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Assessment of Age-Related Degradation of Structures and Passive Components for U.S. Nuclear Power Plants

Brookhaven National Laboratory

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
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ABSTRACT

This report describes the results of the first phase of a multi-year research program to assess age-related degradation of structures and passive components for U.S. nuclear power plants. The purpose of this research program is to develop the technical basis for the validation and improvement of analytical methods and acceptance criteria which can be used to make risk-informed decisions and to address technical issues related to degradation of structures and passive components. The approach adopted for this research program consists of three phases. The Phase I effort included collection and evaluation of plant degradation occurrences, an assessment of the available technical information on age-related degradation, and a scoping study to identify which structures and components should be studied in the subsequent phases of the research program. Based on the results of the Phase I effort, selected structures and passive components are evaluated in Phase II to assess the effects of age-related degradation using existing and enhanced analytical methods. Phase III will utilize the results of the analyses to develop recommendations to the NRC staff for making risk-informed decisions related to degradation of structures and passive components. This report presents the results of the Phase I portion of the research program.

The Phase I assessment of age-related degradation of structures and passive components at nuclear power plants has been completed. This assessment consisted of three activities. In the first activity, instances of age-related degradation have been collected and evaluated. The data were collected from Licensee Event Reports, NRC generic communications, NUREG reports, and industry reports. A computerized database was developed to summarize important parameters which describe the applicable cases of degradation. Trending analyses were performed to identify which structures are most susceptible to age-related degradation, what are the most common aging mechanisms and aging effects, whether degradation occurrences are increasing, and other important observations. In the second activity, additional information such as NRC requirements/guidance, NRC programs, industry programs, degradation information from other countries, and other reports/papers on aging degradation were evaluated to identify the significant aging issues for those structures and passive components which would have the greatest impact on plant risk. In the third activity, the collection of degradation occurrences, trending analyses, available technical information, and risk significance of aging effects were utilized in a scoping study to identify those structures and passive components that warrant further detailed evaluation in Phase II of this program.

The scoping study concluded that the structures and passive components that warrant further detailed evaluation are masonry walls, flat bottom tanks, anchorages, concrete structures (other than containments) and buried piping. The focus of further research will be on developing and improving analytical methods to assess the effects of age-related degradation on the structural performance of structures and passive components, including fragility evaluations for probabilistic risk assessment and seismic margins assessment studies. The methodologies that will be developed could then be used to quantify the impact of age-related degradation of structures and passive components on overall plant risk. This would lead to greater confidence in the use of risk assessment as a tool for making risk-informed decisions for age-degraded structures and passive components.

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EXECUTIVE SUMMARY

As nuclear power plants age, degradation of structures, systems, and components can be expected to occur. The U.S. Nuclear Regulatory Commission (NRC) has sponsored programs in the past to address many concerns related to aging. However, most of these programs studied the effects of age-related degradation of active components. A better understanding of the behavior of structures and passive components is needed to ensure that the current licensing basis is maintained under all loading conditions throughout the life expectancy of a plant. The effect of age-related degradation is also important to ensure the safe operation of plants for the period of operation beyond 40 years for those plants that may apply for license renewal. This report surveys and evaluates age-related degradation occurrences of structures and passive components at nuclear power plants.

The objectives of this research program are (1) to develop the technical basis for the validation and improvement of analytical methods and acceptance criteria which can be used in making risk-informed decisions and (2) to address technical issues related to degradation of structures and passive components. To achieve this, a three-phased approach was adopted. Phase I consisted of (1) collection and analysis of degradation occurrences, (2) review of available technical information such as NRC and industry programs, NUREG reports, and other technical publications, and (3) a scoping study to identify those structures and passive components which should be studied in the subsequent phases of this research program. In Phase II, an assessment of the effects of age-related degradation and enhancement of analysis techniques to evaluate degradation will be performed. Phase III will provide recommendations to the NRC staff for making risk-informed decisions related to degradation of structures and passive components. This NUREG report presents the results of the Phase I research effort and sets the groundwork for the evaluation in the subsequent phases of the research program.

Section 2 of the report describes the process to collect and review degradation occurrences of structures and passive components at nuclear power plants. Instances of degradation occurrences were obtained from Licensee Event Reports, NRC generic correspondences, NUREG reports, and other documents. This information was tabulated and entered into a computerized database to permit sorting, searching, reporting, and evaluating the large amount of information. To determine what can be learned from this information, various trending analyses for degradation occurrences were performed. Trending analyses were developed for degradation occurrence distribution by types of structures/components, calendar years, age of plants, aging effects, aging mechanisms and other parameters.

Section 3 provides a review of the available technical information from existing NRC and industry programs on degradation of structures and passive components. Information from NRC programs and industry programs regarding inspection, testing, assessment, and repair techniques were identified and reviewed. Information related to aging/degradation mechanisms and effects on material properties was also reviewed. NRC requirements/guidance included in this review are 10 CFR 50, Appendix J (Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors); 10 CFR 50.55a (Codes and Standards); 10 CFR 50.65 (Requirements for

Monitoring the Effectiveness of Maintenance at Nuclear Power Plants); 10 CFR 54 (Requirements for Renewal of Operating Licenses for Nuclear Power Plants); Regulatory Guides; Draft Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants; and Generic Correspondences (IE Bulletins and Information Notices). NRC programs included in this review are the Nuclear Plant Aging Research (NPAR) Program, Structural Aging (SAG) Program, Nuclear Power Plant Generic Aging Lessons Learned (GALL), and Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures (NUREG-1522). Industry Reports included in this review are Nuclear Management and Resources Council (NUMARC) Industry Reports, Nuclear Energy Institute (NEI) – Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, American Concrete Institute (ACI) Codes and Standards, American Society of Civil Engineers Standard ASCE 11-90, and other sources of technical information.

Section 4 of this report describes the risk significance of aging effects for structures and passive components. An overview is presented of past probabilistic risk assessments (PRAs) with respect to the effects of age-related degradation of components and the ranking of components according to their risk significance. This is done in three steps. First, past internal event PRA studies on the effects of aging are reviewed regarding the key analysis methodologies and the estimated component ranking. Second, available seismic PRA studies, which addressed the aging of structures and passive components, are reviewed to single out the technical issues that may require further study. Lastly, based on a survey of a large number of past seismic PRAs, structures and components are identified that are potential dominant risk contributors. This information is used as input to the priority ranking of structures/components discussed in Section 5.

Section 5 describes the scoping study performed to identify the technology needs and to identify the important/critical structures and passive components, which should be reviewed in the Phase II scope-of-work. In order to gain an understanding of the technology needs and which structures and components require further assessments, a review was conducted of what NRC and industry programs exist for each structure and passive component and how well they are addressing aging degradation. To identify which structures and components warrant further review, the various structures and passive components were prioritized/ranked considering four key parameters. These parameters are seismic risk significance, number of degradation occurrences, importance to current licensing basis/license renewal, and adequacy of existing NRC and industry programs.

Section 6 summarizes the results of the scoping study and presents the conclusions reached from the Phase I effort and recommendations for performing the research in Phase II and Phase III. Based on the results of the scoping study, the structures and passive components which were ranked highest are masonry walls, flat bottom tanks, anchorages, reinforced concrete structures (other than containment), and buried piping. Therefore, it was concluded that the Phase II research effort should evaluate the effects of age-related degradation of structures and passive components from this selected group. The research effort in Phase II will include developing methods for performing fragility evaluations for Probabilistic Risk Assessment/Seismic Margins Assessment studies. This would lead to greater confidence in the use of risk assessment as a tool in making risk-informed decisions for age-degraded structures and components. The subsequent research effort would include (1) an evaluation and expansion, if necessary, of existing degradation condition assessment techniques, (2) performance of analytical structural evaluations of degraded structures and passive components utilizing methods such as linear or nonlinear finite element methods, (3) development of fragility curves for degraded structures and passive components and evaluation of their effect on overall plant risk, and (4) development of degradation acceptance criteria for structures and passive components based on the above activities, existing codes, standards, and other NRC or industry reports. The results of the Phase II effort will establish the technical bases for the formulation of recommendations during Phase III for regulatory guidance on the assessment of age-degraded structures.

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Some of the results presented in this report were developed by Dr. Young J. Park who died suddenly during the course of this project. Dr. Park made significant contributions to the evaluation of risk significance of aging effects for structures and passive components presented in Section 4 of this report and performed the preliminary study of concrete degradation described in Section 5.3.

The authors would like to thank Mr. D. E. Yielding, NRC Office for Analysis and Evaluation of Operational Data, and Mr. W. P. Poore, Nuclear Operations Analysis Center at Oak Ridge National Laboratory (ORNL), for their assistance in accessing and identifying degradation occurrences of structures and passive components contained in the Sequence Coding and Search System (SCSS). The SCSS is a computerized database for Licensee Event Reports maintained by ORNL for the NRC.

The authors would also like to express special thanks to Ms. S. Signorelli for her secretarial help throughout this program and in the preparation of this report.

1 INTRODUCTION

1.1 Objective

The objective of this research program is to assess the effects of age-related degradation of structures and passive components for U. S. Nuclear Power Plants. The technical basis will be developed for the validation and improvement of analytical methods and acceptance criteria which can be used to make risk-informed decisions and to address technical issues related to degradation of structures and passive components.

A three-phased approach was adopted for this project. Phase I consists of data collection, review of existing technical information, and a scoping study. Phase II consists of assessment of the effects of age-related degradation and improvement of available analysis techniques to evaluate degradation. Phase III consists of providing recommendations to the NRC staff for making risk-informed decisions and for resolving specific technical issues related to degradation of structures and passive components.

The purpose of this Phase I report is to describe the various activities, results, conclusions, and recommendations under the initial phase of the research program. The conclusions and recommendations described in this report identify which structures and components should be included in the subsequent phases of the research program and also present a detailed plan for achieving the stated objectives.

1.2 Background

At the end of 1996, there were 109 operating nuclear power plants (NPPs) in the United States producing approximately 75,000 megawatts of electric power generation. This represents about 22 percent of the Nation's total electric generation. Approximately two-thirds of the NPPs received their construction permit more than 25 years ago and the majority have been operating for 20 years or more. While the performance of safety-related structures and passive components at these plants has been good, the number of occurrences of age-related degradation has been increasing as NPPs age.

Numerous examples of age-related degradation of structures and passive components in NPPs are presented in NUREG-1522, "Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures." Much of the information was obtained from actual walk-downs of structures and components at six older NPPs (licensed before 1977). Instances of degradation were identified in intake structures/pump houses, service water piping, tendon galleries, masonry walls, anchorages, containments, and other concrete structures.

Structures generally have substantial safety margins when properly designed and constructed. However, the available margins for degraded structures are not well known. In addition, age-related degradation may affect the dynamic properties, structural response, structural resistance/capacity, failure mode, and location of failure initiation. A better understanding of the effect of aging degradation on structures and passive components is needed to ensure that the current licensing basis (CLB) is maintained under all loading conditions.

Results from risk evaluation programs conducted by the NRC, such as the Individual Plant Examination of External Events (IPEEE) program, indicate that external events such as earthquakes, high winds, and tornadoes can be significant contributors to core damage frequency (CDF). In some cases, structures and passive components have been found to be significant risk contributors when subjected to these external events. As structures and components age, the effect of age-related degradation will become a more significant factor in assessing risk.

1.3 Program Scope

The program scope covers structures and passive components normally found in nuclear power plants in the United States and not addressed by existing programs. Structures include buildings and civil engineering features such as masonry walls, canals, embankments, underground structures, and stacks. Passive components consist of equipment, which do not move or change their state to perform their intended function. Examples of passive components included in this program are tanks, cable tray systems, conduit systems, and HVAC ducts/supports. After initial review, passive components such as above ground piping, steam generators, reactor pressure vessels, and containments were removed from this study because of existing programs, which are addressing age-related degradation. A more complete definition of the specific structures and passive components included within the scope of this program is presented in Section 2.1 of this Report.

2 COLLECTION AND REVIEW OF DEGRADATION OCCURRENCES

The first part of the Phase I effort consisted of collecting and reviewing age-related degradation occurrences of structures and passive components at nuclear power plants. For the purpose of this research program the term "degradation occurrence" is defined as age-related degradation which was reported in NRC generic correspondences, Licensee Event Reports (LERs), NUREGs, and other referenced documents described below.

2.1 Structures and Passive Components Included

Structures and passive components within the scope of the NRC License Renewal Rule 10 CFR Part 54 and the Maintenance Rule 10 CFR 50.65 were considered for review in this research program. This includes structures and passive components: (1) that are safety-related, (2) whose failure could affect safety-related functions, and (3) that meet several other criteria defined within the scope of the license renewal rule and the maintenance rule.

All structures and components identified to be within the scope of review were placed into one of eighteen categories. A complete listing of the eighteen categories and the included structures and components is presented in Table 2-1. As an example, the category "anchorages" includes embedded anchors, expansion anchors, undercut anchors, drop-in anchors, embedded studs, and the grout beneath the baseplate.

Several items in Table 2-1 (identified with a double asterisk) were removed from further review. This applies to penetrations; electrical conductors; piping (above ground); tubing; and pipe-insulation, fittings, and sleeves. In the early stages of the review effort, other programs were identified which are addressing age-related degradation of these items.

Additional structures and components were subsequently eliminated because after further research, other industry and/or NRC programs were identified which are addressing degradation concerns for these structures/components. This is discussed further in Section 5.2.

2.2 Sources of Information

Various sources were investigated to identify instances of age-related degradation of structures and passive components. These sources primarily consist of LERs, NRC generic correspondence, NUREGs, and industry reports.

The NRC generic correspondence includes IE Bulletins, Generic Letters, and Information Notices. All of the correspondence contained in the Generic Correspondence Library on the Fedworld Information Network (Internet) was investigated. This was done by reviewing all of the generic correspondence titles. Those that apply to structures and passive components or those that may be related in some manner were identified and retrieved for review. If instances of age-related degradation were noted then that occurrence was recorded for use in this research program.

The LERs were obtained from the Sequence Coding & Search System (SCSS) maintained by the Oak Ridge National Laboratory (ORNL) for the NRC. The SCSS database was developed by the NRC's Office for Analysis and Evaluation of Operating Data through the Nuclear Operations Analysis Center at ORNL. The SCSS is an electronic database developed to allow users to retrieve commercial nuclear plant operating experience data from LERs. The database contains over 35,000 LERs from 1980 to the present time.

Instead of providing the actual LER text, the database reduced the LER descriptive text to coded, searchable sequences. It captures the components, system, effects on the plant unit, as well as personnel errors reported in LERs. For each LER, data on component failures include type and number of components involved, system to which components belong, cause and mode of failure, effect of failures on plant systems and unit, and component vendor and model data (if given in the LER). This information is coded for use in searching specific information. For example, there are over 400 specific component codes and there are over 100 cause and effect code designations. In addition to the coded information, an abstract is available which provides a summary of the event.

In view of the very large number of LERs, it was decided to initially review LERs for the period 1990 to 1997. Then, the search was expanded to include LERs extending back to 1985. Thus the total period reviewed covered 1985 to 1997.

A sample printout for an LER obtained from the SCSS database is shown in Figure 2-1. The event was a broken two-inch conduit fitting located near a wall in the service water pump room. As reported in the abstract, the broken conduit fitting was caused by corrosion due to exposure to the salt-water marine environment.

2.3 Degradation Occurrence Database

In order to document and evaluate the enormous amount of data, a computerized database, entitled Degradation Occurrence Database (DOD), was created. The DOD was prepared using the Microsoft database management program "Access". The advantages of this computerized database are: 1) simple entry and update of degradation data, 2), sorting and organizing of data in a meaningful way, 3) quickly finding desired information, 4) creation of tabulated listings or reports, and 5) sharing of data with other authorized users and programs in the system.

