[DNFSB LETTERHEAD]

March 29, 1991

MEMORANDUM FOR: A. J. Eggenberger, Member
R. H. Case, Member

FROM: A. G. Stadnik, Project Officer, Reactors

SUBJECT: Savannah River Site Production Reactors - Review of Seismic Design Capability

Background: As you are aware, I have been reviewing the seismic design capability of the Savannah River Site (SRS) production reactors with Dr. W. J. Hall, Dr. P. C. Rizzo, and Dr. J. D. Stevenson, our three outside experts supporting DNFSB activities in seismic and systems engineering. Monthly meetings between the DNFSB Staff, outside experts, the Department of Energy (DOE), and Westinghouse Savarmah River Company (WSRC) have been conducted since rnid 1990, to review the seisrnic issues related to restart of the SRS reactors as well as meetings to review the design basis of various mechanical and electrical systems in the reactor plant.

Discussion: To date, we have identified fifteen key topics associated with the seismic and systems design capability of the SRS reactors. These topics are as follows:

1. Geotechnical Considerations - Cooling Water Reservoir Basin and Cooling Water System Buried Piping

2. Behavior of Buried Piping and Inter-connections Under Seismic Ground Motions Assuming Adequate Foundation Conditions

3. Behavior of the Cooling Water Reservoir Under Seismic Ground Motions Assuming Adequate Foundation Conditions

4. Seismic Engineering Procedures Review and Comments

5. Seismic Qualification of the Process Water System

6. 105-K Stack and Reactor Building Analysis and Floor Response Spectra

7. Seismic Design Basis and Ground Motion Inputs

8. Seismic Attenuation Experiment

9. Supplementaly Safety System

10. Expansion Joints in the Process Water System

11. Systems Assessments

12. Airborne Activity Confinement System

13. J-bolts and Grouted Plates

14. Probabilistic Risk Analysis - Seismic Issues

15. Design Basis Documentation - Review of SAR Chapter 3 and Seismic Topical Report

A summary description of each topic is attached to this memorandum, and includes a background discussion, current comments and observations by the DNFSB Staff and outside experts, and an approach to resolve the issue. For many of the topics, the associated DOE and WSRC documentation is still being reviewed. Thus, the resolution approach identified for those topics is to complete the review and provide further comments and observations to DOE. Most of the technical documentation reviewed to date has been adequate. However, the topics listed above represent areas where questions and comments remain.

15 Attachments

cc:
J. T. Conway, Chairman, w/attachments
J. W. Gawford, Member, w/attachments
J. C. Kouts, Member, w/attachments
G. W. Cunningham, TD, w/attachments
All Technical Staff, w/attachments
Chron File(2), w/attachments

DNFSB: AG Stadnik, Ext. 208-6407, 3-29-91

Topic 91-1. Geotechnical Considerations Cooling Water Reservoir Basin and Cooling Water System Buried Piping

Background: The 186 Cooling Water Reservoir (CWR) and the associated Cooling Water System Buried Piping that runs from the CWR to the 105-K Reactor Building is the heat sink for the K--Reactor. The Buried Piping connects with Cooling Water System (CWS) piping inside 105-K Reactor Building that runs to the secondary side of the heat exchangers in the Process Water System (PWS). The CWS as a whole is an Accident Prevention System that is to be seismically qualified prior to K--Reactor Restart.

Current Observations and Comments: The geotechnical matters under consideration relate prirnarily to the stability of the foundation support for the CWR and Buried Piping, especially under seismic conditions. Specifically, the following three matters deserve further consideration:

Static and seismic behavior of the "soft zones": These zones were grouted beneath 105-K Reactor Building by the Army Corps of Engineers in 1951, but no such remediation was performed at that time beneath the CWR or the Buried Piping The concern is with the potential for significant surface deformation (total and differential) associated with the behavior of the soft zones; particularly subsequent to a seismic event.

Liquefaction Analysis: No documentation of liquefaction analysis of the foundation soils, including the "soft zones" has been presented.

Amplification/Frequencv Response Analysis: The "soft zones" would seem to have the potential to significantly impact on the frequency response of the near-surface and surface structures and therefore, the Department of Energy (DOE) should consider this matter.

It is noted that the postulated "collapse" of the "soft zones" beneath the CWR and the loss of cooling water associated with a failure of the Buried Piping contnbute a major fraction of the Severe Core Melt Frequency in the Probabilistic Risk Assessment.

In addition, the "soft zones" beneath 105-K Reactor Building have been grouted. No documentation has been received that the grouting is adequate some 40 years after injection to assure foundation support during and after a seismic event.

Resolution Approach: It is recognized that DOE has a completed geotechnical investigation program and that final results are available and are being reviewed. In the meantime, the DOE may wish to consider alternate means, for both the short term and long term, of assuring an adequate cooling water supply, such as, grouting the "soft zones" beneath the CWR and the Buried Piping, or providing other qualified sources of water, in the event that the final results of the geotechnical investigation are inconclusive.

