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LIGHT WATER REACTOR SAFETY TECHNOLOGY PROGRAM
QUARTERLY REPORT
OCTOBER - DECEMBER 1979

D. A. Dahlgren, Program Manager

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LIGHT WATER REACTOR SAFETY TECHNOLOGY PROGRAM

Quarterly Report
October-December 1979

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ACRONYMS

ASTM	American Society of Testing and Materials
B&W	Babcock & Wilcox
BWR	Boiling Water Reactor
CADDs	Computer Aided Design and Drafting System
CCF	Common Cause Failure
CE	Combustion Engineering
DASS	Disturbance Analysis and Surveillance System
DNBR	Departure from Nucleate Boiling Ratio
DOE	Department of Energy (HQ - Headquarters)
ECCS	Emergency Core Cooling System
EI	Energy Incorporated
EPRI	Electric Power Research Institute
ERD	Experiment Requirements Document
ETR	Engineering Test Reactor
FY	Fiscal Year
GE	General Electric Company
INEL	Idaho National Engineering Laboratory
LASL	Los Alamos Scientific Laboratory
LCMNA	Linear Constrained Modified Newton Algorithm
LOA	Lines of Assurance
LOCA	Loss of Coolant Accident
LOFT	Loss of Fluid Test (facility)
LWR	Light Water Reactor

Acronyms (Cont'd)

MAU	Multi-Attributed Utility
MMI	Man-Machine Interaction
NPD	Nuclear Power Division (DOE)
NRC	Nuclear Regulatory Commission
NSF	National Science Foundation
ORNL	Oak Ridge National Laboratory
PCS	Passive Containment System (NucleDyne)
PFAG	Plastic-Fracture Analysis Group
PVRC	Pressure Vessel Research Committee
PWR	Pressurized Water Reactor
RFP	Request for Proposals
SAI	Science Applications, Inc.
SSRI	Social Science Research Institute (UCLA)
TMC	Technology Management Center (Sandia Laboratories)
TMI	Three Mile Island
UCLA	University of California at Los Angeles

LIGHT WATER REACTOR SAFETY TECHNOLOGY PROGRAM

1.0 Introduction

This quarterly report covers the period October 1 through December 31, 1979, and details the progress of the research and development projects of the Light Water Reactor Safety Technology Program of the Department of Energy (DOE). The Program is managed by Sandia National Laboratories which serves as the Technology Management Center (TMC) for DOE. Under the sponsorship of the DOE Nuclear Power Development Division, the Program is intended to develop and assist implementation of light water reactor (LWR) systems and concepts which offer significant improvements in safety.

The motivation for safety improvement stems from the DOE responsibility to assist in the development of viable energy sources that can be used to meet the national objectives of reduction in oil usage and U. S. dependency on foreign oil, and stabilization of energy costs. Furthermore, it is intended that improvements in safety will improve the licensibility of reactors, accelerate the licensing process, increase plant availability, and ultimately increase public acceptance of nuclear power. Achievement of these objectives is consonant with DOE's responsibility to provide safe and reliable energy sources.

The Program was initiated in 1978 and has evolved during 1979. The areas of activity during the last quarter are reported herein and have been broadly categorized as follows:

Management Support Program

Risk Methods Utilization

Improved Safety Systems

Man-Machine Interaction

Safety Data

An Executive Summary is presented in Section II and is intended to be a concise summary of the status and progress of the work in the above areas. A more detailed summary of each area is given in Section III - Progress Reports.

2.0 Executive Summary

The following is a concise summary of the status and progress of the work in each of the program areas for the period October 1 through December 1, 1979.

2.1 Technology Management Center Support Programs

Programs in this area provide the TMC and DOE/HQ with additional, independent viewpoints on the LWR Safety Program.

2.1.1 DOE Support

Technical support was provided by LASL to the President's Commission on the Accident at Three Mile Island (Kemeney Commission). Brief studies were conducted on other LWR safety issues and NRC actions and submitted to DOE/HQ.

2.1.2 Safety Overview

The TMC is maintaining information exchange with informed members of industry, government and the technical community to obtain a broadly-based input in order to ascertain current LWR safety status. As part of this continuing effort a number of meetings were held to assess the new safety perspective. Reports resulting from the TMI accident were carefully reviewed.

2.2 Risk Methods Utilization

This area concentrates on the development of quantitative risk methods which can be applied to guide safety related design/operation trade-offs and to focus safety/licensing activity on the more important problems.

2.2.1 Acceptable Risk and Risk Evaluation

The intent of the Acceptable Risk effort is to delineate a set of workable and justifiable quantitative LWR risk acceptance criteria. The impetus for this project stems from the possibility that delineation of such risk criteria may ultimately provide standards for design guidance and could result in more expeditious licensing. This work is being done at UCLA and all tasks have been essentially completed. Four draft reports have been issued.

The purpose of the LWR Risk Evaluation Methodology project is to develop a framework for evaluating LWR risk within the context of total energy cycle risk estimates and other variables - such as benefit - that may be identified. A methodology will be developed that incorporates the more important parameters in the LWR risk issue to provide a defensible rationale for LWR risk acceptance criteria that can be used for design and licensing guidance.

2.2.2 R&D Selection Methodology

The aim of this program is the development and demonstration of a R&D selection method suitable for application to the LWR Safety Technology Program. Two different methods are being developed to assess safety improvement, cost and acceptability of LWR R&D and design choices. General Electric is following a probabilistic cost-benefit approach which entails a modification of the Lines Of Assurance (LOA) safety analysis used previously

for fast reactor safety evaluation. SAI is developing a decision method based on Multi-Attribute Utility (MAU) theory. Both contractors have concentrated on methodology development during FY79 and will perform demonstration of their respective techniques during FY80.

GE has initiated the first phase application of their R&D selection methodology. A linear-constraint modified Newton algorithm has been selected for this demonstration. Two tasks from the Preliminary LWR Safety Technology Program Plan (Improved Safety Systems and Risk Methods Utilization) were selected to demonstrate and test the methodology.

SAI has identified seven tentative sub-programs for application of their selection methodology process. A group of technical specialists provided input towards development of a complete set of utility curves for the individual subattributes and also specified ranking (weighting) factors for each of the category, attribute, and subattribute elements of the methodology.

