2 Seismic Margins

The objective of an SMA is to describe the additional seismic margin nuclear power plants have, by virtue of their conservative design, to withstand earthquakes larger than their design basis earthquake, i.e., larger than the SSE. This margin may be defined in terms of the high confidence of a low probability of failure (HCLPF) capacity of each of the critical SSCs. A working definition of the HCLPF capacity is that it corresponds to the 95% confidence that there is less than a 5% probability of failure. The plant damage state and the plant HCLPF can then be developed by either the an event/fault tree approach to delineate accident sequences as in the NRC approach to SMAs or by a systems “success path” approach developed by the Electric Power Research Institute (EPRI).

2.1.2.1 NRC Methodology

The NRC method is based on an event/fault tree approach to delineate accident sequences. For example for PWRs, two safety functions are considered to be most important to plant seismic safety: reactor subcriticality and early emergency core cooling. If these functions are ensured for a given earthquake, there is high confidence that core damage would not occur at that level. Further, in an NRC SMA, systems modeling is performed in a manner similar to an SPRA, whereas evaluation of component HCLPF capacities may involve a deterministic or probabilistic assessment. The plant-level HCLPF capacity in an NRC SMA is determined from component-level HCLPFs using a Boolean expression for core damage and a simple min-max approach (minimum HCLPF among “OR” events, and maximum HCLPF among “AND” events).

2.1.2.2 EPRI Methodology

An EPRI SMA involves identifying success paths and finding the weakest link along each success path. This approach defines and evaluates the capacity of those components required to bring the to a stable condition (either hot or cold shutdown) and to maintain that condition for at least 72 hours. The component having the lowest HCLPF capacity comprise the weak links, and they determine the plant-level HCLPF. The EPRI SMA calls for evaluation of a preferred success path and an alternate path. The NRC staff recommended that a reasonably complete set of potential success paths be set down initially, rather than a very small number, since limiting the number of success paths to quickly can prevent the identification of some plant-level HCLPF insights, and mask plant differences regarding defense-in-depth. The staff believed that preliminary analysis to narrow the number of paths to the required two or three should begin with fuller set, and it recommended that this narrowing be documented in detail. For IPEEE objectives, it is desirable that, to the extent practical, the alternative path involve operational sequences, systems, piping runs, and components different from those used in the preferred path.

2.2 Role of Probabilistic Seismic Hazard Assessments

At the time of the development of the IPEEE guidance for seismic initiators, two highly sophisticated probabilistic seismic hazard studies were available for sites in the United States, east of the Rocky Mountains. The development of one had been sponsored by the NRC with Lawrence Livermore National Laboratory (LLNL) as its contractor; the other was developed by EPRI with the Seismicity Owners Group as the sponsor. For many sites, these studies yield significantly different hazard estimates at low probability and high-level ground motions. There was significant concern among the staff that these differences in the seismic hazard estimate could lead to significant differences in IPEEE results. However, the initial PRAs conducted using these two estimate sets indicate that, despite the large differences in the absolute level of the estimates, the identification, ranking, and relative contributions of the dominant seismic sequences are virtually the same for both the LLNL and the EPRI hazard estimates. This equivalence is apparently due to the fact that the slopes of seismic hazard curves are not significantly different over those ground motion levels that, in conjunction with the fragilities, control the relative distribution of the seismically induced core damage frequencies.

In its guidance, the staff warned that there was no guarantee that this would be the case for all sites in the eastern United States. To address this potential issue, the staff noted that while a full seismic hazard uncertainty analysis is not necessary in performing an SPRA for the IPEEE, it preferred that the mean hazard estimates from the EPRI and the LLNL studies should be used to obtain different mean estimates. Further the use of both the LLNL and the EPRI mean hazard curves has another advantage in that the extent of uncertainty will become obvious and the emphasis of the bottom line numbers (HCLPF or CDF) is reduced.

2.3 Review Level Earthquake

The seismic margins method was designed to demonstrate sufficient margin over the SSE to ensure plant safety and to find any vulnerabilities that might limit the plant shutdown capability to safely withstand a seismic event larger than the SSE. The seismic margins methodology utilizes two review or screening levels geared to peak ground accelerations of 0.3g and 0.5g. The specification of the review level earthquake [2](RLE) for use in carrying out an IPEEE was a complex problem involving a search for consistency. It would have been preferable if the selection of the RLEs were completely consistent with the individual plants examinations (IPE) for internal events and the inherent strengths of the seismic margin methodologies, but it is very difficult to satisfy all of the elements in any rigorous quantitative sense. Thus, for example, attempting to equate the RLE to the reporting criteria in the IPE (mean sequence frequency leading to core damage of 10^{-6} per year) is fraught with difficulties because of the large uncertainties in numerical estimates of seismically induced core damage, the appropriateness of a comparison between numerical estimates of seismically and internally induced core damage, and the inherent difficulties in relating the output of a seismic margins study (HCLPF) to estimates of core damage (SPRA output).

