MASTER**URS**

IDENTIFICATION OF SIGNIFICANT PROBLEMS RELATED TO LIGHT WATER REACTOR PIPING SYSTEMS

July 1980

prepared for
U.S. Department of Energy
Light Water Reactor Safety Technology Management Center
Sandia National Laboratories
Albuquerque, New Mexico
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prepared by
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Onder Kustu

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U.S. Department of Energy
Division of Nuclear Power Development
Washington, D.C.

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ABSTRACT

The work done by URS/John A. Blume & Associates, Engineers, on the project *Identification of Significant Problems Related to Light Water Reactor Piping Systems* is reported. The project is part of the Light Water Reactor (LWR) Safety Technology Program within the Nuclear Power Development Division of the U.S. Department of Energy (DOE) and is administered by Sandia National Laboratories, Albuquerque, New Mexico, as Manager of the DOE-LWR Safety Technology Program. Work on the project was divided into three tasks.

In Task 1, past surveys of LWR piping system problems and recent Licensee Event Report summaries are studied to identify the significant problems of LWR piping systems and the primary causes of these problems. Pipe cracking is identified as the most recurring problem. The most significant cause of pipe cracking, and of other piping system problems in general, is determined to be the vibration of pipes due to operating pump-pipe resonance, fluid-flow fluctuations, and vibration of pipe supports. Recent, ongoing, and planned research in the United States relevant to the identified piping system problems is evaluated, and, on this basis, the need for further research toward resolving these problems is determined. Programs to carry out needed research are suggested on topics related to pipe vibration, thermal and dynamic pipe behavior, piping analysis, fatigue, design loads, and the problem of radiation buildup in corrosion products.

Task 2 studies identify typical LWR piping systems and the current loads and load combinations used in the design of these systems. The methodologies and data bases for current load definitions and combinations are reviewed, the current development of new piping data is summarized, and the feasibility and benefits of developing new load definitions and combination procedures are discussed. Recommendations for further research to improve the determination of loads for LWR piping systems are made.

In Task 3, a comparative study is carried out to delineate the benefits that might be gained by using nonlinear analysis methods in the design of LWR piping systems. The study concludes that the current linear-elastic methods of analysis may not predict accurately the behavior of piping systems under

seismic loads and may, under certain circumstances, result in nonconservative designs. Gaps at piping supports are found to have a significant effect on the response of the piping systems. The need for the development of cost-effective nonlinear piping analysis codes is identified.

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EXECUTIVE SUMMARY

Problems with pipes and pipe fittings are responsible for about 10% of all safety-related events* and about 7% of all outage time† at light water reactors (LWRs). Identification and resolution of these problems can lead to increased overall plant safety and significant economic benefits. Also, recent advances in computer technology and analysis methods, and the availability of new load data, provide an opportunity for assessment of the current design loads and analysis methods used in the design of LWR piping systems.

The work described in this report was conducted by URS/John A. Blume & Associates, Engineers, as part of the Light Water Reactor Safety Technology Program within the Nuclear Power Development Division of the U.S. Department of Energy. This program is administered by Sandia National Laboratories, Albuquerque, New Mexico, and its overall objective is to develop and assist in the implementation of reactor systems and concepts that offer significant improvements in safety.

The work described in this report consists of three tasks. The objectives of the first task were to identify the significant problems of the LWR piping systems, to determine the research needed to resolve these problems, and to recommend programs to carry out the needed research. The objectives of the second task were to identify and summarize load definitions and combinations presently used in the design of LWR piping systems, to study the availability of new data, and to investigate the feasibility of developing new load definitions using the new data. The objective of the third task was to study the possible benefits that can be derived from the development and use of non-linear analysis methods in LWR piping system designs.

*Internal memo from W. H. Schmidt to W. A. Von Riesemann on LWR component failure data, Sandia National Laboratories, Albuquerque, New Mexico, December 22, 1978.

†Basin, S. L., and E. T. Burns, *Characteristics of Pipe System Failures in Light Water Reactors*, EPRI NP-438, Topical Report, prepared for Electric Power Research Institute, Palo Alto, California, by Science Applications, Inc., August 1977.

Task 1

In Task 1 work, Licensee Event Report (LER) summaries from the U.S. Nuclear Regulatory Commission (NRC) data base were used as the main source of information for identification and characterization of the problem areas; findings of previously conducted surveys of piping failures were also used. The identified problems and their causes were analyzed, and the research areas relevant to the resolution of the problems were delineated. Then recently completed, ongoing, and planned research in these areas was evaluated to determine the areas most in need of further research. These areas were ranked, and programs on specific research topics were suggested. Some of the more important findings of the LER survey are summarized below:

- Cracking of pipes and fittings was the most significant problem, constituting 52.0% of the reportable occurrences. (A reportable occurrence is an unscheduled incident or event that the NRC determines is significant from the standpoint of public health or safety; Licensee Event Report, or LER, is a generic term for reportable occurrence.) Of the cracks reported, 77.2% (40.1% of all reportable occurrences) were through-the-wall cracks, and were discovered by observed leaks. Causes of cracking were cited as follows:

<u>Cause</u>	<u>Percent of Reportable Occurrences Involving Cracks</u>
Mechanical vibration	34.8
Weld defects	12.0
Stress corrosion	9.2
Thermal fatigue	6.5
Installation error	6.0
Inadequate support	6.0
Material flaws	3.3

- The most-cited causes of reportable occurrences involving all types of failure, including cracking, were as follows:

<u>Cause</u>	<u>Percent of All Reportable Occurrences</u>
Mechanical vibration	22.3
Weld defects	8.5
Personnel error	6.8
Erosion and cavitation	6.2
Installation error	5.9
Stress corrosion	4.8
Corrosion	3.7
Thermal fatigue	3.4

- Of the reported failures, 31.6% occurred at welds and 9.6% at weld-heat-affected zones of pipes, for a combined total of 41.2%.

Because the survey identifies cracking as the most significant problem of LWR piping systems, there is a need for understanding and preventing the cracking phenomenon. Furthermore, because most of the cracks reported in the survey were discovered through observed leaks, there is also a need for discovering cracks before they cause a leak.

Other significant problems identified by the survey are related to erosion and cavitation, general corrosion, and analysis and design of piping systems.

The problems identified as being significant were analyzed as to their causes, their possible consequences, and the research areas relevant to their resolution. A literature survey was conducted to evaluate the recently completed, ongoing, and planned research efforts in each of the relevant research areas and to determine the state of knowledge in these areas. On the basis of this survey, research needs in each area and specific research topics relevant to resolution of the piping problems were identified.

To properly allocate future funds to programs directed toward resolving the significant problems of LWR piping systems -- that is, to allocate funds in such a way that benefits gained from such research in terms of increased safety and reduced costs are maximized -- a rational approach for assigning priorities to the identified research area is needed. In an attempt to rank or assign priorities to the research areas under consideration, an evaluation matrix was developed. In this matrix, each research area was evaluated with respect to the need for research in that area, time and cost requirements for the needed work, costs of implementing the research findings, possible impact of the research on plant safety, construction costs, operation and maintenance costs, plant availability, safety margins, licensing time, and the generic applicability of the research program. As a result of this evaluation, six research areas were selected as being of high priority.

These areas are:

1. Pipe vibration studies
2. Thermal versus dynamic pipe behavior
3. Piping analysis

4. Fatigue studies
5. Piping system design loads
6. Radiation buildup in corrosion products

Suggested program outlines in each of these research areas were developed. These program outlines are presented in the main body of the report.

Task 2

In Task 2 work, LWR piping system design loads currently in use were identified. These loads include weight, normal operating steady-state pressures and temperatures, pressure and temperature transients, loss-of-coolant-accident (LOCA) and other pipe failure loads, seismic loads, anchor movements, and mechanical loads. The events and phenomena that cause these loads, and methods and procedures used for defining them, were also identified.

Research related to nuclear power plant piping loads is being conducted by the NRC, industry, national laboratories, and foreign organizations. The largest single source for new piping load data is the NRC because concerns of this regulatory agency lead to design-basis events and provide, directly or indirectly, the impetus for piping research. A notable void exists for research related to seismic loads on piping. Although a significant effort exists today to improve the understanding of seismic effects in nuclear power plants, the effort is not aimed primarily at piping. However, current research to improve soil-structure interaction models, structure response, and floor response spectra will have a direct effect on the seismic loads for nuclear power plant piping.

The NRC has developed a list of generic tasks that guide NRC-sponsored research. Unresolved safety issues related to piping loads include water hammer, asymmetric blowdown loads on the reactor coolant system, BWR Mark I and Mark II pressure-suppression containments, anticipated transients without scram (ATWS), pipe-rupture design criteria, and seismic design criteria.

Industry groups, including General Electric Company, Combustion Engineering, Inc., Battelle Columbus Laboratories, Science Applications Incorporated, the Electric Power Research Institute, and others, are conducting research related to piping system loads. National laboratories, including Argonne National Laboratory, Brookhaven National Laboratory, Hanford Development

Laboratory, Idaho National Engineering Laboratory (which is the site of the LOFT and Semiscale test facilities), Lawrence Livermore National Laboratory, Oak Ridge National Laboratory, and Sandia National Laboratories, are also conducting research in this area.

Foreign research is being conducted in Germany, Great Britain, France, Italy, Japan, and Sweden. A summary of this research is given in the report.

Benefits can be obtained by developing new or improved load definitions for LOCA, seismic, and water hammer loads and for load combinations that are based on the new data available from recent research. The NRC has recommended alternative plant modifications to reduce the potential effects of ATWS, rather than the development of new procedures for predicting loads. Also, extensive programs for Mark I and Mark II containment systems are being conducted by owners' groups to develop new load definitions. These programs involve data that are generally proprietary in nature. Until these programs are completed and the results studied, it is not possible to speculate on the feasibility and benefits of developing new load definitions for pressure discharge loads. Benefits from new load definitions may be possible for other load types; however, indications of a need for improving the other loads were not found in the literature.

Recommendations for future research are made in the report on the basis of the need for new load definitions, work currently being conducted, and the availability of data. Considerable research on piping system loads is required in the near future. To efficiently develop research programs for some load topics, the research programs currently being conducted should be completed and the results reviewed. Research programs for pressure suppression discharge loads and water hammer loads are in this category. In addition, the limitations created by the proprietary nature of most piping load information must be carefully assessed, and possible means by which such proprietary information can be used for the benefit of the nuclear industry should be investigated.

Task 3

To achieve the objective of Task 3, a typical LWR piping system was selected for a comparative study. This piping system was analyzed using three methods

for earthquake loading, namely, the conventional linear-elastic spectral analysis, linear-elastic time-history analysis, and nonlinear time-history analysis. The response of the piping system as predicted by each analysis was studied. The predicted responses were compared, and, on the basis of this comparison, the design implications and possible benefits of using nonlinear analysis methods were studied. In addition, a survey of computer codes commonly used in piping analyses was done as part of this task.

The results of the comparative study indicate that the conventional spectral analysis method may not accurately predict piping response and could sometimes produce unconservative results. Gaps that exist at piping supports play a significant role in the response of piping systems. The impact between the pipe and the support may sometimes create large reactions, stresses, and displacements within the piping system.

From the results of the Task 3 study, it is concluded that the conventional analysis methods may be unrealistic and may, under certain circumstances, lead to unconservative designs. However, this conclusion is based on the study of one specific piping system under a specific earthquake loading; it cannot be generalized to all piping systems without more extensive analytical and experimental studies. Therefore, further investigations into the true behavior of piping systems under seismic loads are required before an assessment of the effects of analysis methods on the design of piping systems can be made.

ACKNOWLEDGMENTS

Many people contributed to this project. Dr. Roger E. Scholl was Principal-in-Charge of the project. Dr. Onder Kustu was Project Manager. A major part of Task 2 of the project was subcontracted to Jack R. Benjamin & Associates, Inc., Consulting Engineers, of Palo Alto, California, and was done by Dr. John W. Reed and Dr. Charles A. Kircher. Dr. Ahmad F. Kabir and Dr. T. Allan Moore of URS/John A. Blume & Associates, Engineers, also contributed to the project.

The project was made possible by the funding provided by the U.S. Department of Energy, Division of Nuclear Power Development, and was performed under the management of Sandia National Laboratories, Albuquerque, New Mexico. Dr. Walter A. Von Riesemann was Sandia's Project Supervisor. Valuable suggestions, recommendations, and review of this report by Dr. Walter A. Von Riesemann, Mr. Daniel V. Swenson, and Mr. E. P. Chen of Sandia are gratefully acknowledged.

INTRODUCTION

Background

The work conducted by URS/John A. Blume & Associates, Engineers (URS/Blume), and described in this report is part of the Light Water Reactor (LWR) Safety Technology Program within the Nuclear Power Development Division of the U.S. Department of Energy (DOE). The program is administered by Sandia National Laboratories as Manager of the DOE-LWR Safety Technology Program. The overall objective of the program is to develop and assist in the implementation of reactor systems and concepts that offer significant improvements in safety. Identification and resolution of significant problems related to the design, construction, licensing, operation, and maintenance of nuclear power plant piping systems is a specific goal of the overall program.

Problems with pipes and pipe fittings are responsible for about 10% of all safety-related events* and about 7% of all outage time† at LWRs. Identification and resolution of these problems can lead to increased overall plant safety and significant economic benefits.

LWR piping systems are currently designed and qualified using linear-elastic methods of analysis and load descriptions in accordance with standards set forth in Section III of the ASME Boiler and Pressure Vessel Code. The development and use of more realistic load descriptions and analysis methods in the design of LWR piping systems for normal, upset, emergency, and faulted conditions is necessary for safer and more economical designs.

Objectives

The work described in this report consists of three tasks. The objectives of these three tasks are as follows:

*Internal memo from W. H. Schmidt to W. A. Von Riesemann on LWR component failure data, Sandia National Laboratories, Albuquerque, New Mexico, December 22, 1978.

†Basin, S. L., and E. T. Burns, *Characteristics of Pipe System Failures in Light Water Reactors*, EPRI NP-438, Topical Report, prepared for Electric Power Research Institute, Palo Alto, California, by Science Applications, Inc., August 1977.

- Task 1: To identify the significant problems of LWR piping systems, to determine the research needed to resolve these problems, and to recommend programs to carry out the needed research
- Task 2: To identify and summarize load definitions and load combinations presently used in the design of LWR piping systems, to study the availability of new data, and to investigate the feasibility of developing new load definitions using the new data
- Task 3: To study the possible benefits that can be derived from the development and use of non-linear analysis methods in LWR piping system designs

Scope

To achieve the objectives of the three tasks, the following subtasks were performed:

Task 1:

- Previously conducted studies of problems related to LWR piping systems were reviewed. In addition, recent piping failure data based on Licensee Event Reports were obtained from the Nuclear Regulatory Commission data base and evaluated. On the basis of the evaluations, and the conclusions of the previous surveys, the significant problems and their causes were identified.
- Recently completed, ongoing, and planned research in areas relevant to the resolution of the significant problems of LWR piping systems was evaluated, and further research needs were identified.
- Program outlines were developed for research areas most in need of further work.

Task 2:

- Typical piping systems of LWRs were identified.
- Loads and load combinations typically used in the design of LWR piping systems, as well as the current methods and procedures for determining the loads, were identified.
- Background and bases for the design loads and load combinations were studied.
- Possible benefits that may be realized by developing new load definitions based on recent data were identified.

- Recommendations for further work toward development of improved load definitions for LWR piping systems were given.

Task 3:

- A survey of publicly available computer codes and their capabilities was conducted.
- Analysis methods commonly used in current piping design practice were investigated.
- Sources of nonlinearities in LWR piping systems were identified.
- A typical LWR piping system was selected for a comparative study. This system was analyzed for seismic loading using linear-elastic spectral, linear-elastic time-history, and nonlinear time-history analyses.
- The response of the piping system as predicted by each analysis was studied. The predicted responses were compared, and, on the basis of this comparison, the design implications and possible benefits of using nonlinear analysis methods were determined.

Organization of the Report

Because the reported work consists of three tasks that are quite independent of each other, the report is divided into three parts. Each part summarizes the work performed in a single task. Thus, the work performed in Tasks 1, 2, and 3 is presented in Parts 1, 2, and 3 of this report, respectively.

A glossary of abbreviations used in the report follows Part 3.

PART 1

Task 1: Piping System Problems and Research
Needs Relevant to Their Resolution

1. OBJECTIVES AND SCOPE

The objectives of Task 1 are to identify the significant problems of LWR piping systems, to determine the research needed to resolve these problems, and to recommend programs to carry out the needed research. To achieve these objectives, a study made up of three subtasks was conducted. The results of the work performed under these subtasks are summarized in the chapters that follow. The work of the three subtasks involved:

- Reviewing previously conducted studies of problems related to LWR piping systems and evaluating recent piping failure data not covered in the previous studies to identify significant problems (Chapter 2)
- Evaluating the identified problems and the recently completed, ongoing, and planned research in areas relevant to their resolution to identify research needs (Chapter 3)
- Ranking the research needs and developing program outlines for research areas most in need of further work (Chapter 4)

Tables for Task 1 follow the text.

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2. IDENTIFICATION OF SIGNIFICANT PROBLEMS OF LWR PIPING SYSTEMS

To obtain information on the problems of LWR piping systems, it was necessary first to review the findings of previous studies that have attempted to characterize the causes of such problems and then to study the recent data on LWR piping system problems not covered by the previous studies. In this chapter, the information obtained through these efforts is summarized and evaluated and is used to identify and rank significant problems of LWR piping systems.

Review of Previous Studies

A literature search was conducted to identify previous studies that have attempted to characterize the causes of problems related to LWR piping systems. Six such reports or papers were found. The findings of these studies are summarized below.

Gibbons-Hackney Study. In 1964, W. S. Gibbons and B. D. Hackney completed a study¹ on potential failures of piping systems in nuclear power plants. Their study was based on a survey of piping failures in various industries. Although only a very small number of responses to that survey were from the nuclear industry, the information obtained from the survey was used by the authors to assess the nature and consequences of possible failures in nuclear power plant piping systems. Gibbons and Hackney considered this to be acceptable because nuclear power plant piping and nonnuclear industrial piping are similar with regard to the basic considerations applied to their design, engineering, materials, fabrication, and erection.

Gibbons and Hackney studied a total of 399 cases of piping failure obtained from responses to the survey and from published literature. The significant conclusions of their study were as follows:

- Because piping systems that have been designed and constructed within established code criteria exhibit high reliability, a catastrophic, complete-severance rupture would be unlikely to occur in such systems.
- The most probable failure would be cracking due to low-cycle fatigue resulting from the inability of

the designer to predict environmental and service conditions on a local basis.

- The most likely location for failure to occur would be in pipe or in a weld-heat-affected zone of pipe, particularly in association with a structural discontinuity.
- The growth of cracking-type failures can generally be expected to result in leakage prior to gross or major failure. The importance of adequate monitoring and surveillance must therefore be emphasized.
- Review of the corrosion-erosion failures indicated that this failure mechanism was not adequately considered during plant design. Much of the piping that failed in this manner appeared to have been designated as "field run."
- None of the complete-severance failures (4.7% of total failures) was preceded by detected leakage. However, the majority of all failures were discovered as a result of evident leakage, rather than during routine inspections.

A summary of detailed findings of the Gibbons-Hackney study is shown in Table 1-1. Table 1-2 classifies the most significant causes of pipe failures, comprising approximately 76% of the total failures, by operating temperatures, and Table 1-3 classifies pipe failures by fracture or failure mode.

An interesting observation in Table 1-3 is that fatigue is the most recurring mode of failure (29.1%). According to Table 1-2, a rather large number of failures caused by fatigue at all operating temperatures were due to thermal fatigue; however, if consideration is given only to failures at 600°F or less, which approximately corresponds to the typical LWR operating temperature, the number of failures due to mechanical fatigue are twice as many as the failures due to thermal fatigue (8 versus 4).

Phillips-Warwick Survey. The 1968 report by C. A. G. Phillips and R. G. Warwick² outlines a survey done by the Associated Offices Technical Committee (AOTC), a group of British engineering inspection organizations, and the United Kingdom Atomic Energy Authority. The survey spanned five years (July 1, 1962, to June 30, 1967) and primarily covered failures of pressure vessels designed and constructed to Class 1 requirements of AOTC standards, although some failures of pipe work built to comparable standards were also covered.

Failures are defined in the report as either "catastrophic failures," which are disruptions of the vessel or component that necessitate major repairs or scrapping, or "potentially dangerous failures," which are defects that require remedial action because working conditions might result in a dangerous extension of the known defect.

Time and causes of failures, detection methods, and location of defects were studied, classified, and tabulated by the authors. Also, the failures relevant to nuclear power plants were identified, and the statistics were revised. Finally, a survey of defects reported in United States and British power and research reactors was summarized for comparison purposes.

Table 1-4 presents some of the findings of the Phillips-Warwick study for conventional plants and for nuclear plants. Because most of the failures were due to cracks, the authors studied these failures separately, and their findings are reported in Table 1-5. In both tables, only "service failures" are reported, thus excluding failures that have occurred during fabrication, pressure testing, or at first start-up. Table 1-6 compares causes of failures in nuclear plants with causes of failures in conventional plants built to Class 1 standards.

It should be noted that this study included only 17 reported events from nuclear power plants.

Smith-Warwick Survey. The 1974 survey by T. A. Smith and R. G. Warwick³ is a follow-up to the 1968 Phillips-Warwick survey. The Smith-Warwick survey studied 139 failures that occurred in conventional plants only. It was similar to the earlier survey in that it spanned five years (mid-1967 to the end of 1972) and primarily covered failures of pressure vessels designed and constructed to Class 1 requirements of AOTC standards. Although some failures of pipe work built to comparable standards were also covered, the conclusions of the Smith-Warwick survey, like those of the Phillips-Warwick survey, cannot be related to all pipe work.

Tables 1-7 and 1-8 summarize some of the findings of this survey.

NRC Pipe Cracking Study Group Report. The Pipe Cracking Study Group (PCSG), formed by the NRC to investigate the cracking problems encountered in the austenitic stainless steel piping of Boiling Water Reactor (BWR) plants, summarized its findings and conclusions in a report published in 1975.⁴ The PCSG studied similarities and differences between various BWR plants with regard to the affected piping to identify possible causes of cracking. The following factors were reviewed:

- Coolant environment
- Metallurgy
- History of plant operation
- Pipe configuration, modes of support, and stress
- Mode of operation of the plant

The conclusions of the study ruled out pipe configuration and modes of support as causes of cracking. The mode of operation of the plant (operation of the plant with the bypass valve in the open or closed position) was not considered to be a significant factor in preventing the bypass line from cracking; however, operation of the plants with the bypass valve open was recommended because this mode of operation effectively reduces the oxygen content in the bypass line, eliminates some thermal stresses, and reduces pipe vibrations.

After studying the history of operation for BWR plants, the PCSG concluded that the plants with cracks tended to be those with the:

- Earliest operating license and initial criticality dates
- Greatest equivalent thermal full-power hours
- Greatest average thermal load factor
- Greatest number of outages of 24 or more hours
- Greatest mean time between outages of 24 or more hours

Cracks in the 4-in.-diameter bypass lines and the 10-in.-diameter core spray lines were identified as stress corrosion cracks. Principal causes contributing to this type of cracking were listed as:

- A large amount of dissolved oxygen present in the water

- Sensitization of Type-304 stainless steel around the weld-heat-affected zone of pipes
- High residual stresses in the pipes due to welding

Bush Study. In 1976, a paper by S. H. Bush⁵ on the reliability of piping in LWRs was published in *Nuclear Safety*. To estimate failure probabilities for nuclear piping systems, Bush studied nonnuclear service-failure data from several sources and nuclear service-failure data from NRC files.

As part of his study, Bush reviewed a study by S. A. Wilson⁶ and summarized the conclusions, which are given below:

- The relative probability of severance per component is highest for inadequate flexibility and is successively lower for lack of support, flow and temperature fluctuations, and hydraulic transients.
- The relative probability of severance per component is highest for fabricated or nonstandard joints and is successively lower for high-restraint joints, fittings (reducers, ells, bends, and tees), and straight sections.
- The relative probability of severance per component is highest for small pipes (1 to 6 in. in diameter) and is successively lower for large pipes.
- At operating temperatures, secondary stresses will result in higher relative probabilities of severance than primary stresses.
- Virtually any type or degree of piping damage may be considered to result from some kind of human error or lack of knowledge in design, fabrication, or operation.

In addition, Bush studied Abnormal Operation Reports* available from the NRC and summarized his findings, which are presented in Tables 1-9 through 1-13.

Basin-Burns Study. The 1977 study by S. L. Basin and E. T. Burns⁸ was carried out to characterize pipe system failures, including ruptures, leaks, and defects, that have been discovered and that require repair and to estimate the

*Present terminology is Licensee Event Reports (LERs), a generic term for "reportable occurrences." A reportable occurrence is an unscheduled incident or event that the NRC determines to be significant from the standpoint of public health and safety. For details see Reference 7.

impact of pipe system failures on plant performance (availability and capacity). The authors attempted to verify or refute past claims regarding the factors that significantly influence pipe system integrity and to identify new problems, if any, in need of corrective action or further investigation.

The main sources of data used by Basin and Burns were Nuclear Safety Information Center (NSIC) files, Gray Books, and the File of Evaluated Event Data (FEED). The study covered the period from August 1960 through August 1976.

The significant conclusions of the study regarding the reported pipe failures were as follows:

- The principal causes of the reported pipe failures are not related to inconsistencies in or failure of current design requirements. Rather, they appear to be related to phenomena that are not readily predictable; hence, the methods of precluding pipe failures due to such phenomena have depended, for the most part, upon engineering judgment.
- The principal causes of the reported pipe failures ranked by frequency of occurrence are:

Vibration (high cycle fatigue)	25.1%
Fabrication errors	12.8%
Stress corrosion cracking	11.3%
Erosion	6.9%
Thermal fatigue (low-cycle fatigue)	5.4%
Corrosion	2.9%

- The distribution of piping failures among plant subsystems is:

<u>Plant Type and Subsystem</u>	<u>Percent</u>
<u>PWRs</u>	
Primary plant charging system	26.8
Feedwater system	15.9
Decay heat removal system	12.2
Other	45.1
<u>BWRs</u>	
Feedwater system	21.5
Recirculation system	19.0
Gas systems	13.2
Core spray piping	10.7
Other	35.6

- The majority of the reported failures occurred in small-diameter piping. For PWRs, 70.7% of the reported failures occurred in pipes less than 6 in. in diameter, and 51.2% of the reported failures occurred

in pipes 1 in. or less in diameter. For BWRs, failure proportions were 70.2% for pipes less than 6 in. in diameter and 29% for pipes 1 in. or less in diameter.

- Approximately 9.0% of all the pipe failures involved a pipe rupture.
- The data regarding the specific location of the piping failures indicate that:
 - 54% occurred in welds or in the weld-heat-affected zones of pipes.
 - 40% occurred in the pipe wall.
 - 6% occurred in threaded pipe joints.
- The number of pipe failures per plant-year of operation was higher for BWR plants than for PWR plants.
- The average number of pipe failures per PWR plant declines gradually over the first six years of operation whereas the failure rate for BWR plants increases gradually and peaks at about five years from the initial criticality.

The conclusions of the study with regard to the impact of pipe failures on plant availability were as follows:

- Approximately 47% of the pipe failures were discovered during a plant shutdown. The remaining 53% fall into the following categories:

No shutdown is required	23%
A normal plant shutdown is performed	26%
An automatic scram occurs	2%
A manual scram is performed	2%
- A forced reactor shutdown due to a pipe failure may be expected to occur, on the average, once every 3.6 plant-years.

Other Sources. As part of this study, a letter inquiring about the research activities related to LWR piping problems that are being carried out was mailed to various organizations in the nuclear industry. The number of responses received was low. However, some of the responses received commented on some problem areas and recommended research programs toward solving these problems. These recommendations were evaluated and incorporated into this study where appropriate.

Discussion. Each of the six previously conducted studies has used a somewhat different approach in selecting and classifying data on piping problems and

in interpreting the relevance of the data to nuclear piping systems. In addition, the criteria used to determine the significance of piping problems vary from study to study.

The earlier studies -- those by Gibbons-Hackney, Phillips-Warwick, and Smith-Warwick -- had to rely primarily on data available for nonnuclear piping systems to identify the problems most likely to be encountered in nuclear piping systems. Phillips-Warwick and Smith-Warwick have compared the limited nuclear data available at the time of their studies with nonnuclear data judged to be relevant to nuclear piping systems, thereby implying the applicability of the nonnuclear data to the studies of nuclear piping systems.

The Bush study points out the subjectivity involved in determining which non-nuclear data or events are relevant to nuclear piping systems. In addition, the study attempts to estimate relative failure probabilities for different pipe sizes, materials, and other variables of piping systems. The author notes that this attempt was limited due to the incompleteness of the available data.

The Basin-Burns study is the most comprehensive study conducted for the purpose of identifying significant problems of LWR piping systems. The study has utilized a large body of data submitted by nuclear power plants to the NRC. Most of the data used in the study are from LERs, supplemented by plant outage and maintenance data as needed and if available. However, the authors point out that the LERs were incomplete with regard to the causes of failures, time of the events, shutdown times, and modes of failure. They also note an increased availability of LERs during the period from 1974 to 1976.

Study of Recent Failure Data

The studies of piping system problems reviewed in the previous section were based on data for the period from 1960 to 1976. Some of these studies either did not cover problems in nuclear piping systems specifically or were based on incomplete data. More up-to-date and complete data about problems in nuclear piping systems are now available. For this task, LERs for the period from January 1, 1969, through May 16, 1979, were obtained from the NRC. The

reportable occurrence summaries for pipe systems were examined and were used for two surveys, which are described below.

Survey of Reportable Occurrences Involving Pipe Systems. A general survey covering all the data available on reportable occurrences involving all components of pipe systems, including pipes and fittings, pumps, valves, valve operators, and pipe supports, was conducted. Causes of failure were classified as they were given in the LERs (i.e., component failure, defective procedures, design/fabrication errors, personnel errors, external causes, and other causes). The number of reportable occurrences for each component and each classification were counted and tabulated as shown in Table 1-14. According to this table, most of the reportable occurrences (2,026 out of 4,655) involved valves. The majority of the causes of reportable occurrences have been classified as "component failure," with "personnel errors" being a distant second.

Survey of Reportable Occurrences Involving Pipes and Fittings. A separate survey of reportable occurrences involving pipes and fittings was conducted to more specifically study the characteristics of failures that occur in these components of piping systems. Pipe and fitting data from LERs submitted during the period June 1, 1976, through May 16, 1979, were collected. If more than one pipe or fitting failure was reported in a single LER summary, that summary was counted as one reportable occurrence. The survey covered a total of 354 reportable occurrences involving pipes and fittings.* The data on these occurrences were studied in an effort to answer the following questions:

- How did the pipes fail?
- Where did the pipes fail?
- Why did the pipes fail?
- What were the effects of the pipe failures on plant operation?
- What were the safety consequences of the pipe failures?
- Which subsystems suffered significant numbers of failures?

*It should be noted that a reportable occurrence is not necessarily a failure and also that not all events reported in LERs are for Class 1 piping.

Tables 1-15 through 1-20 show the answers to these questions obtained from the study of the 354 reportable occurrences.

Discussion. The two surveys used the most recent LER data not covered by the previous studies to identify piping system problems. The LER format has been set up to give a comprehensive description of the reportable event. Provisions have been made for "cause subcodes," as well as more general "cause codes," and space has been reserved to report plant shutdown effects, radiation release, and safety consequences, as well as other information. However, even the most recent reports are not as complete as they could be: Cause subcodes are rarely specified, shutdown time is almost never given (except to report that no shutdown was required), and radiation release information is not consistently reported. All these limitations in the available data have made it necessary to carefully study each event description to determine, somewhat subjectively, the cause and location of the failure.

Proper classification of causes of failures proved to be a difficult task as well, since ultimately all failures result from lack of knowledge about the physical phenomena, material properties, etc., which can be classified as human or design errors, or from other, more direct human errors, such as using a pipe as a footstep. However, a classification compatible with the event descriptions and with the previous surveys was developed. Sometimes multiple causes were cited for a single failure; for example, a weld failure was classified as being due to a weld defect, as well as being due to personnel error, because a welder's error was implied in the report.

The surveys were directed toward answering several questions to characterize the LWR pipe system failures: How, where, and why do the pipe systems most often fail, and what are the safety and operational consequences of the failures? Answers to these questions were determined for each reportable occurrence and presented in Tables 1-14 to 1-20. Several observations can be made by studying these tabulated findings.

Table 1-14 shows that, if all components of piping systems are considered (including pipes and fittings, pumps, valves, valve operators, and pipe supports), then the components with highest rates of failure are valves (43.5%) and valve operators (18.8%), with a combined failure rate of 62.3%. Fail-

ures of pipes and fittings, pumps, and supports represent 16.5%, 12.7%, and 8.4% of the total reportable occurrences, respectively.

Component failure is the most cited cause of the reportable occurrences. For valves and valve operators, personnel error is the second most cited cause of failure. For all other components, design/fabrication error is the second most cited cause of failure.

Tables 1-15 through 1-20 show that, if only the failures of pipes and fittings are considered, then during the three-year period studied:

- 70% of the reported failures were discovered because of a leak. Partial cracks that were discovered by various means without any leakage constituted 11.9% of the total.
- 40.1% of the failures were in the form of through-the-wall cracks. Complete rupture of pipes was cited in only 2.2% of the reportable occurrences. Cracking, either partial or through the wall, was the mode of failure in 52.0% of the reportable occurrences.
- 31.6% of the failures occurred at welds and 9.6% at weld-heat-affected zones of pipes, for a total of 41.2%; 24.9% occurred at the pipe base metal.
- The most cited causes of failure were as follows (shown as percentage of total reportable occurrences):

<u>Cause</u>	<u>Percent</u>
Mechanical vibration	22.3%
Defective welds	8.5%
Personnel error	6.8%
Installation error	5.9%
Erosion and cavitation	6.2%
Stress corrosion	4.8%
Corrosion	3.7%
Thermal fatigue	3.4%
Unknown causes	25.1%

As can be seen, mechanical vibrations are the most important cause of failure. The vibrations are due to pump-pipe resonance, or they are induced by fluctuations in fluid flow.

- The findings of this study with regard to the principal causes of pipe failure compare with those of the 1977 Basin-Burns survey⁸ as follows:

<u>Principal Causes of Pipe Failure</u>	<u>Percent of Reportable Occurrences</u>	
	<u>Basin-Burns Survey</u>	<u>This Study</u>
Mechanical vibration	25.1	22.3
Installation error/ defective welds	12.8	14.4
Erosion and cavitation	6.9	6.2
Stress corrosion	11.3	4.8
Corrosion	2.9	3.7
Thermal fatigue	5.4	3.4

The comparison indicates that conclusions with regard to the principal causes of pipe failure have not changed much since the Basin-Burns survey was completed. However, one significant exception is the notable decrease in stress corrosion problems during the last three years.

- In 41.2% of the reportable occurrences, the power plants were down when the failure was discovered. In 28.8% of the LER summaries, no information was given on the effect of the failure on the plant operation. In only 4.5% of the reportable occurrences was a manual or automatic shutdown due to the failure reported. It seems impossible to use these data to determine the impact of various pipe failures on plant operation. Closer inspection of the LERs revealed that 8 failures out of the 16 that were reported to cause a shutdown occurred in the coolant recirculation system and that 6 out of these 8 failures were caused by mechanical vibrations.
- In 6.5% of the reportable occurrences, radioactive release was reported. Personnel exposure or injury was reported in 2.5% of the cases. The accuracy of these numbers is questionable because there were cases in which the event description seemed to imply radiation release or personnel exposure, although no specific reference to the radioactivity release or personnel exposure was given in the LER.

Out of 23 cases in which radioactive release was reported, 8 cases occurred in gas and liquid radioactive waste management systems; 4 cases each were reported for chemical, volume control, and liquid poison systems and coolant recirculation systems.

- Since failures in the gas and liquid radioactive waste management systems constitute only 4.2% percent of the total failures, these systems seem to have a very high radioactive release rate per reportable occurrence.
- Even though the relationship between piping failures and pipe diameters was not investigated in this study, it should be noted here that both the Bush and

the Basin-Burns surveys indicate that the majority of failures occur in small-diameter piping that is not necessarily designed to Class 1 requirements.

Identification and Ranking of Significant Problems of LWR Piping Systems

In the first two sections of this chapter, data from previous surveys and from the study of recent LER summaries were presented and findings were discussed. On the basis of the number of occurrences, certain trends characterizing the problems of piping systems were observed.

In determining the significance of each problem or problem area, other factors should ideally be considered in addition to the recurrence rate. These factors include safety consequences, effects on plant operation and availability, effects on licensing procedures, and others. Although the data available for this study do not permit consideration of all these additional factors in a rational way, the limited data were used in an attempt to correlate the failures with safety consequences and with the subsystems in which these failures occurred. Some trends were observed indicating that failures in gas and liquid radioactive waste management systems caused most of the reported radiation release. It was also observed that a large portion of the failures causing the reported plant shutdowns occurred in the coolant recirculation systems. Beyond these few cases where the safety and availability effects can be considered, the main criterion for ranking problem areas of LWR piping systems has to be the recurrence rate.

It was noted previously that most pipe failures occur in the form of cracks and are discovered through observed leaks. Only a small number of defects or cracks are discovered before they cause a through-the-wall crack and leak. These facts point out the need for understanding and preventing the cracking phenomenon and/or for discovering the cracks before they cause failure. Therefore, cracking of piping is ranked as the most significant problem.

Other significant problems related to LWR piping systems are, in order of importance, erosion and cavitation, corrosion, and analysis and design problems. In Chapter 3, these problems are discussed, their causes are analyzed, and the research areas relevant to their resolution are determined.

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3. EVALUATION OF IDENTIFIED PROBLEMS AND RESEARCH AREAS RELEVANT TO THEIR RESOLUTION

In the preceding chapter, an attempt was made to identify the significant problems of LWR piping systems. In this chapter, the problems identified as being significant are analyzed as to their causes, their possible consequences, and the research areas relevant to their resolution. The following is a list of the significant piping problems and the research areas relevant to their resolution.

