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SEISMIC EVALUATION OF SAFETY SYSTEMS AT THE SAVANNAH RIVER REACTORS

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ABSTRACT

A thorough review of all safety related systems in commercial nuclear power plants was prompted by the accident at the Three Mile Island Nuclear Power Plant. As a consequence of this review, the Nuclear Regulatory Commission (NRC) focused its attention on the environmental and seismic qualification of the industry's electrical and mechanical equipment. In 1980, the NRC issued Unresolved Safety Issue (USI) A-46 to verify the seismic adequacy of the equipment required to safely shut down a plant and maintain a stable condition for 72 hours. After extensive research by the NRC, it became apparent that traditional analysis and testing methods would not be a feasible mechanism to address this USI A-46 issue. The costs associated with utilizing the standard analytical and testing qualification approaches were exorbitant and could not be justified. In addition, the only equipment available to be shake table tested which is similar to the item being qualified is typically the nuclear plant component itself. After 8 years of studies and data collection, the NRC issued its "Generic Safety Evaluation Report" approving an alternate seismic qualification approach based on the use of seismic experience data. This experience-based seismic assessment approach will be the basis for evaluating each of the 70 pre-1972 commercial nuclear power units in the United States and for an undetermined number of nuclear plants located in foreign countries. This same cost-effective approach developed for the commercial nuclear power industry is currently being applied to the Savannah River Production Reactors to address similar seismic adequacy issues. This paper documents the results of the Savannah River Plant seismic evaluation program. This effort marks the first complete (non-trial) application of this state-of-the-art USI A-46 resolution methodology.

INTRODUCTION

The Savannah River Site (SRS) material production reactors (K, L and P) were built in the early 1950's for the Atomic Energy Commission and are currently owned by the Department of Energy (DOE). The reactors were operated by E.I. du Pont de Nemours & Company until early 1989 and are currently operated by the Westinghouse Savannah River Company. Because these are nuclear material production reactors which do not generate electricity, their design is significantly different from a commercial nuclear power plant. The K, P and L Reactors are low-pressure reactors moderated and cooled by heavy water. Reactor control is provided by a conventional control rod system, and an emergency shutdown capability is provided by a gravity-driven safety rod system backed up by an independent, liquid-poison (gadolinium nitrate) injection system. Ultimate heat rejection is by a once-through cooling system which, under shutdown conditions, can supply water by gravity feed.

The SRS reactors were designed to both the 1946 Uniform Building Code (UBC) and to a 1951 UBC supplement specifically generated for SRS. This supplement includes a 0.1 g seismic requirement and a 1000 psf blast overpressure requirement. These design criteria generally applied only to the civil structures with very few of the mechanical and electrical systems receiving an initial seismic or blast design. Subsequent to the original plant design, some of the piping and equipment were evaluated for seismic loading (and upgraded if required), but the majority of the equipment remained without a seismic evaluation.

To address the impact that current seismic criteria might have on the SRS reactors, a seismic evaluation program for these three reactors was initiated in 1988. The scope of work covered in this evaluation was based on NRC criteria as outlined in NRC NUREG 1211 "Regulatory Analysis for Resolution of Unresolved Safety Issue A-46, Seismic Qualification of Equipment in Operating Plants" (Reference 12). These criteria include all piping and equipment necessary to safely shut down the reactor for 72 hours following a Design Basis Earthquake (DBE) occurring at 100% reactor power. Systems included in the initial study were selected portions of the Process Water (PW), Cooling Water (CW), Supplementary Safety (shutdown) System (SSS), the D.C. Power System that drives the PW circulating pumps, and associated control and instrumentation components. Also included were facilities to provide a means of returning spilled process water to the system and pumps and piping for flood control in the event of failure of non safety related piping. The seismic characterization of equipment and raceways at SRS was accomplished using the methodology which the NRC has endorsed to resolve Unresolved Safety Issue A-46 for commercial nuclear power facilities. The major portion of this A-46 resolution methodology is based on experience data that was developed by EQE Engineering (EQE) and the Seismic Qualification Utilities Group (SQUG). A similar experience-based seismic evaluation methodology was adopted in this SRS seismic evaluation to address the piping within the systems being addressed.

