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Structural Aging Program Technical Progress for Period January – December 1992

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**STRUCTURAL AGING PROGRAM TECHNICAL PROGRESS REPORT
FOR PERIOD JANUARY 1, 1992, TO DECEMBER 31, 1992**

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ABSTRACT

The Structural Aging Program is conducted for the Nuclear Regulatory Commission (NRC). The program has the overall objective of preparing an expandable handbook or report which will provide NRC with potential structural safety issues and acceptance criteria for use in nuclear power plant evaluations for continued service. Initial focus of the program is on concrete and concrete-related materials which comprise safety-related (Category I) structures in light-water reactor facilities. The program is organized into four tasks: Task S.1 – Program Management, Task S.2 – Materials Property Data Base, Task S.3 – Structural Component Assessment/Repair Technology, and Task S.4 – Quantitative Methodology for Continued Service Determinations. During this reporting period, an annual technical progress report was prepared; program personnel participated in the NRC Water Reactor Safety Information, NRC Aging Research Information, and Fifth Workshop on Containment Integrity meetings; 16 technical papers/reports were published, five technical papers/reports have been completed and accepted for publication, six draft technical papers/reports have been completed, six journal articles have been prepared for publication, eight presentations were made at technical meetings, and six program briefings were made. In addition, lectures on concrete materials and containment structures in nuclear plants were presented at the International Atomic Energy Agency training course on safety aspects of aging and maintenance in nuclear power plant operation; *Structural Materials Handbook* pages and *Structural Materials Electronic Data Base* files were completed for 55 additional materials bringing the total materials in the Structural Materials Information Center to 116; long-term (42 year) properties were developed from aged concrete materials; testing of concrete materials cast in conjunction with several of the United Kingdom nuclear power stations was completed; a survey was conducted to identify in-service inspection procedures used by utilities for concrete structures in U.S. nuclear power plants, problem areas that have been identified for these structures, repair procedures that have been utilized, and the effectiveness of remedial programs; overviews of North American and European procedures for repair of reinforced concrete structures were completed; assessments of corrosion of metals in concrete, potential for occurrence of stray electrical current-induced corrosion in nuclear power plant reinforced concrete structures, and feasibility of using cathodic protection systems to mitigate corrosion of reinforced concrete structures were completed; an approach was developed for formulation of a damage assessment and repair prioritization methodology; and the methodology developed for condition assessment and reliability-based life prediction of concrete structures was refined and used to investigate optimization of inspection and maintenance strategies to maintain failure probability below a specified target value. Appendix A presents a summary of accomplishments under each of the major task areas of the Structural Aging Program.

1. INTRODUCTION

The Structural Aging (SAG) Program has the overall objective of preparing a handbook or report which will provide the Nuclear Regulatory Commission (NRC) license reviewers with the following: (1) identification and evaluation of the structural degradation processes; (2) issues to be addressed under nuclear power plant (NPP) continued service reviews, as well as criteria, and their bases, for resolution of these issues; (3) identification and evaluation of relevant in-service inspection or structural assessment programs in use, or needed; and (4) quantitative methodologies for assessing current, or predicting future, structural safety margins. The results of this study will provide an improved basis for the NRC staff to evaluate NPPs for continued service. More specifically, potential regulatory applications of this research include (1) improved predictions of long-term material and structural performance and available safety margins at future times, (2) establishment of limits on exposure to environmental stressors, (3) reduction in total reliance by licensing on inspection and surveillance through development of a methodology which will enable the integrity of structures to be assessed (either pre- or post-accident), and (4) improvements in damage inspection methodology through potential incorporation of program results into national standards which could be referenced by Standard Review Plans.

The SAG Program consists of three technical tasks and a management task. The organization of these tasks is illustrated by the Level 2 work breakdown structure for the program shown in Fig. 1.1. An overall representation of selected key activities and major milestones for each of the task areas, as well as their interrelationship, is presented in Fig. 1.2. The tasks interface with each other and are augmented by documentation and technology transfer activities. The remainder of this document is arranged according to the work breakdown structure shown in Fig. 1.1 with a chapter in this report addressing each of the four tasks. A work breakdown structure has also been developed for each task and is displayed in the corresponding chapter of this document. Each chapter also contains a brief objective for that task, a background discussion, a plan-of-action, summary of accomplishments for the reporting period, planned FY 1993 activities, and milestone statements and schedule.

Fig. 1.1. Major work breakdown structure for the Structural Aging Program.

Fig. 1.2 Selected key activities and major milestones for each task of the SAG Program.

2. TASK S.1 – PROGRAM MANAGEMENT

2.1 OBJECTIVE

The overall objective of the program management task is to effectively manage the technical tasks undertaken to address priority structural safety issues related to continuing the service of nuclear power plants. Management duties include planning, integrating, monitoring, reporting, and technology transfer. A key part of the management function is the integration of the technical objectives and the efforts of various program participants. Figure 2.1 presents a work breakdown structure for the management task.

2.2 BACKGROUND

The SAG Program is administratively carried out through the Engineering Technology Division of ORNL. A key part of the management function is the integration of the technical objectives and the efforts of various program participants. The participants will include other divisions of ORNL, other national and international laboratories, and universities and industrial organizations. The aim is to utilize capabilities and facilities in a complementing way to meet program objectives. Figure 2.2 illustrates the placement of the program within the ORNL organization and lists key staff members (as of December 1992).

Concerning technical integration, a strong relationship continues to be made with peer groups and committees. These include: the American Society of Mechanical Engineer's (ASME) Boiler and Pressure Vessel Code Subcommittee Section XI Working Group on Concrete Pressure Components, Subgroup on Containment, and Working Group on Plant Life Extension; American Concrete Institute (ACI) committees on Materials Property Data Base, Service Life Prediction, Strength of Structures, Structural Safety, Repair of Concrete, Nondestructive Testing, Concrete Nuclear Structures, and Radioactive and Hazardous Waste Management; and American Society for Testing and Materials (ASTM) Standard groups. In addition, efforts have continued to establish exchange of information with other North American, European, and Asian researchers and pertinent research groups such as Fédération Internationale de la Précontrainte (Prestressed Concrete Pressure Vessels and Containments), International Union of Testing and Research Laboratories for Materials and Structures (RILEM committees on Damage Classification of Concrete Structures and Service Life Prediction of Building Materials and Components), European Committee on Concrete (CEB), International Council for Building Research Studies and Documentation (CIB), etc.

Program developments are transferred to the technical community through progress and topical reports, program review meetings, information meetings, open-literature papers, and committee participation. Informal technical interactions with several foreign organizations have been established and every effort will be expended to enhance and expand these relationships.

2.3 PLAN OF ACTION

The plan for Task S.1 is to effectively manage the program. The Level 3 work breakdown structure for this task is shown in Fig. 2.3. The three subtasks present a more detailed breakdown of the management functions performed. The specific activities and schedules within each subtask are presented later in this chapter (Sect. 2.7) in the form of milestone charts. In addition to planning, monitoring, and reporting on the program performance in a timely manner, efforts are continuously made to maintain effective technology transfer and to maintain liaison with peer groups, committees, and programs in foreign countries.

This task administers the research and development subcontracts and technical consulting agreements that are required to supplement the ORNL work. The subcontractor progress reports are integrated into the overall program technical progress reports. Figure 2.4 presents a listing of subcontracted activities under each of the program's primary task areas as of December 1992.

The monthly progress report is part of a monthly highlights report for the NRC Office of Nuclear Regulatory Research issued by the ORNL NRC Programs Office. An associated part of this reporting procedure is the preparation of an annual technical progress report.

2.4 SUMMARY OF ACCOMPLISHMENTS

During the reporting period covered by this document, activities have been conducted under each of the subtask areas listed in Fig. 2.1 for Task S.1 (Program Management).

2.4.1 Program Planning and Resource Allocation (Subtask S.1.1)

Under this subtask, the program's FY 1993 Project and Budget Proposal was prepared. Two subcontracts were completed [Construction Technology Laboratories (long-term properties of concrete) and National Institute of Standards and Technology (evaluation of in-place methods for concrete strength determination)], and six existing subcontracts were administered (Sargent & Lundy Engineers; Johns Hopkins University; Taywood Engineering Limited; Wiss, Janney, Elstner, Associates; Corpro Companies; and Howard University). Each of these subcontract activities is discussed later in this document under the appropriate program task (see Fig. 2.4). In addition, program personnel participated in NRC short courses on General Electric Boiling Water (Ref. 2.1) and Westinghouse Pressurized Water Reactor Technology (Ref. 2.2).

2.4.2 Program Monitoring and Control (Subtask S.1.2)

Program monitoring and control activities have included the preparation of monthly management reports for inclusion in the highlights report prepared for the NRC Office of Nuclear Regulatory Research and issued by the ORNL NRC Programs Office. An annual technical progress report was prepared (Ref. 2.3) and Program personnel presented papers at the NRC Aging Research Information Conference (Ref. 2.4-2.6), Fifth Workshop on Containment Integrity (Ref. 2.7), and Twentieth Water Reactor Safety Information Meeting (Ref. 2.8).

2.4.3 Documentation and Technology Transfer (Subtask S.1.3)

During this reporting period, 16 technical papers/reports were published (Appendix B), 5 technical papers have been prepared for presentation at international meetings, 6 draft technical papers/reports have been completed, 6 journal articles have been prepared for publication, and 8 presentations were made at technical meetings (Appendix C). Lectures on concrete materials and containment structures in nuclear plants were presented at the International Atomic Energy Agency training course *Safety Aspects of Aging and Maintenance in Nuclear Power Plant Operation* (Refs. 2.9, 2.10). In addition, program personnel participated in technical committees of the American Concrete Institute (Service Life Prediction, Radioactive and Hazardous Waste Management, and Concrete Materials Property Data Base), ASME (Section XI Working Group on Plant Life Extension, Working Group on Concrete Pressure Components, and Subgroup on Containment), International Union of Testing and Research Laboratories for Materials and Structures (Damage Classification of Concrete Structures), and Fédération Internationale de la Précontrainte (Prestressed Concrete Pressure Vessels and Containment Structures).

The program was coordinated with other government agencies-related activities, for example, Low-Level Radioactive Waste and Nuclear Plant Aging Research programs sponsored by the NRC and Department of Energy-sponsored Modular High-Temperature Gas-Cooled Reactor Program and New Production Reactors Working Group on Standards. Discussions continued with the utility sector of the nuclear power industry (for example, Nuclear Management and Resources Council and Electric Power Research Institute), as well as with several utilities (for example, Consumers Power and Commonwealth Edison), national organizations, universities, and consulting/engineering organizations. Table 2.1 presents a listing of domestic organizations which have been contacted. Overviews of the program were presented to Boston Edison (Ref. 2.11-2.12), Southern California Edison (Ref. 2.13), U. S. Army Corps of Engineers Waterways Experiment Station (Ref. 2.14) and at the ASME Pressure Vessels and Piping Conference (Ref. 2.15).

Technology exchange has continued at the international level. One foreign trip was taken to visit Japanese research organizations, construction companies, and technical institutions where materials research and concrete testing are performed (Ref. 2.16). These establishments were visited to obtain information on time variation of material properties, performance of concrete material systems in nuclear power plant structures, techniques for repair of degraded concrete structures, methods for evaluation of concrete structures, and modeling to predict the effects of degradation factors on concrete materials. An overview of the SAG Program was prepared for inclusion in the minutes of the RILEM committee *Damage Classification of Concrete Structures* (Ref. 2.17). Information exchange continued with several European organizations: Fédération Internationale de la Précontrainte Commission on Prestressed Concrete Pressure Vessels and Containments, International Council for Building Research Studies and Documentation (CIB), International Atomic Energy Agency, and Commission of European Communities. Contacts have been made with 100 foreign organizations in 25 countries in pursuit of technology exchange. Table 2.2 presents a summary of foreign organizations with whom SAG Program personnel have been in communication.

2.5 PLANNED FY 1993 ACTIVITIES

Activities planned in FY 1993 will in large part be directed at bringing program technical activities to a timely conclusion. The planning, organization, performance, and reporting of the technical tasks will be managed in keeping with approved program plans. The monthly reporting system will be continued, an annual technical progress report prepared, and the Project and Budget Proposal (189) will be developed and issued. Selection and administration of subcontracts will continue. Participation in technical committees (ACI, ASME, etc.), and industry and NRC nuclear plant aging management programs will continue. Efforts will continue to establish technology transfer activities both domestically and to foreign countries in keeping with NRC guidelines.

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- 2.7 D. J. Naus, "Aging of Concrete Containment Structures in Nuclear Power Plants," *Fifth Workshop on Containment Integrity*, Washington Marriott, Washington, D.C., May 14, 1992.
- 2.8 D. J. Naus, "An Update on the Structural Aging Program," *Twentieth Water Reactor Safety Information Meeting*, Bethesda, Md., October 23, 1992.

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- 2.17 D. J. Naus, C. B. Oland, B. R. Ellingwood, Y. Mori, and E. G. Arndt, "An Update on the Structural Aging Program," Proceedings of the 6th Meeting RILEM Technical Committee 104-DDC *Damage Classification of Concrete Structures*, held 18-20 May 1992 Haifa, Israel, Published by the National Building Research Institute, Technion-Israel Institute of Technology, Technion City Haifa, Israel, I. Soroka editor, 1992.

2.7 MILESTONE STATEMENT AND SCHEDULE

The statement and schedule for the milestones in Task S.1 are presented in the following charts.

[illegible][illegible]

Fig. 2.1. Level 2 work breakdown structure for SAG Task S.1: Program Management.

Fig. 2.2. Organization of the Structural Aging Program.

Fig. 2.3. Level 3 work breakdown structure for SAG Task S.1: Program Management.

Fig. 2.4. Listing of subcontracted activities for each primary task area.

Table 2.1. SAG program contacts with domestic organizations

Type	Organization	Location
Governmental	Argonne National Laboratory	Argonne, Ill.
	Argonne National Laboratory-West	Idaho Falls, Idaho
	Battelle Pacific Northwest Laboratories	Richland, Wash.
	Bureau of Reclamation	Denver, Colo.
	Corps of Engineers - Headquarters	Washington, D.C.
	Corps of Engineers - Waterways Experiment Station	Vicksburg, Miss.
	Defense Nuclear Facilities Safety Board	Washington, D.C.
	Federal Highway Administration	Washington, D.C.
	Hanford Engineering Development Laboratory	Richland, Wash.
	Idaho National Engineering Laboratory	Idaho Falls, Idaho
	National Institute of Standards and Technology	Gaithersburg, Md.
	Navfac	Alexandria, Va.
	Sandia National Laboratory	Albuquerque, NM.
	Savannah River Laboratories	Aiken, S.C.
	Strategic Highway Research Program	Washington, D.C.
	US DOE Reactor & Nuclear Fuels Corporation	Toyko, Japan
	US DOE Headquarters	Germantown, Md.
National	American Concrete Institute	Detroit, Mich.
	American Institute of Mining, Metallurgical and Petroleum Engineers	Warrendale, Penn.
	American Institute of Steel Construction	Chicago, Ill.
	American Iron and Steel Institute	Washington, D.C.
	American Nuclear Society	LaGrange Park, Ill.
	American Society of Civil Engineers	New York, N.Y.
	American Soc. of Mechanical Engineers	New York, N.Y.
	American Society for Testing and Mat'ls	Philadelphia, Penn.
	ASM International	Materials Park, Ohio
	Concrete Reinforcing Steel Institute	Schaumburg, Ill.
	CORRPRO Companies, Inc.	Medina, Ohio
	DYWIDAG Systems Int'l	Lincoln Park, N.J.
	ERICO Products, Inc.	Cleveland, Ohio
	Martin Marietta	Baltimore, Md.
	Master Builders	Cleveland, Ohio
	NUS Corporation	Gaithersburg, Md.
	National Assoc. of Corrosion Engineers	Houston, Tex.
	Prestressed Concrete Institute	Chicago, Ill.
	Post-Tensioning Institute	Phoenix, Ariz.
	U.S. Steel	Pittsburgh, Pa.
	Wire Reinforcing Institute	McLean, Va.

Table 2.1. SAG program contacts with domestic organizations (cont.)

Type	Organization	Location
Consulting/Engineering	BCP Technical Services	Novato, Ca.
	Bechtel Power Corporation	Gaithersburg, Md.
	Bechtel Western	Norwalk, Calif.
	Jack R. Benjamin and Assoc.	Mountain View, Calif.
	T. M. Brown	Bartlett, Ill.
	Burns & Roe	Oradell, N.J.
	Construction Technology Laboratories	Skokie, Ill.
	Consulting Engineers Group	Mt. Prospect, Ill.
	Di Benedetto Associates	North Andover, Mass.
	Dixon & Associates	Stone Mountain, Ga.
	EQE Engineering	San Francisco, Calif.
	N. FitzSimons	Rockville, Md.
	Fluor Daniels, Inc.	Irvine, Calif.
	General Dynamics Services	Groton, Conn.
	General Electric (Vallecitos)	Pleasanton, Calif.
	Innovative Technologies	Monroeville, Pa.
	Metropolitan Water District	Los Angeles, Calif.
	Multiple Dynamics Corporation	Southfield, Mich.
	Newtech Resources	Costa Mesa, Calif.
	NSF Center for Science and Technology of Advanced Cement-Based Materials	Evanston, Ill.
	Olson Engineering Inc.	Lakewood, Colo.
	PDI Technology	Brea, Calif.
	PSC	East Chicago, Ind.
	Raths, Raths & Johnson	Willowbrook, Ill.
	Sargent & Lundy Engineers	Chicago, Ill.
	Singleton Laboratories	Louisville, Tenn.
	Stone & Webster Engineering Corp.	Boston, Mass.
	Teledyne Engineering Services	Jupiter, Fla.
	United Engineers and Constructors	Philadelphia, Pa.
	Westinghouse Scientific & Technology Center	Pittsburgh, Pa.
	Wiss, Janney, Elstner and Assoc.	Northbrook, Ill.

Table 2.1. SAG program contacts with domestic organizations (cont.)

Type	Organization	Location
Universities	Bradley University	Peoria, Ill.
	California Institute of Technology	Pasadena, Calif.
	Cornell University	Ithaca, NY
	Howard University	Washington, D.C.
	Johns Hopkins University	Baltimore, Md.
	Michigan State University	East Lansing, Mich.
	Northwestern University	Evanston, Ill.
	Pennsylvania State University	University Park, Pa.
	Purdue University	West Lafayette, Ind.
	Rice University	Houston, Tex.
	Texas A & M University	College Station, Tex.
	University of California	Berkeley, Calif.
	University of Illinois	Urbana-Champaign, Ill.
	University of North Dakota	Grand Forks, N.D.
	University of Notre Dame	Notre Dame, Ind.
	University of Wisconsin	Madison, Wis.
	University of Wisconsin	Milwaukee, Wis.
Utility	Boston Edison	Boston, Mass.
	Commonwealth Edison Companies	Chicago, Ill.
	Consumers Power	Jackson, Mich.
	Dairyland Power	LaCrosse, Wis.
	Detroit Edison	Detroit, Mich.
	Duke Engineering Services	Charlotte, N.C.
	Electric Power Research Institute	Palo Alto, Ca.
	Metropolitan Water District	Los Angeles, Ca.
	Northeast Utilities Service Company	Hartford, Conn.
	Northern States Power	Minneapolis, Minn.
	Nuclear Management and Resources Council	Washington, D.C.
	Southern California Edison	San Diego, Ca.
	Southern Company Services	Birmingham, Ala.
	Virginia Power Company	Glen Allen, Va.

Table 2.2. SAG program contacts with foreign organizations

Country	Organization	Location
Australia	Cement and Concrete Assoc. of Australia Institute of Minerals, Energy & Construction	Sydney Highett, Victoria
Austria	International Atomic Energy Agency	Vienna
Belgium	Association Européenne du Ciment Center National de Recherches Scientifiques et Techniques pour l'Industrie Cimentière Center Scientifique et Technique de la Construction Studiecentrum voor Kernenergie	Brussels Brussels Brussels Mol
Canada	Atomic Energy of Canada Ltd. Canada Center for Mineral and Energy Technology Chalk River Laboratory Ministry of Transportation National Research Council of Canada Ontario Hydro University of Saskatchewan University of Toronto	Pinawa, Manitoba Ottawa, Ontario Chalk River, Ontario Downsview, Ontario Ottawa, Ontario Toronto, Ontario Saskatoon, Saskatchewan Toronto, Ontario
Czechoslovakia	Brno Technical University Research Institute of Civil Engineering-VUIS Technical University	Brno Bratislava Kosice
Denmark	G. M. Idorn Consultant A/S Technical University of Denmark The Road Directorate-Ministry of Transport	Birkerød Lyngby Copenhagen
England	AEA Technology British Cement Assoc. Building Research Establishment Central Electricity Generating Board Central Electricity Generating Board Construction Industry Research and Information Assoc. Fédération Internationale de la Précontrainte Imperial College Institution of Civil Engineers Institution of Structural Engineers Nuclear Electric	Oxfordshire Wexham Springs, Slough Garston, Watford Barnwood, Gloucester Leatherhead, Surrey London London London London Bedminster Down, Bristol

Table 2.2. SAG program contacts with foreign organizations (cont.)

Country	Organization	Location
England (cont.)	Nuclear Installations Inspectorate Taywood Engineering Ltd. The Concrete Society University of Bristol	Bootle, Merseyside Southall, Middlesex London Bristol
Federal Republic of Germany	Beuth Verlag GmbH Bundesanstalt für Materialforschung und Prüfung (BAM) Deutscher Ausschuss für Stahlbeton Deutscher Beton-Verein e.v. Fachinformationszentrum-Workstoffe e.v. Hochschule für Architektur und Bauwesen IBR Institute für Bautechnologie Institute für Bauforschung (IBAC) Institute für Baustoffe Institute für Massivbau und Baustofftechnologie Institute für Nukleare Sicherheitsforschung der Kernforschungsanlage (KFA) Kernforschungszentrum Karlsruhe (KfK) Lehrstuhl für Baustoffkunde und Werkstoffprüfung der TU München Stransgenberg, Schnellenbach and Partner GmbH Technische Hochschule Darmstadt	Berlin Berlin Berlin Weisbaden Berlin Weimar Ratingen Aachen Braunschweig Karlsruhe Jülich Karlsruhe Munich Bochum Darmstadt
Finland	Technical Research Center of Finland	Espoo
France	Centre Experimental de Recherches et d' Etudes du Batiment et des Travaux Publics (CEBTP) Institut de Protection et Sureté Nucleaire (IPSN/DAS) Electricity de France (EDF) Réunion Internationale des Laboratoires d' Essais et de Recherches sur les Matériaux et les Constructions	Saint Remy les Chevreuse Fontenay-aux-Roses Villeurbanne Cachan
Greece	National Technical University at Athens	Athens
India	Babaha Atomic Research Center	Trombay, Bombay
Israel	Israel Institute of Technology	Haifa

Table 2.2. SAG program contacts with foreign organizations (cont.)