<u>Problem</u>	<u>Relevant Research Area</u>
1. Pipe cracking	Flaw detection and evaluation Fracture mechanics Fatigue studies Stress corrosion cracking Postulated pipe rupture Pipe vibration Welding technology Thermal versus dynamic pipe behavior
2. Erosion and cavitation	Erosion and cavitation
3. General corrosion	Radiation buildup in corrosion products
4. Analysis and design problems	Pipe design loads Piping analysis

A literature search was conducted to evaluate the recently completed, ongoing, and planned research efforts in each of the research areas listed above and to determine the state of knowledge in these areas. On the basis of this literature search, research needs in each area and specific research topics relevant to resolution of the piping problems are identified.

Problem Area 1: Pipe Cracking

Pipe cracking is the most significant mode of failure in LWR piping systems. Part-through and through-the-wall cracking represent over 50% of all reported pipe failures. Therefore, any research program that can lead to a reduction of cracking in LWR piping systems will have a significant effect on the safety and availability of LWR plants.

Causes of the Problem. The survey of reportable occurrences involving pipes and fittings conducted as part of this study indicated that the significant

causes of cracking in pipes are:

<u>Cause</u>	<u>Percent of Report- able Occurrences Involving Cracks</u>
Mechanical vibration	34.8%
Weld defects	12.0%
Stress corrosion	9.2%
Thermal fatigue	6.5%
Installation error	6.0%
Inadequate support	6.0%
Material flaws	3.3%

Pipe cracks are initiated by preexisting material flaws or weld defects. They may also be initiated by intergranular stress corrosion or by stress concentrations at welds or structural discontinuities.

Material flaws and weld defects cannot be completely eliminated. However, the existence of most flaws and defects can usually be determined through nondestructive examination techniques, even though some are not detected through these techniques. At present, acceptable sizes of detected flaws are specified by the ASME Code. Stress concentrations cannot be completely eliminated either, but through careful design and detailing it is possible to minimize them. The mechanism by which intergranular stress corrosion cracking is initiated has been the subject of extensive studies. Such cracking is known to be caused by a combination of heat sensitization of some stainless steels, high stresses, and high oxygen content in the water.

Once initiated, a crack propagates through one or a combination of the following mechanisms:

- Stresses caused by steady loading
- Impact loads such as water hammer
- Cyclic stresses caused by
 - Mechanical vibration
 - Thermal cycles
 - Other cyclic service loads
- Intergranular stress corrosion

The presence of a corrosive environment at the crack location might increase the rate of propagation of the crack.

Consequences of the Problem. Under certain circumstances, a crack might grow in a stable manner and may be arrested (e.g., when it is initiated in the tension zone and penetrates into the compression zone, as happens when cold water is injected into a hot pipe), or it might grow rapidly in an unstable manner, causing a rupture. Where cracking is stable, leakage before a complete rupture is expected. Leakage is one of the most common modes of discovery of cracks. Leak-before-break failure is preferable to sudden rupture failure because in the former case the plant can be shut down safely, and the pipe can be repaired.

The safety significance of piping failures depends on the piping system in which the failures occur, the contents of the pipes, the auxiliary systems available, and other factors. The NRC does not consider pipe failures to be a significant health and safety hazard for the public. The reasoning for this is that nuclear power plants are designed with a defense-in-depth approach to safety. There are three levels of safety built into the design: (1) components are designed and constructed to very high standards, (2) protective, engineered safety features are provided to monitor and correct abnormal operation conditions, and (3) if an accident occurs, safety features exist to mitigate the accident consequences. Hence, if an important pipe breaks, the plant's emergency systems are supposed to come into operation, and, in some cases, an automatic shutdown might occur. Even if radioactivity is released due to the failure, the containment system is expected to prevent the release of the radioactivity into the environment. Therefore, the NRC concentrates its research efforts on the safety of the reactor pressure vessel, and specifically on the postulated loss-of-coolant accident.

In spite of the NRC philosophy, a pipe failure can lead to a significant accident and can endanger public safety if the failure is followed by and combined with malfunction of safety systems and operator errors. Also, even for pipe failures where the radioactivity release is contained, the plant personnel and the maintenance and repair crews will be exposed to radiation. Therefore, the safety significance of piping failures, and specifically of pipe cracking, cannot be overlooked.

Relevant Research Areas. Resolution of the pipe cracking problem and estimation of safety margins require a complete understanding of the mechanisms

by which cracks are formed, propagate, and cause ruptures. Nondestructive examination techniques are needed to discover cracks in their early stages. Analytical fracture mechanics methods are needed to accurately predict a crack's stability, behavior, and safety margin. Operational data are needed to improve understanding of the actual loads, displacements, and environmental effects to which piping system components are subjected during the life of a plant.

The research areas relevant to the resolution of the pipe cracking problem are discussed below.

Flaw detection and evaluation is an important research area because it is recognized that LWR piping components will always contain some material flaws and defects introduced during the manufacturing and construction processes and that cracks will develop in these components during the lifetime of the plant. Benefits to be gained by discovery of these defects before they can lead to failure are significant. The development of reliable methods for detecting material and weld defects and cracking would reduce the possibility of pipe rupture significantly, and the effect of such development on safety would be great.

Early detection of defects is also very important for economic reasons. Repair of cracked piping detected by in-service inspection during scheduled refueling outages is far less costly than shutting down a plant to perform repairs after a leak develops. Therefore, the impact of developments in this research area on plant availability, as well as on operation and maintenance, would be great.

Nondestructive examination techniques are applied to fabricated piping products both before installation and periodically over the plant lifetime (during in-service inspections). These techniques include:

- Visual examination
- Liquid-penetrant examination
- Magnetic particle examination
- Radiographic examination
- Ultrasonic testing

Criteria and details for the examination techniques are delineated in Sections III, V, and XI of the ASME Code. The code specifies allowable flaw sizes, on the basis of linear elastic fracture mechanics, for various locations within the piping system components. Any developments in detection methods leading to increased capability for reliably detecting smaller flaws would increase the margin of safety.

Ultrasonic testing is presently the most accepted and useful volumetric inspection technique. Section XI of the ASME Code specifies such testing for periodic in-service inspection of the welds and the heat-affected zones of pipes. However, the reliability of ultrasonic testing for flaw detection and characterization using the code procedure is often inadequate.⁹ For austenitic stainless steel and welds and for detecting intergranular stress corrosion cracking, these testing methods are especially unreliable and do not give reproducible results.

Significant advances have been made recently in the application of ultrasonic testing techniques and in the interpretation of the results of such testing. Capabilities have been developed in the use of computers for determining the size and shape of defects (synthetic aperture focusing and adaptive learning network), and analytical methods have been developed to interpret the signals.

The NRC has a number of programs aimed at improving ultrasonic testing methods. New techniques are being validated in the laboratory with realistic plate and piping samples. The NRC intends to finalize work on the improved testing methods in 1981.⁹ Combustion Engineering, Inc., is planning to study various parameters such as inspector variables (different personnel at different times), equipment variables (different instruments or transducers at different times), and inspection variables (varying surface finishes, etc.) in order to evaluate the accuracy and repeatability of the ultrasonic testing specifications in Section XI of the ASME Code.¹⁰

The Electric Power Research Institute (EPRI) has recently undertaken research projects to optimize in-service inspection procedures using code-acceptable methods and to develop ultrasonic testing signal-processing methods that are less dependent on operator skills and signal amplitude.¹¹ The Adaptive Learning Network method of signal analysis was applied to Type-304 stainless

steel pipe welds, and the results indicate that this method has good potential to increase the dependability of ultrasonic testing techniques. Further developments are under way and EPRI research programs include improvement of the examination methods and the introduction of these methods into practical use by utilities.¹²

In addition to the nondestructive examination methods, a new area of research is the development of continuous on-line surveillance methods. Detection of flaws and monitoring of flaw or defect growth during service are the most effective means of preventing unexpected failures of large primary-system components during service.⁹ A method under development is acoustic emission, which makes use of the fact that slight deformation or crack growth in metals produces sounds that can be detected by suitable receivers. This method also has promise for detecting leaks.¹³ Acoustic emission probes eventually will be placed on vessels and piping to monitor signals emitted during operation, and other nondestructive examination methods will be used during shutdown periods for further characterization of detected flaws. NRC-supported research on this method is progressing; field testing is expected to begin in 1980,⁹ and recommendations for the ASME Code implication are expected to be developed by 1983. Westinghouse Electric Corporation (Westinghouse) also has an ongoing research program for the development of this method. They are currently involved in field testing and refinement of the method.¹³

Fracture mechanics research programs study the mechanisms by which cracks are initiated from existing flaws and by which they subsequently propagate and are arrested. If these mechanisms are well understood, the flaws discovered by non-destructive examination methods can be analyzed to determine whether or not they constitute a threat to the structure over its design life, and the margin of safety can be accurately predicted. Potential savings to the nuclear industry from reduced shutdowns due to reduced uncertainty in establishing failure criteria from detected flaws can be significant.

Present fracture criteria used in the industry and accepted by the NRC for evaluating piping with cracks are based on the principles of linear elastic fracture mechanics (LEFM). In LEFM, attention is restricted to the initiation of crack growth in materials that behave in an essentially linear elastic manner. Typical pressure boundary materials used in LWR piping are ductile

at temperatures slightly above room temperature, and crack initiation and propagation at operating temperatures are accompanied by large-scale plastic deformations. Therefore, classical LEFM techniques are not applicable to LWR piping materials, and the use of these techniques will result in considerable underestimation of the margin of safety for piping with discovered flaws or cracks.

Considerable research to develop elastic-plastic and fully plastic fracture mechanics methodologies has been initiated in the industry. The NRC's Metallurgy and Materials Research Branch supports a significant research program in fracture mechanics.⁹ Even though the main concern of this program is reactor pressure vessel safety, the results of the research and the technology developed will aid in solving the pipe cracking problem. As part of the NRC program, experimental data on the fracture toughness of various reactor vessel and piping steels are being obtained. Research is under way to develop a theory and methodology for elastic-plastic and fully plastic fracture analysis of reactor pressure vessels and primary piping. The objective of this research is the quantitative prediction of allowable flaw sizes and stress levels using appropriate toughness values under all operating conditions. Once developed, the methods are expected to be validated with experimental data for incorporation into codes and standards. Also, methodologies are being developed to predict the arrest of a running crack before it reaches a critical size.

Under EPRI sponsorship, research programs on the phenomenon of crack arrest, as well as studies of crack initiation and propagation, are being carried out.¹⁴ In addition, General Electric Company (GE) and Battelle Columbus Laboratories are working jointly towards prediction of the behavior of flawed structures in the plastic regime.

The objectives of these studies are to develop the theoretical background necessary to fully describe the plastic behavior of flawed structures and to develop the proper analysis capability for providing the stresses and strains that initiate crack growth and failure instability. Appropriate material properties required for the analyses will be developed, and benchmarks will be established.

Research efforts in the elastic-plastic and fully plastic fracture mechanics area are not yet conclusive, and the need for studying crack initiation, propagation, and arrest behavior under realistic conditions exists.

Fatigue studies constitute another important area of research relevant to the pipe cracking problem. One of the mechanisms by which small flaws, material defects, and inhomogeneities that exist in LWR piping can cause crack initiation and growth is cyclic loading during operation of the plant. Cyclic loading may be caused by start-up and shutdown cycles of the plant (thermal fatigue), or it may be due to mechanical vibration of the piping system.

The effect of cyclic loading on crack growth is important in that a crack could reach a critical size through this mechanism, resulting in a pipe rupture. Because the survey of pipe and fitting failures indicated that mechanical vibrations are the most recurring causes of pipe cracking, the determination of cyclic crack growth rates under realistic conditions would have a significant impact on the accurate evaluation of margins of safety for flawed pipes. At present, the rules for fatigue design are given in Section III of the ASME Code, while Section XI of the ASME Code includes guidelines for interpreting the nondestructive evaluation results and for calculating the remaining safe lifetimes for flaw-containing structures.¹⁵ The data on which the latter guidelines are based were generated in high-temperature, high-pressure environments. However, later research done at Westinghouse¹⁶ has indicated that the code design curve was unconservative for lower load frequencies or higher mean loads. GE studies carried out under NRC sponsorship at the GE BWR test loop have verified this finding and have established threshold values on cyclic loading frequency for intergranular stress corrosion cracking behavior in both Type-304 stainless steel and Inconel 600 in the BWR coolant environment.¹⁵ Because slow cycling more nearly approaches realistic service conditions, there is a need for research on fatigue behavior under varied loading times, long hold times, and various load wave forms.

The NRC research program includes studies of cyclic crack growth rates under realistic load conditions in the BWR and PWR environments.⁹ Planned research will also correlate laboratory research results with the behavior of similar materials under field conditions. NRC research is expected to be concluded in 1981.

The EPRI research program for 1979-1983 includes efforts to determine the corrosion fatigue properties of reactor pressure vessel steels and to develop a design curve for corrosion fatigue aimed at modifying the fatigue crack growth analyses contained in Section XI of the ASME Code.¹²

No research program within the nuclear industry could be found for the study of high-cycle fatigue behavior of piping steels and weldments. Because the survey of pipe and fitting failures indicates that high-cycle fatigue at welds or at weld-heat-affected zones of pipes is a significant mode of failure, there is a need for research on this topic. This research should be coordinated with the study of mechanical vibrations and should consider conditions where high-cycle small-amplitude vibrations occur about high mean stress levels.

Stress corrosion cracking refers to the phenomenon of cracking of metals by a combination of relatively high stress and a corrosive environment.⁴ Either transgranular or intergranular stress corrosion cracking may occur in austenitic stainless steels, depending on the specific metallurgical characteristics of the steel, as well as on stress and environment. Intergranular stress corrosion cracking (IGSCC) of austenitic stainless steel piping is the type of stress corrosion cracking that has been encountered in the LWR stainless steel piping systems.

The first significant cases of IGSCC were discovered in 1974 in the Type-304 stainless steel piping of several BWR plants in the United States. These incidents occurred in 4-in.-diameter recirculation loop bypass lines and 10-in.-diameter reactor core spray lines. Since 1975, IGSCC has been found in other piping systems of BWR plants, including a 24-in.-diameter recirculation loop of a German plant. Most recently, cracking has been discovered in the feed-water lines of eleven PWR plants. Even though the causes of these latest cracks have not yet been identified, preliminary studies indicate the causes to be corrosion fatigue, thermal fatigue, or stress-assisted corrosion.¹⁷

IGSCC has been studied extensively. Major studies have been undertaken by the NRC, EPRI, and GE. The 1975 NRC study by the Pipe Cracking Study Group⁴ (see Chapter 2) identified the causes of IGSCC to be the sensitization of the

stainless steel at the weld-heat-affected zone of pipes, high residual stresses adjacent to welds, and the presence of oxygen in the coolant. The group recommended increased in-service inspections of Type-304 and Type-316 austenitic stainless steel pipes, stricter leak detection specifications, replacement of certain piping with other materials less susceptible to IGSCC, and modification of plant operation procedures to minimize the oxygen content in the coolant and to avoid low-flow or stagnant branch lines.⁴

A new Pipe Cracking Study Group was activated by the NRC in 1978 after discovery of IGSCC in large-diameter pipes. This group reviewed existing information on IGSCC and developed conclusions and recommendations relevant to the current status of IGSCC, the significance of the problem, and the reliability of crack detection methods and methods available to minimize IGSCC.¹⁸

The new group's recommendations, which were in addition to those made by the previous group, include avoiding the use of Type-304 and Type-316 stainless steel in BWR piping systems unless special measures are taken, minimizing the oxygen content in the coolant throughout the operating cycle, incorporating the recent developments in ultrasonic detection methods into the ASME Code, and conducting investigations to determine the effects of actual BWR operating stress and thermal loading on IGSCC.

The new group also concluded that IGSCC in large-diameter BWR stainless steel piping is unlikely to cause unstable crack growth and excessive loss of coolant under normal operating conditions. It further concluded that, even if a loss-of-coolant accident were to occur, the emergency core cooling system would provide protection, and therefore IGSCC would not present a hazard to public health and safety. However, a major safety concern with regard to the IGSCC in LWR piping is that a seismic event or water hammer can cause a pipe rupture at the cracked section, thereby initiating a loss-of-coolant accident. In view of recent incidents, relying on emergency core cooling systems to rule out more serious consequences does not seem to be very realistic. Therefore, the impact of IGSCC on safety, as well as on plant availability and operation and maintenance, is considered to be significant.

As a result of recent studies, the following factors that contribute to IGSCC have been characterized:¹⁹

- Patterns of residual welding stresses for various pipe sizes of Type-304 austenitic stainless steel
- Highly variable material responses within the commercial Type-304 specifications of the American Society for Testing and Materials
- Sensitization effects resulting from welding operations
- Plant water-chemistry variations as a function of temperature and operation
- Electrochemical potentials (in-plant and laboratory) to aid understanding of the phenomenon and to guide testing conditions

A realistic working model has been established to describe the IGSCC phenomenon. The model incorporates the major factors that contribute to IGSCC, such as stress state, sensitization, and environment. The interrelationship between these contributing factors is not yet completely understood or quantified. While the recent research efforts of the NRC, EPRI, and GE have concentrated on short-term remedies, the present efforts are directed toward a complete quantification of the factors contributing to IGSCC and the justification and qualification of corrective actions that can be applied to the materials, environment, and operation of existing and new plants. Planned research includes the statistical qualification of alternate BWR piping materials and application of the research results to finding permanent solutions to the IGSCC problem.¹²

A need exists for incorporating the results of recent research efforts into the codes and standards used for design purposes. Presently the ASME Code does not contain any specific provisions or guidelines for designing against environmental effects such as corrosion, erosion, and IGSCC. The code recommends an increased material thickness to compensate for environmental effects. This approach requires good judgment and experience with the specific phenomenon on the part of the designer and is not adequate. Therefore incorporation of new knowledge into the design codes will have a significant effect on the reliability of future designs.

Postulated pipe rupture has always been considered in the design of nuclear power plants. Presently, the design basis accident assumed for reactor

primary coolant piping is an instantaneous double-ended rupture. The reasons for postulating such an accident are as follows:²⁰

- Calculation of the design basis containment pressure
- Consideration of the loss of coolant in the design basis for the emergency core cooling systems

If such an accident occurs, the ruptured pipe could whip violently, and the safety-related components in the vicinity of the pipe could be damaged by the whipping pipe or by the discharging coolant jet. The preferred design for the consequences of the postulated rupture is the separation or isolation scheme described in Section 3.6.1 of the NRC's Standard Review Plan (SRP), whereby essential equipment and systems will not be affected by the ruptures.²¹ When this approach is not possible or practical, pipe whip restraints must be provided to prevent the ruptured pipe from damaging essential structures, equipment, or systems. Due to space limitations inside a nuclear plant, the increased number and size of these rupture restraints could interfere with in-service inspection and normal operation of systems and also could increase the cost of construction. Establishment of pipe-rupture criteria that are based on rational, realistic mechanisms would have a significant impact on the design effort, the time required for licensing, and the operation and maintenance of the plants. If such criteria proved to be less stringent than the present criteria, safety margins would be better established without sacrificing conservatism.

The major elements specified by the pipe-rupture criteria are postulated pipe-rupture locations, break type (circumferential or longitudinal break); break opening area, break opening time, and jet characteristics. Due to lack of experience with this phenomenon, and lack of theoretical and experimental background, the present pipe-rupture criteria are based on the conservative worst-case assumptions intended to assure upper-bound design considerations. Recent advances in the fields of fracture mechanics and nondestructive testing, as well as developments in analytical and experimental capabilities, now permit the development of more rational criteria.

The developments in nondestructive evaluation techniques are reviewed in a previous section of this report. Improved inspection capabilities are expected to enable integrity evaluations to be performed with fewer arbitrary

assumptions on flaw size and shape and to allow more precise identification of initial crack size and potential effects of any detected flaw.²² Research results in fracture mechanics, and especially the results of studies on elastic-plastic and fully plastic fracture mechanics, crack propagation, and crack arrest, can be used in developing pipe-rupture criteria. However, development of plastic fracture or ductile fracture methodology is still incomplete, and additional work needs to be done in this field.

Significant research programs related to the pipe rupture problem are presently being conducted by Battelle Columbus Laboratories and reactor manufacturers. However, research needs still exist in the following areas:

- Development of increased analytical predictive capability using ductile fracture mechanics principles
- Determination of the effects of all possible realistic load conditions and their combinations on the opening and propagation of cracks (these conditions include evaluation of cracked pipes with various crack configurations, effects of seismic loads, overpressurization, mechanical loads, and thermal shock)²²
- Determination of the effects of existing cracks on the overall behavior of pipe and the redistribution of forces and stresses before, during, and after rupture²²
- Experimental verification of analytical methods and establishment of benchmarks
- Review of the existing criteria for location of pipe breaks, break area, and time of rupture; development of more realistic criteria that are based on newly developed analytical methods and experimental data

Pipe vibration is the most cited cause of LWR piping failures according to the survey of piping problems and their causes conducted as part of this study, as well as some previous surveys. These vibrations can be attributed to:

- Resonance between the pipe and an operating pump
- Dynamic forces generated at elbows and tees by fluctuations in fluid flow
- Support vibrations, including excitation of one pipe by a larger pipe to which it is attached

The stresses generated by low-amplitude vibrations of pipes may be small compared with the stresses generated by other loading conditions. However, if these fluctuating stresses are added to already high residual stresses

existing near welds, the resulting phenomenon is cyclic loading about high mean stress levels, sometimes at or above yield stress.

Recent studies¹⁵ have confirmed that crack propagation rates under cyclic loading are higher for larger mean stress values. Also, most research on fatigue and crack propagation has concentrated on low-cycle fatigue or large strain reversals. Data from tests that consider low-amplitude vibrations at high mean stress levels have not been encountered.

Presently the provision in the ASME Code for pipe vibrations is in the form of a recommendation for "arranging and supporting the pipe so that vibration will be minimized. The designer shall be responsible, by design and by observation under start-up or initial service conditions, for ensuring that vibration of piping systems is within acceptable levels." In general, no systematic analytical effort is made in piping design, especially for small balance of plant (BOP) piping, to account for the effects of vibrations.

The literature search did not yield much information or published data on the vibration of pipes, except one article on the measurement of flow fluctuations to determine the dynamic forces applied on an elbow.²³

Vibration of piping, especially that of small piping, is not by itself a very great safety problem. However, the reduction of pipe cracking and leaks caused by vibration would have a significant effect on the cost of operation and maintenance, and would reduce the exposure of the repair personnel to radiation.

Research toward eliminating pipe cracking due to vibrations is needed. Such research includes:

- Collection of operational data from various piping systems in order to better characterize the problem
- Development of analytical techniques that would consider fluid fluctuations and pipe-pump resonance in piping analysis
- Investigation of possible use of snubbers in vibration damping
- Experimental studies to investigate fatigue and crack propagation behavior in pipes and welds, under low-amplitude and high mean stress vibrations

Welding technology is an important research area in that over 40% of the reported failures considered in the survey of pipe failures occurred at welds or at weld-heat-affected zones of pipes. Improvements in welding technology could help reduce the number of failures occurring at welds and thereby increase the availability of LWR plants.

Problems related to welds can be classified into several groups:

- Problems inherent in the welding process that cannot be avoided but may be minimized (e.g., neither heat sensitization of the metal adjacent to the welds nor residual stresses can be avoided completely, but these problems can be minimized using certain techniques)
- Problems due to machining and weld preparation of pipes
- Problems due to specification of improper welding procedures
- Problems due to welder's error
- Problems due to designs that do not give sufficient consideration to constructibility, such as inadequate space and inaccessible welds
- Problems due to fit-up or cold springing

Welding is a complex system, involving metallurgy, chemistry, thermodynamics, and physics. Scoping studies in the areas related to the problems listed above might be necessary to determine the specific welding methods in use and the problems associated with them. Some pressing research needs exist for further development of automatic welding equipment due to radiation exposure limits on qualified welders. Another area of research is the development of repair techniques, including freeze sealing and similar techniques. Efforts to minimize welding in general, and field welding in particular, are needed. Recent developments in continuous pipe bending procedures are steps in this direction.

Thermal versus dynamic pipe behavior is another research area important to the resolution of the pipe cracking problem. LWR piping systems have to be designed for a variety of normal operating and upset or accident conditions. The general conditions to be considered in design include gravity loads, steady-state pressures and temperatures, temperature and pressure transients

due to system transient operating conditions, seismic loads, and postulated accident conditions.

Thermal expansion due to temperature variations can best be accommodated by providing a free-expanding, flexible system. If flexibility is not provided and thermal expansion is restrained, large stresses can be induced in pipes, as well as at pipe supports. Seismic and accident loads, however, are dynamic loads and can only be accommodated by proper supports and appropriate stiffnesses, which pose a design conflict with the thermal requirement for flexibility.²⁴

These conflicting, simultaneous design requirements for thermal and seismic considerations have been met by introducing snubbers. Snubbers are either hydraulic or mechanical devices that permit free displacement of pipe during slow thermal expansion but lock up and provide rigid support under dynamic conditions when velocity or the acceleration of the support point exceeds a predetermined threshold value.

Although snubbers are theoretically adequate for both thermal and seismic requirements, they have caused serious operation and maintenance problems (especially hydraulic snubbers).²⁵ Accessibility has been a problem for snubbers located inside dry wells. Seal failures of hydraulic snubbers have caused the snubbers to be inoperable and have created clean-up problems of spilled contaminated hydraulic fluid. These problems have led to stricter inspection schedules and thus have had a significant effect on plant availability.²⁶ Mechanical snubbers, although free of some of the problems associated with hydraulic snubbers, still present problems with the variability of their response over a long period of time.

Recently, questions have been raised over the need for snubbers and the validity of stringent seismic requirements. Alternate seismic support techniques have been proposed.²⁷ Currently no trouble-free substitute for snubbers has been found.

Research needs exist for experimental and analytical studies of piping systems that give proper consideration to realistic support and boundary conditions under thermal and dynamic loading conditions. Such studies would make

it possible to better evaluate the need for snubbers and rigid supports and might lead to a significant reduction in the number of snubbers and other supports required to safely respond to seismic and accident loadings.

The following research topics are suggested for resolving the problems of thermal fatigue and excessive support stiffnesses:

- Snubber-Support Development. Development of alternate devices or methods to replace or improve snubbers
- Nonlinear Analysis. Analytical studies of piping system behavior under thermal and seismic loadings with proper consideration of support and boundary conditions (see Problem Area 4: Analysis and Design Problems)
- Thermal/Dynamic Pipe Behavior Monitoring. Experimental studies to verify and provide the basis for analytical studies

Problem Area 2: Erosion and Cavitation

Erosion and cavitation are problems that occur in some nuclear piping because of high velocities, wet steam, and water chemistry. Most reported erosion and cavitation problems have occurred at the downstream side of valves or at bends.

The ASME Code does not have any specific requirements that deal with erosion. However, an increase in pipe wall thickness is recommended if erosion is expected.

The literature search did not yield any recently published reports on erosion problems of LWR piping. The present status of research on erosion problems of LWR piping is thus not known. A study to identify the scope of future work on these problems is recommended.

Problem Area 3: General Corrosion

General corrosion, as distinguished from stress corrosion, is a problem that affects mainly carbon steel pipes. However, corrosion of supposedly corrosion-resistant alloy pipes is also encountered where the coolant has highly corrosive properties. Some piping systems, such as PWR steam generator tubing, are

particularly vulnerable to corrosion attack. These corrosion problems can be minimized by control of local chemistry.

Apart from problems associated with the corrosion of PWR steam generator tubing or other localized corrosion, general corrosion does not seem to be a serious safety or operation and maintenance problem. However, one corrosion-related problem seems to have a significant effect on operation and maintenance. This problem is radiation field buildup in BWR stainless steel piping. Recent studies²⁸ indicate that the radiation buildup is caused by the incorporation of radioisotopes into the corrosion-product film formed on the inside surfaces of stainless steel pipes. Although the mechanism of oxide formation on the out-of-core coolant surfaces of BWRs and the incorporation of radioisotopes into this film is not yet thoroughly understood, a crud-growth model has been developed by GE and EPRI. According to this model, the oxide film on reactor coolant surfaces consists of two distinct layers. The inner layer is formed by outward diffusion of metal ions and inward diffusion of oxygen ions. During this process, cobalt 60 is incorporated into the oxide lattice. The outer layer is formed by deposition of hematite particles from the reactor coolant.

Research is needed to understand the mechanism of radiation field buildup. Such research is important to safety because the radiation eventually becomes a problem for maintenance personnel. Research is also needed to understand the mechanism of crud growth and radioisotope incorporation into the crud and to develop effective methods for decontamination of BWR primary systems.

Because of the character of the problem, research efforts in radiation buildup have to be multidisciplinary, including such disciplines as materials science, radiology, mechanical engineering, and operations.²⁹ Research should be directed toward the following:

- Survey of radiation buildup
 - Radiation field mapping of plants
 - Continued study of pipe samples removed from primary systems
- Understanding the mechanism
 - Development of analytical models that are based on the study of the samples

- Development of analytical capability for prediction of future radiation levels
- Development or improvement of decontamination processes
- Evaluation of decontamination processes

Problem Area 4: Analysis and Design Problems

Nuclear power plants differ from conventional nonnuclear power plants in that failures in the critical piping systems of nuclear plants present a risk of potential release of radioactive contaminants into the environment. Hence, the conventional analysis and design methods and commonly used engineering assumptions that may be adequate for the design of nonnuclear power plant piping systems are not necessarily adequate for the design of nuclear power plant piping systems.

The optimum design of LWR piping systems must utilize the best available knowledge and information in the following areas:

- Plant and piping system layout and geometry
- Normal operation and faulted condition loads that the system might be subjected to during the lifetime of the plant; frequency of occurrence of these loads
- Environmental conditions to which the piping system might be subjected
- Expected lifetime of the system
- Analysis procedures
- Material properties and their variability
- Manufacturing and construction procedures and constructibility considerations
- Operation and maintenance considerations
- Reliability of components

Safety factors accounting for the uncertainties and variabilities involved in these areas should be considered and integrated realistically so that the final design presents an acceptable level of risk to public safety without being too costly. Continuous research to improve knowledge and capabilities in the areas listed above is therefore essential to the optimum and safe design of nuclear power plants and the piping systems.

In the previous sections of this chapter, research activities and needs for further research have been discussed for various problems, each of which falls into one of the areas in the above list. A very effective means of putting new knowledge into use is to continuously incorporate newly available data, knowledge, and technology into the design process. Reference 30 gives an extensive review of the significant recent research on piping systems and correlates that research with existing design practices. The size of this document is an indication of the magnitude of the effort involved in a complete review of the design practices and the relevant research. Table 1-21, which was adapted from Reference 30, provides some of the results of this effort by summarizing areas of concern and needed research work.

The two most significant contributors to a successful design are knowledge of the system loads and knowledge of the analysis procedures. System loads and analysis procedures have been studied as Tasks 2 and 3 of this project. These tasks were undertaken essentially as exploratory studies in order to provide bases for future research in relevant topics. Conclusions and recommendations of Tasks 2 and 3 are reported in Parts 2 and 3 of this report. Here, these two areas will be identified as research areas, and the present status and the further needs for research will be briefly discussed.

Piping System Design Loads. Considerable research is currently under way to verify the validity of present design loads, load combinations, and analytical methods. However, much of the data generated by this research is proprietary to the organizations doing the research and is kept confidential. Therefore, a mechanism to collect and evaluate the data generated by various organizations is needed in order to efficiently utilize the research results in improving the present load definitions and design and analysis methods.

Task 2 studies indicate that the present load definitions used in the design of piping systems for weight, normal operating steady-state temperatures and pressures, and sustained mechanical loads and anchor movements are adequate. Research needs exist for evaluating the data from loss-of-coolant-accident experiments and for developing rational and realistic pipe rupture criteria. Also, there is a need for development of rational procedures to better define seismic loads and for development of capabilities for analyzing piping systems for different earthquake motions at different supports. Another area that is

in need of further research is the development of load combination techniques based on probabilistic concepts.

Piping Analysis. Currently, linear-elastic methods are used, in general, in the analysis of LWR piping systems. The use of linear-elastic methods necessarily requires a number of unrealistic assumptions, omissions, and approximations. The implications of these assumptions, omissions, and approximations with regard to accurate prediction of piping response and the design of piping systems have not been well understood.

The Task 3 study conducted as part of this project compared the response of a typical piping system with seismic loads as predicted by three methods of analysis: linear-elastic spectral analysis, linear-elastic time-history analysis, and nonlinear time-history analysis. The objective of the Task 3 study was to assess qualitatively the benefits, if any, that could be realized by using nonlinear analysis methods in the design and licensing of LWR piping systems. The findings of the Task 3 study indicate that the conventional linear-elastic methods of analyses are not necessarily conservative and may not always predict the true behavior of piping systems. Also, current nonlinear analysis capabilities are found to be inadequate and too costly. Therefore, a strong need for development of special-purpose nonlinear piping analysis codes that are in the public domain and are cost-effective is indicated.

Another area in need of research and development is approximate seismic analysis of piping systems. Analytical methods are needed to perform approximate spectrum analysis (similar to response spectrum analysis) of piping systems with consideration of multiple support input motions.

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4. RANKING OF RESEARCH AREAS AND RECOMMENDATIONS FOR RESEARCH PROGRAMS

To properly allocate future funds to programs directed toward resolving the significant problems of LWR piping systems -- that is, to allocate funds in such a way that benefits gained from such research in terms of increased safety and reduced costs are maximized -- a rational approach for assigning priorities to the identified research areas is needed. Such an approach would involve determination of the state of knowledge and the needed research in a specialized area and thus would ideally require a panel of experts in that area. Also, a rational approach would require quantifying concepts such as significance and safety. Because both of these requirements are beyond the scope of Task 1 of this project, the Task 1 findings and recommendations, which were made using the best available information and some subjective judgment, should be considered only as general guidelines to future research plans. Detailed scoping studies on the suggested research topics are recommended for planning future research.

Evaluation and Ranking of Research Areas

In an attempt to rank or assign priorities to the research areas discussed in the previous chapter, an evaluation matrix was developed and is shown in Table 1-22. In this table, the piping problems and research areas that were evaluated and ranked are listed in the first column. Columns 3 through 12 list the factors that were considered for determining the relative significance of the research areas. The priorities assigned to the areas and shown in Column 2 reflect the need for further research as determined through the literature search. If the literature search indicated a large demand for further research in an area and the present level of effort was not considered to be sufficient, then the research area was assigned a priority of one. If the literature search indicated that there was little need for further research or that the present and planned level of effort in the industry was adequate, a lower priority (two or three) was assigned to the research area.

Research programs related to pipe vibration, fatigue studies, and thermal versus dynamic pipe behavior were assigned priorities of one, indicating that the present level of research in these areas is not sufficient given that

the pipe problems related to them form a large part of pipe failures. Research programs on radiation buildup in corrosion products were also assigned a priority of one, indicating that the research efforts in this area were found to be limited considering that this problem has the potential to become significant as the presently operating plants become older. Research programs related to development of piping system design loads and piping analysis were also assigned high priorities because the present level of research in these areas is not adequate in relation to the potential impact of such research on the overall safety and performance of piping systems.

Other research areas were assigned lower priorities. Assignment of lower priorities to these areas does not necessarily indicate that they will have little impact on the safety of piping systems or on the reduction of problems; rather it may indicate that there is a significant amount of recent and ongoing research in these areas and that the results of this research have to be evaluated and implemented before further programs are initiated. In the case of erosion and cavitation problems, a priority of two was assigned -- even though the present status of research in this area is unknown -- because of the relatively low recurrence rates of failures due to erosion and cavitation.

Columns 3, 4, and 5 of the matrix reflect the expected relative time and cost of the research programs. If the expected time or cost is very small, a zero was assigned. If the program is expected to take too long to complete or to cost too much, a number between zero and minus three was assigned.

Columns 6 through 11 reflect the relative benefits that might be gained from the programs. A zero indicates minimal benefit, whereas a three indicates a high level of benefit.

Column 12, generic applicability, is intended to reflect the fact that the results of some programs would not be applicable to all nuclear power plants due to difficulties in retrofitting and, therefore, would benefit only new plants. For example, the results of programs on flaw detection and evaluation can be applicable to all plants, new and old, whereas the results of stress corrosion cracking studies are more likely to affect newly designed plants.

The numbers given in Columns 3 through 12 of each row of the matrix were summed algebraically in Column 13. On the basis of these sums, the research areas were reordered and ranked as shown in Table 1-23. Table 1-24 ranks the research areas with respect to the priorities given in Column 2 of Table 1-22.

Recommendations for Research Programs

Programs to carry out the needed research in the highest priority areas relevant to the resolution of the significant problems of LWR piping systems are briefly described below.

Pipe Vibration

Estimated Cost: \$300,000 to \$500,000

Estimated Time
to Complete: 2 to 3 years

Objective: To reduce or eliminate pipe cracking due to mechanical vibrations caused by pump-pipe resonance, fluid flow fluctuations, or support vibrations.

Suggested Topics:

1. Operational data collection: Instrumentation of various piping systems in operating LWR plants to collect vibration data and to characterize vibration problems.
2. Analytical developments: Development of analytical techniques to analyze piping systems for predicting potential vibration problems.
3. Investigation of possible use of snubbers in vibration control.
4. Development of design guidelines to account for mechanical vibrations.

Thermal versus Dynamic Pipe Behavior

Estimated Cost: \$300,000 to \$500,000

Estimated Time
to Complete: 2 to 3 years

Objective: To resolve the inherent conflict between piping designs for thermal and dynamic loads (e.g., earthquakes).

Suggested Topics:

1. Thermal and dynamic pipe behavior monitoring: Instrumentation of various piping systems in operating LWR plants located in high seismicity areas; collection of data on thermal behavior of piping systems, performance of forced vibration tests, and possible measurement of seismic response in the event of an earthquake.

2. Analytical studies: Evaluation of operational data and development of alternate approaches to the design of piping systems.
3. Snubber-support development: Development of alternate devices or methods to replace or improve snubbers.