Topic 91-2. Behavior of Buried Piping and Inter-connections Under Seismic Ground Motions Assuming Adequate Foundation Conditions

Background: The Cooling Water System (CWS), which is classified as an Accident Prevention System to be seismically qualified prior to restart, includes large diameter Buried Piping (steel piping with about eight feet of cover). This Buried Piping transports cooling water from the Cooling Water Reservoir (CWR) to the secondary side of the Process Water System (PWS) heat exchangers.

Current Observations and Comments: The Defense Nuclear Facilities Safety Board (DNFSB) has received a seismic qualification study for the K-Reactor which is actually a comparison of conditions between K-Reactor and L-Reactor assuming foundation conditions are adequate. It attempts to justify seismic qualification at the K--Reactor on the basis that conditions are equal or better at the P--or L-Reactors. No basis is presented for concluding that the Buried Piping at the L-Reactor is seismically qualified. In any event, the following matters require further attention:

1. Site specific soils data from the completed geotechnical investigation should be incorporated into the analysis to examine the effect on the ground motion input, behavior of supporting soils, and the seisrnic response of the Buried Piping. Particular care should be taken to characterize the behavior of any "soft zones" (as-is or as-remediated) under both static and dynamic conditions, for example, by widely ranging parametric studies.

2. It is noted that the completed geotechnical investigation does not include test borings or cone penetrometer tests along the alignment of the Buried Piping. This matter should be specifically addressed, including the potential for adverse effects of leaks, if any, on the underlying subsoils.

3. The critical nature of the Buried Piping suggests a need for further documentation as to the effects of surface waves and differential displacements on the system, particularly at the cormection with the structures, including the 105-K Reactor Building and the CWR, and at sharp bends, "Elbows," and Tees."

4. The design and construction of the Buried Piping indicate a need for a conservative assessment of the possible loss of water to the environment through joints and interconnections with structures.

5. The current condition of the pipe, including the joints, '~ees," "Elbows," and interconnections with structures, including the Reactor Building and the CWR, taking into account its age and the surrounding environment requires documentation.

Resolution Approach: A seismic qualification document specific to the Buried Piping at the K-Reactor addressing items such as the above cited matters should be provided by tne Department of Energy.

Topic 91-3. Behavior of the Cooling Water Reservoir Under Seismic Ground Motions Assuming Adequate Foundation Conditions

Background: The 186 Cooling Water Reservoir (CWR) is a part of the Cooling Water System (CWS) which is classified as an Accident Prevention System, and is to be seismically qualified prior to restart. The CWR is a key feature in the CWS loop that begins with water being pumped from the Savannah River to the CWR, then to the secondary side of the Process Water System (PWS) heat exchangers, and finally back to the River.

During normal operations, the CWS operates as a once-through cooling system with the Savannah River as the ultimate heat sink. After a seismic event and a postulated loss of off site power, the heated effluent from the K-Reactor is recirculated back to the CWR, making the CWR the ultimate heat sink for decay heat. Under this scenario, sufflcient cooling water from the CWR is available by gravity flow to the PWS heat exchangers to handle decay heat for a limited time.

The CWR is an embedded reinforced concrete structure consisting of three basins, a pump well, and the 190 Pump House. Pumps located in the 190 Pump House supply cooling water to the K-Reactor through two, large diameter Buried Pipes. Heated effluent flows from the K--Reactor to the Process Effluent Sump (PES) Building and then back to the River under normal operating conditions. Subsequent to a seismic event, seismically qualified pumps with emergency power are available for recirculation back to the CWR in the event of a loss of off site power to the pumps as mentioned above.

Current Observations and Comments: The seismic analyses of the CWR and PES structures have been performed using industry standard practice. The structures have been found to generally meet the conventional acceptance criteria for typical nuclear structures founded on well understood foundation media. Considering the uniqueness of the CWR and the level of understanding of the foundation media, clarification is needed for the following items with additional analyses and/or documentation:

1. Site specific soils data from the completed geotechnical investigation should be incorporated into the analysis to examine the effect on the ground motion input, the structural response, the floor response spectra and the reported Demand/Capacity Ratios. Particular care should be taken to characterize the behavior of any "soft zones" (as-is or as- remediated), for example, by widely ranging parametric studies.

2. The size, mass and dimensions of the CWR suggest a need for further documentation as to the effects of seismic surface waves and differential settlement on the base slab and the exterior walls (lateral earth pressures) and the effects of structure-to-structure interaction with the 105-K Reactor Building and associated coupling phenomena.

3. The design and construction of the CWR indicate a need for a conservative assessment of the possible loss of water to the environment through cracking of concrete walls and base slabs, distress, separation and/or "hammering" damage at expansion joints, overtopping due to sloshing, etc., associated with an earthquake.