2.2.3 LWR Design Decision Methodology

The objective is to develop and test a LWR system design in terms of overall plant safety and life-cycle costs. A design decision methodology will be developed utilizing probabilistic techniques to determine a figure of merit for safety design and allow optimization of cost versus safety on a relative basis. This methodology will be demonstrated on existing safety systems designs.

Energy Incorporated (EI) was selected to develop and test the methodology. Phase I (FY79) of this work was a short-term effort to thoroughly review the current LWR design decision process to assure complete understanding of current practices and to establish a well-defined interface for development of initial quantitative design guidelines. Phase II, which is currently underway, (FY80) involves the development and demonstration of a quantitative-based design decision process, primarily at the system level. A Phase III (FY81) effort is planned to apply the methodology to an overall plant and its associated risk parameters.

2.2.4 Data Base Evaluation/Statistical Methods Development

The analysis of LWR component failure data has been limited by the unavailability of an adequately defined and implemented data base. Such a LWR data base is needed to facilitate safety/risk assessment, reliability analysis, and better maintenance and design. A limited supply of good data has sometimes been used as an excuse for lack of progress in data analysis and further use of existing data.

Results of FY79 work indicate that utilities are ready to support a data base which helps them in their efforts to improve safety analysis, perform maintenance and promote smooth operation. Work in FY79 also showed that a regression model can be used to help alleviate the scarcity of data problems.

Work has begun in FY80 to apply diagnostic techniques to the regression method for reliability data analysis. A small sample investigation has been initiated for the Cox regression method.

2.2.5 Interactive Reliability Analysis

The INEL work incorporates existing reliability and Common Cause Failure (CCF) analysis codes with a Computer Aided Design and Drafting System (CADDs). This provides a safety analysis tool that can be used early in the design phase of nuclear power plants. Efforts in this area are highly motivated since design changes during the late stages of design or during plant construction can be very costly. Further, the integration of three-dimensional graphics with safety and CCF analysis offers a significant improvement in reactor safety assessment technology.

The CADDs software needs and procedures were identified during FY79. Work on CCF analysis software development continued during the initial quarter of FY80.

2.3 Improved Safety Systems

The objective is to identify, design, license, and demonstrate safety systems which are more reliable, simpler, and less costly.

2.3.1 Piping Systems

Problems with pipes and pipe fittings in LWR Plants are responsible for a significant percentage of all safety-related

events and outage times. Resolution of these problems offers potential safety and economic improvements.

The purpose here is to identify piping problems which should be considered for funding by the DOE LWR Safety Technology Program, evaluate design and operational loadings associated with LWR piping systems, and evaluate the benefits to be gained by the use of nonlinear piping analysis.

Contracts were placed with three contractors. Surveys to identify problems in need of further study are nearing completion. Current definitions of piping system loads are under review. Nonlinear analyses of piping systems are being performed.

2.3.2 Seismic Interchange Committee

The purpose of this task is to identify and interchange detailed descriptions of funded programs in the area of seismic analysis of LWRs. Though the emphasis is on activities funded by DOE, NRC, EPRI, and NSF, other programs are followed, such as those conducted by the European and Japanese nuclear power community.

2.3.3 Structural Mechanics Interchange Committee

The objective here is to identify and interchange detailed descriptions of funded programs in the area of structural mechanics. Although activities funded by DOE, NRC, and EPRI are emphasized, other programs, such as those conducted by the European and Japanese nuclear power community, are also followed.

2.3.4 Nonlinear Analysis

The objective is to examine the potential benefits offered by the application of nonlinear analysis methods in the design of LWR systems and components. Three contracts were placed in June and July, 1979. Two of the contracts terminated on September 30, 1979 while the remaining one ended on November 15, 1979. The purpose of these short-term scoping studies was to identify where and how the nonlinear analysis method can be used beneficially in LWR power plant design. The project has essentially been completed. Draft reports are currently under review. Major findings were nonlinear analysis is beneficial to LWR design, and specific areas of research and development work were pointed out. Cost-effectiveness was the major concern for the wide adaption of nonlinear analysis techniques.

2.3.5 Design Constraints Resulting from Considerations of Fire Safety

A methodology is being developed which will identify fire hazards at LWRs and ways to eliminate them. Three reports are in different stages of completion. Plans for barrier exposures to long lasting fires were completed.

2.3.6 NucleDyne Passive Containment System

Gilbert/Commonwealth completed a review of the NucleDyne Passive Containment System and submitted a final report. NucleDyne submitted a preliminary response to that report.

2.3.7 Component Failures (Causes and Solutions)

General Electric submitted a preliminary draft of their final report describing the causes and solutions for recurrent component failures at BWRs. Those programs identified as especially promising will be considered for future funding.

2.4 Man-Machine Interaction

Human errors are estimated to cause a significant part of the risk involved with nuclear power production. The long-range objective of this development area is to develop advanced monitoring and operator assistance techniques that will: (1) provide the operator with timely diagnostic information and recommended corrective procedures during accident sequences; (2) inform the operator of the margins of safety for existing operating conditions throughout the plant and of the prevailing status of the plant safety systems; (3) enable the operator to exercise improved control over randomly occurring plant disturbances that can potentially result in reactor scram; (4) detect and correct errors that occur during maintenance and fuel handling operations. Three programs in this area are being pursued in FY79. These are briefly described below.

2.4.1 Advanced Monitoring and Operator Assistance Program Plan

The purpose here is to formulate a comprehensive and detailed program plan to implement an Advanced Monitoring and Operator Assistance program for reducing human error and improving safe utilization in a commercial LWR. Work on this program is being done by Sandia National Laboratories (SNL) and Oak Ridge National Laboratory (ORNL). Most of the information concerning the tasks of the various subprograms has been assembled. Priorities and funding estimates are being established.

2.4.2 Advanced Digital Control/PPS Development

The objective is to explore the utility of modern optimal estimation, prediction and control theories in the application of assisting the operator to control a commercial LWR. Work on this program is being done by Idaho National Engineering Laboratory (INEL). Comparisons between the performance of the advanced control and protection concepts and the current LOFT system will be made.

2.4.3 Disturbance Analysis and Surveillance System

The purpose of this project is to determine the engineering procedures required to implement a practical, integrated DASS. Development of generic procedures by which a DASS can be designed for a specific plant is of primary importance.

B&W has been selected to be the contractor and will lead a team consisting of Burns and Roe, General Physics Corporation,

and Duke Power Company. It is expected that the contract will be signed by early in January 1980.