Country	Organization	Location
Italy	Comitato Nazionale per la Ricerca e per lo sviluppo del' Energia Nucleare e delle Energie Alternative (ENEA) Ente Nazionale per l' Energia Elettrica (ENEL) Industria Italiana del Cemento Istituto Sperimentale Modelli e Strutture (ISMES)	Rome Milan Rome Bergamo
Japan	Architecture Institute of Japan Building Research Institute-Ministry of Construction Central Research Institute of Electric Power Industry (CRIEPI) Japan Atomic Energy Research Institute (JAERI) Japan Concrete Institute Japan Power Engineering and Inspection Kajima Institute of Construction Technology Koma Research Laboratory Nihon University Obayashi Research Institute of Shimizu Corp. Shimizu Corp. Takenaka Komuten Technical Research Laboratory Technical Research Institute - Obayashi Tokyo Institute of Technology University of Kyoto University of Tokyo	Tokyo Ibarku-ken Toyko Ibarki-ken Tokyo Tokyo Tokyo Tokyo Chiba-ken Tokyo Kota-ku Tokyo Tokyo Tokyo Kanagawa Kyoto Tokyo
Korea	Korea Electric Power Corp. (KEPCO) Korea Power Engineering Co. (KOPEC)	Daejeon Seoul
Luxembourg	Commission of European Communities	
Netherlands	Center for Civil Engineering Research and Codes Conseil International du Batiment Pour la Research L'Etude et la Documentation (CIB)	Gouda Rotterdam
Portugal	Technica de Lisboa	Lisbon
Republic of China	Taiwan Power Co.	Taipei
Romania	Institutul de Percetări in Constructii	Bucharest
South Africa	Portland Cement Institute	Johannesburg

Table 2.2. SAG program contacts with foreign organizations (cont.)

Country	Organization	Location
Sweden	Institute for Byggnadssatik	Stockholm
	Institute for Fire Safety Design (IDEON)	Lund
	Lund University	Lund
	National Swedish Institute of Building Research	Gävle
	Swedish Cement and Concrete Research Institute	Stockholm
Switzerland	Swedish Council for Building Research	Solna
	Basler & Hofmann	Zurich
	Comite Euro-International du Beton (CEB)	Lausanne
	Hauptabteilung für die Sicherheit der Kemonlagen	Würenlingen
	International Association for Bridge and Structural Engineering	Zurich
Yugoslavia	Institute for Materials in Research and Structures (ZRMK)	Maribor

3. TASK S.2 – MATERIALS PROPERTY DATA BASE

3.1 OBJECTIVE

The objective of the materials property data base task is to develop a reference source which contains data and information on the time variation of material properties under the influence of pertinent environmental stressors and aging factors. The data base will have use in the prediction of potential long-term deterioration of critical structural components in nuclear power plants and in establishing limits on hostile environmental exposure for these structures. The results also will have application to the establishment of maintenance or remedial measures programs that will assist in either prolonging component service life or improving the probability of the component surviving an extreme event such as a loss-of-coolant accident. Figure 3.1 presents the Level 2 work breakdown structure for the materials property data base task.

3.2 BACKGROUND

One of the findings in NUREG/CR-4652 (Ref. 3.1) was that materials property data for concrete over an extended time period are limited. This is especially true for concretes which have been subjected to aging factors or environmental stressors characteristic of those that could occur in a nuclear power plant (see Ref. 3.1). Figure 3.2 obtained from Ref. 3.2, presents modes of degradation that were considered for the concrete components in a particular nuclear plant (Surry Unit One). Another limitation on materials property data availability is that in order for the data to be considered as being of "high quality," detailed information such as presented in Table 3.1 is desired. Unfortunately, for most structures that have been in service for the period of interest, 30 to 100 years, either detailed information about constituent materials, mix designs, plastic concrete properties, curing procedures, and exposure conditions are not available, or the time variation of material properties is unknown. Three potential means are available, however, which can be utilized to supplement the modest existing data base: (1) obtaining and testing samples from existing nuclear power plants, non-nuclear power generating facilities, and general civil engineering structures; (2) reviewing, analyzing, and trending existing data available through nuclear power plant prestressing tendon in-service inspection and integrated leak-rate test records; and (3) utilizing accelerated aging techniques.

3.3 PLAN OF ACTION

The Level 3 work breakdown structure for this task is presented in Fig. 3.3.

A vital component of the SAG Program is the development of the Structural Materials Information Center (SMIC). The SMIC consists of two formats so that information is useful to structural engineers, stress analysts, and materials engineers; for instance, the data base is being developed as an expandable, hard-copy handbook version (*Structural Materials Handbook*) and as an electronic version (*Structural Materials Electronic Data Base*) for use on an IBM or IBM-compatible personal computer. The handbook will serve as the information source upon which the electronic data base is built. Initially, information related to concrete, metallic reinforcements, prestressing steels, and structural steels are

being incorporated into the data base. As the data base is developed, other structural materials will be added.

In parallel with the efforts to develop SMIC, numerous activities are being conducted relative to development of materials property data for input into SMIC. Potential sources of information include cognizant foreign research establishments in Europe, North America, and Asia; domestic research establishments such as government facilities, universities, and consulting/engineering firms; and obtaining and testing of aged material samples from prototypical nuclear power plants or civil works structures. The open literature is being reviewed to identify representative (well documented) data for incorporation into SMIC, for example, technical reports, professional journal articles, and communications with fellow researchers.

Prediction or explanation of the complex inter-relationships that occur between concrete's constituents and between concrete and its environment requires the development of mathematical models based on scientific and engineering principles. Such models play a vital role in the development of reliability-based life prediction techniques for concrete structures in nuclear power plants (Task S.4). Under this activity, modeling studies were conducted to complement those being conducted at the National Institute of Standards and Technology under another NRC program, "Performance Criteria for Concrete-Based Barriers for Low-Level Waste Disposal Facilities." Models developed under the SAG Program address aging factors and environmental stressors pertinent to safety-related concrete structures in nuclear power plants, as well as incorporate synergistic effects to predict behavior when more than one environmental stressor or aging factor is present.

3.4 SUMMARY OF ACCOMPLISHMENTS

During the reporting period covered by this document, activities have been conducted under each of the subtask areas shown in Fig. 3.1.

3.4.1 Structural Materials Information Center (Subtask S.2.1)

Utilizing results of a review and assessment of materials property data bases (Ref. 3.3), and a plan which had been prepared for development of the Structural Materials Information Center (SMIC) (Ref. 3.4), initial formatting of SMIC has been completed and the results presented in a report (Ref. 3.5). Contained in the report are more detailed descriptions of the *Structural Materials Handbook* and the *Structural Materials Electronic Data Base* than provided below.

3.4.1.1 Structural Materials Handbook

The *Structural Materials Handbook* is being developed as an expandable, hard-copy reference document that contains complete sets of data and information for each material in SMIC. The handbook consists of four volumes that are provided in loose-leaf binders for ease of revision and updating. Volume 1 contains design and analysis information useful for structural assessments and safety margins evaluations, for example, performance curves for mechanical, thermal, physical, and other properties presented as tables, graphs, and mathematical equations. Test results and data used to develop the performance curves in Volume 1 are provided in Volume 2. Volume 3 contains material data sheets which

provide general information, as well as material composition and constituent material properties, for each material system contained in the handbook. Volume 4 contains appendices describing the handbook organization, as well as updating and revision procedures.

Initially, Volumes 1, 2, and 3 of the handbook will each contain four chapters of materials property data and information, with the chapters consistent between the volumes. Each material in the data base is assigned a unique seven-character material code which is used in the handbook and the electronic data base to organize materials with common characteristics. This code consists of a chapter index, a group index, a class index, and an identifier. The chapter index is used to represent the various material systems in the data base. The group index is used to arrange materials in each chapter into subsets of materials having distinguishing qualities such as common compositional traits. The class index is used to organize groups of materials with common compositional traits into subsets having a similar compositional makeup or chemistry. The identifier is used to differentiate structural materials having the same chapter, group, and class indices according to a specific concrete mix, ASTM standard specification for metallic reinforcement, etc. The arrangement of the material code parameters is shown in Fig. 3.4. Table 3.2 lists material code parameters that have been developed for the concrete and concrete-related materials presently in the data base.

A wide variety of information and materials property data is being collected and assembled for each material system included in the data base, for example, general description, composition, mechanical property data, etc. In setting up the data base, each material property has been identified by a unique four-digit property code selected from an established set of material property categories. Table 3.3 lists the property code ranges and corresponding material property categories. Reference 3.5 presents a breakdown of the individual material property code values in each of these ranges.

Associated with each entry of data (numerical results of tests) or values (results of evaluation of data) into the data base is an assessment of the quality of the entries presented in the form of a letter grade. Although the criteria for assessing the quality of data and values are somewhat subjective, five quality levels have been developed. These levels are represented, in order of descending quality, by the letters A through E, Table 3.4. The 11 requirements utilized in the evaluation of the quality of data and values are listed in Table 3.5 with specific criteria for each of the quality levels provided in Ref. 3.6.

Each reference document that is used as an information source is assigned a unique integer identifier. In Volumes 1 and 2, reference numbers are listed to identify each information source, and all references that are used to develop a reported property for a particular material are provided in Volume 3. Since each reference may be used for more than one property or structural material, a complete listing of references appears in Appendix E of Volume 4. The integer identifier assigned to each reference source is consistent in both the handbook and the electronic data base.

3.4.1.2 Structural Materials Electronic Data Base

The *Structural Materials Electronic Data Base* is an electronically accessible version of the *Structural Materials Handbook*. It has been developed on an IBM-compatible personal computer using a commercially-available data base management system designed specifically for maintaining and displaying properties of engineering materials. To ensure that the handbook and electronic data base are compatible, each material included in the electronic data base is identified by the same common name and material code that has

been used to represent the material in the handbook. Also, each electronic data base material record contains data and information taken directly from the handbook. Due to software limitations, the electronic data base is not as comprehensive as the handbook, but it does provide an efficient means for searching the various data base files to locate materials with similar characteristics or properties.

The electronic data base management system includes two software programs: Mat.DB (Ref. 3.7) and EnPlot (Ref. 3.8). Mat.DB is a menu-driven software program that employs window overlays to access data searching and editing features. This software is capable of maintaining, searching, and displaying textual, tabular, and graphical information and data contained in electronic data base files. Although Mat.DB has been developed for metallic materials, its formatting can be modified to accommodate nonmetallic and composite materials such as portland cement concretes. EnPlot is a software program that incorporates pop-up menus for creating and editing engineering graphs. This software includes curve-fitting and scale-conversion features for preparing engineering graphs and utility features for generating output files. The engineering graphs generated with EnPlot can be entered directly into the Mat.DB data base files. These graphs are compatible with Microsoft Word, the word processing software used to prepare the handbook. The software was developed to run under Microsoft Windows Version 3.0 on IBM personal computers, or compatibles, using an Intel 80286 Processor, or higher, and DOS 3.1, or higher.

Each material record in the electronic data base could include up to nine major categories of data and information: designations, specifications, composition, notes, forms, graphs, properties, classes, and rankings (see Ref. 3.5). The user may search an entire data base file to locate materials with similar material properties. During the search each material may be screened for selected tabular data and certain property values based on comparison operators (for instance, =, >, <, >=, <=, and <>). The user may elect to perform property searches using either the International System (SI) of units or customary (English) units.

3.4.2 Data Collection (Subtask S.2.2)

One of the findings of Ref. 3.1 was that documented long-term and environment-dependent materials property data for concrete are limited. This is especially true for concretes which have been subjected to aging factors or environmental stressors characteristic of those that could occur in a nuclear power plant. Another limitation on material property data is that in order for the data to be considered "high quality," detailed information such as presented in Table 3.1 is desired. Unfortunately, for most concrete structures that have been in service for the period of interest (30 to 100 years), either detailed baseline information or the time variation of the properties of the construction materials is not available. Two approaches have been utilized in an attempt to expand the somewhat limited data base: (1) pursuing technology exchange with U.S. and foreign research establishments and (2) obtaining and testing of prototypical concrete materials.

3.4.2.1 Technology Exchange

Japanese research organizations were visited to participate in meetings with technical managers and research engineers to learn about Japanese aging-management programs and research activities that focus specifically on issues similar to those being addressed by the SAG Program. Organizations visited included Central Research Institute of Electric Power

Industry, Japan Atomic Energy Research Institute, Japan Power Engineering and Inspection Corporation, Kajima Technical Research Institute, Obayashi Corporation, Shimizu Corporation, and Takenaka Technical Research Laboratory. During these meetings discussions were held to identify potential sources of long-term and environment-dependent materials property data suitable for entry into the Structural Materials Information Center. In addition, efforts were made to obtain background information on the in-service inspection programs, nondestructive examination and repair procedures, and service life prediction methodologies for existing concrete structures. A total of 62 documents were provided for use by the SAG Program. Contained in these documents were several papers presenting results of elevated temperature testing of concrete, data obtained from testing of concrete cores removed from the biological shield wall of the Japan Power Demonstration Reactor, results of studies investigating various coating materials to protect steel reinforcement from corrosion in a salt environment, development of methods to estimate the 500-year behavior of concrete, fabrication of concrete mixes containing boron (colemanite) to act as neutron absorbing material at a Japanese fuel reprocessing facility, formulation of admixtures to increase the durability of concrete by reducing the concrete air content and absorbing the carbon dioxide and chloride ions that enter the concrete surface, a description of a rebar inspection system based on a conventional eddy current device, and a methodology for predicting the service life of reinforced concrete buildings based on the rate of corrosion of the reinforcing steel. Reference 3.9 presents a summary of the visit to Japan and a listing of the documentation that was obtained as a result of the visit.

In addition to the activity described in the previous paragraph, efforts have continued to obtain concrete material properties information suitable for input into the data base. Table 3.6 presents a listing of organizations from which information and reports on concrete properties have been acquired.

3.4.2.2 Prototypical Sample Procurement

In parallel with the efforts to obtain concrete properties data, several U.S. utilities, national laboratories, and concrete research organizations have been contacted to pursue the possibilities of obtaining concrete core samples from prototypical structures for testing. A summary of these contacts and the status of the discussions is presented in Table 3.7. These contacts have resulted in procurement of samples from Shippingport Power Station (Battelle Pacific Northwest Laboratories); EBR-II site (Argonne National Laboratory-West); Palisades and Midland Power Stations (Consumers Power Co.); Braidwood, Byron, Dresden, LaSalle, Quad Cities, and Zion Power Stations (Commonwealth Edison Co.); and Vallecitos Nuclear Center (General Electric). Discussions have been terminated relative to obtaining samples from the Fermi (Detroit Edison), Pathfinder (Northern States Power), LaCrosse (Dairyland Power Cooperative), and CintiChem Plant (Union Carbide) stations due to either the utility not being interested, or baseline records not being available. Discussions are still in progress relative to the possibility of obtaining concrete core samples at the Experimental Boiling Water Reactor and Chicago Pile-5 reactors (Argonne National Laboratory). Subcontracts have been placed with Sargent & Lundy Engineers and Taywood Engineering Ltd. to provide data on prototypical concrete and concrete-related materials properties.

The purpose of the subcontract with Sargent & Lundy Engineers (Chicago, IL) is to locate, review, evaluate and provide baseline material property information regarding the concrete materials utilized in the construction of Commonwealth Edison Company's

(CECO) nuclear power stations at Braidwood, Byron, Dresden, LaSalle, Quad Cities, and Zion. Sargent & Lundy Engineers also is assisting in the obtaining and shipment of concrete core samples to ORNL for testing. The samples are provided from CECO facilities when modifications are made that require the coring and removal of concrete materials. In addition to the core samples, information is being provided which identifies the plant from which the samples were obtained, the location in the plant, and any available background information (constituents, mix designs, environments, etc.). To date, background information has been provided pertaining to the construction specifications and reference concrete compressive strength data (90-day) for each of the six stations. Also, concrete core specimens have been provided from each of the plants. Over 100 test specimens have been prepared for testing from the concrete cores provided to date.

The overall objective of the subcontract with Taywood Engineering Ltd. (London, England) was to provide data from archived test specimens that had been cast in conjunction with the fabrication of several of the United Kingdom nuclear power stations. Over 100 test specimens, generally 450-mm long by 150-mm diameter and having ages from 4 to 24 years, were available from the Wylfa, Heysham I, Heysham II, Hartlepool, Torness and Sizewell "B" stations. A small number of specimens 300-mm long by 100-mm diameter were also available from the Wylfa station. With the exception of the Wylfa station specimens, all the specimens had been initially cured under heat cycled conditions to simulate the early in-situ temperature rise due to cement hydration. Each specimen had been fitted with embedded measurement studs for strain measurements (200-mm gage length) using a Demec gage. Most of the specimens also contained vibrating wire strain gages (200-mm gage length). The specimens had been continuously stored in a sealed, stable moisture state at temperatures from 10° to 95°C with some having been under sustained loading (13.8 MPa). Available baseline data for the six concretes include compressive strength (up to 1 year), thermal expansion, thermal conductivity, and details on constituent materials and mixture proportions. Also, a limited number of cylinders for the Heysham I and Heysham II concretes had been tested for elastic and creep recovery, and for compressive strength at ages to 4 years. Elastic modulus and creep results are available for the Wylfa concrete up to an age of 12.5 years. Twenty-nine specimens of the over 100 available were selected for testing to provide information on the long-term performance of nuclear grade concretes. Variables investigated included age of specimen, concrete mix design, loaded or unloaded while curing, and storage temperature. The test procedure followed consisted of six steps: (1) measurement of the static modulus of elasticity up to loads of one-third the estimated compressive strength; (2) cutting the specimen into three 150-mm lengths and placing the center section containing the Demec gage points and vibrating wire strain gage in storage; (3) sealing the two remaining end sections with moisture-resistant plastic and storing until time of test; (4) unsealing each of the specimen end sections and subjecting them to a series of nondestructive tests (density, ultrasonic pulse velocity, Schmidt hammer, surface hardness, and dynamic modulus of elasticity); (5) resealing one of the two end sections and returning it to its original storage temperature for future monitoring and testing; and (6) saturating the remaining end section in water, testing it for saturated density, capping the section ends, and testing to determine its compressive strength. Results indicate that, except for one or two exceptions, there was a consistent trend for the concrete moduli of elasticity and compressive strengths to increase with age for each of the mixes. From the limited results which were available for specimens that were continuously stored unloaded at 10° to 20°C, increases in modulus of elasticity relative to 28-day reference values ranged from 3% for the Sizewell "B" concrete (test age of 4.8 years) to 112% for the Wylfa concrete (test age of 21 years). Relative to

the 28-day reference values, increases in compressive strength values obtained from cube test specimens ranged from 12% for the Torness concrete (test age of 11.7 years) to 40% for the Heysham I concrete (test age of 23.6 years). Details on specimens tested, test procedures and test results are presented in Ref. 3.10.

Taywood Engineering Ltd. (London, England), as part of their overall subcontract in support of the SAG Program, collated surveillance data for several prestressed concrete pressure vessels in the United Kingdom. Prestressing tendon anchorage lift-off load measurement data, results of corrosion examinations, and visual examination results were compiled from surveillance records for the vessels at the Wylfa, Hartlepool, and Heysham I stations. Two prestressed concrete pressure vessels are located at each station. Results presented cover time periods since prestressing of 23, 14.5, and 13.4 years for the Wylfa, Hartlepool, and Heysham I stations, respectively. As expected, the tendon anchorage lift-off load measurements showed a general trend for the loads to decrease with time due to a combination of prestressing steel relaxation and concrete creep. Examination of prestressing strands removed for inspection and testing revealed only a few minor, structurally insignificant pits indicating that the combination of waxes and greases used to inhibit corrosion of the ungrouted prestressing systems has been effective. Tensile test results for the prestressing strands exceed design requirements. Visual examination of the concrete surfaces revealed a few surface cracks, with the cracks <0.30-mm wide and when active, their rate of growth was <20 microns per year. The cracks were associated with drying shrinkage. More detailed surveillance results for each of the three power stations are contained in Ref. 3.11.

3.4.2.3 Materials Property Data Bases Completed for SMIC

Materials property data bases have been prepared for each material listed in Tables 3.8 - 3.12. A summary description of each of the data base sources is presented below.

Portland Cement Concretes

Concrete properties for the data bases listed in Table 3.8 are contained in Chapter 1 (Vols. 1–3) of the handbook and electronic data base file CONCRETE.DB.

University of Wisconsin long-term study. Three experimental studies, Series A, B, and C, have been conducted at the University of Wisconsin on the long-term behavior of concrete. These studies began in 1910 and focused on compressive strength, unit weight, and volume changes of concrete materials exposed to the climate in Madison, Wisconsin. Results from Series B, which began in 1923, have been entered into the Structural Materials Information Center.

In Series B, test specimens were prepared from six concrete mixes. Each mix consisted of four batches of concrete. Each batch contained a portland cement that was produced by one of four different vendors and one type of coarse aggregate. Three types of 38-mm maximum size coarse aggregate were used in the mixes: a crushed dolomite, a crushed red granite, and a gravel. Cement contents in the mixes ranged from 200 to 369 kg/m³ and water-cement ratios from 0.41 to 0.69. No chemical or mineral admixtures were used in the mixes. The test specimens were cast, cured for 28 days, stored either inside (controlled environment) or outside, and tested at selected intervals over a 50-year period. Each of the six data bases include average compressive strength values as a function of time. Average reference 28-day compressive strengths ranged from 10 to

19.2 MPa. Since tests were performed at selected times over a 50-year period, the compressive strength vs time performance curves are considered high quality and have been rated Quality Level A.