4. Ground motion and associated time histories should be provided.

5. Technical assurance and/or justification of a two dimensional planar analysis as opposed to a three dimensional analysis.

Resolution Approach: Additional analysis and/or documentation addressing the observations and comments cited above should be provided by the Department of Energy.

Topic 914. Seismic Engineering Procedures (SEP's) Review and Comments

Background: The Westinghouse Savannah River Company (WSRC) has identified a total of 27 seismic engineenng procedures as part of their Seisn~ic Improvement Project. Of these, all but two, SEP-13 and SEP-14, have been identified as requiring implementation prior to restart.

Current Observations and Comments:

(1) The following are general comments related to the indhidual SEP's as identified at the beginning of the comment:

- (a) SEP-5 and SEP-6: In what ways do the current SEP-5 and SEP-6 formats fail to meet the Generic Implementation Procedure (GIP) Rev. 2 format?
- (b) SEP-7: What is the date of the code of record being used for B31.1 re-evaluation?
- (c) SEP 8: Why is American Institute of Steel Construction (AISC) Standard N690 not being used as the code or standard of reference?
- (d) SEP-15: What is basis of GIP acceptance of instrument tubing? Documentation that the GIP addresses seismic qualification of instrument tubing needs to be provided.
- (e) SEP-16: The GIP addresses only anchorage of tanks and heat exchangers. To what extent will noz~le loads and seismic evaluations of the body and internal components of tanks and heat exchangers be addressed and how?
- (f) SEP-18: Since much of the buried piping is welded steel pipe, to what extent will the "B31.1" Code requirements be addressed? Specifically, what consideration will be given to stress intensification factors or stress indices in the pipe?
- (g) SEP-19: To what extent will Nuclear Regulatory Commission (NRC) Regulatory Guide 1.12 or Arnerican National Standards Institute (ANSI)/American Nuclear Society (ANS) 2.2 be followed with respect to seismic instrumentation?
- (h) SEP-21: To what extent and to what acceptance critena will stresses in piping and piping supports be evaluated?
- (i) SEP-22: To what extent will the requirements of NRC Regulatory Guide 1.100 Rev. 2 be complied with?
- (j) SEP-24: Will Welding Research Council (WRC) Bul. 353 also be considered as a code and standard reference for piping supports?

(2) Several SEP's have yet to be completed; SEP-9, SEP-12, SEP-15, SEP-16, SEP-17, SEP-19, SEP-26, and SEP-27. Further detailed comments await their completion.

(3) Detailed comments on SEP-24 Rev. 2, its associated field guide and SEP-7 will be transmitted separately. Of these, the only comment of primary concern for SEP-24 and SEP-7 is shown on page 2 of SEP-24 regarding the potential for a ratchet mode of failure. Unless it can be demonstrated generally as it was done for fatigue that the ratchet mode of failure is not a safety concern, it should be considered on a line-by-line basis.

Resolution Approach: A response to the questions and comments in paragraphs (1) and (2) above should be provided by the Department of Energy. The SEP's still under development should be provided in a timely manner after issuance. A technical exchange between Dr. J. D. Stevenson, Defense Nuclear Facilities Safety Board expert, Department of Energy and Westinghouse Savannah River Company to resolve comments associated w~th paragraph (3) above should be arranged.

Topic 91-5. Seismic Qualification of the Process Water System (PWS)

Background: The PWS, consisting of six (6) parallel loops, circulates process water (deuterium oxide or heavy water) as a coolant through the reactor core and therefore, it is the primary means of maintaining the heavy water inventory and the sole means for removing decay heat. Each loop of the PWS includes a main pump, two heat exchangers, and motor operated rotovalves--all connected by large diameter pipe. This is an Accident Prevention System and is, in fact, one of the most critical systems as regards safety of the reactor. According to the Level 1 Probabilisffc Risk Assessment (PRA) currently available, failure of the ability to circulate process water after a seismic event contrlbutes over 30 percent of the Severe Core Melt Frequency due to external events.

The Seismic Qualificaffon Program (SQP) to support Restart utilizes a series of Seismic Engineering Procedures (SEP's) to evaluate systems, structures and components. Specifically, SEP-6 utilizes Seismic Qualificaffon Uffliffes Group (SQUG) methodology for evaluating piping components, such as valves. Simplified acceptance criteria in SEP24 are being used for piping analysis prior to restart in most of the systems.

Current Observations and Comments: SEP-10 of the Seismic Qualification Program lists the PWS as a Seismic Category I Accident Prevention System with Priority 1 status to be seismically qualified prior to restart. SEP-7 applies to dynamically analyzed Seismic Category I piping systems. Considering the importance of the PWS system to overall safety, use of SQUG and SEP-24 methodologies alone for the PWS requires exceptional documentation and justification if an adequate dynamic analysis has not be performed.

Resolution Approach: The Department of Energy recently submitted a dynamic analysis of the PWS and a review of the adequacy of this analysis is currently underway.