2.5 Safety Data

This area is intended to provide information which can be used for more realistic licensing of nuclear plants, to provide the information to support the development of new systems and concepts, and to assist in the resolution of certain NRC Unresolved Safety Issues.

2.5.1 Fuel Damage Limits

The ultimate objective of this effort is to develop a mechanistic fuel damage criterion, i.e., one based on an actual fuel damage mechanism, to replace the DNBR limit currently in use for PWRs. The present study will define a test program for providing the data necessary to establish a mechanistic limit and implement that criterion as a PWR fuel design limit. A draft test proposal, or Experiment Requirement Document (ERD), has been prepared by the contractor and reviewed by a panel of experts. Recommendations were made for improving the proposed test program, and those recommendations are being addressed by the contractor in a revised ERD. Work is also continuing on preconceptual design and cost estimates for the required test facilities.

2.5.2 Containment Sump Reliability (Unresolved Safety Issue)

The test facility at Alden has been completed. System checkout is in progress and is expected to be complete in mid-

February. An initial test plan has been agreed upon and will be begun following system checkout.

2.5.3 Reactor Vessel Material Toughness (Unresolved Safety Issue)

Collation and organization of the NRC Reactor Vessel data base has been completed. All information has been received from the PWR vendors. A draft topical report has been produced and reviewed by a peer review group. Review comments have been incorporated into a final report to be issued about February 1, 1980.

3.0 Progress Reports

This section details the technical progress in each of the program areas.

3.1 Technology Management Center Support Programs

Two programs currently provide the TMC with additional, independent viewpoints on the LWR Safety Improvement Program.

3.1.1 DOE Support

Reports on technical investigations conducted for the President's Commission on the Accident at Three Mile Island were submitted to that Commission during this period. These documents are part of the Commission permanent records that will be available in the National Archives.¹⁻⁴

Brief studies of other LWR safety issues and NRC actions were conducted during this period on the following topics:

- a. Short-term lesson learned (NUREG-0578),
- b. NRC emergency planning basis (NUREG-0396),
- c. Vented containment, and
- d. TMI-2 radioactive releases compared to expectations based on models and data in WASH-1400.

Two additional reports were drafted on the topics of, "A Reliable Plant is a Safe Plant" and "The Licensing Pause and Planned Actions."

3.1.2 Safety Overview

The TMC is maintaining information exchange with informed members of industry, government and the technical community to obtain a broadly-based input in order to ascertain current LWR safety status. As part of this continuing effort, a number of meetings were held to assess the new safety perspective. Reports resulting from the TMI accident were carefully reviewed.

3.2 Risk Methods Utilization

This area concentrates on the development of quantitative risk methods which can be applied to guide safety-related design/operation trade-offs and to focus safety/licensing activity on the more important problems.

3.2.1 Acceptable Risk and Risk Evaluation

The development of probabilistic acceptable risk criteria could assist in the development of a more quantitative and streamlined basis for the licensing of nuclear power plants. This study, conducted at UCLA, undertook tasks which work toward the development of such criteria, employing a data base which includes alternative energy technologies. Specifically, these tasks were defined as follows: (1) Recommend a scope of work and a program plan based on a review of the Inhaber Report for DOE to utilize in conducting a study of U. S. Energy Technologies/Systems; (2) review, develop and evaluate proposals for probabilistic risk criteria; (3) assess the effects of uncertainties in

risk estimates; (4) recommend an action plan for DOE/NRC cooperation on developing acceptable risk criteria; (5) recommend ways in which probabilistic criteria can be used in the licensing process.

Sandia has received completed preliminary drafts of four reports as follows:

1. "Framework for Nuclear Reactor Risk Acceptance Criteria" by J. Griesmeyer and D. Okrent, UCLA-ENG7969.
2. "The Use of Risk Aversion in Risk Acceptance Criteria" by J. M. Griesmeyer, M. Simpson and D. Okrent, UCLA-ENG-7970.
3. "Canvey: The Reaction of a Community Exposed to Substantial Risks" by L. Cave, UCLA-ENG-7967.
4. "Tentative Outline of Work Scope and Organization for a Study of the Risks from Energy Production in the U. S." by L. Cave, D. Okrent and G. Pery, UCLA-ENG-7941.

All four of these papers have been reviewed by Sandia staff and are being revised as is appropriate. Final drafts are expected during the next quarter. After extensive revision, the first report will provide the basis for appropriate follow-on studies.

A request for proposals (RFP) for development of a risk acceptance methodology was sent out during this quarter. The methodology was delineated as follows: (1) Methodology Development for Integrating Risk/Benefit "Hard" Estimates and Measures of

"Soft" (i.e., social and psychological) Variables Characterizing the acceptance of Nuclear Electric Power; (2) Methodology Development for Identification and Quantification of "Hard" and "Soft" Variables that relate to the acceptance of Nuclear Electric Power Plants.

The contract was awarded to the Social Science Research Institute (SSRI) at the University of Southern California. Both parts of the methodology are in development as a unified effort. Decision Research of Eugene, Oregon, is working with SSRI on a subcontract basis.

The methodology development is planned to take place over a two-year period. The following major tasks are now completed: (1) Development of a methodology to identify groups concerned with risks from nuclear and alternative energy sources, and (2) adoption of decision analytic value tree techniques to elicit values and concerns of groups. Further work will adapt multi-attribute measurement methods to quantify and partially aggregate values and concerns to group specific risk/benefit indices, conduct research to determine behavior relationships between "hard" risk/benefit estimates and "soft" values and concerns identified in Task II, and adapt decision analysis for the overall integration of "hard" risk benefit estimates and "soft" perceptions and evaluations. A final report will be issued and a workshop will be conducted to demonstrate the results.

The methodology will be illustrated by applying it to an example decision by a utility on meeting projected increases in the demand for electricity by one of the following: (1) advocating conservation, (2) adding a coal plant, (3) adding a nuclear plant.

3.2.2 R&D Selection Methodology

The aim of this program is the development and demonstration of a R&D selection method suitable for application to the LWR Safety Technology Program. Two different methods are being developed to assess safety improvement, cost and acceptability of LWR R&D and design choices. General Electric is following a probabilistic cost-benefit approach which entails a modification of the Lines of Assurance (LOA) safety analysis used previously for fast reactor safety evaluation. SAI is developing a decision method based on multi-attribute utility (MAU) theory. Both contractors have concentrated on methodology development during FY79 and will perform demonstration of their respective techniques during FY80.