SRS SEISMIC PROGRAM FORMAT

The seismic evaluation program at Savannah River closely paralleled the Generic Implementation Procedure (GIP) generated for the Seismic Qualification Utilities Group (SQUG). The GIP (Reference 1) documents the experience based methodology developed to resolve USI A-46 and has been reviewed and approved by the NRC (Reference 2). While the use of earthquake experience data is the primary thrust of the SRS seismic evaluation program, there are several other innovations which also improve traditional seismic qualification techniques. These innovations include the assimilation and use of seismic shake table data in a generic manner; the development of simplified analytical tools and realistic criteria for and equipment anchorage; and the development of realistic criteria for the generation of seismic demand (i.e. floor response spectra).

The Savannah River Seismic Evaluation Program represents an expansion of the SQUG GIP in several areas. Programatic changes were incorporated to add credibility and defendability to the SRS program. In addition, several technical changes were added to the SQUG procedure to address conditions unique to Savannah River. These changes included:

- o Site specific procedures were developed to be used in conjunction with the SQUG procedures.
- o A Systems Engineer was made a member of each walkdown team.
- o More detailed documentation was provided.
- o Overview by a Technical Review Team (TRT).
- o Independent reviews of procedures and applications.
- o Additional verification for shell type expansion anchors.
- o A special investigative program was initiated to develop allowable loads for lead shield expansion anchors.

The TRT consisted of senior specialists from Westinghouse Savannah River Co. (formerly DuPont), EQE, and United Engineers and Constructors (UE&C) whose function was to develop the implementation and acceptance criteria for the project and to resolve technical issues. The TRT also conducted a program review and a walkdown of the safe shutdown equipment to ensure that the procedures were being applied correctly and to verify uniformity between walkdown groups. The two independent reviews were conducted by SQUG steering group members who pioneered the A-46 resolution methodology, SQUG consultants, and by the Senior Seismic Review and Advisory Panel (SSRAP) who serve as technical advisors to SQUG (Reference 3). These two reviews provided valuable insight relative to the SRS procedures and to the judgments being made by the SRS walkdown teams.

The SRS seismic evaluation procedure is documented in SRS reports (References 4 & 11). These procedures include the floor response spectrum and instructions to walkdown personnel regarding documentation requirements and deviations from the GIP (such as reduction of anchor bolt capacity because of lower concrete strength). The procedure requires a one-week training program for each participant in the seismic evaluation project coupled with independent study in each of the technical areas. This training is considered to be necessary due to the high degree of judgment and experience required when applying the subject seismic evaluation methodology. The primary sections of the procedures are:

- Identification of Safe Shutdown Equipment
- Plant Walkdown
- Seismic Demand
- Equipment Review
- Anchorage Evaluation
- Cable Tray, Conduit and Piping Review
- Seismic Spacial System Interaction Assessment

A brief description of each of these areas is given below together with a description of the results found at Savannah River.

IDENTIFICATION OF SAFE SHUTDOWN EQUIPMENT

SRS personnel that were knowledgeable of the system functions and requirements developed documents that identified the safe shutdown path, system boundaries and piping and/or equipment that must function during or after a DBE in order to shut down the reactor and maintain it in a safe shutdown condition for 72 hours.

In identifying the safe shutdown path and equipment, the following conditions were assumed:

- Offsite power may not be available for up to 72 hours following the earthquake.
- No other extraordinary events or accidents (e.g., LOCAs, fires, floods, extreme winds, sabotage) are postulated to occur other than the earthquake itself and loss of offsite power.
- Redundant systems and/or equipment must be qualified and be available to ensure that failure of a single item does not compromise the ability to meet the shutdown criteria.
- Instrumentation and controls necessary to monitor critical conditions and ensure appropriate corrective action is taken was included.
- Where operator actions are relied upon to achieve and maintain safe shutdown, the time required for the action was considered.

