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BENCHMARK ANALYSIS FOR THE DESIGN OF PIPING SYSTEMS IN ADVANCED REACTORS*

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ABSTRACT

To satisfy the need for the verification of the computer programs and modeling techniques that will be used to perform the final piping analyses for an advanced boiling water reactor standard design, three piping benchmark problems were developed. The problems are representative piping systems subjected to representative dyna- c loads with solutions developed using the methods being proposed for analysis for the advanced reactor standard design. It will be required that the combined license holders demonstrate that their solutions to these problems are in agreement with the benchmark problem set. A summary description of each problem and some sample results are included.

INTRODUCTION

Recent changes in licensing regulations permit NRC review and certification of standard designs for the next generation of U.S. nuclear plants even before a utility applies for a Construction Permit. Under the rules of Title 10, Part 52 of the Code of Federal Regulations (10CFR52, 1989), the licensing process for advanced reactors may be separated into three distinct areas: early site permits, standard design certifications, and combined licenses. Applications for site permits and design certifications may be filed independently of applications for combined licenses (construction permit and operating license). These new regulations should provide a more efficient and stable licensing environment for new plants in which safety issues can be resolved prior to initiating construction. Utilities applying for a new plant combined license (COL) will be able to reference a pre-approved standard plant design to be constructed at a pre-approved site.

There are currently four standardized advanced reactor designs which are undergoing NRC staff review for design certification. Although 10CFR52 specifies the overall requirements for granting standard design certification for nuclear power facilities, the development of detailed procedures to implement these requirements is still evolving. The NRC staff and its consultants have been working closely with industry and vendors to develop an effective set of procedures that satisfy these requirements.

According to the rules of 10CFR52, an application for design certification must provide an essentially complete design with sufficient level of design detail to enable the NRC to reach a final conclusion on all safety questions. In addition, the design certification applicant must propose a series of inspections, tests, analyses and acceptance criteria (ITAAC) that must be implemented and satisfied by the COL holder to demonstrate that

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the as-built plant conforms to the certified design. However, in reviewing the lead design certification application submitted for an evolutionary light water advanced reactor, the NRC staff identified a number of technical review areas where the vendor did not provide design and engineering information in sufficient detail to make a final safety decision. One of these areas was piping and pipe support design where the vendor did not have as-built or as-procured information to complete the final design. As an alternate approach, the concept of design acceptance criteria (DAC) was introduced. The DAC are a set of prescribed limits, parameters, procedures, and attributes upon which the NRC relies, in a limited number of technical areas, in making a final safety determination to support a design certification. The DAC are objective (measurable, testable, or subject to analysis using pre-approved methods), and must be verified as part of the ITAAC used to demonstrate that the as-built facility conforms to the certified design. The DAC become a part of the ITAAC which are required for design certification. The DAC concept will enable the NRC staff to make a final safety determination subject only to satisfactory design implementation and verification by the COL holder through appropriate ITAAC.

By applying the DAC approach to piping design, the vendor will not be required to provide final piping designs and stress analyses in the design certification application. Instead the vendor must provide a detailed description of the methodologies, design processes, and acceptance criteria that will be used to complete the design. Sample analyses of representative piping systems demonstrating the implementation of the methodology must also be provided. During the design certification review process, the NRC staff and its consultants will perform detailed technical reviews of the proposed methods and the sample analyses. Based on these reviews, the NRC will be able to determine whether the final piping stress analyses using the approved methodology and acceptance criteria will result in a design that adequately addresses all applicable safety concerns. The review of final piping design analyses using as-built information will be performed during the COL review stage as part of the implementation of the ITAAC program.

During recent NRC staff reviews of the DAC for piping and pipe supports for the design certification of an advanced boiling water reactor standard design, the NRC identified a need for verification of computer programs and modeling techniques that will be used to perform the final piping analyses. The COL holder will be responsible for verifying the adequacy of the computer program that will be used to complete the final piping design analysis in accordance with the NRC benchmark program as part of the ITAAC. The benchmark program will require the combined license holder to construct mathematical models of specific NRC piping benchmark problems, analyze the problems for given loads, and demonstrate that the results are in agreement with the benchmark problem results within a given range of acceptable values. In order to satisfy this requirement, a set of piping benchmark problems were developed and are described in this paper.