Portland Cement Association long-term study. Under a subcontract with Construction Technology Laboratories (CTL), Inc., in Skokie, Illinois, results have been assembled and provided for three of four series of tests from a study that has been ongoing at the Portland Cement Association since the early 1940s to investigate the strength-producing properties of several cements. The overall program encompassed about 500 concrete mixes fabricated using a wide variety of cement and aggregate materials. Beam and cylindrical test specimens were either moist cured (continuous storage in a moist room at 100% relative humidity and 23°C), air cured (moist cured for 7 days followed by laboratory storage at ~50% relative humidity and 21° to 23°C), air cured followed by soaking (moist cure for 7 days, air cured till 48 hours before testing, soaking in water at 23°C for 48 hours), cured outdoors at Skokie, Illinois (moist cure for 7 days, store outdoors on a clay loam), or cured outdoors at Dallas, Texas (moist cure for 7 days, store outdoors on a sandy soil) until testing. Periodically, at ages from 1 day to 34-years, specimens were tested to determine values of compressive strength, modulus of elasticity, and modulus of rupture as a function of materials, environment, and time. Reference 3.12 presents summary descriptions of the overall program, material characteristics, mixture proportions, specimen geometries, curing conditions, test methods, and available data. Detailed results on the material characteristics, mix designs, curing conditions, and test results for the three test series are provided as appendices to the report. Each of the three appendices to the report corresponds to a particular test series and is divided into two volumes (Volume 2 - Supporting Documentation and Volume 3 - Material Data Sheets) in accordance with formatting requirements for the ORNL Structural Materials Information Center (Ref. 3.5). Of the over 500 concrete mixes prepared, 51 have been selected to provide information on a wide variety of cement types and curing conditions and material property (flexure strength, compressive strength and dynamic modulus of elasticity) versus time performance curves developed and entered into the Structural Materials Information Center.

A second study was conducted at Construction Technology Laboratories (CTL), Inc. to provide additional data for one of the four test series that had been cast to investigate the strength-producing properties of several cements. Twenty-two archived beam test specimens that had been continuously moist cured (100% relative humidity and 23°C) since being cast in 1950 were tested. Each 152.4- by 152.4- by 762-mm specimen was first tested in flexure using a 457.2-mm end of the beam to obtain a modulus of rupture value. Pulse velocity and unit weight were then measured on the short end of the beam. Cubes were cut from the two ends of the beam for compression tests. Also, a 25.4-mm slice of the beam was cut for petrographic examination and two nominal 101.6-mm diameter cores were extracted from the remainder of the beam for compression and unit weight tests. These tests increased the period for which the performance curve versus time results presented in Ref 3.12 for this test series were available from 34 to 42 years. Average compressive strength results obtained from these specimens at the test age of 42 years ranged from 30.9 to 78.9 MPa. Relative to the 28-day reference compressive strength values, increases in compressive strength for the mixes ranged from 24% (253 kg/m³ Type III cement; w/c of 0.6, by weight) to 265% (168 kg/m³ Type IV cement; w/c of 0.78, by weight). Additional details on the mix designs, testing procedures and test results are presented in Ref. 3.13. Since tests were performed at selected times throughout the duration of the study, baseline data on constituent materials and mix designs are

available, and exposure conditions are well documented, the material property versus time performance curves are considered high quality and have been rated Quality Level A.

University of Saskatchewan (Canada) elevated-temperature study. The effect of elevated temperature on the modulus of elasticity and compressive strength of concrete was investigated in a 1981 study at the University of Saskatchewan. The concrete specimens (235-mm long by 75-mm diameter) were made using ASTM C 150, Type V sulfate-resistant portland cement, ASTM C 618, Type C fly ash from lignite coal, and 19-mm maximum size crushed stone coarse aggregate. Also included in the mix were an air-entraining agent, a conventional water-reducing admixture, and a high-range water-reducing admixture (superplasticizer). The average 28-day compressive strength and modulus of elasticity values for the concrete mix were 24.8 MPa and 24.5 GPa, respectively. After moist curing for 28 days, the specimens (sealed and unsealed) were placed into environmental chambers and heated or cooled to their designated soak temperatures. Sets of specimens were subjected to one of seven temperature levels ranging from -11° to 232°C. At the end of each of ten exposure periods, which ranged from 1 day to six months, a minimum of three sealed and three unsealed specimens were removed from the environmental chambers, permitted to thermally stabilize at room temperature, and then tested in compression. Since only average compressive strength results are presented, the data have been rated Quality Level B.

CANMET (Ottawa) long-term elevated temperature study. Six normal-weight concretes were prepared as part of a Canada Centre for Mineral and Energy Technology research program to develop long-term strength data on concretes exposed to sustained elevated temperature. The first phase of the study was initiated in 1979 for the purpose of examining normal portland cement, normal portland cement plus slag, and normal portland cement plus fly ash concretes.* Either limestone or dolostone were used as the coarse aggregate materials in the mixes which had water-cement ratios ranging from 0.33 to 0.60. Cylindrical test specimens (203-mm long by 102-mm diameter) were cast from each mix, moist cured for 28 days, and then permitted to air dry in the laboratory for 26 weeks prior to elevated temperature exposure. Each specimen was slowly heated to a prescribed thermal soak temperature which covered a temperature range from 75° to 600°C, maintained at the soak temperature for up to four months, and then permitted to cool to room temperature. Tests conducted at room temperature following thermal exposure included pulse velocity, resonant frequency, and compressive strength. Average reference 28-day compressive strengths obtained from moist-cured specimens ranged from 32.9 to 57.7 MPa for the concrete mixes. Since only average test results are presented, results have been rated Quality Level B.

Fuel cycle facility at Argonne National Laboratory-West. The fuel cycle facility for the EBR-II reactor was constructed by the Argonne National Laboratory in 1959 and 1960 near Idaho Falls, Idaho, using normal-weight and heavyweight concrete materials. The normal-weight concrete mixes were fabricated using ASTM C 150, Type I portland cement ranging in content from 288 to 307 kg/m³, 38-mm maximum size gravel aggregate, and water-cement ratios ranging from 0.42 to 0.57. The heavyweight concrete mix used a ASTM C 150, Type I portland cement content of 301 kg/m³, crushed stone coarse aggregate, and a water-cement ratio of 0.42. In addition, 16-mm diameter steel shot were added to the mix to increase its density and shielding effectiveness. During construction, plastic concrete properties were obtained and cylindrical concrete test samples

* Only the test results for the normal portland cement concretes are presented in the data base.

were cast for each of the mixes. These samples were moist cured and tested at ages of 7 and 28 days to verify that the material met its design compressive strength requirements. Average 28-day reference compressive strengths for the normal-weight and heavyweight concrete mixes were approximately 35 and 38 MPa, respectively.

In 1989, 102-mm diameter core samples were removed from the footings, subbasement wall, biological shield inside the reactor pit building, and argon cell shield walls and shipped to ORNL for testing. Using these core test results and baseline data available from construction records, data bases for these materials were developed. Each data base includes a compressive strength vs time performance curve, but because only limited compressive strength (28-day and approximately 30-year) test results were available, each performance curve was rated Quality Level C.

Walz (Germany) long-term study. Concrete cube and slab sections were cast in 1941 by the Research and Materials Test Center for Construction at the Technical University in Stuttgart as part of a long-term study to investigate the effects of weathering on concrete and stone materials. The eight concrete materials used to cast the 200-mm cube and 150-by 400- by 700-mm slab test specimens were fabricated using cement which essentially met ASTM C 150 requirements for Type I portland cement, gravel aggregates, and water. No chemical or mineral admixtures were included in the mixes. Two nominal cement contents were used in the mixes, 200 kg/m³ and 300 kg/m³. Water-cement ratios for the 200 kg/m³ concrete mixes ranged from 0.78 to 1.29, and from 0.50 to 0.82 for the 300 kg/m³ cement-content concretes. Six cube and 20 slab test specimens were cast from each mix. Three of the cubes were moist cured for 28 days and then tested in compression. Reference 28-day compressive strength values for the 200 kg/m³ concrete mixes ranged from 5.4 to 18.6 MPa, and from 15.5 to 36.5 for the 300 kg/m³ cement-content concretes. The other three cubes were continuously moist cured for 14 days and then alternately wetted and dried in air until tested in compression at an age of 2.5 years. Two of the slab sections from each mix were exposed to 100 rapid freezing-thawing cycles in the laboratory prior to testing in flexure. Six of the slab sections from each mix were stored outdoors near Stuttgart, and six each were half submerged in slow moving rivers near Blaubeuren and Partenen. After 2-3/4 and 12 years of exposure, two of the slab sections stored near Stuttgart were tested in flexure. The remaining two slab sections remained near Stuttgart until an age of about 30 years at which time 100-mm-diameter cores were removed and tested to assess depth of carbonation and compressive strength. After 12 years of exposure, three each of the slab sections at Blaubeuren and Partenen were tested in flexure. Two additional slab sections for each mix were removed from these sites at an age of about 29 years and 100-mm-diameter cores taken and tested to assess depth of carbonation and compressive strength. Due to the limited number and differences in size of the test specimens, the time-dependent compressive strength properties for these materials have been rated Quality Level D.

Kondo (Japan) stray electrical current corrosion study. Nine normal-weight concrete mixes were prepared at Kyoto University as a part of a research program to investigate the electrolytic corrosion of reinforcing steel in concrete resulting from stray electrical currents. The test specimens were cast from concrete mixes that contained various combinations of fly ash (0 to 30%, by total weight cementitious materials), calcium chloride (0 to 1.0%, by weight cement), and calcium lignosulfate (0 to 0.25%, by weight cement). No other chemical or mineral admixtures were used in the mixes. Total cementitious materials contents for all the mixes was 320 kg/m³ and the water-cement ratios ranged from 0.49 to 0.58. A 25-mm maximum size river gravel was used as the coarse aggregate material in

all the mixes. Twelve bond test specimens (150-mm cubes) were typically cast from each concrete mix. Embedded in each concrete cube were a 0.423-mm thick galvanized iron plate and a No. 6 deformed steel reinforcing bar. After casting and demolding, specimens were submerged in 21°C water for five days. Upon removal from the water, each specimen was immediately sealed with a membrane compound to prevent moisture transfer. After nine additional days of curing in a laboratory environment, nine specimens from each batch were connected to direct current power supplies with the reinforcing bars serving as anodes and the galvanized plates as cathodes. Three each of the specimens were connected to 5-volt, 10-volt and 20-volt power supplies. Current was applied to each specimen for 3 hours daily for 14 days. Three of the specimens were not subjected to direct current and served as controls. Additional specimens were cast from two of the concrete mixes to investigate the effect of 20-volt alternating current supply and the effect of switching the anode and cathode locations. At conclusion of the 14-day exposure period, the specimens were tested in accordance with ASTM C 234-54 requirements to evaluate the effect of electrical current on the concrete-steel reinforcement bond. Since the compressive strength at the time of testing was not reported, the bond stress versus slip performance curves have been rated Quality Level B.

SABNGS long-term study. Concrete test cylinders (305-mm long by 152-mm diameter) were cast by Ontario Hydro in conjunction with fabrication of the Sir Adam Beck Niagara Generating Station (SABNGS) No. 2 in 1953. This study is part of Ontario Hydro's dam safety program in which a wide variety of concretes were prepared, ranging from very lean, without admixtures, to air entrained with high cement content. The objective of the program is to investigate the long-term performance of these materials. Sufficient control specimens were cast at each site to periodically obtain concrete compressive strength data at ages from 7 days to 100 years. After fabrication, each specimen was cured in a moist room located at the site until the field laboratory was dismantled. The remaining specimens were then either temporarily buried at the site or placed into tunnels at the facility. Later all remaining samples were sent to the Ontario Hydro Research Center (Toronto) where they were buried in the ground to a minimum depth of one meter to expose the specimens to moist curing without the threat of freezing and thawing. During the summer prior to testing, specimens were removed from the ground and stored in a laboratory moist room until scheduled for testing.

The SABNGS concrete was made using normal portland cement, fly ash, natural sand, and 38-mm maximum size coarse aggregates. A total of nine different mixes were used to fabricate the generating station, three mixes each having cementitious materials contents of either 281, 311, or 363 kg/m³. The first set of three mixes contained no chemical or mineral admixtures, the second set contained an air-entraining admixture and the third set contained no air-entraining admixture but fly ash provided 30% of the total weight of cementitious materials. Reference 28-day compressive strengths for the mixes ranged from 25.0 to 39.4 MPa. To date, compressive strength tests have been conducted at 7, 28, and 90 days, and 1, 2, 3, 5, 10, 20, and 25 years. Sufficient specimens remain for testing at ages of 50 and 100 years. Since tests were conducted at selected times over a 25-year period and all baseline characteristics for the materials were reported, the compressive strength versus time performance curves are considered high quality and rated Quality Level A.

Stewartville (Canada) Generating Station long-term data. Concrete cylinders (305-mm long by 152-mm diameter) were cast by Ontario Hydro in conjunction with fabrication of the Stewartville Generating Station in 1948. This study was also part of Ontario Hydro's dam safety program and thus utilized the same specimen preparation,

curing, and testing procedures as the SABNGS study. The Stewartville concrete was made using normal portland cement, natural sand, and 38-mm maximum size coarse aggregate. No mineral or chemical admixtures were added to the mixes. Two main mixes were used in construction of the generating station, a face concrete and a core concrete. The face concrete mix had a water-cement ratio of 0.70 and a cement content of 274 kg/m³. The core concrete had a water-cement ratio of 0.98 and a cement content of 334 kg/m³. Reference compressive strengths for the two mixes at 28 days age were 25.1 and 14.1 MPa, respectively. To date, compressive strength tests have been conducted at 28 and 90 days, and 1, 2, 3, 5, 10, 15, and 25 years. Sufficient specimens remain for testing at ages of 50 and 100 years. Since tests were conducted at selected times over a 25-year period and all baseline characteristics for the materials were reported, the compressive strength versus time performance curves are considered high quality and rated Quality Level A.

High-density shielding wall long-term study. Background data and test results were provided by Ontario Hydro for a high-density shielding wall concrete. The 2.13-m thick wall was constructed as part of a Canadian nuclear power facility using a concrete mix containing hematite, ilmenite, and feldspar aggregate materials (20-mm maximum size) to produce a mix having a unit weight over 3400 kg/m³. An ASTM C 150, Type IV, low heat of hydration cement was used in a quantity of 348 kg/m³. The mix used a water-reducing admixture and had a 0.49 water-cement ratio. During construction a number of 300-mm long by 152-mm diameter cylindrical test specimens were cast, moist cured, and tested at ages of 7 and 28 days. At an age of 20 years, two 150-mm diameter cores 1.8-m long were taken from the 2.13-m thick wall. The cores were used to determine compressive strength, modulus of elasticity, specific heat, thermal diffusivity, conductivity, and coefficient of thermal expansion values for the shielding concrete. Due to the limited number and ages at which specimens were tested, the time-dependent compressive strength properties for these materials have been rated Quality Level C.

General Electric test reactor at Vallecitos Nuclear Center. Six 152-mm diameter by 750-mm long concrete core specimens were provided by the Vallecitos Nuclear Center for testing. The cores were removed from different sections of the biological shield of the 50-MW, pressurized, light-water cooled, high-flux reactor that began full-power operation in 1959. Three cores were normal-weight concrete and three cores were heavyweight concrete. The normal-weight concrete had been prepared using Type I portland cement, natural sand and 32-mm maximum size gravel coarse aggregate. The heavyweight concrete had been prepared using Type I portland cement, magnetite sand, 38-mm maximum size crushed magnetite coarse aggregate and a water-reducing chemical admixture. Design compressive strength at 28-days was 20.7 MPa for both concretes. Compressive strength test results are available for both mixes at concrete ages of 7, 28, and 90 days, and 20 and 34 years. Since constituent material information, mix designs, and plastic concrete properties are available as well as strength results at several concrete ages, the compressive strength versus time performance curves have been rated Quality Level B.

Midland Nuclear Power Station. Two normal-weight concrete cores from a floor slab located inside the auxiliary building and seven normal-weight concrete cores from two floor slabs located inside the Unit 2 containment building were provided by Consumers Power for testing. The concrete from the auxiliary building floor slab had been prepared using Type II portland cement, flyash, 19-mm maximum size crushed stone aggregate, and air-entraining and water-reducing admixtures. The concrete from the containment building floor slabs had been prepared using Type II portland cement, flyash, 38-mm maximum size crushed stone aggregate, and air-entraining and water-reducing admixtures. The auxiliary and containment building concretes had 28-day compressive strengths

of 34.3 and 41.4 MPa, respectively. Compressive strength test results are available for both mixes at concrete ages of 7, 28 and 90 days and 14 years. Although constituent material information, mix designs, and plastic concrete properties are available for these materials, no compressive strength results are available for concrete ages between 90 days and 14 years. The compressive strength versus time performance curves for these materials therefore have been rated Quality Level C.

Japan Power Demonstration Reactor (JPDR). The JPDR is a boiling water-type reactor that was constructed in the early 1960's at the Tokai Research Center and operated by the Japan Atomic Energy Research Institute from 1963 to 1976. Beginning in 1986, decommissioning and dismantling operations were initiated during which concrete cores were removed from the biological shielding walls for testing. The coring operations took place in three phases. During Phases I and II, horizontal cores were taken at several elevations and radial locations completely through the biological shielding wall which was as much as 3-m thick. During Phase III, cores were obtained from five vertical borings located adjacent to the steel liner that formed the reactor pressure vessel cavity boundary. During the coring operations, the exact location of each core was noted and fast neutron fluence for that location recorded. The concrete cores were tested to develop performance curves presenting the effect of fast neutron fluence on the concrete compressive strength, splitting-tensile strength and modulus of elasticity. In addition, several samples representing different fluence levels were examined using a scanning electron microscope. Fast neutron fluence exposures ranged from about 10^{11} to 10^{17} neutrons/square centimeter ($E > 0.1$ MeV). The concrete used to construct the biological shielding wall had a 28-day design compressive strength of 24.5 MPa and was prepared from ASTM Type I portland cement (290 kg/m³), washed sea sand available from the site, and gravel from the Nakagawa River. The water-cement ratio (by weight) was 0.48 and a water-reducing admixture was added to the mix. The performance curves for this data have been rated Quality Level B.

Metallic Reinforcements

Properties for uncoated, deformed, and uncoated, plain carbon steel reinforcing bar material conforming to ASTM A 615 Grades 40, 60 and 75 requirements (Ref. 3.14) are contained in Chapter 2 (Vols. 1–3) of the handbook and electronic data base file REBAR.DB. Information provided includes ambient and temperature-dependent engineering stress-strain performance curves for both plain and deformed Grade 40, 60 and 75 materials and S-N (fatigue) performance curves for the Grade 40 material. Since the data presented do not reflect test results for each possible bar size, the performance curves were rated Quality Level B.

Properties for uncoated, deformed, and uncoated, plain billet-steel reinforcing bar material conforming to ASTM A 15 requirements (Ref. 3.15) are contained in Chapter 2 (Vols. 1–3) of the handbook and electronic data base file REBAR.DB. The material is produced in three grades (Structural, Intermediate, and Hard) by either the open-hearth, basic oxygen, electric-furnace, or acid bessemer processes. Information provided includes ambient engineering stress-strain performance curves for each of the grades, and S-N (fatigue) performance curves and temperature-dependent stress-strain performance curves for the deformed, uncoated, Intermediate Grade material. Since the data presented do not reflect test results for each possible bar size, the performance curves were rated Quality Level B.

Prestressing Tendons

Properties for a carbon steel wire material conforming to ASTM A 421, Type BA wire (Ref. 3.16) having a diameter of 7.01 mm and used to post-tension concrete components are contained in Chapter 3 (Vols. 1–3) of the handbook and electronic data base file TENDON.DB. Information provided includes a temperature-dependent engineering stress–strain performance curve, a tensile yield strength vs temperature performance curve, an ultimate tensile strength vs temperature performance curve, and an ultimate tensile elongation vs temperature performance curve. Since these performance curves reflect test results from one reference source, each performance curve has been rated Quality Level B.

Structural Steels

Properties for a hot-rolled, carbon steel material conforming to ASTM A 36 requirements (Ref. 3.17) are contained in Chapter 4 (Vols. 1–3) of the handbook and electronic data base file STEEL.DB. Information provided includes a temperature-dependent engineering stress–strain performance curve, a tensile yield strength vs temperature performance curve, an ultimate tensile strength vs temperature performance curve, and an ultimate tensile elongation vs temperature performance curve. Since these performance curves reflect test results from only one reference source, each design curve has been rated Quality Level B.

Rubbers

Temperature-dependent hardness curves versus time for an ethylene propylene diene rubber conforming to ASTM D 1418 (Ref. 3.18) are contained in Chapter 5 (Vols. 1–3) of the handbook and electronic data base file RUBBER.DB. The test results were obtained from a study sponsored by the Swedish Standards Institute and Swedish Institute of Rubber Technology. The purpose of the study was to develop a data base on the effect of various liquids on the behavior of some common rubber polymers. Samples, 25-mm diameter by 2-mm thick, were supplied by several laboratories to a hardness specification of ~70 IRHD (International Rubber Hardness Degrees). The test specimens were immersed in water maintained at one of three temperatures, 23°, 70°, or 100°C. After a prescribed test period of up to 56 days, each specimen was removed from the water, weighed in air and water, and tested for hardness. Since the materials were supplied from several sources and the compositions were not reported, this data has been rated Quality Level C.

3.4.3 Material Behavior Modeling (Subtask S.2.3)

Modeling studies under this activity are designed to complement, as closely as possible, those conducted at the National Institute of Standards and Technology under another NRC program, “Performance Criteria for Concrete-Based Barriers for Low-Level Waste (LLW) Disposal Facilities.” Program personnel have participated in meetings of the American Concrete Institute Committee 365, “Service Life Predictions,” RILEM Committee TC-104, “Damage Classification of Concrete Structures,” and RILEM 100-TSL, “Techniques for Service Life Prediction.” Also, program personnel participated in the “Fifth International Conference on Durability of Building Materials and

Components," the CANMET/ACI "Second International Conference on Durability of Concrete," and met with researchers at Taywood Engineering Laboratories (England), the British Cement Association, AEA Technology (England), the Building Research Establishment (England), Danish Road Directorate, Atomic Energy of Canada, Ltd., Canada Centre for Mineral and Energy Technology, University of Toronto, Ontario Ministry of Transportation, National Research Council of Canada, Takenaka Technical Research Center, and Shimizu Corporation to discuss concrete modeling and service life predictions. References 3.9 and 3.19 - 3.23 summarize participation in the RILEM meetings, durability conferences, and discussions at foreign research facilities. During the previous reporting period a review and evaluation was conducted to identify and evaluate models and accelerated aging techniques and methodologies which can be used in making predictions of the remaining service life of concrete in nuclear power plants (Ref. 3.24). Recent activities have focused on development of prototypical data for input into the service life prediction models that have been identified for corrosion of reinforced concrete.