PLANT WALKDOWN

The responsibilities and qualifications of those individuals participating in the SRS walkdowns follow those outlined in the GIP. Each walkdown team is comprised of at least two seismic capability engineers (required), systems/plant operations engineers (optional), plant maintenance personnel (optional) and a relay engineer if the relay walkdown is done concurrently. The seismic capability engineers are expected to exercise engineering judgment during the walkdown and to apply the methodology developed to resolve USI A-46. Collectively the walkdown team must have knowledge of the performance of equipment and structures in past earthquakes, nuclear plant walkdown experience, knowledge of nuclear design standards, and expertise in the seismic design, seismic analysis and vibration test qualification practices relative to nuclear plant equipment. Each individual seismic capability engineer must possess a portion of the collective experience discussed above, they must be a degreed engineer or equivalent, and they must have at least five years of applicable nuclear experience. At least one member of each walkdown team must also be a registered professional engineer. The SRS project had four walkdown teams (one per major system) normally with four to five members per team.

The purpose of the plant walkdown is to examine each component within the safe-shutdown list and to evaluate parameters specific to the as-built condition of the item. There are four basic criteria that must be satisfied in order to verify the seismic adequacy of equipment during the walkdown. The first criterion is that the seismic capacity response spectra envelops the seismic demand spectrum curve over the frequency range of interest. The second criterion is that the equipment being reviewed is similar to equipment in the experience data base and that all specific caveats and inclusion rules associated with that equipment class are satisfied. The third criterion is that the anchorage capacity, installation, and rigidity are adequate for the seismic demand loads. The last criterion is that seismic interactions must not cause equipment to fail to perform its safety-related function. Each of these 4 areas which are addressed on the walkdown are described in the following sections.

SEISMIC DEMAND

Several aspects of seismic demand are necessary to show applicability of and implement the SQUG methodology:

- o Earthquake experience data base applicability
- o Applicability of the Generic Equipment Ruggedness Spectra (GERS)
- o Anchorage demand

To use the earthquake experience data base to demonstrate the seismic adequacy of equipment, the design basis earthquake motions for the facility being evaluated must be shown to be enveloped by estimates of the motion experienced by facilities in the earthquake experience data base. Two options exist to satisfy this criteria:

- o The facility design free-field ground motion as specified by design ground response spectra are enveloped by the SQUG Bounding Spectrum. The earthquake experience data base approach is then applicable to equipment located in the facility within 40 feet above grade elevation.
- o Floor response spectra at equipment locations are shown to be enveloped by a spectrum equal to 1.5 times the SQUG Bounding Spectrum.

<u>SQUG Bounding Spectrum</u>. In order to verify the applicability of the earthquake experience data base for seismic qualification of equipment, SSRAP has developed and SQUG has adopted a generic seismic motion bounding spectrum (Reference 3). The purpose of this SQUG Bounding Spectrum is to compare the potential seismic demand on equipment in the facility being evaluated with the estimated seismic demand that similar equipment experienced in data base facilities subjected to earthquakes. For convenience, the SQUG Bounding Spectrum is expressed in terms of free-field ground motion at a facility rather than floor response or equipment response. The SQUG Bounding Spectrum represents approximately two-thirds of the estimated average free-field ground motion to which the data base equipment was actually subjected. The SQUG Bounding Spectrum is based on the free-field ground response spectra from four data base sites: Sylmar Converter Station (1971 San Fernando), El Centro Steam Plant (1979 Imperial Valley), Pleasant Valley Pumping Plant (1983 Coalinga), and Llolleo Pumping Plant (1985 Chile). All had average peak ground accelerations greater than 0.4g. These earthquake response spectra were selected based on earthquake characteristics (highest ground motion, duration, and frequency content), and presence and performance of representative equipment. The SQUG Bounding Spectrum is defined in terms of a 5% damped horizontal ground response spectrum. This spectrum bound is intended for comparison with the 5% damped horizontal design ground response spectrum for the facility to be evaluated. Hence, the earthquake experience data base is demonstrated to be applicable with respect to seismic demand when the horizontal design ground response spectrum for the facility is less than the SQUG Bounding Spectrum at the approximate frequency of vibration of the equipment and all higher frequencies. This comparison of ground response spectra is judged to be applicable for equipment mounted less than about 40 feet above grade and for reasonably stiff structures. If equipment frequencies are less than about 8 Hz, floor response spectra must be compared with 1.5 times the SQUG Bounding Spectrum as discussed below. Reference 3 contains an expanded discussion of aspects of the derivation of the bounding spectrum.