A number of tables were created as part of the DOD to fully describe the age-related degradation of structures and passive components. The various tables that were developed are described below. The complete set of tables is contained in Appendix A to this report. Representative copies of some pages from these tables are included in this section as noted below in order to explain the development and content of the tables.

	Table No.
1. Structures and Passive Components	2-1
2. Degradation Occurrence Table (Sample)	2-2
3. Aging Effects and Mechanisms	2-3
4. System Definition Codes	2-4
5. Stress Corrosion Codes	2-5

The Structures and Passive Components Table identifies the various types of structures and passive components included in the scope-of-work. The structures and components that are within the initial scope-of-work were described in Section 2.1 and a detailed listing is also presented in Table 2-1.

The Degradation Occurrence Table (DOT) contains all of the degradation occurrences identified as applicable under this research program. A total of 492 degradation occurrences were included in the DOT. It should be noted that there are certainly many more occurrences of degradation than what were identified and reported in this DOT. However, if they were not reported in LERs or other publicly available documents then they would not be included in this database. For example, some degradation occurrences may not be reported in LERs if the event or condition does not seriously affect the plant or result in an unanalyzed condition that significantly compromised plant safety.

A representative copy of one page from the DOT is shown in Table 2-2. For each occurrence the following type of information is provided in the DOT:

1. Component
2. Subcomponent
3. System
4. Aging Effect
5. Aging Mechanism
6. Plant
7. Month, Day, Year
8. How Identified
9. Evaluation Method
10. Repair Method
11. Docket No.
12. Reference Document
13. Reference Document No.

The "Component" and "Subcomponent" entries identify the type of structure or passive component as listed in Table 2-1. The "System" refers to the plant system such as service water system or containment system. Since many acronyms are used, another table entitled System Definition Codes was prepared and is presented in Table 2-4.

The "Aging Effect" and corresponding "Aging Mechanism" entries are obtained from the reference document (LER, NRC generic correspondence, etc.). As with any of the other entries in the database, if the required information is not given or is insufficient then an "NA", meaning not available, is noted. A listing of the various types of aging effects and aging mechanisms used in the DOT is shown on another table entitled Aging Effects and Mechanisms (Table 2-3).

Table 2-3 lists the aging effects and aging mechanisms separately for concrete, steel, and "other" (e.g. seals, coatings, insulation) materials. Table 2-3 is not intended to be a complete listing of all possible aging effects and mechanisms but rather, those aging effects and mechanisms for the occurrences presented in the DOT. The entries for aging effects and aging mechanisms are listed next to each other and are not intended to suggest which aging effect is caused by which aging mechanism. Such information is available in NUREG-1557.

For the aging effect of cracking in steel, specific types of aging mechanisms are sometimes given in the referenced documents. Examples of this are stress corrosion cracking (SCC) and hydrogen stress corrosion cracking (HSC). A definition for these types of stress corrosion codes is given in the table entitled Stress Corrosion Codes (see Table 2-5).

The "Plant" entry identifies the nuclear power plant where the age-related degradation occurred. All nuclear power plants were included; operating, shutdown, and plants that have been or are going through a decommissioning process. Where degradation occurrences of some foreign plants were identified in the literature, they were also included in the DOT.

The entry for the "Month," "Date," and "Year" corresponds to the date that the degradation occurrence was identified. This is normally given in the referenced document. When this information is not available, then the date used in the table corresponds to the date of the reference, which described the occurrence. When this occurred an asterisk was placed next to the year entry in the table to indicate that an exact date for the occurrence was not available and so the date of the publication is presented.

The next three columns in the DOT describe how the degradation was identified, evaluated, and repaired. Identification methods include visual, inspection, leaking, alarm, test, low flow and other methods. The evaluation methods consist of how the degradation was investigated/reviewed. These methods generally consisted of visual examinations; tests such as leak rate tests, ultrasonic tests, eddy current tests; and engineering judgements. For repair methods, designations such as repair, replacement, monitoring, tightening, and cleaning were noted. It should be noted that for these three headings in the DOT, apparently many of the referenced documents did not have sufficient information so that NA was denoted in the corresponding box in the table.

The next column in the DOT provides the docket number, which is the unique NRC assigned number for each of the plants, even sites with multiple units. Following the docket number is the reference document name/type and the specific document number.

As described in Section 2.2, data were obtained by identifying and reviewing LERs, NRC generic correspondence, NUREGs, and industry reports. After evaluating each degradation occurrence, the information was entered into the DOD. The analysis of the data and observations that can be derived from this data are described in the next section.

2.4 Analysis of Degradation Trends

A total of 492 degradation occurrences were identified related to structures and passive components. Using the DOD, a tabulation of the total number of degradation occurrences for each structure/component category was made. The results of this tabulation are shown in Table 2-1.

Since all of the data have been entered into a computerized database program, the information can also be searched, sorted, and tabulated in any order or form. For example, the degradation occurrences can be easily sorted by types of components, types of degradation, causes of degradation, plant names, dates, or systems. To evaluate the degradation occurrences, the data were sorted to obtain trending information. Trending data were developed for the following types of distribution:

TRENDING DATA - DISTRIBUTION BY	FIGURE NO.
1. Components/Sub-components	2-2
2. Years (1985-1997)	2-3
3. Age of Plants	2-4
4. Steel Degradation Aging Effects	2-5
5. Concrete Degradation Aging Effects	2-6
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7. Types of Cracking Induced by Corrosion	2-8
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11. Subcomponents for Filters	2-12
12. Subcomponents for RPV	2-13
13. Systems	2-14
14. Methods of Identification	2-15

The distribution of degradation by types of components/subcomponents (Figure 2-2) was obtained by compiling the number of occurrences for each of the components listed in Table 2-1. Where a subcomponent had an extremely large number of occurrences such as piping and steam generators, it was included as a separate item on the bar chart in Figure 2-1. Where a component had no occurrences identified such as structural seismic gap and vessels (other than steam generators) it was not included on the bar chart.

From this distribution of degradation by types of components/subcomponents, it is evident that piping & tubing, steam generators, RPV, and containments have the largest number of degradation occurrences. This is not surprising since it has been known in the industry that these structures and components have had numerous instances of degradation. Following these, the structures and passive components with the greatest number of occurrences in descending order are filters, concrete, structural steel, heat exchangers, piping supports, tanks, pressurizers, electrical conductors, and anchorages. All of the remaining items have six or less occurrences.

Since the NRC and industry have been studying and addressing the age-related degradation concerns related to piping, steam generators, RPV, and containments, it was decided after consultation with the NRC staff to eliminate these components from the subsequent phases of this research program.

The distribution of degradation by years (Figure 2-3) was developed by adding the number of occurrences in each of the years from 1985 through 1997. The bar corresponding to 1997 is lower than the others because LERs were only made available up to February 1997 when this compilation of occurrences was made. Looking at the rest of the bar chart it appears that with the exception of 1988 and 1994, there was a moderate increase from 1985 until 1991 and then the number of occurrences has remained constant at approximately 27 per year.

Figure 2-4 shows the distribution of degradation occurrences by age of plants. The graph represents the average number of occurrences per plant per year for different plant vintages. This was developed by categorizing all U.S. nuclear power plants by their age (1997 minus year of construction permit). Then the total number of occurrences for each group of plants in a given age category was divided by the number of plants in that age category and the age of the plants in that category. Although the actual number of occurrences are not high, this curve demonstrates that as the age of plants increase, the number of occurrences per plant per year also increases. Using the best fit curve, the actual number of occurrences per plant per year over a 14 year period (19 year to 33 year old plants) shows a growth more than three times (from about .065 to .24).

The distribution of degradation occurrences by steel aging effects is shown in Figure 2-5. The most predominant type of aging effect is cracking with 215 occurrences. Most of these cracking occurrences have been induced by some form of stress corrosion. The other aging effects consisting of loss of material, failure, wall thinning, plugging, and fouling have much fewer occurrences (below 60 occurrences each).

For concrete elements, the distribution by degradation aging effects is presented in Figure 2-6. The major aging effects for concrete degradation in descending order are cracking, spalling, general deterioration, and loss of material. Cracking was the most predominant with 30 occurrences, while the other aging effects had less than 7 occurrences each. As noted earlier there may have been other instances of degradation, however, if they were not interpreted to be severe, then the occurrences would not have been reported.

Another distribution of interest is what caused most of the age-related degradation occurrences. Figure 2-7 shows the distribution of degradation occurrences by aging mechanisms. The predominant aging mechanisms in descending order are SCCs, corrosion (general or not identified as SCC), erosion, moisture, organisms, fatigue, chemical attack, foreign object, mechanical wear, and vibration. All of the remaining aging mechanisms had less than 11 occurrences.

Since there were so many occurrences of SCCs, a distribution of SCC by types was developed and presented in Figure 2-8. This figure shows that intergranular SCC is the most predominant cause, followed in descending order by primary water SCC, stress corrosion cracking (no specific type given), outside diameter SCC, intergranular attack/intergranular SCC, and hydrogen stress corrosion cracking.

The distribution of subcomponents for some of the significant structures and passive components are shown in Figures 2-9 through 2-13 for structural steel, concrete, containment, filters, and RPV, respectively. For structural steel, instances of degradation occurred most often with steel doors, liners, and spent fuel racks. For concrete elements, degradation occurrences were most predominant for masonry walls, concrete walls, concrete ceilings, and intake structures. For containment, the greatest number of occurrences were with the liners, prestressed systems, steel shell, and penetrations. For filters, the most occurrences were identified with the screens (typically travelling screens in the intake structures), strainers, and charcoal filters. For RPVs, the predominant occurrences were with the core shroud, jet pump assembly, core spray piping, and CRD. Additional details for each of these items can be found by reviewing the DOT provided in Appendix A.

To evaluate which plant systems have the most degradation occurrences; a distribution by systems was developed. Figure 2-14 presents the results, which show that the RCS by far has the most degradation occurrences followed by containment, feedwater, ERCSW, circulating water, RHR, and service water. The remaining systems all have less than 7 occurrences each. Although the RCS shows up as having many more occurrences (190 versus 54 for containment which follows it), part of the explanation may be that the RCS is more closely scrutinized and inspected than most of the other systems.

The last distribution that was developed is shown in Figure 2-15. This figure shows the distribution by methods of identification. The methods that identified the greatest number of degradation occurrences in descending order were inspection, visual, leaking fluid, NRC notification, test, preventive maintenance, and low or change in flow reading or annunciation. All the other methods of identification had less than 9 occurrences.

FORM 3 LER SCSS DATA 03-26-97

****

DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
293	1991	015	0	9108190070	222638	06/28/91

****

DOCKET:293 PILGRIM 1 TYPE:BWR
REGION: 1 NSSS:GE
ARCHITECTURAL ENGINEER: BECH
FACILITY OPERATOR: BOSTON EDISON CO.
SYMBOL: BEC

STEP	LK	SLK	CAUSE	PSYS	ISYS	COMP	VEND	QUAN	TR	CH	DI	T	P	D	EFF	PCC	CORR
1	0		SZ	PZ		PZ		1			1	M	T	R	UA	M	E
2	1		EF	KF	SP	CON		1	1		1	M	TR	K	DA		
3	2	X	CA	KF	SP	CON		1	1		1	M	TR	K	DA		
4	2		RC	KF	SP	CND		1	1		1	M	TR	K	AI		
5	4		RC	KF	SP	PND		1	1		1	M	T	K	AI		
6				XX								H	XX		YC		
7				YY								N	N		YC		

WATCH-LIST CODES FOR THIS LER ARE:
20 EQUIPMENT FAILURE

REPORTABILITY CODES FOR THIS LER ARE:
10 10 CFR 50.73(a)(2)(i): Shutdowns or technical specification violations.

REFERENCE LERS:
1 293/84-007 2 293/90-019 3 293/91-009

------STEP: 1
CAUSE:SZ --UNKNOWN HUMAN FACTOR CAUSE
PRIMARY SYSTEM:PZ --UNKNOWN ACTIVITY
COMPONENT:PZ --UNKNOWN PERSONNEL
EFFECT:UA --COMMISSION OF UNDESIRED TASK, ANALYSIS, OR STEP

------STEP: 2
THIS STEP IS DIRECTLY LINKED TO STEP 1
CAUSE:EF --CORROSION/ OXIDATION
PRIMARY SYSTEM:KF --FIRE PROTECTION
SECONDARY SYSTEM:SP --PUMPING STATIONS

COMPONENT:CON --CONNECTOR
EFFECT:DA --BREAK/ SHEAR

------STEP: 3
THIS STEP IS DIRECTLY LINKED TO STEP 2
SUBLINK:X
CAUSE:CA --MECHANICAL OVERLOAD
PRIMARY SYSTEM:KF --FIRE PROTECTION
SECONDARY SYSTEM:SP --PUMPING STATIONS
COMPONENT:CON --CONNECTOR
EFFECT:DA --BREAK/ SHEAR

Figure 2-1 Sample Printout of an LER from the SCSS Database

```
------STEP:  4
  THIS STEP IS DIRECTLY LINKED TO STEP  2
            CAUSE:RC  --RESULTANT COMPONENT FAULT
   PRIMARY SYSTEM:KF  --FIRE PROTECTION
 SECONDARY SYSTEM:SP  --PUMPING STATIONS
        COMPONENT:CND --CONDUIT
           EFFECT:AI  --OPEN

------STEP:  5
  THIS STEP IS DIRECTLY LINKED TO STEP  4
            CAUSE:RC  --RESULTANT COMPONENT FAULT
   PRIMARY SYSTEM:KF  --FIRE PROTECTION
 SECONDARY SYSTEM:SP  --PUMPING STATIONS
        COMPONENT:PND --PENETRATION, ELECTRICAL
           EFFECT:AI  --OPEN


INITIAL UNIT CONDITIONS: H    REFUELING
            UNIT EFFECT: XX   NO SIGNIFICANT EFFECT

  EFFECT ON ENVIRONMENT: N    NO RELEASE
    EFFECT ON PERSONNEL: N    NO EXPOSURE
```

ABSTRACT

POWER LEVEL - 000%. ON 6/28/91, AT 1506 HOURS DURING A REFUELING OUTAGE, THE NORTH WALL OF THE "A" TRAIN SALT SERVICE WATER (SSW) PUMP ROOM IN THE INTAKE STRUCTURE WAS FOUND BREACHED. THE BREACH CONSISTED OF A BROKEN TWO INCH CONDUIT FITTING LOCATED WHERE THE CONDUIT PENETRATES THE NORTH WALL OUTSIDE THE "A" TRAIN SSW PUMP ROOM. THE WALL IS A TS APPENDIX R FIRE BARRIER THAT SEPARATES THE SSW PUMP ROOM FROM THE SERVICE WATER PUMP FILTER ROOM. THIS CONDITION WAS DETERMINED TO BE REPORTABLE ON 7/11/91. THE BROKEN CONDUIT FITTING WAS A RESULT OF CORROSION DUE TO EXPOSURE TO A MARINE ENVIRONMENT. THE CONDUIT WAS IDENTIFIED AS BEING CORRODED IN 7/89. A FIRE BARRIER PENETRATION WALKDOWN CONDUCTED IN 12/89 FOUND THE BARRIER INTACT. SUBSEQUENT TO THAT INSPECTION IT IS POSTULATED THAT PERSONNEL USED THE CONDUIT AS A FOOT/HAND HOLD THAT EVENTUALLY BROKE THE CONDUIT FITTING. A CONTINUOUS FIREWATCH WAS IMMEDIATELY ESTABLISHED WHEN THE BREACH WAS IDENTIFIED. A FIRE SEAL WAS INSTALLED ON 7/25/91 AND THE FIRE WATCH WAS

DISCONTINUED. OTHER ACTIONS PLANNED INCLUDE: REPAIRING THE CONDUIT FITTING; PERFORMING A THOROUGH WALKDOWN OF THE INTAKE STRUCTURE; AND REVIEWING WALKDOWN PROCEDURES AND TRAINING TO IDENTIFY IMPROVEMENTS. THIS CONDITION WAS IDENTIFIED WITH THE REACTOR MODE SELECTOR SWITCH IN THE REFUEL POSITION. THE REACTOR VESSEL (RV) WATER TEMPERATURE WAS 83F AND THE RV PRESSURE WAS O PSIG.