Piping Analysis

Estimated Cost: \$300,000 to \$500,000

Estimated Time to Complete: 2 to 3 years

Objective: Development of analytical capabilities to more accurately predict piping system response and safety margins under all service and accident conditions.

- Suggested Topics:
1. Development of efficient nonlinear piping analysis codes.
 2. Development of realistic approximate seismic analysis techniques, taking into account the multiple support seismic input motions.

Fatigue Studies

Estimated Cost: \$500,000 to \$700,000

Estimated Time to Complete: 2 to 3 years

Objective: To study the high- and low-cycle fatigue properties of LWR piping steels and weldments.

- Suggested Topics:
1. High-cycle fatigue studies: Performance of realistic tests of piping systems.
 2. Low-cycle fatigue studies: Performance of realistic tests of piping systems.

Piping System Design Loads

Estimated Cost: \$500,000 to \$700,000

Estimated Time to Complete: 3 to 4 years

Objective: To better define LWR piping system design loads.

- Suggested Topics:
1. Collection and evaluation of recent data relevant to piping loads: Development of more realistic design load definitions.
 2. Pipe rupture criteria revision: Development of more realistic, rational pipe rupture criteria.
 3. Seismic loads: Evaluation of current seismic design loads and assumptions.
 4. Load combinations: Development of rational, probabilistic methods for combining design loads.

Radiation Buildup in Corrosion Products

Estimated Cost: \$300,000 to \$500,000

Estimated Time
to Complete: 2 to 3 years

Objective: To understand the mechanism for crud growth and radioisotope incorporation into this crud in stainless steel LWR piping; to develop effective methods for decontamination of BWR primary systems.

Suggested Topics:

1. Surveys of radiation buildup: Radiation field mapping of operating BWR plant primary piping.
2. Analytical models: Development of analytical models for predicting future radiation levels.
3. Decontamination processes: Development and testing of processes for decontamination of BWR primary systems.

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TABLES

TABLE 1-1
FAILURE DATA SUMMARY - NUMBER OF FAILURES*

(1) Primary Cause of Failure	(2) Up to 600° F	(3) 600° F to 1200° F	(4) Total-All Temperatures	(5) Percent of Total	(6) Occurred Without Leakage	(7) Complete Severance Cases	Location of Failure					
							(8) "Thermowell"	(9) Nozzle	(10) Fitting or Component ^(d)	(11) Pipe	(12) Casting	(13) Weld
1. Design												
a. Mech. fatigue	8	10	18	4.5	2	6	7	2	3	6		
b. Expansion/flexibility	5	5	10	2.5		1				7		3
c. Section transition	1	1	2	0.5						1		1
d. Support		2	2	0.5						2		
e. Design-operation:												
1) Thermal-fatigue	4	28	32	8.0	3	1		1	1	28		2
f. Miscellaneous	5	4	9	2.3			1		1	3	1	3
Sub-total	23	50	73	18.3	5	8	8	3	5	47	1	9
2. Material Selection												
a. Misapplication		9	9	2.3		1			3	1		5
b. Metallurgical												
1) Type 347 stainless steel		59	59	14.8	(a)					59(b)		
2) Graphitization		41	41	10.3	11(a)	1			1	38(c)	1	1
3) "Stress-rupture"		5	5	1.2	1					3		2
4) "885" embrittlement		2	2	0.5		1		1		1		
c. Miscellaneous	1	3	4	1.0						4		
Sub-total	1	119	120	30.1	13+	3		1	4	105	1	8
3. Manufacturing												
a. Wrong material supplied		5	5	1.2					1	3		1
b. Base material defects	17	14	31	7.8	1	1			2	16	13	
c. Weld defects		1	1	0.3								1
d. Associated with welding	1	3	4	1.0						1	3	
e. Material not to specifications		1	1	0.3						1		
Sub-total	18	24	42	10.6	1	1			3	21	16	2
4. Fabrication-Erection												
a. Weld defects	19	8	27	6.8	1	1						27
b. Associated with welding	4	6	10	2.5				2	1	4	1	2
c. Wrong material used		4	4	1.0		2				2		1
d. Miscellaneous	2	7	9	2.3	1		1		1	7		1
Sub-total	25	25	50	12.6	2	3	1	2	2	13	1	31
5. Operation-Service												
a. Mis-operation	4	5	9	2.3	1	2			4	4		1
b. Thermal shock and shock	1	5	6	1.5	1	1			2	2	2	
c. Corrosion and/or erosion	82	12	94	23.4	2				5	80	7	2
Sub-total	87	22	109	27.2	4	3			11	86	9	3
6. Unknown	1	4	5	1.2		1		1		3		1
Totals	155	244	399	100.0	25+	19	9	7	25	276	28	54

Footnotes:

(a) Some of these occurred without leakage; precise data was not supplied.

(b) Most of these occurred in the weld heat-affected zone, some were in components elbows; specific data distribution was not supplied.

(c) Most of these were located in the weld heat-affected zone.

(d) Elbow, tee, wye, etc.

*Source: Reference 1 (Table IV)

TABLE 1-2
CLASSIFICATION OF SIGNIFICANT CAUSES OF
FAILURE BY OPERATING TEMPERATURE*

Cause	Number of Failures, by Temperature	
	600°F or Less	All Temperatures
Design:		
Mechanical fatigue	8	18
Thermal fatigue	<u>4</u>	<u>32</u>
	12	50
Materials:		
Graphitization	0	41
Type-347 stainless steel	<u>0</u>	<u>59</u>
	0	100
Manufacturing: Base metal defects	17	31
Fabrication-erection: Weld defects	19	27
Service: Corrosion-erosion	<u>82</u>	<u>94</u>
Total	130	302

*Based on information contained in Reference 1

TABLE 1-3
CLASSIFICATION OF FAILURES BY FRACTURE OR FAILURE MODE*

Mode	Number of Failures	Percent
Cleavage	7	1.7
Fibrous	7	1.7
Creep rupture:		
General	15	3.8
Graphitization	41	10.2
Type-347 stainless steel	59	14.8
Fatigue:		
High cycle - elastic	20	5.1
Low cycle - plastic	93	23.3
Not clear (either of above)	3	0.7
Corrosion		
Austenitic:		
General	5	1.2
Transcrystalline	3	0.7
Intercrystalline	2	0.5
Ferritic:		
General	18	4.6
Transcrystalline	--	--
Intercrystalline	2	0.5
Erosion	82	20.4
No fracture, defect leaked	24	6.2
Insufficient data to attempt classification	<u>18</u>	<u>4.6</u>
Total	399	100.0

*Adapted from Reference 1 (Table VI)

TABLE 1-4
CAUSES OF SERVICE FAILURES IN
CONVENTIONAL AND NUCLEAR PLANTS*†

Cause	Conventional Plants		Nuclear Plants	
	Number of Failures	Percent	Number of Failures	Percent
Cracks	118	89.3	12	70.5
Corrosion (including stress corrosion, assisted fatigue, and wastage)	2	1.5	3	17.7
Maloperation (including water shortage and control malfunction)	8	6.1	0	0.0
Preexisting from manufacture (material or fabrication causes)	3	2.3	1	5.9
Creep failure	1	0.8	0	0.0
Miscellaneous	<u>0</u>	<u>0.0</u>	<u>1</u>	<u>5.9</u>
Total	132	100.0	17	100.0

*Based on information contained in Reference 2.

†Majority of data are for pressure vessels.

TABLE 1-5
CAUSES OF CRACKS LEADING TO SERVICE FAILURES
IN CONVENTIONAL AND NUCLEAR PLANTS*†

Cause	Conventional Plants			Nuclear Plants		
	Number of Failures Due to Cracks	Percent of Failures Due to Cracks	Percent of Total Failures	Number of Failures Due to Cracks	Percent of Failures Due to Cracks	Percent of Total Failures
Fatigue (including mechanical and thermal fatigue)	47	40.0	35.6	5	41.7	29.4
Corrosion (including stress corrosion, assisted fatigue, and wastage)	24	20.3	18.2	4	33.3	23.4
Preexisting from manufacture	10	8.4	7.6	2	16.7	11.8
Not ascertained	35	29.6	26.5	0	0.0	0.0
Miscellaneous	<u>2</u>	<u>1.7</u>	<u>1.4</u>	<u>1</u>	<u>8.3</u>	<u>5.9</u>
Total	118	100.0	89.3	12	100.0	70.5

*Based on information contained in Reference 2.

†Majority of data are for pressure vessels.

TABLE 1-6
COMPARISON OF CAUSES OF FAILURES IN CONVENTIONAL PLANTS
BUILT TO CLASS 1 STANDARDS AND IN NUCLEAR PLANTS*†

Cause	Percent of Failures in Conventional Class 1 Plants	Percent of Failures in Nuclear Plants
Fatigue	25.8	29.4
Preexisting from manufacture	19.4	17.7
Corrosion (including stress corrosion and assisted fatigue)	30.6	41.1
Not ascertained or miscellaneous	<u>24.2</u>	<u>11.8</u>
Total	100.0	100.0

*Based on information contained in Reference 2.

†Majority of data are for pressure vessels.

TABLE 1-7
CAUSES OF SERVICE FAILURES (CONVENTIONAL PLANTS)*

Cause	Number of Failures	Percent
Cracks	117	84.2
Corrosion (including stress corrosion, assisted fatigue, and wastage)	--	--
Maloperation (including water storage and control malfunction)	10	7.2
Preexisting from manufacture (material or fabrication causes)	11	7.9
Creep failure	<u>1</u>	<u>0.7</u>
Total	139	100.0

*Based on information contained in Reference 3

TABLE 1-8
CAUSES OF CRACKS LEADING TO SERVICE FAILURES
(CONVENTIONAL PLANTS)*

Cause	Number of Failures Due to Cracks	Percent of Failures Due to Cracks	Percent of Total Failures
Fatigue (including mechanical and thermal fatigue)	20	17.0	14.5
Corrosion	4	3.4	2.9
Preexisting from manufacture	48	41.0	34.6
Not ascertained	18	15.4	12.9
Miscellaneous (including maloperation and creep)	<u>27</u>	<u>23.2</u>	<u>19.3</u>
Total	117	100.0	84.2

*Based on information contained in Reference 3

TABLE 1-9
DISTRIBUTION OF REPORTABLE OCCURRENCES
ACCORDING TO PLANT TYPE*

Plant Type	Number of Occurrences	Percent
<u>BWRs</u>		
GE-1	7	5.6
GE-2	3	2.4
GE-3	38	30.0
GE-4	<u>9</u>	<u>7.2</u>
	57	45.2
<u>PWRs</u>		
W-2	15	11.9
W-3	9	7.2
W-4	13	10.3
B&W	16	12.7
CE	<u>16</u>	<u>12.7</u>
	<u>69</u>	<u>54.8</u>
Total	126	100.0

*Based on information contained in Reference 5
 (Table 9)

TABLE 1-10
DISTRIBUTION OF REPORTABLE OCCURRENCES
ACCORDING TO PIPE SIZE*

Diameter (in.)	Number of Occurrences	Percent
$d \leq 1$	75	59.5
$1 < d \leq 2$	5	4.0
$2 < d \leq 4$	9	7.1
$4 < d \leq 6$	2	1.6
$6 < d \leq 8$	1	.8
$8 < d \leq 10$	7	5.6
$10 < d \leq 20$	5	3.9
$20 < d$	2	1.6
Unknown	<u>20</u>	<u>15.9</u>
Total	126	100.0

*Based on information contained in Reference 5
 (Table 9)

TABLE 1-11
DISTRIBUTION OF REPORTABLE OCCURRENCES
ACCORDING TO FAILURE LOCATION*

Location	Number of Occurrences	Percent
Welds:		
Socket	31	24.6
Other	31	24.6
Pipe	22	17.5
Joints	22	17.5
Other	4	3.2
Unknown	<u>16</u>	<u>12.6</u>
Total	126	100.0

*Based on information contained in Reference 5 (Table 9)

TABLE 1-12
DISTRIBUTION OF REPORTABLE OCCURRENCES
ACCORDING TO CAUSE*

Cause	Number of Occurrences	Percent
Fatigue	59	46.8
Construction error	27	21.4
Erosion	15	11.9
Water hammer	6	4.8
Design error	5	4.0
Brittle failure	3	2.4
Corrosion	3	2.4
Missile	1	.8
Unknown	<u>7</u>	<u>5.5</u>
Total	126	100.0

*Based on information contained in Reference 5
 (Table 9)

TABLE 1-13
CLASSIFICATION OF SOME CAUSES OF REPORTABLE
OCCURRENCES BY PIPE DIAMETER*

Cause	Number of Occurrences, by Pipe Diameter (in.)		
	≤1	>1	Unknown
Fatigue	45	4	10
Construction	15	6	6
Erosion	0	7	8
Water hammer	1	5	0
Brittle failure	1	1	1
Design	0	5	0

*Based on information contained in Reference 5
 (Table 10)

TABLE 1-14
RESULTS OF SURVEY OF REPORTABLE OCCURRENCES
INVOLVING PIPE SYSTEMS*

Cause Classification	Number of Reportable Occurrences, by Component					Total Reportable Occurrences
	Pipes and Fittings	Pumps	Valves	Valve Operators	Pipe Supports	
Component failure	401	334	1,285	660	191	2,871
Defective procedures	31	30	106	17	16	200
Design/fabrication errors	171	92	161	79	106	609
Personnel errors	103	92	359	85	51	690
External causes	12	6	6	4	3	31
Other causes	<u>52</u>	<u>38</u>	<u>109</u>	<u>30</u>	<u>25</u>	<u>254</u>
Total	770	592	2,026	875	392	4,655

*BWR and PWR data combined

TABLE 1-15
DESCRIPTION OF PIPE FAILURE

Description	Number of Reportable Occurrences	Percent
Leak from through-wall crack	142	40.1
Leak at the nipples and at threaded or bolted connections	50	14.1
Partial crack without leak	42	11.9
Pinhole leak	25	7.1
Leak due to corrosion, erosion, or water hammer	23	6.5
Complete rupture	8	2.2
Other	64	18.1
Total	354	100.0

TABLE 1-16
LOCATIONS OF PIPE FAILURE

Location	Number of Reportable Occurrences	Percent
Weld	112	31.6
Base metal (pipe)	88	24.9
Fittings	54	15.3
Weld-heat-affected zone	34	9.6
Other	<u>66</u>	<u>18.6</u>
Total	354	100.0

TABLE 1-17
CAUSES OF PIPE FAILURE

a. Summary of Causes

Cause	Number of Reportable Occurrences	Percent
Design errors	41	11.6
Manufacturing and construction errors	67	18.9
Material-related causes	31	8.8
Environmental and operational conditions	148	41.8
Personnel error	24	6.8
Unknown causes	<u>89</u>	<u>25.1</u>
Total	400*	113.0*

*The reason for the sum being larger than the total number of reportable occurrences (400 versus 354) is that in some cases multiple causes have been cited for a single occurrence.

b. Detail of Causes: Design Errors

Cause	Number of Reportable Occurrences	Percent
Inadequate support	15	36.6
Materials related	3	7.3
Load related	1	2.4
Analysis related	1	2.4
Design drawings	1	2.4
Other design errors	<u>20</u>	<u>48.9</u>
Total	41	100.0

c. Detail of Causes: Manufacturing and Construction Errors

Cause	Number of Reportable Occurrences	Percent
Defective welds	30	44.8
Installation errors	21	31.3
Machining errors	1	1.5
Other	<u>15</u>	<u>22.4</u>
Total	67	100.0

d. Detail of Causes: Material-Related Causes

Cause	Number of Reportable Occurrences	Percent
Stress corrosion cracking	17	54.8
Material flaws	6	19.4
Gasket failure	4	12.9
Other	<u>4</u>	<u>12.9</u>
Total	31	100.0

e. Detail of Causes: Environmental and Operational Conditions

Cause	Number of Reportable Occurrences	Percent
Mechanical vibrations	79	53.3
Erosion	17	11.5
Corrosion	13	8.8
Thermal cycling	12	8.1
Cavitation	5	3.4
Water hammer	2	1.4
Other	<u>20</u>	<u>13.5</u>
Total	148	100.0

TABLE 1-18
EFFECTS OF PIPE FAILURE ON PLANT OPERATION

Effect	Number of Reportable Occurrences	Percent
Plant was not operational at the time	146	41.2
Effect on plant operation not reported	102	28.8
No effect on plant operation	86	24.3
Manual shutdown	15	4.2
Automatic shutdown	1	0.3
Other (partial power reduction, etc.)	<u>4</u>	<u>1.2</u>
Total	354	100.0

TABLE 1-19
SAFETY CONSEQUENCES OF PIPE FAILURE

Consequence	Number of Reportable Occurrences	Percent of Total Reportable Occurrences
Radiation release reported	23	6.5
Personnel exposure/injury reported	9	2.5

TABLE 1-20
SUBSYSTEMS WITH SIGNIFICANT NUMBERS OF FAILURES

Subsystem	Number of Reportable Occurrences	Percent
Chemical, volume control, and liquid poison system	54	15.3
Emergency core cooling system and controls	35	9.9
Coolant recirculation system and controls	32	9.1
Reactor coolant clean-up system and controls	31	8.8
Residual heat removal system and controls	22	6.2
Station service water system and controls	15	4.2
Liquid radioactive waste management systems	11	3.1
Feedwater system and controls	10	2.8
Reactor core isolation coolant system and controls	10	2.8
Gas radioactive waste management systems	4	1.1
Other	<u>130</u>	<u>36.7</u>
Total	354	100.0

TABLE 1-21
SUMMARY OF AREAS OF CONCERN AND BRIEF STATEMENTS
OF NEEDED RESEARCH WORK*

Area Number	Priority	Area of Concern/Needed Research Work
1	1	Dynamic analysis. Conduct in-depth review of present design practices and develop improved, standardized methods.
2	2	End effects on curved pipe or elbows. Utilize analytical methods to perform a parametric study. Develop simplified equations and/or charts for routine design application.
3	3	Thermal stresses and thermal fatigue cracking at branch connections, nozzles in pressure vessels, or tees. Review cases of thermal stress fatigue. Review thermal sleeve design practice and attempt to identify where thermal sleeves are needed and the best way to design them.
4	2	ANSI piping product standards. Review ANSI B16.9 and develop appropriate revisions commensurate with the reliability needs of Class I piping.
5	2	Continuous recording of piping system failures that contains data relevant to the failure origin, cause, etc., that allows a determination of how near the failure was to being a full area opening. A questionnaire should be sent to reactor operators requesting the desired information.
6	4	Buckling of piping products with $D/t > 40$. Run tests on piping products with $D/t \gg 40$, e.g., $D/t = 80$.
7	1	Code rules for emergency or faulted conditions. Code rules need to be reexamined, but Areas Number 10 and 12 are needed before adequate evaluation can be completed.
8	2	Welded joints. A need exists for a summary report on the characteristics of welded joints as used in nuclear power plant piping systems. This report should correlate code rules on fabrication and inspection with the code design rules and evaluate the reliability of welded joints.

*Adapted from Reference 30 (Table 3)

TABLE 1-21 (Continued)

Area Number	Priority	Area of Concern/Needed Research Work
9	3	Girth butt welded joints with radial weld shrinkage -- no test data. Tests need to be conducted.
10	3	Girth butt welded transition joints -- no test data. Tests need to be conducted.
11	1	Branch connections (nozzles) -- combined moment and pressure capacity. Additional tests need to be conducted.
12	2	Branch connections (nozzles) are closely spaced -- lack of design information. Analytical methods should be used to conduct parametric studies, and those studies should be reduced to simple equations or charts for routine design use.
13	1	B16.9 Tees and reducers -- lack of information on combined moment and pressure loading. Tests need to be conducted.
14	4	Flanged joints -- leakage. Test data need to be evaluated and additional tests conducted.
15	3	Attachments -- fatigue characteristics. Tests need to be conducted.
16	1	Effects of dynamic loadings, water hammer, cavitation, etc., on nonleaking flaws (corrosion or stress corrosion cracks). Needs analytical and experimental study.
17	2	Attachments and branch connections can apply both a secondary circumferential stress and longitudinal stress to the pipe. Their effect on existing local flaws is unknown. Tests need to be conducted.
18	2	Nozzles, elbows, girth welds, and seam welds -- very little experimental work found. Tests need to be conducted.
19	2	Records of "incidents." To maximize the usefulness of the Experience Approach, a more complete and systematic abstract of "incidents" should be established and maintained. The Experience Approach to evaluating the reliability of piping systems should become more useful as experience accumulates and better abstracts of incidents become available.

TABLE 1-21 (Continued)

Area Number	Priority	Area of Concern/Needed Research Work
20	2	Elements Reliability Approach -- complete applications. At present, GEAP-10452 constitutes a good foundation. It needs to be simplified (if possible), applied to different basic assumptions and to various safety-related piping systems.

TABLE 1-22
EVALUATION OF RESEARCH AREAS

Research Area (1)	Priority (2)	R/D Time Requirements (3)	Cost of R/D (4)	Cost of Implementing Results (5)	Positive Impact On						Generic Applicability (12)	Sums (13)
					Safety (6)	Construction Costs (7)	O/M Costs (8)	Plant Availability (9)	Safety Margins (10)	Licensing Time (11)		
Pipe cracking:												
Flaw detection and evaluation	3	-2	-2	-1	3	0	3	3	3	1	3	11
Fracture mechanics	2	-2	-2	0	2	0	3	3	3	0	3	10
Fatigue studies	1	-2	-2	0	2	0	3	3	3	0	2	9
Stress corrosion cracking	3	-3	-2	-2	2	0	2	3	3	1	1	5
Postulated pipe rupture	2	-1	-2	0	2	3	2	1	3	2	2	12
Pipe vibration	1	-1	-1	0	2	1	3	3	2	1	2	12
Welding technology	2	-2	-2	0	2	1	3	3	2	0	2	9
Thermal versus dynamic pipe behavior	1	-1	-1	-1	1	1	3	3	2	1	2	10
Erosion and cavitation	2	0	0	-1	2	0	2	2	1	0	2	8
General corrosion: Radiation buildup in corrosion products	1	-2	-1	-1	2	1	3	3	0	0	3	8
Design/analysis problems:												
Piping system design loads	1	-3	-2	0	2	3	1	1	3	2	2	9
Piping analysis	1	-2	-1	-1	2	1	2	2	3	2	2	10

TABLE 1-23
RANKING OF RESEARCH AREAS WITHOUT
CONSIDERATION OF PRIORITY

Sum*	Priority†	Research Area
12	1	Pipe vibration
12	2	Postulated pipe rupture
11	3	Flaw detection and evaluation
10	1	Thermal versus dynamic pipe behavior
10	1	Piping analysis
10	2	Fracture mechanics
9	1	Fatigue studies
9	1	Piping system design loads
9	2	Welding technology
8	1	Radiation buildup in corrosion products
8	2	Erosion and cavitation
5	3	Stress corrosion cracking

*From Column 13, Table 1-22

†From Column 2, Table 1-22

TABLE 1-24
RANKING OF RESEARCH AREAS WITH RESPECT TO PRIORITY

Priority	Sum [†]	Research Area
1	12	Pipe vibration
1	10	Thermal versus dynamic pipe behavior
1	10	Piping analysis
1	9	Fatigue studies
1	9	Piping system design loads
1	8	Radiation buildup in corrosion products
2	12	Postulated pipe rupture
2	10	Fracture mechanics
2	9	Welding Technology
2	8	Erosion and cavitation
3	11	Flaw detection evaluation
3	5	Stress corrosion cracking

*From Column 13, Table 1-22

[†]From Column 2, Table 1-22

PART 2

Task 2: Piping System Loads

1. OBJECTIVE AND SCOPE

The objective of Task 2 was to describe piping system loads for LWR nuclear power plants. Piping systems in LWR nuclear power plants are defined to be the pressure-retaining components, consisting of straight or curved pipe and pipe fittings (e.g., elbows, tees, and reducers), used to transport fluids. Piping system loads include the following:

- Weight
- Normal operating steady-state pressures and temperatures
- Pressure and temperature transients (includes water hammer)
- Loss-of-coolant-accident load and other pipe failure loads
- Seismic loads
- Anchor movements and other mechanical loads

The description of piping loads includes information up to, but not including, the static or dynamic analysis procedures used to analyze piping systems for the effects of loads. This boundary is somewhat fuzzy for certain loading conditions. For example, the size (hence, load) of a pipe rupture may depend on how much the piping system displaces and rotates, which is part of the response analysis domain. However, in order to be complete, the description of piping system loads includes discussions of pertinent piping analysis subjects, where appropriate.

The Task 2 studies are organized into the following chapters:

- Nuclear Power Plant Piping Systems (Chapter 2)
- Load-Causing Events and Phenomena (Chapter 3)
- Methods and Procedures for Definition of Loads (Chapter 4)
- Load Combinations (Chapter 5)
- Methodologies and Data Bases for Current Load Definitions and Combinations (Chapter 6)
- Development of New Piping Load Data (Chapter 7)

Feasibility and Benefits of Developing New Load Definitions and Combination Procedures (Chapter 8)

Recommendations for Future Research (Chapter 9)

Tables and figures for Task 2 follow the text.

2. NUCLEAR POWER PLANT PIPING SYSTEMS

The purpose of Chapter 2 is to summarize the types of piping systems found in LWR nuclear power plants. These systems are referred to in subsequent chapters, which discuss piping loads.

In the first two sections of this chapter, the piping systems associated with the two types of LWR facilities -- the boiling water reactor (BWR) and the pressurized water reactor (PWR) -- are discussed. Although all LWR facilities have common features, it is not possible to list generically all piping systems for both BWR and PWR facilities; thus, the most important piping systems for BWR and for PWR plants are listed and described in separate sections. The systems are described according to the nuclear steam supply system (NSSS) manufacturer's designated functions.*

Nuclear power plant piping systems may also be listed according to the NRC's "General Design Criteria for Nuclear Power Plants," Chapter 10, Code of Federal Regulations, Part 50, Appendix A (referred to herein as the NRC General Design Criteria). However, the systems listed in the NRC General Design Criteria do not always use the same name as that specified by an NSSS manufacturer. For example, several piping systems defined by one NSSS manufacturer may be specified to meet the requirements for a single system defined by the NRC General Design Criteria. The NRC designations for LWR piping systems are given in Regulatory Guide (RG) 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants." The third section of this chapter lists LWR piping systems as categorized in RG 1.70.

In the last section of this chapter, the piping systems obtained from recent safety analysis reports for two PWR plants and two BWR plants are identified. These piping systems are listed by the names given in the safety analysis reports but are organized according to the format of RG 1.70.

*Note that the individual pieces of equipment (e.g., piping) are often designated as components for more than one system and that therefore some components serve more than one function.

BWR Plants

The BWR system marketed today in the United States is a direct-cycle steam-generating system. Water is boiled by the fissioning core in the reactor vessel, and the wet steam passes up through moisture separators and steam dryers, leaving the reactor vessel with 0.30% moisture at 970 to 1,000 psia where it goes to the turbine. After passing through a suitable heat cycle, the steam is condensed. From the condenser, the condensate is pumped back through the full-flow polishing demineralizers and feedwater heaters and then back to the reactor.

The industrial BWR was developed and is marketed by General Electric Company (GE), which is today the only NSSS manufacturer for BWRs in the United States. Balance-of-plant (BOP) piping is supplied by a number of architectural/engineering (A/E) firms. Of the approximately 60 BWR plants existing or under construction in the United States, Bechtel Corporation, with 18 plants, is the leading A/E firm. GE has produced a number of different designs, from the BWR/1 (introduced in 1955) through the most current model, the BWR/6 (introduced in 1972). Figure 2-1 shows a schematic diagram of the BWR primary loop and turbine power train. The most significant systems of the BWR/6 (similar to other BWR systems) that include piping elements are given below, along with a brief description of their design function.

Reactor Water Recirculation System. The function of the reactor water recirculation system is to circulate coolant through the reactor core. The system consists of two loops external to the reactor vessel, each containing a pump with a directly coupled water-cooled (air-water) motor, a flow-control valve, and two shutoff valves.

Main Steam and Feedwater System. The limits of the BWR primary system (i.e., reactor coolant pressure boundary) are defined by the Code of Federal Regulations, Chapter 10, Part 50, as the outermost of the two containment isolation valves on the main steam lines, which connect to the turbine, and on the reactor feedwater lines, which connect to the reactor feedwater pumps. Steam exits from the vessel several feet below the reactor vessel flange through four nozzles. Carbon steel lines welded to the vessel nozzles run parallel to the vertical axis of the vessel, downward to an elevation where they then run horizontally and finally emerge from the containment.

Control Rod Drive System. Positive core reactivity control is maintained by the use of movable control rods interspersed throughout the core. The rods are moved vertically by hydraulically actuated, locking-piston-type drive mechanisms. The control rod drive system consists of a number of locking-piston control rod drive mechanisms, a hydraulic control unit for each drive mechanism, a hydraulic power supply for the entire system, and instrumentation and controls with necessary interconnections.

Control Rod Drive Hydraulic Supply System. The control rod drive hydraulic system, located close to, but outside of, the primary containment, supplies pressurized demineralized water to the control rod drive system to provide hydraulic operating pressure and cooling water for the drive mechanisms.

Reactor Water Cleanup System. The purpose of the reactor water cleanup system is to maintain high-quality reactor water by removing fission products, corrosion products, and other soluble and insoluble impurities. In addition, the system provides a means for removing water from the primary system during periods of increased water volume.

Fuel and Containment Pools Cooling and Cleanup System. The fuel and containment pools cooling and cleanup system removes the spent fuel heat, as well as dry-well heat transferred to the upper containment pool.

Closed Cooling Water System for Reactor Service. The closed cooling water system consists of a separate, forced-circulation loop that uses water piped from the site service water source to provide a heat sink for selected nuclear system equipment. Its purpose is to provide a second barrier between the radioactive primary systems and the final heat sink.

Emergency Equipment Cooling System. The emergency equipment cooling system services certain equipment required for normal and emergency shutdown of the plant. The system provides cooling water for the residual heat removal system pump motor and pump seal cooler and the high-pressure and low-pressure core spray system pump motors and pump seal coolers. Upon loss of normal ventilation, the emergency equipment cooling system provides cooling water for the high-pressure and low-pressure core spray systems, the residual heat removal system, and the reactor core isolation cooling system.

Standby Liquid Control System. The standby liquid control system is a redundant control system capable of shutting the reactor down from rated power operation to the cold condition in the postulated situation of control rod failure.

Reactor Core Isolation Cooling System (RCIC). The reactor core isolation cooling system maintains sufficient water in the reactor pressure vessel to cool the core and maintain the nuclear boiler in standby condition in the event the vessel becomes isolated from the turbine steam condenser and from feedwater makeup flow.

Emergency Core Cooling System (ECCS). The emergency core cooling system comprises the low-pressure coolant injection function of the residual heat removal system, the high- and low-pressure core spray systems, and the automatic depressurization systems. The ECCS is designed to protect the reactor core against fuel-cladding damage for the entire range of pipe break accidents.

Subsystems of the ECCS are:

High-Pressure Core Spray System: The purpose of the high-pressure core spray system is to depressurize the nuclear boiler system and to provide makeup water in the event of a loss of reactor coolant inventory, as well as to prevent damage to fuel cladding in the event the core becomes uncovered.

Low-Pressure Core Spray System: The function of the low-pressure core spray system is to prevent fuel-cladding damage in the event the core is uncovered due to loss of coolant.

Automatic Depressurization System (ADS): Blowdown, through selected safety-relief valves, in conjunction with the operation of the low-pressure coolant injection mode of the residual heat removal system and/or low-pressure core spray system, functions as an alternative to the operation of the high-pressure core spray system for protection of fuel against cladding damage due to postulated loss-of-coolant accidents (LOCA).

Residual Heat Removal System (RHR). The residual heat removal system removes residual heat generated by the core under normal and abnormal shutdown conditions. The objectives of the system are:

- To restore and maintain water level in the reactor vessel after a LOCA
- To limit suppression pool water temperature
- To remove heat from the nuclear boiler system while the reactor is shut down
- To condense reactor steam so that decay and residual heat may be removed if the main condenser is unavailable
- To supplement the fuel and containment pools cooling and cleanup system capacity when necessary

The subsystems of the RHR are:

Low-Pressure Coolant Injection (LPCI): The low-pressure coolant injection function (also part of ECCS) of the RHR system, in conjunction with the low-pressure core spray system, the high-pressure core spray system, and/or the ADS, restores and maintains the desired water level in the reactor vessel after a LOCA.

Suppression Pool Cooling: The suppression pool cooling function of the RHR system ensures that the temperature of the suppression pool, immediately after blowdown and when the reactor vessel pressure is greater than 135 psig, does not exceed a predetermined limit.

Reactor Steam Condensing (Hot Standby): During nuclear boiler system isolation, and in conjunction with the operation of the reactor core isolation cooling system and steam blowdown to the suppression pool, steam at reduced pressure and temperature is directed from the main steam lines to the RHR system heat exchanger.

Nuclear Boiler Shutdown Cooling: The shutdown cooling function of the residual heat removal system removes residual heat from the nuclear boiler system after normal reactor shutdown.

Containment Spray Cooling: The containment spray cooling function of the RHR system condenses and removes heat of any steam that bypasses the dry well and prevents over-pressurization of the containment.

PWR Plants

A PWR nuclear power plant contains a closed loop of pressurized water that removes the heat energy from the core and transfers the energy to a second water system, where steam is generated. The steam, in turn, drives a turbine generator set that produces electric power. The reactor coolant is demineralized water and at normal operating conditions is about 2,300 psia and

600°F. The steam generated in the second system is in the range of 850 to 1,000 psia and usually has a moisture content of 0.20% to 0.25%.

PWRs are marketed in the United States by Westinghouse Electric, Incorporated (Westinghouse), Combustion Engineering, Inc. (C-E), and the Babcock & Wilcox Company (B&W). Of the approximately 130 PWR plants existing or under construction in the United States, approximately 70 are of Westinghouse design, 30 are of C-E design, and 30 are of B&W design. The A/E providing the largest amount of BOP piping is Bechtel Corporation, followed by Stone and Webster Engineering and Sargent and Lundy Engineers. Although there are three manufacturers of the NSSS, many of the associated piping systems have a common function and classification. Figure 2-2 shows a schematic diagram of the PWR primary loop and turbine power train. The most significant systems for a current Westinghouse PWR that contain piping elements are given below, along with a description of their design function.

Reactor Coolant System (RCS). The reactor coolant system consists of similar heat transfer loops connected in parallel to the reactor pressure vessel. The number of loops may vary between two and four depending on the plant size (and the NSSS manufacturer). Each loop contains hot-leg and cold-leg piping, at least one circulation pump, and one steam generator. The RCS transfers heat from the reactor pressure vessel via the hot leg to the steam generator, where heat energy is used to create saturated steam. The circulation pump on the cold leg(s) returns reactor coolant from the steam generator to the reactor vessel. A single pressurizer is connected to the RCS on one of the heat transfer loops and is used to maintain the RCS pressure during steady-state operation and to limit pressure changes during transient conditions.

Chemical and Volume Control System (CVCS). The chemical and volume control system performs the following functions:

- Fills the RCS
- Provides a source of high-pressure water for pressuring the RCS when the RCS is cold
- Maintains the water level in the pressurizer when the RCS is hot
- Reduces the concentration of corrosion and fission products in the reactor coolant

- Adjusts the boric acid concentration of the reactor coolant
- Provides high-pressure seal water for the reactor coolant pump seals

Residual Heat Removal System (RHRS). The primary function of the residual heat removal system is to transfer heat energy from the core and RCS during plant shutdown and refueling. The system is also employed in conjunction with the safety injection system for emergency core cooling during pipe rupture accident conditions.

Safety Injection System (SIS). The primary function of the safety injection system is to supply borated water to the RCS to limit fuel-rod-cladding damage during a postulated LOCA. A secondary SIS function is to provide a means of introducing a spray of borated water into the containment as an additional dynamic heat sink.

Waste Processing System. The waste processing system handles liquid and gaseous plant effluents during power operation and plant shutdowns. The system is composed of two sections, one for liquid handling and one for gas handling. The liquid-handling section processes both potentially tritiated and nontritiated effluents. The gas-handling section is capable of retaining radioactive gases within the gas decay tanks at slightly above atmospheric pressure for the life of the plant. The gas-handling section is designed to receive a purge of a gas mixture of hydrogen and fission products from the volume control tank.

Containment Spray System. The containment spray system is designed to do the following tasks, in the event of a LOCA:

- Limit the pressure in the containment atmosphere to below the containment design pressure
- Remove sufficient iodine from the containment to limit the off-site and site boundary doses to values below those set by the Code of Federal Regulations, Chapter 10, Part 100

Steam, Condensate, and Feedwater Systems. During operation, the steam generators deliver saturated steam through the main steam lines to the high-

pressure turbine. The main steam lines are crosstied near the main turbine to ensure minimal differences between pressures in the steam generators and to ensure balanced steam and uniform heat removal by the RCS. A portion of the steam in the high-pressure turbine is exhausted to moisture separator heaters, where it is dried, superheated, and supplied to low-pressure turbines. Exhausted steam from the turbines is condensed and pumped through the feed-water heating system, which removes air and prepares the condensate for return to the steam generators.

Auxiliary Feedwater System. The function of the auxiliary feedwater system is to provide adequate cooling water to the steam generators in the event of a unit trip coupled with a loss of off-site power.

Circulating Water System. The circulating water system serves as the major heat sink for the plant by providing large amounts of screened cooling water for the main condensers, the service water system, the fire protection system, and the screen wash system.

Service Water System. The service water system supplies all the equipment cooling water for the plant, including equipment cooling water to meet the emergency shutdown requirements of the plant.

Component Cooling System. The component cooling system is designed to remove heat from heat exchangers, pumps, and waste disposal systems. Component cooling system water picks up heat from the various components and transfers this heat to the service water system. The component cooling loop thus serves as an intermediate system between the RCS and the service water system.

Spent Fuel Pit Cooling System. The spent fuel pit cooling system is designed to remove the spent fuel pit heat generated by the stored spent fuel elements.