Taywood Engineering Ltd. has completed the collation of survey data and a durability assessment review of reinforced concrete structures contained as a part of several nuclear power stations located in the United Kingdom. The nuclear power station surveys were aimed primarily at providing data which can be used to predict the onset of corrosion of steel reinforcement due to carbonation or chloride ingress. Corrosion of steel reinforcement was recognized to be a two stage process: activation and propagation. The present state-of-the-art for modeling carbonation and chloride migration into concrete is reviewed along with factors which affect the time to corrosion activation, e.g. concentration level, environmental conditions, and diffusion coefficient. Survey data obtained from the Windscale Advanced Gas-Cooled Reactor, Hinkley Point "A," Bradwell and Wylfa nuclear power stations at selected locations include chloride content, carbonation depth, compressive strength, resistivity, moisture content, sorptivity, oxygen diffusion, and petrography. Environmental conditions (temperature, relative humidity, and rainfall) and survey results were used to predict the onset of steel reinforcement corrosion due to either carbonation or chloride ingress at selected locations in these stations. Predicted maximum age for steel reinforcement corrosion activation due to carbonation or chloride ingress ranged from 31 to >140 years and 43 to >140 years, respectively. Visual surveys at the stations indicate that for interior structures the steel reinforcement and concrete are sound with no corrosion, whereas the external concrete in most instances exhibited only a few localized areas of cracking and spalling. The exception to this is at Wylfa where the cooling plant exhibited severe rust staining and spalling due to chloride ion penetration. Other survey results were that the depths of carbonation reached up to 50 mm for internal concrete when it was uncoated, chloride ingress was generally low (<0.05% in surface 10 mm) except for Wylfa, and the concretes were free of alkali-silica reactivity and sulfate attack. More detailed information on these results is presented in Ref. 3.25.

3.5 PLANNED FY 1993 ACTIVITIES

Procurement and testing of aged concrete test specimens from prototypical structures will continue. Aged concrete test specimens from Palisades, Braidwood, Byron, Dresden, LaSalle, Quad Cities, and Zion power stations will be prepared, tested, and data bases prepared and entered into the SMIC. Reports prepared by Taywood Engineering Ltd. on testing of archived concrete samples, surveillance data from prestressed concrete pressure vessels, and collation of survey data from several of the British nuclear power stations will

be published. Criteria for use in development of customized data base management system software for the Structural Materials Information System will be developed.

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3.7 MILESTONE STATEMENT AND SCHEDULE

The statement and schedule for the milestones in Task S.2 are given in the following charts.

MILESTONE STATEMENT AND SCHEDULE

TASK: S.2 MATERIALS PROPERTY DATABASE

[illegible]

Fig. 3.1. Level 2 work breakdown structure for SAG Task S.2: Material Property Data Base.

Fig. 3.2. Listing of potential concrete degradation factors (Source: R. E. Weyers et al., "Concrete Deterioration Inspection System for Extending the Operating Life of Nuclear Power Plants Applied to Surry Unit One," Charles E. Via Department of Civil Engineering, Virginia Polytechnic Institute and State University, Blacksburg, February 1988).

Fig. 3.3. Level 3 work breakdown structure for SAG Task S.2: Material Property Data Base.

Fig. 3.4. Material code arrangement.

Table 3.1. Baseline information to classify
data base entry as "high quality."

1. Material Selection

Mixture proportions
Constituent composition/characteristics
Admixtures

2. Mixing/Placing/Curing

Mixing conditions
Schedule
Placing/casting conditions
Curing/environmental conditions

3. Properties

Unit weight
Air content/porosity/pore size distribution/permeability
Time of set
Bleeding
Compressive strength
Modulus of elasticity
Poisson's ratio
Tensile strength
Bond strength
Coefficient of thermal expansion
Coefficient of thermal diffusivity
Coefficient of conductivity

Table 3.2. Material code identification and description.

Portland Cement Concretes Handbook Chapter Index 01 - Electronic Data Base File CONCRETE.DB			
Group Index	Group Index Description Type of Concrete	Class Index	Class Index Description Type of Aggregate
A	Insulating	A	Stone
B	Structural Lightweight	B	Gravel
C	Normal-Weight	C	Manufactured or By-product
D	Heavyweight		
Metallic Reinforcements Handbook Chapter Index 02 - Electronic Data Base File REBAR.DB			
Group Index	Group Index Description Type of Reinforcement	Class Index	Class Index Description Characteristic Feature
A	Carbon Steel Bars	A	Uncoated without Deformations
B	Stainless Steel Bars	B	Coated without Deformations
C	Steel Wires	C	Uncoated with Deformations
D	Bar Mats/Wire Fabric	D	Coated with Deformations
Prestressing Tendons Handbook Chapter Index 03 - Electronic Data Base File TENDON.DB			
Group Index	Group Index Description Type of Tendon	Class Index	Class Index Description Characteristic Feature
A	Carbon Steel Bars	A	Materials without Deformations
B	Carbon Steel Wires	B	Materials with Deformations
C	Strand		
D	Nonmetallic Materials		
Structural Steels Handbook Chapter Index 04 - Electronic Data Base File STEEL.DB			
Group Index	Group Index Description Type of Reinforcement	Class Index	Class Index Description Characteristic Feature
A	Carbon Steels	A	Hot- or Cold-Rolled Steels
B	Stainless Steels	B	Bolting Materials
		C	Special Materials

Table 3.3. Property code range description.

Property Code Ranges	Property Code Range Description
1000-1999	General Information
2000-2999	Constituent Material and Plastic Concrete Properties
3000-3999	Mechanical Properties
4000-4999	Thermal, Physical and Other Properties
5000-9999	Available for Data Base Expansion

Table 3.4. Quality level definitions.

Quality Levels and Corresponding Term Definitions		
Quality Level	Quality Level Term Description	Relative Quality Level Rating
A	Recommended Property	Highest ↓ Lowest
B	Selected Property	
C	Typical Property	
D	Provisional Property	
E	Interim Property	

Table 3.5. Requirements for evaluating the quality of data and values.

Requirements for Determining the Quality of Data and Values	
Requirement Number	Requirement Description
1	Completeness of the material description
2	Stability of the material
3	Type of input from the references or sources [actual experimental observations (data) or results of previous analyses (values)]
4	Completeness of the data or values search
5	Completeness of the resources (completeness of the consideration given to all available data or values)
6	Quality of the references or sources
7	Availability of the data or values (completeness of data or values field coverage)
8	Consistency of the data or values with respect to related properties
9	Precision or scatter of the data or values
10	Uncertainty of the data or values (systematic error or bias)
11	Method used to determine the reported property (averaging, curve fitting, synthesization, derivation or extrapolation of the source data or values)

Table 3.6 Organizations providing material property data.

Organization	Result
a. U.S. Research Establishments	
Bradley University	Report on hydraulic conductivity of cement pastes
Bureau of Reclamation	Reports containing baseline information and 10-year property data on dams
Concrete Reinforcing Steel Institute	Information on properties of concrete reinforcing materials
Construction Technology Laboratories	Reports on long-term properties of concrete and fatigue of rebars
Corps of Engineers - Vicksburg, Miss.	Information on maintenance and repair materials, and petrographic and physical examination of radioactive concrete
CORRPRO Companies, Inc.	Information on chloride- and non-chloride induced corrosion of reinforced concrete
Howard University	Information on damage classification of concrete structures
Idaho National Engineering Laboratory	Report on service life modeling of reinforced concrete
Materials Research Laboratories (Pennsylvania State University)	Information on cementitious materials for waste fixation, and service-life prediction
Michigan State University	Reports on concrete durability and repair
Multiple Dynamics Corp.	Reports on durability of concrete materials and structures in nuclear power plants
NSF Center for Science and Technology of Advanced Cement-Based Materials	Information on high performance concretes and concrete fracture
National Institute of Standards and Technology	Reports on modeling to predict service life of concrete components
Prestressed Concrete Institute	Information on properties of concrete prestressing materials
Singleton Laboratories	Reports on properties of concrete at elevated temperature
Strategic Highway Research Program	Information on multiyear concrete research program addressing alkali-silica reactions, durability, inspection and repair
University of Notre Dame	Reports on damage and damage classification of concrete due to chemical processes
University of Wisconsin	Data on concrete properties to 50 years age
Virginia Power Co.	Information on inspection and aging of nuclear power plant concrete structures
Wiss, Janney, Elstner Assoc.	Information on petrographic examinations, mix design development, and mechanical testing of concrete materials for nuclear power plants

Table 3.6 (cont.) Organizations providing material property data.

Organization	Result
b. Foreign Research Establishments	
AEA Technology (England)	Reports on long-term durability of concrete, corrosion of steel in concrete, and non-destructive testing techniques
Basler & Hofmann (Switzerland)	Report on repair techniques; information on concrete structures in nuclear power plants
British Cement Assoc.	Information on European research related to service-life prediction and damage classification of concrete structures; reports on alkali-silica reactions
Building Research Establishment (England)	Information and reports on alkali-silica reactions, structural assessments and long-term performance of concrete
Canada Center for Mineral and Energy Technology CEC (Luxembourg)	Reports on long-term durability and strength development of concrete Report on historical examination of concrete; listing of publications; annual report on R&D program for decommissioning nuclear installations
Central Electricity Generating Board	Report on elevated temperature effects on concrete
Central Research Institute of Electric Power Industry (CRIEPI)	Reports on long-term and elevated temperature behavior of concretes associated with design of high-level radioactive waste facilities
Darmstadt Technische Hochschule	Reports on concrete properties, nondestructive testing and structural behavior
Deutscher Ausschuss für Stahlbeton Electricite de France	Report on 30-year properties of concrete Reports and information on containment design and long-term performance of nuclear power plant concrete structures
ENEA (Italy)	Information on elevated temperature effects on concrete properties
German Commission on Reinforced Concrete	Reports on water content and porosity of concrete at high temperature
Institut für Massivbau und Baustofftechnologie	Reports on creep and repair of concrete
Industria Italiana del Cemento	Reports on reliability evaluation of existing structures
Institute for Fire Safety and Design (IDEON) (Sweden)	Reports on elevated temperature behavior of concrete and reinforcing materials
Japan Atomic Energy Research Institute (JAERI)	Data from testing of concrete cores obtained from Japan power demonstration reactor

Table 3.6 (cont.) Organizations providing material property data.

Organization	Result
b. Foreign Research Establishments (cont.)	
Japan Power Engineering and Inspection Corporation (JAPEIC)	Information on Japanese nuclear power plant life extension program and how it relates to the concrete materials
Kajima Technical Research Institute (Japan)	Reports on properties of concrete at elevated temperature and concrete removed from Japan power demonstration reactor; information on rehabilitation of concrete structures
Lehrstuhl für Baustoffkunde und Werkstoffprüfung der TU München	Information on long-term, elevated temperature and multiaxial performance of concrete; information on damage classification
National Swedish Council for Building Research	Information on performance of building materials and structures
Nuclear Electric (England)	Information on long-term performance of prestressed concrete pressure vessels
Obayashi Corporation (Japan)	Reports on properties of concrete at elevated temperature; information on corrosion of concrete reinforcing materials, cementitious materials use in radioactive waste depositories, and boron-containing concretes for neutron absorption
Ontario Hydro	Reports on service performance of nuclear containment concrete and aging of materials and components
Ontario Ministry of Transportation	Reports on cathodic protection of reinforced concrete structures
Universität Karlsruhe	Reports on effects of elevated temperature and irradiation on concrete and its constituents
University of Toronto	Reports on concrete permeability and migration of moisture and their effects on aging of concrete containments

Table 3.7 Organizations contacted to provide prototypical data
or concrete samples for testing

Organization	Candidate facility	Status
Argonne National Laboratory	EBR-II	Samples provided/tested from air cell foundation, shield wall, and containment building
	EBWR and CP-5	Under consideration
Battelle Pacific Northwest Laboratory	Shippingport Power Station	Samples provided/tested from reactor enclosure refueling ring and reactor enclosure basement
Commonwealth Edison	Braidwood	Samples provided from each plant, samples prepared for testing
	Bryon	
	Dresden	
	LaSalle	
	Quad Cities	
Construction Technology Laboratories	Zion	Reports completed providing concrete data for ages to 42 y
	Long-term study	
Consumers Power Co.	Midland Palisades	Samples provided from Midland and Palisades plants, Midland samples tested
Dairyland Power Cooperative	LaCrosse	Terminated, not cost effective
Detroit Edison	Fermi	Terminated, utility not interested
General Electric	Vallicetos	Samples of normal-weight and heavyweight concretes tested
Northern States Power	Pathfinder	Terminated, original records unavailable
Taywood Engineering, Ltd.	Several British nuclear power stations	Specimens tested from Wylfa, Heysham I, Hartlepool, Heysham II, Torness, and Sizewell B
Union Carbide	CintiChem	Terminated, original records unavailable

Table 3.8. Portland cement concretes in the Structural Materials Information Center.

Material Code	Common Name	Material Description	Property Code	Property Code Description
01CA001	Series B, Lannon Dolomite University of Wisconsin	Portland Cement Concrete Normal-Weight Crushed Stone Aggregate	3621	Ultimate Compressive Strength versus Time
01CA002	Series B, Red Granite University of Wisconsin	Portland Cement Concrete Normal-Weight Crushed Stone Aggregate	3621	Ultimate Compressive Strength versus Time
01CA003	Nasser and Chakraborty, 1981 University of Saskatchewan	Portland Cement Concrete Normal-Weight Crushed Stone Aggregate	3612	Modulus of Elasticity versus Time (Temperature Dependent)
			3622	Ultimate Compressive Strength versus Time (Temperature Dependent)
01CA004	CANMET, Limestone Mix 1 Canada Centre for Mineral and Energy Technology	Portland Cement Concrete Normal-Weight Crushed Stone Aggregate	3622	Ultimate Compressive Strength versus Time (Temperature Dependent)
01CA005	CANMET, Limestone Mix 2 Canada Centre for Mineral and Energy Technology	Portland Cement Concrete Normal-Weight Crushed Stone Aggregate	3622	Ultimate Compressive Strength versus Time (Temperature Dependent)
01CA006	CANMET, Limestone Mix 3 Canada Centre for Mineral and Energy Technology	Portland Cement Concrete Normal-Weight Crushed Stone Aggregate	3622	Ultimate Compressive Strength versus Time (Temperature Dependent)
01CA007	CANMET, Dolostone Mix 4 Canada Centre for Mineral and Energy Technology	Portland Cement Concrete Normal-Weight Crushed Stone Aggregate	3622	Ultimate Compressive Strength versus Time (Temperature Dependent)
01CA008	CANMET, Dolostone Mix 5 Canada Centre for Mineral and Energy Technology	Portland Cement Concrete Normal-Weight Crushed Stone Aggregate	3622	Ultimate Compressive Strength versus Time (Temperature Dependent)
01CA009	CANMET, Dolostone Mix 6 Canada Centre for Mineral and Energy Technology	Portland Cement Concrete Normal-Weight Crushed Stone Aggregate	3622	Ultimate Compressive Strength versus Time (Temperature Dependent)
01CA010	PCA - Series 356, Mix XL1 Portland Cement Association	Portland Cement Concrete Normal-Weight Crushed Stone Aggregate	3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time

Table 3.8. (cont.) Portland cement concretes in the Structural Materials Information Center.

Material Code	Common Name	Material Description	Property Code	Property Code Description
01CA011	PCA - Series 356, Mix XL2 Portland Cement Association	Portland Cement Concrete Normal-Weight Crushed Stone Aggregate	3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CA012	PCA - Series 356, Mix XL3 Portland Cement Association	Portland Cement Concrete Normal-Weight Crushed Stone Aggregate	3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CA013	PCA - Series 356, Mix XhL1 Portland Cement Association	Portland Cement Concrete Normal-Weight Crushed Stone Aggregate	3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CA014	PCA - Series 356, Mix XhL2 Portland Cement Association	Portland Cement Concrete Normal-Weight Crushed Stone Aggregate	3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CA015	PCA - Series 356, Mix XhL3 Portland Cement Association	Portland Cement Concrete Normal-Weight Crushed Stone Aggregate	3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CA016	Midland, Mix E-2 Midland Nuclear Power Plant Midland, Michigan	Portland Cement Concrete Normal-Weight Crushed Stone Aggregate	3621	Ultimate Compressive Strength versus Time
01CA017	Midland, Mix C-1 Midland Nuclear Power Plant Midland, Michigan	Portland Cement Concrete Normal-Weight Crushed Stone Aggregate	3621	Ultimate Compressive Strength versus Time
01CB001	ANL-West Fuel Cycle Facility Argonne National Laboratory - West, Idaho Falls, Idaho	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
01CB002	Series B, Janesville, 0.51 University of Wisconsin	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time

Table 3.8. (cont.) Portland cement concretes in the Structural Materials Information Center.

Material Code	Common Name	Material Description	Property Code	Property Code Description
01CB003	Series B, Janesville, 0.51 University of Wisconsin	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
01CB004	Series B, Janesville, 0.67 University of Wisconsin	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
01CB005	Series B, Janesville, 0.69 University of Wisconsin	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
01CB006	EBR-II, Lift 5, Subbasement Argonne National Laboratory - West, Idaho Falls, Idaho	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
01CB007	EBR-II, Biological Shield Argonne National Laboratory - West, Idaho Falls, Idaho	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
01CB008	Walz, Series A Stuttgart, Germany	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
01CB009	Walz, Series B Stuttgart, Germany	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
01CB010	Walz, Series C Stuttgart, Germany	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
01CB011	Walz, Series D Stuttgart, Germany	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
01CB012	Walz, Series E Stuttgart, Germany	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
01CB013	Walz, Series F Stuttgart, Germany	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
01CB014	Walz, Series G Stuttgart, Germany	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time

Table 3.8. (cont.) Portland cement concretes in the Structural Materials Information Center.

Material Code	Common Name	Material Description	Property Code	Property Code Description
01CB015	Walz, Series H Stuttgart, Germany	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
01CB016	Kondo, Control University of Kyoto, Japan	Portland Cement Concrete Normal-Weight Gravel Aggregate	3651	Bond Stress versus Slip
01CB017	Kondo, 10% Fly Ash University of Kyoto, Japan	Portland Cement Concrete Normal-Weight Gravel Aggregate	3651	Bond Stress versus Slip
01CB018	Kondo, 20% Fly Ash University of Kyoto, Japan	Portland Cement Concrete Normal-Weight Gravel Aggregate	3651	Bond Stress versus Slip
01CB019	Kondo, 30% Fly Ash University of Kyoto, Japan	Portland Cement Concrete Normal-Weight Gravel Aggregate	3651	Bond Stress versus Slip
01CB020	Kondo, 0.25% Calcium Chloride University of Kyoto, Japan	Portland Cement Concrete Normal-Weight Gravel Aggregate	3651	Bond Stress versus Slip
01CB021	Kondo, 1.0% Calcium Chloride University of Kyoto, Japan	Portland Cement Concrete Normal-Weight Gravel Aggregate	3651	Bond Stress versus Slip
01CB022	Kondo, Calcium Lignosulfonate University of Kyoto, Japan	Portland Cement Concrete Normal-Weight Gravel Aggregate	3651	Bond Stress versus Slip
01CB023	Kondo, 0.25% and 0.25% University of Kyoto, Japan	Portland Cement Concrete Normal-Weight Gravel Aggregate	3651	Bond Stress versus Slip
01CB024	Kondo, 1.0% and 0.25% University of Kyoto, Japan	Portland Cement Concrete Normal-Weight Gravel Aggregate	3651	Bond Stress versus Slip
01CB025	SABNGS No.2, 5.4 Bag Plain Ontario Hydro Toronto, Ontario, Canada	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
01CB026	SABNGS No.2, 6 Bag Plain Ontario Hydro Toronto, Ontario, Canada	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time

Table 3.8. (cont.) Portland cement concretes in the Structural Materials Information Center.

Material Code	Common Name	Material Description	Property Code	Property Code Description
01CB027	SABNGS No.2, 7 Bag Plain Ontario Hydro Toronto, Ontario, Canada	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
01CB028	SABNGS No.2, 5.4 Air Ontario Hydro Toronto, Ontario, Canada	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
01CB029	SABNGS No.2, 6 Air Ontario Hydro Toronto, Ontario, Canada	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
01CB030	SABNGS No.2, 7 Air Ontario Hydro Toronto, Ontario, Canada	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
01CB031	SABNGS No.2, 5.4 Fly Ash Ontario Hydro Toronto, Ontario, Canada	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
01CB032	SABNGS No.2, 6 Fly Ash Ontario Hydro Toronto, Ontario, Canada	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
01CB033	SABNGS No.2, 7 Fly Ash Ontario Hydro Toronto, Ontario, Canada	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
01CB034	Stewartville G.S., Core Ontario Hydro Toronto, Ontario, Canada	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
01CB035	Stewartville G.S., Face Ontario Hydro Toronto, Ontario, Canada	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
01CB036	PCA - Series 356, Mix AV1 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Dynamic Modulus of Elasticity versus Time
			3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time

Table 3.8. (cont.) Portland cement concretes in the Structural Materials Information Center.

Material Code	Common Name	Material Description	Property Code	Property Code Description
01CB037	PCA - Series 356, Mix AV2 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Dynamic Modulus of Elasticity versus Time
			3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB038	PCA - Series 356, Mix AV3 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Dynamic Modulus of Elasticity versus Time
			3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB039	PCA - Series 356, Mix DV1 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Dynamic Modulus of Elasticity versus Time
			3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB040	PCA - Series 356, Mix DV2 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Dynamic Modulus of Elasticity versus Time
			3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB041	PCA - Series 356, Mix DV3 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Dynamic Modulus of Elasticity versus Time
			3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time

Table 3.8. (cont.) Portland cement concretes in the Structural Materials Information Center.

Material Code	Common Name	Material Description	Property Code	Property Code Description
01CB042	PCA - Series 356, Mix EV1 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Dynamic Modulus of Elasticity versus Time
			3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB043	PCA - Series 356, Mix EV2 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Dynamic Modulus of Elasticity versus Time
			3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB044	PCA - Series 356, Mix EV3 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Dynamic Modulus of Elasticity versus Time
			3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB045	PCA - Series 356, Mix XV1 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Dynamic Modulus of Elasticity versus Time
			3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB046	PCA - Series 356, Mix XV2 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Dynamic Modulus of Elasticity versus Time
			3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time

Table 3.8. (cont.) Portland cement concretes in the Structural Materials Information Center.

Material Code	Common Name	Material Description	Property Code	Property Code Description
01CB047	PCA - Series 356, Mix XV3 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Dynamic Modulus of Elasticity versus Time
			3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB048	PCA - Series 356, Mix XW1 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB049	PCA - Series 356, Mix XW2 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB050	PCA - Series 356, Mix XW3 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB051	PCA - Series 356, Mix AhV1 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Dynamic Modulus of Elasticity versus Time
			3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB052	PCA - Series 356, Mix AhV2 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Dynamic Modulus of Elasticity versus Time
			3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time

Table 3.8. (cont.) Portland cement concretes in the Structural Materials Information Center.