Figure 2-1 Sample Printout of an LER from the SCSS Database (continued)

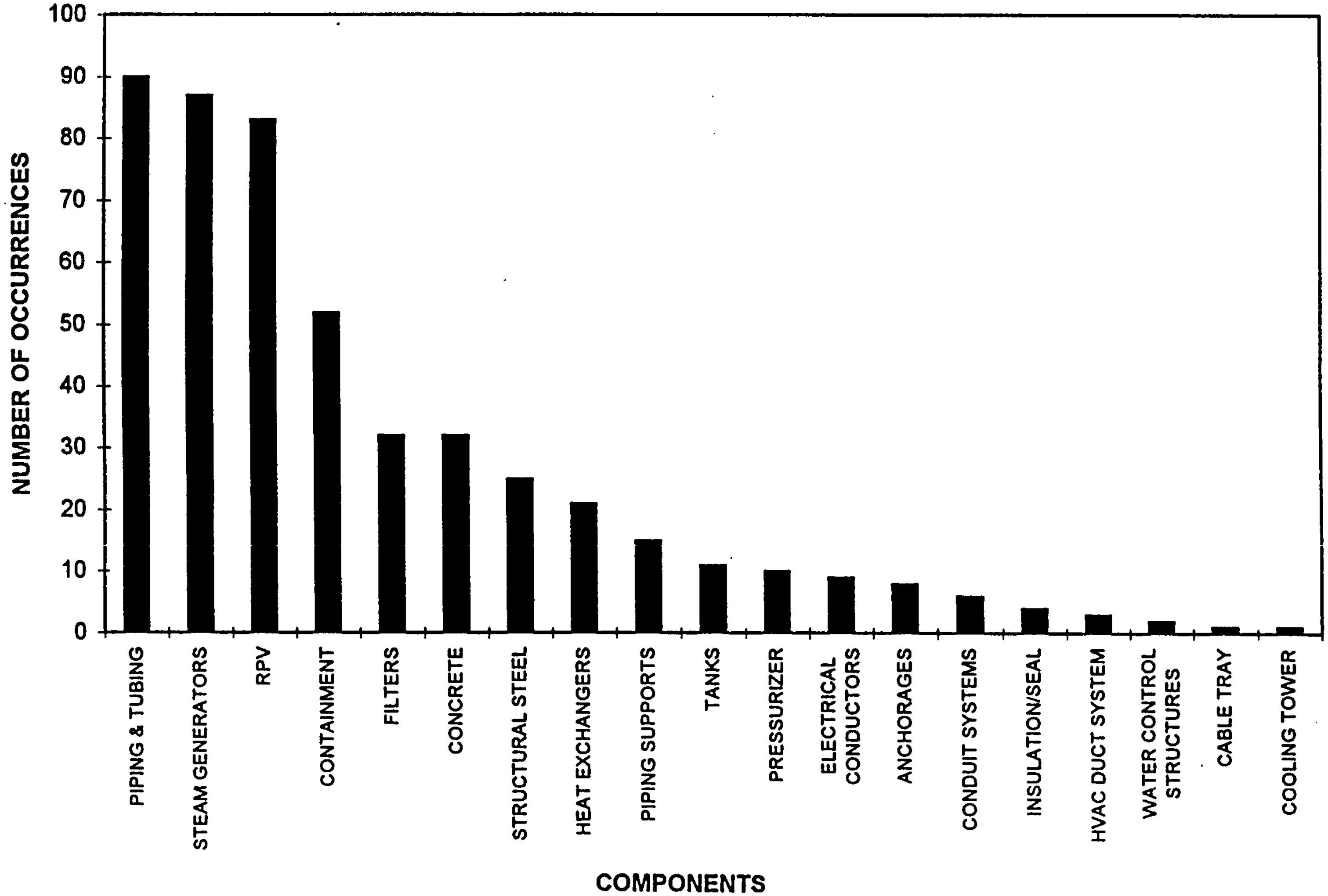

Figure 2-2 Passive Structures and Components - Degradation Occurrences Distribution By Components/Subcomponents

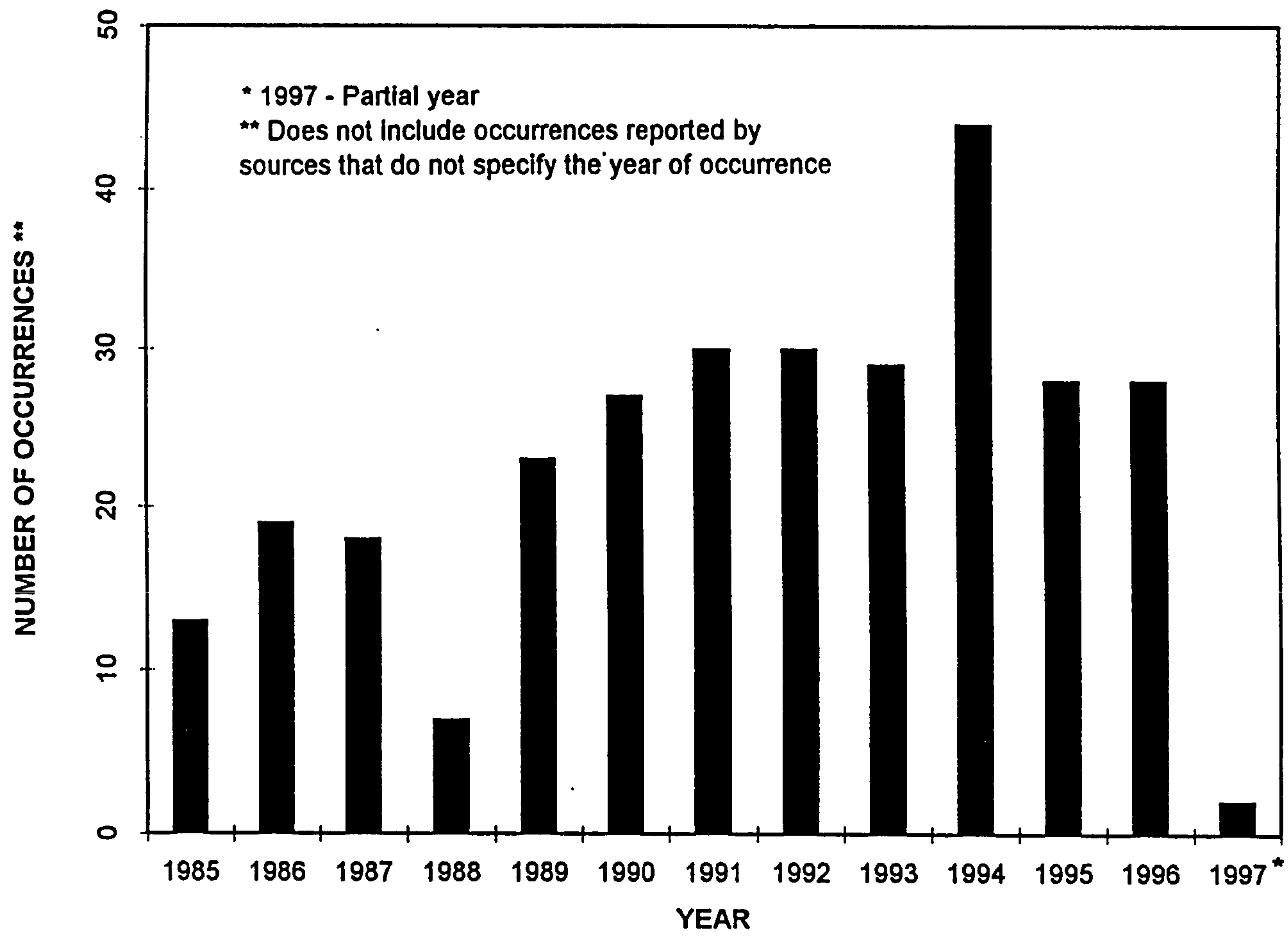

Figure 2-3 Passive Structures and Components - Degradation Occurrences Distribution By Years (1985-1997)

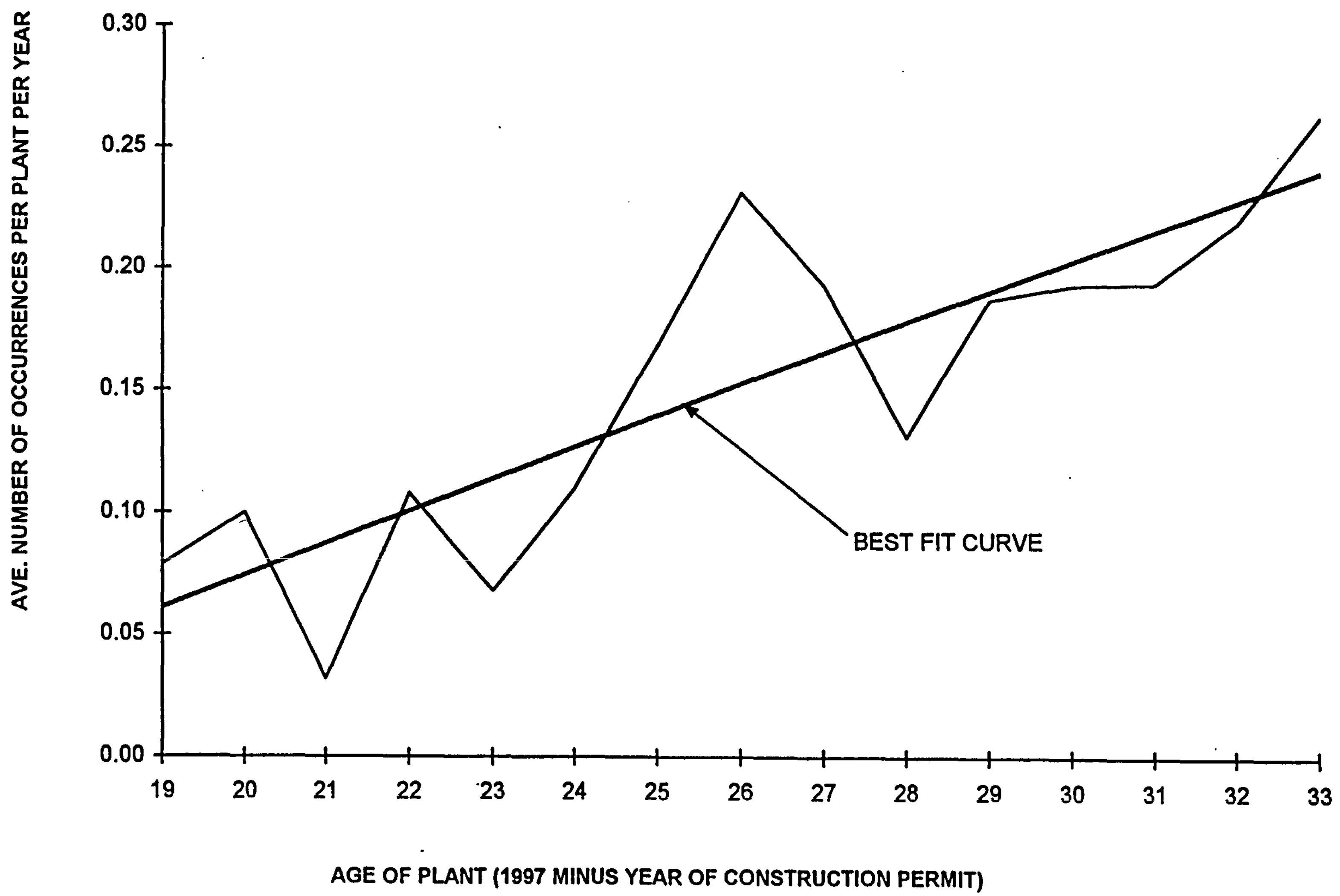

Figure 2-4 Passive Structures and Components - Degradation Occurrences Distribution By Age of Plants

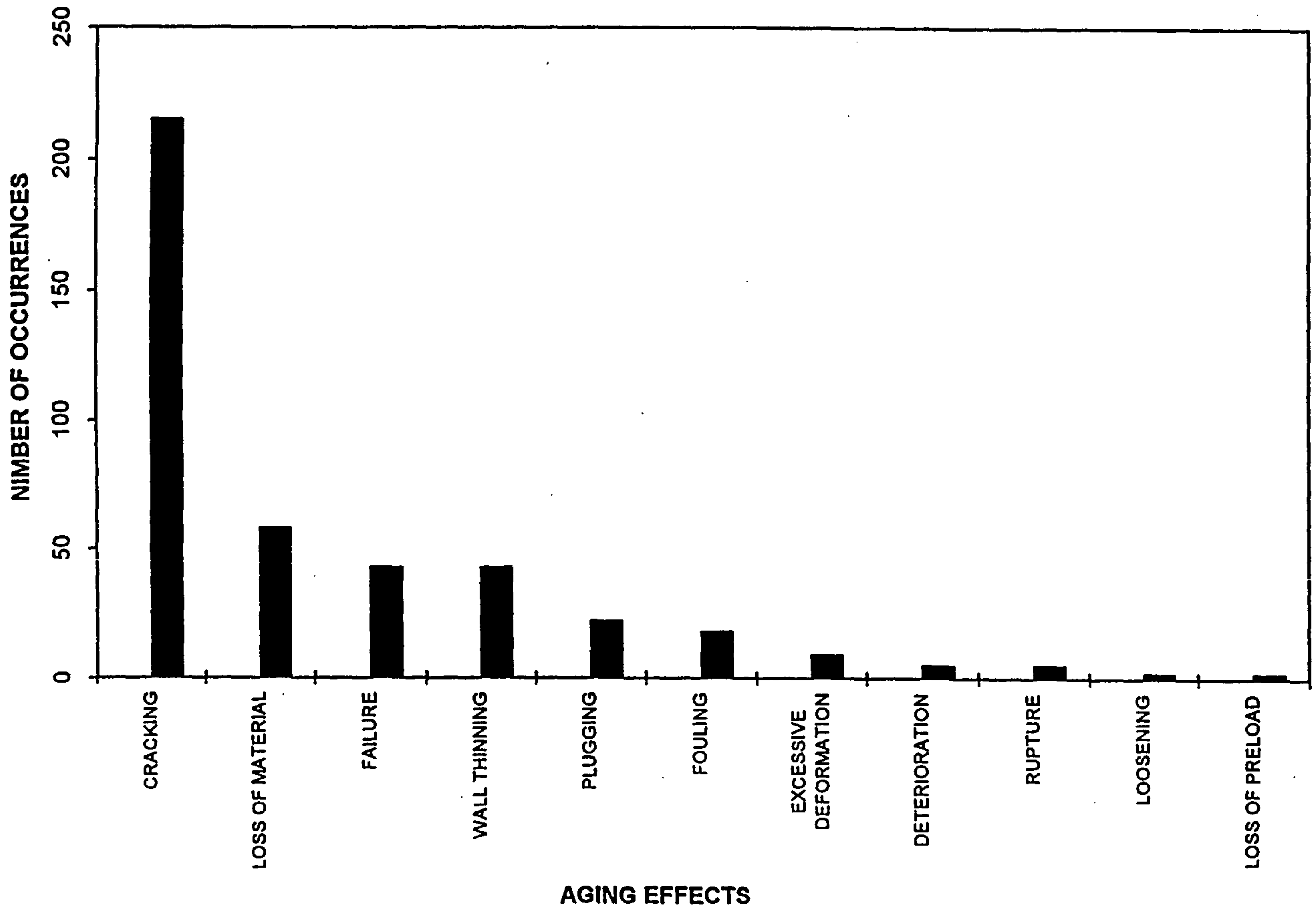

Figure 2-5 Passive Structures and Components - Degradation Occurrences
Steel Aging Effects