In the past, the NRC had developed and published benchmark problems for general verification of computer programs which perform dynamic analysis by the response spectrum method. These problems and their solutions were well documented by Bezler (1980, 1985). The problems covered both uniform support motion and independent support motion methods. The problems ranged from very small simple configurations to large complex configurations taken from actual nuclear plant piping systems. Most piping analysis programs that have been used by the industry to design nuclear plant piping systems have been verified against these benchmark problems.

Since the time when the original benchmarks were published, piping analysis technology has further advanced. Most piping analysis programs have since been modified to incorporate more sophisticated methods for performing dynamic analysis than were covered by past NRC benchmark problems. The new benchmark problems for advanced reactors have therefore incorporated methods proposed by advanced reactor vendors which had not been included in the previous NRC benchmark problem analyses. The new problems are limited to representative piping systems subjected to representative dynamic loads for the proposed standard advanced reactor designs.

PROBLEM DESCRIPTIONS

For the advanced boiling water reactor design, three benchmark problems involving two representative piping systems were developed. The first piping system represents one loop of the feedwater piping system extending from the reactor pressure vessel to the containment penetration. The piping system, Figure 1, consists of a 22 inch nominal diameter carbon steel pipe anchored at the containment wall penetration. Three 12 inch branch lines connect to the 22 inch line and are anchored at three reactor vessel nozzles. Pipe supports include snubbers, a spring hanger and a guide. Pipe mounted equipment consists of the inboard check and gate valves and their operators which are included in the model.

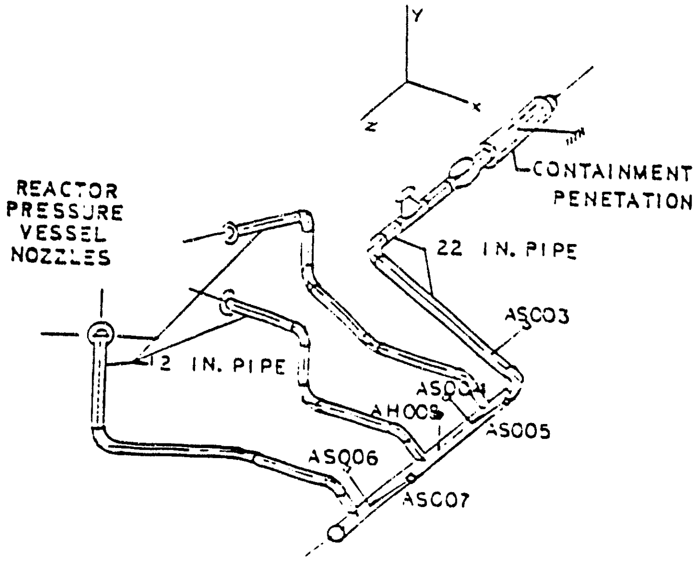

Figure 1
Feedwater Piping

The second piping system, Figure 2, represents one of the safety/relief valve discharge lines (SRVDL) in the wetwell which extends from the diaphragm floor penetration to the X-quencher in the suppression pool. The piping system consists of a 10 inch nominal diameter stainless steel pipe anchored at the diaphragm floor penetration. In the upper portion, the pipe connects to a 12 inch stainless steel pipe through a reducer. The 12 inch pipe extends down and connects to the X-quencher which is anchored to the quencher basemat on the wetwell floor. The lower portion of the piping system and the quencher are submerged in water. The piping is supported by a number of struts. There are no snubbers, spring hangers or valves in this system.

The PSAFE2 program was used to develop the mathematical models and to analyze the piping systems. PSAFE2 was developed by Brookhaven National Laboratory (BNL). It is a modified version of the general purpose finite element analysis program SAP IV, (Bathe, 1973) for performing piping analysis. PSAFE2 and its predecessor, EPIPE, (Subudhi, 1981) had been used to develop and analyze the earlier NRC piping benchmark problems discussed above. The program has been extensively tested and verified against other analytical solutions as well as test results.