<u>Floor Response Spectra</u>. An alternative to comparing the SQUG Bounding Spectrum with the horizontal design ground motion is to compare horizontal floor response spectra to 1.5 times the SQUG Bounding Spectrum. This alternative may always be invoked and must be invoked when the equipment item is located

greater than 40 feet above grade, has a natural frequency less than about 8 Hz, or the ground response spectra criteria fails. Floor response spectra to be compared should be realistic, i.e. median-centered conditional on the occurrence of the design ground motion. Conservatively calculated in-structure spectra may be compared but are not required and this extra conservatism should be recognized as unnecessary.

Generic Equipment Ruggedness Spectra (GERS). The applicability of GERS from the excitation standpoint is related to floor response spectra at the equipment support location (Reference 6). Conservatively calculated floor response spectra may be directly compared with GERS. If median-centered or realistic floor response spectra are calculated, they need to be multiplied by 1.5 for comparison purposes which ensures conservatism in the demand specification.

Anchorage Demand. Anchorage evaluations are performed by comparing seismic demand on the anchor to its seismic capacity. To evaluate the structural integrity of the equipment anchorage and its load path, applied loads are derived from response spectral accelerations at the equipment support location. If median-centered floor response spectra are the basis of the evaluation, a load factor of 1.25 is applied to introduce conservatism.

Vertical Components of Motion. Although inclusion criteria for the earthquake experience data base and GERS are cast in the form of horizontal ground and floor response spectra, vertical motions need to be included in all quantitative assessments such as anchorage evaluations.

SRS Reactors Evaluation. Evaluation of the SRS K- and L- reactors was performed for design ground response spectra defined by US NRC Regulatory Guide R.G. 1.60 anchored to 0.2g PGA. A comparison of the SQUG Bounding Spectrum to R. G. 1.60 is shown in Figure 1. Figure 2 shows selected floor response spectra comparisons with 1.5 times the SQUG Bounding Spectrum. These comparisons show the applicability of the earthquake experience data base up to 40 feet above grade. Floor response spectra at higher elevations are being reviewed and will extend applicability of the earthquake experience data base.

EQUIPMENT EVALUATION

The guidelines used for verification of the seismic adequacy of electrical and mechanical equipment at SRS closely followed the GIP developed by SQUG (Reference 1). Slight deviations from the GIP equipment review sections by the SRS specific procedure (Reference 4) were necessary mainly due to site specific quality documentation requirements. The GIP provides the technical approach and generic procedures for operating nuclear plants to evaluate seismic adequacy of mechanical and electrical equipment needed to achieve a safe shutdown condition following a safe shutdown earthquake.

These equipment evaluation guidelines present an alternative qualification approach, based primarily on the performance of equipment in past earthquakes. Reviews of equipment experience data from conventional power plants and industrial facilities subject to past earthquakes shows that with established inclusion rules and caveats, and below certain seismic motion bounds, it is unnecessary to perform explicit seismic qualification in order to demonstrate functionality of many classes of equipment following an earthquake. The guidelines address twenty classes of equipment, established by their representation in the seismic experience data base and also by their similarity to nuclear plant equipment including construction, operation, capacity, and application (Reference 5). Where higher equipment seismic capacity or function during an earthquake needs to be demonstrated, the review guidelines include generic criteria based on shake-table test results (Reference 6).

The scope of equipment covered by the procedure includes active mechanical and electrical equipment such as motor control centers, switchgear, transformers, distribution panels, pumps, valves, HVAC equipment, batteries and their racks, engine and motor generators, and instrumentation and control panels, cabinets, and racks. Relays are also reviewed to determine if plant safe shutdown systems could be adversely affected by relay (contact) chatter as a result of an earthquake. The equipment review guidelines evaluate the seismic capacity versus demand of a component item, and include a detailed in-plant evaluation of the component structural integrity, anchorage load path, and functionality parameters.

The results from the SRS equipment evaluations were in general confirmatory in nature, demonstrating seismic ruggedness for most equipment. The isolated cases of outliers were generally associated with unique, plant specific mounting details such as customized frame supports and vibration isolation support details which required further detailed analysis. There was one case of an outlier due to a deep well casing on a vertical pump which exceeded the length of those in the experience data base; additional study verified its seismic adequacy. Further evaluation was also required to assess the seismic ruggedness of the cast iron material of yokes of isolated air operated valves in the review scope. The data packages assembled during the evaluations proved to be extremely helpful throughout the review process due to their completeness and clarity. Although the documentation procedure at SRS exceeded the minimum requirements outlined in the GIP, the added effort proved beneficial in the long run.