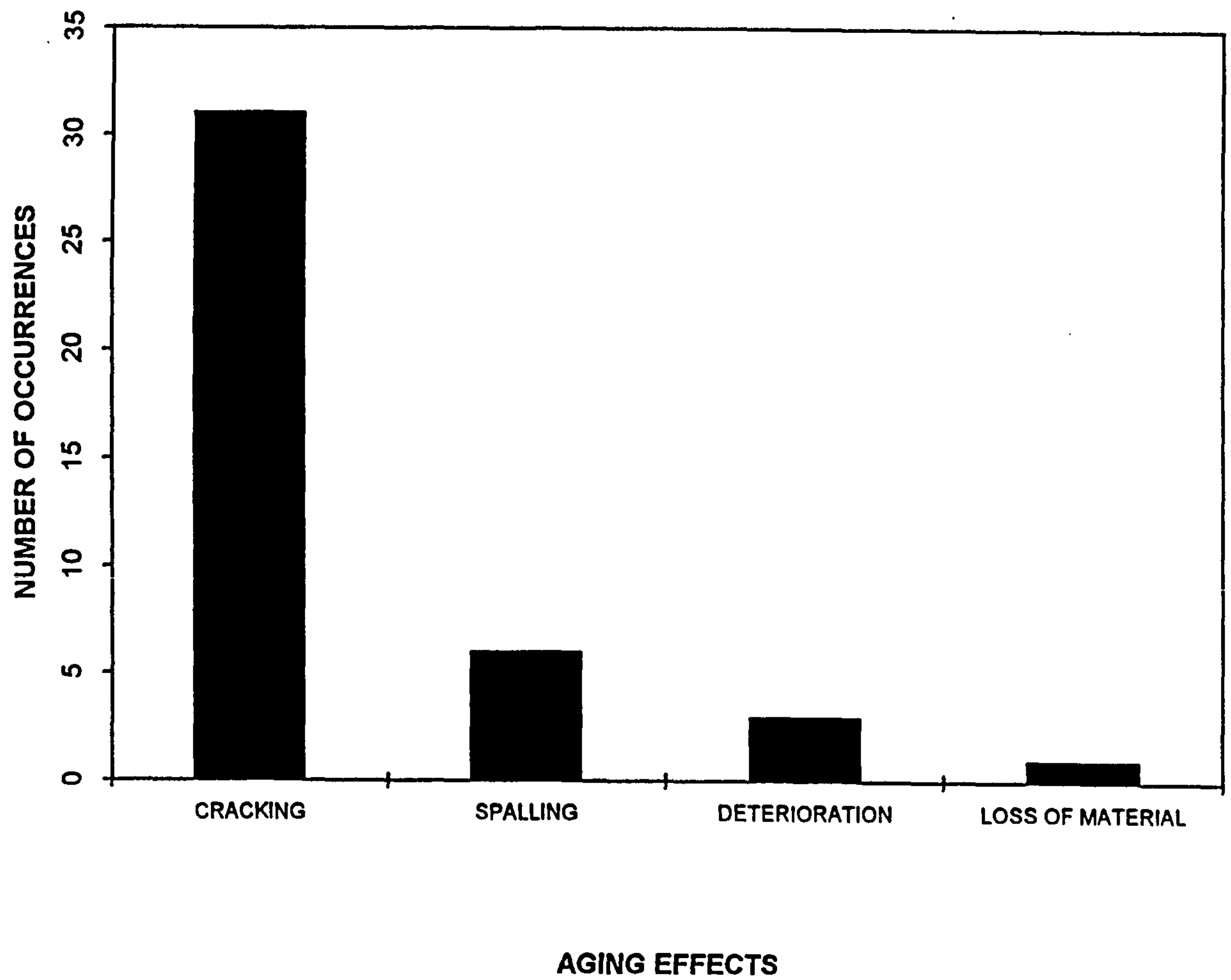

Figure 2-6 Passive Structures and Components - Degradation Occurrences
Concrete Aging Effects

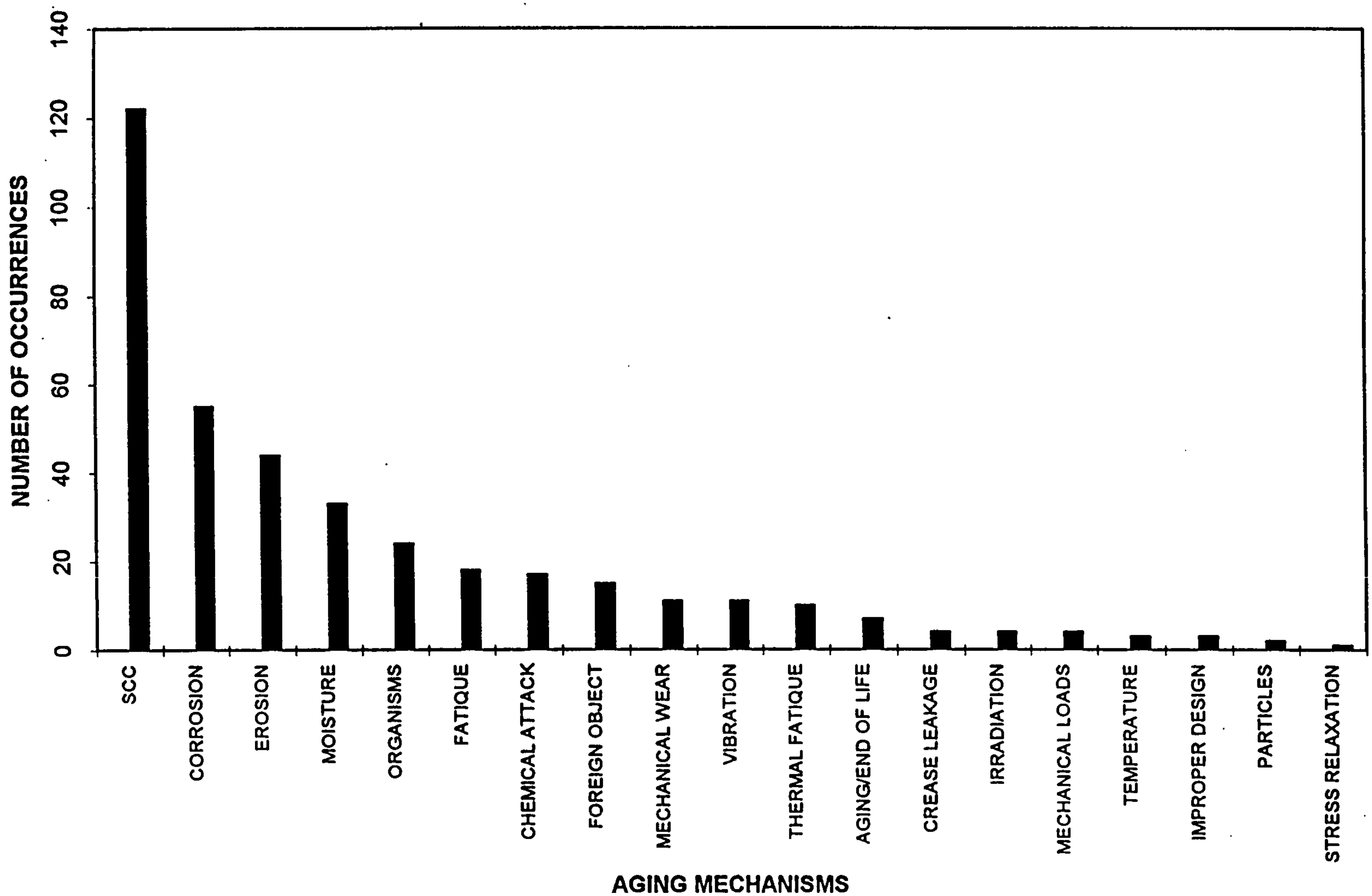

Figure 2-7 Passive Structures and Components - Degradation Occurrences Distribution By Aging Mechanisms of Degradation

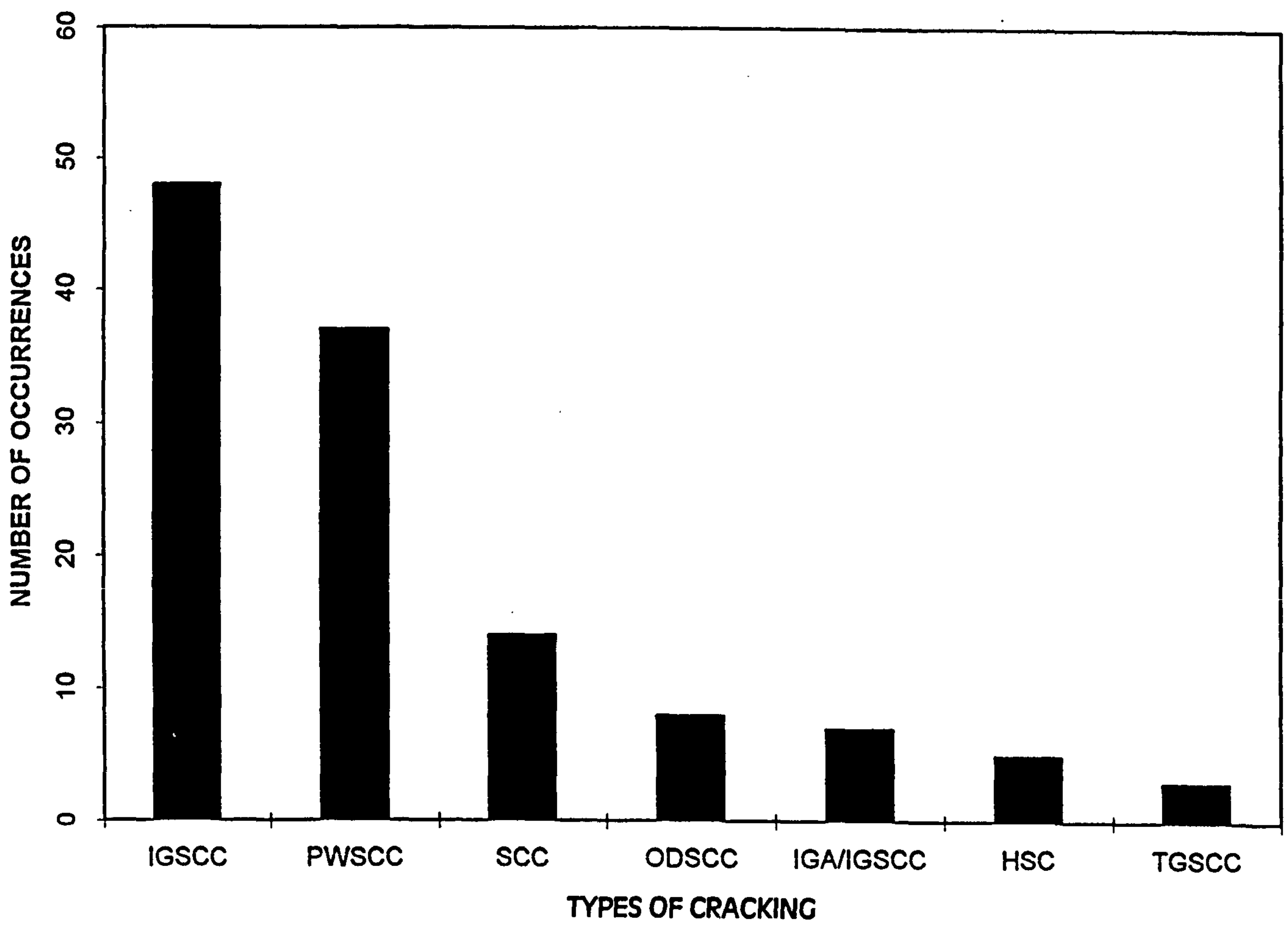

Figure 2-8 Passive Structures and Components - Degradation Occurrences Distribution By Type of Cracking Induced By Corrosion

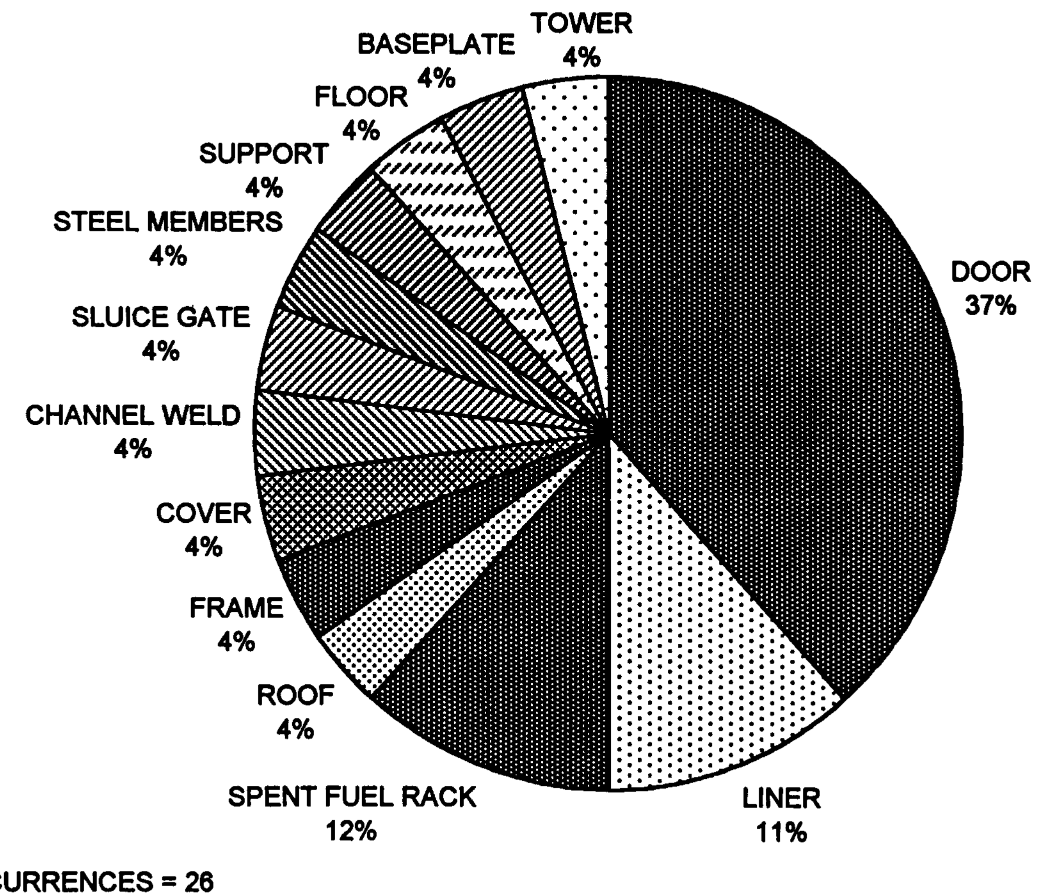

Figure 2-9 Passive Structures and Components - Degradation Occurrences Distribution of Subcomponents for Structural Steel