Systems Designated by RG 1.70

In the identification of piping systems (i.e., systems with piping elements) for LWRs, it is important to recognize that more than one system name may be associated with a particular piping system. This is the case for systems with multiple designated functions; e.g., in a PWR, the safety injection system

(the NSSS name), is also the emergency core cooling system (the NRC name for a required engineered safety feature). To clarify the specification and identification of LWR piping systems, names as given by RG 1.70 are reviewed. That document delineates the format and names used in LWR preliminary safety analysis reports (PSAR) and final safety analysis reports (FSAR) and in the standard safety analysis reports supplied by the NSSS and A/E firms.¹ RG 1.70 is intended as a document generically applicable to all types of LWRs. Certain elements are, however, necessarily BWR or PWR specific. The chapter and sections in RG 1.70 pertinent to the identification of piping systems in LWRs have been summarized and are given in Table 2-1. Systems that are PWR and BWR specific are so designated. From the table it is seen that there are six categories of systems (corresponding to the six chapters of RG 1.70) that contain piping. The following comments apply to the six categories.

Reactor: BWR plants contain piping systems related to control rod hydraulics and reactor monitoring.

Reactor Coolant Systems and Connected Systems: The reactor coolant system includes those systems and components that contain or transport fluids coming from, or going to, the reactor core.

Engineered Safety Features (ESF): The engineered safety features of LWRs are those systems provided to mitigate the consequences of postulated serious accidents, in spite of those accidents' being unlikely. The features can be divided into four categories: containment systems, emergency core cooling systems, habitability systems, and fission product removal and control systems.

Auxiliary Systems: Auxiliary systems are systems used for normal plant operation (e.g., water systems) or in support of RCS or ESF systems.

Steam and Power Conversion System: Steam and power conversion systems are associated with the transport and conversion of steam heat energy into mechanical and electrical energy.

Radioactive Waste Management: Radioactive waste management systems are those systems used in the transport and processing of radioactive waste associated with the LWR operation.

Piping Systems of Current LWR Nuclear Power Plants

Relevant sources for identifying LWR piping systems are PSARs and FSARs prepared for BWR or PWR plants. These documents identify specific equipment and classify this equipment in terms of safety class, seismic category, and

applicable design code. The most recent BWR plant for which a PSAR is available (in the NRC Western Region) and the most recent BWR plant for which an FSAR is available are, respectively, Skagit Nuclear Power Project² and Washington Public Power Supply System (WPPSS) Nuclear Project No. 2.³ The most recent PWR plants for which a PSAR is available and the most recent PWR plants for which an FSAR is available are, respectively, Sundesert Nuclear Power Plant Units 1 and 2⁴ and San Onofre Nuclear Generating Station (SONGS) Units 2 and 3.⁵

The significant piping systems that were identified from these safety analysis reports are listed in Table 2-2 for PWR nuclear power plants and in Table 2-3 for BWR nuclear power plants. For each piping system, the seismic category and applicable design code are identified. Seismic category (or seismic design, for Sundesert Nuclear Power Plant definitions) and design code definitions are provided with Tables 2-2 and 2-3. Tables 2-2 and 2-3 show that many piping systems contain elements that have different specifications for seismic category and applicable design code. The systems are organized within the tables according to the general areas used in RG 1.70 (discussed in the previous section), although the name of each system is taken directly from the related safety analysis report.

3. LOAD-CAUSING EVENTS AND PHENOMENA

The six types of piping loads listed below are described in the six sections that make up this chapter:

- Weight
- Normal operating steady-state pressures and temperatures
- Pressure and temperature transients
- Loss-of-coolant accident load and other pipe failure loads
- Seismic loads
- Anchor movements and other mechanical loads

Anticipated transients without scram (ATWS), pressure discharge loads in pressure-suppression-type containments, and water hammer loads are included in the section on pressure and temperature transients. Although LOCAS or other pipe failures result in transient conditions, the effects of these loads are discussed in a separate section.

Weight

The weight of pipes, valves, fittings, insulation, and pipe contents is equal to the mass multiplied by the acceleration due to gravity. Weight-induced forces are resisted by hangers and supports. Valve weight can be relatively high for large isolating valves and actuators.

Normal Operating Steady-State Pressures and Temperatures

Both PWR and BWR power systems produce saturated steam at about 1,000 psia.⁶ In PWR steam supply systems, the core coolant hot- and cold-leg pressures are approximately 2,240 psia and 2,300 psia, respectively.⁷ In both systems, the temperatures of water and steam are between 500°F and 600°F. Because of the relatively long runs of nuclear power plant piping, thermal expansion, when restrained, can cause large piping loads. Table 2-4 lists pressures and temperatures for example nuclear power plant systems.

Three pressure terms are referred to in the ASME Code: design pressure, range of operating pressure, and peak pressure. The design pressure is the

specified difference between the pressure inside and outside of a pipe in the most severe loading for the normal operating condition (called Level A in the current ASME Code edition). The design pressure also includes allowance for pressure surges. The other two pressure terms are related to transient conditions and are discussed in the next section.

Pressure and Temperature Transients

In the operation of a nuclear power plant, there are many anticipated conditions that result in pressure and temperature transients. Accidental transients may also occur. Although their probability is low, they must be considered explicitly because of their potential severity. At the end of normal plant operations, transient conditions occur frequently, and their aggregate effect on fatigue is considered in the piping stress analysis using the ASME Code. Example plant start-up and shutdown temperature and pressure transient time histories for a PWR are shown in Figure 2-3.

Although postulated accidents may never occur, the potential exists for breaching the reactor fuel cladding and the pressure boundary, causing the subsequent release of radioactive material into the environment. Hypothetical accidents include:⁸

- Loss of reactor coolant due to major ruptures in the primary coolant system piping (i.e., LOCA, which is discussed in the following section)
- Reactor fuel damage due to rapid power increase beyond design limits (caused by fast removal of the control rod due to control rod system failure)
- Reactor fuel damage caused by dropping an irradiated fuel assembly during refueling operations
- System transients in PWR systems due to major ruptures in secondary (steam) system piping

Other accidents that can be envisioned are included in the design process. System transients are classified by plant operating condition (normal, upset, emergency, and faulted*). The operating conditions are analyzed to allowable stresses using the ASME Code. Summaries of design transients and

*These are called Levels A, B, C, and D in the 1977 edition of the ASME Code.⁹

number of occurrences for PWR and BWR systems are given in Tables 2-5 and 2-6, respectively.^{3,4} These transients and the number of occurrences are established by the plant owner (i.e., utility) and not by the NRC. The design pressures and temperatures are given as a function of time for each plant operating condition in the design specifications for each piping component.

The effects of temperature and pressure transients are given below. Anticipated transients without scram, pressure discharge loads in pressure-suppression-type containments, and water hammer loads are also discussed below.

The LOCA transient load and other pipe failure loads are discussed in the next section.

Effects of Pressure Transients. Pressure transients cause the fluid pressure to vary along the pipe as a function of time, but they cause essentially equal fluid pressure at any cross section in a piping system. The range of operating pressure is defined in the ASME Code provisions for Class 1 piping. This pressure is the difference between the maximum and minimum pressure as the piping system goes from one system load level to another. There is a corresponding temperature range that is also considered, as discussed below. Fatigue analyses are performed to verify that a piping system will not be adversely affected by the repeated change in loads.

The peak pressure is referred to in the ASME Code provisions for Class 2 piping. This pressure is the maximum or peak pressure that occurs due to a system transient. It is similar to the range of operating pressures used in the Class 1 design, except the lower limit pressure at the start of the load cycle is not included (i.e., it is considered to be zero).

Effects of Temperature Transients. Thermal transients cause time-varying temperature distributions across the pipe wall. These temperature distributions result in wall stresses that can be conceptualized as consisting of three parts: a uniform part, a linear part, and a nonlinear part. The uniform part results in expansion loads; the linear part causes a bending moment in the pipe wall; and the nonlinear part results in a surface stress,

which is often referred to as a "skin" stress. These temperature terms and their general transient characteristics are shown in Figure 2-4.

Due to the high temperatures in nuclear power plant piping, the following loads are caused in piping systems and are explicitly recognized in the ASME Code.

- Uniform expansion
- Radial linear gradient
- Radial nonlinear gradient
- Axial discontinuity difference
- Differential anchor movement

The first three loads are discussed above. The axial discontinuity difference load is due to the range of average temperature on each side of a gross structural or material discontinuity (e.g., between a pipe and a socket-welded fitting). The differential anchor movement load is due to temperature-induced displacements or rotations at the piping supports. Differential movements between supports load the piping system, and the potential for this loading is considered in the piping system analysis.

Anticipated Transients Without Scram. The consequences of anticipated temperature and pressure transients are normally limited by safety and control systems. Some deviations from normal operating conditions are minor, while others, occurring less frequently, can impose significant demands on plant equipment. Some transients require rapid shutdown of the nuclear reactor (i.e., initiating a scram), which quickly reduces the generation of heat in the reactor core. If a potentially severe anticipated transient occurred and the reactor shutdown system did not scram, then an ATWS would have occurred.

Table 2-7 lists initiating events for both PWRs and BWRs that might lead to an ATWS accident. Transients that isolate the reactor from normal cooling systems have the greatest likelihood of occurring and, for most reactors, the most severe potential consequences if the scram system fails. Of principal concern is the peak pressure that could occur in the primary cooling system. An example of a loss of feedwater ATWS pressure time history is shown in Figure 2-5. Table 2-8 lists the peak pressure in the primary sys-

tem for the most critical ATWS event for each NSSS manufacturer. These pressures were obtained through system model analyses, which are discussed in Chapter 6. Note that temperature effects, which are also important, generally occur some time after the system pressure has peaked.

Pressure Discharge Loads. In pressure-suppression-type containments, as developed by GE for their BWR system, energy can be absorbed by stored water to reduce the potential buildup of pressure inside the containment due to a LOCA or to relieve pressure in the primary coolant system during a severe transient. In the three GE containment systems (i.e., Mark I, II, and III) the reactor and primary coolant piping are located in a dry well that is connected by vent pipes to a suppression pool filled with water. In the Mark I system, the suppression chamber consists of a steel tank in the shape of a torus. In the Mark II system, a wet well with concrete walls is located below the dry well and is separated from it by a concrete floor diaphragm. The Mark III system suppression pool is wrapped around the bottom of the dry well and is connected to it by horizontal vent pipes.

During a postulated LOCA, a steam pipe breaks inside the dry well, causing pressure to increase due to blowdown of the reactor system. Pressurization of the dry well then causes the water initially in the vent system to be accelerated outward through the vent pipes into the suppression pool. As a result, a jet of water may impinge on structures located near the vent outlet, causing both direct jet and drag loads. In addition, jet formation can occur asymmetrically, leading to lateral reaction loads on the vents. Immediately after the vent water clearing, air from the dry well starts to form a bubble in the vent exit. The steam flowing through the vents condenses in the pool. The dynamic bubble pressure results in loads on submerged structures and on the walls of the suppression pool. As the bubble increases in size, the suppression pool begins to rise. Impact loads occur on structures near the pool surface. As the surface of the water continues to rise, structures normally above the surface become submerged and are subjected to drag forces. Finally the bubble collapses, and the water level falls back. The pool-swell transient associated with the dry-well air venting to the suppression pool generally lasts about 5 sec.

Following this stage, there is a relatively long period of steam flow through the vent system. During the steam condensation, lateral loading on the vents occurs due to the random movement of the steam-water interface. Depending on the size of the break, condensation oscillations and chugging may occur over a long period of time.

During severe transients, such as a turbine trip, safety-relief valves (SRV) are actuated, and steam from the primary coolant system is diverted through vents into the suppression pool. In a manner similar to LOCA-induced discharge, a water column is accelerated into the suppression pool, causing pressure and velocity transients that result in drag or jet impingement loads on submerged structures. Following water clearing, SRV air clearing drag loads on submerged structures occur. Pure steam is then injected into the pool, increasing the pool temperature. This process is relatively stationary so long as the local pool temperature is below the stability threshold temperature. At the threshold level, the condensation process becomes unstable; steam bubbles that are formed oscillate and collapse, giving rise to severe pressure oscillations that affect structures in the pool and the pool boundary.

The forces caused by LOCA and SRV discharges directly affect the vent pipe and discharge devices (i.e., ramsheads and quenchers). Water, air, and steam passing through the vents cause both reaction loads and lateral loads on the vent pipes. Fluids discharging into the suppression pool create jet loads that impact on submerged pipes near the outlets. Swelling of the pool causes impact forces on piping located above the normal water surface and also causes drag forces as the water level rises. Pressure forces due to condensation oscillations also affect submerged structures. Because of the severe pressure disturbances, vibrations are caused in other parts of the reactor building. These in turn influence piping systems in much the same way as phenomena such as earthquakes.

Water Hammer. A water hammer event is caused by an intense pressure pulse in fluid systems due to gradual or sudden changes in the water velocity. When moving water is decelerated, a compression wave forms and is followed by rarefaction waves due to reflection off distant boundaries. These pressure waves bounce back and forth along the piping system, causing forces on the various components.

Since 1971, about 120 water hammer incidents in nuclear power plants have been reported. These incidents have involved many types of fluid systems, including steam generator feed rings, feedwater and steam supply piping, residual heat removal systems, containment spray systems, emergency core cooling systems, and service water systems. The causes of water hammer include rapid valve motions, rapid condensation of steam pockets, pump start-up with partially empty lines, and steam-driven slugs of water.

Damage to piping systems generally has been minor, although there have been several cases of failure or partial failure of piping (see Chapter 2 and Part 1). Most of the reported damage is caused by forces due to pressure pulses at pipe bends and flow area changes. Failure can occur if the load exceeds the ultimate strength of the material. Failure also can occur due to the cumulative effect of the loads from successive water hammer and other events. This latter type of failure results when the number of stress cycles exceeds the fatigue limit of the material. The pressure boundary, internal system components (e.g., valve trim and pump impellers), and external components (e.g., pipe hangers, snubbers, valve operators, and pump motors) can be affected by water hammer.

Means to prevent the rapid condensation of steam in the steam generator feed rings of some PWRs are being instituted.⁶ Also, other analysis and testing requirements are being added to the design bases. Water hammer is listed by the NRC as an unresolved safety issue.¹⁰

Loss-of-Coolant-Accident Load and Other Pipe Failure Loads

Pipe failures are postulated in all safety-related systems that are classified as being either low- or high-energy systems. Failures in the primary coolant systems are called loss-of-coolant accidents, while failures in other systems are referred to as "other piping failures." A piping failure causes forces in the system to which it is connected. In addition, the fluid jet discharging from the break may impinge on adjacent piping systems, thus creating additional jet loads.

Criterion 4 (Environmental and Missile Design Bases) of the NRC General Design Criteria states:

Structures, systems and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents including loss-of-coolant accidents.

Potential effects of a LOCA or other pipe failure, including jet impingement, environmental influence, and pipe whip, are analyzed as part of the design process, and appropriate protection is provided for safety-related structures, systems, and components. A postulated LOCA or other pipe failure is defined as an instantaneous full double-ended circumferential pipe rupture (i.e., guillotine rupture), circular or elliptical break resulting in an axial split without pipe severance (i.e., longitudinal rupture), or through-the-wall leakage crack. Loss-of-coolant accident longitudinal split and circumferential rupture geometries are shown in Figure 2-6. The circumferential and longitudinal breaks are postulated for high-energy fluid system piping (i.e., design temperature greater than 200°F and design pressure greater than 275 psig), while the leakage crack is postulated for low-energy fluid system piping.

Forces associated with a LOCA or other pipe failures consist of a time-dependent thrust force caused by jet flow from the postulated pipe break. At the instant of rupture, a thrust pulse occurs due to a sudden pressure drop. Then a decompression wave propagates into the fluid contained by the pipe. The wave, traveling at the local speed of sound within the fluid, reaches a boundary and is reflected back as a compression wave. The process of wave propagation and reflection results in a wave thrust transient. Successive wave transmissions and associated fluid acceleration forces decay as the discharge flow rate builds up to a steady blowdown thrust that occurs due to the momentum of the fluid leaving the end of the pipe (assuming a fluid energy reservoir having sufficient capacity is available). Hence, forces resulting from a pipe failure are a function of time and space and depend upon the state of the fluid (i.e., saturated steam, subcooled water, or two-phase fluid) within the pipe prior to and during rupture, the break flow area, frictional losses, piping system characteristics, and other factors. The entire blowdown process occurs in no more than a few seconds.

The time-dependent pressures associated with a postulated pipe failure impose dynamic forces on the piping systems, which require installation of protective structures, such as pipe whip restraints. The analyses for pipe failure forces require that the dynamic nature of the pipe rupture phenomena and the nonlinear response characteristics of the piping and support systems be appropriately modeled. Pipe ruptures are postulated at numerous locations in the various piping systems. However, only one postulated rupture is analyzed at a time, and current NRC requirements state that associated piping must withstand the induced forces with no additional ruptures.⁷

Seismic Loads

Two earthquakes are defined and used in the design of nuclear power plants. The safe-shutdown earthquake (SSE) is an earthquake that is based upon an evaluation of the maximum earthquake potential of the site and region surrounding the plant (also called the design basis earthquake in older plants). For this event, it is necessary to ensure that:

- Reactor coolant pressure boundary is maintained
- Reactor can be shut down and maintained in a safe condition
- Potential off-site radioactivity exposures can be prevented

As given in Tables 2-5 and 2-6, this event is classified as a faulted plant operating condition.

The operating basis earthquake (OBE), which also is based on an evaluation of the site and region, is one that can reasonably be expected to occur during the life of the plant. The OBE is often assumed to be equal to one-half the SSE; however, there is some debate about whether an event of this size can reasonably be expected to occur during the life of the plant. Design for the OBE must ensure that, if an OBE-level event or smaller event occurs, the plant can continue operation without undue risk to the health and safety of the public.

Two effects associated with earthquake motions are used to analyze and/or design a piping system -- inertial effects and differential anchor movements. Inertia-induced loads are due to the forces caused by the accelerations of the masses (piping products and fluid contents) during dynamic

response of the piping system. Because earthquake forces cause stresses (hence strains) in all parts of the nuclear power plant, piping supports generally will displace and rotate. The differential movements between anchor points cause the piping system to be stressed and also must be considered in design.

Anchor Movements and Mechanical Loads

In addition to stresses caused by pressure, temperature, and accident loads, stresses due to anchor movements and mechanical loads also are considered in the analysis and design of piping systems. Anchor movements can be caused by any source that creates relative movements between the piping system supports. For piping systems that are attached directly to the concrete structure, relative anchor movements, including translations and rotations, can result from seismic motions (as discussed in the previous section) that cause distortions in the concrete structure between the supports. For smaller piping systems that are attached to larger systems, anchor movements are similarly caused by relative motions of the larger system to which they are attached. Anchor movements due to temperature changes are discussed above in the section on pressure and temperature transients.

Mechanical loads due to valve operation also can cause forces in the piping system elements. Water hammer and SRV discharge loads are included in this category. Because of the importance of potential loadings from these sources, water hammer and SRV effects are discussed above as part of pressure and temperature transients. Mechanical loads can also be sustained loads, such as the weight of a valve and operator that is supported by the piping elements. During normal operations, vibrations in the piping systems are caused by the operation of various types of mechanical equipment (e.g., pumps, motors, etc.).

4. METHODS AND PROCEDURES FOR DEFINITIONS OF LOADS

The current definitions for piping loads are discussed in this chapter. Because of the phenomenological nature of each load type, the intermediate steps required to determine each load type are reviewed. For example, the development of seismic loads involves definition of free-field motion, soil-structure interaction analyses of various building and soil models, and parametric studies. Also, the particular characteristics of each load type determine what aspects are pertinent to the discussion. For example, LOCA loads involve break locations, break types, directions, and dynamic analysis load characteristics. The methods and procedures for defining each of the loads identified in Chapter 2 are presented below.

Weight

The weights of piping products, insulation, and their fluid contents are based on equipment manufacturers' specifications and design fluid states. Loads due to weight can be accurately and reliably predicted and usually involve almost no uncertainty. Weights are often conservatively estimated at the beginning of a nuclear power plant design because the final selection of components (hence size and weight) have not been made at that time.

Normal Operating Steady-State Pressures and Temperatures

The temperature and pressure loads associated with the normal operating conditions of the piping systems are specified by the NSSS manufacturer and other equipment suppliers. Like weight loads, normal operating steady-state pressures and temperatures are reliably predictable because they are constrained by the design and operations of each plant. Deviations from normal conditions are either anticipated occurrences or are classified as abnormal and thus analyzed as part of upset, emergency, and faulted plant operating conditions. Pressure and temperature transients and pipe failure loads are discussed below.

Pressure and Temperature Transients

Pressure and temperature transients are caused by either normal or abnormal (accident) changes in the current operating state. Temperature and pressure transients are usually idealized as step or ramp gradients with time. One-di-

mensional finite-difference heat conduction computer codes are used in the reactor system analysis to solve thermal transient problems.⁴ The following discussion of analysis methods for ATWS is also applicable to other transients (e.g., transients with scram). The output from a system analysis includes time histories of temperatures and pressures that are used in the subsequent stress analysis.

For thermal gradient loading, the nominal stress is related to the change of temperature across the thickness of the pipe. As discussed in the section on temperature and pressure transients in Chapter 3, uniform, linear, and nonlinear portions of the temperature transient are calculated and appropriate equations of the ASME Code are solved to establish that the induced stresses are acceptable.

The methods and procedures for analyzing ATWS, pressure discharge loads, and water hammer are discussed below. Loss-of-coolant accident and other pipe failure loads are discussed in the next section.

Anticipated Transients Without Scram. The issue of anticipated transients without scram has been discussed throughout the nuclear industry since the early 1970s. The question of whether an ATWS event probability is sufficiently high to warrant inclusion of this type of event as a design basis for nuclear power plants has been hotly debated. At first the NRC excluded ATWS events from the design basis. In September 1973 the then-AEC staff published WASH-1270,¹¹ which gave "acceptance criteria" required to protect against ATWS events. Following this report, each of the four NSSS manufacturers was required to submit analyses and supporting information on ATWS for review by the NRC staff. The vendors and plant owners have questioned whether the NRC staff's requirements are necessary and justified. The industry contends that the probability of an ATWS event is sufficiently low and that therefore ATWS occurrences should not be considered as design basis events. The NRC stated its position in Volumes 1 and 2 of the NUREG-0460,^{12,13} which gave recommendations for mitigating the consequences of ATWS events. These recommendations have been criticized by industry and some members of the NRC staff. These concerns have been reviewed, and NRC has issued Volume 3 of NUREG-0460,¹⁴ which restates their position. Rather than basing their position on a numerical safety objective (i.e., a probability of an occurrence) the principal criterion for choosing among alter-

native plant modifications, necessary to prevent or mitigate ATWS consequences, is the level of safety judged by the NRC to be necessary.

The procedures for obtaining pressure and temperature values for ATWS events involves ATWS evaluation models. A model developed by each NSSS manufacturer is used to evaluate the behavior of the reactor systems, including the reactor coolant systems (e.g., piping), associated systems, and the reactor core during a postulated ATWS event. Each vendor model includes one or more computer programs and all information necessary to specify the calculation procedures.

Pressure Discharge Loads. The expulsion of water, air, and subsequently steam in BWR plants following a postulated LOCA or SRV actuation induces a flow velocity and acceleration field within the suppression pool. Piping either submerged within the pool, or located close to the pool surface, will be loaded due to this induced motion. The methods and procedures for defining these loads is different depending on the type of BWR containment (i.e., Mark I, II, or III). Reference 15 gives the NRC position on licensing of pressure suppression containments for the three containment systems.

The major concern for the Mark I system is the loading on the containment torus structure. The vibration of the torus is transmitted to the torus support and to certain piping systems attached to the torus. Other significant concerns include: (1) drag loads on components submerged in the pool; (2) pool swell impact and drag loads on structures above the pool surface; and (3) vent system lateral loads occurring during the chugging period. The NRC position is given in Reference 16. The NRC staff is currently in the process of reviewing and developing load criteria for the Mark I system.

The NRC position on pipe loads for the Mark II containment is given in Reference 17. Currently, review of Mark II plants is divided into a Lead Plant Program (LPP) and a Long-Term Program (LTP). Reference 17 gives the load criteria acceptable for the LPP. Table 2-9 gives the Mark II containment hydrodynamic loads on piping systems and the NRC acceptance criteria for computing the loads.

The LOCA loads receiving primary attention in the Mark III system evaluation program include: (1) drag loads on components submerged in the suppression

pool; and (2) pool swell impact and drag loads on structures above the pool surface. General Electric Company has conducted tests and performed analyses to establish LOCA dynamic loads for the Mark III design.¹⁸ The NRC has reviewed these loads and has issued a set of design criteria that are acceptable for preliminary design approval. These design criteria are described in Section 6.2.1.9 of NUREG-75/110, *GESSAR 238 Nuclear Island Standard Design Safety Evaluation Report*,¹⁹ and the modified criteria for impact loads are described in Supplement 2, Section 6.2.1.9, to the GESSAR safety analysis report.¹ The loads due to SRV quencher discharge have been established by tests, and the evaluation of these loads by the NRC is documented in the section reference immediately above.

Water Hammer. A description of procedures for defining water hammer loads is given in Reference 20, which is titled: *Water Hammer in Nuclear Power Plants*. Classical water hammer problems in liquid systems and methods for solution are given in Reference 21. Standard charts for estimating peak pressure rise due to valve closure downstream of a reservoir have been developed.²² Steam bubble collapse and impact due to steam-driven slugs of water in steam lines cannot be predicted with sufficient accuracy for design calculations.²⁰ The present approach is to design to prevent the occurrence of water hammer for these events.

Piping systems are analyzed in two steps. First the transient water hammer pressures and forces are calculated, then they are input to a structural model to obtain piping and component stresses and support loads as a function of time. These two analyses are decoupled, which assumes that the piping system is fixed during the fluid transient calculation. However, it is possible to include the effect of local elastic or plastic behavior of the pipe wall by modifying the wave speed of the pressure pulse.

Current design codes require that impact forces (e.g., water hammer) be considered; however, none of the standard codes provides guidance for the load definition or type of analysis that should be performed. Currently, water hammer is specifically identified in the NRC Standard Review Plan (SRP) only for the feedwater and condensate system and emergency core cooling system. RG 1.68 requires testing of various fluid systems, and RG 1.70 requests information on dynamic analyses of safety-related piping systems.

Loss-of-Coolant-Accident Load and Other Pipe Failure Loads

Loss-of-coolant accident and other pipe failure loads used to analyze and design piping systems are defined in the following NRC documents.

- SRP 3.6.1: Plant Design For Protection Against Postulated Pipe Failure in Fluid Systems Outside Containment
- Branch Technical Position APCS 3-1: Postulated Piping Failures in Fluid Systems Outside Containment (attached to SRP 3.6.1)
- Appendix A to APCS 3-1: Definitions
- Appendix B to APCS 3-1: Giambusso Letter
- Appendix C to APCS 3-1: O'Leary Letter
- SRP 3.6.2: Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping
- Branch Technical Position MEB 3-1: Postulated Break and Leakage Locations in Fluid System Piping Outside Containment (attached to SRP 3.6.2)
- RG 1.46: Protection Against Pipe Whip Inside Containment

SRP 3.6.2 provides general discussion of NRC review responsibility, references other NRC documents for pipe break locations, and provides dynamic analysis criteria for the effects of piping failure loads. Relevant requirements of the NRC documents are discussed below. The piping loads due to a pipe failure are described in terms of break locations, break types and force direction, and dynamic analysis load criteria.

Postulated Pipe Break Locations. Specifications for pipe break locations are given in NRC documents for piping inside containment--RG 1.46 and Branch Technical Position (BTP) MEB 3-1--and outside containment--SRP 3.6.1, BTP APCS 3-1 with Appendices A, B, and C, and BTP MEB 3-1. For pipe break locations inside containment, RG 1.46 is the primary basis for locating LOCA pipe breaks. Branch Technical Position MEB 3-1 will be considered by the NRC for LOCA failures if the criteria specified in RG 1.46 are impractical to implement. Note that BTP MEB 3-1 was developed for piping outside containment and normally would not be used for determining failure points for reactor coolant piping or other piping systems located inside containment.

For piping systems located inside containment, RG 1.46 specifies that LOCA piping breaks are postulated to occur at the following locations in each piping run or branch run.

- The terminal ends
- Any intermediate locations between terminal ends where stresses* (or stress intensities) exceed defined values
- Any intermediate locations where the cumulative usage (fatigue) factor* exceeds 0.1
- Minimum of two intermediate locations

Example LOCA break locations in the primary coolant system for a PWR are shown in Figure 2-7.

For piping systems located outside containment, SRP 3.6.1 provides the basis for review of plant design for protection against postulated piping failures and references BTP MEB 3-1, which defines postulated break and leakage locations. For plants with construction permit (CP) applications tendered before July 1, 1975, the following documents may be used in lieu of BTP MEB 3-1 to define break locations:

CP tendered before July 1, 1973: Appendix B to BTP
APCSB 3-1
(Giambusso Letter)

CP tendered before July 1, 1975, but after July 1, 1973: Appendix C to BTP
APCSB 3-1
(O'Leary Letter)

These documents specify less restrictive requirements and are superseded by BTP MEB 3-1 for new plants.

BTB MEB 3-1 encourages the practice of separating or isolating fluid systems from essential safety-related systems and components. Requirements are given for locating breaks in high-energy fluid system piping and leakage cracks for moderate-energy fluid system piping. Failure locations are defined for piping systems depending on their location (e.g., in containment penetration areas, enclosed within protective structures, not enclosed within protective structures, between containment isolation valves, etc.). Specific re-

*Calculated according to Section III of the ASME Boiler and Pressure Vessel Code⁹

quirements for locating failures outside containment are related to stress and fatigue constraints in a manner similar to limits for piping located within containment.

Postulated Pipe Break Types and Force Directions. RG 1.46 and BTP MEB 3-1 give requirements for pipe break types and force directions for piping inside and outside containment, respectively. The requirements for the two piping categories are similar; however, specific exceptions do occur. The requirements for circumferential pipe breaks and longitudinal pipe breaks are given below. Through-wall leakage cracks are considered for environmental effects (e.g., fluid spray and flooding) and are generally not analyzed for force effects.

Circumferential pipe breaks are postulated for piping exceeding a nominal pipe size of 1 in. For piping inside containment, RG 1.46 states:

Circumferential breaks are perpendicular to the pipe axis, and the break area is equivalent to the internal cross-sectional area of the ruptured pipe. Dynamic forces resulting from such breaks are assumed to separate the piping axially and cause whipping in any direction normal to the pipe axis.

The corresponding requirements from BTP MEB 3-1 for piping outside containment are as follows:

The dynamic force of the jet discharge at the break location should be based on the effective cross-section of the pipe...

and

Pipe whipping should be assumed to occur in the plane defined by the piping geometry and configuration, and to cause pipe movement in the direction of the jet reaction.

Note that RG 1.46 states for circumferential breaks that the pipe can whip in any direction normal to the pipe axis, whereas BTP MEB 3-1 states that pipe whip can occur in a direction defined by the piping geometry, stiffness, and fluid forces. This apparent inconsistency was noted in the PSAR for the Sundesert Nuclear Plant, where exception was taken to the RG 1.46 provision.⁴

BTP MEB 3-1 also states that, when the maximum stress range in the circumferential direction is at least 1.5 times that in the axial direction, only longitudinal breaks will be postulated. The PSAR for Sundesert adopted this requirement for LOCAs inside containment.

Longitudinal pipe breaks are postulated for piping with nominal sizes 4 in. and larger. For piping inside containment, RG 1.46 states:

Longitudinal breaks are parallel to the pipe axis and oriented at any point around the pipe circumference. The break area is equal to the sum of the effective cross-sectional flow area upstream of the break location and downstream of the break location or is equal to a break area determined by test data which defines the break geometry. Dynamic forces resulting from such breaks are assumed to cause lateral pipe movements in the direction normal to the pipe axis.

The corresponding requirements from BTP MEB 3-1 for piping outside containment are as follows:

Longitudinal breaks should be assumed to result in an axial split without pipe severance. Splits should be oriented (but not concurrently) at two diametrically opposed points on the piping circumference such that the jet reaction causes out-of-plane bending of the piping configuration. Alternatively, a single split may be assumed at the section of highest tensile stress as determined by detailed stress analysis (e.g., finite element analysis).

and

The dynamic force of the fluid jet discharge should be based on a circular or elliptical ($2D \times 1/2 D$) break area equal to the effective cross-sectional flow area of the pipe...

and

Piping movement should be assumed to occur in the direction of the jet reaction unless limited by structural membrane, piping restraints, or piping stiffness...

Note that the required failure area for LOCAs outside containment (i.e., from BTP MEB 3-1) appears to be equal to one-half the area for LOCAs inside containment (i.e., from RG 1.46). This inconsistency was noted in the PSAR for

the Sundesert Nuclear Plant, where exception was taken to the RG 1.46 provision.⁴ However, the RG 1.46 requirement has been interpreted by others to mean one full area break in each pipe segment.

BTP MEB 3-1 also states that when the maximum stress range in the axial direction is at least 1.5 times that in the circumferential direction, only a circumferential break will be postulated.

Dynamic Analysis Load Criteria. The load on a piping system due to a LOCA or other pipe failure is expressed either as a time-dependent function, which includes the effects of the thrust pulse (due to sudden pressure drop), the thrust transient (due to wave effects), and the blowdown thrust (due to build-up of discharge flow rate), or as a steady-state jet thrust function. SRP 3.6.2 defines the requirements for establishing the forcing functions for a LOCA or other pipe failure. The following criteria from SRP 3.6.2 define how the forcing time functions are to be developed.

- The rise time for the initial pulse should not exceed one millisecond (unless a longer time is substantiated by data or theory).
- The jet thrust time variation should be related to pressure, enthalpy, fluid volume in the upstream reservoir, and reservoir capacity.
- The shape of the transient function may be modeled on the basis of break area, system flow constrictions, piping friction losses, and flow-limiting devices.

SRP 3.6.2 permits various approaches for the dynamic analysis of a LOCA or other pipe failure used to calculate piping and restraint system responses. However, each method is required to adequately account for the following effects:

- Mass inertia and stiffness properties of the system
- Impact and rebound
- Elastic and inelastic deformation of piping and restraints
- Support boundary conditions

Specifically, lumped-parameter, energy balance, and static analysis models are acceptable. Other models may be used, if justified. The lumped-parameter

analysis approach requires that a time-history force analysis be performed using force time histories as input. A hydraulic systems analysis is conducted to determine the forcing functions. The method of characteristics or the finite-difference approach is used to perform a one-dimensional two-phase fluid analysis. In a typical analysis, finite control volumes are selected, and the equations representing conservation of mass, momentum, and energy, supplemented by the equations of state and the constitutive equations for wall shear and wall heat transfer are solved for each volume. Key parameters calculated as a function of time are pressure, mass flow rate, and fluid density. These are appropriately combined to produce the forcing functions used in the piping analysis. The resulting time history, F , at any flow area is calculated considering the pressure, area, and momentum flux as follows:⁷

$$F = \left[(p - p_a) + \frac{|G|G}{\rho g} \right] A$$

where:

- p = system pressure
- p_a = external pressure
- G = mass flux
- ρ = fluid density
- A = flow area of control volume
- g = acceleration due to gravity

As discussed in Chapter 6, on the methodologies for developing forcing functions, different approaches may be used in the dynamic analysis. Some analysts use only the forcing function at the break plane as input, while others apply multiple time histories at the break plane and at all bends or discontinuities in the system. In either case, a two-step approach is required for each postulated LOCA or other pipe failure to obtain piping response where the hydraulic systems analysis is required to determine the input forces to the dynamic response analysis. An example LOCA pipe thrust time history is shown in Figure 2-8.

When an energy balance or static model is used in the piping analysis, an amplified step function with the following form may be used.

$$T = kpA$$

where:

- T = jet thrust force (unamplified)
- p = system pressure prior to break
- A = pipe break area
- k = thrust coefficient ($k = 1.26$ for steam-saturated water or steam-water mixture and $k = 2.0$ for sub-cooled, nonflashing water)

If pipe rebound occurs upon impact at a restraint, a factor of 1.2 should be included to amplify the forcing function if the energy balance model is used. When the static analysis model is used, a dynamic amplification factor is required that is conservatively based on comparisons with factors derived from detailed dynamic analyses performed for comparable systems (typical values range from 2 to 3).

Jet Impingement. Requirements for jet impingement forces are given in SRP 3.6.2. The total jet thrust force, given by the above equation, is specified to be time and distance invariant and to expand uniformly at a half angle not to exceed 10° along a straight path. Jet expansion of steam or water-steam mixtures within a zone of five pipe diameters from the break may be determined by testing or may be based on a valid analysis (e.g., Moody's expansion model).²³ The break opening through which the fluid is expelled may be assumed to be a circular shape with a flow area equal to the area of the break. The impingement force is considered to be uniformly distributed across the cross-sectional area of the jet, and only the portion intercepted by a pipe is applied to the pipe. An example LOCA jet impingement on an adjacent pipe is shown in Figure 2-9.

Seismic Loads

In the analysis of piping systems for earthquake motion, the seismic input is defined by either time-history support motions or response spectra. The process for obtaining seismic input for a piping system consists of the following steps:

- Definition of free-field peak ground acceleration and ground response spectra
- Development of free-field ground motion time histories

- Deconvolution of free-field ground motion to bedrock motion
- Soil-structure interaction analysis of building and soil model to obtain base mat input motions
- Building-dominant model and equipment-dominant model analyses to obtain input motion for the piping systems
- Parameter studies to provide a basis for modifying floor time-history and response spectra input

The free-field peak ground acceleration for the SSE is based upon an evaluation of the maximum earthquake potential at the site. A response spectrum shape defined by NRC RG 1.60 is anchored to the peak ground acceleration value (this is performed for both vertical and horizontal directions). A similar process is conducted for the OBE, where the peak ground acceleration is usually 50% of the SSE value. Statistically, independent vertical and horizontal time-history motion records are developed that envelope the free-field response spectra. Example design and synthetic-time-history ground response spectra are shown in Figure 2-10. Bedrock time-history motions are obtained by deconvolution, and a soil-structure interaction (SSI) analysis of a coupled soil/building system is performed to obtain base mat motions. This type of analysis usually involves a finite-element two-dimensional representation of the building/soil system. The lumped mass-spring method or "compliance function method" can be used for cases where the foundation is shallow and the soil is relatively uniform. Damping values for the structure are based on RG 1.61, and values for the soil include material and radiation damping components.