Topic 91-6. 105-K Stack and Reactor Building Analysis and Floor Response Spectra

Background. The following documents related to the 105-K Stack Building have been reviewed:

(1) "Phase I, II and II of the Comprehensive Seismic Analysis and Evaluation of the 105-K Exhaust Stack Building at the Savannah River Site," December 1989, IJRS/John ~ Blume and Associates.

(2) "RLCA Review Comments on the 105-K Stack Building Analyses (Phase L IL and m) Revised Report," February 8, 1990, Robert L Cloud & Associates, Inc.

(3) RLCA Letter P171-12-L002 to NUS (Ousley), "Resolution of Issues Related to the 105-K Stack Building Seismic Analysis," July 27, 1990, Robert L Cloud & Associates, Inc.

(4) BNL Letter to DOE (Dr. M. Davister) from Dr. C Miller, June 13, 1990, Brookhaven National Laboratory.

(5) Topical Report, WSRC-RP-9~993 Rev. Q "Reactor Restart Seismic Program Topical Report" (U), December 1990.

The only document related to the reactor building which has been reviewed has been the "Evaluation of 105-K Reactor Building Floor Slab Frequencies and Basemat Stresses (U)," WSRC-RP-90 1328. With regard to the WSRC-RP-90-1328 report, the major purpose of this report appears to be justification of the use of horizontal floor spectra in the vertical direction.

Current Observations and Comments:

A. With regard to the 10-K Reactor Building;

(1) The analysis described in the above-referenced report evaluates the natural frequencies of the floors in the reactor building and the stresses in the basemat under design basis earthquake (DBE) loading.

(2) The floor frequencies were determined to justify the use of horizontal amplification factors in the vertical floor response spectra for qualifying the equipment and piping supported in the building.

(3) The floor frequencies are deterrnined on the basis of linear elastic analysis of assumed uncracked concrete sections. It was not possible to verify the geometric properties of the slabs, such as size, shape, thickness, and boundary conditions. Some drawings, sketches, or figures should be included to complete the documentation.

(4) No justification has been provided for assuming a live load of 100 pounds/square foot. The calculations apparently use uncracked sections for the structural elements. This

implies very low strains. It may be appropriate to consider fixed ends for most cases since at these strains, tension in the reinforcement is expected to be small. On the other hand, uncracked sections may overestimate element stiffness and therefore, the frequencies may be calculated to be higher than actual.

(5) Vertical ground motions typically contain higher frequencies than horizontal ground motions. Motions in the 10 to 20 hertz range may be amplified if the fixed base modes have significant participation. Although the vertical floor response spectra typically exhibit smaller spectral accelerations than horizontal in absolute terms, the peaks at higher frequencies may exceed the horizontal spectral accelerations at the corresponding frequencies. Floor flexibility may further add to the potential amplification.

(6) The computer code verification should be documented. Explanatory notes to the summary table on Page 14 should be presented.

(7) In the calculation of the basemat stresses the maximum bearing stress and its distribution under the mat should be reported. Further, it is not clear if the vertical seismic acceleration is included in calculating the foundation bearing stresses. T~e report should also indicate the location of the critical demand/capacity ratios.

(8) In summary, the basemat stresses are within allowable limits. Additional documentation as suggested above will enhance the report. Additional justification is required to substantiate the vertical floor response spectra used in quali~ring equipment and piping in the 105-K Reactor Building especially at higher frequencies.

B. With regard to the 105-K Stack Building analysis the following comments are made:

(1) Phase I of the Blume report included preliminary analysis and was performed to identify structural components that may be overstressed under the DBE, and evaluate remedial schemes to upgrade the structure. Phase II of the Blume report included a detailed analysis to qualify the structure including the recommended upgrading. Phase III refined the Phase II analysis to update the structural model for as-built conditions disclosed during Phase II and includes further modifications resulting from the Phase II analysis.

(2) Phases II and III represent the more significant part of the overall effort. The analysis methodology, the recommended remedial measures and the cAteria used in qualify,ing the structure following the industry practice. However it is not clear what specific remedial measures have been implemented to date. The current status of the remedial program needs to be documented and made available.

(3) With reference to the Blume report, the following are specific comments:

a. Synthetic ground motions are defined by Blume at the surface. Comparison of the esponse spectra of the synthetic time histones with the ground design response spectra at all damping values used in the analysis should be reported to demonstrate that the time histories meet the acceptance criteria more or less uniformly at all damping values. The PSD of the synthetic time histories for 2 percent damping should also be reported.

b. The soil structure interaction parameters have been based on correlations with measured SPT values. These should be verified by in-situ shear wave velocity measurements in the K-area.

c. The seismic analysis is based on vertically propagating shear and compression waves. This is generally a conservative assumption for surface structures. It is not clear how the effects of surface wave components on the dynamic soil pressures on walls of embedded structures (embedded tunnel) are considered in the analysis. This requires further documentation.

d. The potential for dynamic consolidation of the foundation soils due to earthquake shaking, and the resulting relative displacements of the supported structures should be evaluated and reported.

e. The Blume report presents conceptual upgrading schemes. The design details were to be established by others. These should be reviewed to assure that the intent of the conceptual upgrading is properly implemented. For example, the recommended upgrading of the crane girders consists of adding 21 inches of concrete and reinforcement to the bottom of the girders near the main column locations. the reinforcement detail needs to provide for the shear transfer bet veen the new and old concrete in order that the improved section acts as a composite element. Does the detail design satisfy this for all upgrading cases?