GE will analyze two R&D program areas from the LWR Safety Technology Plan: Improved Safety Systems and Risk Methods Utilization. These areas were selected for the first phase application because of their different modeling and data requirements. Priority ranking will involve two indices: a risk ranking index and a composite ranking index. The risk ranking index will

express the reduction in the expected consequences in man-rem per dollar spent on the R&D effort. A composite ranking index will express the reduction in the expected LWR plant cost burden in dollars per dollar spent on the R&D project. A Linear Constrained Modified Newton Algorithm (LCMNA) has been selected to find the optimum R&D budget allocation. To test the validity of LCMNA allocation, a test problem was analyzed during this quarter using hypothetical R&D projects to minimize the public risk under an R&D budget constraint.

The SAI FY80 work on the project involves introduction of refinements to the R&D Selection Methodology and the associated interactive programming, along with a demonstration of its capabilities by applying the method to a set of R&D program descriptions to obtain a relative prioritization of the individual programs. A group of technical specialists was gathered to produce a set of utility curves and a set of scaling factors for the collection of subattributes, attributes and categories. The intent is to evaluate the sensitivity of R&D program selection results to these different sources during the application. In order to illustrate the advantages of the Multi-Attribute Utility (MAU) technique, seven R&D subprograms have been identified for evaluation and utility ranking. The major effort on development of the computer package has been completed. The remaining work involves completing the rather straightforward programming job on

the unfinished modules and introducing limited features to improve the efficiency and ease of use of the overall package.

3.2.3 LWR Design Decision Methodology

The overall objective of this study is the systematic incorporation of safety considerations into the design decision process in a manner that optimizes the cost versus safety relationship while minimizing the impact on current designs, management and philosophy. Energy Incorporated (EI) was selected to develop and test the methodology and a multi-year schedule is envisioned to accomplish this work.

Phase I (FY79) completed the review of the current design process. Phase II, to be accomplished during FY80, will include the development and demonstration of a quantitative-based design decision process, primarily at the system level. Phase III (the final phase) will be an extension of the methods developed in the previous phase wherein component, subsystem, and system design and operational parameters will be related to an overall plant and its associated risk characteristics. Active industry participation will be sought for this phase of the work.

Activities during the first quarter of FY80 have involved design decision and analysis techniques identification as well as safety criteria definition.

A master list of all nuclear design decisions has been developed. Key decisions were divided into three groups which

were representative of key phases of the design process; the conceptual phase, the initial design development phase, and the main design development phase. This categorization helped in defining the effectiveness of analysis techniques in various phases of design. The list of decisions will be subjected to periodic review as the integrated decision process is developed.

A review of existing codes and regulations is being carried out to identify those criteria that affect the design process in either a qualitative or quantitative manner. The selected criteria will define either specific quantitative goals or express specific qualitative goals for design such as redundancy and separation of subsystems.

Standards and techniques of safety and reliability programs developed and implemented for the military, aircraft, and aerospace design processes have been reviewed and evaluated. The techniques that have produced the most useful results have been identified. Work is continuing to evaluate their effectiveness in the nuclear design flow.

3.2.4 Data Base Evaluation/Statistical Methods Development

A further development of statistical methods is necessary to allow full use of available nuclear plant reliability data. In particular, work in FY79 demonstrated that a regression model can improve use of a limited supply of data.

The Cox regression model permits the estimation of the effect of exogenous variables on the life-time of a component without

assumptions about the exact form of the failure rate function. Although this method has been extensively studied in the last decade, almost no work has been done to investigate its properties when applied to small samples. This is now being done at LASL by computer simulation.

Among the questions being addressed in FY80 are: (1) How well do asymptotic (i.e., large sample) results for the variance of the estimate compare with what is actually observed for small samples? In particular, how reliable are tests for the importance of a given exogenous variable based on such asymptotic results? (2) How is the analysis affected when in reality the failure rate functions are not proportional as assumed by the Cox mode? (3) How does the presence of censoring (termination of observation prior to the failure of all components) affect the analysis?

During the first quarter of FY80, a simulation program was written to explore these questions.

3.2.5 Interactive Reliability Analysis

The use of interactive graphics (i.e., Computer Aided Design and Drafting System [CADD]) capabilities in the design analysis of nuclear systems can benefit both the safety analyst and the system designer. CADD systems are being used by many major corporations throughout the country including Boeing, General Motors, Monsanto, and Sandia National Laboratories. The uses include designing electrical and mechanical circuits and airplane engines.

The coupling of interactive 3D graphics capabilities with safety and common cause failure analyses offers a powerful new approach to reactor safety assessment. Presently, the results of many safety and common cause failure analyses require changes in plant design during construction. Such changes can be costly and impact the schedule. By factoring common cause failure and safety analyses into the design process as early as possible, such changes can be minimized. Through the use of computer-aided design, potential changes to the system design can be easily made and analyzed before construction begins and before the design is finalized. This can result in reduced construction cost.

The objective is to develop and demonstrate a reliability analysis method that can be incorporated early in the design phase of a nuclear reactor power station. The method uses interactive graphics capabilities to examine the proposed plant for potential common mode/common cause failures and to suggest solutions for these. The main advantages to such an approach include minimization of costly changes during the construction phase of the plant, ready accessibility of the latest system design drawings, and rapid evaluation of proposed design changes.

The tasks identified for FY80 are:

- (1) Determine CADDs software needs.
- (2) Modify existing software.

- (3) Develop CADDs software.
- (4) Develop and implement an interface between CADDs and the INEL CYBER computer.
- (5) Develop guidelines for drawing fault trees on CADDs and for collecting qualitative Common Cause Failure (CCF) data.
- (6) Initiate the sample problem and analysis.
- (7) Begin 3D modeling capabilities on the CADDs.
- (8) Issue an interim report on the project.

The majority of Task 1 was performed in FY79. The work left to be completed will be finished after the CADDs arrives and is implemented. Work on Task 2 continues now in FY80. Work on Tasks 3 through 6 will start next quarter. Task 7 will be performed in FY81.

3.3 Improved Safety Systems

The objective of this area is to identify, design, license, and demonstrate safety systems which are more reliable, simpler, and less costly.