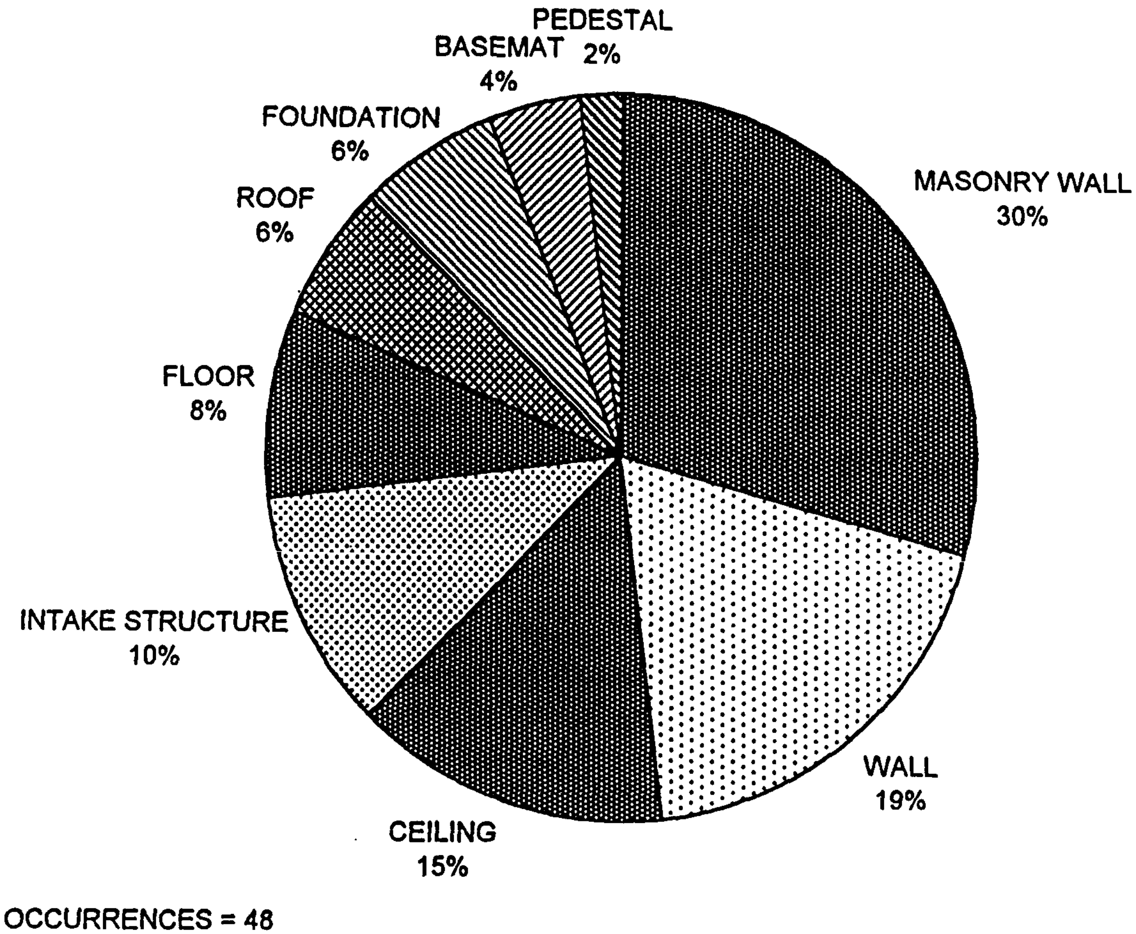

Figure 2-10 Passive Structures and Components - Degradation Occurrences Distribution of Subcomponents for Concrete

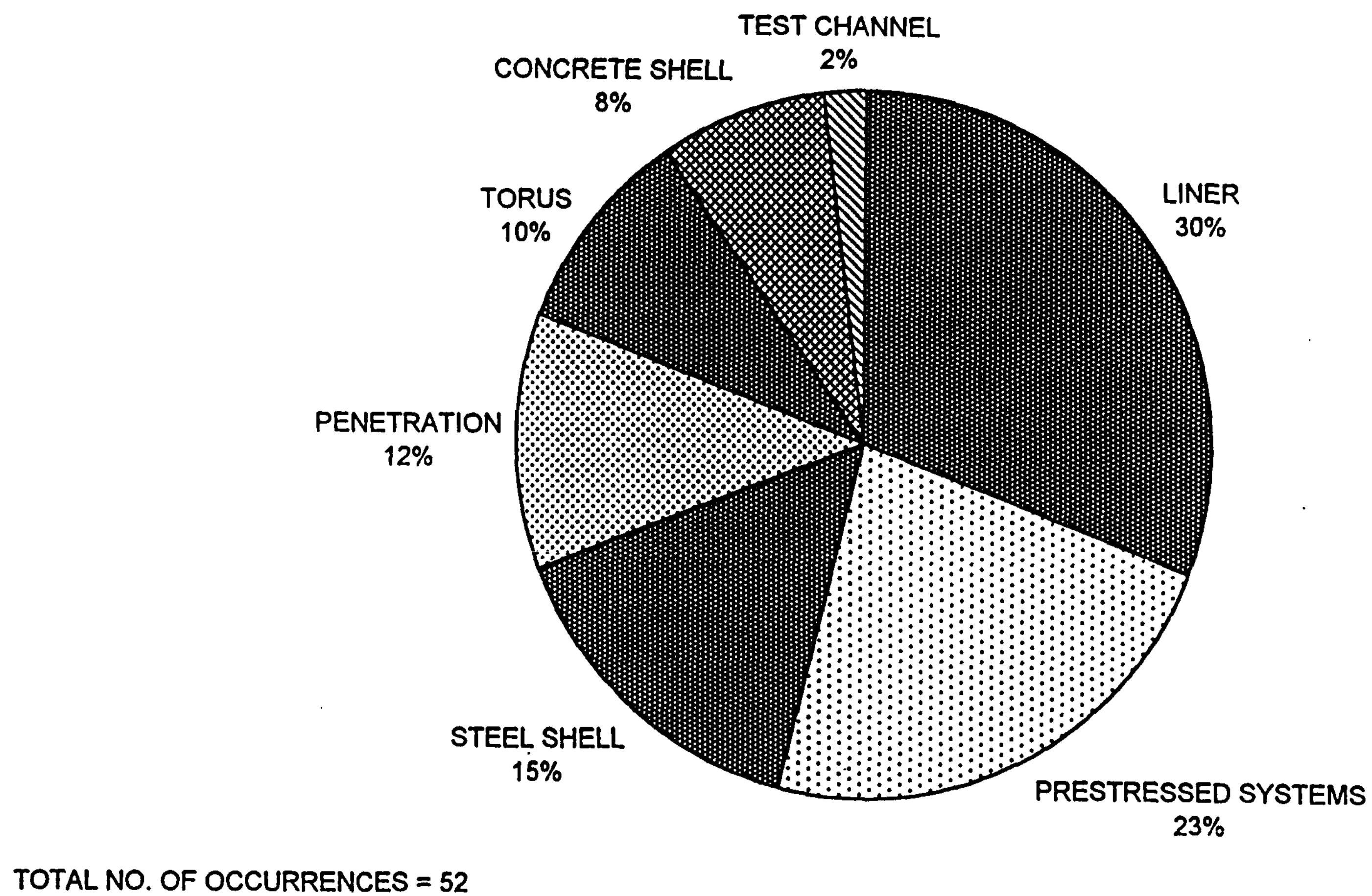

Figure 2-11 Passive Structures and Components - Degradation Occurrences Distribution of Subcomponents for Containment

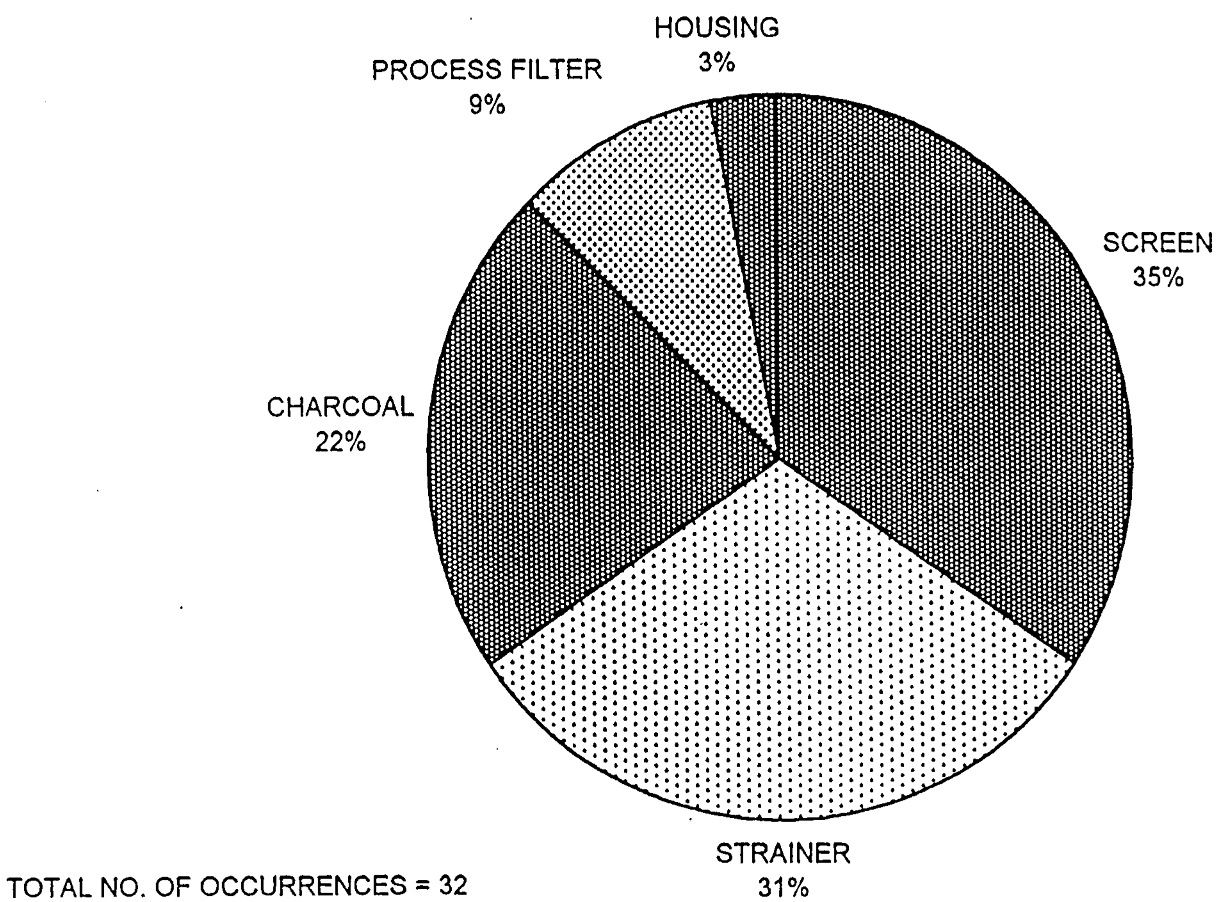

Figure 2-12 Passive Structures and Components - Degradation Occurrences Distribution of Components for Filters

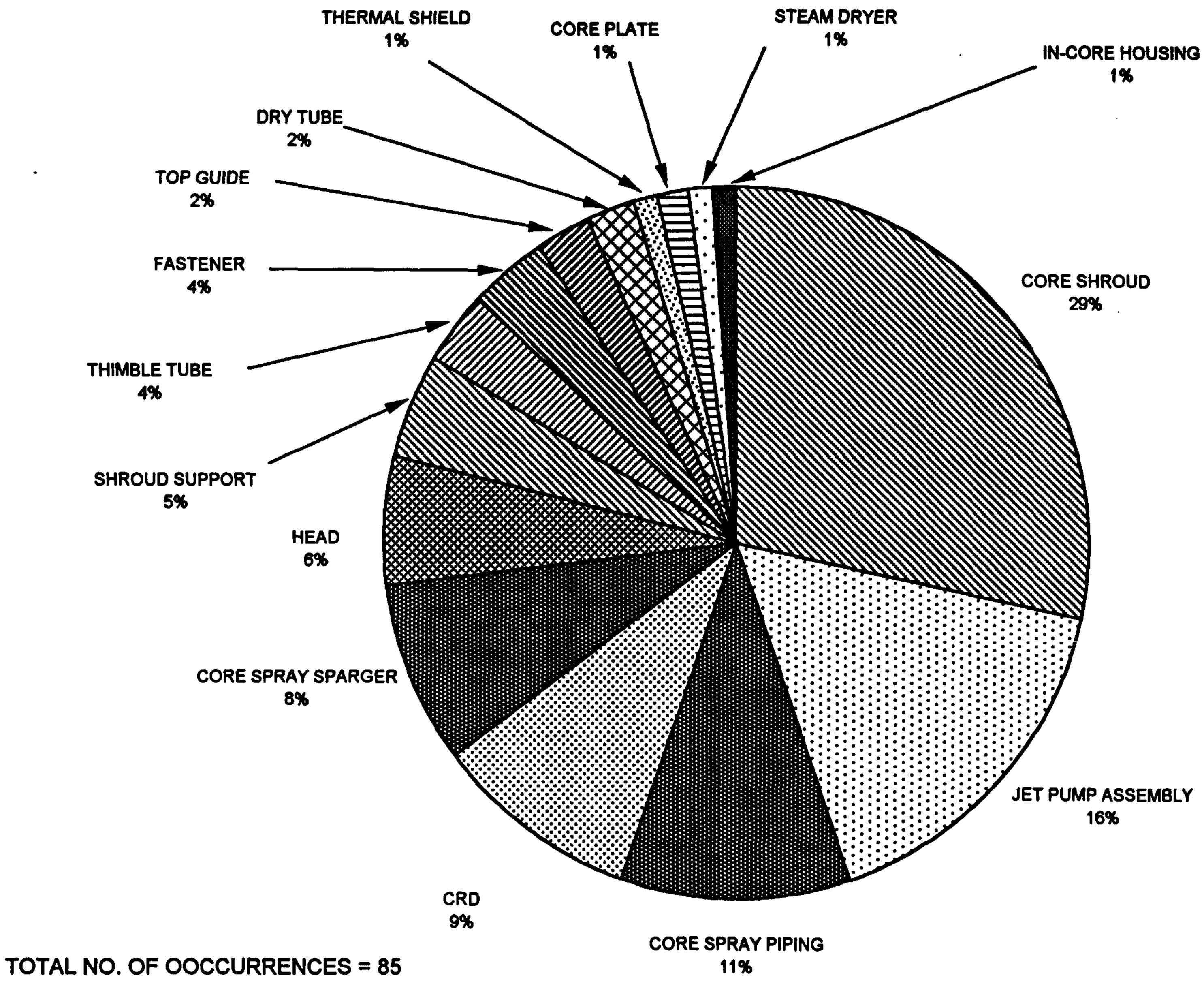

Figure 2-13 Passive Structures and Components - Degradation Occurrences Distribution of Components for RPV