Material Code	Common Name	Material Description	Property Code	Property Code Description
01CB053	PCA - Series 356, Mix AhV3 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Dynamic Modulus of Elasticity versus Time
			3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB054	PCA - Series 356, Mix DhV1 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Ultimate Compressive Strength versus Time
			3621	Flexural Strength versus Time
			3631	Ultimate Compressive Strength versus Time
01CB055	PCA - Series 356, Mix DhV2 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Flexural Strength versus Time
			3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB056	PCA - Series 356, Mix DhV3 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Dynamic Modulus of Elasticity versus Time
			3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB063	PCA - Series 356, Mix XhW1 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB064	PCA - Series 356, Mix XhW2 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time

Table 3.8. (cont.) Portland cement concretes in the Structural Materials Information Center.

Material Code	Common Name	Material Description	Property Code	Property Code Description
01CB065	PCA - Series 356, Mix XhW3 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB068	PCA - Series 374, Mix 11T1 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Dynamic Modulus of Elasticity versus Time
			3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB069	PCA - Series 374, Mix 11T2 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Dynamic Modulus of Elasticity versus Time
			3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB093	PCA - Series 374, Mix 19B1 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Dynamic Modulus of Elasticity versus Time
			3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB094	PCA - Series 374, Mix 19B2 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Dynamic Modulus of Elasticity versus Time
			3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time

Table 3.8. (cont.) Portland cement concretes in the Structural Materials Information Center.

Material Code	Common Name	Material Description	Property Code	Property Code Description
01CB097	PCA - Series 374, Mix 21A Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Dynamic Modulus of Elasticity versus Time
			3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB098	PCA - Series 374, Mix 21B Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Dynamic Modulus of Elasticity versus Time
			3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB099	PCA - Series 374, Mix 21T1 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Dynamic Modulus of Elasticity versus Time
			3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB100	PCA - Series 374, Mix 21T2 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Dynamic Modulus of Elasticity versus Time
			3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB110	PCA - Series 374, Mix 31A Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Dynamic Modulus of Elasticity versus Time
			3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time

Table 3.8. (cont.) Portland cement concretes in the Structural Materials Information Center.

Material Code	Common Name	Material Description	Property Code	Property Code Description
01CB111	PCA - Series 374, Mix 31B Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Dynamic Modulus of Elasticity versus Time
			3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB124	PCA - Series 374, Mix 43A3 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Dynamic Modulus of Elasticity versus Time
			3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB125	PCA - Series 374, Mix 51A Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Dynamic Modulus of Elasticity versus Time
			3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB126	PCA - Series 374, Mix 51B Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3613	Dynamic Modulus of Elasticity versus Time
			3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB127	PCA - Series 436, Mix 1 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB128	PCA - Series 436, Mix 2 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time

Table 3.8. (cont.) Portland cement concretes in the Structural Materials Information Center.

Material Code	Common Name	Material Description	Property Code	Property Code Description
01CB129	PCA - Series 436, Mix 3 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB130	PCA - Series 436, Mix 4 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB131	PCA - Series 436, Mix 5 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB132	PCA - Series 436, Mix 6 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB133	PCA - Series 436, Mix 7 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB134	PCA - Series 436, Mix 8 Portland Cement Association	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
			3631	Flexural Strength versus Time
01CB151	GETR, Mix 57-1842 General Electric Test Reactor Vallecitos, California	Portland Cement Concrete Normal-Weight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time

Table 3.8. (cont.) Portland cement concretes in the Structural Materials Information Center.

Material Code	Common Name	Material Description	Property Code	Property Code Description
01CB152	JPDR, Biological Shield Walls Japan Atomic Energy Research Institute (JAERI), Tokai Research Establishment, Japan	Portland Cement Concrete Normal-Weight Gravel Aggregate	3615	Molulus of Elasticity versus Fast Neutron Fluence
			3623	Ultimate Compressive Strength versus Fast Neutron Fluence
			3633	Splitting Tensile Strength versus Fast Neutron Fluence
01DA001	ANL-West Argon Cell Wall Argonne National Laboratory - West, Idaho Falls, Idaho	Portland Cement Concrete Heavyweight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
01DA002	HD (High-Density) Shielding Wall Concrete	Portland Cement Concrete Heavyweight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time
01DA003	GETR, Mix 58-1970 General Electric Test Reactor Vallecitos, California	Portland Cement Concrete Heavyweight Gravel Aggregate	3621	Ultimate Compressive Strength versus Time

Table 3.9. Metallic reinforcements in the Structural Materials Information Center.

Material Code	Common Name	Material Description	Property Code	Property Code Description
02AA001	ASTM A 15, Structural Plain	Metallic Reinforcement Carbon Steel Reinforcing Bar Uncoated without Deformations	3701	Engineering Stress-Strain (Ambient Conditions)
02AA002	ASTM A 15, Intermediate Plain	Metallic Reinforcement Carbon Steel Reinforcing Bar Uncoated without Deformations	3701	Engineering Stress-Strain (Ambient Conditions)
02AA003	ASTM A 15, Hard Plain	Metallic Reinforcement Carbon Steel Reinforcing Bar Uncoated without Deformations	3701	Engineering Stress-Strain (Ambient Conditions)
02AA004	ASTM A 615, Gr. 40, Plain	Metallic Reinforcement Carbon Steel Reinforcing Bar Uncoated without Deformations	3701	Engineering Stress-Strain (Ambient Conditions)
02AA005	ASTM A 615, Gr. 60, Plain	Metallic Reinforcement Carbon Steel Reinforcing Bar Uncoated without Deformations	3701	Engineering Stress-Strain (Ambient Conditions)
02AA006	ASTM A 615, Gr. 75, Plain	Metallic Reinforcement Carbon Steel Reinforcing Bar Uncoated without Deformations	3701	Engineering Stress-Strain (Ambient Conditions)
02AC001	ASTM A 15, Structural Deformed	Metallic Reinforcement Carbon Steel Reinforcing Bar Uncoated with Deformations	3701	Engineering Stress-Strain (Temperature Dependent)
02AC002	ASTM A 15, Intermediate Deformed	Metallic Reinforcement Carbon Steel Reinforcing Bar Uncoated with Deformations	3701	Tensile Yield Strength versus Temperature
			3731	S-N Diagram

Table 3.9. (cont.) Metallic reinforcements in the Structural Materials Information Center.

Material Code	Common Name	Material Description	Property Code	Property Code Description
02AC003	ASTM A 15, Hard Deformed	Metallic Reinforcement Carbon Steel Reinforcing Bar Uncoated with Deformations	3701	Engineering Stress-Strain (Ambient Conditions)
02AC004	ASTM A 615, Gr. 40, Deformed	Metallic Reinforcement Carbon Steel Reinforcing Bar Uncoated with Deformations	3701	Engineering Stress-Strain (Ambient Conditions)
			3731	S-N Diagram
02AC005	ASTM A 615, Gr. 60, Deformed	Metallic Reinforcement Carbon Steel Reinforcing Bar Uncoated with Deformations	3701	Engineering Stress-Strain (Ambient Conditions)
			3702	Engineering Stress-Strain (Temperature Dependent)
			3711	Tensile Yield Strength versus Temperature
			3712	Ultimate Tensile Strength versus Temperature
			3721	Ultimate Tensile Elongation versus Temperature
			3731	S-N Diagram
02AC006	ASTM A 615, Gr. 75, Deformed	Metallic Reinforcement Carbon Steel Reinforcing Bar Uncoated with Deformations	3701	Engineering Stress-Strain (Ambient Conditions)
			3731	S-N Diagram

Table 3.9. (cont.) Other materials in the Structural Materials Information Center.

Material Code	Common Name	Material Description	Property Code	Property Code Description
04AA002	ASTM A 36	Structural Steel Carbon Steel Hot-Rolled Steel	3702	Engineering Stress-Strain (Temperature Dependent)
			3711	Tensile Yield Strength versus Temperature
			3712	Ultimate Tensile Strength versus Temperature
			3721	Ultimate Tensile Strength versus Temperature
05AA001	SIS, EPDM, 75 IRHD	Rubber ASTM D 1418, Class M EPDM	3662	Hardness versus Time (Temperature Dependent)

Table 3.10. Prestressing tendons in the Structural Materials Information Center.

Material Code	Common Name	Material Description	Property Code	Property Code Description
03BA003	ASTM A 421, Type BA, 7.01 mm	Prestressing Tendon Carbon Steel Wire Without Deformations	3702	Engineering Stress-Strain (Temperature Dependent)
			3711	Tensile Yield Strength versus Temperature
			3712	Ultimate Tensile Strength versus Temperature
			3721	Ultimate Tensile Strength versus Temperature

Table 3.11. Structural steels in the Structural Materials Information Center.

Material Code	Common Name	Material Description	Property Code	Property Code Description
04AA002	ASTM A 36	Structural Steel Carbon Steel Hot-Rolled Steel	3702	Engineering Stress-Strain (Temperature Dependent)
			3711	Tensile Yield Strength versus Temperature
			3712	Ultimate Tensile Strength versus Temperature
			3721	Ultimate Tensile Strength versus Temperature

Table 3.12. Rubbers in the Structural Materials Information Center.

Material Code	Common Name	Material Description	Property Code	Property Code Description
05AA001	SIS, EPDM, 75 IRHD	Rubber ASTM D 1418, Class M EPDM	3662	Hardness versus Time (Temperature Dependent)

4.0 TASK S.3 – STRUCTURAL COMPONENT ASSESSMENT/REPAIR TECHNOLOGY

4.1 OBJECTIVE

The overall objectives of this task are to (1) develop a systematic methodology which can be used to make a quantitative assessment of the presence, magnitude, and significance of any environmental stressors or aging factors which could impact the durability of safety-related concrete components in nuclear power plants (NPPs) and (2) provide recommended in-service inspection or sampling procedures which can be utilized to develop the data required both for evaluating the current structural condition as well as trending the performance of these components for use in continued service assessments. Associated activities in meeting the objectives of this task include the identification and evaluation of techniques for mitigation of any environmental stressors or aging factors which may act on critical concrete components, and an assessment of techniques for repair, replacement, or retrofitting of concrete components which have experienced an unacceptable degree of deterioration. Results of this task will have a major impact on NPP continued service considerations (see Chapter 5 – Task S.4) since residual life determination cannot be made unless all degradation mechanisms are evaluated and environmental factors quantified and each structure is examined and evaluated in detail. Figure 4.1 presents the work breakdown structure for the structural component assessment/repair technology task.

4.2 BACKGROUND

4.2.1 Evaluation of Concrete Material Systems

Since the ability of a concrete component to meet its functional and performance requirements over an extended period of time is dependent on the durability of its constituents, techniques for the detection of concrete component degradation should address evaluation of the concrete, mild steel reinforcing, prestressing system, and anchorage embedments.

Concrete cracking, voids, and delaminations can be detected by visual inspections, nondestructive testing (ultrasonic and stress wave, acoustic impact, radiography, penetrating radar, thermal mapping), and core examination. In-situ concrete strength determinations are through either direct (core tests) or indirect techniques (surface hardness, rebound, penetration, pullout resistance, break-off resistance, and ultrasonic pulse velocity). The primary distress to which mild steel reinforcement could be subjected would be corrosive attack. Techniques available for corrosion monitoring and inspection of steel in concrete include visual, mechanical and ultrasonic tests, core sampling in conjunction with chemical and physical tests, potential and thermal mapping, and rate of corrosion probes. The condition and functional capability of unbonded post-tensioning systems is assessed through selection of a random, representative sample of tendons, examination of the anchorage assembly hardware of the selected tendons, determination of the stress level in each sample tendon, examination of previously stressed wires or strands from one tendon of each type in the structure, and an analysis of a grease sample from each tendon in the surveillance. The present basis for conducting tendon inspections is

presented in RG 1.35, “Inservice Inspections of Ungerouted Tendons in Prestressed Concrete Containments (Rev. 3)” and companion RG 1.35.1, “Determining Prestressing Forces for Inspection of Prestressed Concrete Containments.” Failure of an embedment will generally occur as a result of either improper installation or deterioration of the concrete within which it is embedded. A combination of visual examinations and mechanical tests is used to evaluate the general condition of an embedment.

Table 4.1 (Ref. 4.1) summarizes primary and secondary nondestructive evaluation techniques for inspection of concrete components. More detailed information on many of these techniques can be obtained from Ref. 4.2. Quantitative interpretation of the results obtained from many of the nondestructive evaluation methods can be difficult, however, due to the requirement for correlation curves. Also, many of the methods only make surface determinations of concrete properties which can be quite different from internal properties, particularly where a component may be several meters thick. In addition, none of the techniques provide rate effect data which can be used for continued service considerations or residual life predictions.

4.2.2 Inservice Inspection of NPPs

Inservice inspection requirements are imposed on nuclear plants through documents such as the following: 10CFR50, NRC Regulatory Guides, Plant Technical Specifications, Inspection and Enforcement Bulletins, NRC letters, and American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (Ref. 4.3). However, because each nuclear plant has its unique construction permit issue and operating license issue dates, each plant could potentially have a different set of minimum inservice inspection requirements. Therefore, to simplify continued service evaluations, it would be advantageous to have a standardized inservice inspection program that could be used to identify and also to quantify any deteriorating influences. The availability of such a standardized inspection program would also help ensure that a uniform set of evaluation criteria are applied to each plant.

Limited information on criteria, inspection, and testing requirements for development of such a procedure are available in the form of documents published by the American Concrete Institute, for example, *Guide for Making a Condition Survey of Concrete in Service* (Ref. 4.4), *Strength Evaluation of Existing Concrete in Service* (Ref. 4.5), *Practices for Evaluation of Concrete in Existing Massive Structures for Service Conditions* (Ref. 4.6), and *Guide for Concrete Inspection* (Ref. 4.7). Additional information is also contained in Refs. 4.8–4.10. However, the application of requirements presented in these documents to nuclear safety-related concrete structures requires evaluation with respect to items such as accessibility, service history, functional requirements, construction materials, etc.

4.2.3 Remedial Measures

Objectives of remedial work are to restore the component’s structural integrity, arrest the mechanism producing distress, and ensure, as far as possible, that the cause of distress will not recur. Basic components of a program to meet these objectives include diagnosis (damage evaluation), prognosis (can repair be made and is it economical), scheduling (priority assignments), method selection (depends on nature of distress, adaptability of proposed method, environment, and costs), preparation (function of extent and type of distress), and application (Ref. 4.11).

Typical types of distress that occur in LWR concrete facilities include cracking, spalling or delamination, nonvisible voids, and fracturing or shattering. Although a wide variety of materials are available for the repair or maintenance of concrete exhibiting distress, they generally include one or more of the following materials: epoxy resins, shotcrete, preplaced aggregate concrete, epoxy ceramic foams, replacement mortar or concrete, wedge anchors and additional reinforcement, and miscellaneous sealant materials (Ref. 4.12). Selection of the technique for repair of a concrete structure depends to a large degree on the size, depth, and area of repair required. Existing elements can also become inadequate due to either a change in performance requirements or occurrence of an overload condition. Under these conditions retrofitting may be required to reestablish serviceability. Retrofitting can be accomplished by either strengthening of existing elements, replacement, addition of new force-resisting elements, a combination of element strengthening and addition, or use of supplemental connecting devices (Ref. 4.12). Reference 4.11 notes that a satisfactory repair meeting requirements for strength, durability, appearance, and economy can be made if the cause of the distress is eliminated, the area is prepared by removal of degraded materials, and the proper repair technique selected and correctly implemented.

4.3 PLAN OF ACTION

Assessment of the functional and performance characteristics of concrete components is an important consideration in continuing the service life of nuclear facilities. Given the complex nature of the various environmental stressors and aging factors that potentially can exert deteriorating influences on the concrete components, a systems approach is probably best in addressing an evaluation of a structure for continued service. Basic components of such an approach would encompass the development of (1) classification scheme for structures, elements, and deterioration causes and effects; (2) a methodology for conducting a quantitative assessment of the presence of active deteriorating influences; and (3) remedial measure considerations to reestablish the capability of degraded structures or components to meet potential future requirements, such as a loss-of-coolant accident (LOCA). The Level 3 work breakdown structure for this task is presented in Fig. 4.2.

4.3.1 LWR Critical Concrete Component Classification

Basic activities under this subtask include the identification of safety-related concrete structural components in LWR plants as well as aging factors and environmental stressors which can impact the ability of these components to meet their functional and performance requirements. Critical components are described and a listing of their functional and performance requirements provided. A ranking scheme has been developed and each component classified in terms of its relative importance to safety. Critical elements are then divided into subelements and their importance to the integrity of the component assessed, for instance, rated. The expected type and range of degradation factors (aging, environmental stressors, operation, testing, etc.) that can impact each subelement are identified. Possible degradation mechanisms (for instance, how degradation manifests itself) for each subelement are identified and a rating system developed which can be used to indicate the significance of each mechanism. Critical performance characteristics and properties which can serve as indicators of the severity of each type of degradation are defined. The significance of each degradation factor has been quantified through

development of a grading system in which each element in the system is differentiated according to intensity or degree of deterioration.

4.3.2 NDE/Sampling Inspection Technology

4.3.2.1 Inservice Inspection Techniques

Detection of age- or environmental stressor-related degradation, as well as its magnitude and rate of occurrence, is a key factor in maintaining the readiness of safety-related concrete components to continue their functions in the unlikely event that a condition, such as a LOCA, would occur. In order to simplify continued service evaluations of nuclear safety-related concrete components, it would be advantageous to have a standardized inservice inspection program which would provide data that could be used to identify and quantify deteriorating influences as well as provide rate of change information on these influences, for example, degradation vs time relationships such as shown in Fig. 4.3.

Basic activities under this subtask are related to an evaluation of nondestructive and sampling procedures which are available for performing inservice inspections of the critical concrete components (see Sect. 4.3.1). Also, recommended criteria will be provided for use in inspection of these components to assess their current condition as well as to develop trending information for use in continued service assessments. Existing destructive and nondestructive testing and inspection methods have been reviewed. These methods were evaluated to identify capabilities, accuracies, and limitations in the detection/assessment of aging factors and environmental stressors to which the critical concrete components in NPPs may be subjected. Included in the review are any potential new techniques which, although they may not be widely used at present, possess significant potential (for example, modal analysis/dynamic testing). Methods which can be utilized to evaluate the severity of each of the various degradation factors are identified and rated according to their effectiveness. Included in the rating are any uncertainties associated with each method as well as its effectiveness for performing volumetric examinations. Also included as a part of this review is an activity to provide correlation curves and other statistical data for the most common techniques used to assess in-situ concrete compressive strength.

4.3.2.2 Structural Integrity Assessments

Once the critical components, degradation factors which can act on these components, and techniques which can be utilized to evaluate the components have been identified, a methodology for conducting a structural assessment of the components needs to be developed as well as criteria for its application. A schematic of such a concrete component evaluation methodology is presented in Fig. 4.4. When developed and implemented, the concrete component evaluation methodology will provide the required data and procedures for performing a safety condition (reliability) assessment of the critical safety-related concrete components in NPPs.

Basic activities under this subtask are related to providing recommended criteria for inspection of critical concrete components to assess their current structural reliability as well as to develop data for use in assessments of future performance (see Chapter 5). Available methodologies (for instance, the application of inspection/evaluation techniques) which have been or are presently being utilized to perform a condition assessment of

concrete components/structural members have been reviewed. Included in the review are techniques which have been used in NPPs, what these techniques have revealed in the form of degradation, an assessment of their effectiveness, and if any of the techniques have been utilized for residual life determinations. Data requirements and recommended inspection intervals for use in developing the required data for use in continued service evaluations also will be established, for example, implementation of predictive monitoring techniques to better evaluate component performance. A methodology will be developed which can be utilized for performing an assessment of the structural reliability of safety-related concrete components in NPPs as well as developing the information required to trend concrete component performance for use in continued service evaluations. Periodic inspections performed in compliance with this methodology will help identify any potential problems before they become safety-significant and allow corrective actions to be taken at an early stage.

4.3.3 Remedial/Preventive Measures Considerations

The life of reinforced concrete components in NPPs is expected to be greater than any likely period for which the plant would operate (original design lifetime plus continued service lifetime), provided neither environmental factors, applied load, nor a combination of load and environmental factors compromise the structural integrity (Ref. 4.13). In fact, when concrete structures have been fabricated with close attention to the details related to the production of good concrete [material selection, production control, desirable properties, economy (Ref. 4.14)] the concrete should exhibit extended durability; however, where there has been a breakdown in any one of these details or if the component was subjected to an extreme environmental stressor or adverse aging factor, distress can occur.

Results of a literature review (Ref. 4.1), in which examples were identified where the pre- and post-repair performance of structural members were compared, indicates that remedial measures are capable of completely restoring a component's structural integrity. The effectiveness of the repair techniques in these studies was the direct consequence of careful consideration given to the selection and application of the particular repair technique under controlled (laboratory) conditions. Successful application of these techniques to concrete structures in NPPs requires development of criteria for their application as well as an assessment of their overall effectiveness under representative (field) conditions.

Basic activities under this subtask are related to an assessment of repair procedures for concrete material/structural systems and establishment of criteria for their utilization. Techniques which are available for the repair, replacement or retrofitting of degraded structural subelements will be reviewed and an assessment performed of their effectiveness. Methods available for evaluating the performance of repair materials as well as any potential impact of a repair on the inspection procedures will be addressed. Techniques which can be used to mitigate the effects of environmental stressors or aging factors will be identified. Recommended preventative measure procedures which can be used to effectively offset, counteract, or minimize any minor deterioration effects to prevent them from becoming significant will be established.

4.4 SUMMARY OF ACCOMPLISHMENTS

4.4.1 LWR Critical Concrete Component Classification (Subtask S.3.1)

Activities under this subtask area were essentially completed during the last reporting period when the report on a structural aging assessment methodology was published. The methodology developed provides a logical basis for identifying the critical concrete structural elements in a nuclear plant as well as the degradation factors which can potentially impact their performance. Because of the variability in likelihood of occurrence of degradation of concrete structures in U. S. light-water reactor plants due to design differences, material utilization, geographical location, etc., the grading system established for degradation factors is stated in terms of a possible range of values. The resulting degradation factor grading values for the individually evaluated subelement (between 1 and 10, with 10 being highest) are then combined into a single degradation factor significance value by summing the degradation factor grading values and dividing by the number of degradation factors, for instance,

$$DFS = (\sum_{i=1}^n DFG_i) / n, \quad (4.1)$$

where

DFS = degradation factor significance value, rounded to nearest integer,
 DFG = degradation factor grading value, and
 n = number of degradation factors, up to a total of three.

Determination of the relative ranking of Category I structures and their subelements is based on the weighted contributions of the (1) structural importance of subelements, (2) safety significance, (3) environmental exposure and (4) degradation factor significance as follows:

$$SR = w_1(I) + w_2(SS) + w_3(DEG), \quad (4.2)$$

where

SR = subelement rank,
 I = subelement importance,
 SS = safety significance
 DEG = (EE + DFS)/2, rounded to nearest integer,
 EE = environmental exposure,
 DFS = degradation factor significance [Eq. (4.1)], and
 w₁, w₂, w₃ = weighting factors having recommended values of 4, 9, and 7, respectively.