3.3.1 Piping Systems

Problems with pipes and pipe fittings in LWR plants are responsible for a significant percentage of all safety-related events and outage times. Resolution of these problems offers potential safety and economic benefits.

The study will identify piping problems which should be considered for funding by the DOE LWR Safety Technology Program,

evaluate design and operational loadings associated with LWR piping systems, and evaluate the benefits to be gained by the use of nonlinear piping analysis.

The work consists of three tasks.

Task 1 will identify the significant LWR piping problems, perform a literature search to determine the recently completed and ongoing studies of these problems, and identify the problems in need of further study. Outlines of a research program for studies of these problems will be developed.

Task 2 will review existing load definitions, data bases, and design procedures used in the design of LWR piping systems.

In Task 3, the benefits that can be derived from using nonlinear analysis methods in the design and licensing of LWR piping systems will be studied. Nonlinearities in typical piping systems will be identified, and nonlinear mathematical models will be constructed and compared to linear-elastic analysis models.

Contracts were placed with URS/J. A. Blume, Engineers and Teledyne Engineering Services in July 1979, and with EDS Nuclear in October 1979 (for Task 2 only).

The following summarizes progress on the three tasks for each of the contractors.

URS/J. A. Blume completed the following:

Task 1: Major part of this task was completed and the first draft of the final task report was written. The

report will be finalized after completion of its in-house review.

Task 2: Work on this task was completed. In-house review of the first draft task report was begun.

Task 3: Some difficulties were encountered in the nonlinear analysis of snubbers. Problems in this area have now been resolved. Linear time history analyses of the example piping system were completed. This task is expected to be completed in January 1980.

Teledyne Engineering accomplished the following:

Task 1: A survey of industry parties was begun to solicit comments and suggestions on potentially useful programs related to LWR piping systems.

Task 2: Additional sources of recorded operating data are being sought.

Task 3: Computer analysis of nonlinear structural effects are continuing.

EDS Nuclear completed the following (Task 2 only):

The identification and categorization of present design loads is nearly complete. Contacts are being made with utilities to obtain operating load data.

3.3.2 Seismic Interchange Committee

The purpose of this task is to identify and interchange detailed descriptions of funded programs in the area of seismic analysis of LWRs. Though the emphasis is on activities funded by DOE, NRC, EPRI, and NSF, other programs are followed, such as those conducted by the European and Japanese nuclear power community.

Continuing activities in this task include: (1) review of reports; (2) attendance, as a member, at meetings of the Seismic Analysis Committee of the ASCE Nuclear Structures and Materials Committee; (3) contact with cognizant NRC, EPRI, and NSF personnel and; (4) visits to Sandia by persons working in the seismic field.

A United States team consisting of 11 members visited Japan on October 17-26, 1979, to discuss with Japanese officials, from government and industry, the various aspects of structural and mechanical design, analysis and testing criteria, especially those aspects related to earthquake loads. Particular attention was given to experience with operating nuclear plants subjected to earthquakes. Another objective of the visit was to explore the possibility of cooperative research projects between the United States and Japan, and the exchange of test data and criteria, and the exchange of operating experience.

The team consisted of two representatives for DOE. In addition, the U. S. Nuclear Regulatory Commission, Lawrence

Livermore Laboratory and the Electric Power Research Institute were represented.

Meetings were held in the Tokyo area with the Ministry of International Trade and Industry, Science and Technology Agency, Power Reactor and Nuclear Fuel Development Corp., Japan Atomic Energy Research Institute, Central Research Institute of the Electric Power Industry, Institute of Industrial Science of the University of Tokyo, Muto Institute, and Kajima Institute of Construction Technology. Visits were made to the Building Research Institute in Tsukuba, the Nuclear Power Engineering Test Center in Isogo and to Fukushima No. 2 Nuclear Power Plant. Full details are given in a published trip report.

3.3.3 Structural Mechanics Interchange Committee

The purpose of this task is to identify and interchange detailed descriptions of funded programs in the area of structural mechanics. Though the emphasis is on activities funded by DOE, NRC, and EPRI, other programs are followed, such as those conducted by the European and Japanese nuclear power community.

Continuing activities in this task include: (1) review reports, (2) attend, as a member, meetings of the ASME Section III Boiler and Pressure Vessel Code committees and the Pressure Vessel Research Committee (PVRC) of the Welding Research Council, (3) attend the EPRI-sponsored Plastic-Fracture Analysis Group (PFAG) meetings, (4) attend American Society of Testing and Materials (ASTM) subcommittee meetings on fracture testing and

allied topics, (5) contact cognizant NRC and EPRI personnel, and (6) confer with persons working in the field.

3.3.4 Nonlinear Analysis

LWR systems and components are currently designed to meet standards set forth in the ASME Boiler and Pressure Vessel Code, which in general specifies the use of linear elastic analysis methods. However, for events which produce severe loadings (Service Level D) such as LOCA, earthquakes, etc., linear elastic response may not be necessary and, in fact, may cause excessive conservative design and increase costs. Therefore, it is possible that the development and application of analysis methods which include nonlinear behavior (both geometric and material) will better quantify existing margins and may lead to designs which are as safe as those currently used but less costly. The present project intends to examine this hypothesis through a qualitative study of the changes in LWR component designs that might result from the use of nonlinear analysis.

LWR systems and components of interest include: the reactor vessel, vessel internals (control rods, core barrel, etc.), nozzles and penetrations, component (e.g., steam generators, reactor coolant pumps) support structures, and containment structures. Piping systems are excluded because they are being addressed by a separate study. Two tasks are defined in the work scope for contractors:

1. Review and summarize the current design procedures and methods of analysis for LWR systems and components (reactor vessel, vessel internals, nozzles and penetrations, component support structures and containment structures). The objective of this task is to identify those systems and components where improvement in design may be made. The work performed to accomplish this task should: (a) identify, review and summarize the current practices used to design LWR systems and components of interest (this literature survey will indicate the time period to be covered and the source of information to be used); (b) review and describe any assumptions underlying current design practices; (c) for events which produce significant loadings (e.g., startup, shutdown, transients, reactor trip, earthquake, LOCA), define and envelope the loadings caused by these events; (d) specify those LWR systems and components for which significant design improvements may be made.