In the SSI analysis, the model of the structure is usually a simplified representation of the structure. Thus, detailed building models are also analyzed to obtain input to the piping systems. Two types of models are analyzed using the base mat motions obtained from the SSI analysis as input: a building-dominant model, where the building elements are discretely represented, and an equipment-dominant model, where the large equipment components are modeled in detail. The purpose of the building-dominant model is to obtain stresses, moments, displacements, etc., in the various parts of the structure and to obtain motions (either time histories or response spectra) at the base of equipment and piping supports. An example earthquake floor response spectrum is shown in Figure 2-11. These motions are then used to analyze the

piping systems that do not significantly modify the building response. In contrast, in the equipment-dominant model, the reactor coolant system and heavy equipment are coupled with the structure and do influence the building response. Even with this model, the piping system may still be coarsely represented, but sufficient detail is provided so that the interaction between building and piping system is modeled and the support motions are accurately computed. The support motions that are obtained are then used as input to detailed models for the reactor coolant system and other large equipment items.

Because of uncertainties in the various parameters (e.g., soil shear modulus, damping, material density, etc.) that are part of the analytical models, a single set of analyses may not conservatively represent the actual physical phenomenon. If response spectrum input to piping systems is used, the peak of each spectrum is broadened on the basis of individual frequency variations due to uncertainty of each building/soil system parameter (or arbitrarily broadened $\pm 15\%$). Similarly, if time histories are used as excitations to the piping systems, at least three different time intervals, Δt , between the acceleration ordinates are defined and analyses performed. For these analyses, the time intervals are established based on frequency variations due to parameter uncertainty.

The seismic load input to piping systems consists of either three-component broadened response spectra or three-component time histories (with variable time intervals to reflect the uncertainty in the building/soil system parameters). Spectra and time histories are obtained from the building model at the elevations of equipment and piping systems. These spectra are computed for various damping values corresponding to the piping system properties because damping varies with the size of piping system components (recommended damping values can be found in RG 1.61).

There are various approaches for inputting the response spectra or time histories to the piping system models. A common method when using response spectra is to develop envelope curves that are everywhere above the corresponding response spectra at all locations of the piping system anchors. The use of an envelope spectrum can result in an overly conservative response. Thus, various procedures have been developed to input a different time history or response spectrum at each piping support.^{24,25,26,27}

For some simple piping systems, equivalent static procedures are used in lieu of a dynamic analysis. For these analyses, a constant acceleration is applied to the piping system masses. The equivalent acceleration is determined by applying a factor to the acceleration response spectrum. Guidance for this approach is given in NRC Standard Review Plan 3.7.2 and Reference 28.

Both inertial effects and relative support displacements due to earthquake motions are used to design piping systems. Thus, both types of effects must be identified in the building/soil system analysis prior to performing the ASME Code analysis. Reference 29 discusses procedures for separating relative support displacements from inertial effects.

Anchor Movements and Mechanical Loads

As a by-product of the dynamic analyses of the building support structure, relative movements between piping anchor points are computed. These data are used as input to the analysis of the piping systems. The methods and procedures for defining seismic input are discussed above.

Water hammer and SRV-induced mechanical loads are discussed as part of the section on pressure and temperature transients. Sustained mechanical loads due to the weight of pipe components are based on equipment manufacturers' specifications, and, like other piping products (see section on weight), these loads are accurately and reliably predicted and involve almost no uncertainty.

Mechanical vibrations due to both normal operations and abnormal occurrences are considered in the fatigue analyses of piping systems. The number of cycles and stress levels are estimated and checked against the allowable values given in the ASME Code.

5. LOAD COMBINATIONS

RG 1.48, titled *Design Limits and Load Combinations for Seismic Category I Fluid System Components*, specifies load combinations to be used in the design of piping systems. The combinations and associated design limits are categorized by plant operating condition (i.e., normal, upset, emergency, and faulted). In addition to these categories, various safety analysis reports also define the design and test conditions.^{3,5} The design condition loads are generally the design pressure or the design pressure combined with weight and OBE inertial effects. The test condition load is due to testing pressure combined with the load due to weight.

Tables 2-10 and 2-11, which are based on References 3, 4, 5, and 9, give load combinations for Class 1 piping and Class 2 and 3 piping, respectively. The loads associated with each plant operating condition and stress category are grouped in the tables. The load combinations are based on the stress equations of the ASME code, Sections NB-3600, NC-3600, and ND-3600, for Class 1, 2, and 3 piping products, respectively.

The combination of earthquake loads with transient-induced loads varies between plants. For example, the WPPSS Nuclear Project No. 2 FSAR³ indicates that transient loads, such as those associated with safety-relief valve discharge and turbine stop valve closure, were combined with earthquake loads using the square-root-of-the-sum-of-the-squares (SRSS) procedure. In contrast, the FSAR for San Onofre Units 2 and 3⁵ indicates that these loads were combined absolutely.

The current requirements of the NRC for combining loads are given in Reference 30. On the basis of a review of recent studies of load combinations, the NRC allows LOCA and SSE loads to be combined by the SRSS procedure; however, OBE plus SRV loads are to be combined by absolute sum until an appropriate basis for a different procedure has been demonstrated. Reference 30 mentions that B&W and C-E use the absolute-sum method to analyze the primary coolant system components and supports while GE and Westinghouse have been using the SRSS method in selected applications.

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6. METHODOLOGIES AND DATA BASES FOR CURRENT LOAD DEFINITIONS AND COMBINATIONS

Introduction

The methodologies and data bases for current load definitions and combinations are discussed in this chapter. The stress equations given in the ASME Code, which specify and combine the various loads that affect piping components, are the result of research dating back to work published by Von Karman in 1911.³¹ The results of research work over the last 65 years have been incorporated into various codes for industrial piping, such as American National Standards Institute (ANSI) B31.1,³² which were used to design the piping systems in early nuclear power plants. It was felt that the degree of reliability provided by ANSI B31.1 was not consistent with the reliability desired for Class 1 nuclear piping systems. Consequently, a new standard called ANSI B31.7 - 1969, Nuclear Power Piping, was issued and used for several years as the basis for the design of nuclear power plant piping. ANSI B31.7 covered three classes of piping: Classes 1, 2, and 3. The design requirements for Classes 2 and 3 were almost identical to those of ANSI B31.1. For Class 1, the requirements of ANSI B31.1 formed the foundation for ANSI B31.7, but additional requirements analogous to those found in the ASME Code for Class 1 pressure vessels were also included.

Finally, the 1971 edition of the ASME Code included rules for Class 1, 2, and 3 piping systems. The stress analysis portion of the requirements was generally taken from ANSI B31.7. There have been changes in recent years to the rules, but the basic philosophy can be traced back to ANSI B31.7, ANSI B31.1, and ultimately back to research work on piping systems over the past 65 years to the work of Von Karman.

The methodologies and data bases for pressure and temperature transients, LOCA loads, and seismic loads are described below. Ultimately, each of the loads determined is combined with other loads and is checked according to the rules of the ASME Code.

As discussed in Chapter 4, weight and normal operating steady-state pressures and temperatures are based on NSSS and other equipment manufacturers' speci-

fications. These loads are accurately and reliably predicted, and deviations of pressure and temperature from normal conditions are classified as other types of loads.

Pressure and Temperature Transients

The current methods and procedures for analyzing ATWS, pressure discharge loads, and water hammer are discussed below. LOCA and other pipe failure loads are discussed in the next section.

Anticipated Transients Without Scram. The computer codes and bases used by the four NSSS vendors for ATWS analyses are given in Table 2-12. The referenced computer codes and analysis reports are also applicable to other transients (i.e., those that do scram). In the reactor cooling system analysis, models for fluid flow, core components, pressurizer, and steam generator (i.e., PWR system) are included. Parameters for the core, reactor coolant system, pressurizer, and secondary system (i.e., PWR system) are assumed in the analysis. Sensitivity studies are then performed to ensure that conservative results are obtained. The NRC has used the RELAP3-B³³ (see Reference 34 for current version) code and actual transient data to check the results of the transient analyses for each of the four NSSS vendors' codes.¹³

The effects of pressures and temperatures during an ATWS event are checked against the ASME Code emergency condition stress limits (now called Service Level C) to ensure that failure would not occur and that the reactor can be safely brought to cold shutdown without violating any other ATWS acceptance criteria.

Pressure Discharge Loads. The bases for postulated LOCA and SRV-discharge loads for pressure-suppression-type containments include testing and analytical work. Until the early 1970s, the only significant hydrodynamic loads considered in design were postulated LOCA loads, and SRV-discharge loads were considered to be small and secondary. Tests in the early 1960s at the Humboldt Bay Nuclear Power Plant and at Moss Landing, California, had been the basis for this position.³⁵ In 1972, two German BWR plants experienced severe vibratory loads during extended SRV operation. Also, in 1973 a large-scale test program

for the Mark III system demonstrated that wet well internal structures located above the suppression pool could be significantly loaded during a LOCA-induced pool swell process. This new information indicated the need for a reassessment of both Mark I and Mark II containment designs.

The affected utilities formed Mark I and Mark II ad hoc owners groups in 1975 with GE as the program manager. The Mark I owners group started the Short-Term Program (STP), which involved a series of tests to demonstrate that licensed plants could continue operation without undue risk to the health and safety of the public for a two-year period until the completion of a comprehensive Long-Term Program (LTP).

The Mark II owners group developed a program to support their pool dynamic loads application methods. The Lead Plant Program (LPP) was started to establish conservative loads for the Mark II plants (i.e., Zimmer, Shoreham, and LaSalle). The LTP, which is to be completed during the second quarter of 1980, was formulated to provide justification for reducing the conservative loads established from the LPP and to provide additional confirmation of certain loads utilized in the LPP.¹⁵

A large-scale test program was started in 1973 to confirm the analytical models used in development of the Mark III design. The basic design and performance of the Mark III containment system is now well established, and the remaining tests to be conducted are of a confirmatory nature.

A summary of the technical basis for piping loads in pressure-suppression containment systems is given below for the three GE containment systems. General information for the three systems is found in Reference 15, and detailed data for the Mark II system are found in Reference 17.

The test programs for the Mark I STP and LTP are given in Table 2-13. On the basis of the results of the STP tests,^{36,37,38} the NRC has concurred that the licensed Mark I plants can safely continue to operate.¹⁶ Final evaluation of the Mark I loads is being performed as part of the NRC staff's generic Task Action Plan A-7, "Mark I Long-Term Program."³⁹

The Mark I owners group is developing a quencher device to replace the rams-head device originally used. Because of space limitations in the torus, a Mark I unique T-quencher has been developed as part of the LTP and tested at the Monticello plant.⁴⁰ GE has developed an analytical model to predict loads associated with steam vent cleaning and has compared the results with in-plant test data.^{41,42} Some of the data for the quencher device were provided by German manufacturers.⁴³

A test program was established by the Mark II owners group in 1975 to determine LOCA-induced loads for the Mark II containment. Tests have been performed at the 4T facility to obtain data on pool swell, froth, and break-through loads for the Mark II design. As part of this program, downcomer lateral loads during condensation and chugging have been determined.¹⁵ In addition, 1/6-scale single-vent and multivent test series are being conducted by Japanese BWR utilities and manufacturers.¹⁵

The Mark II owners group is sponsoring research on the GE cross quencher, although the three lead plants (Zimmer, Shoreham, and LaSalle) are considering a T-quencher designed by Kraftwerk Union Aktiengesellschaft (KWU) of Germany.¹⁵ The Japanese are also performing testing of quenchers at the Tokai-2 Nuclear Power Plant, which is a Mark II facility using a GE cross quencher device.¹⁵

Table 2-14 lists the methodology and test bases for Mark II containment hydrodynamic loads on piping systems. A complete description of the technical bases for the Mark II pressure suppression containment system piping loads is given in Reference 17. Appendix 6 of Reference 17 lists all the supporting program tasks for LOCA and SRV loads. The activity types, target completion dates, and documentation are given in this appendix.

A major element in the LOCA load-predicting process is the pool swell analytical model developed by GE.⁴⁴ This model has been qualified by a series of comparisons between predictions and experimental results and also has been independently checked by the NRC.¹⁷

A large-scale test program for the Mark III pressure-suppression containment system was initiated in the fall of 1973 to confirm analytical models used

in the Mark III design.^{45,46,47} The objectives of the test program were to confirm the conservatism in the analytical models being used to predict Mark III containment pressure response. The LOCA pool dynamic loads for the Mark III containment design are described in Reference 45.

All applicants for the Mark III system propose to use a quencher-type discharge device that has been tested to justify the loads used in the containment structural design. NRC evaluation of loads is documented in Section 6.2.1.9, Supplement 1, of Reference 19.

Water Hammer. The fluid transient conditions due to water hammer are calculated by solving the one-dimensional equations of pipe flow. At changes in the flow direction or flow area, the transient pressures and velocities are used to obtain time-dependent forces. The computer codes WHAM⁴⁸ and PTA-1⁴⁹ are examples of programs used to compute fluid transient response for a liquid. The Sundesert PSAR⁴ cites computer program WATHAM for water hammer transient analysis. The WHAM code uses the method of wave superposition to solve the one-dimensional water hammer equations. This program includes the effect of pipe friction and locally elastic pipe wall behavior. The PTA-1 code uses the method of characteristics to solve the water hammer problem. Both elastic and inelastic pipe wall response is approximated, and pipe wall friction is simulated by steady-state flow friction factors.

For steam hammer, RELAP-4/MOD 5³⁴ has been used to solve the one-dimensional compressible flow equations. This program has been checked against test data.⁵⁰ Computer program STEHAM was referenced in the Sundesert PSAR⁴ for performing the steam hammer transient analysis. RELAP-4/MOD 5 also can solve the two-phase flow situation, such as that encountered following the actuation of safety-relief valves with loop seals.

Once the transient solution is obtained, forces are input to a structural analysis code. Many of the codes used by the NSSS vendors and A/Es are proprietary. An example general-purpose code that is commonly available is PIPESD.⁵¹

Loss-of-Coolant-Accident Load and Other Pipe Failure Loads

General Design Criterion 4, "Environmental and Missile Design Bases," of Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," requires that systems and components important to safety: "shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures." Originally, the primary system pipe break requirements developed from the need to define a design basis for the containment building.⁵² It was assumed that the entire fluid energy in the primary or secondary system was released into the containment building. In this context, the type of break in a pipe wall, speed of propagation, and final break size were not significant because the entire primary or secondary system inventory would be eventually released to the containment.

In the early 1960s, nuclear reactor power plants were designed for a LOCA that was defined as a sudden rupture of the largest branch pipe joining the primary system.^{7,53} Structural consequences were not considered at that time. By 1967, the instantaneous double-ended rupture of the largest reactor coolant system pipe was postulated. At first, the pressure loads were applied statically. A need arose for postulating pipe breaks to demonstrate that the plant could be safely shut down following a LOCA. In order to verify the effectiveness of the ECCS, the rupture speed and size became important. By the late 1960s and early 1970s, questions concerning support loads, pipe whipping, and jet impingement began to emerge. These breaks were postulated, and no consideration was given to the mechanism that would cause the failures.

Research by PWR vendors has shown by experimental and analytical evidence that the probability of a complete rupture is very small.^{54,55,56} Efforts of the NRC and NSSS vendors over a period of five to seven years resulted in RG 1.46, which currently defines pipe break criteria (see Chapter 4). As discussed in Chapter 8, it is felt by professionals in the nuclear field that the current criteria are unrealistic and have caused new problems.

The bases for the current procedures for LOCA forcing functions that are used in the dynamic analysis of piping systems are discussed below. Meth-

odologies for computing the piping forces used by various NSSS vendors and the NRC are summarized. As explained in the section on load definitions in Chapter 3, both time-history analysis and simplified methods (e.g., methods based on energy balance and static models) can be used. The bases for the forcing functions for these two approaches are given. These bases are also generally applicable to other pipe failures besides LOCA events.

NSSS Vendors. The methodologies for analyzing LOCA events for Westinghouse, GE, B&W, and C-E are discussed below.

Reference 7 is a state-of-the-art paper that reviews the application of structural mechanics as practiced today in PWR power systems of Westinghouse design. In this paper, the time-history approach for the analysis of the reactor coolant loop is described. As discussed below, simplified methods are also used by Westinghouse for some piping systems. In applying the time-history method, Westinghouse uses a computer code called MULTIFLEX,⁵⁷ which calculates hydraulic transients within the primary coolant system. This program considers coupled fluid-structure interaction by accounting for the deflection of the core support barrel within the reactor pressure vessel. Because there is a difference in pressure across the decompression wave traveling down a ruptured pipe during the thrust transient, there will be different forces at each change in direction or volume in the piping system. These forces are three-dimensional, and thus three-component forcing functions are calculated at many points in the system (Reference 7 states that 15 functions are obtained for each primary loop).

The PSAR for Sundesert Nuclear Plant, whose NSSS vendor is Westinghouse, describes the THRUST computer code, which uses MULTIFLEX results as input. This program calculates the thrust force time histories, which are then used as input to the piping dynamic analysis. The Sundesert PSAR also describes the approach for analyzing BOP piping for the effects of a LOCA. The blowdown force calculation method is similar to the method applied by Moody.⁵⁸ In this approach, a thermal-hydraulic analysis is performed using the method of characteristics to solve simultaneously the continuity and momentum equations derived in a one-dimensional space. The calculated momentum and pressure forces are then applied at each directional change in the piping

system. The effects of line friction are included in the calculation for steady-state blowdown, and provision is made for the fluid transient response. Various modifications are also provided for different flow assumptions.

It is not clear from the discussion in Reference 4 what the difference is between the detailed approach used for BOP piping and the MULTIFLEX code, which is used for the reactor coolant system LOCA analysis. For BOP piping, a simplified blowdown analysis approach is also described.⁴ The description given conforms identically to the requirements of SRP 3.6.2.

Reference 59 is a topical report for LOCA analysis of the Westinghouse primary loop. This report reviews the results of pipe whip and ductile fracture and fatigue tests, which form part of the basis for selection of locations of LOCA events. In a Westinghouse series of pipe whip tests,⁵⁵ it was determined that a failed pipe will not cause failure of a pipe of equal or larger size with the same thickness. In addition, it was determined that pipes whipping in bending will not cause missiles. Battelle Memorial Institute⁶⁰ and GE⁶¹ have performed fracture tests on pipes with flaws. The results confirm that a LOCA is extremely unlikely and that it is more likely that a pipe will leak without crack propagation.⁶¹

Additional bases for determination of pipe rupture locations for the Westinghouse nuclear steam supply system are given in Reference 55. Circumferential cracks are postulated at points of high fatigue or stress. Essentially, Westinghouse complies with the NRC requirements for break locations as discussed in Chapter 4. It is argued that postulating circumferential breaks at points of high fatigue or high axial primary plus secondary stress is a practical equivalent of postulating breaks everywhere. It is shown in Reference 55 that code-allowable shallow cracks in straight free runs will not grow to the burst depth during plant life. Also, it is shown by analysis that code-allowable deep cracks in straight free runs will yield a detectable leak before rupture. The possibility of leak detection before rupture has not been investigated for end-of-run locations.⁵⁵

The procedure for determining pipe break locations in Westinghouse systems is described in Reference 59. The procedure used to determine potential lo-

cations follows the requirements of RG 1.46. The conclusions as to rupture locations are also supported by test results (as described above) and a review of service experience with existing piping systems.

A 1-msec break-opening time is assumed in the analysis of Westinghouse systems. Data from Reference 60 indicate that a guillotine failure will take approximately 15 msec to traverse the circumference. A full double-ended guillotine break area is assumed; however, a break opening of lesser area will be used if justified by analysis, experiments, or considerations of physical restraints.

In a personal communication with an engineer from the piping design section of General Electric, it was learned that simplified dynamic analysis methods are used to analyze piping products and pipe whip restraints in GE systems. The FSAR for WPPSS Nuclear Project No. 2,³ which is a GE BWR system, confirms that this approach is used. In Reference 3, the procedures for applying the energy balance method are described. However, in addition to this approach, the WPPSS FSAR also describes the time-history approach based on Reference 62.

A recent topical report⁶³ replaces Reference 62 as the basis GE is currently using to analyze major reactor coolant pressure boundary piping systems for LOCA inside the containment. Break type and locations for the recirculation, main steam, RCIC/turbine, feedwater, and ECCS piping systems are given. GE's procedures comply with the NRC requirements for break locations as discussed in Chapter 4. The methods for selecting break types for elbows, tees, sweepolets, and straight sections of pipes are given in Reference 63.

The pipe break analysis is performed by GE using computer program PDA.⁶⁴ This program assumes that bending deformation and energy absorption occur at selected locations. The thrust load time histories that are applied at the break locations are based on the method developed by Moody,²³ which includes the effects of unsteady wave propagation and blowdown thrust components for two-phase jet expansion properties. Charts have been prepared that provide the resulting forcing function input to PDA.⁶³

The procedures for defining LOCA loads for Babcock & Wilcox nuclear steam supply systems are given in three topical reports.^{65,66,67} Break locations

are discussed in Reference 65. Essentially, B&W complies with the NRC requirements for break locations as discussed in Chapter 4. Exceptions to the general requirements are justified in the report. It is argued that the reactor coolant pumps are not terminal ends of the cold-leg piping (thus break locations at these locations are not automatically assumed). B&W also feels that it is unduly conservative to impose the $2.0S_m$ stress criterion of RG 1.46 for break locations, but rather argues that a value of $2.4S_m$ for carbon steel is more appropriate. Finally, B&W demonstrates by analysis that split-type longitudinal failures are not credible. Thus, only guillotine breaks are considered. Because of the close proximity of pipe restraints, an opening area equal to the full piping cross-sectional area may not occur. Analyses using nonlinear techniques described in Reference 66 are performed to justify smaller areas where appropriate. An interactive analysis is performed whereby the blowdown areas are computed assuming both longitudinal separation of the pipe and displacement in the radial direction. At each trial, the forcing function is recomputed on the basis of the previous trial area opening time history.

The procedure for developing forces on the piping due to the thrust caused by a LOCA consists of a hydrodynamic analysis of the reactor cooling system during the subcooled decompression phase. This analysis is performed with the CRAFT2 digital computer code.⁶⁸ FORCE2, which is a subroutine of the CRAFT2 code, calculates the hydraulic forces induced in the piping and other components. The computer code JET calculates the impingement loads produced by the fluid emanating from a LOCA rupture.

CRAFT2 is applicable to a variety of thermal hydraulic problems and can accurately predict the pressure fluctuations produced throughout the primary loop due to the complex interaction between expansion and compression waves. As part of the code verification of CRAFT2, the following analyses were performed.⁶⁸

- Analysis of sudden valve closure in a long horizontal pipeline
- Two-volume analysis of a reservoir pipeline system
- Multivolume analysis of a reservoir pipeline system
- Comparison of CRAFT2 and RELAP-4 analyses of a reservoir pipeline system

- Comparison of CRAFT2, MOC, and WHAM solutions of a line packing problem
- CRAFT2 analysis of Semiscale Test 711
- Two-dimensional analysis of Semiscale Test 711

Break-opening times for a full break are assumed to be 10 msec. This is based on empirical data obtained by Battelle Memorial Institute⁶⁹ and the results of dynamic analyses of pipe separation times for circumferential breaks.⁵⁴ For breaks that are limited by pipe supports, a linear ramp opening time that corresponds to the final break area is assumed.

FORCE2, which uses CRAFT2 results as input data, computes the two-dimensional structural reaction forces in the coolant loop on the basis of the solution of the fluid momentum equation. The force reaction components are calculated as a function of time from the fluid pressures, mass flow rate, and densities obtained from the hydrodynamic CRAFT2 solution. Hydraulic forcing functions are provided at each postulated break location and pipe bend for a spectrum of break sizes or for specific sizes determined by iteration between structural and hydraulic analyses. The output of FORCE2 consists of time histories of the resultant forces, x and y components, and force directions for up to 60 locations.

The JET computer program is used to determine the effects of LOCA-induced fluid jets impinging on piping and other nuclear steam supply system-related targets. JET uses fluid jet intensity data generated from the CRAFT2 program using the FORCE2 subroutine input model. The JET program combines the jet intensity and expansion characteristics with the target geometry input data to calculate the transient impingement force. For a saturated discharge, the Moody expansion model²³ is used. Modifications are made for disc and partial jets caused by limited break openings.

Procedures used by Combustion Engineering to perform LOCA analyses are given in References 70 and 71. Pipe break locations are postulated at pipe ends and at two of the three elbows in each cold leg of the primary piping. The approach used to determine the pipe break locations follows the requirements of RG 1.46. No intermediate break locations are postulated for the very short hot leg of the primary piping that has stress and usage factors below criteria values. The discharge legs exhibit stress characteristics similar

to the hot leg. Since they are also short in length, no intermediate break locations are considered. Circumferential breaks would occur first at the pipe ends, and longitudinal breaks would occur first at the elbows.

Pressure transients due to LOCA effects are determined using the blowdown program CEFLASH4A. In the analysis of the reactor coolant piping for circumferential failure, a conservative, simplified approach is used. Because of the large break sizes relative to the volume of the C-E system, the pressure decreases very rapidly during blowdown, and the combined effect of the pressure decrease and the slow momentum increase results in net bend forces that are less than the initial conditions. Thus it is conservative to neglect bend forces and to apply an instantaneous constant compression force at the break end of the pipe. An example analysis with and without bend forces was performed to demonstrate that this simplified procedure is conservative.⁷¹

The flow area characteristics for a circumferential break are determined by structural dynamic analysis methods. A nonlinear analysis of a model that includes pipe stops is performed to determine break opening areas and times. Although the results are of little importance to the piping analysis (since the simplified approach described above is used), the results are of importance to containment pressurization and effects on the reactor pressure vessel. Longitudinal break characteristics (i.e., area, break shape, and opening time) are also determined by dynamic finite-element analysis techniques. Detailed finite-element stress analyses are also performed for each postulated elbow break location in the cold legs to verify that a longitudinal break is significantly more likely than a circumferential break.

Nuclear Regulatory Commission. The NRC, under the management of the Analysis Development Branch, develops reactor coolant system codes. The purpose of the program is to provide better digital computer codes for computing the behavior of reactor systems for postulated accident conditions. In addition to providing a rational approach for obtaining blowdown forces, a primary goal is to predict fuel rod surface temperatures that could lead to cladding damage and radioactive material release. Existing codes developed by the NRC⁵⁰ are based on the assumption that liquid and vapor (i.e., two-phase steam-water mixtures) are in thermal and mechanical equilibrium (i.e., tem-

peratures are equal and both flow with the same velocity). These assumptions allow the thermal-hydraulic behavior to be represented by equations that conserve the mass, momentum, and energy of the assumed homogeneous mixture. Equations of steam and the constitutive equations for wall shear and wall heat transfer are also included in the analysis formulation. The numerical solution of these equations is based on the lumped-parameter approximation, in which the system is represented by a collection of control volumes that are interconnected. Current codes are one-dimensional.

The RELAP-4/MOD5 code, which was developed for the NRC by the Idaho National Engineering Laboratory (INEL) and released in 1976, is applicable to the LOCA blowdown regime only (as compared to reactor pressure vessel refill and flooding stages of the emergency core cooling process). However, this is the time period during the LOCA phenomenon in which severe forces are imposed on the piping systems and other safety-related components. Other modifications to RELAP currently are being made to extend the capability to the reflood regime. A user-convenient version of RELAP-4/MOD5 has been assembled and released as code package WRAP (Water Reactor Analysis Package), which is applicable to both PWR and BWR plants.

Advanced reactor codes including TRAC, THOR, and RELAP-5 are currently being developed by the NRC. The ultimate goal in developing advanced system analysis codes is to model, in the most realistic and feasible manner, the thermodynamic and flow processes of the coolant during blowdown or other transients, in all flow regimes. Numerous special-purpose component codes (as compared to the system code discussed above) are also being developed.⁵⁰

A limited amount of independent assessment has been performed for the RELAP-4/MOD5 computer code. The NRC is sponsoring experimental research at the Loss of Fluid Test (LOFT) facility and at the Semiscale facility, both located at INEL. The LOFT facility is a 55-MWe PWR developed to simulate the thermal-hydraulic behavior of 1,000-MWe PWRs in LOCA experiments. The smaller Semiscale facility is a scale model of LOFT (one-thirtieth the power) used to demonstrate how well scaling procedures have been developed. The data generated by the Semiscale tests are being used to assess analytical procedures for computing the blowdown phenomena (e.g., RELAP-4/MOD5).

Five nonnuclear tests have been conducted in the LOFT facility, and a limited amount of independent assessment has been performed for the RELAP-4/MOD 5 code. More independent assessment will be conducted in the future.⁵⁰

Other LOCA Analysis Work. Reference 72 presents a method of coupling the thermal-hydraulic computer code to the structural dynamic computer codes. The system analysis codes are not specifically organized for the development of piping system forces (e.g., RELAP-4). A new geometry descriptor termed the force equivalent area (FEA) has been developed that couples the system program to a structural dynamic computer code. This approach was originally developed as a postanalysis method for use with RELAP-3 and also has been used for RELAP-4. In the future, it will also be easily adapted to the various new system codes. This will enable the structural designer to benefit from the latest developments in fluid thermal-hydraulic systems programs.

Other experimental work is being performed in foreign countries to verify system codes. One code, DEPCO-MULTI (subcooled Decompression Process in a Loss-of-Coolant Accident Multiple Pipe Network), which is a one-dimensional system code based on conservation of mass, momentum, and energy, has been developed and compared to experimental results from the ROSA (Japan) and the LOFT semiscale facilities. It has been found that the agreement between the calculated and measured pressure time histories is generally good.⁷³

The application of RELAP to the case of a rapid pressure transient (e.g., BWR valve closure transient) can produce unconservative results.⁷⁴ Because RELAP is a finite-difference code, artificial viscosity effects become dominant and tend to damp down the pressure surge unless the control volumes are smaller than a critical size. The authors of Reference 74 propose an analytical procedure based on the method of characteristics to eliminate this problem. The computer program MCHAP, based on this approach, produced results substantially more conservative. RELAP, MCHAP, and experimental data were compared for steam dome pressure transient after turbine valve closure for the Brown's Ferry and Peach Bottom nuclear plants.⁷⁴ MCHAP's results were closer to the experimental data than were RELAP's.

Seismic Loads

The calculation of seismic loads for nuclear power plant piping systems begins with the specification of seismic hazard at the plant site, which includes estimation of the size and the likelihood of future events. This process involves reviewing the historical record and converting the information to a free-field input consisting of a peak ground acceleration value (different values for OBE and SSE and for vertical and horizontal components) plus a response spectrum shape (different for vertical and horizontal components). A soil-structure interaction analysis is performed, and various detailed building models are then analyzed to develop input at the piping supports. Either floor response spectra or time histories are then input to the piping systems to obtain piping component seismic stresses. Reference 75 gives a review and evaluation of analytical and experimental techniques for seismic design of nuclear power plants. Included in Chapter 4 of Reference 75 are the data bases for current analysis methodologies for the following areas:

- Estimating free-field responses to earthquakes
- Soil-structure interaction
- Reactor building response

Reference 75 includes a comprehensive discussion of each of these subjects and cites pertinent references, which form the basis for reactor building analysis. Recent state-of-the-art reviews for these areas are presented in twelve documents prepared by the LLL/NRC Division of Operating Reactors Seismic Conservatism Program, which are currently in press.⁷⁶⁻⁸⁷

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7. DEVELOPMENT OF NEW PIPING LOAD DATA

This section presents summaries of research conducted by the NRC, industry, national laboratories, and foreign organizations related to nuclear power plant piping and loads. Some of the research identified (e.g., pipe rupture studies) is not specifically directed to improving the definition of loads for nuclear power plant piping. Nonetheless, piping research that is not specifically directed to the piping loads is summarized below if the research may cause a redefinition of existing design basis loads. In this sense, fracture mechanics studies of pipe rupture are included since they may eventually form the basis for an improved definition of design basis accidents (e.g., LOCA) even though fracture mechanics studies, in themselves, do not specifically assess piping loads.

The largest single source for new piping load data is the NRC, since the concerns of this regulatory agency lead to design basis events and provide, directly or indirectly, the impetus for piping research. The intention of the NRC is to only support research required to provide the NRC with suitable bases for evaluating nuclear facility applications or establishing regulations and policies. The NRC, as the responsible regulatory agency, identifies and establishes potential piping problems and then monitors research work performed by A/E firms, NSSS vendors, and nuclear power plant owners necessary to ensure that each problem identified has been adequately resolved.

Since the impact of the NRC on the definitions of loads and on the research for new load data is so significant, the first two subsections of this chapter review the NRC's confirmatory research programs related to piping. Other subsections describe research being conducted by industry, national laboratories, and foreign countries, much of which is directly or indirectly performed for the NRC. A notable void exists for research related to seismic loads on piping. While a significant effort exists today to improve the understanding of seismic effects in nuclear power plants, the effort is not aimed primarily at piping and therefore is not included here. Nonetheless, current research to improve soil-structure interaction models, structure response, floor response spectra, etc., will have a direct effect on the seismic loads for nuclear power plant piping.

NRC Confirmatory Research

Prior to the establishment of the NRC in January 1975, all activities related to nuclear reactors, including both development and regulation, were the responsibility of the Atomic Energy Commission (AEC). In May 1973, safety-related research in the AEC was made the responsibility of a new Division of Reactor Safety Research (DRSR). When the NRC was established in 1975, the functions of the DRSR were transferred, essentially without change, to the NRC office of Nuclear Regulatory Research (RES). Today, much of the current research effort by the RES is a continuation of work begun under the AEC DRSR.

In accordance with the act of Congress establishing the NRC, the RES is responsible for:

- developing recommendations for research to be performed by the NRC
- engaging in or contracting for research that the NRC deems necessary

In spite of this broad mandate, the RES has limited its activities primarily to confirmatory research, as distinguished from the developmental research carried out by industry or nonregulatory governmental agencies (e.g., the DOE). Confirmatory research has been considered to be that necessary to provide the NRC with suitable bases for evaluating applications or for establishing regulations or policies.

The principal areas of confirmatory safety research of the NRC are listed in Table 2-15. A description of current and planned research for each of the principal areas is given in NUREG-0006, *Water Reactor Safety Research Program*.⁵⁰ Not all of the research areas apply to or will have an impact on piping and piping loads. Those areas of research that have the potential for influencing piping loads are identified as follows.

Systems Engineering Branch. The LOCA is one of the principal accidents studied by this branch of the NRC, with particular emphasis on the ECCS. The largest cost item in this program is the LOFT facility, which, with the Semiscale facility, is located at INEL. The Systems Engineering Branch also has programs related to BWR plants, which include a 1/5-scale BWR suppression pool test program, conducted at LLL.

Analysis Development Branch. The analysis development program is designed to provide better digital computer codes. Many of the codes are being developed in a flexible, modular fashion to be easily applicable to postulated system disturbances in addition to LOCA, such as ATWS and reactivity-initiated accidents. Improved modeling of complex phenomena such as LOCA will provide an improved understanding of the forces that act on reactor vessels, supports, and associated piping.

Metallurgy and Materials Research Branch. This branch is primarily concerned with the integrity of the primary-system pressure boundary, which includes primary piping up to 4 in. in thickness. As such, the program consists of three subelements: (a) fracture mechanics, (b) operational effects, (c) flaw detection. In general, the primary-system integrity subelements are geared to produce information including crack propagation and arrest, steam-line break, thermal shock, and pneumatic loading cyclic crack growth. Ultimately, the results will be incorporated into improved industry code rules and standards for improved LWR safety designs.

NRC Generic and Unresolved Safety Issues

As a result of congressional action in 1977, the NRC budget was amended to include the new Section 210, which specified that the NRC would develop a plan providing for specification and analysis of unresolved safety issues relating to nuclear reactors and should take such action as might be necessary to implement corrective measures. The plan submitted to Congress on January 1, 1978, pursuant to Section 210, is published as NUREG-410, *NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants*,³⁹ and contains lists of 133 generic issues of varying significance. The issues were placed in one of four categories. Those issues judged to warrant priority attention (i.e., Category A) are given in Table 2-16. Table 2-16 also designates 17 safety issues from among the Category A generic issues identified in a subsequent NRC publication NUREG-510, *Identification of Unresolved Safety Issues Relating to Nuclear Power Plants*.¹⁰ The generic issues, particularly the unresolved safety issues, are of the greatest concern to the NRC and require action to be taken by the NRC in accordance with Section 210. NUREG-0371, *Approved Task Action Plans for Category A Generic Activities*,⁸⁸ provides a description of Category A tasks and the work scheduled to be performed.

Category A tasks given in Table 2-16 that relate to piping (and potentially to piping loads) are given below along with a brief description of each task.

Water Hammer (A-1). Studies on water hammer (and steam hammer) by the Division of Systems Safety (DSS) of the NRC-RES are undertaken to ensure that water hammer is given appropriate consideration in construction permit (CP) and operating license (OL) licensing reviews and to obtain more information on the potential water hammer phenomena through systematic investigations. A recent NRC publication, NUREG-0582, *Water Hammer in Nuclear Power Plants*,²⁰ contains the results of a review of reported water hammer events and is the first of several reports scheduled to be issued by the NRC on water hammer. This report identifies technical studies currently under way that include existing computer programs to calculate piping system response to dynamic water hammer loads. Further, this report recommends that additional reviews be made to determine the adequacy of current methods for calculating dynamic loads on valves and associated piping and the need for any confirmatory experiments.