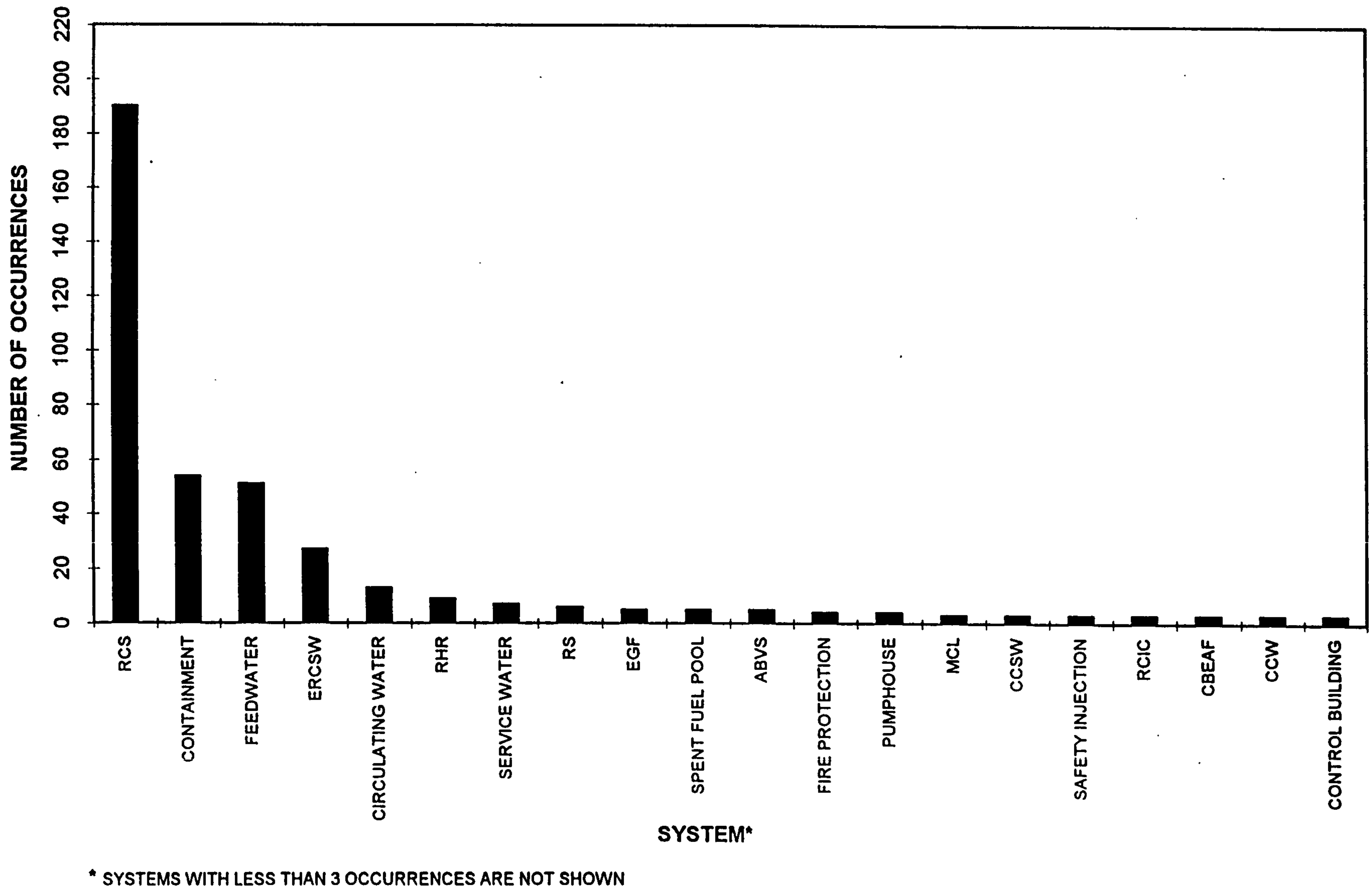

Figure 2-14 Passive Structures and Components - Degradation Occurrences Distribution By System

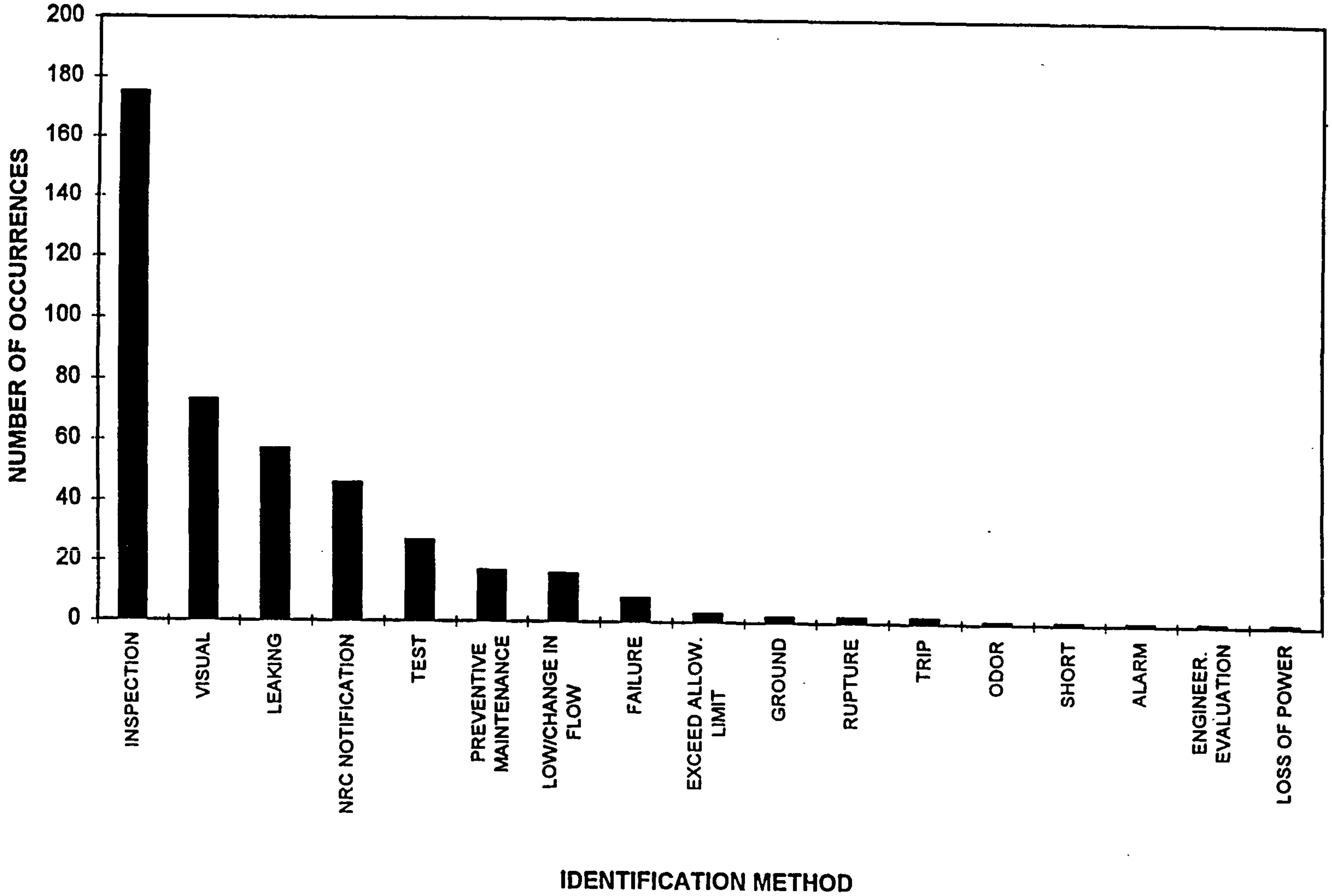

Figure 2-15 Passive Structures and Components - Degradation Occurrences Distribution By Methods of Identification

Table 2-1 Structures and Passive Components

ITEM	SUBTOPICS	NO OF OCCURRENCE
ANCHORAGES	EMBEDDED ANCHORS, EXPANSION ANCHORS, UNDERCUT ANCHORS, DROP-IN ANCHORS, EMBEDDED STUDS, GROUT. ALL COMPONENTS INCLUDING EQUIPMENT	8
CABLE TRAY SYSTEMS	ELECTRICAL CABLE TRAYS, TRAY SUPPORTS, JUNCTION BOXES	1
CONCRETE	REINF. CONC. BLDGS; WATER INTAKE STRUCTS.; UNDERGROUND STRUCTS.; CONCRETE - WALLS, FLOORS, CEILINGS, MATS, FNDTNS., CANALS, POOLS, PITS, PEDESTALS, PRESTRESSED, & MANHOLES; MASONRY	32
CONDUIT SYSTEMS	ELECTRICAL CONDUITS, CONDUIT SUPPORTS	6
CONTAINMENT	SHELL - STEEL & CONCRETE, PRESTRESSING SYSTEM, PENETRATIONS**, TORUS, BELLOWS, LINERS, SUPPORTS	52
COOLING TOWER		1
ELECTRICAL CONDUCTORS **	CABLE/WIRES INCLUDING INSULATION, BUS DUCT	9
EXCHANGERS	STEAM GENERATOR, HEAT EXCHANGER, CONDENSER (INCLUDING ICE) & SUPPORTS	108
FILTERS	MECHANICAL & HVAC - SCREEN, SEPARATOR, STRAINER, ADSORBER, SUPPORTS, HOUSING. ONLY MATERIAL TYPE DEGRADATION; EXCLUDE REGULAR MAINTENANCE ITEMS	32
HVAC DUCT	DUCT AND ITS SUPPORTS	3
INSULATION/SEAL	PIPE INSULATION**,CONTAINMENT INSULATION, CERAMIC INSULATORS, FLOOR SEALS, FLOOD PROTECTION SEALS	4
PIPING SYSTEM	PIPING**, FITTINGS**, SMALL BORE PIPING** & TUBING**, SLEEVES**, PIPE SUPPORTS (EXCLUDING HYDRAULIC & MECHANICAL ASSEMBLY OF SNUBBERS), UNDERGROUND PIPING	105***
RPV	SHELL, INTERNALS, CRD (PASSIVE COMPONENTS ONLY), SUPPORTS	83
STRUCTURAL SEISMIC GAP	(MAINTAINING THE PHYSICAL GAP BETWEEN STRUCTURES TO ACCOMMODATE SEISMIC MOVEMENT)	0
STRUCTURAL STEEL	FRAMES, TRUSSES, PLATFORMS, SUPPORTS, BOLTS, STUDS, FASTENERS, LINERS, DOORS, COVERS, HATCHES, SUPPORT TO ALL TYPES OF EQUIPMENT	25
TANKS		11
VESSELS	PRESSURIZER, OTHER PRESSURIZED VESSELS, AND SUPPORTS	10
WATER-CONTROL STRUCTURES	DAMS, EMBANKMENTS, SPRAY PONDS	2
	TOTAL =	492
*	OCCURRENCES ARE DEFINED AS INSTANCES OF AGING DEGRADATION REPORTED IN LER'S, IE BULLETINS, GENERIC LETTERS, IN'S, NUREGS, AND OTHER REFERENCE DOCUMENTS.	
**	THESE ITEMS REMOVED FROM FURTHER REVIEW DUE TO OTHER EXISTING NRC PROGRAMS	
***	NUMBER OF OCCURRENCES WELL IN EXCESS OF NUMBER REPORTED	

Table 2-2 Degradation Occurrence Table (Sample)

COMPONENT	SUBCOMPONENT	SYSTEM	AGING EFFECTS	AGING MECHANIS	PLANT	MO	DA	YR	HOW IDENT	EVALN METH	REPAIR METH	DKT	REF DOC	REF NO
ANCHORAGES	ANCHOR BOLTS - STRAINER	SERVICE WATER	DETERIORATION	N. A.	ROBINSON 2			95*	VISUAL	N. A.	N. A.	261	NUREG	1522
ANCHORAGES	EXPANSION ANCHOR	ERCSW	FAILURE	CORROSION	MILLSTONE 2	4	29	86	VISUAL	VISUAL	N. A.	336	LER	860100
ANCHORAGES	EXPANSION ANCHOR - NUTS	ERCSW	LOOSENING	VIBRATION	QUAD CITIES 1	5	8	87	N. A.	VISUAL	REPLACEMENT	254	LER	870801
ANCHORAGES	GROUT & BASEPLATES	N. A.	DETERIORATION	CORROSION	BEAVER VALLEY 1			95*	VISUAL	N. A.	N. A.	334	NUREG	1522
ANCHORAGES	GROUT- EQUIPMT. SUPPORT	PUMPHOUSE	CRACKING DETERIORATION	MOISTURE	POINT BEACH 2			95*	VISUAL	N. A.	N. A.	301	NUREG	1522
ANCHORAGES	GROUT- EQUIPMT. SUPPORT	PUMPHOUSE	CRACKING DETERIORATION	MOISTURE	POINT BEACH 1			95*	VISUAL	N. A.	N. A.	266	NUREG	1522
ANCHORAGES	N. A.	SEVERAL	DETERIORATION	N. A.	COOPER			95*	VISUAL	N. A.	N. A.	298	NUREG	1522
ANCHORAGES	STUDS - EMBEDDED	N. A.	FAILURE	MECHANICAL LOADS	INDIAN POINT 2	2	12	91	VISUAL	VISUAL	REPAIR	247	LER	910401
CABLE TRAY SYSTEM	ELECTRICAL CABLE TRAY-SEAL	CBEAF	DETERIORATION	N. A.	BRUNSWICK 1	10	25	95	TEST	N. A.	REPAIR	325	LER	952001
CONCRETE	CEILING	FHB	CRACKING DETERIORATION	MOISTURE	TURKEY POINT 3			95*	N. A.	N. A.	N. A.	250	NUREG	1522
CONCRETE	CEILING/FOUNDATION	SECONDARY CONTMT.	DETERIORATION	N. A.	DRESDEN 2	4	7	89	TEST	N. A.	REPAIR	237	LER	891400
CONCRETE	FLOORS, WALLS, FOUNDATION	VARIOUS STRUCTURES	CRACKING SPALLING	N. A.	COOPER			95*	VISUAL	N. A.	N. A.	298	NUREG	1522
CONCRETE	INTAKE STRUCT. - BEAMS	CIRCULAT. WATER	CRACKING	CORROSION - EMBED. STL.	TURKEY POINT 4			89	N. A.	N. A.	REPAIR	251	NUREG	1522
CONCRETE	INTAKE STRUCT. - BEAMS	CIRCULAT. WATER	CRACKING	CORROSION - EMBED. STL.	TURKEY POINT 3			89	N. A.	N. A.	REPAIR	250	NUREG	1522
CONCRETE	INTAKE STRUCT. - BEAMS & WALLS	SERVICE WATER	CRACKING	CORROSION - EMBED. STL.	SAN ONOFRE 1			84	INSPECTION	N. A.	REPAIR	206	NUREG	1522
CONCRETE	INTAKE STRUCTURE	SERVICE WATER	CRACKING	N. A.	ROBINSON 2			95*	VISUAL	N. A.	N. A.	261	NUREG	1522
CONCRETE	INTAKE STRUCTURE	RAW WATER - INTAKE STR.	CRACKING	N. A.	BEAVER VALLEY 1			95*	VISUAL	N. A.	N. A.	334	NUREG	1522
CONCRETE	MASONRY WALL	N. A.	CRACKING	N. A.	TURKEY POINT 3			95*	N. A.	N. A.	N. A.	250	NUREG	1522
CONCRETE	MASONRY WALL	N. A.	CRACKING	N. A.	TURKEY POINT 4			95*	N. A.	N. A.	N. A.	251	NUREG	1522
CONCRETE	MASONRY WALL	FAN HOUSE	CRACKING	N. A.	INDIAN POINT 2	9	16	85	INSPECTION	VISUAL	REPAIR	247	NRC IN	87-67-1
CONCRETE	MASONRY WALL	N. A.	CRACKING	N. A.	OYSTER CREEK	5	5	86	INSPECTION	VISUAL	REPAIR	219	NRC IN	87-67-1
CONCRETE	MASONRY WALL	N. A.	CRACKING	N. A.	YANKEE ROWE	1	26	87	INSPECTION	VISUAL	REPAIR	29	NRC IN	87-67-1