2. Perform a brief study that will qualitatively assess the benefits, if any, to be gained by the use of nonlinear analysis in the design and licensing of each LWR system or component of interest. The work to complete this task should: (a) identify those geometric features and material properties that can only be realistically analyzed using nonlinear methods; (b) select or construct mathematical and/or material models which semi-quantitatively represent or bound the nonlinear behavior of each geometric or material property identified in part (a); (c) use

those models to perform a qualitative analysis (central estimates, bounding calculations) of several representative LWR systems and components which will delineate the benefits (e.g., licensing time, safety margins, costs of construction and maintenance), if any, that would result from the use of nonlinear analysis methods in the design and licensing of LWR power plants.

The accomplishment of the above two tasks will identify the potential benefits from the application of nonlinear analysis methods in LWR systems and components design; further, the areas where nonlinear methods can be used beneficially will also be identified.

A RFQ was issued on May 15, 1979. Contracts were placed to ANATECH International Corporation on June 8, 1979; to O'Donnell Associates, Inc., on June 12, 1979; and to Teledyne Engineering Services on July 18, 1979. The work for ANATECH and O'Donnell was scheduled for completion by September 30, 1979, while November 15, 1979, is the completion date for Teledyne. Visitations by Sandia personnel to ANATECH and O'Donnell were made in July 1979, and to Teledyne in November. The work has essentially been completed. Preliminary results indicate that nonlinear analysis methods are beneficial to LWR design. The major concern for the wide application of such techniques is the cost-effectiveness question. Areas where nonlinear analysis are needed were pointed out. The draft reports are currently under review. Final reports will be issued when this is completed.

3.3.5 Design Constraints Resulting from the Consideration of Fire Safety

A report titled "Nuclear Power Plant Fire Protection-Philosophy and Analysis" has been written. The report discusses an integrated methodology for assessing nuclear power plant fire safety and presents procedures for identifying plant vulnerabilities to fires. Treatment of fire-spread paths as a flow network provides a promising approach to identifying a minimum set of barriers and rooms which need to be protected against fire in order to assure plant safety.

The vulnerability of barriers to long lasting (~3 hour) cool fires (peak temperatures less than those for which standard 3 hour barriers are rated) will be examined by two small-scale burn tests. Concrete pours for these two tests have been completed. A critical review of fire barrier data has been completed and will be issued as a report during the next quarter.

3.3.6 NucleDyne Passive Containment System

A qualitative evaluation of the NucleDyne Passive Containment System (PCS) as applied to a two loop Pressurized Water Reactor (PWR) has been completed by Gilbert/Commonwealth (G/C). G/C concludes (1) that the pressure suppression behavior of PCS appears likely to function as claimed; (2) that, if shock jet injectors can be shown to work as claimed, then the unique passive safety systems of PCS will provide enhanced safety performance, at least for large LOCAs, due to increased system redundancy and

diversity; (3) that the use of free-standing cells to enclose the primary loop components necessitates handling all loads by compression, which may be unacceptable and will significantly complicate the construction of PCS; (4) that the cost of a reactor with PCS will be comparable to that of a conventional reactor (decreased size and the elimination of some conventional safety systems is offset by a more complicated construction sequence, increased design pressures, and the addition of unique passive safety systems); and (5) that an NSSS vendor should be consulted in order to determine whether the elimination of pipe whip restraints requires significant design modifications for the primary loop components.

3.3.7 Component Failures (Causes and Solutions)

During FY79 three studies of the cause of and solutions for recurrent component failures at nuclear power plants were initiated. Final reports for the SAI study of scrams and manual shutdowns at LWRs and for the CE study of component failures at PWRs have been received. The major conclusions of these two studies have been summarized in previous reports. A draft final report for the GE study of component failures at BWRs was received in November. This report identifies 14 programs which would lead to significant reductions in scram frequency and outage time at BWRs. Seven of these programs have the potential to save at least one-half day of downtime per year at most operating BWRs. These seven programs would develop (1) an analog transmitter/trip unit system, (2) an

improved moisture separator level control and drain system, (3) a redundant feedwater control system, (4) an improved recirculation control system, (5) a thermal power simulator, (6) a safety relief valve leakage monitor, and (7) an improved feedwater regulation valve.

3.4 Man-Machine Interaction (MMI)

Human errors are estimated to cause a significant part of the risk involved with nuclear power production. Efforts to improve the safety of nuclear power production through improved man-machine interaction design is justified on the basis of the following list of findings: (1) Operator error is involved with 50 percent of possible sequences that lead to release of radioactive material. (2) Human error accounts for 27 percent of abnormal occurrences at U. S. LWR plants. Fifty percent of the errors are due to carelessness of operators during maintenance and operation. (3) Of all recorded U. S. PWR outages, about one-fourth could have been averted by correct and timely operator action. (4) Control room designs are not optimized to promote reliable, effective and safe operator performance. (5) An adequate data base for estimation of human error rates does not exist. (6) Reduction in the number of plant shutdowns provides a significant safety benefit. Furthermore, the operator has inherent limitations in his ability, during emergencies when time is a critical factor, to analyze large amounts of information, distinguish important information from the irrelevant, and devise and

take appropriate correct actions. This is a significant limitation because the potential adverse consequences of nuclear accidents on public safety are very major.

The long range objective of the man-machine interaction program is to develop advanced monitoring and operator assistance techniques that will: (1) Provide the operator with timely diagnostic information and recommended corrective procedures during accident sequences. (2) Inform the operator of the margins of safety of existing operating conditions throughout the plant and of the prevailing status of the plant safety systems. (3) Enable the operator to exercise improved control over randomly occurring plant disturbances that can potentially result in reactor scram. (4) Detect and correct errors that occur during maintenance and fuel handling operations.

This program began in FY79 with a scoping study to identify the required methodologies and technologies, an exploratory control techniques program to investigate applications from the fields of optimal control and optimal estimation and prediction of process variables, and a scoping and feasibility study of a disturbance analysis and surveillance system (DASS) for LWRs. The objectives and progress to date for the three FY79 projects are summarized below.

3.4.1 Advanced Monitoring and Operator Assistance Program Plan

The purpose here is to prepare a comprehensive and coordinated program plan to implement Advanced Monitoring and Operational Assistance Techniques in a commercial power LWR with the

objective of reducing the potential for human error and improving safe operation of the reactor. Work on this program is being done by Sandia National Laboratories (SNL), Oak Ridge National Laboratory (ORNL), and nuclear steam system vendors.