The cumulative rank for each Category I concrete structure is determined as follows:

$$CR = \sum_{i=1}^N SR_i / N, \quad (4.3)$$

where

CR = cumulative rank,

SR = subelement rank, and

N = number of subelements for the particular primary structure.

The methodology has been applied to a PWR with large dry metal containment, a BWR with reinforced concrete Mark II containment, and PWR with large dry prestressed concrete containment. The highest ranking primary concrete structure for each of these plants was found to be the shield building, containment building, and containment building, respectively. More detailed information on the structural aging assessment methodology is provided in Ref. 4.15.

4.4.2 NDE/Sampling Inspection Technology (Subtask S.3.2)

This subtask evaluates nondestructive and sampling/analysis procedures which are available for performing inservice inspections of the critical concrete components in nuclear power plants. Also, recommended criteria will be provided for use in inspection of these components to assess their current condition as well as to develop trending information for use in continued service assessments. During this reporting period, primary activities under this subtask were conducted through subcontracts with Construction Technology Laboratories, Inc. (Skokie, Illinois) and the National Institute of Standards and Technology (Gaithersburg, Maryland). During the previous reporting period a review and assessment of inservice inspection techniques and methodologies for application to concrete structures in nuclear power plants was completed and the results summarized in Ref. 4.16. Recent activities have focused on the development of statistical data for selected techniques used to determine the in-place strength of concrete. These data are required for use with the reliability-based condition assessment methodology being developed under Task S.4 (see Chapter 5).

As noted above, nondestructive testing methods for concrete structures require correlation curves. Typically a small number of destructive and nondestructive tests are conducted in tandem at noncritical locations in a structure to develop a regression relation between the two tests. If destructive tests are not permitted in the structure and laboratory tests can not be performed using specimens fabricated from the same concrete mix design, the assessment of in-place strength must be based on published results. Correlation curves and other statistical data were developed for selected nondestructive testing techniques by applying monovariant linear regression analyses (Ref. 4.17) to data obtained from publications on selected nondestructive testing techniques, i.e. break-off, pullout, rebound hammer, ultrasonic pulse velocity, and probe penetration. These techniques were selected since they comprise an overwhelming majority of the nondestructive tests performed. For each of the nondestructive techniques investigated, the data identified were subdivided by coarse aggregate type and coarse aggregate content (by weight). This subdivision was based on results provided in the literature indicating that the techniques are influenced by aggregate characteristics, e.g., the pullout and break-off tests are dependent on the aggregate type and maximum aggregate size, and the probe penetration and rebound hammer results are influenced by the aggregate hardness. Unfortunately, insufficient data were available to further subdivide the data by maximum coarse aggregate size. Since all of the data used in the study were not the result of careful experimentation, a quality rating system was developed based on the 11 criteria used to evaluate data quality for the Structural Materials

Information Center (see Table 3.5). The data source rating system developed includes nine criteria, e.g., completeness of material description, type of input, completeness of data, completeness of resources, quality of resources, consistency of results, precision and scatter, uncertainty and bias, and statistical methods used. Each data source used was given a rating from A to D (A being highest), based on these criteria. Figure 4.5 presents an example of the relationship between concrete cylinder compressive strength and rebound number that was obtained from gravel concretes having water-cement ratios from 0.37 to 0.56 (by weight) and coarse aggregate contents from 0.45 to 0.49 (by mass). Results of the regression analyses, along with the parameters required to estimate concrete compressive strength from subsequent nondestructive tests, are provided in Ref. 4.18.

4.4.3 Remedial/Preventative Measures Considerations (Subtask S.3.3)

Under this subtask, basic activities are related to an assessment of repair procedures for concrete material/structural systems and establishment of criteria for their utilization. Techniques which can be used to mitigate the effects of environmental stressors or aging factors will be identified. Recommended preventative measure procedures which can be used to effectively offset, counteract, or minimize any minor deterioration effects to prevent them from becoming significant will be established. Current work is directed at reviews of European and North American practices for repair of reinforced concrete structures, an assessment of corrosion of reinforced concrete structures, and development of a damage assessment and repair prioritization methodology.

Taywood Engineering Ltd. has completed an assessment of repair procedures from the European perspective, primarily addressing those for corrosion damage of reinforced concrete. Results obtained through Taywood Engineering Ltd.'s participation in collaborative European projects addressing repair of reinforced concrete structures and methods that can be used to extend the lifetime of a structure, either during the construction stage (incorporation of chemical admixtures or alternative cementing materials) or post-construction (use of surface coating and treatments), have been used in the assessment. Included in the report presenting results of this activity are discussions of (1) criteria used in selection of a particular repair procedure, (2) descriptions of the various repair materials and procedures currently used, (3) an evaluation of the effectiveness of the various repair techniques as determined by both in situ evaluation (testing) or performance history, and (4) an indication of the future direction of concrete repairs in Europe. Depending on the degree of deterioration and the residual strength of an element, the function of a repair may be structural, protection, cosmetic, or a combination of these three requirements. Basic elements of a repair procedure include assessment, choice of repair strategy, detailed design of repair procedure and choice of materials, planning and execution of repair procedure, and maintenance and monitoring of structure. In order to repair corrosion-damaged reinforced concrete it is sufficient to stop one of the three processes governing corrosion occurrence: anodic process, cathodic process, or electrolytic process. Basic principles of repair to stop one or more of the above corrosion processes include realkalization, limiting concrete water content, steel reinforcement coatings, cathodic protection, and avoiding oxygen access to steel reinforcement. Damage occurring from chloride presence and carbonation are important sources of concrete distress in Europe. For chloride attack, efforts are underway to provide an improved understanding of the corrosion mechanisms, the mechanism of incipient anode development, and the use of cathodic protection to overcome the problem. For carbonation, the emphasis has been placed on anti-carbonation surface treatments, protective properties of patch materials, and the

durability/compatibility of these materials. Currently in Europe the patch repair is the most widely applied method to restore both mechanical damage and spalling caused by corroding reinforcement. Cementitious, polymer-modified cementitious, and polymer-based material systems are used to effect patch repairs. Table 4.2 presents typical properties for these material systems. Coating systems have also been used effectively to increase the resistance to chloride penetration and carbonation, and reduce the concrete moisture content. Applications and an indication of the relative protection provided by several coating systems is provided in Table 4.3. The importance of repairs for damaged or deteriorated concrete structures has only recently attained any degree of prominence in Europe. Also, there is a much greater awareness of potential problems and a consensus is developing with regard to specific technical requirements for repair systems, as opposed to repair materials. Although many European countries have a range of national standards for design and construction of new buildings, and Eurocodes are in the course of preparation, standards and guidelines for repair of damaged structures are much less frequently available. At present the most widely developed regulations are those developed by the German Committee on Reinforced Concrete (Ref. 4.19). The German guidelines address four major areas: general regulations and basic design rules, design and performance, quality assurance and execution, and technical delivery conditions and test regulations. Within the next five years, or so, European Standards are expected to be in place. More detailed information on European practices for repair of degraded reinforced concrete structures is available in Ref. 4.20.

Wiss, Janney, Elstner Associates, Inc. (Northbrook, IL) is performing a complementary activity to that described above in which North American practices for repair of reinforced concrete structures are being reviewed. The distinction between the two activities is that European repair practices have primarily (but not exclusively) addressed those for corrosion-damaged reinforced concrete in building and general civil engineering structures, while in North America, repair activities have concentrated on road and bridge structures. The overall objective of this activity is to identify, describe, and quantify the effectiveness of techniques and materials which are available for use in the repair of civil engineering concrete structures, with particular emphasis on those structures in nuclear power plants. A state-of-the-art manual is being prepared which will identify methodologies and materials for repair, and rank the repair procedures based on historical performance (reliability). Basic components of the manual will include discussions of when a specific repair technique is applicable, e.g. specific crack sizes; how the techniques or materials are used, e.g. injection, routing; how to evaluate and test a repair; how to maintain the repair after it has been installed; the expected life of the repair technique; methods for determining when a repair has failed; and methods for re-repair. Information specifically addressing repair of reinforced concrete structures in light-water reactor plants is being assembled through responses to a questionnaire which was sent to U. S. utilities. In addition to a general description of the particular plant, each utility has been requested to provide information on inspection procedures utilized (frequency, technique), types of deterioration that have occurred (voids, spalling, degradation of prestressing steel, etc.), the deterioration mechanism(s) (chemical attack, freeze-thaw, etc.), repair actions that have been undertaken (grout injection, dry pack, etc.), repair locations (basemat, vertical walls, etc.), research investigations on repair materials, and performance history for repair procedures that have been utilized. To date, 29 responses have been received covering 38 units. Preliminary results indicate that the majority of the plants perform only a general visual inspection of the concrete structures in association with each integrated leak rate test. Plants which have inspection programs for the concrete structures do so at intervals

ranging from annual to five-years with visual methods being used primarily. In general, deterioration of the concrete structures has been minor due to the high initial quality of the construction and the relatively young age of the plants. Twenty-six of responses reported that they had experienced some type of damage or deterioration to the concrete. The most common locations of deterioration in the pressurized-water reactor plants were in the containment dome and in the walls and slabs of the auxiliary structures. Slabs, walls and equipment supports or pedestals were the most common areas of damage in the boiling-water reactor plants. The most common problem was concrete cracking with several of the plants also reporting spalling and voids or honeycombed concrete. Most of the problems occurred during construction and have been repaired. Other problems included delaminations, acid attack, freeze-thaw damage, sea water attack, water leakage and seepage, temperature or fire damage, leakage of post-tensioning tendon corrosion inhibitor, steel reinforcement corrosion, aggregate reactions, and sulfate attack. Twenty-seven of the responses reported that damaged concrete had been repaired. Dry packing has been used primarily to repair cracks and spalls. Other methods commonly used to repair cracks include epoxy injection, grout injection, flexible sealing, drilling and plugging, stitching, and routing and sealing. Four of the plants utilize cathodic protection for conventionally reinforced structures and two plants reported having cathodic protection systems on post-tensioned concrete members. Results of this survey will be included in the manual.

Corpro Companies, Inc. (Medina, OH) has completed a review of corrosion of reinforced concrete structures, with an emphasis on stray electrical current-induced corrosion and use of cathodic protection to control the occurrence of corrosion in these structures. Four components are required for corrosion to occur: anode, cathode, electrolyte, and metallic path. Types of corrosion that can occur on metals embedded in concrete include uniform, bimetallic, fretting, crevice, pitting, selective leaching, intergranular, stress corrosion cracking, corrosion fatigue, and hydrogen embrittlement. Conditions that affect the corrosion rate include oxygen level, conductivity of electrolyte, concentration of ions, anode/cathode area, temperature, corrosion resistance of metals, electrical relationship between metals, radiation (affects crystal structure of metal potentially making it more susceptible to localized corrosion), and bacteria. Methods which are available to detect corrosion include visual observations, half-cell potential measurements, delamination detection, electrolyte chemistry, corrosion monitors, acoustic emission, radiography, ultrasonics, magnetic perturbation, metallurgical properties, and electrical resistance. Advantages and disadvantages of each of these methods have been identified. Remedial methods to control corrosion include damage repair, cathodic protection, inhibitors, chloride removal, membrane sealers, stray current shielding, dielectric isolation, coatings, structural modifications to prevent water accumulation, material selection, and environmental modification. Stray electrical current is any current flowing in a path other than its intended circuit. Stray current will take all available conductive parallel paths back to its source and has a magnitude inversely proportional to the resistance of the path(s). With the exception of prestressing tendons, the portion of the structure picking up stray current will not experience corrosion damage. However, the area of discharge will result in metal dissolution. The amount of metal lost will be directly proportional to the current density and duration of exposure. In the case of prestressing tendons, hydrogen embrittlement corrosion may result where the structure is picking up the stray current. For the nuclear power industry, stray current can be anticipated from a variety of sources with the most significant being cathodic protection systems, high voltage direct current systems, and direct current welding operations. Half-cell potential versus time study, half-cell potential versus distance study, and cooperative (interference) testing are techniques

commonly used to detect stray current. Mitigation measures for stray current include prevention or elimination of the current source, installation of cathodic protection, draining the current from the affected source, and shielding the structure from the source. Cathodic protection is not only a rehabilitation technique for corroding structures but is also a corrosion prevention technique for steel that may lose its inherent passivity by contamination of the concrete at a later time or even due to occurrence of stray currents. It mitigates the corrosion reaction by imposing direct current flow between an anode placed on the concrete surface and the metal to be protected. Cathodic protection systems are of the impressed current or sacrificial anode type. Impressed current systems use a DC power supply (rectifier) to force current flow from a relatively inert electrode (anode) through the concrete to the steel surface to be protected. In sacrificial systems a metal that is more anodic (higher tendency to corrode) than the steel embedded in the concrete is used as the source of energy. The theory of cathodic protection is basically that all galvanic corrosion of steel is halted when the potential difference between the anodic and cathodic areas on the steel can be made equal to zero. Criteria commonly used for atmospherically exposed bridge structures to indicate when this condition is sufficiently met include 100 mV polarization decay, $E \log I$ analysis, macrocell current reversal, fixed current density, voltage probe measurement, half-cell potential measurement and rebar probe polarization decay. To be most effective, cathodic protection requires electrical continuity of all metallic components within a concrete structure. Since cathodic protection systems are DC, they have the potential to cause stray current corrosion in other structures, and application of cathodic protection to high-strength steel used in prestressing wires or strands may result in embrittlement due to generation of hydrogen at the cathode. More detailed information on corrosion of steel embedded in concrete, stray current-induced corrosion of reinforced concrete structures, and cathodic protection systems is contained in Ref. 4.21.

Howard University is developing a systematic rating system for application in the repair of degraded reinforced concrete structures in nuclear power plants. Similar computerized rating systems have been developed for evaluating general civil works concrete structures, e.g., BRAIN (Building Rating Analysis and Investigation System). The first of two planned phases for this work has been completed (Ref. 4.22). In this phase existing damage assessment and repair prioritization systems for bridge and building structures were reviewed to determine their applicability to nuclear power plant concrete structures and an approach was identified for development of a rating system for nuclear power plant concrete structures. Input developed under the first phase of the study is presently being used in the second phase. A computer program has been developed to simplify use of the aging assessment methodology (see Sect. 4.4.1) and has been validated through its application to the three examples presented in Ref. 4.15. The approach used in development of the rating system is similar to that for the aging assessment methodology, except that instead of using four factors to develop a rating (repair prioritization number) two factors are considered, i.e. the environmental exposure and the damage significance. The resulting repair rating system will be in the form

$$RPN = w_1(EE) + w_2(DS), \quad (4.4)$$

where

RPN	= repair priority number,
EE	= environmental exposure, and
DS	= damage significance.

A qualitative presentation of the repair rating system for reinforced concrete structures subjected to chloride environments is shown in Figure 4.6. Criteria are being developed so that a quantitative assessment of the damage to a structure can be made, i.e., environmental exposure and damage values are being developed so that scales can be applied to Fig. 4.6 and regions indicating testing and evaluation or repair actions delineated. When completed, results developed under this phase will provide a logical connection between the structural aging assessment methodology and the repair activities noted above.

4.5 PLANNED FY 1993 ACTIVITIES

The reports on European and North American repair practices for concrete structures, corrosion of steel in concrete, and the repair prioritization methodology will be published. Work will initiate on development of a recommended in-service inspection methodology for reinforced and post-tensioned concrete structures in nuclear power plants.

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4.7 MILESTONE STATEMENT AND SCHEDULE

The statement and schedule for Task S.3 milestones given in the following charts.

MILESTONE STATEMENT AND SCHEDULE

TASK: S.3 STRUCTURAL COMPONENT ASSESSMENT/REPAIR TECHNOLOGY							
SUBTASK/MILESTONE	FY 92						FY 96 BEYOND FY 96
	1	2	3	4	O N D J F M A M J J A S	FY 94 1 2 3 4	FY 95
S.3 STRUCTURAL COMPONENT ASSESSMENT/REPAIR TECHNOLOGY							
S.3.1 LWR Critical Component Classification							
(Activities Completed)							
S.3.2 NDE/Sampling Inspection Technology							
A. Issue Reports on Inservice Inspection and Structural Assessment Techniques for Concrete Materials/Structures	a	b					
a. Complete Draft Report Presenting Surveillance Data for Several United Kingdom PCPVs							
b. Issue Report on Effectiveness of Testing Techniques for Determination of In-Place Concrete Strength							

MILESTONE STATEMENT AND SCHEDULE

TASK: S.3 STRUCTURAL COMPONENT ASSESSMENT/REPAIR TECHNOLOGY																									
	SUBTASK/MILESTONE	FY 92				FY 1993												FY 94				FY 95	FY 96	BEYOND FY 96	
		1	2	3	4	O	N	D	J	F	M	A	M	J	J	A	S	1	2	3	4				
S.3.3	Remedial/Preventative Measures Considerations																								
	A. Issue Reports on Review of Techniques for Repair of Concrete Materials and Structures																								
	a. Complete Draft Report Assessing European Repair Practices																								
	b. Complete Draft Report Assessing North American Repair Practices and Presenting Results of Questionnaire on Concrete Performance in Nuclear Power Plants

Fig. 4.1. Level 2 work breakdown structure for SAG Task S.3: Structural Component Assessment/Repair Technology.

Fig. 4.2. Level 3 work breakdown structure for SAG Task S.3: Structural Component Assessment/Repair Technology.

Fig. 4.3. General time rate of degradation curve for concrete structures. Source: Adaptation of diagram showing typical rate of deterioration of concrete subjected to sulfate attack or alkali-silica reaction presented in J. R. Clifton and L. I. Knab, *Service Life of Concrete*, NUREG/CR-5466, National Institute of Standards and Technology, Gaithersburg, MD, November 1989.

Fig. 4.4. Concrete component current condition assessment methodology. Source: Adaptation of a procedure presented in T. L. Rewerts, "Safety Requirements and Evaluation of Existing Buildings," *Concr. Int'l*, 7(4), American Concrete Institute, Detroit, April 1985.

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Figure 4.5 Relationship between cylinder compressive strength and rebound number for selected gravel concretes.

Fig. 4.6 Model for development of repair prioritization methodology.

Table 4.1. Nondestructive evaluation methods for inspection of concrete materials

Material and characteristic	Available methods of detection	
	Primary	Secondary
<u>Concrete</u>		
General quality	Ultrasonic pulse velocity Rebound hammer Penetrating probe	Ultrasonic pulse echo Gamma radiography ^a
Cracking/voids	Visual inspection Ultrasonic pulse velocity Acoustic impact	Ultrasonic pulse echo Gamma radiography ^a
Strength	Penetrating probe Rebound hammer Pullout methods	Breakoff methods Surface hardness methods
<u>Mild steel reinforcing</u>		
Location/size	Pachometer Gamma radiography ^a	Ultrasonic pulse echo Penetrating radar
Corrosion	Visual inspection ^b Electrical potential measurements	Rate of corrosion probes
<u>Prestressing tendons</u>		
Loads	Tendon liftoff tests	Load cells
Corrosion	Visual inspections Mechanical property tests Tendon load vs elongation tests	Corrosion inhibitor analysis
<u>Concrete embedments</u>		
	Visual inspections Mechanical testing	

^aLimited to concrete thickness ≤ 450 mm.

^bReflected through cracking and staining observed at concrete surface.

Table 4.2. Typical properties of patch repair materials.

Material Property	Patch Repair Material		
	Concrete	Mortar	Polymer Modified Mortar
Compressive Strength, MPa	20 - 50	20 - 50	55 - 100
Elastic Modulus, GPa	20 - 40	15 - 35	10 - 20
Thermal Expansion Coefficient (10 ⁻⁶ mm/mm/°C)	7 - 13	8 - 20	8 - 20
Maximum Service Temperature Under Load (°C)	>300	>300	100 - 300
			40 - 80

Table 4.3. Relative performance for several coating systems for concrete materials.

Coating	Application*					Protection*			
	Damp Conditions	Alkali Resistance	Easy to Apply	Low Hazard	Resist Cl ⁻	Resist CO ₂	Vapor Transmission	Resist Rain	
<u>Film Forming</u>									
Epoxy resin	4	1	5	5	1	1	5	1	
Coal tar epoxy	4	1	5	5	1	1	5	1	
Polyurethane	4	1	5	5	1	1	5	1	
Chlorinated rubber	2	1	1	5	1	1	5	1	
Bituminous	2	3	1	3	3	3	3	1	
Acrylic resin	2	3	1	1	3	1	1	1	
<u>Densifying</u>									
Silicate/flurosilicate	1	1	1	1	5	5	1	3	
Cementitious	1	1	1	1	3	3	1	3	
<u>Non-film Forming</u>									
Silane/siloxane	2	2	1	5	1	5	1	1	
Oil impregnation	4	5	1	1	3	5	1	4	
Silicone	4	5	1	2	3	5	1	3	

* 1 = very good, 5 = very poor.

5. TASK S.4 – QUANTITATIVE METHODOLOGY FOR CONTINUED SERVICE DETERMINATIONS

5.1 OBJECTIVE

The goal of this task is to develop a methodology to facilitate quantitative assessments of current and future structural reliability and performance of concrete structures in nuclear power plants, taking into account those effects that might diminish the ability of these structures to withstand future operating, extreme environmental or accident conditions. Specific objectives associated with accomplishing this goal are to: (1) identify models to evaluate changes in strength of concrete structures over time in terms of initial conditions, service load history, and aggressive environmental factors; and (2) formulate a methodology to predict structural reliability of existing concrete structures during future operating periods from a knowledge of initial conditions of the structure, service history, aging, nondestructive condition assessment techniques, and inspection/maintenance strategies. When completed, the condition assessment methodology will be in a form such as summarized in Fig. 5.1 and will provide answers to the following questions, or at least frame the questions in a logical manner.

1. Which aging factors are particularly significant for concrete structures in terms of their future reliability?
2. Has the original strength of the structure degraded over time as a result of environmental stressors or aging factors?
3. What is the residual life of a structure and how would it respond to a design-basis event?
4. Which NDE techniques or in-situ strength measurements are most useful for demonstrating reliability of an existing structure?
5. What inspection procedures should be required and how frequently should they be conducted?
6. Which repair techniques are most effective in extending usable life and enhancing reliability?
7. What documentation (analysis or data) regarding condition assessment of Category I concrete structures should be provided to support an application to continue operation of a nuclear power plant past its original design period?

Figure 5.2 presents the Level 2 work breakdown structure for the quantitative methodology for continued service determinations task.