It was the staff's judgement that the use of a 0.3g review level earthquake for most of the nuclear power plant sites east of the Rocky Mountains would meet the objectives of the IPEEE. However, all the sites east of the Rocky Mountains are not subject to the same level of earthquake hazard. For some sites where the seismic hazard is low, a reduced-scope margins approach centered on a comprehensive walkdown is acceptable. For western US site other than California coastal sites, a 0.5g RLE could be used. For California coastal sites, the seismic margins approach was not available.

2.5 Coordinated Safety Issues

In addition to using the IPEEE process to subsume several outstanding seismic issues, the NRC staff recommended that one issue be resolved concurrently with the IPEEE review as a potential resource saving. The first issue is Unresolved Safety Issue (USI) A-46, entitled "Verification of Seismic Adequacy of Equipment in Operating Plants." This issue was identified during the Systematic Evaluation Program, mentioned above. Equipment in nuclear power plants, for which construction permit applications had been docketed before 1972, had not been reviewed according to then current (1980-1981) licensing criteria for seismic qualification of equipment. Therefore the seismic adequacy of the equipment in these older plants to survive and function in the event of a safe shutdown earthquake was questionable. Equipment in plants with a construction permit docketed after 1972 was qualified according to the then-current criteria. The affected utilities formed the Seismic Qualification Utility Group (SQUG) to independently assess and review the viability of using earthquake experience data and test data to demonstrate equipment ruggedness and to provide expert advise and consultation. This group sponsored the development of the "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment. Following its review and acceptance by the NRC staff, ir provided a basis for resolution of the issue. Because of the similarities between the resolution of USI A-46 and the seismic portions of the IPEEE, licensees of A-46 plants tended to combine the implementation of these two programs.

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[2] The review level earthquake is defined as the ground motion against which the seismic portion of a plant's IPEEE is conducted ; it is characterized by the peak acceleration used to anchor a NUREG-0098 spectra. The review level earthquake is set above the plant's SSE .

In 1982, the U.S. Geological Survey identified the possibility of a large, damaging earthquake occurring at locations in the eastern US that had not been considered in licensing decisions. This issue has been referred to as the Eastern US (EUS) Seismicity Issue. It provided a large portion of the impetus for the development of the LLNL and EPRI probabilistic seismic hazard assessment methodologies, discussed above. The staff decided and included in the IPEEE guidance that submittal of the seismic IPEEE would resolve the EUS seismicity issue without requiring additional analyses or documentation.

The following seismic issues were subsumed by the seismic IPEEE:

1. USI A-45, AShutdown Decay Heat Removal Requirements@ - the adequacy of shutdown decay heat removal at operating plants, and
2. GSI-131, APotential Seismic Interactions Involving the Movable In-Core Flux Mapping Systems in Westinghouse Plants@

3. RESULTS

The NRC staff has received and reviewed 70 IPEEE submittal representing all currently operating nuclear power plants. There are more than 70 operating plants in the United States, but sites that have more than one plant on a site were permitted to conduct a single IPEEE package if the plants were essentially identical. The staff has been able to conclude that the objectives of the IPEEE have been achieved, in that the licensees have a greater appreciation and a better understanding of the severe accident response of their plants to seismic initiators.

Most licensees reported to the NRC that they have no specific vulnerabilities, but most did report what they called seismic “outliers” or “concerns”. Most of the plants made improvements to address concerns and to improve seismic ruggedness. The licensees used different definitions of the term, “vulnerability,” and the plants that did report vulnerabilities reported the same kinds of problems that other utilities had not called vulnerabilities.

Summary of Specific Conclusions and Observations

- Most plants that completed SPRAs reported seismic CDFs between 10^{-5}/yr and 10^{-4}/yr,
- The next largest group reported seismic CDFs between 10^{-6}/yr and 10^{-5}/yr,
- Very few plants reported seismic CDFs that were either > 10^{-4}/yr or < 10^{-6}/yr,
- SPRA results for older plants were similar to results for newer plants,
- Plants conducting SMAs reported HCLPF values between 0.12g and 0.3g,
- HCLPF values predicted to be either equal to or greater than plant’s SSE after taking credit for proposed improvements,
- The dominant contributors to SPRA were:
- building and structure failure,
- failures in front-line and support systems,
- offsite and electrical system components,
- emergency diesel generators, and
- tanks

REFERENCES

1. U.S. Nuclear Regulatory Commission, Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, NUREG-1407, June 1991

2. U.S. Nuclear Regulatory Commission, Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program, NUREG-1742, April 2002

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