The program plan, based on the principle of defense-in-depth, will be completed in the first half of the fiscal year. The program plan will describe tasks included in the following subprograms: Disturbance Analysis and Surveillance System (DASS), Post Accident Monitoring and Procedures, Readiness Monitoring, Advanced Automatic Process Control, Plant Protection System, Human Factors and Human Engineering, Reliability, Computer Design, Control Room Design, Maintenance Errors, and Fuel Handling Errors. An Advisory Task Force will be organized to provide input from the nuclear industry. The plan will include information on priorities of subprograms, funding requirements and schedules. Also included will be plans to demonstrate and implement beneficial development in a nuclear power plant. Methods of technology transfer to the industry will be taken into account.

ORNL staff members met with development personnel at B&W's Research Center in Lynchburg, Virginia, early in December. The purpose of this meeting was to increase ORNL's awareness of the status of advanced control and modeling concepts at B&W and to review portions of the MMI program for technical accuracy.

Most of the information concerning the tasks of the various subprograms has been assembled. Priorities and funding estimates are partially established. Integration of the data into report

form is presently underway. A review of a preliminary version of the program has been completed by Westinghouse, Babcock and Wilcox, Combustion Engineering, NRC, and EG&G.

3.4.2 Advanced Digital Control/PPS Development

The utility of modern optimal estimation and control theories in nuclear systems applications is currently an active area of research both in this country and abroad. Protection and control systems using these methods have been implemented on a number of foreign plants. The research and implementations to date indicate that the on-line digital optimal estimation and prediction of plant variables and unmeasurable parameters, coupled with optimal control algorithms, does offer increased safety and plant efficiency. This project is important because it will produce the most comprehensive investigation of nuclear systems applications of these techniques in this country to date. The approach is to simulate an LWR on the analog side of the INEL hybrid computer. The digital side will contain the estimation, prediction, and control algorithms. Protection and control action effectiveness, under conditions of transient inputs and process and measurement noise, will be determined. Operator interface requirements will be studied. System reliability studies will be performed.

During FY80, the current INEL hybrid computer laboratory will be augmented with an additional PDP 11/55 minicomputer. The estimator and control algorithms incorporating a low-order LOFT

plant model will be implemented on this additional computer. Evaluations of the algorithm performance using the hybrid LOFT model will be used as a basis for algorithm refinement. The hybrid LOFT model will be run at various operating points simulating operational transients for the advanced LOFT control system to act upon. Sensitivity, stability, and speed of operation of the new control system will be examined. Different control laws will be studied. Comparisons between the performance of the advanced control and protection concepts and the current LOFT system will be made. Advantages and disadvantages of each system will be noted. Specific attention will be paid to the control of the steam generator water level.

The PDP-11 control computer is currently scheduled to arrive at the end of January 1980. Studies are being done to identify reduced-order models of the LOFT plant. This work involves driving the previously developed low-order LOFT model with random disturbances and identifying correlations between various plant variables.

Documentation of the LOFT steam generator modeling effort is continuing. The high and low order steam generator programs are being incorporated in a single program to permit simultaneous execution for simulation of on-line estimation and control. A macro program has been written for on-line system linearization and has been added to the Advanced Continuous System Language macro library.

3.4.3 Disturbance Analysis and Surveillance System (DASS)

The purpose of this project is to determine the engineering resources, design guidelines, and procedures required to implement a practical, integrated DASS. Development of generic procedures by which a DASS can be designed for a specific plant is of primary importance.

A useful set of DASS goals will be established in the following four categories: safe shutdown with no radiation release, monitor plant for violations of assumptions made in deriving trip surfaces, keep plant on-line, and shutdown with no damage to plant component.

The scope and depth of DASS functions will be selected where scope of the DASS refers to both the number of plant components or subsystems and the number of functions that are included in one system and where depth refers to the level of detail and analysis required by the system.

The modes of operation will be defined relative to DASS functions. Modes of operation are major classifications of plant loading such as start up, maneuvering, base load, and includes the operators intervention with respect to each major piece of equipment in the plant.

Given the list of DASS goals, prioritized for operating modes and scoped in terms of implementation complexity, general engineering procedures and estimates to guide DASS design and development will be produced.

A specific DASS function will be selected and the design and engineering analysis will be performed according to the procedures formulated in the above task.

B&W has been selected to be the contractor and will lead a team consisting of Burns and Roe, General Physics Corporation and Duke Power Company. The contract has not yet been placed. Meetings were held at B&W and the work scope was more carefully defined and agreed upon between SLA and B&W. It is expected that the contract will be signed early in January 1980.

3.5 Safety Data

This area is intended to provide information which can be used for more realistic licensing of nuclear plants, to provide information to support the development of new systems and concepts, to assist in the resolution of certain NRC Unresolved Safety Issues.

3.5.1 Fuel Damage Limits

One of the fuel design limits currently used for PWRs is the departure from nucleate boiling ratio (DNBR) criterion. The DNBR limit, however, is not based on an actual fuel damage mechanism and, thus, does not provide an accurate indication of safety margin. The objective of the present program is to determine the data requirements for developing a mechanistic fuel damage limit which can replace the DNBR limit. The contractor for this study is EG&G-Idaho at the Idaho National Engineering Laboratory (INEL).

In meeting the objective of this program, the contractor, EG&G, will propose a test series, test train, and test facility

for generating the necessary data, and will provide cost estimates for developing the test capability and performing of the tests. A parallel program is being conducted by the Electric Power Research Institute (EPRI) to estimate the benefits (e.g., increased power, improved thermal efficiency, reduced power cycling due to unnecessary reactor trips, etc.) which could result through replacement of the DNBR limit.

The contractor proposes to perform fuel bundle tests in the Engineering Test Reactor (ETR) at INEL. A new in-pile tube and water loop, capable of simulating typical PWR thermal hydraulic conditions, is being proposed for these tests. The objective of the proposed bundle tests will be to establish a fuel damage threshold (a threshold cladding temperature, as currently conceived) for PWRs under normal and anticipated transient conditions, and to provide the data base needed to implement this threshold as a limit on the design and operation of commercial PWRs.