ANCHORAGE

Seismic experience data and shake table tests have demonstrated that adequate anchorage of equipment and distribution system installations is a critical parameter for component survivability during strong motion earthquakes. The Electric Power Research Institute (EPRI) conducted a detailed program to develop simple and effective anchorage evaluation guidelines (Reference 7) to support the SQUG efforts towards resolution of USI A-46. The EPRI anchorage evaluation guidelines were developed to provide a consistent and cost effective manner for assessing the seismic capacity of equipment anchorage, in conjunction with developing technical justification for elimination of unnecessary sources of conservatism in the anchorage evaluation procedure. These anchorage criteria are considered applicable to evaluation of existing anchorages as well as for upgrading or designing new anchorages.

The EPRI anchorage guidelines address several types of fasteners including expansion anchor bolts, cast-in-place bolts, and welding to embedded or exposed steel. The fastener strength criteria were developed by compiling and analyzing a vast quantity of test data. Criteria for other fastener types were adopted from existing codes and standards with appropriate elimination of unnecessary conservatism. The anchorage capacity evaluation guidelines include an inspection checklist for in-plant review and assessment of an anchor's as-installed condition and other critical anchorage parameters, to assist in fastener strength determination. Simple equivalent static analysis methods are then used to determine the equipment anchorage lateral acceleration level capacity. This is defined as the horizontal load that when applied at the equipment component center of mass will cause the anchorage to reach its assigned strength (which includes appropriate factors of safety).

For the SRS seismic review implementations, plant specific anchorage evaluation guidelines had to be developed. The SRS guidelines differed from the EPRI methodology (Reference 7) due to unique plant-specific conditions, and also included an in-plant inspection guideline refinement whose necessity became evident as the seismic reviews progressed. The SRS specific conditions not covered by the EPRI anchorage evaluation guidelines include lower strength concrete than considered by the EPRI study, and lead sleeve expansion-type concrete anchor bolts.

Anchor bolt capacities for cast-in-place bolts (covered by the EPRI guidelines) were estimated by reducing the EPRI capacities which were based on higher strength concrete. For expansion anchor bolts of the types covered by the EPRI study, capacities were based on manufacturers' test-data for lower strength concrete as stated in the original SRS (DuPont) design standards, with factors of safety consistent with the EPRI study recommendations. For the lead sleeve expansion anchors, the SRS embarked upon a testing program of abandoned lead sleeve anchors to establish ultimate capacities. Allowable "generic" capacities for the seismic reviews utilized appropriate factors of safety consistent with the EPRI recommendations. Due to the generally low capacities that resulted from the large variance in ultimate capacities from the SRS lead shell anchor testing program, another test phase was conducted to develop bolt-specific proof torque test relationships versus allowable capacity on an as-needed basis. Using this relationship, bolt-specific torque proof load tests were conducted where seismic demand exceeded the generic lead shell anchor capacities. Bolts failing the proof torque load test were replaced, typically with conventional non-shell type expansion anchors.

The EPRI guidelines utilize detailed in-plant inspection requirements for verifying expansion anchor installation adequacy. The inspection guidelines include a tightness check that ensures expansion anchor set. During the SRS seismic reviews, it was determined that the tightness checks for shell-type anchors were at times meaningless as the bolt tightening was simply forcing the concrete insert shell up against the equipment component base plate. As a refinement to the EPRI procedure, shell type expansion anchor inspection guidelines for the SRS adopted a check requiring removal of hold-down bolts from the concrete anchor expansion sleeve (after initial tightening) to verify that a gap existed between the top of the anchor shell and the bottom of the component base plate.

Of the hundreds of expansion anchors reviewed at Savannah River, seismic adequacy was verified for about 80 - 90 percent using the SRS specific guidelines. The majority of outlier anchors identified were due to improperly installed shell type anchors and low capacity lead shell expansion anchors. Bolt replacement for the outlier conditions typically was not difficult and considerably increased component seismic margin. Based on the SRS reviews, recommendations were made to the SQUG program to adopt the developed revisions to the EPRI bolt in-plant inspection procedure.