Asymmetric Blowdown Loads on the Reactor Coolant System (A-2). In the unlikely event of primary coolant piping rupture, large nonuniformly distributed loads would be imposed upon the reactor vessel, internals, and other components (e.g., piping) in the reactor coolant systems. As part of this task, the NRC has reviewed computer models and codes developed by reactor vendors to calculate asymmetric blowdown loadings for PWRs. In addition, the NRC has been investigating this phenomenon as it applies to BWRs and is expanding Task A-2 to resolve this issue.

BWR Mark I and Mark II Pressure Suppression Containments (A-6, A-7, A-8, A-39).

In performing large-scale testing of a Mark III containment, and during in-plant testing of Mark I containments, new suppression pool hydrodynamic loads were identified which had not been explicitly included in the original design basis for Mark I or Mark II BWRs. These additional loads result from dynamic effects of dry well air and steam being rapidly forced into the suppression pool (torus) during a postulated LOCA and from suppression pool response to various modes of SRV operation generally associated with plant transient operating conditions.

The affected utilities (Table 2-17 lists Mark I BWR owners and Table 2-18 lists Mark II BWR owners) formed ad hoc Mark I and Mark II owners groups and each engaged GE as its program manager. Both owners groups developed two-phase programs (STP and LTP) for resolution of their concerns about pool dynamic loads for their respective containment designs. The owners groups' programs include a number of comprehensive experimental and analytical programs to establish generic pool dynamic loads, load combinations, and design criteria.

The NRC has identified and initiated a number of tasks to review and evaluate the results of the Mark I and Mark II owners groups' programs (i.e., Tasks A-6, A-7, A-8, and A-39). The Mark I STP (i.e., Task A-6) is complete, and the NRC conclusions are documented in NUREG-0408, *Mark I Containment Short Term Program Safety Evaluation Report*.¹⁶ The objectives of the Mark I LTP include establishing design basis loads that are appropriate for the anticipated life of each Mark I facility. Documentation by the NRC including final load definitions is currently being developed and should be available by early 1980. The objective of the NRC's efforts related to the Mark II STP includes the review and evaluation of pool dynamic load associated with a postulated LOCA. NUREG-0487, *Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria*,¹⁷ documents the Mark II STP and provides an evaluation of the Mark II owner's methodologies for establishing LOCA and SRV loads. The evaluation of the loads by the NRC and consultants from the Brookhaven National Laboratory (BNL) made use of experimental data obtained from (1) Marviken Tower Station Tests,⁸⁹ (2) small-scale tests conducted at the Massachusetts Institute of Technology, and (3) small-scale tests conducted at the University of California. With regard to the Mark II LTP, the NRC is evaluating the results of confirmatory experimental and analytical programs to assess the margin of safety for selected loads. The NRC reviews and evaluates the results of the Mark I and Mark II owners groups' experimental and analytical programs to establish and justify the SRV-related dynamic loads for Mark I and Mark II containment designs. Documentation of the results should be available in 1980.

ATWS (A-9). In September 1973, the AEC published WASH-1270,¹¹ which contained acceptance criteria to protect against ATWS. Over the next two years, the four NSSS manufacturers submitted analyses and information on ATWS events, which were reviewed by the NRC; subsequently, status reports were released by the

NRC. The NSSS vendors and owners questioned whether the NRC requirements were necessary, which led to the issuance of the three-volume report, NUREG-0460, *Anticipated Transients Without Scram*,^{12,13,14} for LWRs. NUREG-0460 concluded that current reactor scram system reliability has not been demonstrated and recommended that means of mitigating ATWS consequences be provided in plant design. Currently, work sponsored by the NRC and NSSS vendors is under way to quantify the pressure and temperature transients of ATWS events and to evaluate plant safety with respect to these events.

Pipe Rupture Design Criteria (A-18). Current criteria for postulating pipe breaks have evolved over a long period of time, lack consistency when applied inside and outside the containment, and are subject to misinterpretation in some areas. Although these criteria may be improved, the NRC feels that adequate protection is currently provided, and this task is not designated as an unresolved safety issue. NUREG-0371⁸⁸ describes planned research for this task.

Seismic Design Criteria (A-40). The NRC is conducting Generic Task A-40 as a summary of short-term efforts to support the reevaluation of the seismic design of operating reactors, particularly older reactors designed prior to current seismic standards. The objective of this task is to investigate selected areas of the seismic design sequence to determine their conservatisms for all types of sites, to investigate alternative approaches to parts of the design sequence, and to quantify overall conservatisms. Long-term efforts include improving the state-of-the-art methodology for defining near-field ground motion, which is scheduled for completion in 1981.

Industry

General Electric Co. GE is the program manager for the BWR Mark I and II owners groups' studies of the hydrodynamic loads associated with LOCA and SRV events (Tables 2-17 and 2-18 list, respectively, Mark I and Mark II nuclear power plants in the United States). As such, GE has continuing involvement with research projects that will provide additional data on piping loads.

Results of hydrodynamic load research projects are contained in a large number of GE topical reports (many of which are proprietary) that form the basis for quantifying Mark I and Mark II loads. Test programs related to the BWR Mark I and Mark II containment design are as follows:

- Pressure suppression tests at the 4T test facility consisted of approximately 60 blowdowns through a single downcomer into a vessel simulating a prototypical wet well.^{90,91} In addition to pool swell hydrodynamic loads, data were obtained for lateral loads on downcomers and pressure fluctuation loads on the submerged wet well during steam condensation and chugging. The data from these tests form the basis for qualification of the pool swell analytical model and submerged wet well pressure loads during steam condensation.⁹²
- Pressure suppression test facility (PSTF) tests emphasized the examination of the effects of pool impact on structures located above the suppression pool.⁹³
- Tests at the Monticello Mark I BWR plant were conducted with the objective of providing the necessary data base for pressures and temperatures in the torus and SRV piping associated with SRV actuation. The report *Final Report: In-Plant Safety/Relief Valve Discharge Load Test-Monticello Plant*⁴⁰ contains a summary of the test results.

Since 1965, GE has conducted pipe rupture research studies for the NRC (previously for the AEC). These studies have been designed to extend the understanding of the failure-causing mechanism and to provide improved capability for evaluating reactor piping systems to minimize the probability of failures. Progress reports (e.g., *Reactor Primary Coolant System Pipe Rupture Study Progress Report No. 40*⁶¹) summarize accomplishments in this broad program to study basic fatigue crack growth in LWR primary piping materials and to study weld sensitization in Type-304 stainless steel. GE also prepares topical reports pertinent to NRC licensing reviews. The GE licensing topical report NEDO-23649, *Application of Pipe Break Criteria for Major Piping Systems Inside Containment for BWR/6 218, 238 and 251 Mark III Product Line Plants*,⁶³ is an example of such reports, which, in this case, describes the criteria for pipe break and the application thereof.

Combustion Engineering, Inc. C-E is an NSSS manufacturer of PWR plants and has conducted safety-related research to establish mechanistic pipe rupture criteria based on the realistic behavior of pipes, components, supports, and stops. This research has also examined the nature of flaws inherent in primary piping systems and the manner in which they can enlarge and threaten

system integrity. Future C-E work on hypothetical pipe breaks, as stated in *Safety-Related Research and Development for C-E Pressurized Water Reactors*,⁹⁵ includes the establishment of possible causal relationships for pipe breaks and predictions of the resulting consequences, such as crack opening time and area. Research work by C-E supports NRC programs by providing realistic inputs or by reviewing proposed new pipe break criteria.⁹⁶

Battelle Columbus Laboratories. Research related to the failure of cold-leg piping in PWR plants was conducted by Battelle Columbus Laboratories for the NRC. These research studies examined the growth of small defects in the weld of base material due to repeated stress cycles. In a report prepared for the NRC, NUREG/CR-0325, *Relevance of Fatigue Tests to Cold Leg Piping*,⁹⁷ researchers at Battelle examined fatigue test data on piping products to determine the margin of safety of the present ASME Code methods, to determine leak-rupture trends, and to determine Implications of fatigue tests for PWR cold-leg piping.

Science Applications Incorporated. Science Applications Incorporated is currently conducting a two-year study of failure of LWR piping systems for the NRC. Two objectives have been identified for this program. The first objective is to determine failure mechanisms and to compute the probability of pipe failure. The reliability of typical piping systems both inside and outside containment will be calculated and used to develop rational criteria for determining potential locations of piping failures. The ultimate use of the research findings will be to modify the applicable regulatory guides and standard review plan sections that specify piping failure locations.

The second objective is to examine the likelihood of a LOCA and a simultaneous earthquake. Currently, the NRC requires that the effects of these phenomena be combined by the square-root-of-the-sum-of-the-squares (SRSS) method. The goal of the study is to show that the probability of an earthquake causing a large LOCA is low enough that the primary system of a PWR plant should not be designed for the simultaneous occurrence of both events. A deterministic fracture mechanics approach is being coupled with a stochastic representation of all loads (both normal and accident), initial crack size, and material properties. The effect of leak detection activities also is being included in the probabilistic analysis. Crucial to the results is whether the crack size re-

quired to produce a 5-gal/min flow rate (which is the level at which a plant is shut down) is stable during severe seismic loading conditions.

The piping loads project was started in July 1979, and a preliminary report on the large LOCA problem for one nuclear plant will be completed by June 1980. The entire study is scheduled to be finished by July 1981.

Electric Power Research Institute. The Electric Power Research Institute (EPRI) has provided research studies related to piping and piping loads. As part of the Mark II hydrodynamic load studies (see section on GE), EPRI performed 1/13-scale tests.⁹⁸ Continuing research at EPRI related to piping loads is outlined in the EPRI report PS-331-SR, *Research and Development Program Plan for 1979-1983*.⁹⁹ The 1979-1983 research program for the Nuclear Power Division of EPRI places a major emphasis on the assurance of safety and reliability of LWR facilities. Of high priority is to gain a better knowledge of the behavior of structural materials used for piping.

Within the framework of the Nuclear Power Division of EPRI, four major programs are currently under way, of which the Water Reactor System Technology Program is the first. This program is being funded through 1983 at a rate of approximately \$20 million per year. The Water Reactor System Technology Program is divided into five subprograms:

- Loss-of-Coolant Accident Emergency Core Cooling Systems (LOCA-ECCS)
- LWR System Behavior
- Structural Integrity
- Probabilistic Safety Analysis
- Primary and Secondary Pressure Boundary

The objective of the LOCA-ECCS subprogram is to establish the capability to predict LWR behavior under postulated LOCA conditions through the development and application of analytical methods, empirical test data, and verified computer codes. Elements of this subprogram include improved descriptions of the LOCA event and associated hydrodynamic loads.

The objective of the LWR System Behavior subprogram is to develop system and component performance analysis methods that provide realistic predictions of LWR behavior under operational, off-normal, and accident (non-LOCA) transient conditions.

The Structural Integrity subprogram's objective includes developing synthesized methods for design by integrating the results from other related areas (soils, seismology, etc.). This subprogram contains seismic-related research on soil-structure interaction and structural and component response.

The objective of the Probabilistic Safety Analysis subprogram is to quantify the risk and consequences associated with nuclear power reactors and to develop methods and models to predict operational behavior. A main element of this subprogram is to quantify the consequences associated with ATWS.

The objectives of the Primary and Secondary Pressure Boundary subprogram are to insure the continued safe operation and integrity of the nuclear pressure boundary through research projects that include improved characterizations of material properties, material response, and the application of improved analytical techniques to present engineering problems.

National Laboratories

The DOE national laboratories are major contributors to piping-related research. These laboratories include: Argonne National Laboratory, which has been involved with fatigue behavior in piping materials; BNL, which has been heavily involved in such matters as pool swell and SRV loads for BWR plants; Hanford Engineering Development Laboratory (HEDL); INEL, which is the site of the LOFT and Semiscale test facilities; LLL, which is conducting the Seismic Safety Margin Research Program (SSMRP) to assess LWR facility safety; Oak Ridge National Laboratory (ORNL); and Sandia Laboratories.

Since 1967, ORNL has been providing information for the development of piping design criteria and piping stress indexes to be used in the analysis of piping components for nuclear power plants. From January 1967 through June 1975, this program was sponsored by ERDA. In June 1975, the NRC DRSR assumed sponsorship for the program. Results to date include a survey of the piping manu-

facturing industry products. A review of the technical literature and pertinent engineering data has been conducted, and theoretical and experimental analysis of nuclear-piping components has been performed. A paper titled "Summary and Accomplishments of the ORNL Program for Nuclear Piping Design Criteria"¹⁰⁰ describes, documents, and reports published research.

Sandia Laboratories of Albuquerque, New Mexico, is conducting an LWR safety research program for the NRC. A quarterly report for October-December 1978¹⁰¹ contains a description of five areas where research is presently being conducted. One of the research projects is the study of two-phase jet loads. The purpose of this study is to develop an improved approximate engineering model to characterize two-phase jets emanating from circumferential or longitudinal breaks in a typical PWR piping system. This model is proposed as a replacement for the current Moody model, which assumes asymptotic jet expansion and thermodynamic equilibrium.

Foreign Countries

Research conducted outside of the United States related to piping loads is summarized by country as follows

Germany

GKM test facility

Lateral loads were investigated on the ends of downcomers due to chugging loads associated with the suppression chamber of a BWR.¹⁰² Investigations were made of pressure loads experienced by the suppression chamber boundaries during the steam blowdown phase of a LOCA event.¹⁰³

KWU-Karlstein

Downcomer lateral loads and submerged boundary pressure loads during chugging in a multivent configuration were studied.¹⁰⁴ Studies of the effects on SRV loads in a Mark I BWR suppression chamber, equipped with KWU-designed T-quencher devices, were conducted at Karlstein, Germany.

KWU-Erlangen

Two-phase jet impingement studies were performed by KWU at Erlangen, Germany. Final results appear in reports NRC-477¹⁰⁵ and NRC-478.¹⁰⁶

Battelle Institute/
Frankfurt

Research using a 1/64-volumetric-scale blowdown facility based on the 1200-MW BIBLIS PWR is being conducted by Battelle Institute of Frankfurt, Germany.¹⁰⁷

HDR

Blowdown tests are being performed in Germany at the Heissdampfreaktor (HDR), the superheated steam test facility.

Great Britain

Berkeley Nuclear
Laboratories

Burst tests of nuclear piping have been performed by the Central Electricity Generating Board at the Berkeley Nuclear Laboratories in Great Britain. Experimental programs have investigated growth of defects directed at examining current assumptions for pipe rupture.¹⁰⁸

France

CEA/Framatome

Through an agreement between the Commissariat à l'Energie Atomique (CEA) and Framatome of Courbevoie, France, an experimental program at the Cadarache Center EDITH facility has been conducted to study the mechanical behavior of PWR piping subjected to longitudinal and guillotine pipe rupture breaks.¹⁰⁹ Additional studies of fatigue crack growth have also been performed jointly by CEA and Framatome under the Aquitaine 1 program.¹¹⁰

Italy

University of Pisa/
CNEN

Experimental research, directed at forecasting the fracture behavior of pipes, was performed jointly by the Istituto di Impianti Nucleari of the University of Pisa in cooperation with the Comitato Nazionale per l'Energia Nucleare (CNEN) of Italy. The experimental program involved studies of crack initiation in through-thickness cracked pipes.¹¹¹

Caorso

In-plant testing at the Caorso Nuclear Power Plant in Italy was performed to verify performance of GE-designed quenchers during SRV discharges in a Mark II BWR.¹¹²

Japan

Tokai-2

In-plant testing at the Tokai-2 Nuclear Power Plant in Japan was performed to examine GE-designed quencher performance and resulting containment loads due to SRV discharge in a Mark II BWR suppression chamber.

Mitsubishi

Experimental studies of pipe reaction forces and jet impingement loads due to pipe break accident conditions were conducted by MAPI Engineering Center, Mitsubishi Atomic Power Industries, and the Takasago Technical Institute, Mitsubishi Heavy Industries.¹¹³

Sweden

Marviken

A large international project at the Marviken Power Station is being conducted in which tests are being performed to investigate critical flow in large pipes and to examine suppression chamber blowdown loads and resulting pressures.⁸⁹

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8. BENEFITS OF DEVELOPING NEW LOAD DEFINITIONS AND COMBINATION PROCEDURES

Benefits of developing new definitions for LOCA, seismic, and water hammer loads and for load combinations are given in the following text. These loads are particularly important to the safety of the reactor coolant systems in LWR plants. As discussed in Chapter 4, the NRC has recommended alternative plant modifications to reduce the potential effects of ATWS, rather than to develop new procedures for predicting loads. Also, extensive programs for Mark I and II containment systems are being conducted by owners groups to develop new load definitions. Those programs are discussed in Chapter 6 and involve data which are generally proprietary in nature. Until these programs are completed and the results studied, it is not possible to speculate on the feasibility and benefits of developing new load definitions. Benefits from new load definitions may be possible for other load types; however, indications of a need for improving the other loads were not found in the literature.

LOCA Loads

A recent report identifies and assesses the basis for the design assumptions for LOCA loads.⁵³ This report lists the background, design approaches, current impact, recommendations, and resulting anticipated benefits for the various design steps. The following recommendations and anticipated benefits regarding LOCA loads were made. The last subsection discusses general information concerning LOCA loads from other references.

Postulated Break Opening Time. An analytical and testing program should be conducted to better define realistic crack propagation times. This would lead to a reduction in LOCA forces and associated requirements for pipe whip restraints. Two goals are recommended:

- Develop a new enveloping break opening time
- Develop a break opening time that is a function of the various participating factors

The benefits of developing a more rational break opening time are given below:

- Eliminate need for utilities to individually obtain approval of alternative break opening times
- Eliminate or minimize the number of fracture mechanics analyses required to be performed
- Produce lower LOCA loads, fewer design problems, and less restraint material
- Allow less lead time for fabrication and installation of restraints
- Reduce in-service inspection interference
- Lower compartment pressurization (due to less flow blockage of venting paths) if a LOCA occurs
- Reduce maintenance time for equipment blocked by pipe whip restraints

In addition to research work to better define break opening times, additional analytical and experimental studies should be conducted to determine realistic break opening configurations. Many of the above benefits would also apply.

Blowdown Load Modeling Techniques. Analytical results from thermal-hydraulic codes (e.g., RELAP) should be compared with experimental results. Blowdown parameters that should be investigated include the following items:

- Time to achieve maximum (peak) thrust and time to reach steady-state blowdown
- Maximum thrust loads for typical high-energy piping systems
- Discharge flow rate (velocity), pressure, and temperature transients for various steam quality regimes

The benefits of the results from this research would include insight into the amount of conservatism inherent in analyses currently being conducted with the present thermal-hydraulic computer codes.

Jet Characteristics. Tests or additional investigations (i.e., analysis or evaluations of existing test data) should be performed to predict forces in

steam, steam-water, and flashing liquid jets. The characteristics of free stream jets should be studied to better define pressure distributions along the centerline of the jet axis and in the direction normal to the centerline of the jet. Temperature data should also be obtained at the same locations where pressure data are measured. Jet characteristics for longitudinal split and circumferential rupture (both full rupture and limited-displacement rupture) should be included as part of the research.

The benefits of this research would be to reduce design loads and to decrease corrective modifications to resist the effects of jet impingement. These results would have a twofold effect. First, modification construction costs would decrease; second, it would be easier to inspect piping systems during operations.

General. A recent paper expresses the need for new design basis pipe break criteria.⁵² The author states that present criteria, which define the location, type, and size of breaks, are not based on realistic mechanisms that lead to failure. Originally, LOCA requirements were developed for the containment building analysis. Later, pipe break requirements were extended to the piping systems themselves in order to demonstrate that a plant could be safely shut down following a LOCA. Finally, LOCA and maximum seismic effects are now arbitrarily combined, which further increases the provisions required to protect safety-related piping systems. All these requirements lead to stiff systems that do not perform as well for normal temperature-induced loads, and the increase in protective hardware makes in-service inspection more difficult.

Studies cited in Reference 52 conclude, from analytical and experimental results, that the probability of a complete rupture of a primary pipe is highly unlikely. Both axial and circumferential cracks must be longer than half the pipe diameter before rapid propagation can occur. In addition, the cracks that cause leaks are many times smaller than critical cracks. Thus, there is a significant margin between leak and break.

Rational break criteria should parallel the usual design process, where actual conditions are determined and conservative safety margins provided to protect

against failure. Using this approach, the effects of small cracks, which represent flaws many times larger than those that could occur, should be determined. In order to develop this approach, the following research should be accomplished.⁵²

- Improvement in the detection of the size and nature of flaws in piping and piping welds
- Better definition of fracture characteristics of piping material in both ductile and brittle conditions
- Evaluation of consequences of all loading conditions on the opening and propagation of cracks, including the effects of cracked pipes, seismic loads, overpressurization, mechanical loads, and thermal shock

In regard to the last topic, Reference 114 presents the results of an analysis in which PWR primary piping with an assumed circumferential crack, one-half circumference long, is subjected to an SSE transient applied simultaneously with the initiation of the crack. The results indicated that the crack does not extend significantly and that the opening areas are small fractions of the pipe flow area. These results have not been verified experimentally, however.

Seismic Safety of LWR Nuclear Power Plants

A recent study, performed for Sandia Laboratories, Albuquerque, New Mexico, recommended research for improving the seismic safety of LWR nuclear power plants, primarily on the basis of the results of a questionnaire returned by 55 persons working in seismic safety of nuclear power plants.¹¹⁵ The recommendations were also based on ideas expressed at a meeting of an ad hoc group of professionals, on a review of literature, on current research programs, and on engineering judgment. The following responses in regard to piping loads were made by respondents to the first part of the questionnaire, which consisted of four general requests or questions.

1. Areas of Needed Research to Increase Seismic Safety of LWR Systems:
 - Development of techniques for different response spectrum inputs at different supports
 - Development of earthquake cycle amplitude distributions for crack propagation analysis

- Rational evaluation of loads such as LOCA in combination with seismic loads and of the procedures used to combine loads (e.g., use of peak loads)
2. Responses Identifying the Single Most Important Problem in the Seismic Design of Nuclear Power Plants:
 - Conservatism due to current procedures used in dynamic analysis of reactor buildings, leading to time histories and response spectra used in piping system design
 - Loads induced by excessive number of piping restraints and failure of snubbers during normal operation
 3. Areas Where Excess Conservatism Exists:
 - Methods for generating floor response spectra
 - Procedure for enveloping response spectra due to multiple-component earthquakes
 - Enveloping response spectrum input to be used in the analysis of multiple-support piping systems
 - Unrealistic loads that are combined with earthquakes, especially LOCA
 - Procedures used to combine loads, such as using peak values where statistical independence is ignored
 - Unrealistic broadening of response spectrum peaks
 - Use of uncoupled analyses of buildings, piping systems, and equipment
 4. Promising New Design Concepts for Increasing Seismic Safety:
 - Probabilistic methods for developing load combinations
 - Multiple independent support motions and multi-spectrum analysis techniques
 - New procedures for analyzing buildings for SSE input

Based on the second part of the questionnaire, which involved rating research topics for various attributes, and on other sources as discussed above, recommendations were made for future research on the following topics, which are closely related to piping loads.

- Experimental verification of piping behavior to validate analytical methods
- Development of piping models to accommodate uncorrelated multiple support input
- Dynamic interaction of small and large piping systems
- Procedures for obtaining design spectra from raw input spectra

On the basis of a review of seismic analysis methods for piping and equipment, problems with current procedures for developing piping system loads were identified in another report.¹¹⁶ A general criticism of the present approach for developing loads by the author of Reference 116 is that conservatism is compounded due to the high factors of safety independently assigned to each of the analysis steps. Because of the high factors for extreme load conditions, such as seismic and pipe whip, the total reliability is actually less than what it would be with less conservative standards because of the provisions made for extreme loads, where flexibility for thermal expansion is jeopardized. The author of Reference 116 cites the need for research for the following topics related to piping loads:

- Procedures for providing consistent criteria for the time-history and response spectrum methods
- Present methods for dealing with parameter uncertainty
- Reevaluation of the arbitrary definitions of primary and secondary stresses in light of multiple input motions where piping has more than one source of excitation
- Development of probabilistic procedures for combining extreme dynamic loadings caused by different phenomena

Water Hammer

Water hammer has been identified as a major concern and incorporated into Task Action Plan A-1, which is described in NUREG-0371.⁷⁶ The purpose of the task action plan is to develop systematic review procedures to ensure that water hammer is given appropriate consideration in licensing and operating reactor reviews. There is also a need to provide guidance

and methods for understanding and resolving water hammer problems in existing plants. As discussed in Chapter 4, water hammer occurrences resulting from steam bubble collapse and steam-driven slugs of water cannot currently be predicted with adequate accuracy. There may be considerable benefit gained in eliminating these occurrences if this technical problem is solved.

Load Combinations

The NRC in Reference 30 endorses a limited application of the SRSS methodology, although this report encourages research to extend the approach to other specific design situations and also to a generic basis. The approaches that are cited as being feasible are given below:

- Show that the conditional probability for the combination of the dynamic response to exceed the SRSS value is acceptably low
- Demonstrate that the dynamic reserve strength margin exceeds the difference between dynamic responses calculated by the SRSS and the absolute-sum procedure

The NRC also encourages further work to justify decoupling LOCA and SSE loads beyond the SRSS level of conservatism. To date, Westinghouse has demonstrated analytically that an SSE could not reasonably be an initiation of a large pipe break.³⁰

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9. RECOMMENDATIONS FOR FUTURE RESEARCH

Procedures for determining piping loads and current research programs for developing new test data were discussed in the previous chapters of this part. Considerable research is currently being conducted, which, when completed, will have an impact on the present methodologies and load definitions. From a review of the state of the art, the load definitions for weight, normal operating steady-state temperatures and pressures, and sustained mechanical loads (such as weights of valves) and anchor movements are adequate. ATWS piping loads can be calculated; however, the NRC is requiring plant modifications to avoid those loads. Other load types require additional research and improvement.

Much of the data being generated are proprietary to the organizations doing the research. This is particularly true of new data and analytical procedures being sponsored and developed by the private sector of the nuclear power plant industry (i.e., NSSS vendors, A/Es, and engineering consultants). All work pertinent to nuclear power plant licensing is reviewed by the NRC; however, all proprietary material is kept confidential and is not released to other members of the industry. This practice creates some obstacles to future research.

The following recommendations for a future research program are based on the need for new load definitions, on work currently being conducted, and on the availability of data. Considerable research on piping system loads is required in the near future. In order to efficiently develop research programs for some load topics, the research programs currently being conducted should be completed and the results reviewed. Research programs for pressure suppression discharge loads and water hammer loads are included in this category. In addition, the limitations created by the proprietary piping load information must be carefully assessed to determine if the nuclear power plant industry, as a whole, would benefit by duplicate studies. With this basis, the following recommendations for a future research program are made.

LOCA Loads

The LOCA is a mechanistic accident in the reactor coolant system postulated to ensure that the consequences of the worst possible failure can be mitigated. It is generally believed that it is nearly impossible for a full guillotine break to occur in 0.001 sec (which is the current criterion). As discussed in Chapter 8, analyses have been performed and tests conducted to verify that cracks do not open larger than a definite size and that the crack growth process requires a relatively long time to occur (i.e., greater than 0.001 sec). It is recommended that the various results from the analytical studies and testing projects be reviewed and new crack size and propagation time criteria be established. The following tasks should be conducted:

- Obtain all available analytical studies and results that relate to the formation, propagation, and arrest of cracks.
- Obtain results of experimental work related to cracking phenomena.
- Correlate the analytical and testing results and develop conclusions.
- Perform additional analytical studies, as required, to verify that the conclusions are generically applicable.
- Develop new LOCA crack size and failure time criteria.
- Determine the effect of the new criteria on piping system safety margins (i.e., degree of conservatism) and the anticipated impact on piping construction, operation, and maintenance.

In addition to the above work, the various comparisons between analytical predictions of piping forces due to LOCA loads and test data (e.g., LOFT and Semiscale) should be systematically documented for future reference. This task should include review of NRC-developed computer programs (e.g., RELAP-4) and codes developed by NSSS vendors. Both types of codes would be compared with each other and with the experimental data. Although it is believed that the comparisons are acceptable, the degree of variation has not been systematically documented.

Seismic Loads

Several earthquake-related topics directly related to piping loads have been identified for future research. Because of the importance of this load type for piping systems in plants located in seismic areas, the following research is recommended.

Response Spectra Input. Rational procedures for inputting different support response spectra to piping system supports should be developed. Currently, procedures exist for inputting different support time histories, but often the analyst is constrained to use response spectra for piping analysis. The current procedure is to envelop response spectra at all supports, which can lead to excessively conservative loads. In order to provide a better procedure for performing piping system analyses using response spectrum input, which is less costly than using time-history input, the following tasks should be performed:

- Develop an analytical formulation for inputting different support response spectra at various supports of a piping system.
- Compare the results of the response spectrum approach with the "exact" time-history procedure.
- Determine the effects of the new procedures on piping system safety margins (i.e., degree of conservatism) and the anticipated impact on piping construction, operation, and maintenance.

Response Spectra Broadening. Current procedures for broadening floor response spectra are generally more conservative than required to meet safety-related objectives. It is recommended that the following tasks be performed:

- Review current procedures for broadening response spectra and compare them to results of specific analyses from typical nuclear power plants.
- Perform parametric studies to determine response spectra with uniform levels of probability exceedance.
- Investigate alternative procedures for peak broadening that consider the variability of both input motions and the frequencies of structures and piping systems.
- Develop new criteria for response spectra peak broadening.

- Determine the effect of the new procedures on piping system safety margins (i.e., degree of safety) and the anticipated impact on piping construction, operation, and maintenance.

Load Combinations

General procedures for combining loads (e.g., SRV, LOCA, seismic, etc.) should be developed to provide a rational basis for combining dissimilar transient loadings. Currently SRV and seismic loads are combined using the absolute-sum procedure, and LOCA and seismic responses are combined by the SRSS procedure. In order to have a rational procedure(s) for load combinations, the following tasks are recommended:

- Develop probabilistic procedures to combine transient loadings of unknown synchronization in time (e.g., LOCA and SSE).
- Validate these procedures using time-history load data obtained either from tests or analytical results.
- Develop or identify simplified criteria (e.g., SRSS) to combine loads that satisfy the results of the detailed probabilistic procedures.
- Determine the effect of the new procedures on piping system safety margins (i.e., degree of conservatism) and the anticipated impact on piping construction, operation, and maintenance.

Impact of Various Design Loads on Safety of Piping Systems

In addition to the effects of loads on primary piping systems discussed above, which provide the basis for the recommended research, the amount of impact that each load type has on the safety of each piping system in LWR nuclear power plants should be established. Because of the general nature of PSARs and FSARs, this information cannot be generally obtained from these documents. In order to determine how much each load type affects each piping system (see Tables 2-2 and 2-3 for lists of piping systems), a detailed review of piping system stress analysis reports is required.

The results of this work would provide a basis for determining which loads have the greatest safety impact on each piping system and would provide the basis for any additional needed research.

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- 5.4.4 Main Stream Line Flow Restriction System
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TABLE 2-2
SEISMIC CATEGORY AND DESIGN CODE SPECIFICATION
FOR PWR SYSTEMS WITH PIPING ELEMENTS

System	SONGS Units 2 and 3		Sundesert Units 1 and 2	
	Seismic Category*	Design Code†	Seismic Design‡	Design Code‡
<u>Reactor</u>				
<u>Reactor Coolant System</u>				
Reactor coolant piping system	I	III-1	1	III-1
Pressurizer relief discharge system				
Upstream of safety valve	I	III-1		
Downstream of safety valve	II	B31.1		
Surge line			1	III-1
Crossover leg			1	III-1
Lower oil reservoir (pump)			1	III-3
<u>Engineered Safety Features</u>				
Containment heat removal systems				
Containment spray system	I	III-2	6	III-2
Containment isolation system	I	III-2	6	III-2
Combustible gas control system	I	III-2	6	III-2
Safety injection system				
Reactor coolant pressure boundary	I	III-1		
Drain lines	II	B31.1		
Other	I	III-2		
Residual heat removal system			1	III-2
Fission product removal and control systems	I	III-2		

TABLE 2-2 (Continued)

System	SONGS Units 2 and 3		Sundesert Units 1 and 2	
	Seismic Category*	Design Code†	Seismic Design‡	Design Code†
<u>Auxiliary Systems</u>				
Fuel pool cooling systems			6	III-3
Spent fuel cooling system	I	III-3		
Purification system				
Containment penetration	I	III-2		
Other	II	B31.1		
Salt water cooling system	I	III-3		
Components cooling water system			6	III-3
Containment penetration	I	III-2		
Other	I	III-3		
Condensate storage facility				
Auxiliary feedwater portion	I	III-3		
Turbine plant portion	II	B31.1		
Nuclear service water system				
Containment penetrations	I	III-2		
Other	II	B31.1		
Reactor plant service water system			6	III-3
Turbine plant cooling water system	II	B31.1		
Equipment and floor drainage				
Radioactive sump and drain system				
Component cooling water area	II	B31.1		

TABLE 2-2 (Continued)

System	SONGS Units 2 and 3		Sundesert Units 1 and 2	
	Seismic Category*	Design Code†	Seismic Design‡	Design Code‡
<u>Auxiliary Systems (Cont.)</u>				
Containment area	II	B31.1		
Fuel handling building	II	B31.1		
Penetration area	II	B31.1		
Safety injection area	II	B31.1		
Storage tank area	II/III	B31.1		
Radwaste area	II	B31.1		
Containment penetration	I	III-2		
Chemical and volume control system			1/2	III-2/ III-3
Containment penetrations	I	III-2		
Letdown portion: containment pene- tration to back- pressure control valve	I	III-2		
Letdown portion: backpressure control valve to radwaste diversion valve	II	III-2		
Volume control tank	II	III-2		
Charging portion	I	III-2		
Within boric acid makeup system	I	III-3		
Boric acid blending tee	I	III-2		
From refueling water tank to charging pump suction	I	III-2		
Boron recovery system			6	III-3

TABLE 2-2 (Continued)

System	SONGS Units 2 and 3		Sundesert Units 1 and 2	
	Seismic Category*	Design Code†	Seismic Design‡	Design Code‡
<u>Auxiliary Systems (Cont.)</u>				
Containment building ventilation systems (normal operation)				
Containment normal cooling units				
Containment penetration	I	III-2		
Other (inside containment)	II	B31.1		
Other (outside containment)	II	B31.1		
Auxiliary building ventilation system (normal operation)				
Normal chilled water	III	B31.1		
Auxiliary building ventilation system (emergency operation)				
Emergency chilled water	I	III-3		
Turbine building ventilation system				
Steam-air ejector exhaust	II	B31.1		
Fire protection system				
Containment penetration	I	III-2		
Other	III	NFPA		
Diesel generator fuel oil storage and transfer system	I	III-3	6	III-3
Diesel generator cooling water system	I	III-3	6	III-3
Diesel generator starting system	I	III-3	6	III-3

TABLE 2-2 (Continued)

System	SONGS Units 2 and 3		Sundesert Units 1 and 2	
	Seismic Category*	Design Code†	Seismic Design‡	Design Code‡
<u>Auxiliary Systems (Cont.)</u>				
Diesel generator lubrication system	I	III-3	6	III-3
Diesel generator air intake and exhaust system			6	III-3
<u>Steam and Power Conversion</u>				
Turbine-generator				
Lube oil system	III	B31.1		
Other	II	B31.1		
Main steam supply system				
From steam generator through containment isolation valves	I	III-2	6,2	III-2
To auxiliary feed-water pump turbine	I	III-3	6,3	III-3
Main condenser				
Turbine gland sealing system	II	B31.1		
Turbine bypass system	II	B31.1		
Circulating water system	II	B31.1		
Condensate and feedwater				
Containment penetration to steam generator inlet	I	III-2	6	III-2
Other	II	B31.1		
Steam generator blowdown				
Steam generator through containment penetration	I	III-2	6	III-2
Other	II	B31.1		

TABLE 2-2 (Continued)

System	SONGS Units 2 and 3		Sundesert Units 1 and 2	
	Seismic Category*	Design Code†	Seismic Design‡	Design Code‡
<u>Steam and Power Conversion (Cont.)</u>				
Auxiliary feedwater				
Containment isolation valves to steam generator inlet	I	III-2	6	III-2
Other	I	III-3		
Turbine plant chemical addition system	II	B31.1		
<u>Radioactive Waste Management Systems</u>				
Coolant radwaste, miscel- laneous liquid				
Waste and boric acid recycle				
Containment penetration	I	III-2		
Other	II	B31.1		
Waste gas			6	III-3
Containment penetration	I	III-2		
Waste gas surge tank drain	II	B31.1		
Waste gas discharge header	II	B31.1		
Vent gas collection header	II	B31.1		
Other	I	III-3		
Radwaste solidification				
Radioactive	II	B31.1		
Nonradioactive	III	B31.1		
Containment atmosphere radiation monitoring				
Containment penetration			6	III-2
Outside containment			5	NA

TABLE 2-2 (Continued)

*Seismic Category:

- I = Defined by NRC RG 1.29, Seismic Design Classification.
- II = Those structures, components, and systems that are not Seismic Category I but whose limited damage could interrupt generation of power.
- III = Those structures, components, and systems that are not Seismic Category I or II but whose failure could inconvenience normal plant operation.

†Design Code:

- III-1 = ASME Code, Section III, Class 1.
- III-2 = ASME Code, Section III, Class 2.
- III-3 = ASME Code, Section III, Class 3.
- B31.1 = ANSI B31.1-1973 Power Piping.
- NFPA = National Fire Protection Association

‡Seismic Design:

Seismic Category I Qualification By:

- 1 = Individual analysis dynamic method.
- 2 = Generic analysis dynamic method.
- 3 = Static methods using dynamic load factors.
- 4 = Testing method.
- 5 = Nonseismic design.
- 6 = To be provided in FSAR.