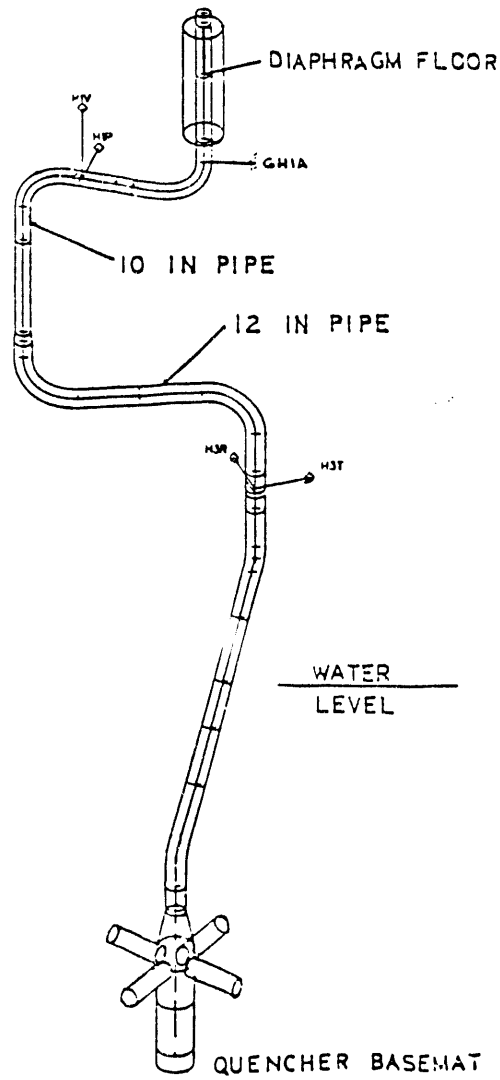

Figure 2
SRVDL Wetwell Piping

Details of the two representative piping systems, proposed analysis methods, and representative loads were provided by the vendor. Both systems required dynamic analysis for various seismic and hydrodynamic load definitions. For the benchmark program, three representative load cases which employed three different dynamic analysis methods were selected:

1. For Benchmark Problem 1, the Feedwater piping system was analyzed for Chugging loads. The analytical model shown in Figure 3, was analyzed by the uniform support motion response spectrum analysis method. The loading was the response spectrum for chugging in the X direction shown in Figure 4. A listing of the predicted natural frequencies with a character of the associated modes are provided in Table 1.

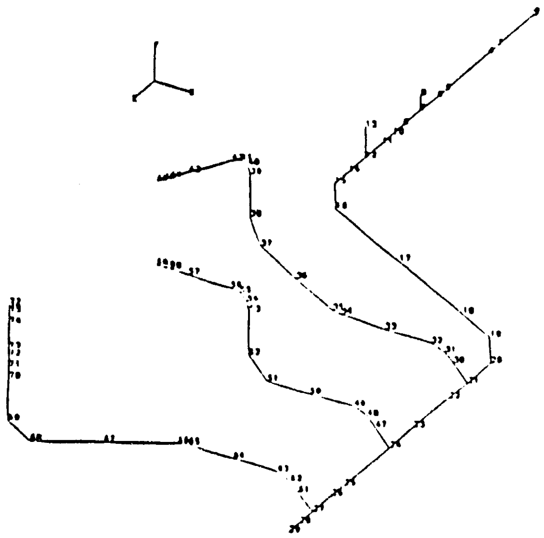

Figure 3, Finite Element Model Feedwater Piping

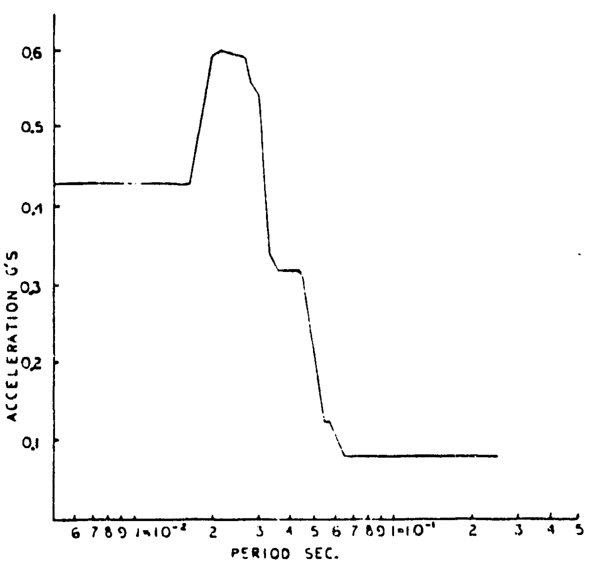

Figure 4, Response Spectrum CHUG X, Feedwater Piping

TABLE 1
FEEDWATER PIPING MODAL ANALYSIS RESULTS

MODE NO.	FREQUENCY CPS	MODE DOMINANT DIRECTION
1	8.17	X
2	10.87	Y
3	11.56	Y
4	11.84	Y
5	13.03	X
6	15.28	X
7	15.81	Y
8	18.00	X
9	19.90	X
10	21.60	X

The effects of high frequency modes (missing mass effects) were incorporated into the analysis in accordance with the methodology described in the Standard Review Plan Section 3.7.2 Appendix A (USNRC, 1989). The high frequency modes effects analysis had not been included in previous NRC benchmark problems. This methodology involves the calculation of pseudostatic inertial forces associated with the response of all modes above the cutoff frequency (60 Hz for this hydrodynamic load) for each degree of freedom. The piping system is statically analyzed for this set of forces and the response is treated like an additional modal response. The high frequency modes response is combined by the square-root-of-sum-of-squares (SRSS) method with the response from the lower frequency modes dynamic analysis to obtain the total response of the piping system.