CONDUIT AND CABLE TRAYS EVALUATION

Plant specific evaluation guidelines were developed for seismic evaluation of conduit and cable trays at SRS, based on a conservative interpretation of the current efforts of the SQUG raceway evaluation guideline development program at the time the SRS reviews commenced (Reference 8). The approach is based on seismic experience data, shake table test data, component test data, and bounding analyses. Seismic experience data have shown that cable tray and conduit systems consistently perform well at conventional power and industrial facilities subject to past strong-motion earthquakes, even though the systems are typically not designed for earthquakes. A number of shake table tests on portions of cable tray and conduit systems confirm the observations from past earthquakes and demonstrate that typical configurations perform well under repeated high level seismic input test motion on the order of 1g zero period acceleration.

The SRS evaluation guidelines for electrical cable and conduit raceway systems included screening criteria and procedures for verification of seismic adequacy. Seismic ruggedness of raceway systems was defined as protecting electrical cable function and maintaining overhead support. The evaluation guidelines address seismic ruggedness by walkdown guidelines and limited analytical review guidelines. The walkdown guidelines are for in-plant seismic ruggedness reviews which have two purposes. First, the in-plant review screens raceway systems to check that they are representative of the experience data base that forms the basis for the guidelines. The screening guidelines also check for certain details that may lead to undesirable seismic performance as shown by past experience. Second, the in-plant review selects worst-case, bounding samples of as-installed raceway system supports for limited analytical review. The limited analytical review guidelines check that the bounding sample supports are as rugged as those that have been shown to perform well by past earthquake experience. The checks assess the raceway support dead load integrity, ductility, vertical capacity, and lateral capacity.

The raceway evaluations at SRS generally demonstrated acceptability of the as-installed configurations. Identified outliers were associated with isolated conditions of unique anchorage details requiring additional study, and isolated cases of expansion anchors not fully set. The raceway systems at SRS were observed to be well constructed, lightly loaded with short spans, and with high capacity support systems for seismic load. The rigid support systems were evaluated to have high seismic margins inherent with their design. The many flexible support systems were evaluated as acceptable due to their high vertical capacity and ductility for lateral seismic loading, consistent with the experience based criteria.

PIPING EVALUATIONS

Seismic evaluation of piping systems at SRS included review of systems with and without available dynamic analyses. Certain piping systems had been previously dynamically analyzed to evaluate their ability to survive the design basis earthquake. For these systems, the scope of this evaluation was limited to assuring that the piping dynamic analyses were available, and that the analysis configurations were representative of the as-installed conditions. The piping system boundary included the piping configuration from component to component or to other anchor points. Other interconnections to the subject piping were included from the connection to the first anchor point or component. Anchors were defined as the point at which the pipe is restrained in three directions and three rotations, or through a series of supports which provide equivalent restraint of the pipe.

For piping systems lacking existing available dynamic analysis, the reviews were based primarily on seismic experience data and in part on engineering judgment and simple calculations. Piping systems in this category were reviewed for identified favorable seismic performance installation attributes which are typical of piping systems found in conventional plants (i.e., process, industrial and power plants). A significant number of piping systems in conventional plants, which have been subject to earthquakes with peak ground motions in excess of the SRS design basis earthquake, have been reviewed by utility sponsored organizations such as SQUG and EPRI (Reference 9). Also, piping configurations representative of those found in conventional plants have been tested as part of seismic qualification activities in commercial nuclear power plants.

The comprehensive experience and test data indicate that welded steel piping can withstand earthquakes with ground motions exceeding 0.5g peak ground acceleration. The few reported failures are attributed to conditions that are considered inadequate in well designed conventional systems, and the effect of large seismic induced displacements on the same inadequate features. Also, failures have been due to certain other design aspects which are particularly sensitive to differential building motions. Evaluation of the SRS piping without dynamic analysis was performed by assuring that the piping systems do not contain design details otherwise considered inadequate in conventional plants, and critical attributes that are sensitive to large seismic induced movements. The piping system reviews at SRS identified a few configurations requiring additional study due to lack of sufficient lateral support strength, and also limited cases requiring bracing to preclude potential problems associated with large seismic induced motion. Other isolated problems were observed due to as-installed conditions of expansion anchor bolts to the building structure. Seismic adequacy was verified for several of the as-installed configurations and no "generic" SRS issues related to seismic adequacy of the piping systems were identified.