5.2 BACKGROUND

Once it has been established that a component has been subjected to environmental stressors or aging factors that have resulted in deteriorating influences,* the effects of these influences must be related to a condition or structural reliability assessment, especially when the component is being considered for a continuation of its service past its initial licensing period. In order to continue the operation of NPPs, evidence should be provided that the critical safety-related concrete structures in these facilities in their current condition are able to withstand potential future design events (over the continued service period) with a level of reliability adequate to meet requirements for protecting the public health and safety. Barriers to prediction of the future performance of a structure or component are listed in Ref. 5.1 and include the availability of: (1) a systematic approach or methodology for treating the problem, (2) an effective mechanism for obtaining and reporting data on the actual inservice performance of materials, (3) knowledge of the mechanisms of degradation, (4) knowledge of the environmental factors causing degradation, (5) the ability to simulate or account for the synergism between degradation factors, and (6) mathematical models describing the material behavior in specific environments or applications. With the exception of the systematic approach or methodology, each of these barriers has been addressed previously.

A methodology for conducting such an assessment presently does not exist (Refs. 5.2 and 5.3); however, structural reliability techniques have advanced to the point where it should be possible to make a quantitative evaluation of the durability (residual life) of a concrete structure based on a knowledge of the condition of the structure when it was built, its service history, its present condition, and projected use during a time interval past its initial licensing period (Ref. 5.4). Such techniques would provide guidance to utilities and regulators alike regarding the technical data to be submitted in support of an application for continued service, current structures condition, and the need, if any, for future inspections or regular maintenance as a condition for granting a license for continued operation.

5.3 PLAN OF ACTION

This task will implement results obtained under Tasks S.2 and S.3 to develop a reliability-based evaluation methodology for concrete structures which will enable the factors that affect structural durability to be taken into account (Fig. 5.3). The methodology will integrate information on degradation and damage accumulation, environmental factors, and load history into a decision tool that will enable a quantitative measure of structural reliability and performance under projected future service conditions based on a condition assessment of the existing structure. The interaction of this task with the condition assessment methodology of Task S.3 (Fig. 4.4) is illustrated schematically in Fig. 5.4. The Level 3 work breakdown structure for this task is presented in Fig. 5.5.

5.3.1 Predictive Model Development (Subtask S.4.1)

Predictive or damage models will be identified or developed that will be utilized in the formulation of procedures for assessing the current condition as well as the development of

* Even a structure that exhibited no obvious signs of deteriorating influences might have to be evaluated for continued service considerations.

reliability-based future condition assessment of critical safety-related concrete components. The models will enable the change in strength of concrete structures over time to be evaluated in terms of initial conditions, applied load history, and a parameterization of environmental stressors and aging factors. Since models developed under this subtask will deal only with a subset of the parameters needed to predict future structural reliability, many of these models may have to be synthesized to encompass all significant factors and to account for any synergistic effects.

5.3.2 Data Assemblage/Implementation (Subtask S.4.2)

Included under this subtask is the assemblage of pertinent data for use in establishing functional and performance requirements for critical components and in development of the predictive models. Requirements used in the design, fabrication, erection, and testing of the critical components will be identified through documents such as General Design Criteria 1, "Quality Standards and Records;" 2, "Design Bases for Protection Against Natural Phenomena;" and 4, "Environmental and Missile Design Bases" of Appendix A, "General Design Criteria for Nuclear Plants" to 10CFR50, "Licensing of Production and Utilization Facilities" (Ref. 5.5). Additional information will be obtained from codes such as American Concrete Institute (ACI) Standard 318, "Building Code Requirements for Reinforced Concrete" (Ref. 5.6), ACI 349, "Code Requirements for Nuclear Safety Related Structures" (Ref. 5.7), and ACI 359, "ASME Section III—Division 2, Code for Concrete Reactor Vessels and Containments" (Ref. 5.8). Quantitative performance data on the factors which can impact structural durability will be identified, including age, operating conditions, environmental stressors, design bases, material quality, and construction methods. Uncertainties associated with this data will be identified for use in the reliability-based methodologies in Task S.4.3. In-situ strength determinations by destructive or nondestructive evaluation methods will be assessed, with particular attention to quantifying measurement uncertainties associated with various condition assessment methods. The impact of remedial measures will be evaluated under this subtask.

5.3.3 Methodology Development (Subtask S.4.3)

This subtask will develop stochastic models of the variables in the deterministic prediction models that have been identified and refined in Subtasks S.4.1 and S.4.2. Techniques also will be developed for analyzing current or future reliability of concrete structures. Structural loads arising from service, extreme environmental and accident conditions will be modeled as stochastic processes, as will environmental conditions under which the nuclear power plant must operate. The models of structural behavior used to predict response to various postulated conditions or strength changes over time also are random in nature (Ref. 5.9). Uncertainties in life prediction models, in strength, and in the ability of nondestructive evaluation methods to predict strength will be included in development of the methodology. The methodology to be developed for predicting future strength, or continued service period, will reflect the stochastic characterization of the plant environment to determine probability distributions of structural capacity at some future time. Conversely, results developed under this subtask will make it possible to determine a probability distribution of additional usable life associated with maintaining a minimal structural capacity.

5.3.4 Validation Studies (Subtask S.4.4)

The predictive models and supporting data identified in previous subtasks will be refined and validated under this subtask through the use of laboratory and prototypical structures testing. Laboratory tests will enable degradation factor effects (load and environmental stressor) to be evaluated under controlled conditions. Ideally, the prototypical structures tests will involve an existing concrete structure such as a decommissioned nuclear plant. Initial conditions will be defined from the codes used in design, available structural design documents, and inspection and construction data. Best estimates of applied service loads and environmental parameters will be factored into the evaluation as well as available information from inspection data or repairs. The analysis will be run forward to the present time and the current predicted strength compared to an estimate of the actual present strength. Results will then be utilized to modify the predictive methodology.

5.4 SUMMARY OF ACCOMPLISHMENTS

During the reporting period covered by this document, a report was completed by the principal investigators for Task S.4. which describes a methodology for condition assessment and reliability-based life prediction of concrete structures in nuclear power plants (Ref. 5.10). The methodology includes models to predict structural deterioration due to environmental stressors, a data base to support the use of these models, and methods for analyzing time-dependent reliability of concrete structural components subjected to stochastic loads. The methodology can be used to provide a basis for selecting appropriate periods for continued service and/or determining optimum intervals and extent of inspection and maintenance. Inspection/maintenance strategies have been identified to minimize the expected future cost of keeping the failure probability of a structure at or below an established target failure probability during its anticipated service period. Results of this evaluation are provided in a report which has been prepared (Ref. 5.11). A summary is provided below.

5.4.1 Introduction

The failure probability of a structural component under stationary random loading can be evaluated as a function of time if the degradation function defining the fraction of initial strength remaining at time, t , and the probabilistic characteristics of the initial strength and loads modeled as stochastic processes are known (Ref. 5.12). In order to evaluate the effect of periodic inspection and maintenance on the failure probability of a structure, it is necessary to relate the strength degradation to the damage intensities and to determine the impact of various repair strategies on strength.

5.4.2 Degradation Function Based On Individual Damage Intensities

The damage intensity is modeled in the abstract as a state variable taking a value within the interval $[0, 1]$; the values 0 and 1 indicate no damage and no residual strength, respectively. An example of this state variable would be the ratio of area of reinforcement lost due to corrosion to the original area. The following assumptions are made:

- (a) Initiation of damages in a component is described by a Poisson process in which the expected number of damages in time interval $(t, t + \Delta t]$ is $\int_t^{t+\Delta t} v(\tau) d\tau$ for $t > 0$. $v(\omega)$ is dependent on the surface area or volume of the component.
- (b) Damages initiate homogeneously over the surface area or volume of the component.
- (c) Once damage initiates at location j , it grows according to

$$X_j(t) = \begin{cases} 0 & ; 0 \leq t < T_{Ij} \\ C_j(t - T_{Ij})^\alpha & ; t \geq T_{Ij} \end{cases} \quad (5.1)$$

in which $X_j(t)$'s are the intensity of damage at time t , T_{Ij} 's are the random times at which damage initiates, C_j 's are damage growth rates which are assumed to be identically distributed and statistically independent random variables described by a cumulative distribution function (cdf) $F_C(c)$, and α is a deterministic parameter. Parameters C and α depend on the degradation mechanism (e.g., Ref. 5.13).

- (d) The degradation function, $G(t)$, for a component can be given in terms of damage intensities as,

$$G(t) = 1 - \max_{all j} \{X_j(t)\}. \quad (5.2)$$

Consider damages which initiate within interval $(t_1, t]$. Given that the number of these damages equals n , the rank-ordered initiation times, $T_{I1}, \dots, T_{In}$ are n order statistics of random variables $W_{I1}, \dots, W_{In}$ which are statistically independent and identically distributed with probability density function (pdf) expressed as [Ref. 5.14],

$$f_{W_I}(w) = \begin{cases} \frac{v(w)}{\int_{t_1}^t v(\tau) d\tau} & ; t_1 \leq w \leq t \\ 0 & ; \text{Otherwise} \end{cases} \quad (5.3)$$

From assumption (c), the cdf of $X_j(t)$, $F_X(x; t_1, t)$, is expressed as,

$$F_X(x; t_1, t) = \int_{t_1}^t F_C\left(\frac{x}{(t-\tau)^\alpha}\right) f_{W_I}(\tau) d\tau. \quad (5.4)$$

The cdf of $X_{max}(t_1; t) = \max_j \{X_j(t) \text{ initiating within } (t_1, t]\}$ is expressed as [Ref. 5.15],

$$F_{X_{max}}(x; t_1, t) = \exp\left[-\int_{t_1}^t v(\tau) d\tau \{1 - F_X(x; t_1, t)\}\right]. \quad (5.5)$$

From assumption (d), the mean of the degradation function is evaluated by,

$$E[G(t)] = 1 - \int_0^1 [1 - F_{X_{max}}(x; 0, t)] dx \quad (5.6)$$

In the course of the analysis, it was found that the variability in $G(t)$ has a secondary effect on the time-dependent reliability of a component, and thus the reliability can be evaluated considering only the mean of $G(t)$, defined as $g(t)$ (Ref. 5.15).

5.4.3 Degradation Function After Repair

No nondestructive evaluation (NDE) method can detect a given defect with certainty. The imperfect nature of NDE methods must be described in statistical terms. Figure 5.6 illustrates conceptually the probability, $d(x)$, of detecting a defect of size x . Such a relation exists, at least conceptually, for each in-service inspection technology.

Assume that during inspection/maintenance the entire component is inspected, that all detected damages are repaired immediately and completely, and that the repaired parts of the component are restored to their initial strength levels. Then the effect of inspection/maintenance on $g(t)$ depends on the detectability function, $d(x)$, associated with the NDE method. The inspection with higher $d(x)$ makes repair more likely and, accordingly, leads to higher values of the degradation function, $g(t)$. In the limit, if an inspection is perfect, i.e., $d(x) = 1$ for $x > 0$, then the component is restored to its original condition by the repair.

5.4.3.1 Step Detectability Function

Let us first assume that the detectability function, $d(x)$, is defined as

$$d(x) = \begin{cases} 0 & ; \quad 0 \leq x < x_{th} \\ 1 & ; \quad x_{th} \leq x \leq 1 \end{cases} \quad (5.7)$$

where x_{th} is the minimum detectable value of damage. The same detection threshold values are assumed for all inspections. Following m inspections at $t_R = \{t_{R_1}, \dots, t_{R_m}\}$, some of the damages are repaired and the cdf describing $X(t)$ and the number of damages existing at time $t > t_{R_m}$, $N(t)$, changes. The intensities of damages that initiate after t_{R_m} are independent of repair, and only the pdf of the intensities of damages initiating before t_{R_m} is updated. Let us consider damages which exist at time t and initiate with (A) $(0, t_{R_m}]$ and (B) $(t_{R_m}, t]$. The number of damages left unrepaired after t_{R_m} , $N_{(A)}$, can be described by a filtered Poisson process with a parameter $p \cdot v(w)$, where

$$p = P[\text{A damage is not repaired by } t_{R_m}] = F_X(x_{th}, 0, t_{R_m}) \quad (5.8)$$

while the number of damages initiating within $(t_{R_m}, t]$, $N_{(B)}$, is described by a Poisson process with a parameter $v(w)$. In other words, the number of defects existing at time t can be described by a nonstationary Poisson process within a parameter $v''(w)$ given by

$$v''(w) = \begin{cases} p \cdot v(w) & ; 0 < w \leq t_{R_m} \\ v(w) & ; t_{R_m} < w \leq t \end{cases} \quad (5.9)$$

Therefore, the procedure to estimate the degradation function for a component before an inspection/repair discussed in the preceding section can be used to estimate the function after multiple inspection/repair, replacing $v(w)$ by $v''(w)$, and $F_X(x; 0, t)$ by the updated cdf $F''_X(x; t_{R_m}, t)$.

By the theorem of total probability, $F''_X(x; t_{R_m}, t)$ can be expressed as,

$$F''_X(x; t_{R_m}, t) = F_{X(A)}(x; t_{R_m}, t) \cdot P(A) + F_{X(B)}(x; t_{R_m}, t) \cdot P(B) \quad (5.10)$$

in which $F_{X(A)}(x; t_{R_m}, t)$ and $F_{X(B)}(x; t_{R_m}, t)$ are the cdf of intensity of damages in group (A), $X_{(A)}(t)$, and intensity of damages in group (B), $X_{(B)}(t)$, respectively, and $P(A)$ and $P(B)$ are the probabilities that a defect belongs to group (A) or (B):

$$P(A) = P[W_I \leq t_{R_m}] = \int_0^{t_{R_m}} f_{W_I}(w) dw \quad (5.11)$$

$$P(B) = 1 - P(A) \quad (5.12)$$

in which $f_{W_I}(w)$ is evaluated by Eqn. (5.3) replacing $v(w)$ with $v''(w)$. $F_{X(A)}(x; t_{R_m}, t)$ is expressed as,

$$F_{X(A)}(x; t_{R_m}, t) = P[X(t) < x | X(t_{R_m}) < x_{th}]$$

$$= \frac{\int_0^{t_{R_m}} F_C \left(\min \left\{ \frac{x}{(t-\tau)^\alpha}, \frac{x_{th}}{(t_{R_m}-\tau)^\alpha} \right\} \right) f_{W_I}(\tau) d\tau}{\int_0^{t_{R_m}} F_C \left(\frac{x_{th}}{(t_{R_m}-\tau)^\alpha} \right) f_{W_I}(\tau) d\tau} \quad (5.13)$$

The cdf $F_{X_{max}(B)}(x; t)$ is given by

$$F_{X_{max}(B)}(x; t) = F_X(x; t_{R_m}, t) \quad (5.14)$$

5.4.3.2 General Detectability Function

In general, the detectability function, $d(x)$, is not a step function but rather a non-decreasing function of damage intensity (see Fig. 5.6). Noting that a detectability function has the same property as a cdf, i.e., non-decreasing and taking a value within $[0, 1]$, $d(x)$ can be interpreted as describing an inspection with a step detectability function in which the threshold value, X_{th} , is a random variable with cdf, $F_{X_{th}}(x) = d(x)$. In short,

$$\begin{aligned} d(x) &= P[\text{Detect damage with intensity as small as } x] \\ &= P[X_{th} \leq x]. \end{aligned} \quad (5.15)$$

$F_{X_{th}}(x)$ can be interpreted as the distribution function of the smallest detectable defect size. The cdf of damage intensity in group (A) after inspection is evaluated as (cf. Eqn. (5.13)),

$$\begin{aligned} F_{X_{(A)}}(x; t_{R_m}, t) dx &= P[X(t) \leq x | X(t_{R_m}) < X_{th}] \\ &= k \cdot \int_0^1 F_{X_{Step}}(x; x_{th}) \cdot F_X(x_{th}; 0, t_{R_m}) \cdot f_{X_{th}}(x_{th}) dx_{th} \end{aligned} \quad (5.16)$$

in which k is a normalizing constant so that $F_{X_{(A)}}(x; t_{R_m}, t)$ is a legitimate cdf, $F_{X_{Step}}(x; x_{th})$ is the cdf of damage intensity for a step detectability function with a threshold value x_{th} , evaluated by Eqn. (5.13), and

$$f_{X_{th}}(x) = \frac{d}{dx} d(x). \quad (5.17)$$

The cdf of damage intensity in group (B) after inspection remains unchanged from Eqn. (5.14).

5.4.4 Effect of Inspection/Repair Operations on Reliability

In order to study a simple case, let us assume that:

- (a) The initiation of damages is described by a stationary Poisson process with $v = 5/\text{yr}$.
- (b) Damage grows linearly as a function of time as described by Eqn. (5.1) with $\alpha = 1$; moderate reinforcement corrosion is typified by such growth (Ref. 5.15).
- (c) The degradation rate, C , is lognormally distributed with mean values, $\mu_C = 0.00125$, which corresponds to $E[X(40)|T_I = 0] = 0.05$, and with coefficient of variation, $V_C = 0.5$.

The effect on the mean degradation function of inspection/repair described by several detectability functions is illustrated in Fig. 5.7. The first detectability function considered is a step detectability function in which $x_{th} = 0.03$; in the second, X_{th} is uniformly distributed

(i.e., $d(x)$ is linear between x_{min} and x_{max} , where $d(x_{min}) = 0$ and $d(x_{max}) = 1$); in the third and fourth, X_{th} is lognormally distributed with mean, $\mu_{X_{th}}$, equal to 0.03, and coefficient of variation, $V_{X_{th}}$, equal to 0.3 or 0.5. It is assumed that inspection/repair is carried out at $t_{R_m} = 20$ years. The mean degradation function decreases as $V_{X_{th}}$ increases (which would result in lower reliability); however, the effect of the general shape of $d(x)$ (Fig. 5.6) is not significant and decreases with time elapsed since inspection. This insensitivity of the mean degradation to the choice of detectability function suggests that a general detectability function might be approximated for practical purposes by a step function with $x_{th} = \mu_{X_{th}}$. This would be advantageous for NDE technologies currently used for reinforced concrete structures because information on $\mu_{X_{th}}$ may be more readily available than information on $d(x)$.

The effect of multiple inspections/repairs on the mean degradation function is illustrated in Fig. 5.8, considering the step detectability function. Inspections/repairs are carried out at 20, 30, 40 and 50 years with $x_{th_j} = x_{th} = 0.05$ or at 30 years with $x_{th} = 0.01$. With multiple inspection/repair, the mean degradation function can be kept within a narrow range during the service life of a structure; the width and the location of the range can be controlled by the frequency and the threshold value of the inspections.

Failure probabilities for a service life of 60 years evaluated with the degradation functions illustrated in Fig. 5.8 are presented in Fig. 5.9 for a reinforced concrete component designed for flexure using the design load combination for dead plus live load in ACI Standard 318 (Ref. 5.16),

$$0.9R_n = 1.4D_n + 1.7L_n \quad (5.18)$$

in which R_n = nominal strength, and D_n and L_n = nominal dead and live loads, respectively. The probabilistic models of load intensity used in this illustration are summarized in Table 5.1. It is assumed that the initial strength has a lognormal distribution with $\mu_R = 1.15R_n$ and coefficient of variation (c.o.v.) $V_R = 0.15$ and that $D_n = L_n$. The basis for these load and resistance statistics is given elsewhere (Ref. 5.12, Ref. 5.15). The slope of $F(t)$ changes at the time of repair. The combined effect of multiple repairs with $x_{th} = 0.05$ leads to a failure probability during 60 years that is comparable to that of the single more intensive repair.

5.4.5 Conclusions

Results such as shown in Fig. 5.9 suggest the existence of an optimum inspection/maintenance strategy in which the failure probability of a component is kept below an established target probability during its service life and the total expected cost, defined as the sum of the cost of inspections/repairs and the expected loss due to failure, is minimized. The development of such a strategy is presented in detail elsewhere (Ref. 5.15). In situations where the cost of failure dominates the expected total cost, the optimum policy involves inspection/repair at nearly uniform intervals during the service life of the structure.

5.5 PLANNED FY 1993 ACTIVITIES

The report on reliability-based condition assessment of concrete structures in nuclear power plants will be published (Ref. 5.11). Predictive models for reinforced concrete strength degradation that have been incorporated into the time-dependent reliability analysis will be refined as necessary. Data on nondestructive and destructive in-situ strength evaluation methods will continue to be implemented into the reliability analysis. Guidelines will be developed for selecting optimum in-service inspection/maintenance strategies to provide appropriate intervals of inspection, identify desirable attributes of nondestructive evaluation methods, and identify the degree of maintenance necessary to assure a predetermined level of reliability over a prescribed period of time. Validation studies will be initiated to demonstrate the reliability-based methodology developed for predicting residual strength or future safe operating conditions for concrete structures in nuclear power plants. Also, an assessment of the importance of aging of concrete structures on seismic probabilistic risk assessment or seismic margin analysis of an existing nuclear power plant will be initiated.

5.6 REFERENCES

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- 5.3 B. M. Morris and J. P. Vora, *Nuclear Plant Aging Research (NPAR) Program Plan*, NUREG/1144 (Rev. 1), Division of Engineering Technology, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC, July 1985.
- 5.4 J. T. P. Yao et al., "Stochastic Fatigue, Fracture and Damage Analysis," *J. Str. Safety* 3(4), 231-267, 1986.
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- 5.9 B. R. Ellingwood and H. Hwang, "Probabilistic Descriptions of Resistance of Safety-Related Structures in Nuclear Plants," *Nucl. Eng. Des.* 88(2), 1985.
- 5.10 B. R. Ellingwood and Y. Mori, "Condition Assessment and Reliability-Based Life Prediction of Concrete Structures in Nuclear Plants," ORNL/NRC/LTR-92/4 (Subcontract Report 19X-SD684V, The Johns Hopkins University, Baltimore, Maryland), Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., Oak Ridge, Tenn., January 1992.
- 5.11 B. R. Ellingwood and Y. Mori, "Reliability-Based Condition Assessment of Concrete Structures in Nuclear Plants," ORNL/NRC/LTR-93/1 (Subcontract Report 19X-SD684V, The Johns Hopkins University, Baltimore, Maryland), Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., Oak Ridge, Tenn., January 1992.
- 5.12 B. R. Ellingwood and Y. Mori, "Probabilistic Methods for Life Prediction of Concrete Structures in Nuclear Power Plants," Transactions of the 11th International Conference on Structural Mechanics in Reactor Technology, Vol. D, pp. 291 - 296, Tokyo, Japan, August 1991.
- 5.13 J. R. Clifton and L. I. Knab, "Service Life of Concrete," NUREG/CR-5466, U. S. Nuclear Regulatory Commission, Washington, D. C., November 1989.
- 5.14 H. M. Taylor and S. Karlin, "An Introduction to Stochastic Modeling," Academic Press, Orlando, Fla., 1984.
- 5.15 Y. Mori, "Reliability-Based Condition Assessment and Life Prediction of Concrete Structures," thesis presented to the Johns Hopkins University, Baltimore, Maryland, in partial fulfillment of the requirements for the degree of Doctor of Philosophy, 1992.
- 5.16 American Concrete Institute Committee 318, "Building Code Requirements for Reinforced Concrete," ACI Standard 318-89, American Concrete Institute, Detroit, Mich., 1989.