Resolution Approach: Information, in addition to that provided in the Seismic Topical Report, regarding the above comments on the 105-K Stack Building should be provided by the Department of Energy (DOE) in a timely manner. The report on the 105-K Reactor Building and any other information used to evaluate the structural and seismic capability of the 105-K Reactor Building should also be provided by DOE.

Topic 91-7. Seismic Design Basis and Ground Motion Inputs

Background: The Seismic Design Basis and associated ground motion input for the Savannah River Site (SRS), and specifically for the K--Reactor has been a topic of discussion from the early reviews conducted by the Defense Nuclear Facilities Safety Board (DNFSB). Initially, investigations to substantiate an "assumed" seismic design basis were to be conducted in Phase 4 of the four phase Seismic Qualification Program undertaken by the Department of Energy (DOE). This aspect of the Program was subsequently changed to Phase 1.

The Reactor Restart Seismic Program Topical Report states that the seismic design basis being used currently by DOE consists of Nuclear Regulatory Commission (NRC) Regulatory Guide 1.60 Spectra anchored to 0.20 g and NRC Regulatory Guide 1.61 for damping values.

Current Observations and Comments: Resolution of the following matters regarding theseismic design basis prior to K--Reactor restart is appropriate:

1. Deterministic Analysis: A draft report has been received for review and this is underway.

2. Probabilistic Risk Analvsis (PRA) The PRA uses a Seismic Hazard Analysis prepared by the Electric Power Research Institute (EPRI) which is generally less conservative than similar curves developed by Lawrence Liverrnore National Laboratory (LLNL). For example, the PRA states that seismic ground motions corresponding to a frequenc~,r level of 1*E-5 dominate seismic risk EPRI indicates that this corresponds to 0.4g whereas LLNL indicates 1.0g. Therefore, the applicability of EPRI vs. LLNL should be addressed by DOE.

3. New Production Reactor (NPR): The seismic design basis for the NPR is currently being determined. The concern is with the methodology and the results at the NPR as compared with the restart design basis and the long term design basis. To the extent there may be differences between NPR and the restart and long terrn seismic design bass, reasons therefor should be provided.

4. Surface Faulting: There is substantial evidence as summarized in the Seismic Topical Report that surface faulting cannot be dismissed out of hand at the SRS. DOE has field work underway, but conclusive age dating may be lacking.

5. Other Nearby Facilities: The SeismiG Topical Report presents the basic seismic design criteria for a number of facilities in the area around the SRS. This discussion should be supplemented to include additional facilities, as well as discussion of the potential hazard under seismic events associated with the facility.

6. UCRL,15910: Regardless of its specific applicability to the SRS production reactors, an assessment as to how UCE~15910 relates to the seismic design basis for the restart of K-Reactor and in the longer term should be provided.

Resolution Approach: DOE is currently addressing these comments and will be submitting additional reports substantiating the restart seismic design basis.

Topic 91-8. Seismic Attenuation Experiment

Background: The Department of Energy (DOE) is participating in a long term experiment to measure attenuation of ground motion in basement rock and in the geologic column beneath the Savannah River Site (SRS). The main objective of the experiment is to measure key attenuation parameters used in ground motion studies, particularly the deterministic analysis cited under Topic 91-7. The experiment consists a series of deep boreholes on a line between Charleston, South Carolina and the SRS. Explosive charges will be set off in several of the deep boreholes and resulting ground motions will be measured on the ground surface and at depth.

Current Observations and Comments: The documentation, planning, and procedures for this experiment should, as a minimurn, address the effect of triassic basins penetrating into basement rock particularly at the SRS, and the difference in the strain level and magnitude of energy, of an earthquake and of that associated with a borehole explosion.

Resolution Approach: More details of this experimental program should be provided by DOE.

Topic 91-9.   Supplementary Safety System (SSS)

Background: The shutdown system involving the insertion of safety rods is not seismically qualified for the Savannah River Site (SRS) production reactors. Therefore, the SRS production reactors have been modiSed to include the SSS which basically injects a "poison" or "ink" into the reactor tank and pump suction piping upon activation of the system. The SSS can be activated by five different means, only two of which are to be seismically qualified to any ground acceleration levels. The two seismically qualified means for SSS activation are the seismic triggers set at 0.05g located at the -40 feet elevation in the 105-K Reactor Building and a pull ring in the Central Control Room to be pulled by the Central Control Room Operator (CCRO) when an alarm is heard or if the operator senses an earthquake is occurring. These two activation mechanisms are to be seismically qualified for restart up to the design basis earthquake.