Table 2-3 Aging Effects and Mechanisms

CONCRETE AGING EFFECTS	AGING MECHANISMS CONCRETE	STEEL AGING EFFECTS	AGING MECHANISM STEEL	"OTHER" AGING EFFECTS	AGING MECHANISMS
CRACKING	FREEZE-THAW	CRACKING	MOISTURE	CRACKING	SEE CONCRETE/STEEL CAUSES
SPALLING	LEACHING	LOSS OF MATERIAL	TEMPERATURE - ELEVATED OR SUBFREEZING	DETERIORATION	AGING/END OF LIFE
SCALLING	CHEMICAL ATTACK	WALL THINNING	CORROSION	PEELING	GREASE LEAKAGE
POPOUTS	CORROSION OF EMBEDDED STEEL	REDUCED STRENGTH	CHEMICAL ATTACK		IRRADIATION EMBRITTLEMENT
DETERIORATION	IRRADIATION	LOSS OF FRACTURE TOUGHNESS	MECHANICAL WEAR		
LOSS OF MATERIAL	ELEVATED TEMPERATURE	EXCESSIVE DEFORMATION	EROSION		
EXCESSIVE DEFORMATION	EROSION	LOSS OF PRELOAD	MECHANICAL LOADS		
FAILURE	MOISTURE	FAILURE	VIBRATION		
DISINTEGRATION		LOOSENING	ORGANISMS		
INCREASE POROSITY & PERMEABILITY		RUPTURE	IMPROPER DESIGN		
		FOULING	STRESS CORROSION CODES (IGSCC, HSC, TGSCC, SCC, PWSCC, IGA, ODSCC)		
		PLUGGING	FATIGUE		
		DETERIORATION	THERMAL FATIGUE		
			PARTICLES/FOREIGN OBJECTS		
			IRRADIATION EMBRITTLEMENT		
			STRESS RELAXATION		

Table 2-4 System Definition Codes

ABBREVIATION	DEFINITION
ABVS	AUXILIARY BUILDING VENTILATION SYSTEM
AWCT	AERATED WASTE CONCENTRATE TANK
BWMS	BORATED WATER MAKE-UP SYSTEM
CBEAF	CONTROL BUILDING EMERGENCY AIR FILTRATION
CCGC	CONTAINMENT COMBUSTIBLE GAS CONTROL
CCSW	CONTAINMENT COOLING SERVICE WATER
CCW	COMPONENT COOLING WATER
CIC	CONTAINMENT ICE CONDENSER
CRDM	CONTROL ROD DRIVE MECHANISM
CSI	CORE SPRAY INJECTION
CVCS	CHEMICAL & VOLUME CONTROL SYSTEM
ECCS	EMERGENCY CORE COOLING SYSTEM
EDG	EMERGENCY DIESEL GENERATOR
EGF	EMERGENCY GENERATOR FUEL
ERCSW	ESSENTIAL RAW COOLING SERVICE WATER
ERCSW/CS	ESSENTIAL RAW COOLING SERVICE WATER/CONTAINMENT SPRAY
FHB	FUEL HANDLING BUILDING
HPCI	HIGH PRESSURE COOLANT INJECTION
IASL	INSTRUMENT AIR SUPPLY LINE
JPIL	JET PUMP INSTRUMENT LINE
MCL	MAKE-UP COOLANT LINE
MSR	MOISTURE SEPARATOR REHEATER SYSTEM
PRPS	PRIMARY RECIRCULATION PIPING SYSTEM
RBCCW	REACTOR BUILDING CLOSED COOLING WATER
RBCLC	REACTOR BUILDING CLOSED LOOP COOLING
RBS	REACTOR BUILDING SPRAY
RCIC	REACTOR CORE ISOLATION COOLING
RCS	REACTOR COOLANT SYSTEM
RHR	RESIDUAL HEAT REMOVAL
RS	RECIRCULATION SYSTEM
RSHX	RECIRCULATION SPRAY HEAT EXCHANGER
RSS	RECIRCULATION SPRAY SYSTEM
RWCS	REACTOR WATER CLEAN-UP SYSTEM
SAT	STATION AUXILARY TRANSFORMER
SBGT	STANDBY GAS TREATMENT
SFCS	SPENT FUEL POOL COOLING SYSTEM
SGB	STEAM GENERATOR BLOWNDOWN
SIPSL	SAFETY INJECTION PUMP SUCTION LINE
SWS	SERVICE WATER SYSTEM

Table 2-5 Stress Corrosion Codes

ABBREV	DEFINITION
HSC	HYDROGEN STRESS CORROSION
IGA/SCC	INTERGRANULAR ATTACK/ STRESS CORROSION CRACKING
IGSCC	INTERGRANULAR STRESS CORROSION CRACKING
ODSCC	OUTER DIAMETER STRESS CORROSION CRACKING
PWSCC	PRIMARY WATER STRESS CORROSION CRACKING
SCC	STRESS CORROSION CRACKING
TGSCC	TRANSGRANULAR STRESS CORROSION CRACKING

3 AGE-RELATED DEGRADATION TECHNOLOGY INFORMATION

In Phase I of this program, existing technical information was collected and reviewed to provide input into the research effort. Information from NRC programs and industry programs regarding inspection, testing, assessment, and repair techniques were identified and reviewed. In addition, information related to aging/degradation mechanisms and effects on material properties/strengths was also reviewed.

3.1 NRC Requirements/Guidance

In the past, there had been very limited requirements for the inspection, maintenance, monitoring, and evaluation of structures and passive components for the effects of degradation. The requirements and guidance were available primarily for containment structures, water-control structures, masonry walls, above ground piping and supports, steam generators, and RPV. This section of the report describes current NRC requirements and available guidance related to degradation of structures and passive components.

<u>Containments</u>

Containments are subject to periodic leak rate testing in accordance with 10 CFR Part 50, Appendix J. The leak rate testing includes Type A, B, and C tests. Type A tests measure the primary reactor containment overall integrated leakage rate. Type B tests are intended to detect local leaks for penetrations. Type C tests are intended to measure containment isolation valve leakage rates. In addition to these three types of tests, a general visual inspection of the accessible interior and exterior surfaces of the containment structures and components must be performed prior to any Type A test to identify any evidence of structural deterioration which may affect either the containment structural integrity or leak-tightness.

Additional requirements for prestressed concrete containments are provided in Regulatory Guides 1.35 and 1.35.1 for ungrouted tendons. Regulatory Guide 1.35 describes a basis acceptable to the NRC staff for developing an appropriate inservice inspection and surveillance program for ungrouted tendons in prestressed concrete containments. Regulatory Guide 1.35 provides guidance for performing visual inspections, prestress monitoring tests (lift-off tests), tendon material tests and inspections, inspection of filler grease, evaluation of inspection results, and reporting requirements. Regulatory Guide 1.35.1 provides a basis acceptable to the staff for developing appropriate presstressing tolerance bands for tendons so that these limits can be compared against the lift-off forces measured in the sample inspection program of Regulatory Guide 1.35.

In part, because the visual inspection requirements of 10 CFR Part 50, Appendix J do not provide specific guidance, have not been applied in a consistent manner, and the rate of age-related degradation occurrences have been increasing, 10 CFR 50.55a has been revised to provide more precise requirements. The objective of the revised 10 CFR 50.55a is to assure that the critical areas of containments are routinely inspected to detect and to take corrective action for defects that could compromise a containment's structural integrity. The final rulemaking, which was effective on September 9, 1996, endorses the 1992 Edition with 1992 Addenda of Section XI, Subsection IWE (Class MC Containments) and Subsection IWL (Class CC Containments) of the ASME

Code. A recent revision to 10 CFR 50.55a has also endorsed as acceptable the 1995 Edition with the 1996 Addenda of the ASME Code. Licensees must incorporate Subsection IWE and Subsection IWL into inservice inspection programs for containments. The rulemaking includes exemptions from and additional requirements to those in Subsections IWE and IWL.

Water-Control Structures

For water-control structures such as intake structures, canals, dams, earthen embankments and slopes associated with emergency cooling water systems or flood protection, Regulatory Guide 1.127 describes a basis acceptable to the staff for developing an appropriate inservice inspection and surveillance program. Guidance is provided for the compilation of engineering data, onsite inspection program, technical evaluation, frequency of inspections, and preparation of reports.

Masonry Walls

NRC IE Bulletin 80-11 "Masonry Wall Design" initiated a major re-evaluation effort in the nuclear industry to demonstrate the structural adequacy of reinforced and unreinforced masonry walls. Of the seventy plants originally in the scope of 80-11, two were shut down; three were reviewed under the Systematic Evaluation Program (SEP); one plant had no safety-related masonry walls; four were qualified by analytical methods verified by full-scale testing; and the remaining sixty plants were qualified in accordance with the Structural Engineering Branch (SEB) Interim Criteria.

In December 1987, NRC issued Information Notice No. 87-67, "Lessons Learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11". In this notice, a number of deficiencies uncovered during site audits were described. They were grouped into the following categories:

- Unanalyzed Conditions – existing cracks in unreinforced masonry
- Improper Assumptions – mortar properties, boundary conditions, presence of reinforcement
- Improper Classification – specification of safety-related versus non-safety walls
- Lack of Procedural Controls – walk-down surveys, record keeping, modification activities

It was also noted that "NRC inspectors observed that mechanisms did not exist at certain facilities to ensure that the physical conditions of masonry walls remained as previously analyzed."

In August 1988, an internal NRC report was issued: "Status of Multi-Plant Action (MPA) B-59, Masonry Wall Design." This report recommended that the MPA be considered closed and also summarized the current status of each plant included in the action. The report also stated that the Office of Inspection and Enforcement had responsibility of inspection related activities.

10 CFR 50.65 – Maintenance Rule

On July 10, 1991, the NRC published 10 CFR 50.65 entitled, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," referred to as the Maintenance Rule. The regulation was effective July 10, 1996. The purpose of the Maintenance Rule is to monitor the effectiveness of maintenance activities for safety significant plant equipment in order to minimize the likelihood of failures and abnormal events caused by the lack of effective maintenance. The final rule requires that licensees monitor the performance or condition of structures, systems, and components (SSCs) against licensee-established goals in a manner sufficient to provide reasonable assurance that the SSCs will be capable of performing their intended functions. Such monitoring needs to be established commensurate with safety and, where practical, take into account industry operating experience.

Several other documents related to the Maintenance Rule contain additional technical information and guidance: Regulatory Guide 1.160, Rev. 2, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants;" NUMARC 93-01, Rev. 2, "Nuclear Energy Institute - Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants;" NRC Inspection Manual - Inspection Procedure 62706, "Maintenance Rule;" NRC Inspection Manual - Inspection Procedure 62002, "Inspection of Structures, Passive Components, and Civil Engineering Features at Nuclear Power Plants;" and NRC Inspection Manual - Inspection Procedure 62003, "Inspection of Steel and Concrete Containment Structures at Nuclear Power Plants."

<u>10 CFR Part 54 - License Renewal Rule</u>

Nuclear power plants were initially licensed to operate for 40 years. The requirements for obtaining the renewal of a nuclear power plant operating license for up to an additional 20 years are presented in 10 CFR Part 54 - License Renewal Rule. Under Part 54, applicants are required to identify all structures, systems, and components (SSCs) that are within the scope of the rule. A screening review is then required to identify those SSCs that are "passive and long-lived" structures and components. For the passive, long-lived structures and components, the applicant must demonstrate that the effects of aging will be managed so that the intended function(s) will be maintained consistent with the current licensing basis through the period of extended operation.

The license renewal rule also requires the applicant to identify and update all time-limited aging analyses which are part of the current licensing basis. An example would be a design basis fatigue analysis of a piping system which assumed a specified number of loading events based on a 40-year period of operation.

In August 1996, the NRC issued a draft regulatory guide, DG-1047, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses" for public comment. In addition to presenting a uniform format and content acceptable to the staff for its review, the draft regulatory guide proposes to endorse the Nuclear Energy Institute (NEI) guidance document NEI 95-10 as an acceptable method for complying with the documentation requirements for a license renewal application.

On September 21, 1997, the NRC made available to the public a working draft "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants" (SRP-LR). The draft SRP-LR was prepared to provide guidance for staff reviewers in performing safety reviews of applications to renew licenses of nuclear power plants in accordance with 10 CFR Part 54. The staff is currently revising the SRP-LR to reflect the experience gained from review of the initial applications and communication with the industry. A major effort is currently in progress to develop specific guidance for use by both applicants and staff reviewers to ensure effective, efficient and consistent satisfaction of the LR Rule requirements. This work is a continuation of Generic Issues Lessons Learned (GALL). The revised SRP-LR will reference the GALL Report for descriptions of generic aging management programs which the staff has evaluated and found applicable to license renewal.

3.2 NRC Programs

The research program described in this report surveys and evaluates degradation occurrences of structures and passive components at nuclear power plants not addressed by previous or existing programs. The survey and trending analyses performed in this research effort build on the results of other programs and establish the aging performance of structures and passive components from actual plant experience. The evaluations in this program also permit a determination of the susceptibility of the various structures and components to aging degradation and prioritize the structures and components for further study.

The other major NRC programs related to aging degradation of structures and passive components in nuclear power plants are summarized below. A more complete listing of the NRC programs in this area has been compiled in the Degradation Reference Database described in Section 3.5 of this report.

Nuclear Plant Aging Research (NPAR) Program

The NPAR program was implemented in 1985 to identify and resolve technical safety issues related to aging of SSCs in operating nuclear power plants. The principal goals of the program were to understand the effects of age-related degradation in NPPs and how to manage and mitigate them effectively. NUREG-1144, Rev. 2 describes the objectives of the program, the current status of research, and summarizes the utilization of the research results in the regulatory process. As a result of the NPAR program approximately 100 NUREG/CR reports have been developed as of June 1991, plus numerous published papers and proceedings.

A listing of past research activities through September 1993 is presented in NUREG-1377, Rev. 4. This NUREG contains summaries of NRC sponsored reports that were generated in the NPAR Program. Each summary describes the objectives of the research, the contractor and authors, and outlines significant research results. Although most of the items included in this NUREG cover hardware oriented plant components and systems, there are some summaries given for structural and passive components.

Structural Aging (SAG) Program

Another major research program, which was sponsored by the NRC, is the SAG Program. The SAG Program was initiated in 1988 with the objective of developing technical bases for addressing aging of safety-related concrete structures and providing guidance for use in evaluating continued service of these concrete structures. Over 90 technical reports and papers have been published describing the results of the program.

The SAG Program consisted of a management task and three technical task areas. The purpose of the management task was to effectively manage the technical tasks related to the safety issues of aging NPP concrete structures. The first technical task was to develop a materials property database. This consisted of a reference source containing data and information on the time variation of material properties under exposure to applicable environmental stressors (mechanisms) and aging factors. The materials database covered various concrete types, steel reinforcements, prestressing tendons, structural steels, and rubber materials. The information contained in the database can be used to predict deterioration of structural components in NPPs and in developing limits on detrimental environmental exposures. All of the material property data have been compiled into a Structural Materials Information Center (SMIC). The SMIC has been developed in two formats – a Structural Materials Handbook and a Structural Materials Electronic Database.

The second technical task described a methodology that can be used to (1) make quantitative assessments of environmental stressors or aging factors that could affect safety-related concrete structures at NPPs and (2) provide recommended in-service inspection (ISI) or sampling procedures for use in evaluating the structural condition and for trending the performance of these components. Also included in this task are the identification and evaluation of techniques for mitigation of stressors or aging factors that may affect critical concrete components, and an assessment of techniques for repair, replacement, or retrofitting of deteriorated concrete components.