TABLE 2-3
SEISMIC CATEGORY AND DESIGN CODE SPECIFICATION
FOR BWR SYSTEMS WITH PIPING ELEMENTS

System	WPPSS No. 2		Skagit	
	Seismic Category*	Design Code†	Seismic Category*	Design Code†
<u>Reactor</u>				
CRD hydraulic system			G	G
Discharge volume lines	I	III-2		
Insert and withdraw lines	I	III-2		
Other	II	B31.1		
Neutron monitoring system	I	III-2	G	G
<u>Reactor Coolant System</u>				
Main stream isolation valves leakage control system				
Within outermost isolation valve	I	III-1		
Beyond outermost isolation valve	I	III-2		
Reactor recirculation system	I	III-1	G	G
Reactor core isolation cooling system			G	G
Within outmost isolation valves of reactor coolant pressure boundary (RCPB)	I	III-1		
Beyond outermost isolation valves	I	III-2		
Return test line to condensate storage tank beyond second stop valve	II	B31.1		
Residual heat removal system			G	G
Within outermost isolation valves, RCPB	I	III-1		
Other	I	III-2		
Reactor water cleanup system			G	G
Within outermost isolation valves, RCPB	I	III-1		
Other	I	III-3		

TABLE 2-3 (Continued)

System	WPPSS No. 2		Skagit	
	Seismic Category*	Design Code†	Seismic Category*	Design Code†
<u>Reactor Coolant System (cont.)</u>				
Nuclear boiler system			G	G
Relief valve discharge from relief valve to suppression pool	I	III-3		
Relief valve discharge within suppression chamber and suppression pool	I	III-2		
Main steam and feedwater within outermost isolation valve	I	III-1		
Other, within outermost isolation valves	I	III-1		
<u>Engineered Safety Features</u>				
Primary containment cooling system	I	III-2		
Primary containment atmospheric control system	I	III-2		
Hydrogen control system			I	III-2
Low-pressure core spray system			G	G
Within outermost isolation valves	I	III-1		
Beyond outermost isolation valves	I	III-2		
High-pressure core spray system			G	G
Within outermost isolation valve	I	III-1		
Return test line to condensate storage beyond second isolation valve	II	B31.1		
Beyond outermost isolation valve	I	III-2		
Standby gas treatment system	I	III-2	I	III-2

TABLE 2-3 (Continued)

System	WPPSS No. 2		Skaqit	
	Seismic Category*	Design Code†	Seismic Category*	Design Code†
<u>Auxiliary Systems</u>				
Fuel pool cooling and cleanup system				
Suppression pool to outer isolation valves	I	III-2		
Other	I/II	III-3		
Containments isolation related			I	III-2
Standby makeup system and spent fuel pool cooling loop related			I	III-3
Other			NA	B31.1
Turbine plant service water system	II	B31.1		
Reactor building closed cooling water system				
Inside containment	I	III-3		
Containment isolation related	I	III-2		
In reactor building	II	B31.1		
Other	II	B31.1		
Reactor component cooling water system				
Piping forming part of containing boundary			I	III-2
Other			NA	B31.1
Condensate storage and transfer system			NA	B31.1
Cleanup system	II	B31.1	NA	B31.1
Standby service water system	I	III-3		
Chemical treatment make-up, blowdown			NA	B31.1
Other			I	III-3

TABLE 2-3 (Continued)

System	WPPSS No. 2		Skagit	
	Seismic Category*	Design Code†	Seismic Category*	Design Code†
<u>Auxiliary Systems (cont.)</u>				
Service water system				
Piping forming part of containment boundary			I	III-2
Other			NA	B31.1
Instrument and service air system	I	B31.1		
Piping in lines between accumulators supporting safety-related systems			I	III-3
Piping forming part of containment boundary, ADS piping			I	III-2
Other			NA	B31.1
Containment instrument air system				
Inside containment to and including outboard isolation valve	I	III-2		
To main steam relief valves	I	III-2		
Other	II	B31.1		
Instruments and sample lines	I/II	III-1/ III -2/ B31.1		
Standby liquid control system			G	G
Within isolation valves	I	III-1		
Beyond isolation valves	I	III-2		
Diesel generator systems (HPCS)	I	B31.1		
Piping, ESF			I	III-3
Start air system			I	III-3
Piping, non-ESF			II	B31.1
Standby AC power system (other than HPCS)	I	III-3		

TABLE 2-3 (Continued)

System	WPPSS No. 2		Skagit	
	Seismic Category*	Design Code†	Seismic Category*	Design Code†
<u>Auxiliary Systems (cont.)</u>				
Heating, ventilating, and air conditioning systems				
Control room			I	III-3
Control building			I	III-3
General area A/C units			I	III-3
Access control area A/C units			NA	B31.1
<u>Steam and Power Conversion</u>				
Power conversion system				
Main steam piping from outermost isolation valves up to turbine stop valves	I	III-2		
Main steam branch piping to first valve capable of timely actuation	I	II-2		
Main turbine bypass piping up to bypass valve	II	B31.1		
Main steam leads from turbine control valve to turbine casing	II	B31.1		
Feedwater and condensate system beyond outermost isolation valve	II	B31.1		
Feedwater treatment system	II	B31.1		
Turbine bypass system beyond turbine bypass valve	II	B31.1		
Turbine gland sealing system	II	B31.1		
Other	II	B31.1		
Main steam and reheat systems				
Main steam piping from outer containment isolation valve up to and including the moment-				

TABLE 2-3 (Continued)

System	WPPSS No. 2		Skagit	
	Seismic Category*	Design Code†	Seismic Category*	Design Code†
<u>Steam and Power Conversion (cont.)</u>				
limiting restraint system downstream of the main steam shutoff valve			I	III-2
Piping of 2-1/2 in. or larger nominal pipe size connected to above piping up to but not including the turbine and bypass valves			NA	B31.1
Main steam leads from turbine control valve to turbine casing			NA	B31.1
Turbine bypass system beyond turbine bypass valve			NA	B31.1
Turbine gland sealing system			NA	B31.1
Other steam line piping			NA	B31.1
Main condenser and evacuation system			NA	B31.1
Feedwater system				
Reactor feedwater piping to the movement restraints at the auxiliary-turbine building interface, and between the outermost containment isolation valve up to and including the moment restraint at the auxiliary-turbine building interface			NA	B31.1
Other feedwater piping			I	III-2
Feedwater pump turbine related			NA	B31.1
Circulating water and cooling tower makeup water systems	I	II-2	NA	B31.1

TABLE 2-3 (Continued)

System	WPPSS No. 2		Skagit	
	Seismic Category*	Design Code†	Seismic Category*	Design Code†
<u>Radioactive Waste Management</u>				
Radwaste system				
Piping forming part of containment boundary	I	III-2		
Other	II	III-3		
Liquid radwaste				
Piping forming part of containment boundary			I	III-2
Other			NA	B31.1
Solid radwaste			NA	B31.1
Off-gas system	II	III-3	G	G

*Seismic Category:

I = Constructed in accordance with the seismic requirements of Seismic Category I structures and equipment for the Safe-Shutdown Earthquake (SSE).

NA, II = The seismic requirements for the SSE are not applicable.

G = Defined by GESSAR (Reference 1).

†Design Code:

III-1 = ASME Code, Section III, Class 1.

III-2 = ASME Code, Section III, Class 2.

III-3 = ASME Code, Section III, Class 3.

B31.1 = ANSI B31.1-1973 Power Piping.

TABLE 2-4
EXAMPLE NUCLEAR POWER PLANT PIPING NORMAL
OPERATING PRESSURES AND TEMPERATURES

Piping System	PWR*		BWR†	
	Pressure (psia)	Temperature (°F)	Pressure (psia)	Temperature (°F)
Main Coolant Piping				
Hot Leg	2,250	621	NA	NA
Cold Leg	2,300	565	NA	NA
Main Steam Line	1,070	550	1,000	545

*Values given for C-E System 80 design (Reference 117).

†Values given for GE BWR/6 design (Reference 6).

TABLE 2-5
SUMMARY OF REACTOR COOLANT SYSTEM DESIGN
TRANSIENTS IN A TYPICAL PWR*

Transient	Lifetime Occurrences
<u>Normal Conditions</u>	
Heatup and cooldown at 100°F/hr (pressurizer cooldown 200°F/hr)	200 (each)
Unit loading and unloading at 5% of full power/min	13,200 (each)
Step load increase and decrease with steam dump	2,000 (each)
Large step load decrease with steam dump	200
Steady-state fluctuations	
Initial fluctuations	1.5×10^5
Random fluctuations	3.0×10^6
Feedwater cycling at hot shutdown	2,000
Loop out of service	
Normal loop shutdown	80
Normal loop startup	70
Unit loading and unloading between 0 and 15% of full power	500 (each)
Boron concentration equalization	26,400
Refueling	80
<u>Upset Conditions</u>	
Loss of load, without immediate reactor trip	80
Loss of power (blackout with natural circulation in the RCS)	40
Partial loss of flow (loss of one pump)	80
Reactor trip from full power	
Without cooldown	230
With cooldown, without safety injection	160
With cooldown and safety injection	10
Inadvertent reactor coolant depressurization	20
Inadvertent startup of an inactive loop	10
Control rod drop	80

TABLE 2-5 (Continued)

Transient	Lifetime Occurrences
<u>Upset Conditions (Cont.)</u>	
Inadvertent ECCS actuation	60
Operating basis earthquake (20 earthquakes of 10 cycles each)	200 cycles
<u>Emergency Conditions</u>	
Small LOCA	5
Small steam break	5
Complete loss of flow	5
<u>Faulted Conditions</u>	
Main reactor coolant pipe break (large LOCA)	1
Large steam break	1
Feedwater line break	1
Reactor coolant pump locked rotor	1
Control rod ejection	1
Steam generator tube rupture	(included under upset conditions, reactor trip from full power with safety injection)
Safe-shutdown earthquake	1
<u>Test Conditions</u>	
Turbine roll test	20
Primary side hydrostatic test	10
Secondary side hydrostatic test	10
Primary side leak test	200
Secondary side leak test	80
Tube leakage test	800

*From Reference 4.

TABLE 2-6
SUMMARY OF REACTOR COOLANT SYSTEM
DESIGN TRANSIENTS IN A BWR†

Transient	Lifetime Occurrences
<u>Normal, Upset, and Testing Conditions</u>	
Bolt up*	123
Design hydrostatic test	130
Startup (100°F/hr heatup rate)**	120
Daily reduction to 75% power*	10,000
Weekly reduction to 50% power*	2,000
Control rod pattern change*	400
Loss of feedwater heaters (80 cycles total)	80
Operating base earthquake event at rated operating conditions	1****
Scram:	
Turbine generator trip, feedwater on, isolation valves stay open	40
Other scrams	140
Loss of feedwater pumps, isolation valves closed	10
Single safety or relief valve blowdown	8
Reduction to 0% power, hot standby, shutdown (100°F/hr cooldown rate)**	111
<u>Emergency Conditions</u>	
Scram:	
Reactor overpressure with delayed scram, feedwater stays on, isolation valves stay open	1***
Automatic blowdown	1***
Improper start of cold recirculation	1***
Sudden start of pump in cold recirculation loop	1***
Improper startup with reactor drain shut off followed by turbine roll and increase to rated power	1***

TABLE 2-6 (Continued)

Transient	Lifetime Occurrences
<u>Faulted Condition</u>	
Pipe Rupture and Blowdown	1***
Safe-shutdown earthquake at rated operating conditions	1***

*Applies to reactor pressure vessel only.

**Bulk average vessel coolant temperature change in any 1-hour period.

***The average encounter probability of the one-cycle events is 10^{-2} for emergency and 10^{-4} for faulted events.

****Includes 10 maximum load cycles per event.

TABLE 2-7
EVENTS INITIATING ANTICIPATED
TRANSIENTS WITHOUT SCRAM*

Pressurized Water Reactors

- Rod withdrawal
- Boron dilution
- Loss of primary flow
- Inactive primary loop startup
- Loss of electrical load
- Loss of normal electrical power
- Load increase
- Primary system depressurization
- Excessive cooldown

Boiling Water Reactors

- Primary coolant flow decrease
- Reactor water temperature decrease
- Reactor coolant flow increase
- Primary pressure increase
- Rod withdrawal
- Loss of normal electrical power
- Safety-relief valve stuck open

*From Reference 13.

TABLE 2-8
SUMMARY OF ATWS PEAK REACTOR PRESSURES*

Vendor/System	ATWS Event	Peak Pressure (psia)
General Electric		
BWR/4	MISV closure with safety valve stuck open	1,350
BWR/5		1,270
BWR/6		1,322
Westinghouse	Loss of load with one relief valve failed	3,197
Combustion Engineering		
2,560-MWt	Loss of feedwater with one relief valve failed	4,508
3,800-MWt	Loss of feedwater	4,087
Babcock & Wilcox		
145 FA	Loss of feedwater with one relief valve failed	5,404
177 FA		4,978
205 FA - 3,600-MWt		4,555
205 FA - 3,800-MWt		4,372

*From Reference 12.

TABLE 2-9
MARK II CONTAINMENT HYDRODYNAMIC LOADS ON PIPING SYSTEMS

Load or Phenomenon	NRC Criterion*
<u>LOCA-Related Hydrodynamic Loads</u>	
Pool swell loads - impact on small structures (e.g., pipes)	I.A.6
Steam condensation and chugging loads - downcomer lateral loads	
Single-vent loads	I.B.1
Multiple-vent loads	I.B.2
<u>SRV-Related Hydrodynamic Loads</u>	
Quencher tie-down loads	
Quencher arm loads	
Four-arm quencher	Mark II owner's group position acceptable
KWU T-quencher	Review continuing
Quencher tie-down loads	
Four-arm quencher	Mark II owner's group position acceptable
KWU T-quencher	Review continuing
<u>LOCA/SRV Submerged Structure Loads</u>	
Jet loads	
LOCA/ramshead SRV loads	III.A.1
SRV quencher	III.A.2
Air bubble drag loads	
LOCA	III.B.1
SRV ramshead	III.B.2
SRV quencher	III.B.3
Steam condensation drag loads	Plant review on unique basis
<u>Secondary Loads</u>	
Vent thrust loads	Mark II owner's group position acceptable
Friction drag loads on vents	Mark II owner's group position acceptable

*Reference 17, Appendix D.

TABLE 2-10
CLASS 1 LOAD COMBINATIONS

Plant Operating Condition and Stress Category	Load Combination
<u>Normal/Upset</u>	
a. Primary stress	Weight Design pressure Sustained mechanical loads OBE inertia effects
b. Primary plus secondary stress	Range of operating pressure* OBE Inertia effects Differential anchor movement Other anchor movements (including thermal) Thermal* Uniform expansion Radial linear gradient Radial nonlinear gradient Discontinuity difference Other mechanical loads
(Or if a. cannot be satisfied)	Weight Range of operating pressure* Sustained mechanical loads OBE inertia effects Thermal* Uniform expansion Discontinuity difference Anchor movements
c. Peak stress	Range of operating pressure* Sustained mechanical loads OBE Inertial effects Differential anchor movement

TABLE 2-10 (Continued)

Plant Operating Condition and Stress Category	Load Combination
<u>Normal/Upset (cont.)</u>	
c. Peak stress (cont.)	Other anchor movements (including thermal) Thermal* Uniform expansion Radial linear gradient Discontinuity difference
<u>Emergency</u>	
a. Primary stress	Weight Peak pressure* Sustained mechanical loads OBE inertia effects Mechanical emergency condition loading*
<u>Faulted</u>	
a. Primary stress	Weight Peak pressure* Sustained mechanical loads SSE inertia effects Mechanical emergency condition loading*

*Effects due to transient loading are analyzed. See Tables 2-5 and 2-6 for lists of transients.

TABLE 2-11
CLASS 2 AND 3 LOAD COMBINATIONS

Plant Operating Condition and Stress Category	Load Combinations
<u>Normal/Upset</u>	
a. Sustained loads	Weight Design pressure Sustained mechanical loads
b. Occasional loads	Weight Peak pressure* Sustained mechanical loads OBE inertia effects Occasional mechanical loads*
c. Thermal expansion loads	Thermal expansion Anchor movements Thermal OBE
d. Combined sustained and thermal loads (if c. cannot be satisfied)	Weight Design pressure Sustained mechanical loads Thermal expansion Anchor movements Thermal OBE
<u>Emergency</u>	
a. Occasional loads	Weight Design pressure Sustained mechanical loads OBE inertia effects Mechanical emergency loading condition*

TABLE 2-11 (Continued)

Plant Operating Condition and Stress Category	Load Combination
<u>Faulted</u> a. Occasional loads	Weight Peak pressure* Sustained mechanical loads SSE inertia effects Mechanical faulted loading condition*

*Effects due to transient loading are analyzed. See Tables 2-5 and 2-6 for lists of transients.

TABLE 2-12
ANALYTICAL METHODS AND BASES FOR ATWS EVENTS

Category	NSSS Manufacturer			
	B & W	C-E	GE	Westinghouse
Computer Codes*	CADDS (Ref. 118)	CESEC (Ref. 121)	REDY (Ref. 124)	LOFTRAN (Ref. 125)
	CRAFT2 (Ref. 119)	CEFLASH4 (Ref. 122)		
Transient Analysis Basis Report	BAW-10099 Rev. 1 (Ref. 120)	CENPD-158 (Ref. 123)	NEDO-10802-02 (Ref. 124)	WCAP-8330 (Ref. 126)

*Required for piping load determination.

TABLE 2-13
TESTS PERFORMED TO DEVELOP LOADS FOR
MARK I CONTAINMENT SYSTEM

Test	Reference	Purpose
<u>Short-Term Program</u>		
1/10-scale	71	Performed by EPRI to determine pool surface response during LOCA initial phase (0 to 0.6 sec)
1/12-scale	72	Performed by GE to defined torus hydrodynamic loads
Foreign test facility	73	Sponsored by GE to investigate loads at ends of vent pipes during a LOCA
<u>Long-Term Program</u>		
Full-scale testing program	83	Being performed by GE to investigate hydrodynamic loads and response from chugging on a 22.5° torus sector
1/5-scale tests	75,88	Performed at LLL to compare two-dimensional versus three-dimensional behavior
Quad Cities nuclear plant tests	90	GE-sponsored program to provide data for analytical model for predicting steam vent clearing loads

TABLE 2-14
METHODOLOGY AND TEST BASES FOR MARK II CONTAINMENT
HYDRODYNAMIC LOADS ON PIPING SYSTEMS

Load or Phenomenon	Methodology and/or Test Bases
<u>LOCA-Related Hydrodynamic Loads</u>	
Pool swell loads - impact on small structures (e.g., pipes)	Reference 127
Steam condensation and chugging loads - downcomer lateral loads	
Single-vent loads	Reference 127
Multiple-vent loads	Reference 127
<u>SRV-Related Hydrodynamic Loads</u>	
Quencher tie-down loads	
Quencher arm loads	
Four-arm quencher	Reference 127
KWU T-quencher	See note
Quencher tie-down loads	
Four-arm quencher	Reference 127
KWU T-quencher	See note
<u>LOCA/SRV Submerged Structure Loads</u>	
Jet loads	
LOCA/ramshead SRV loads	References 127 and 128
SRV-quencher	See note
Air bubble drag loads	
LOCA	References 127, 128, and 129
SRV ramshead	Reference 128
SRV quencher	See note
Steam condensation drag loads	See note
<u>Secondary Loads</u>	
Vent thrust loads	Reference 127
Friction drag loads on vents	Reference 127

NOTE: Loads not specified at this time.

TABLE 2-15
PRINCIPAL AREAS OF NRC RESEARCH IN FIELD OF LWR SAFETY

Research Area	Cognizant NRC Research Branch
Thermal-hydraulic tests of postulated accidents and the effectiveness of the engineered safety features	Systems Engineering Branch
Fuel-rod behavior in postulated accidents and associated failure limits	Fuel Behavior Research Branch
Computer code development for accurately predicting the consequences of postulated reactor accidents	Analysis Development Branch
Safety design and protection of the integrity of the reactor pressure vessel and piping	Metallurgy and Materials Research Branch
Operational safety aspects of nuclear-power-plant operation	Research Support Branch

TABLE 2-16
CATEGORY A GENERIC TASKS IN THE NRC PROGRAM

Task No.	Title
A-1*	Water Hammer
A-2*	Asymmetric Blowdown Loads on PWR Primary Coolant Systems
A-3*	Westinghouse Steam Generator Tube Integrity
A-4*	Combustion Engineering Steam Generator Tube Integrity
A-5*	Babcock & Wilcox Steam Generator Tube Integrity
A-6*	Mark I Short-Term Program
A-7*	Mark I Long-Term Program
A-8*	Mark II Containment Pool Dynamic Loads
A-9*	Anticipated Transients Without Scram
A-10*	BWR Nozzle Cracking
A-11*	Reactor Vessel Materials Toughness
A-12*	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports
A-13	Snubber Operability Assurance
A-14	Flaw Detection
A-15	Primary Coolant System Decontamination and Steam Generator Chemical Cleaning
A-16	Steam Effects on BWR Core Spray Distribution
A-17*	Systems Interaction in Nuclear Power Plants
A-18	Pipe Rupture Design Criteria
A-19	Digital Computer Protection Systems
A-20	Impacts of the Coal Fuel Cycle
A-21	Main Steam Line Break Inside Containment - Evaluation of Environmental Conditions for Equipment Qualification.
A-22	PWR Main Steam Line Break - Core, Reactor Vessel and Containment Response
A-23	Containment Leak Testing
A-24*	Qualification of Class IE Safety-Related Equipment
A-25	Nonsafety Loads on Class IE Power Sources
A-26*	Reactor Vessel Pressure Transient Protection (Overpressure Protection)
A-27	Reload Applications
A-28	Increase in Spent Fuel Pool Storage Capacity

TABLE 2-16 (Continued)

Task No.	Title
A-29	Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage
A-30	Adequacy of Safety-Related DC Power Supplies
A-31*	RHR Shutdown Requirements
A-32	Missile Effects
A-33	NEPA Reviews of Accident Risks
A-34	Instruments for Monitoring Radiation and Process Variables During Accidents
A-35	Adequacy of Offsite Power Systems
A-36*	Control of Heavy Loads Near Spent Fuel
A-37	Turbine Missiles
A-38	Tornado Missiles
A-39*	Determination of Safety Relief Valve (SRV) Pool Dynamic Loads to Temperature Limits for BWR Containments
A-40*	Seismic Design Criteria - Short Term Programs
A-42*	Pipe Cracks in Boiling Water Reactors
A-43*	Containment Emergency Sump Reliability
A-44*	Station Blackout

*Unresolved safety issues relating to nuclear power plants

TABLE 2-17
BWR MARK I NUCLEAR POWER PLANTS

Owner	Plant
<u>Licensed to Operate</u>	
Tennessee Valley Authority	Browns Ferry, Units 1, 2, and 3
Carolina Power & Light Co.	Brunswick, Units 1 and 2
Nebraska Public Power District	Cooper
Commonwealth Edison Co.	Dresden, Units 2 and 3
Iowa Electric Light & Power Co.	Duane Arnold
Power Authority of the State of New York	Fitzpatrick
Georgia Power Co.	Hatch, Units 1 and 2
Northeast Utilities	Millstone, Unit 1
Northern States Power	Monticello
Niagara Mohawk Power Corp.	Nine Mile Point, Unit 1
Jersey Central Power & Light Co.	Oyster Creek
Philadelphia Electric Co.	Peach Bottom, Units 2 and 3
Boston Edison Co.	Pilgrim, Unit 1
Commonwealth Edison Co. and Iowa-Illinois Gas & Electric Co.	Quad Cities, Units 1 and 2
Vermont Yankee Nuclear Power Corp.	Vermont Yankee
<u>Under Operating License Review</u>	
Detroit Edison Co.	Fermi, Unit 2
<u>Under Construction</u>	
Public Service Electric & Gas Co.	Hope Creek, Units 1 and 2

TABLE 2-18
BWR MARK II NUCLEAR POWER PLANTS

Owner	Plant
<u>Licensed to Operate</u>	
None	
<u>Under Operating License Review</u>	
Washington Public Power Supply	WPPSS-2
Commonwealth Edison Co.	LaSalle, Units 1 and 2
Long Island Lighting Co.	Shoreham
Pennsylvania Power and Light Co.	Susquehanna, Units 1 and 2
Cincinnati Gas and Electric Co.	Zimmer
<u>Under Construction</u>	
Northern Indiana Public Service Co.	Bailly I
Philadelphia Electric Co.	Limerick, Units 1 and 2
Niagara Mohawk Power Co.	Nine Mile Point, Unit 2

FIGURES

FIGURE 2-1 BWR PRIMARY LOOP AND TURBINE POWER TRAIN

FIGURE 2-2 PWR PRIMARY LOOP AND TURBINE POWER TRAIN

FIGURE 2-3 EXAMPLE PLANT START-UP AND SHUTDOWN TEMPERATURE AND PRESSURE TIME HISTORIES

Temperature Terms

$\bar{T}$ = Average (uniform)

T_1 = Linear

T_2 = Nonlinear

FIGURE 2-4 TEMPERATURE TERMS AND SAMPLE THERMAL TRANSIENT TIME HISTORY FOR A RAMP CHANGE IN FLUID TEMPERATURE

FIGURE 2-5 LOSS OF FEEDWATER ATWS PRESSURE TIME HISTORY

Longitudinal Split or
Through-the-Wall Leakage Crack

Circumferential Rupture

FIGURE 2-6 LOCA LONGITUDINAL SPLIT AND CIRCUMFERENTIAL
RUPTURE GEOMETRIES

FIGURE 2-7 EXAMPLE LOCA BREAK LOCATIONS

FIGURE 2-8 EXAMPLE LOCA PIPE-THRUST TIME HISTORY

FIGURE 2-9 LOCA JET IMPINGEMENT ON ADJACENT PIPE

Operating Basis Earthquake
Horizontal Component
Damping = 0.05

FIGURE 2-10 EXAMPLE DESIGN AND SYNTHETIC-TIME-HISTORY GROUND RESPONSE SPECTRA

2-143 - 2-144

FIGURE 2-11 EXAMPLE FLOOR RESPONSE SPECTRUM

PART 3

Task 3: Nonlinear Analysis Methods as Applied to the Design of LWR Piping Systems

1. OBJECTIVE AND SCOPE

The objective of Task 3 was to assess qualitatively any benefits that might be realized by using nonlinear analysis methods in the design and licensing of LWR piping systems. To achieve this objective, a study made up of several subtasks was conducted. The findings and conclusions of the subtasks that were performed for the Task 3 study are summarized in the chapters that follow. These subtasks are:

- Survey of publicly available computer codes commonly used in piping analysis (Chapter 2)
- Identification of commonly used analysis methods and assumptions used in current piping design practice (Chapter 3)
- Identification of sources of nonlinearities in LWR piping systems (Chapter 4)
- Comparative study of a typical LWR piping system (Chapter 5)

Tables and figures for Task 3 follow the text.

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2. SURVEY OF ANALYSIS CODES

Analysis of piping systems for various service and accident conditions is accomplished using commercially available or proprietary computer codes. Most of the currently used codes are based on linear elastic principles. However, codes with various nonlinear capabilities are also available.

As part of this study, a survey of computer codes with linear and nonlinear analysis capabilities was performed. The results of this survey are presented in the Appendix. The information given for each code includes the developer of the code, the agency maintaining the code, the availability and capabilities of the code, and the methods and assumptions used by the code. In addition, the capabilities of each code are presented in tabular form.

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3. CONVENTIONAL PIPING ANALYSIS AND DESIGN METHODS

Piping systems in LWR plants are currently analyzed and designed in accordance with the provisions of Section III of the ASME Code. There are several types and classes of piping in modern nuclear power plants, and the design requirements for the different classes vary. In general, nuclear-safety-class piping systems are designed for the following loads:

- Weight
- Normal operating steady-state pressures and temperatures
- Pressure and temperature
- Accident loads
- Seismic loads
- Anchor movements and other mechanical loads

Currently, linear-elastic deterministic methods are used to analyze piping systems for most of these loads. In general, the use of linear-elastic methods in analysis necessarily requires a number of unrealistic assumptions, omissions, and approximations. For example, the gaps that commonly exist at piping supports are neglected. Snubbers are treated as rigid supports under dynamic loading, neglecting the locking and unlocking thresholds and bleed rates. When spectrum analysis techniques are used for seismic analysis, the differences between various support motions are neglected, and an average envelope spectrum is used. Modal responses and responses in different directions are combined with approximate methods, such as the square-root-of-the-sum-of-the-squares (SRSS) method.

There is a hypothesis that the cumulative effect of these various assumptions, omissions, and approximations on the piping design is excessive conservatism and, therefore, unnecessary cost. The increased conservatism leading to an overly stiff system would also cause large stresses in piping due to thermal loads. Thus, the development and application of analysis methods that include nonlinear behavior may better quantify existing safety margins and may lead to cost savings without any reduction in safety.

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4. NONLINEARITIES IN LWR PIPING SYSTEMS

Both geometric and material nonlinearities exist in piping systems. Descriptions of the most significant sources of nonlinearities in LWR piping systems are presented below.

Gaps

Gaps are the most frequently occurring source of nonlinearities at piping supports. Typically a 1/16-in. to 1/8-in. gap is introduced between the pipe and the restraint to accommodate radial thermal expansion of the pipe.

One-Way Stops

One-way stops are introduced into piping systems in various forms. A rod hanger with a very small buckling load may be considered to be a one-way stop. At some supports, the pipe may simply sit on the support without tie-down for uplift. In vertical runs of piping, lugs are welded onto the pipe, and these lugs are allowed to rest on the support, permitting free upward movement of the pipe.

Sliding Interfaces

At some supports, where free horizontal movements of the pipe are allowed but a vertical restraint is provided, a condition exists where there will be a certain resistance to horizontal movements due to friction forces at the interface. The behavior of this type of support is very complex because the friction force depends on the force perpendicular to the interface, which varies continuously during a dynamic event.

Constant Load Springs

Constant load springs are hangers that provide a constant reaction force to a pipe system independent of the displacement response of the pipe at the support point. Constant load spring supports are used when piping stresses are critical and the pipe is subject to large thermal movements. The constant restraint force is achieved through the entire deflection range by counterbalancing the load and spring moments about a movable spring pivot.

Buckling

Buckling is a nonlinearity that exists when slender hangers are compressed. For rod hangers with a very small compression capacity, buckling behavior may be modeled as one-way stop.

Yielding

Yielding may occur both in the pipe and in its supports. The material behavior may be modeled as elastic-perfectly plastic or strain hardening.

Snubbers

Snubbers are devices that permit free movement of pipes under thermal expansion and provide resistance under dynamic loading when the accelerations or velocities at the supported point exceed a predetermined threshold value. There are two types of snubbers: hydraulic and mechanical.

The hydraulic snubber is a velocity-sensitive device similar to hydraulic shock absorbers. A control valve permits unrestricted motion of the support point at low velocities. At a predetermined locking velocity, the control valve closes and resistance to motion develops. In some snubbers, a bleed system is provided to permit some movement after locking by opening a bleed orifice. The reactions developed are proportional to the bleed velocity.

Mechanical snubbers are acceleration controlled. They may be locking or non-locking types. The former acts as a rigid strut when the acceleration of the support point exceeds a threshold value. The latter limits the acceleration of the support point to a predetermined value after the locking threshold acceleration has been exceeded.

In conventional analysis, the snubbers are omitted for static analysis and are assumed to be rigid for seismic or other dynamic analyses.

5. COMPARATIVE STUDY OF A TYPICAL LWR PIPING SYSTEM

To qualitatively assess any benefits that might be realized by using non-linear analysis methods in the design of LWR piping systems, an actual LWR piping system was selected for a comparative study. This piping system was analyzed for gravity, pressure, and earthquake loadings using three methods: conventional linear-elastic spectrum analysis, linear-elastic time-history analysis, and nonlinear time-history analysis. Piping system responses obtained from these three analyses were compared, and the results of the comparison were studied for possible design implications. (A detailed review of current piping design loads and design codes in use is given in Part 2 of this report.)

Linear-elastic spectrum analysis is the method most commonly used for seismic analysis (both OBE and SSE) of LWR piping systems. The method is approximate and predicts only the maximum response of a piping system to an earthquake defined by a set of spectra. Due to cost considerations, only the natural modes of the piping system, with natural frequencies below a somewhat arbitrary upper limit frequency, are considered. The modal responses are combined by techniques such as the absolute sum or the SRSS. Also, the spectral analyses are performed, in general, for planar earthquakes with one horizontal and one vertical component, and the maximum responses obtained from the analyses in two perpendicular directions are combined, again using such techniques as the absolute sum or the SRSS. In addition to the uncertainties and possible over-conservatism introduced by combining modal responses and the responses in two perpendicular directions using SRSS or other techniques, further conservatism is introduced into the analysis by using envelope response spectra corresponding to the envelope of spectra at various supports. Furthermore, because of the linear-elastic assumptions inherent in the spectrum analysis, the nonlinearities in the piping system and in the response cannot be accounted for.

Nonlinear time-history analysis of a piping system, if performed accurately, should predict the response of the system to a given earthquake motion very precisely. Therefore, if the response obtained from the nonlinear time-history analysis is assumed to correspond to the true response of the piping system, a comparison of this response with the response obtained from the

conventional spectral analysis should allow the assessment of the spectral analysis method with respect to its conservatism in predicting the true response. The differences between the responses from the two analyses (i.e., the spectral and nonlinear time-history analyses) can be due to the analysis method used and the consideration of the nonlinearities in the system. Therefore, a third analysis, namely the linear-elastic time-history analysis, was performed. Comparison of piping system responses obtained from the three analyses allows the effects on the safety margins of using different analysis methods, as well as the effects of considering the system nonlinearities, to be delineated.

Selection of the Piping System

The criteria used for selecting an example piping system for this study were as follows:

- The selected system should be a Class 1 piping system
- The system should be subjected to typical LWR operating pressures and temperatures
- The diameter of the piping should be between 6 and 12 in.
- The system should have several straight runs with multiple supports.
- The system should have supports representing as many types of nonlinearities as possible

Several piping systems from actual LWR power plants were considered, and an injection line of an actual PWR plant was selected for the study. The example piping system is a 10-in.-diameter Schedule-140 piping system with a design pressure of 2,250 psi and a design temperature of 544°F. Figure 3-1 shows an isometric diagram of this piping system.

Selection of the Computer Codes

For the linear-elastic spectrum analysis, the PIPESD computer code was used. This code was developed by URS/Blume and has been used extensively by the nuclear industry for piping analysis (see the Appendix for a description of the PIPESD code). For the linear-elastic and nonlinear time-history analyses, a URS/Blume in-house version of the ANSR-1 code was used. The reasons for selecting ANSR-1 were:

- Availability of the code in-house and its familiarity to the URS/Blume staff
- Its cost efficiency, especially for nonlinear analysis
- Its flexibility for making modifications and additions and the ease with which alterations can be made
- Its availability on the Lawrence Berkeley Laboratory computer, for which the costs are much lower compared with the commercial networks

For generating the floor response spectra and the matching time histories, the URS/Blume version of the SAP IV code (see Appendix) and the URS/Blume proprietary code SMSPC, respectively, were used.

Description of the Loads

Descriptions of the loadings considered in the analyses of the example piping system are given below.

Gravity and Pressure Loads. For gravity loading, the weight of the pipes, their contents, and their insulation and the weight of the valves were considered. An internal pipe pressure of 2,250 psi was used. No analysis was made for the steady-state temperatures.

Seismic Loading for Spectral Analysis. For seismic analysis using the linear response spectrum method, two floor acceleration response spectra (one horizontal and one vertical) representing the Safe-Shutdown Earthquake (SSE) were developed. Each of the spectra was an envelope of several spectra determined to exist at various support points along the pipe. It was assumed for the present analysis that all the supports move in phase (i.e., multiple support excitation was not considered). The final horizontal and vertical spectra are presented in Figures 3-2 and 3-3.

Seismic Loading for Linear-Elastic Time-History Analysis. As a first step towards the linear-elastic time-history analysis, it was necessary to generate time histories whose response spectra matched the corresponding horizontal and vertical spectra developed for the response spectrum analysis. These two spectra were not ground motion spectra but rather those at an

elevated height in the interior structure of the containment building. To obtain the filtered motion at the elevation where the pipe supports are attached to the structure, the ground motion time history, with horizontal and vertical components, was input to a lumped-mass model of the containment interior structure. The SAP IV code was used for this purpose.

The horizontal and vertical floor response time histories were passed through the SMSPC computer code to generate modified horizontal and vertical time histories whose response spectra match the envelope floor response spectra described in the previous section. SMSPC transforms the input time histories in the frequency domain by Fourier transformation and then scales the amplitudes of the frequency spectra to match the ordinates of the given floor response spectra at selected frequencies. Iterations are performed to make the response spectra computed from the generated time histories converge towards the input floor response spectra. Figures 3-4 and 3-5 present the final generated horizontal and vertical floor acceleration time histories, and Figures 3-2 and 3-3 compare the input floor response spectra to those obtained from the generated time histories. The matching was excellent between the two spectra. Damping of 2% was assumed in the computation of the response spectra.

The horizontal and vertical time histories were input to the ANSR-1 computer code for linear-elastic time-history analysis. A constant digitizing interval of 0.01 sec was taken to define the input time histories.

Seismic Loading for Nonlinear Time-History Analysis. The horizontal and vertical time histories generated for the linear-elastic time-history analysis were also used for the nonlinear time-history analysis.

Analytical Modeling and Computer Code Development

In all analyses of the example piping system, a lumped-mass computer model was used. This computer model is shown in Figure 3-6. The support characteristics are summarized in Table 3-1.

Using the lumped-mass model and the elastic support properties, a static analysis was performed, using the PIPESD code, for gravity and pressure

loadings. The snubber supports were assumed to be inactive for the static analysis. In accordance with current practice, the gaps at the supports were ignored, and the snubber supports were assumed to be rigid for the spectral and linear-elastic time-history analyses.

For the linear-elastic time-history analysis, the ANSR-1 code was used. The seismic loads in this case were the time histories generated to match the response spectra.

Only the support nonlinearities were considered in the present study. For the nonlinear analyses, the following types of support nonlinearities were modeled: (1) buckling in compression (e.g., as in hanger-type supports), (2) gaps between the supports and the pipes (i.e., geometrical nonlinearity), (3) yielding of supports (i.e., material nonlinearity), and (4) hydraulic snubbers.