Current Observations and Comments: As regards activation of the system, the Seismic QualiScation Utilities Group approach is being used to seismically qualify the pull ring mechanism and the related nitrogen system. Additional documentation of the methodology and its supporting database as it applies to the pull ring mechanism is needed. Also, an analysis of the time required by the CCRO to activate the pull ring versus the time available and the related impact on reactivity, including a discussion as to the acceptability of the "defense in depth" available with this approach is required.

The second means of activation is the pair of seismic triggers located at the -40 ft. elevation. These triggers have been presented as being seismically qualified, but clarification is required as to the redundancy of the tie-in to the existing uninterruptible power source; specifically, it is not clear that if the power source fails, whether the system automatically actuates and injects ink into the reactor tank.

Also, related to the seismic triggers, additional documentation provided by the Department of Energy (DOE) pertaining to the shutdown of the reactor for events with a peak ground acceleration less than 0.05g has been reviewed. The DOE has stated that the plant will withstand and function through a seismic event with peak ground motion less than 0.05g. The DOE has proposed to substantiate this position with a compilation of relevant industry references on non-damaging effects of seismic events less than 0.05g on nuclear facilities and equipment The DOE approach inherently implies the SRS production reactors and all the support systems, including the safety rods, are as rugged up to 0.05g as other plants even though the reactors were designed and built in the early 1950's. Also, it is noted that the seismic triggers are set with a plus or minus tolerance of 20 percent, and consequently the threshold for seismically qualified SSS activation could be 0.06g. The detailed basis and analysis for concluding the reactor can be shutdown safely up to 0.05g needs to be provided.

As regards the volume and distribution of ink in the reactor tank, the DOE has developed a mathematical model of the flow patterns in the tank and the associated reactivity. The model is based in part of full-scale testing at the C-Reactor in 1959. DOE is proposing to conduct new model testing for the K-Reactor configuration.

Resolution Approach: Additional documentation on the overall rational of all the seismic instrument trigger levels is required which, also addresses the comments discussed above. For example, the current system, although not seismically qualified, is designed to initiate a safety rod scram at Q02g via the seismoscope. This might suggest that the plant is not intended for operation beyond 0.02g as compared to the 0.05g setting of the SSS seismic triggers.

On the assumption that DOE can provide additional analysis and/or documentation related to the activation of the SSS, DOEs information related to the effectiveness of the operating system, specifically the volume of the ink and its distribution within the reactor tank has been reviewed. The confidence level associated with the flow modeling tools would appear to be sufficient so as to not require full-scale model testing prior to restart if the following substantiating documentation can be made available by DOE:

1. Documentation, including graphic presentation, of the correlation between modeled flow characteristics and the 1959 "CMX" model test results. This calibration is essential to demonstrating confidence in the modelling tools and approach.

2. An explicit statement that adequate dispersion of the poison occurs prior to full coastdown of the pumps to DC power.

3. A complete discussion of reactivity levels under DC flow conditions. Available data do not address this condition.

4. The "WISR-19" closure package (Essential Core Monitoring) for review with respect to the need and ability for flu~c monitoring with the SSS in operation.

Topic 91-10. Expansion Joints in the Process Water System (PWS)

Background: The failure of one of the expansion bellows in a reactor PWS loop is the basis of a postulated pipe break area of 55.0 square inches. It is understood that it is assumed that the bellows fails but the sleeve remains intact with no loss of geometry. It is further understood that there have been 19 bellows leaks and 3 restraint rod ruptures during the operating life of the Savannah River Site production reactors.

Current Observations and Comments: The abilit,v of the sleeve to retain its geometry after a postulated bellows failure is essential to the limitation of the postulated break size to 55.0 square inches. The loading on the sleeve during a seismic event concurrent with the postulated rupture of the bellows and resultant stresses and deformation of the sleeve are essential to qualify the limiting break area.

Resolution Approach: An analysis should be provided by the Department of Energy of the basis for determining that the structural integrity and deformation characteristics of the sleeve can be maintained so as to limit the postulated break area to 55.0 square inches given the rupture of one of the bellows concurrent with the design basis earthquake.

Topic 91-11. Systems Assessments

Background: The Defense Nuclear Facilities Safety Board (DNFSB) initiated a review of various production reactor systerns beginning with the Process Water (PW), Moderator Recovery (MR), Emergency Cooling (EC), and Supplementary Safety systems in November, 1990. (The Supplementary Safety System is discussed Topic 91-9.) The objective of this review is to focus on the system design basis with an emphasis on the seismic design capability of the systems. Some of the key items being reviewed include the system design and layout drawings, the seismic boundary definition, and the mechanical and electrical desigrl supporting documentation. The first technical interchange on the PW, MR, and EC systems occurred on January 8-9, 1991. A followup session was held on February 6, 1991 and March 20, 1991.