During the last quarter, the draft Experiments Requirements Document (ERD) prepared by the contractor was presented for comment to a panel of industry, university and national laboratory experts. The major recommendations made by the panel were: (1) that further work (e.g., out-of-pile tests or subchannel analyses) needs to be done to confirm that the three-foot (axial length) ETR core can produce thermal-hydraulic conditions in the proposed test train which are typical of conditions in corresponding

sections of a standard PWR 12 foot core; (2) that subchannel analyses be initiated to determine the range of inlet conditions (flow, temperature, quality) which will be needed for this program; and (3) that further attention be given to the proposed instrumentation of the test train to insure that all the necessary data will be obtained from the tests. The contractor is presently revising the ERD to address the issues raised by the review panel.

Work on the preconceptual design of the ETR water loop which would be needed for the proposed fuel bundle tests continued under a subcontract to Kaiser Engineers. Also, physics and engineering scoping calculations were continued through this quarter in support of the ETR irradiations and in-pile tube design effort. These preliminary design studies are directed toward providing realistic cost estimates for the major construction items anticipated in the program: namely, construction of a new in-pile tube and water loop system, and modification of the ETR to accommodate that test hardware.

3.5.2 Containment Sump Reliability (Unresolved Safety Issue)

Were a pipe break or its equivalent to occur in the primary system of a pressurized water reactor and were the discharge rate sufficiently high, an emergency core cooling system (ECCS) would be activated that would inject water into the system to maintain core cooling. The water would initially be supplied from a large emergency supply tank. Water that would be discharged from the break (and from containment spray) collects in the containment

building sump. Upon the supply tank being emptied to a specific level, the ECCS system is reconfigured from the injection mode to the recirculation mode. In the recirculation mode, water is drawn from the containment building sump, cooled and then circulated through the core to provide core cooling. Under this mode, the containment spray system is also reconfigured to use the containment building sump as a source of coolant.

In the recirculation mode, cooling systems must have the capability to reliably operate for extended periods of time. The reliable operation of pumps in this mode must, therefore, be assured. Conditions to be avoided when the containment building sump is a coolant source include: (1) blockage by debris, (2) air entrainment, (3) vortex formation, and (4) excessive pressure drops.

The above conditions were identified by the NRC and grouped as a single Unresolved Safety Issue, "Containment Emergency Sump Reliability," (Generic Task A-43).

Sandia contacted various hydraulic laboratories during January and February 1979, to ascertain interest in this project. Sandia also worked closely with NRC to determine the extent of the role the Laboratory would play in the resolution of this issue.

In early March, NRC included Sandia in their overall Task Action Plan to resolve Task A-43. Sandia has concentrated on the understanding of vortex formation and the development of vortex suppression techniques which are crucial to the successful resolution of the Task. Although a number of model tests have been

performed by independent hydraulic laboratories for specific reactors, a generic systematic solution for vortex suppression is still lacking. The development of vortex suppression technology is being conducted in two phases. The first phase involves new facility construction for an experimental parametric study of the vortex formation phenomenon. This was initiated during the last quarter of FY79. The second phase includes actual experimentation with vortices and development of vortex suppression technology and will be conducted during FY80 and FY81. A RFP was prepared by Sandia in close cooperation with the NRC and issued for bid in late March 1979. A contract was awarded to Alden Research Laboratories in June following competitive bidding. Procurement of long-lead items (such as recirculation pumps) and preliminary site excavation were initiated immediately following the award of the contract. Design of the facility and submission of a draft experimental program report were completed at the end of FY79.

A modified test plan was proposed by Sandia in collaboration with Alden Research Laboratory and in consultation with NRC Licensing and Research staff. A firm first phase test plan was established in meetings held in Boston and Holden, Massachusetts in January 1980, participated in by Sandia, NRC, DOE, and Alden and agreed to by all parties. Experiments will begin in July 1980, and further program guidance will be provided in meetings to be held in June 1980.

3.5.3 Reactor Vessel Materials Toughness (Unresolved Safety Issue)

Because of the possibility of failure of nuclear reactor pressure vessels designed to the ASME Boiler and Pressure Vessel Code is remote, the design of nuclear facilities does not provide protection in the event of reactor vessel failure. Prevention of reactor vessel failure depends on maintaining the reactor vessel material fracture toughness at levels that will resist brittle fracture during plant operation. For the service times and operating conditions typical of current operating plants, reactor vessel fracture toughness properties provide adequate margins of safety against vessel failure. However, as plants accumulate more and more service time, neutron irradiation may reduce the material fracture toughness and the initial safety margins. Results from reactor vessel surveillance programs indicate that up to 20 operating PWRs may be beltline materials with marginal toughness after comparatively short periods of operation.

The NRC included the above problem in their list of Unresolved Safety Issues as Generic Task A-11, "Reactor Vessel Material Toughness."

During January and February 1979, Sandia contacted a number of leading experts in the reactor materials field including those from national laboratories, consulting firms and the NSSS vendors to obtain views and ascertain interest in the project. Cognizant NRC personnel were also contacted to determine the role the Laboratory would play in the resolution of A-11.

In early March 1979, NRC included Sandia in their overall Task Action Plan. It was decided that Sandia's role would include a three part program:

The first part was the collation and organization of the existing NRC data base into a computerized data bank which would lend itself more easily to further analysis.

The RFP for this work was released for bid after consultation with NRC in March 1979. Award was made to International Energy Associates in early May. The data base was computerized and a user's manual prepared in September 1979.

The second part of the program involved the PWR vendors and with a request they provide any additional data (other than NRC information) that would aid in the characterization of toughness of each reactor vessel.

The RFP for this part was prepared and released for bid in March 1979, following coordination with NRC. Awards to each PWR vendor resulted during May and June. As of September 30, some, but not all, of the information had been received from the vendors. This was due to increased workloads resulting from the TMI-2 incident. It is anticipated the remainder of the information will be available by February 1980.

The third and final part required the use of an independent review organization. This organization would be requested to evaluate and assess all data, methods, and information provided and to issue a draft topical report summarizing their conclusions

and findings. The draft report was then to be circulated for review to the NRC, reactor vendors and other recognized national authorities on the subject.

The draft report was expected to support the hypothesis that all operating PWR vessels have sufficient toughness margins for the next five year period. Were this hypothesis supported, then a crucial portion of the NRC Task Action Plan would be successfully completed. Following a peer review process, a final topical report would be released.

The RFP for this final portion was prepared in cooperation with the NRC and released for bid in late March. The award was made to Washington University Technical Associates in May 1979. The draft technical report was released September 30, 1979, and peer review was completed on October 15. Comments from the review are being incorporated into the final report which is anticipated to be released by February 1, 1980.

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