5.7 MILESTONE STATEMENT AND SCHEDULE

The statement and schedule for the milestones in Task S.4 are given in the following charts.

MILESTONE STATEMENT AND SCHEDULE

[illegible]

Fig. 5.1. System reliability approach to condition assessment.

Fig. 5.2. Level 2 work breakdown structure for SAG Task S.4: Quantitative Methodology for Continued Service Determinations.

Fig. 5.3. Structural durability is based on a systematic evaluation of several factors, including synergism of the various factors.

ORNL-DWG 93-2609 ETD

Fig. 5.4 LWR concrete component evaluation methodology. Source: Adaptation of a procedure presented in T. L. Rewerts, "Safety Requirements and Evaluation of Existing Buildings," *Concr. Int'l.* 7(4), American Concrete Institute, Detroit, April 1985.

Fig. 5.5. Level 3 work breakdown structure for Task S.4: Quantitative Methodology for Continued Service Determinations.

ORNL-DWG 93-2611 ETD

Fig. 5.6. Probability of detection of a defect of size x .

ORNL-DWG 93-2612 ETD

Fig. 5.7. Effect of several detectability functions on mean degradation function of inspection/repair.

Fig. 5.8. Effect of multiple inspections/repairs on mean degradation function.

Fig. 5.9. Failure probability with multiple inspections/repairs .

Table 5.1. Load process parameters

Type	Mean*	Coefficient of variation, c.o.v.	Probability distribution function, pdf	Mean rate of occurrence, λ , yr^{-1}	Mean duration,** τ
Dead load	$1.00 D_n$	0.07	Normal	-	-
Live load	$0.40 L_n$	0.50	Type I	0.5	3 months

* D_n and L_n are nominal loads.

** τ is the mean duration of a load event.

APPENDIX A

SUMMARY OF ACCOMPLISHMENTS

Task S.1 – Program Management

- Program Planning and Resource Allocation – Subtask S.1.1
 - Issued two five-year program plans (ORNL/NRC/LTR-90/1, ORNL/NRC/LTR-92/3)
 - Participated in eight program briefings with NRC personnel
 - Implemented twelve subcontracts (6 completed, 1 pending)
 - Participated in ANS topical meetings on Plant License Renewal (Orlando) and Nuclear Power Plant Life Extension (Snowbird), Second International Seminar on Containment of Nuclear Reactors (Lausanne), Fourth (Alexandria) and Fifth (Washington, D.C.) Workshops on Containment Integrity, American Power Conference (Chicago), ASME Power Plant Systems/Components Aging Management and Life Extension (San Diego, New Orleans), 3rd Int'l Seminar on Containment of Nuclear Reactors (UCLA), NRC Nuclear Power Plant License Renewal Workshop (Reston), SMiRT Conferences (Lausanne and Anaheim), National Concrete Engineering Conference (Chicago), ASCE Materials Engineering Conference (Atlanta), U. S. Army Corps of Engineers Workshop on Deterioration and Condition Assessment of Concrete Navigational Structures (Washington, D. C.), National Research Council Transportation Research Board 71st Annual Meeting (Washington, D. C.), and U. S. National Standard for the Exchange of Product Model Data Conference (Gaithersburg).
 - Participated in short courses on nuclear power reactor safety (MIT), containment in-service inspection and testing (ASME), General Electric Boiling Water Reactor Technology (USNRC), and Westinghouse Pressurized Water Reactor Technology (USNRC)
 - Completed investigative trip to Fort Calhoun Nuclear Power Plant to participate in assessment of impact on structural integrity of containment building of occurrence of post-tensioning system grease leakage
- Program Monitoring and Control – Subtask S.1.2
 - Presented papers at Sixteenth (one paper), Eighteenth (three papers), Nineteenth (one paper), and Twentieth (one paper) WRSMS
 - Presented three papers at NRC Aging Research Information Meeting
 - Completed technical reviews and provided comments to NRC on NUMARC license renewal technical reports on PWR containments and Class 1 Structures, and on Chap. 4.0 "Civil Structures" of SRP – License Renewal
 - Implemented annual technical reporting system and issued three reports
 - Presented program overviews to ASME Advisory Committee on Technical Issues (Nashville), New Production Reactors Working Group on Standards (Washington, D. C.) and Atomic Safety Licensing Board (Gatlinburg)
 - Completed technical review of INEL reports "Life Assessment Procedures for Metallic Containments (Chapter 5)" NUREG/CR-5314, and "Quantification of Degradation Damage in Light Water Reactor Concrete Containments" (INEL-12-6002, Rev. 0)

Task S.1 – Program Management (cont.)

- Documentation and Technology Transfer – Subtask S.1.3
 - Coordinated activities with other government agencies-related programs
 - Low-level radioactive waste program (NRC)
 - Modular high-temperature gas-cooled reactor (NRC)
 - Nuclear plant aging research program (NRC)
 - New production reactors working group (DOE)
 - Thirty-nine technical presentations at national, international, or information meetings
 - Twenty-seven reports prepared and distributed (six pending)
 - Twenty papers prepared for domestic and international conferences (five pending)
 - One journal article published (six pending)
 - Established informal technology exchange with four international technical organizations
 - Federation Internationale de la Precontrainte (FIP)
 - Int'l Union of Testing and Research Laboratories for Materials and Structures (RILEM)
 - Int'l Council for Building Research Studies and Documentation (CIB)
 - Commission of European Communities (CEC)
 - Contacted 100 foreign organizations in 25 countries to pursue informal exchange of technology
 - Participated in technical society committee activities
 - American Concrete Institute (Service Life Prediction, Radioactive and Hazardous Waste Management, Concrete Materials Property Data Base, Fracture Mechanics, Composite Concrete/Steel High-Pressure Vessels)
 - American Society of Mechanical Engineers (Section XI Working Group on Plant Life Extension, Working Group on Concrete Pressure Components, Subgroup on Containments)
 - RILEM (Damage Classification of Concrete Structures, Techniques for Prediction of Service Life)
 - FIP (Commission on Prestressed Concrete Pressure Vessels and Containments: Working Group 2 - Containment Structures and Working Group 3 - Non-nuclear PCPVs)
 - Contacted 100 domestic organizations: governmental (17), universities (17), national organizations (21), consulting/engineering groups (31), and utility organizations/utilities (14) to pursue exchange of information
 - Completed seven foreign trips to coordinate activities and exchange technology
 - Presented lectures on "Concrete Material Systems" and "Containments for Nuclear Power Plants" at IAEA Interregional Training Course on Safety Aspects of Aging and Maintenance in Nuclear Power Plant Operation
 - Completed draft proposal for participation in IAEA co-ordinated research program on Management of Aging of Concrete Containment Building

Task S.2 – Materials Property Data Base

- **Structural Materials Information Center Formulation/Implementation – Subtask S.2.1**
 - Completed review and assessment of domestic and foreign materials property data bases (ORNL/NRC/LTR-89/3)
 - Completed plan for development of Structural Materials Information Center (SMIC) (ORNL/NRC/LTR-89/8)
 - Implemented plan and provided first issue of SMIC (ORNL/NRC/LTR-90/22)
 - Selected software for electronic data base and formatted to handle composite material systems, i.e., concrete
 - Developed format for Structural Materials Handbook and Structural Materials Electronic Data Base
 - Established data and information requirements for concrete, metallic reinforcement, prestressing tendon, and structural steel materials
 - Developed handbook pages and electronic data base files for 116 concrete-related materials (102 portland cement concretes, 12 metallic reinforcements, 1 prestressing tendon, and 1 structural steel) and entered into the Structural Materials Information Center
 - Expanded SMIC to include nonconcrete materials (rubber, 1 entry)
 - Completed report on SMIC and potential applications (ORNL/NRC/LTR-92/8)
- **Data Collection – Subtask S.2.2**
 - Completed procurement and testing of aged concrete samples from three nuclear facilities
 - Procured concrete cores from five additional nuclear facilities and prepared specimens for testing
 - Completed report presenting results of five-year compressive strength testing of variable fly ash content high-strength concretes (ORNL/NRC/LTR-89/12)
 - Completed subcontracts with Construction Technology Laboratories, Inc., to provide 30-year data from long-term study of performance of cement in concrete (ORNL/NRC/LTR-91/26) and testing additional specimens (t > 35 years) (ORNL/NRC/LTR-92/26)
 - Implemented subcontract with Sargent & Lundy Engineers to provide baseline material property information on concrete materials utilized in Commonwealth Edison Companies (CECo) nuclear power plants and to assist in setting up a material sampling program with CECo
 - Completed draft report presenting results of structural concrete condition surveys conducted at several United Kingdom nuclear power stations (Taywood Engineering Subcontract Report 1303/92/6163)
 - Completed draft report presenting results obtained from testing aged concretes obtained from several United Kingdom nuclear power stations (Taywood Engineering Subcontract Report 1303/92/6037)
 - Completed draft report presenting results obtained from surveillance data for several United Kingdom nuclear power stations (Taywood Engineering Subcontract Report 1322/92/5957)

Task S.2 – Materials Property Data Base (cont.)

- **Material Behavior Modeling – Subtask S.2.3**
 - Exchanged technology with National Institute of Standards and Technology (NIST) on modeling of concrete material behavior and permeability studies
 - Exchanged information with Technical Research Center of Finland, Taywood Engineering Laboratories, British Cement Association, British Research Establishment, AEA Technology, CEB, RILEM, University of Toronto, and Danish Road Directorate on concrete modeling and service life prediction
 - Exchanged information with CIB/RILEM and ACI committees on service life prediction and damage classification of concrete structures
 - Participated in CANMET/ACI Conference on Durability of Concrete (Montreal) and Fifth International Conference on Durability of Building Materials and Components (Brighton)
 - Published NIST report evaluating models for use in predicting remaining service life of concrete and reviewing accelerated aging techniques and tests (NISTIR 4712)

Task S.3 – Structural Component Assessment/Repair Technology

- **LWR Critical Concrete Component Classification – Subtask S.3.1**
 - Completed development of concrete component classification and rating system
 - Completed development of degradation factor classification and rating system
 - Published report on structural aging assessment methodology for concrete structures in nuclear power plants (ORNL/NRC/LTR-90/17)
- **NDE/Sampling Inspection Technology – Subtask S.3.2**
 - Published report to review and assess inservice inspection techniques and methodologies for application to concrete structures in nuclear power plants (ORNL/NRC/LTR-90/29)
 - Completed report reviewing and assessing nondestructive methods relative to provision of statistical data for use with condition assessment and reliability-based life prediction models (NISTIR 4874)
 - Participated in government- and industry-sponsored short courses and seminars: Nondestructive Testing of Concrete Materials and Structures (University of Wisconsin - Milwaukee), Workshop on Nondestructive Evaluation for Performance of Civil Structures (NSF), CBT Symposium on Facility Diagnostics (NIST), Nondestructive Testing of Civil Structures and Materials (NSF), and Diagnosis of Concrete Structures (Bratislava)
- **Remedial/Preventative Measures Considerations – Subtask S.3.3**
 - Participated in short courses and seminars: Structural Repair of Concrete and Masonry (ASCE), and Concrete Durability and Repair (Michigan State University)
 - Contacted 10 domestic and 11 foreign research establishments to obtain information on repair procedures and techniques to evaluate the performance of repair materials
 - Implemented subcontract with Wiss, Janney, Elstner Associates and to provide an assessment of repair procedures that have been utilized for concrete materials and structures
 - Completed first phase of subcontract with Howard University to develop damage assessment and repair prioritization methodologies
 - Completed draft report on assessment of European repair practices for corrosion damaged reinforced concrete (Taywood Engineering Subcontract Report 1303/91/5328)
 - Completed draft report on corrosion of metals either embedded or in contact with concrete, potential for stray electrical current-induced corrosion in nuclear power plants and assessment of performance of cathodic protection systems (ORNL/NRC/LTR-93/0X)

Task S.4 – Quantitative Methodology for Continued Service Considerations

- Completed discussions with recognized experts on subject of reliability-based techniques for current and future condition assessments of concrete structures
- Implemented subcontract with the Johns Hopkins University to develop a methodology for use in performing current condition assessments and reliability-based life predictions for concrete structures in nuclear power plants
 - Completed review of literature on methods for time-dependent reliability analysis
 - Developed probabilistic approaches for assessing time-dependent reliability and deterioration of reinforced concrete structural components subjected to stochastic load events
 - Completed initial framework for a probability-based method to evaluate time-dependent reliability of a single or series of structural components
 - Completed report describing development of a probabilistic method for condition assessment and reliability-based life prediction of concrete structures (ORNL/NRC/LTR-92/4)
 - Completed draft report presenting methodology to facilitate quantitative assessments of current and future structural reliability and performance of concrete structures in nuclear power plants and applied it to provide a basis for selecting appropriate periods for continued service and determining optimum intervals and extent of inspection and maintenance (ORNL/NRC/LTR-93/1)
- Completed peer review board meeting

APPENDIX E

LIST OF TECHNICAL REPORTS AND PAPERS PREPARED
UNDER THE SAG PROGRAM

1. D. J. Naus, *Concrete Component Aging and Its Significance Relative to Life Extension of Nuclear Power Plants*, NUREG/CR-4652 (ORNL/TM-10059), Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., Oak Ridge, Tenn., September 1986.*
2. D. J. Naus, "Aging of Concrete Components and Its Significance Relative to Life Extension of Nuclear Power Plants," pp. 229-234, in *Trans. of the 9th Int'l. Conf. on Str. Mech. in Reactor Tech.*, Paper D5/1, Lausanne, Switzerland, A. A. Belkema Publisher, August 1987.*
3. D. J. Naus and C. E. Pugh, *Report of Foreign Travel of C. E. Pugh and D. J. Naus, Engineering Technology Division*, ORNL/FTR-2694, August 8-26, 1987 (September 21, 1987).*
4. D. J. Naus, M. F. Marchbanks, and E. G. Arndt, "Evaluation of Aged Concrete Structures for Continued Service in Nuclear Power Plants," pp. 57-67, in *Proc. of Topical Meeting on Nuclear Power Plant Life Extension*, July 31-August 3, 1988, Snowbird, Utah, Session 2, Paper 1, American Nuclear Society, 1988.
5. D. J. Naus, M. F. Marchbanks, C. B. Oland, and E. G. Arndt, *Structural Aging Program Five-Year Plan: FY 1988-1992*, ORNL/NRC/LTR-89/1, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., Oak Ridge, Tenn., March 1989.
6. M. F. Marchbanks, *A Review and Assessment of Materials Property Databases with Particular Reference to Concrete Material Systems*, ORNL/NRC/LTR-89/3, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., Oak Ridge, Tenn., March 1989.
7. D. J. Naus, M. F. Marchbanks, and E. G. Arndt, "Evaluation of Aged Concrete Structures for Continued Service in Nuclear Power Plants," *Proceedings of the United States Nuclear Regulatory Commission Sixteenth Water Reactor Safety Information Meeting* held at National Institute of Standards and Technology, Gaithersburg, Md., NUREG/CP-0097, Vol. 3, *Structural and Seismic Engineering*, March 1989.
8. D. J. Naus, M. F. Marchbanks, C. B. Oland, and E. G. Arndt, "Structural Aging Program to Assess Adequacy of Critical Concrete Components in Nuclear Power Plants," pp. 109-118, in *Transactions of the 10th International Conference on Structural Mechanics in Reactor Technology*, Session D, Paper 122, Anaheim, Calif., August 1989.

* Prepared under NPAR Program.

9. C. B. Oland, M. F. Marchbanks, and D. J. Naus, *Plan for Use in Development of the Structural Materials Information Center*, ORNL/NRC/LTR-89/8, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., Oak Ridge, Tenn., September 1989.
10. D. J. Naus and C. B. Oland, *Five-Year Compressive Strength Test Results for Moist-Cured and Sealed High-Strength Variable Fly Ash Content Concretes*, ORNL/NRC/LTR-89/12, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., Oak Ridge, Tenn., November 1989.
11. D. J. Naus, M. F. Marchbanks, C. B. Oland, E. G. Arndt, and T. M. Brown, "Considerations in the Evaluation of Concrete Structures for Continued Service in Aged Nuclear Power Plants," *Proceedings of the American Power Conference*, Volume 51, pp. 827-32, Illinois Institute of Technology, Chicago, Ill., 1989.
12. D. J. Naus, C. B. Oland, M. F. Marchbanks and E. G. Arndt, *Structural Aging (SAG) Program Five-Year Plan: FY 1989 - 1993*, ORNL/NRC/LTR-89/15, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., Oak Ridge, Tenn., December 1989.
13. D. J. Naus, C. B. Oland, and M. F. Marchbanks, *Structural Aging Program Annual Technical Progress Report for Period October 1, 1988, to September 30, 1989 (FY 1989)*, ORNL/NRC/LTR-90/1, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., Oak Ridge, Tenn., January 1990.
14. C. B. Oland, *Report of Canadian Travel of C. B. Oland, Pressure Vessel Technology Section, Engineering Technology Division*, Letter Report to E. G. Arndt, U. S. Nuclear Regulatory Commission, Rockville, MD, from C. B. Oland, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., Oak Ridge, Tenn., April 10, 1990.
15. D. J. Naus, *Report of Foreign Travel of D. J. Naus, Engineering Technology Division, June 9-23, 1990*, ORNL/FTR-3641, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., Oak Ridge, Tenn., July 9, 1990.
16. C. B. Oland and D. J. Naus, *Structural Materials Information Center for Presentation of the Time Variation of Materials Properties*, ORNL/NRC/LTR-90/22, Martin Marietta Energy Systems, Inc., Oak Ridge Nat'l Lab. Oak Ridge, Tenn., November 1990.
17. D. J. Naus, *Report of Foreign Travel of D. J. Naus, Engineering Technology Division, November 3-15, 1990*, ORNL/FTR-3827, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., Oak Ridge, Tenn., November 27, 1990.
18. C. J. Hookham, *Structural Aging Assessment Methodology for Concrete Structures in Nuclear Power Plants*, ORNL/NRC/LTR-90/17 (Subcontract Report 11X-SD343V from Multiple Dynamics Corporation, Southfield, Michigan), Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., Oak Ridge, Tenn., March 1991.

19. D. J. Naus and C. B. Oland, *Structural Aging Program Technical Progress Report for Period October 1, 1989, to December 31, 1990*, ORNL/NRC/LTR-91/2, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., Oak Ridge, Tenn., March 1991.
20. D. J. Naus, C. B. Oland and E. G. Arndt, "Management of the Aging of Critical Safety-Related Concrete Structures in Light-Water Reactor Plants," *Proceedings of United States Nuclear Regulatory Commission Eighteenth Water Reactor Safety Information Meeting* held at Holiday Inn - Crowne Plaza, Rockville, Md., NUREG/CP-0114, Vol. 1 - *Structural and Seismic Engineering*, pp. 527-552, April 1991.
21. C. B. Oland and D. J. Naus, "Development of the Structural Materials Information Center," *Proceedings of United States Nuclear Regulatory Commission Eighteenth Water Reactor Safety Information Meeting* held at Holiday Inn - Crowne Plaza, Rockville, Md., NUREG/CP-0114, Vol. 1 - *Structural and Seismic Engineering*, pp. 579-596, April 1991.
22. B. R. Ellingwood and Y. Mori, "Probabilistic Methods for Condition Assessment and Life Prediction of Concrete Structures in Nuclear Power Plants," *Proceedings of United States Nuclear Regulatory Commission Eighteenth Water Reactor Safety Information Meeting* held at Holiday Inn - Crowne Plaza, Rockville, Maryland, NUREG/CP-0114, Vol. 1 - *Structural and Seismic Engineering*, pp. 553-577, April 1991.
23. D. J. Naus, C. B. Oland, and E. G. Arndt, "Aging Management of Safety-Related Concrete Structures in Nuclear Power Plants," *Technical Sessions on Nuclear Plant Systems/Components Aging Management and Life Extension*, ASME Pressure Vessel and Piping Conference 1991, ASME PVP-Vol. 208, pp. 1-7, San Diego, Calif., June 23-27, 1991.
24. D. J. Naus, C. B. Oland and E. G. Arndt, "Aging Management of Safety-Related Concrete Structures to Provide Improved Bases for Continuing the Service of Nuclear Power Plants," *J. of Mat'ls and Struct.* 24 (142), pp. 308-316, The International Union of Testing and Research Laboratories for Materials and Structures, Chapman and Hall, New York, July 1991.
25. B. R. Ellingwood and Y. Mori, "Probabilistic Methods for Life Prediction of Concrete Structures in Nuclear Power Plants," *Transactions of the 11th International Conference on Structural Mechanics in Reactor Technology*, Tokyo, Japan, August 18-23, 1991.
26. C. B. Oland, *Report of Foreign Travel of C. B. Oland, Engineering Technology Division, August 4-16, 1991*, ORNL/FTR-4004, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., Oak Ridge, Tenn., August 30, 1991.

27. D. J. Naus, C. B. Oland and E. G. Arndt, "Aging of Concrete Structures in Nuclear Power Plants," M91-50997, *Minutes, RILEM TC-104 Damage Classification of Concrete Structures*, Technical University, Kosice, Czechoslovakia, August 30, 1991.
28. D. J. Naus, C. B. Oland and E. G. Arndt, "Aging of Concrete Structures in Nuclear Power Plants," *RILEM-IMEKO International Conference on Diagnosis of Concrete Structures*, EXPERTCENTRUM CZECHOSLOVAKIA, September 3-6, 1991, Slovak Technical University, Bratislava, Czechoslovakia, 1991.
29. T. M. Refai and M. K. Lim, *Inservice Inspection and Structural Integrity Assessment Methods for Nuclear Power Plant Concrete Structures*, ORNL/NRC/LTR-90/29 (Subcontract Report 17X-SE611V from Construction Technology Laboratories, Inc., Skokie, Illinois), Martin Marietta Energy Systems, Inc., Oak Ridge Nat'l Lab., Oak Ridge, Tenn., September 1991.
30. D. J. Naus, *Report of Foreign Travel of D. J. Naus, Engineering Technology Division, August 26-September 7, 1991*, ORNL/FTR-4029, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., Oak Ridge, Tenn., September 23, 1991.
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APPENDIX C

LIST OF TECHNICAL PRESENTATIONS
UNDER THE SAG PROGRAM

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