Current Observations and Comments: To date, the following issues have been identified and need to be addressed by the Department of Energy (DOE):

a. The rationale, technical basis, and supporting documentation for using different codes and standards, and different code and standard revisions for different parts of the various structures, systems, and components should be provided. There does not appear to be a consistent or rationale logic for what has been applied that has also been properly documented in a technical specification or appropriate design basis document.

b. The seismic qualification requirements for various plant systems, for restart and post restart need to be clarified due to the inconsistencies identified by the DNFSB in its January 11, 1991, letter to DOE forwarding the DNFSB staff trip report on the SSS.

c. The system boundary document (Westinghouse Savannah River Company (WSRC) Report OPS-RSE-902451 of 1V28/90)) lists many subsysterns and components that were, or are, to be examined. It is not clear that the various plant systems have been examined from the total system perspective in order to ensure that all important elements were examined. For example, is the basis for qualification of piping and tubing equal to or less than one inch in diameter for the restart and long terrn included? Are all instrumentation penetrations evaluated?

Additional related information requested at the January 8-9, 1991, meeting is scheduled to be submitted by DOE by April 1991.

Resolution Approach: The infornnation obtained to date is currently being reviewed and evaluated. The overall review cannot be completed until receipt of the balance of the remaining documentation from DOE and WSRC, as well as a report from DOE addressing the three issues discussed above.

Topic 91-12 Airborne Activity Confinement System (AACS)

Background: The AACS is the primary means of controlling the release of radioactive particulates and non-noble gases to the atmosphere following an accident. This accident mitigation system consists of moisture separators, particulate filters, halogen absorbers, ductwork motor driven fans, an exhaust stack and associated instruments and controls. Two independent diesel generators provide emergency power for the AACS.

The seismic upgrade of the AACS was originally scheduled for Phase 4 in the Department of Energy (DOE) Seismic Qualification Program. After consideration of the importance of the AACS to the public health and safety as pointed out by the Defense Nuclear Facilities Safety Board, the DOE adopted an alternate strategy for the AACS.

Current Observations and Comments: DOE's strategy is to pursue an analytical approach to seisrnically qualify the AACS prior to restart; however, in anticipation of the difficulty with an analytical approach for predicting by-pass and/or behavior of the filter elements during or after a seisrnic event, as well the probability of finding possible deficiencies with the filter compartments and filter elements, DOE elected to commence a procurement prograrn to purchase and install new filter compartments for the Savannah River Site production reactors. The new filter compartments will be seismically qualified upon purchase, probably by testing. The DOE has periodically provided an update on the progress of the procurement.

Resolution Approach: DOEs submittals received through February 25, 1991, which address some of the matters mentioned above, are being reviewed. DOE's renewed and continued emphasis on the AACS effort and a commitment to carry out the stated strategy on a timely basis will continue to be evaluated.

Topic 91-13. J-bolts and Grouted Plates

Background: The seismic structural supports in the Savannah River reactors are being reviewed. Some of the supports were modified in the 1960's and 1970's by chipping around them, inserting a "J bolt" in the support plate, tack welding the bolt to the structural concrete re-bar, and then back filling with grout. Since April 1990, the Department of Energy (DOE) has provided briefings on the progress of the testing of the J-bolts to determine their current structural capabilities, and the review of the as-built information to ensure that the e~sting conditions are consistent with the tested configurations. This has been presented and discussed several times at periodic seismic revlews.

Current Observations and Comments: Several points remain open concerning this topic:

a. DOE needs to document and justify the technical basis for using a minimum safety factor of less than 2.0 in some of the assessments of the grouted structural support plates.

b. DOE needs to provide the documentation and technical rationale for not inspecting and qualifying all of the grouted support plates since the as-built data are not completely consistent with the design drawings.

c. DOE needs to adequately document why the results of the tested configurations in the "R" reactor are transferable to the "K: and "L" reactors including the rationale and basis for same.

Resolution Approach: DOE has submitted a report addressing the above comments and it is currently under review.

Background: The Department of Energy (DOE) has performed a PRA (Level 1-Exterrlal Events) that provides considerable insight into the relative contribution of the various systems, events, and operator functions to the Severe Core Melt Frequency (SCMF). The analysis considers the effect of all external events generally considered by the profession in such analyses. For seismic, the analysis utilizes a site specific seismic hazard curve based on Electnc Power Research Institute (EPRI) methodology.

It is recognized that there is a difficulty in drawing conclusions regarding SCMF at K--Reactor in absolute terms, particularly with comparison to established cAteria, however, the PRA allows one to make judgments and draw conclusions in a relative sense as to the various contributions to the SCMF.