To accommodate the requirements of the linear and nonlinear time-history analyses of the example piping system, the following elements were developed as part of this project and incorporated into the ANSR-1 code:

- Straight linear-elastic pipe element
- Curved linear-elastic pipe element
- Snubber element with provision for locking and unlocking thresholds and bleed rates

The hanger-type support was modeled by the truss element available in the ANSR-1 program. The load-deformation relationship for this element is shown in Figure 3-7.

The supports with gaps between the pipe and the support may be visualized as two springs on opposite sides of the pipe with initial gaps separating them, as illustrated in Figure 3-8. Figure 3-8 presents the load-deformation relationship for such an element. Note that the gap seems to open up more and more as the springs go into inelastic ranges with cyclic loadings. The yielding in supports is represented by the yield load YFP or YFN , and the load-deformation part with reduced stiffness is represented by $K2P$ or $K2N$.

The hydraulic snubber is very difficult to model accurately, and only the most important parameters are chosen to represent it. Conceptually, the element is assumed to be attached to any specified node on the piping system (Figure 3-9a) with node I defining the direction of the snubber support. The load-deformation relationship is presented in Figure 3-9b. The load on the snubber remains zero as long as the gap δ is not closed.

Once the gap is closed, the snubber starts picking up load due to both stiffness K_1 and viscosity C_1 . When the velocity of node N exceeds a specified locking velocity, the snubber locks and the load is picked up at a steeper rate with stiffness and viscosity coefficients K_2 and C_2 . The load on the snubber drops to zero if displacement is reversed. If velocity drops below the locking velocity, the stiffness and viscosity coefficients drop back to K_1 and C_1 . This snubber element was coded and incorporated in the ANSR-1 program as part of this study.

Some convergence problems were encountered in the nonlinear time-history analysis when all the nonlinearities in the system were included in the same computer model. The problem was traced to the supports with bilinear characteristics that also included gap elements. This yielding support-gap combination apparently requires extremely small time steps during iteration by the iteration routine available in the ANSR-1 code. Attempts were made to use an alternative code, ANSYS. However, cost estimates indicated that the cost of a 10-sec time-history run would be prohibitively high.

In lieu of one nonlinear time-history analysis that included all the support nonlinearities, two sets of nonlinear analyses were made using the ANSR-1 code. For the first set, the supports were assumed to have bilinear properties without any geometric gaps between the supports and the pipe. The yielding force at which the supports were assumed to yield was taken to be 80% of the maximum support reactions obtained from the linear ANSR time-history run. This was done under the assumption that the support designs were perfectly balanced, with each having a ductility factor of 1.2 as provided by the ASME code. This analysis is termed *bilinear analysis*. A strain-hardening stiffness equal to 10% of the elastic stiffness of the supports was assumed. A direct-integration time-history approach was used with the same time histories as in linear ANSR-1 analysis.

For the second set, the supports were assumed to have geometric gaps and linear stiffnesses. This run is termed *gap-linear* ANSR-1 analysis. The hanger elements, however, were assumed to take tension load only and to buckle in compression in both sets of analyses. The snubber elements were assumed to have nonlinear properties as described previously. A direct-integration time-history approach was applied using the same time histories as in the previous analyses.

Results of the Analyses

Since several analyses of the example piping system were done with different computer codes and different analysis techniques, proper parameters had to be selected to compare the results of these analyses. The parameters selected for comparison were the maximum pipe displacements, support reactions, and pipe stresses.

Dead-load analysis was performed by PIPESD code only. For the ANSR-1 analyses, initial forces in the system as obtained for the PIPESD dead-load analysis were considered. The displacements, stresses, and reactions computed by the program were, however, relative to the dead-load state. Therefore, the comparisons between the analyses are for seismic loads only.

Dynamic characteristics of the piping system as obtained from the eigenvalue computations in PIPESD show that the first three modal frequencies with predominant motions in the north-south (x), vertical (y), and east-west (z) directions are 6.9, 6.3, and 7.6 Hz, respectively. Following the NRC Regulatory Guide, only the modes below the cutoff modal frequency of 33 Hz were considered for the response spectrum analyses. There were nine modes below this cutoff frequency.

Table 3-2 presents the maximum horizontal and vertical displacements at selected nodes for the following four cases:

- Case 1: Linear-elastic spectral analysis using the PIPESD code
- Case 2: Linear-elastic time-history analyses using the ANSR-1 code
- Case 3: Nonlinear time-history analyses with non-yielding gap supports using the ANSR-1 code

Case 4: Nonlinear time-history analysis with bilinear supports using the ANSR-1 code

The results shown in Table 3-2 indicate that the maximum displacement responses for the first 10 sec of the linear-elastic time-history analysis (Case 2) are, in general, smaller than the responses obtained from the spectral analysis (Case 1). The largest displacements are given by Case 3, which is the nonlinear analysis with nonyielding supports containing gaps. Case 4 displacements are of a magnitude similar to Case 1 and larger than Case 2. These results seem to point out that the conventional spectral analysis may overestimate displacement responses compared with the elastic time-history analysis. However, the more realistic displacement responses obtained from the nonlinear analyses are somewhat larger than those predicted by the conventional analysis.

Table 3-3 shows the longitudinal stresses at various nodes of the piping system model as computed by the four cases. Case 2, the linear-elastic time-history analysis, has resulted in the smallest piping stresses. Stresses of Case 4 (with yielding supports) are of the same order of magnitude as the Case 1 (spectral analysis) stresses. However, Case 4 analysis with gap supports indicates significantly higher pipe stresses compared with the conventional analysis (Case 1).

A comparison of the support reactions obtained for the four analyses is given in Table 3-4. Again, the reactions predicted by the conventional spectral analysis are larger than those predicted by the linear-elastic time-history analysis. Analysis with yielding supports (Case 4) indicates reactions of magnitudes similar to the ones predicted by the conventional analysis. However, the analysis with linear supports containing gaps (Case 3) results in reactions that are much larger than those predicted by any other analysis.

Discussion of Results and Conclusions

The comparison of maximum displacements, reactions, and pipe stresses obtained from the four analyses presented in the previous section indicates that, for the specific piping system and loading selected for the analysis:

- Conventional spectral analysis tends to overestimate piping response compared with linear-elastic time-history analysis.
- When nonlinearities in the piping system are considered, the response is, in general, larger than that predicted by the conventional linear analyses. If the nonlinear analysis is presumed to predict the true response more closely, then it follows that the conventional spectral analysis can sometimes produce nonconservative results.
- The most significant support nonlinearities that affect the piping system response seem to be the gaps. The impact between the pipe and its supports causes large reactions, stresses, and displacements within the pipe system.

These conclusions are subject to some limitations due to the approach taken within the scope of this study. One limitation is that the study was done for a single piping system and therefore its conclusions cannot be generalized. To derive general conclusions, it would be necessary to study several such piping systems. Also, the response of a piping system to different earthquake time histories can vary considerably depending on the frequency content of the time history, its duration, and its periodicity. Therefore, several representative earthquake time histories would be required for a comprehensive study leading to generalized conclusions.

Another limitation of the study is the inadequacy of the available nonlinear analysis codes for the problem under consideration. Numerical problems were encountered when gap elements were used together with yielding supports. The nonlinear analysis results need to be confirmed by using other codes or, better yet, with experimental studies (e.g., shaking table tests).

Even though the conclusions of this study cannot be generalized without further, more detailed studies, indications are that the conventional analysis methods may be unrealistic and may lead to unconservative designs. Therefore, further investigations into the true behavior of piping systems under seismic loads are required before an assessment of the analysis methods on the design of piping systems can be made. The behavior of supports with gaps that may yield during an earthquake is most significant with respect to the overall response of the pipe system and requires special attention.

Recommendations for Further Studies

This study, even though its conclusions cannot be generalized because of its limited scope, has raised concern with regard to two specific items. One item, as described above, is the combination of gap and yielding support. Although the behavior of this type of support is not well understood, it could have a very significant effect on the overall response of the pipe system. The possible behavior of this type of support is such that at every contact between the pipe and the support large reactions develop from impact forces. These reactions lead to yielding of the supports, which result in enlarged gaps. If the gaps get progressively larger, the impact forces also will tend to get larger, and progressive failures of supports are possible. Therefore, further analytical and experimental studies concerning the behavior of piping systems containing this type of nonlinearity would be strongly recommended.

The other significant item that raised concern was the limitations of the existing nonlinear analysis capabilities. Convergence problems were incurred when physical gaps were assumed to exist between the supports and the piping. No convergence at all was obtained for the case where supports were modeled to have bilinear yielding load-deformation relationships with specified gaps. A time step of 0.0001 sec was chosen for this case, and 100,000 time steps were required for a 10-sec-duration earthquake. It was estimated that the contact-impact problem due to the closing of the gap requires extremely small time steps, perhaps of the order of 10^{-6} sec. Such small time step requirements make it forbiddingly expensive to do nonlinear analyses of a realistic piping system, such as the one studied, subjected to an earthquake load of reasonable duration. Therefore, there is a significant need to develop special-purpose nonlinear piping analysis codes that are in the public domain and are cost-effective. Development of such codes will also make detailed analytical studies of piping systems possible and may lead to improved designs.

TABLES

TABLE 3-1
SUPPORT CHARACTERISTICS FOR THE EXAMPLE PIPING SYSTEM

Support Number	Direction	K (10 ⁶ lb/in.)	Gap (in.)	Type
1	+Y	2.8	0.0625	Restraint
	-Y	8.9	--	
	(+X)	2.6	0.0625	
	(-X)	7.3	0.0625	
2	+Y	4.6	0.125	Restraint
	-Y	1.1	--	
3	+Y	4.3	0.125	Restraint
	-Y	6.0	--	
4	+X	3.0	0.0625	Restraint
	-X	3.0	0.0625	
5	+Y	0	--	Spring Hanger
	-Y	0.0009	--	
6	+X	7.0	0.0625	Restraint
	-X	7.0	0.0625	
7	+Y	1.0 × 10 ⁶	0.0625	Restraint
	-Y	1.0 × 10 ⁶	--	
8	+Y	1.0 × 10 ⁶	0.0625	Restraint
	-Y	1.0 × 10 ⁶	--	
9	+X	0.5	0.0625	Restraint
	-X	0.5	0.0625	
10	+Y	15.0	0.0625	Restraint
	-Y	15.0	--	
11	+Z	1.0 × 10 ⁶	--	Snubber
	-Z	1.0 × 10 ⁶	--	
12	+Y	0	--	Spring Hanger
	-Y	0.00068	--	
13	+Z	2.4	--	Snubber
	-Z	2.4	--	
14	+Y	0	--	Spring Hanger
	-Y	0.0008	--	
15	+Y	0	--	Spring Hanger
	-Y	0.0006	--	

TABLE 3-2
DISPLACEMENT RESPONSE (in in.) AT SELECTED NODES
FOR EAST-WEST AND VERTICAL SEISMIC LOADING

Node	Case 1: Spectral Analysis	Case 2: Linear-Elastic Time-History Analysis	Case 3: Gap Supports	Case 4: Bilinear Supports
X Displacement				
16	0.0173	0.0142	0.0393	0.1685
19	0.0177	0.0145	0.0395	0.1713
23	0.3589	0.3395	0.5152	0.5283
24	0.3475	0.2126	0.3621	0.7207
29	0.4365	0.2638	0.3606	0.9091
30	0.3447	0.2074	0.2806	0.7461
Y Displacement				
16	0.0216	0.0135	0.0189	0.1433
19	0.0905	0.0699	0.1159	0.2198
23	0.7130	0.5781	0.8086	1.1423
24	0.8451	0.6332	0.8388	1.1154
29	0.5103	0.3747	0.4909	0.6914
30	0.3415	0.2530	0.3309	0.4884
Z Displacement				
16	0.6744	0.4470	0.5821	0.6404
19	0.3448	0.1987	0.3017	0.4895
23	0.5815	0.4658	0.7164	0.5811
24	0.3614	0.3021	0.4700	0.4499
29	0.0945	0.0710	0.1353	0.2486
30	0.0933	0.0726	0.1273	0.2452

TABLE 3-3
LONGITUDINAL STRESSES (in psi) IN PIPE ELEMENTS
FOR EAST-WEST AND VERTICAL SEISMIC LOADING

Element	Node	Case 1: Spectral Analysis	Case 2: Linear-Elastic Time-History Analysis	Case 3: Gap Supports	Case 4: Bilinear Supports
14	15	0.2733E+04	0.1975E+04	0.2338E+05	0.2786E+04
14	16	0.3681E+04	0.2476E+04	0.2559E+05	0.3475E+04
15	17	0.5487E+04	0.3749E+04	0.2512E+05	0.5178E+04
17	18	0.5166E+04	0.3346E+04	0.2480E+05	0.4514E+04
17	19	0.6213E+04	0.4886E+04	0.2558E+05	0.7439E+04
18	20	0.1146E+05	0.8401E+04	0.2823E+05	0.1274E+05
21	22	0.4413E+04	0.2365E+04	0.2329E+05	0.4031E+04
21	23	0.8917E+04	0.4805E+04	0.2769E+05	0.8614E+04
22	24	0.7684E+04	0.4548E+04	0.2476E+05	0.6609E+04
23	25	0.1134E+05	0.6946E+04	0.2234E+05	0.1234E+05
27	27	0.7107E+04	0.4109E+04	0.2650E+05	0.8274E+04
27	29	0.4261E+04	0.2897E+04	0.2323E+05	0.5206E+04
28	30	0.4474E+04	0.2817E+04	0.2766E+05	0.4278E+04
30	32	0.1145E+05	0.7440E+04	0.1915E+05	0.1053E+05

TABLE 3-4
SEISMIC SUPPORT REACTIONS (in lb) FOR EAST-WEST
AND VERTICAL SEISMIC LOADING

Node	Direction	Case 1: Spectral Analysis	Case 2: Linear-Elastic Time-History Analysis	Case 3: Gap Supports	Case 4: Bilinear Supports
5	Y	3,417	4,186	141,325	6,082
5	X	1,095	1,793	149,419	1,801
8	Y	18,577	11,883	88,015	12,796
10	Y	13,041	11,105	126,127	11,837
10	X	4,216	3,173	86,345	4,014
12	Y	55	35	51	50
13	X	3,082	2,400	152,753	3,646
15	Y	9,474	7,718	58,325	8,129
17	Y	7,194	5,817	58,026	9,501
18	X	8,574	7,045	54,084	7,785
20	Y	12,325	8,973	243,611	13,605
20	Z	7,418	6,946	33,036	8,289
22	Y	396	291	515	458
25	Y	676	506	830	671
25	Z	27,706	19,844	41,998	20,249
33	Y	1	1	2	1

FIGURES

FIGURE 3-1 EXAMPLE PIPING SYSTEM: INJECTION LINE FOR A PWR PLANT

FIGURE 3-2 INPUT AND GENERATED HORIZONTAL FLOOR RESPONSE SPECTRA

FIGURE 3-3 INPUT AND GENERATED VERTICAL FLOOR RESPONSE SPECTRA

FIGURE 3-4 FLOOR ACCELERATION TIME HISTORY GENERATED FROM HORIZONTAL SPECTRUM

FIGURE 3-5 FLOOR ACCELERATION TIME HISTORY GENERATED FROM VERTICAL SPECTRUM

FIGURE 3-6 LUMPED-MASS COMPUTER MODEL OF EXAMPLE PIPING SYSTEM

FIGURE 3-7 HANGER SUPPORT LOAD-DEFORMATION BEHAVIOR

(a) Gap Element Configuration

(b) Load-Deformation Relationship of Gap Element

FIGURE 3-8 GAP ELEMENT MODEL

(a) Element Orientation

(b) Element Behavior

FIGURE 3-9. SNUBBER ELEMENT MODEL

APPENDIX

Survey of Computer Codes with Linear and Nonlinear Analysis Capabilities

Program name: STARDYNE

Developed by: Mechanics Research Inc.
9841 Airport Blvd.
Los Angeles, CA 90045

Maintained by: Systems Development Corporation
9841 Airport Blvd.
Los Angeles, CA 90045

Operational on: CDC only

Commercially available: Under license from CDC only

Documentation level: User's manual

Program features: STARDYNE is a general-purpose finite-element program for the static and dynamic analysis of elastic structures. It is based on the displacement method of analysis. It has capabilities for Guyan reduction, eigenvalue extraction (inverse power, Givens, and Householder), modal and direct (Newmark Beta and Wilson Theta) transient response, frequency response, seismic time-history (using relative displacements), and response spectra analysis. The program has a restart capability and permits the use of substructures. STARDYNE has several elements for piping analysis such as straight pipes, beams, hangers, linear and nonlinear springs, elbows and tees (using flexibility factors), and gap elements. The program has no capabilities for large deformations or elastic-plastic analysis nor does it have a special capability for mesh generation of piping systems and stress computations in the manner prescribed by the piping codes.¹

Program name: PIPESD

Developed by: URS/John A. Blume & Associates, Engineers
130 Jessie Street
San Francisco, CA 94105

Maintained by: Developer

Operational on: CDC only

Commercially available: Under license from CDC only

Documentation level: User's manual

Program features: The PIPESD (pipe static and dynamic analysis) is a comprehensive special-purpose computer program designed to perform linear-elastic analysis of three-dimensional piping systems subjected to sets of user-prescribed static, thermal, and dynamic (earthquake) loadings.² The dynamic response of the piping system to earthquake loading is determined using the response spectrum, modal superposition approach. Natural frequencies and mode shapes are calculated using the Householder-QR technique.

PIPESD performs stress calculations in accordance with ASME Section III, Class 1 and 2, and ANSI B31.1 power piping specifications.

Program name: ADLPIPE
 Developed by: Arthur D. Little, Inc.
 Acorn Park
 Cambridge, MA 02140
 Maintained by: Developer
 Operational on: IBM, CDC, UNIVAC
 Commercially available: On a royalty basis, through license agreement with the developer, or under license from CDC and McDonnell Douglas Automation Company
 Documentation level: User's manual and some verification example problems
 Program features: ADLPIPE is a special-purpose program for the static and dynamic analysis of complex piping systems. It is basically an elastic analysis that uses the method of transfer matrices in its formulation. It has capabilities for modeling straight pipes, springs, elbows (flexibility and intensity factors), and linear constraints and also has a general stiffness matrix input capability. A static condensation scheme is available, and eigenvalues and eigenvectors for shock response spectra analysis can be obtained by the Jacobi, Givens, or Householder methods.¹
 Stresses can be computed in accordance with a number of piping codes, including ANSI B31.1, Power Piping; ANSI B31.3, Petroleum Refinery Piping; ANSI B31.4, Liquid Petroleum Transportation Piping Systems; and ASME Section III, Class 1, 2, and 3.³

Program name: PIPDYN

Developed by: Franklin Institute Research Laboratories
20th Street and Franklin Parkway
Philadelphia, PA 19103

Maintained by: Developer

Operational on: IBM, CDC, UNIVAC

Commercially available: On a royalty basis from the developer or purchase from the developer, Utility Network of America, or Westinghouse Electric Corporation

Documentation level: User's manual, theoretical manual, and examples manual

Program features: PIPDYN is a finite-element piping program based on the displacement method of analysis. It has a capability for linear-elastic analysis with temperature-dependent material properties and proportional viscous damping. It can handle concentrated pressure, gravity, stationary, and transient thermal and sinusoidal types of loading. It is also capable of handling prescribed and time-dependent displacement boundary conditions, as well as velocity and acceleration inputs. The program has a capability for condensation of the dynamic degrees of freedom. Using a modal formulation response spectra, harmonic response, transient, and reduced modal transient analyses can be obtained. The eigenvalue routines are based on the Givens and Householder-QR methods. A direct transient analysis capability using explicit time integration is also available.¹

Program name: PISTAR

Developed by: Nuclear Technology, Inc., and
McDonnell Douglas Automation Company

Maintained by: McDonnell Douglas Automation Company

Operational on: IBM

Commercially available: Under license

Documentation level: User's manual and examples manual

Program features: PISTAR is a special-purpose linear-elastic piping program designed for static analysis, modal extraction, response spectrum analysis, dynamic time-history analysis, and transient one-dimensional heat-transfer analysis. PISTAR uses extensive built-in and streamlining features to eliminate most of the tedious research and cross-checking required to input standard data into a piping system analysis. The program performs analysis and evaluations in accordance with ASME Section III, Class 1, 2 and 3, and ANSI B31.1 and B31.7 specifications.⁴

Program name: SAP IV

Developed by: E. L. Wilson
College of Engineering
University of California
Berkeley, CA 94720

Maintained by: Earthquake Engineering Research Center (EERC)
University of California
Berkeley, CA 94720

Operational on: IBM, CDC, UNIVAC

Commercially available: Purchase at a nominal fee from EERC

Documentation level: User's manual and examples manual

Program features: The SAP IV program is a general-purpose finite-element program for linear-elastic analysis of structural systems. The program has capabilities for static, eigenvalue-eigenvector extraction (determinant method, subspace iteration method), dynamic modal, and direct (Wilson Theta) transient response analysis. In addition, seismic response spectra and time histories based on relative displacements can be obtained with the program. A restart option is available to the user.

SAP IV can perform the elastic static and dynamic analyses of piping systems consisting of straight pipe, elbow, spring, and beam elements. The program does not have a tee element. A variety of loadings and boundary conditions can be applied, including ground accelerations.¹

Program name: NUPIPE II

Developed by: Nuclear Services Corporation
1700 Dill Avenue
Campbell, CA 95008

Maintained by: Developer

Operational on: IBM, CDC

Commercially available: On a royalty basis

Documentation level: User's manual

Program features: The program NUPIPE II is a special-purpose computer program that performs linear-elastic analysis of three-dimensional piping systems subjected to thermal, static, seismic, and other dynamic time-history loads. The analysis is performed in accordance with the specifications of ASME Code, Section III, for Class 1, 2 and 3 components, and ANSI B31.1 for power piping.

NUPIPE II is designed to minimize the input-output effort involved in conventional piping analyses. The code has optional capability to operate in metric units.

Dynamic analyses are performed using a lumped-mass model with response spectrum, modal superposition, or time-history integration methods.⁵

Program name: ANSYS Revision 3

Developed by: DeSalvo and J. A. Swanson.
Swanson Analysis Systems, Inc.
P.O. Box 65
Houston, PA

Maintained by: Developer

Operational on: IBM, CDC, UNIVAC

Commercially available: Under license from CDC and utility networks

Documentation level: User's manual, theory manual, and examples manual

Program features: ANSYS is a large, three-dimensional finite-element program capable of structural, heat transfer, and fluid-flow analysis. ANSYS performs both transient (time-history) and harmonic (response spectra) dynamic, linear and nonlinear, elastic and elasto-plastic analysis of structures.

The element library includes elastic and elasto-plastic pipe elements, a spring element with linear or nonlinear damping, a gap element, and a friction element. The library does not contain a tee element.

ANSYS has the capability of generating substructures (or superelements). These substructures may be stored in a library file for use in other analyses. Also dynamic (or kinematic) matrix condensation may be used to reduce the number of dynamic degrees of freedom.

The nonlinear capabilities include plasticity (small strain), creep (thermal- and irradiation-induced), irradiation-induced swelling, large deflection, and buckling. The dynamic capabilities include eigenvalue-eigenvector, steady-state harmonic response, and linear and nonlinear transient response. The materials may be either isotropic or anisotropic and may include nonlinear temperature dependency. This program does not have a concrete-type material with tension cutoff.

The dynamic analysis employs a consistent mass matrix and an explicit quadratic integration routine. Extensive plotting capabilities exist, including geometry, stresses, displacements, and temperatures.

Loadings on the structure may be forces, displacements, pressures, temperatures, or response spectra. Loadings may be arbitrary time functions for linear and nonlinear dynamic analyses.

A postprocessor exists for processing ANSYS solutions of piping networks to make the results more meaningful for code interpretation and evaluations. The results are presented for the ASME Boiler and Pressure Vessel Code, Section III, Subsection NC (Class 2 components), and ANSI B31.1 Power Piping.

Program name: MARC

Developed by: MARC Analysis Research Corporation
260 Sheridan Avenue
Suite 200
Palo Alto, CA 94306

Maintained by: Developer

Operational on: IBM, CDC, UNIVAC

Commercially available: Purchase under license from McDonnell Douglas Automation Company or CDC or via a utility network (on a royalty basis). Also available from the developer.

Documentation level: User information manual, program input manual, and demonstration problems manual

Program features: MARC is a general-purpose three-dimensional finite-element program designed for the linear and non-linear analysis of structures in the static and dynamic regime. The dynamic analysis may be carried out by either the modal or the direct integration procedure. The direct integration is performed using the Newmark Beta method of step-by-step integration ($\beta = 1/4$, $\gamma = 1/2$) or the Houbolt operator, or the central difference (explicit) operator.⁶

However, the program does not presently have a capability for response spectra or acceleration time-history loading, thereby inhibiting application to seismic analysis.

The element library includes elastic and elastic-plastic pipe elements, gap and friction elements, and linear and nonlinear spring elements. A special element suitable for elastic-plastic and large deformation analysis of elbows is available. This element allows for deformation of the elbow cross-section. MARC does not have a tee element.¹ The elastic-plastic and large displacement analysis is achieved with a series of piecewise linear increments. A Von Mises yield criterion with isotropic strain hardening is used with a Lagrangian formulation of the large displacement equations of motion.

Viscous damping may be introduced by assuming a system damping matrix that is a linear combination of the mass and stiffness matrices of the system.

Program name: WECAN

Developed by: Westinghouse Electric Corporation
1310 Beulah Road
Pittsburgh, PA 15235

Maintained by: Westinghouse Electric Corporation
Advanced Systems Technology
2040 Ardmore Boulevard
Pittsburgh, PA 15221

Operational on: CDC

Commercially available: On royalty basis from Westinghouse

Documentation level: User's manual, demonstration manual, pre- and post-processor manual, and verification manual

Program features: WECAN is a large, general-purpose finite-element program. It is based on the finite-element displacement method of analysis. The program can perform static analysis, thermal and flow analysis, modal analysis, harmonic response analysis, and linear and nonlinear dynamic transient analysis.

WECAN's dynamic capabilities include Guyan reduction, several routines for eigenvalue extraction (inverse power, Jacobi, Householder-QR), frequency response, direct transient response (Houbolt, Newmark Beta), seismic time-history, and response spectrum analysis. For piping analysis, WECAN's element library contains straight and curved pipes, linear and nonlinear springs, hangers, spring-dampers, and gap and friction elements. The program has a restart capability for linear and nonlinear dynamic analysis.

Program name: ANSR-II

Developed by: D. P. Mondkar and G. H. Powell
Division of Structural Engineering and Structural
Mechanics
University of California
Berkeley, CA 94720

Maintained by: Developer

Operational on: CDC

Commercially available: Purchase from University of California

Documentation level: User's manual and theory manual

Program features: ANSR-II is a general-purpose three-dimensional finite-element program capable of either static or direct time-history dynamic analysis. Both material and geometric nonlinearities may be considered. Dynamic loading may consist of nodal force or ground acceleration time histories and may be different from one point of application to another so that out-of-phase motions and relative anchor movement effects can be modeled.

New finite elements and/or material models can be incorporated conveniently because of the modular structure of the program. Computations associated with specific element types, materials, and nonlinear modes of behavior are carried out within subroutines that constitute the element library. ANSR-II incorporates an efficient and flexible equation-solving strategy, in which the program user can specify step-by-step, constant stiffness iteration, or Newton-Raphson iteration schemes, or combinations of these schemes. This is desirable because no single scheme is optimal for all types of nonlinear behavior.

The element library includes a number of elastic and inelastic one-dimensional bar and support elements, a gap element, elastic and inelastic straight beam/pipe elements, and an elastic pipe bend element. Most elements have been developed for research rather than production purposes. The library does not contain a tee element.

Viscous damping may be imposed on the dynamic structural response of elements by assuming that the viscous damping results from a linear combination of mass-dependent and stiffness-dependent effects. Stiffness-proportional damping may be dependent on the tangent stiffness or the initial stiffness.

Program name: ADINA

Developed by: K. J. Bathe
Mechanical Engineering Department
Massachusetts Institute of Technology
Cambridge, MA 02139

Maintained by: Developer

Operational on: CDC

Commercially available: Under license from CDC

Documentation level: User's manual, theory manual, and examples manual

Program features: ADINA is a three-dimensional finite-element program capable of linear and nonlinear dynamic structural analysis.

The element library includes elastic and elastic-plastic beam elements and linear and nonlinear truss elements, which may be used to model springs.

The nonlinearities may be due to large displacements, large strains, and nonlinear material behavior. The Von Mises or Drucker-Prager yield condition may be adopted.

In dynamic analysis, implicit time integration (the Newmark or Wilson methods) or explicit time integration (central difference method) can be employed. A modified Newton iteration is used in the nonlinear incremental analysis, with the nonlinearities being dealt with in the time integration. The total Lagrangian or the updated Lagrangian formulation may be used for large displacement and large strain analysis.

Damping can only be specified in the form of concentrated nodal point dampers. The loading history is generally given as a force history at a node. This imposes a severe restriction on the use of the program for seismic analysis because earthquake records are generally measured as acceleration time histories of the ground motion.

Program name: NASTRAN

Developed by: National Aeronautics and Space Administration
Washington, D.C.

Maintained by: MacNeal-Schwendler Corporation
7442 Figueroa Street
Los Angeles, CA 90041

Operational on: IBM, CDC, UNIVAC

Commercially available: Under license from MacNeal-Schwendler Corporation
or CDC or on a royalty basis from COSMIC, University
of Georgia, Athens, GA

Documentation level: User's manual, theory manual, and examples manual

Program features: NASTRAN is a large, three-dimensional finite-element
program for the dynamic analysis of structures, using
frequency response, modal, and direct (Newmark
Beta) time integration methods. The element library
contains straight elastic pipe elements, linear and
nonlinear springs, and gap and friction elements.
The program does not have elbow and tee elements
for piping analysis.

The time-history response of a structure due to
ground motion can be computed with NASTRAN by using
special techniques. Its capabilities for elastic-
plastic analysis are limited. Nonlinear elements,
including multipliers and arbitrary function gener-
ators that simulate Coulomb damping, plastic defor-
mation, and other nonlinear relationships are avail-
able for transient dynamic analysis.

Damping properties can be specified as viscous
dampers, material damping coefficients, direct in-
put matrices, and modal damping factors.

**Checklist of Piping
Program Capabilities**

CHECKLIST OF PIPING PROGRAM CAPABILITIES: ANALYTICAL/LOAD CAPABILITIES

Analytical/Load Capability		Program												
		STARCYN	PIPESD	ADLPIPE	PIPDYN	PISTAR	SAP IV	NUPIPE II	ANSYS REV 3	MARC	WECAN	ANSR-II	ADINA	NASTRAN
Static	Deadweight	X	X	X	X	X	X	X	X	X	X	X	X	X
	Thermal	X	X	X	X	X	X	X	X	X	X	X	X	X
	Pressure	X	X	X	X	X	X	X	X	X	X	X	X	X
	Anchor movement	X	X	X	X	X	X	X	X	X	X	X	X	X
Dynamic	<u>Modal response spectra:</u>													
	Seismic	X	X	X	X	X	X	X	X		X			X
	<u>Modal time history:</u>													
	Seismic	X		X	X	X	X		X					
	<u>Linear-elastic direct time integration:</u>													
	Seismic	X	X	X	X	X	X	X	X	X	X	X	X	X
	Thermal		X	X	X	X			X	X			X	
	<u>Materially nonlinear time integration:</u>													
	Seismic								X	X	X	X	X	X*
	<u>Geometrically nonlinear time integration:</u>													
	Seismic								X	X	X*	X		X*

*Limited capability

CHECKLIST OF PIPING PROGRAM CAPABILITIES: ELEMENT LIBRARY

Element Library	Program												
	STARDYN	PIPED	ADLPIPE	PIPDYN	PISTAR	SAP IV	NUPIPE II	ANSYS REV 3	MARC	WECAN	ANSR-II	ADINA	NASTRAN
Straight pipe	X	X	X	X	X	X	X	X	X	X	X	X	X
Elbow	X	X	X	X	X	X	X	X	X	X	X		
Tee	X	X	X	X	X		X						
Reducer		X	X		X		X						
Linear spring	X	X	X	X	X	X	X	X	X	X	X	X	X
Dash pot								X		?			
Gap	X							X	X	X	X	X*	X
Nonlinear spring	X							X	X	X	X	X	
Spring damper with nonlinear damping									X				
Friction element								X	X	X			X
Plastic hinge with strain hardening								X			X	X	
Hanger (strut with compression buckling)								X	X		X		

*Modeled with bilinear strain-hardening element

CHECKLIST OF PIPING PROGRAM CAPABILITIES: SPECIAL FEATURES

Special Features	Program												
	STARDYN	PIPESD	ADLPIPE	PIPDYN	PISTAR	SAP IV	NUPIPE II	ANSYS REV 3	MARC	WECAN	ANSR-II	ADINA	NASTRAN
Bandwidth optimization	X	X	X	?	X		X	X*	X*	X	X	X	X
Free format input		X			X			X					
Save and recall eigensolution, restart	X	X		X	X	X		X	X	X		X	X
Hangar design/load tables		X	X		X		X						
Specify elements of 6 x 6 stiffness matrix	X		X	X		X		X				X	
Independent motion for multiple supports	X							X			X	X	
Condensation	X	X	X					X	X	X		X	X
Substructuring	X							X				X	X

*Wave front solution

CHECKLIST OF PIPING PROGRAM CAPABILITIES: PREPROCESSING

Preprocessing	Program												
	STARDYN	PIPED	ADLPIPE	PIPDYN	PISTAR	SAP IV	NUPIPE II	ANSYS REV 3	MARC	WECAN	ANSR-II	ADINA	NASTRAN
<u>Loading input may be:</u>													
Force time history	X	X	X	X	X		X	X	X	X	X	X	
Displacement time history					X			X	X	X		X	
Velocity time history								X					
Acceleration time history	X	X		?	X	X	X	X			X	X	X
Displacement response spectra	X				X			X					
Velocity response spectra								X					
Acceleration response spectra	X	X	X	X	X	X	X	X		X			
<u>Input data preparation:</u>													
Specifically for piping	X	X	X	X	X		X						

CHECKLIST OF PIPING PROGRAM CAPABILITIES: POSTPROCESSING

Postprocessing	Program												
	STARDYN	PIPESD	ADLPIPE	PIPDYN	PISTAR	SAP IV	NUPIPE II	ANSYS REV 3	MARC	WECAN	ANSR-II	ADINA	NASTRAN
<u>Stresses are computed to meet:</u>													
ANSI B31.1 Power Piping	X	X	X	X	X		X	X					
B31.2 Fuel Gas Piping													
B31.3 Petroleum Refinery Piping													
B31.5 L. P. Transportation Piping													
B31.6 Chemical Process Systems													
B31.7 Nuclear Power Piping		X	X	X	X			X					
B31.8 Gas Transmission and Distribution Piping													
ASME Section III, Nuclear Class 1	X	X	X	X	X		X	X					
Components Code Class 2	X	X	X	X	X		X	X					
Class 3	X		X		X		X	X					
<u>Plotting:</u>													
Static Deformation	X	X		X	X			X	X	X			X
Dynamic: Mode Shapes		X	X	X	X			X	X				
Displacement Time History	X				X	X		X			X		
Strain Time History								X					
Stress Time History			X										
Force Time History	X			X	X						X		

COMPUTER SURVEY

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GLOSSARY OF ABBREVIATIONS

ADS	Automatic depressurization system
A/E	Architect/Engineer
AEC	Atomic Energy Commission
ANSI	American National Standards Institute
AOTC	Associated Offices Technical Committee
ASME	American Society of Mechanical Engineers
ASME Code	American Society of Mechanical Engineers Boiler and Pressure Vessel Code
ATWS	Anticipated transient without scram
B&W	Babcock & Wilcox Company
BNL	Brookhaven National Laboratory
BOP	Balance of plant
BTP	Branch technical position
BWR	Boiling water reactor
C-E	Combustion Engineering, Inc.
CEA	Commissariat a l'Energie Atomique
CNEN	Comitato Nazionale per l'Energia Nucleare
CP	Construction permit
CVCS	Chemical and volume control system
DOE	Department of Energy
DRSR	Division of Reactor Safety Research
DSS	Division of Systems Safety
ECCS	Emergency core cooling system
ESF	Engineered safety features
EPRI	Electric Power Research Institute
FEED	File of Evaluated Event Data
FSAR	Final Safety Analysis Report
GE	General Electric Company

HDR	Heissdampfreaktor
HEDL	Hanford Engineering Development Laboratory
IGSCC	Intergranular Stress Corrosion Cracking
INEL	Idaho National Engineering Laboratory
KWU	Kraftwerk Union Aktiengesellschaft
LEFM	Linear Elastic Fracture Mechanics
LER	Licensee Event Report
LLL	Lawrence Livermore Laboratory
LOCA	Loss-of-coolant accident
LOFT	Loss-of-fluid test
LPCI	Low-pressure coolant injection
LPP	Lead Plant Program
LTP	Long-Term Program
LWR	Light water reactor
NRC	Nuclear Regulatory Commission
NSIC	Nuclear Safety Information Center
NSSS	Nuclear steam supply system
OBE	Operating basis earthquake
OL	Operating license
ORNL	Oak Ridge National Laboratory
PCSG	Pipe Cracking Study Group
PSAR	Preliminary Safety Analysis Report
PWR	Pressurized water reactor
RCIC	Reactor core isolation cooling system
RCPB	Ractor coolant pressure boundary
RCS	Reactor coolant system
REC	Office of Nuclear Regulatory Research
RG	Regulatory guide
RHR	Residual heat removal system (BWR system)

RHRS	Residual heat removal system (PWR system)
RSR	Reactor safety research
SIS	Safety injection system
SMiRT	Structural Mechanics in Reactor Technology
SONGS	San Onofre Nuclear Generating Station
SRP	Standard review plan
SRSS	Square root of the sum of the squares
SRV	Safety-relief valve
SSE	Safe-shutdown earthquake
SSI	Soil-structure interaction
SSMRP	Seismic Safety Margin Research Program
STP	Short-Term Program
Westinghouse	Westinghouse Electric Corporation
WPPSS	Washington Public Power Supply System

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