The response thus obtained indicated peak displacements of 0.16 mm., 0.21 mm, and 0.22 mm in the X, Y, and Z directions respectively on the outer most leg; a peak axial force and transverse moment of 12.9 KN and 11.4 KN-m respectively in the 22 in. pipe and a peak support force and anchor moment of 18.2 KN and 11.4 KN-m respectively.

2. For Benchmark Problem 2, the Feedwater piping system was analyzed for Safe Shutdown Earthquake loads. The model was analyzed by the independent support motion response spectrum analysis method. Figure 5 shows the response spectra corresponding to the Z coordinate direction for the three support groups.

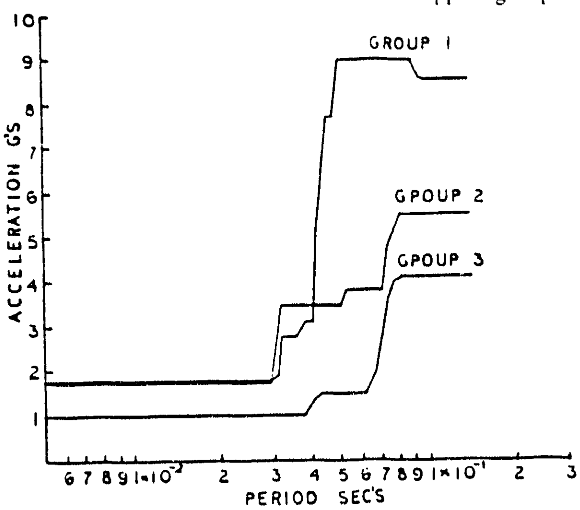

Figure 5

Response Spectra SSE Z Direction Feedwater Piping

The effects of high frequency modes were also incorporated into this analysis using a methodology comparable to the one described above. Since the analysis involved three different support groups subjected to three different response spectra, three separate sets of pseudostatic inertial forces associated with the response of modes above the cutoff frequency (33 Hz for this seismic load) had to be generated for each degree of freedom. Three static load cases were consequentially needed to determine the high frequency response for each support group. The three static responses were combined by the SRSS method to obtain the total high frequency response. The combined modal responses for each of the three support groups from the dynamic analysis (for the modes below the cutoff frequency) were also combined by the SRSS method. Finally, the low frequency responses and the high frequency responses were combined by SRSS to obtain the total piping system response.

The predicted total responses indicated peak displacements of 7.1 mm, 6.2 mm and 14.1 mm in the X, Y, and Z directions respectively near the juncture of the inner most leg and 22 in. header; a peak axial force and transverse moment of 243 KN and 430 KN-m respectively and a peak support force and anchor moment of 334 KN and 99.1 KN-m respectively.

3. For Benchmark Problem 3, the SRVDL wetwell piping system was analyzed for adjacent quencher loads. This is an air clearing load from an active quencher acting on an adjacent inactive quencher. This is a dynamic load which is characterized as a force time history on the submerged portion of piping. Figure 6 shows the computer generated mathematical model of the SRVDL piping system along with the assigned node numbers. A force time history is applied at each node (below water level) corresponding to the pressure time history times the contributory pipe area. Figure 7 shows one of these force time histories at node 60. The model was analyzed by the direct integration time history analysis method. This method of analysis was not included in earlier NRC benchmark problems. As applied, the solution of the equations of motion is obtained by direct integration using the Wilson-Θ method. In order to ensure that the integration time