SYSTEM INTERACTIONS

System interaction is characterized by the inter-system dependencies that may result in potential harmful effects of initiating events to required safety systems. The seismic system interaction issue is addressed in three NRC documents: Regulatory Guide 1.29, and Unresolved Safety Issues (USI) A-17 and A-46. NRC Regulatory Guide 1.29 (Reference 10) stipulates "Those portions of structures, systems, or components whose continued function is not required but whose failure could reduce the functioning of any plant Category I component to an unacceptable safety level should be designed and constructed so that the Safe Shutdown Earthquake would not cause such failure". NRC USI A-17 defines the seismic system interaction concern and describes five separate areas relative to seismic spacial interactions: Category II over Category I (II/I) failure and falling conditions, seismic deflection/impact, differential motion induced failures, seismic induced spray and flooding, and seismic induced fires. Some of these seismic system interaction issues within USI A-17 are being addressed and resolved as a part of the USI A-46 program resolution. The SRS program has committed to follow the USI A-46 scope and will address the first three of these areas: II/I failure and falling (up to the first support), seismic deflection/impact, and differential displacement failures.

The II/I seismic system interaction issue relates to the effects of a non-seismically designed component (Category II) failing during an earthquake and subsequently falling or sliding into a Category I safety related component. The seismic deflection interaction issue relates to potential impacts between Category II and Category I components due to relative displacements during the earthquake. The differential displacement interaction issue relates to distortion-controlled behavior which occurs when a component is constrained and thereby forced to undergo the same movement as that of a major structure (e.g. damage resulting from components crossing seismic separations between buildings or systems, such as piping connected to equipment that experience large seismic movements).

The primary methods utilized on the SRS project to evaluate seismic spacial interaction concerns included engineering judgment, earthquake experience data and simple calculations. Engineering judgment is useful for obvious situations which are judged not to be a credible concern. Simple seismic calculations can be useful to calibrate an engineer's judgment. The earthquake experience database developed by EQE under the sponsorship of SQUG contains detailed seismic data on the performance of a wide variety of components in past earthquakes. This database is very useful in evaluating the credibility of postulated potential interactions through a comparison to the performance of similar configurations in past earthquakes. Specific objectives of the experience data approach include the documentation of the most common sources of seismic damage, identification of the threshold of seismic motion corresponding to various types of damage, and determination of installations that are typically undamaged by earthquakes in facilities that are representative of critical nuclear power plant systems.

There were relatively few credible interactions identified during the SRS walkdowns, and many of those identified were resolved with minimal effort. The majority of the interactions identified related to unanchored equipment being in close proximity to safety related equipment.

CONCLUSIONS

The Savannah River Seismic Characterization Program reviewed the seismic adequacy of the safe shutdown systems for the K and L production reactors. The methodology utilized in that review is described in this paper and was found to be cost-effective, expedient, thorough and defendable. In general, most of the components within the safe-shutdown systems were found to possess adequate inherent capacity to survive the 0.2 g design basis earthquake at SRS. Retrofits were recommended and are currently being implemented in those cases where seismic adequacy could not be established by more detailed analyses. The principal areas where retrofits were required included equipment anchorage (particularly where the lead shield expansion anchors existed), relays, relocation of proximate non-safety components (system interaction concerns) and structural load path concerns. These findings are consistent with the types of situations which have been found on SQUG trial plant walkdowns and on previous seismic margin studies and seismic risk assessments for commercial nuclear power plants.

The SRS seismic evaluation program had two independent reviews in addition to DOE's review. These independent reviews were conducted by the Seismic Qualifications Utility Group and by the Senior Seismic Review and Advisory Panel. Both of these reviews concluded that the SRS program was comprehensive, technically correct and conducted in a professional manner. Comments and suggestions from these two reviews were incorporated where deemed appropriate and have resulted in a stronger program at SRS. Due to the fact that the SRS program commenced before the SQUG methodology was completed, many of the open issues to the SQUG program were resolved as a result of the SRS study. In addition, the SRS study provided new direction to the requirements of SQUG guidelines in areas such as documentation, anchorage inspection, independent review and piping evaluation. Many of these SRS procedures will prove helpful to all of the older reactors (DOE and Commercial) which will be undergoing seismic reviews in the future.