The results of the PRA indicate that the total mean contribution to SCMF from external events is 2.22* per year, to which seismic contributes 1.2*e4 per year. A review of the sequences making up the contribution of a seismic event to the SCMF indicates that failure of the foundation soils underlying the 186 Cooling Water Reservoir (CWR) and the loss of foundation support for the Buried Piping portion of the Cooling Water System (CWS) are key failure mechanisrns.

It is understood that the PRA currently available is to be considered for K--Reactor restart. However, it is based on the P--Reactor configuration, P--Reactor fragility relationships, and L--Reactor soils data. The DOE considers the impact on risk of seismic upgrades at the K--Reactor as if they had been made to P--Reactor.

Current Observations and Comments: The following with regard to the PRA and its applicability to K--Reactor restart need to be considered:

Incorporation of the results of the geotechnical investigation at K--Reactor, including the postulated collapse, liquefaction, and/or amplification of ground motion frequencies associated with any "soft zones" (as-is or asremediated) beneath the 186 CWR and/or the Buried Piping. Inconsistencies between recent results of predicted behavior of the "soft zones" and those postulated by the PRA analysts need to be resolved in this effort.

Consideration of the Lawrence Livermore National Laboratory (LLNL) seismic hazard curves or other seismic hazard culves that might be available from work being done for new facilities at the site.

Topic 91-14

The appropriateness of a stand-alone PRA for K--Reactor that is based on the seismically upgraded K--Reactor, the K--Reactor configuration and K--Reactor fragility relationships, and the geotechnical conditions beneath the K--Reactor, the 186 CWR and the Buried Piping portion of the CWS.

4. The fault trees for internal events that were used as a basis for establishing the sequences for the external events PR.

Resolution Approach: The effects of the geotechnical investigation and other seismic hazard curves on the PRA need to be determined by DOE. Documents relating the internal events sequences to external event sequences are currently under review. A schedule for the completion of a stand-alone PRA for K--Reactor based on the actual site conditions, fragility curves and configuration at the K--Reactor and the technical basis for going forward with K--Reactor restart with the existing or modified PRA also should to be provided by DOE.

Topic 91-15. Design Basis Documentation - Review of SAR Chapter 3 and Seismic Topical Report

Background: The Seismic Topical Report (WSRC-RP-90-993) and the seismic chapter from the Safety Analysis Report (SAR) (Chapter 3) are two documents summarizing the seismic design basis of the Savannah River Site production reactors. The topical report was provided in December, 1990, and the updated SAR section has yet to be completed and provided to the DNFSB. The objective of this topic is to the review and comment on the adequacy of these design basis documents which are intended by the Department of Energy (DOE) and Westinghouse Savannah River Company (WSRC) to support reactor restart and subsequent operations.

Current Observations and Comments. The topical report is under review. The following major comments, which also relate to comments under other topics, have been noted

Is the 0.20g Zero Period Acceleration (ZPA) to be assumed as the final recommendation for the design basis seismic event? The rationale for this is not clearly presented and/or explained in the report. Additional work has been ongoing since issuing the report to better address this issue. Appropriate discussion of this item in the report/SAR chapter 3, along with resolution of items discussed under Topic 91-7 is needed.

Chapter 4 indicates that the K-area is deemed to be free of liquefaction concerns, but later in the report it is stated that this is still under study. This should be clarified, and addressed properly given the items presented in Topics 91-1 and 91-7.

The equipment seismic verification presented in Chapter 7 fails to present or identify that satisfactory evidence that the entire system has been examined, including piping, valves, supports, sensors, controls, etc. The extent of the system walkdowns needs to be better presented including evidence that minor sensors have been considered. While a thorough and complete presentation on every system is not necessary for this report, this should be provided for several key systems. A similar concern has been expressed under Topic 91-11.

Resolution Approach: When the additional information identified in the above comments and the SAR chapter are provided, they will be reviewed March 29, 1991

[DNFSB LETTERHEAD]

Mr. Victor Stello, Jr.
Deputy Assistant Secretary for Facilities
Office of Defense Programs
Department of Energy
Washington, DC 20585

Dear Mr. Stello:

Enclosed for your consideration is a memorandum from the Defense Nuclear Facilities Safety Board (DNFSB) Staff concerning the Board's ongoing review of the seismic capability, seismic design basis, and overall systems design basis and capability of the Savannah River Site (SRS) production reactors. The DNFSB Staff and contractors that have been conducting this review have identified fifteen key topics that require further resolution. Some of these topics may require resolution prior to restart, while others may concern longer term efforts. A surnmary description of these topics including the background, current observations and colr~nents, and resolution approach is attached to the memorandum.

It is requested that during the next planned SRS seismic review session on April 16 and 17, 1991, the Department of Energy and its contractors be prepared to discuss these key topics, with particular emphasis on the resolution approach and schedule for each.

If you or your staff need further information, please let me know,

Sincerely,

John T. Conway
Chairman

Enclosure

Memorandum dated March 28, 1991 toA. J. Eggenber,ger and E G. Case from A. G. Stadnik