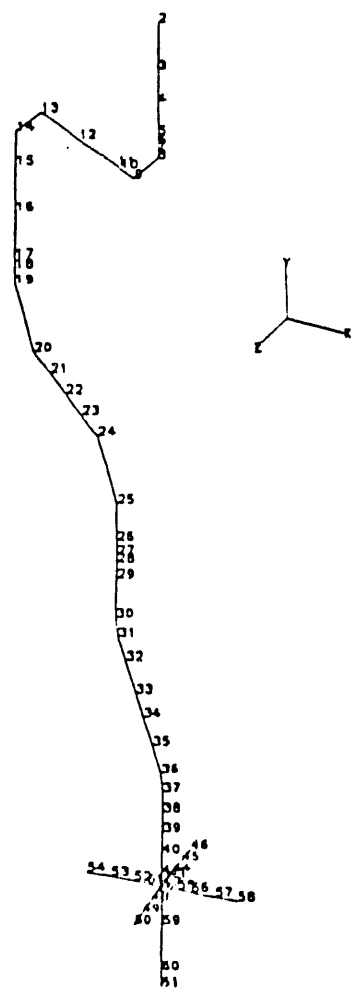

Figure 6

Finite Element Model SRVDL Wetwell Piping step used was acceptable, successive analyses were performed with smaller time steps. The time step was considered acceptable when smaller time steps did not introduce more than a 10% change in response.

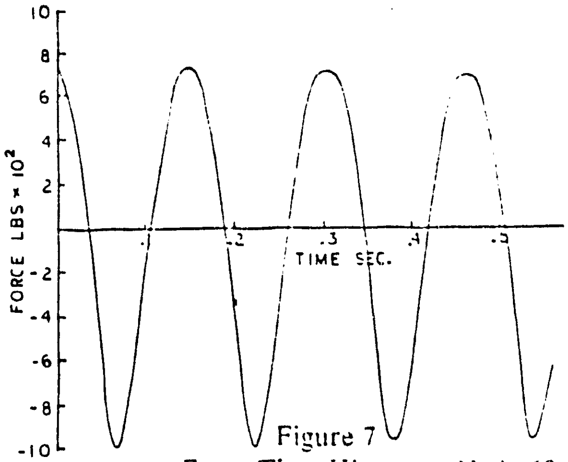

Figure 7
Force Time History at Node 60
Z Quencher Load, Wetwell Piping

A modal analysis was performed on the SRVDL to obtain some information on the dynamic characteristics of the piping system. Table 2 provides the natural frequencies for the first 10 modes along with a characterization of the associated modes. From Figure 7 the frequency of the forcing function is determined to be approximately 6.25 Hz. This frequency falls between the first and second mode so resonance is not a major concern.

TABLE 2
SRVDL WETWELL PIPING MODAL ANALYSIS RESULTS

MODE NO.	FREQUENCY (CPS)	MODE DOMINANT DIRECTION
1	3.90	Z
2	9.32	Y
3	11.3	Y
4	12.0	X
5	12.5	Z
6	18.0	X
7	20.1	X
8	21.3	Z
9	25.0	X
10	28.4	X

The predicted response of the SRVDL piping included peak displacements of 0.64 mm in the X direction at node 24, 0.69 mm in the Y direction at node 16, and 0.61 mm in the Z direction at node 36. The corresponding peak axial pipe force was 14.2 KN while the peak transverse moment was 4.90 KN-m.

CONCLUSION

In summary, three benchmark problems have been developed and will be used to assess the adequacy of the analysis techniques that will be used by COL holders to qualify piping for advanced boiling water reactor standardized designs. A complete description of the input, a comprehensive listing of the output and the acceptance criteria that will be used to assess the adequacy will be presented in a NUREG report to be issued. As stated, COL holders will be required to develop solutions to these problems and to demonstrate that those solutions meet the acceptance criteria. It is anticipated that similar benchmark problem sets will be developed for each advanced reactor standardized design.

REFERENCES

Bathe, J.K., Wilson, E.L., Peterson, F.E., "SAP IV - A Structural Analysis Program for Static and Dynamic Responses of Linear Systems", Report No. EERC 73-11, University of California, Berkeley, CA, 1973.

Bezler, P., Hartzman, M., Reich, M., "Piping Benchmark Problems", NUREG CR/1677, August 1980.

Bezler, P., Subudhi, M., Hartzman, M., "Piping Benchmark Problems", NUREG/CR1677, Vol. II, August 1985.

10CFR52, Office of Federal Register National Archives and Research Administration, "Code of Federal Regulations", Title 10, Part 52, (54 FR 15386), May 1989.

Subudhi, M., Bezler, P., "EPIPE, Piping Analysis Program User's Manual - Version 1981", BNL NUREG/30784, December 1981.

U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants", Section 3.7.2, 1989.

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