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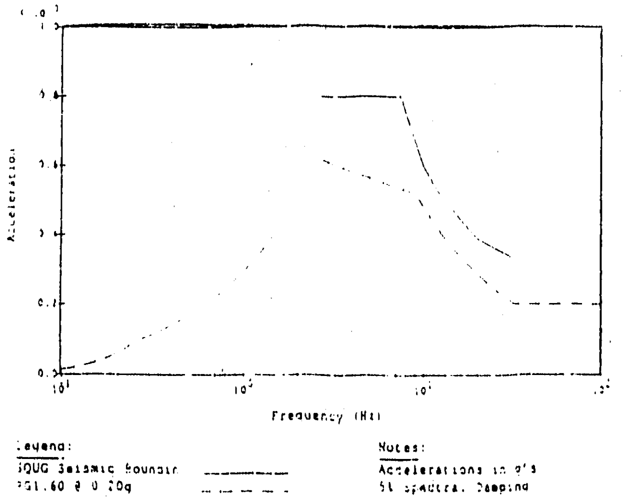

Figure 1 COMPARISON OF SQUG BOUNDING SPECTRUM WITH R. G. 1.60

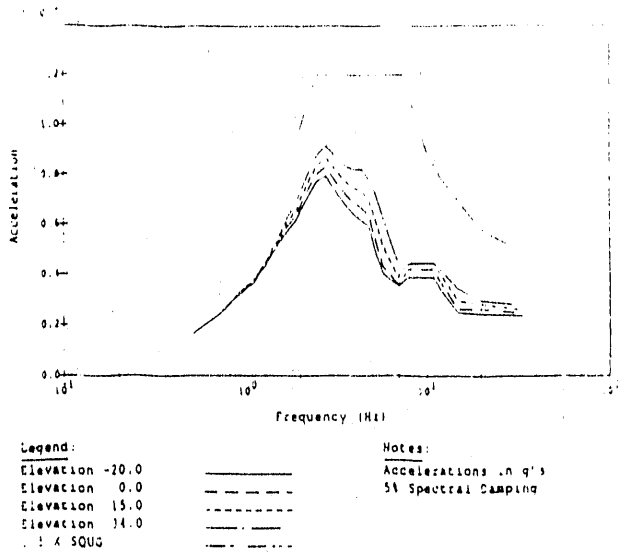

Figure 2 COMPARISON OF FLOOR RESPONSE SPECTRA WITH 1.5X SQUG BOUNDING SPECTRUM
(A) TRANSVERSE

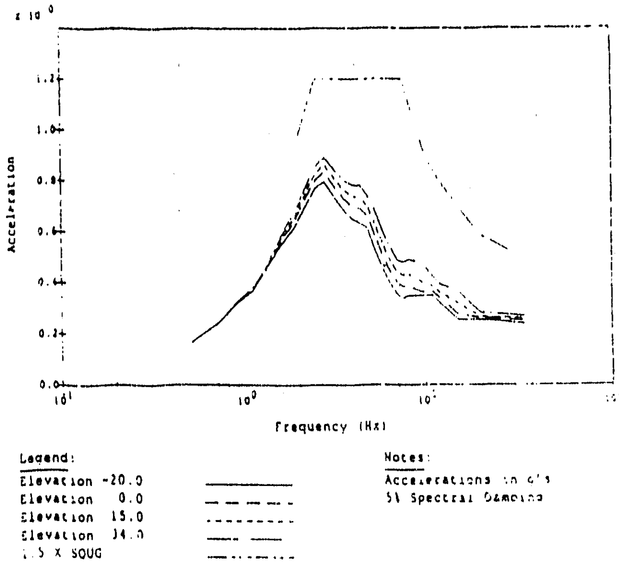

Figure 2 (B) LONGITUDINAL

END

DATE FILMED

4/28/92