The third technical task developed a quantitative methodology for continued service determinations. This included development of predictive models to assess the current and future reliability and performance of concrete structures.

A summary of the entire SAG Program is provided in NUREG/CR-6424. It describes the SAG Program including a description of safety-related concrete structures and longevity considerations; inservice inspection, condition assessment, and remedial measure considerations; evaluation of NPP reinforced concrete structures; reliability-based methodology for condition assessments; and summary, conclusions, and recommendations. The NUREG includes an excellent description of the aging mechanisms and aging effects for concrete and associated steel components of reinforced concrete structures. Appendix B to the NUREG provides a listing of the numerous reports and papers that were developed under the SAG Program.

Some of the conclusions as reported in NUREG/CR-6424 are:

- The performance of the reinforced concrete structures in NPPs has been good. However, as these structures age, incidences of degradation due to environmental stressor effects are likely to increase to potentially threaten their durability. Items of note would be corrosion of steel reinforcement due to carbonation of the concrete or presence of chloride ions, excessive loss of prestressing force, leaching of concrete, and leakage of post-tensioning system corrosion inhibitor through cracks in the concrete.

- Techniques for detecting the effects of environmental stressors are sufficiently developed to provide qualitative data.

- Methods for conducting condition assessments of reinforced concrete structures are fairly well established. Few standards or criteria are available for interpreting the results obtained from condition assessments. Current inspection requirements for NPP reinforced concrete structures are fairly limited with the exception of concrete containments.

- Techniques for repair of concrete structures are well established and when properly selected and applied are effective. At the time, no codes or standards are available for repair of reinforced concrete structures, although some are being developed. Criteria that may be used to determine when a repair action should be implemented are not available.

- A reliability-based methodology has been developed that can be used to facilitate quantitative assessments of current and future structural reliability and performance of reinforced concrete structures in NPPs.

Nuclear Power Plant Generic Aging Lessons Learned (GALL)

NUREG/CR-6490 entitled Nuclear Power Plant Generic Aging Lessons Learned (GALL) describes the effort sponsored by the NRC to perform a systematic review of plant aging information in order to assess materials and component aging issues related to continued operation and license renewal of operating plants. A literature review was performed for mechanical, structural, thermal-hydraulic components and systems, and electrical components and systems.

The results of these reviews were tabulated and included in a two-volume report. The NUREG concluded, "all ongoing significant component aging issues are currently being addressed by the regulatory process. However, the aging of what are termed passive components have been highlighted for continued scrutiny." The NUREG lists the aging issues significant to passive components. Most of the structural components evaluated pertain to the RPV (instrumentation and CRD housing nozzles, closure studs, jet pump and holddown beams, reactor internals, core shroud, etc.); piping and feedwater nozzles and interfacing tanks and components; concrete shield walls; and other concrete elements.

The NUREG also concluded, "passive components are not as extensively or thoroughly covered by current plant maintenance procedures. Furthermore, surveillance and monitoring methods and instrumentation and procedures have not been as extensively developed or employed for passive components subjected to the highlighted aging mechanisms, nor are some of the passive component aging mechanisms as well understood." In addition, the NUREG points out that passive components are often the most costly and most difficult to replace. Therefore, the knowledge base for predicting applicable aging effects behavior and significance is very important for passive components.

Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures (NUREG-1522)

In June 1995, the NRC published NUREG-1522, entitled "Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures." This report describes the condition of structures and civil engineering features at operating nuclear power plants and provided information that would help identify, monitor, and correct degraded conditions of these structures. The NUREG contains descriptions of age-related degradation, which were obtained from many different sources. The most significant information came from site visits, conducted by the NRC staff and its contractor BNL, at six older nuclear power plants (licensed before 1977).

Some of the observations noted in the report identify certain types of structures (e.g. water intake structures, masonry walls, anchorages, tanks, buried piping, and inaccessible areas) as requiring special considerations. The report also concludes that based on the observations and information collected, structures and civil engineering features should be periodically inspected and a systematic maintenance program be implemented to ensure the expected useful life of the structures.

3.3 Industry Programs

There are many industry programs developed over the years that address age-related degradation of structures and passive components. Some of these programs are described below and a more complete listing is provided in the Degradation Reference Database that is described in Section 3.5 of this report.

NUMARC Industry Reports (IRs)

DOE and EPRI sponsored the preparation of ten industry reports under the direction of the Nuclear Management and Resources Council (NUMARC). The IRs covered items such as PWR and BWR vessels, internals, primary coolant boundary, containments, and Class I structures. The purpose of the IRs is to address age-related degradation of these components on a generic basis. The IRs would provide the technical basis, which could be referenced by licensees in support of their license renewal application.

Each IR identifies the components that comprise the subject item (e.g. BWR containment) and evaluates each component in terms of possible age-related degradation mechanisms. Thus, certain aging mechanisms were eliminated and only those age-related degradation mechanisms that could affect the component were identified and described. In addition, the IRs evaluated the capability of programs to manage aging mechanisms that are applicable, and where generic effective programs cannot be shown to be capable of managing the effects of age-related degradation, aging management options for plant-specific programs are described.

NEI – Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants

The Nuclear Energy Institute (NEI) has developed an industry guidance document (NUMARC 93-01, Rev. 2) entitled, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." This guideline was developed to assist the industry in implementing the final Maintenance Rule (10 CFR 50.65). The guideline describes the process for the identification of the SSCs within the scope of the Maintenance Rule and the process of establishing plant-specific risk significant criteria and performance criteria.

Areas covered in the guideline include methodology to select plant structures, systems, and components; establishing risk and performance criteria/goal setting and monitoring; SSCs subject to effective preventive maintenance programs; evaluation of systems to be removed from service; and periodic maintenance effectiveness assessments.

NUMARC 93-01, Rev. 2 specifically addresses monitoring of structures under the Maintenance Rule (MR). The applicability of the MR to structures was a subject of considerable confusion within the industry during initial implementation of the MR. It is clearly stated in Section 10.2.3 of NUMARC 93-01, that structures which perform intended functions, in accordance with the criteria provided in NUMARC 93-01, are within the scope of the MR and require a monitoring program which ensures that degradation is detected before there is loss of any intended function.

Regulatory Guide 1.160, Rev. 2 endorses NUMARC 93-01, Rev. 2 as an acceptable method to satisfy the general requirements of the MR. Regulatory Guide 1.160, Rev. 2 also addresses monitoring of structures under the MR and provides specific guidance for satisfying the requirements of the MR, as it pertains to structures.

American Concrete Institute (ACI) Codes and Standards

Over the years, the ACI has developed a number of codes and standards that relate to degradation of reinforced concrete structures. ACI 201.1R-68, "Guide for Making a Condition Survey of Concrete in Service" provides a system for reporting on the condition of concrete in service. This guide includes a checklist for making a survey of the condition of concrete, provides a definition of the terms associated with the durability of concrete, and presents actual photographs to demonstrate the different types of aging effects.

ACI 201.2R-77, "Guide to Durable Concrete" discusses the more important causes of concrete degradation and gives recommendations on how to prevent such damage. Topics covered include freezing and thawing, aggressive chemical exposure, abrasion, corrosion of steel and other materials embedded in concrete, chemical reactions of aggregates, repair of concrete, and the use of coatings to enhance concrete durability.

ACI 207.3R-79, "Practices for Evaluation of Concrete in Existing Massive Structures for Service Conditions" describes methods for evaluating the physical properties of concrete in existing concrete structures. The report covers the review of preconstruction data, construction, operation and maintenance records; review of in-service inspections; condition surveys; nondestructive testing; and destructive testing.

ACI 224.1R-93, "Causes, Evaluation, and Repair of Cracks in Concrete Structures" summarizes the causes of cracks in concrete and the means for their control. The report also describes evaluation procedures and methods for crack repair such as epoxy injection, routing (enlarging the crack) and sealing, stitching (U-shaped metal units), use of additional reinforcement, and grouting.

ACI 349.3R-96, "Evaluation of Existing Nuclear Safety-Related Concrete Structures" presents recommendations for developing an effective evaluation procedure for nuclear safety-related concrete structures. The report describes the selection process of critical structures, the various degradation mechanisms, inspection techniques, evaluation criteria, evaluation frequency, qualifications of evaluation team, and repairs. Under the evaluation criteria recommendations, ACI 349.3R-96 presents a three tiered evaluation criteria: acceptance without further evaluation, acceptance after review, and conditions requiring further evaluation. It is in this area that the technical basis for some of the acceptance criteria need to be developed, expanded, and documented.

Other ACI standards such as ACI 224R-90, "Control of Cracking in Concrete Structures" and 222R-89, "Corrosion of Metals in Concrete" are listed in the Reference section of ACI 349.3R-96. ACI 349.3R-96 also lists related standards from ASCE, ASME, and ASTM.

American Society of Civil Engineers Standard

American Society of Civil Engineers (ASCE) Standard ASCE 11-90, "Guideline for Structural Condition Assessment of Existing Buildings" describes guidelines and a methodology for the structural assessment of existing buildings. Assessment techniques are provided for conventional buildings (non-nuclear) constructed from materials consisting of concrete, metals, masonry, and wood. The standard describes assessment procedures, condition assessment of materials, and evaluation procedures. Tables are presented in the guideline, which provide for each test method, a description of the application, principle of operation, user expertise, advantages, limitations, and references. Also included in the guideline are tables, which identify the various test methods which are most appropriate to evaluate chemical and physical properties of the material.

3.4 Other Sources of Technical Information

Information from Japan

Other sources of technical information regarding age-related degradation of structures and passive components have been identified from international sources. A review of Japanese literature for degraded concrete structures was conducted by BNL under a separate research program for the NRC. A report by Park (September 1998) entitled "Effects of Aging Degradation on Seismic Performance of Reinforced Concrete Structures: Summary of Japanese Literature in Related Areas" summarizes the results of the review.

The report provides a summary of a literature survey of available Japanese publications. Key observations are described in detail regarding age-related degradation mechanisms and seismic performance of degraded reinforced concrete structures. The report covers experimental studies on reinforced concrete members such as shear walls and beams in degraded conditions. Some of the observations and preliminary conclusions noted are:

- Vertical cracks in beams (normal to member axis) reduce the bending stiffness. However, vertical cracks do not significantly reduce the bending strength. Vertical cracks, in general, do not affect shear strength, unless they are located at the compression failure zone. Horizontal cracks (along component axis) affect the shear strength more than the bending strength.

- The orientation of cracks in concrete shear walls determines whether cracks affect the seismic capacity of components. Cracks would affect the shear capacity if they coincide with cracks caused by applied seismic loads or when they alter the failure mode.

- The size and number of cracks indirectly affect the seismic performance of all concrete structural members since the extent of corrosion is largely affected by crack size.

- There are indications that some initial levels of corrosion of steel reinforcement would increase the flexural strength of beams.

Organization for Economic Co-operation and Development (OECD) – Nuclear Energy Agency (NEA)

The NEA is an intergovernmental body within the OECD located in Paris, France. The objective of the Agency is to contribute to the development of nuclear energy as a safe, environmentally acceptable, and economical energy source through co-operation among its participating countries. Currently there are 27 countries including the United States that are members of the NEA. One of the committees within NEA, the Committee on the Safety of Nuclear Installations (CSNI) has a Principal Working Group PWG-3 which is entitled "Integrity of Structures and Components."

PWG-3 has the mandate of studying the integrity of components, systems and structures and to propose general principles on the optimal ways of dealing with challenges to integrity in particular from aging. Specifically the mandate is:

- to exchange views on generic technical aspects of integrity and aging of components and structures, and follow and take account of, as necessary, national and international programs concentrating on research, operational aspects and regulation;

- in the relevant technical areas, stimulate the establishment of new required research and recommend possible international co-operative projects;

- to develop common technical positions on specific integrity issues and to identify areas where further work is needed;

- to discuss the potential impact of aging and other challenges to integrity on the safety, regulation and operability of nuclear power plants.

An OECD - NEA Workshop on FE Analysis of Degraded Concrete Structures was sponsored by the U.S. NRC and the OECD-NEA. This workshop was held at BNL on October 29-30, 1998. During the workshop over seventeen papers were presented related to the topic of the workshop. Many of the papers described technical approaches to utilize FE analysis methods for degraded concrete structures. A list of CSNI reports produced by or relevant to PWG-3 subgroups on the aging of concrete structures and the seismic behavior of structures is shown in Table 3-1.

3.5 Degradation Reference Database

To aid the process of collecting and reviewing the various documents related to aging degradation of structures and passive components, a Degradation Reference Database (DRD) was created. The DRD includes the codes, industry standards and guidelines, NUREG reports, technical papers, presentations (at conferences), regulatory documents, and other reports that were collected and reviewed in Phase I of this research program. The regulatory documents include 10 CFRs; NRC generic correspondences such as IEs, INs, GLs, etc.; NRC inspection reports; NRC regulatory guides; and NRC SECY papers.

All of the documents and summary information for each was entered into a computerized database. Currently there are over 160 documents in the database, which can be sorted in any manner, or specific documents can be located by identifying a subject of interest. A copy of this database is presented in Appendix B sorted by type of document.

The information contained in the database consists of the type of document, the identification or ID (document no.), title of the document, date of publication, author/organization, a summary description, types of components covered, and potential aging issues identified in the document.

Since the DRD, like the DOD described earlier in Section 2.3 of this report, was created using the Microsoft program ACCESS, a copy of all the data is available on floppy disks, which would allow any user to get access and sort or locate specific information on aging degradation of structures and passive components.

Table 3-1 List of CSNI Reports Produced by or Relevant to PWG-3 Subgroup on Aging Of Concrete Structures and the Seismic Behavior of Structures

Concrete

NEA/CSNI/R(99)1	Finite Element analysis of degraded concrete structures - proceedings of workshop at BNL, October 1998
NEA/CSNI/R(98)6	Development priorities for NDE of concrete structures in nuclear plant
NEA/CSNI/R(97)9	Proceedings of workshop on loss of tendon prestress in NPP containments, Civaux, August 1997
NEA/CSNI/R(99)11	NPP containment prestress loss - summary statement
NEA/CSNI/R(95)19	Report of the task group reviewing national and international activities in the area of ageing of NPP concrete structures
NEA/CSNI/R(95)25	Containment by-pass and leaktightness (PWG-4/CAM)
NEA/CSNI/R(97)28	Proceedings of workshop on development priorities for NDE of concrete structures in nuclear plants, Risley, November 1997

Seismic

NEA/CSNI/R(98)5	Status report on seismic re-evaluation
NEA/CSNI/R(96)10	Seismic shear wall ISP - NUPEC's seismic ultimate dynamic response test - comparison report
NEA/CSNI/R(95)19	Report of the task group reviewing national and international activities in the area of ageing of NPP concrete structures
NEA/CSNI/R(97)22	State of the art report on the current status of methodologies for seismic PSA

4 RISK SIGNIFICANCE OF AGING EFFECTS

This section of the report presents an overview of past probabilistic risk assessments (PRAs) with respect to the effects of age-related degradation of components and describes the ranking of components according to their risk significance. First, past internal event PRA studies on the effects of aging are reviewed regarding the key analysis methodologies and the estimated component ranking. Second, available seismic PRA studies, which addressed the aging of structures and passive components, are reviewed to single out the technical issues that may require further study. Lastly, based on a survey of a large number of past seismic PRAs (without aging consideration), structures and components are identified that are potential dominant risk contributors. This information is used as input to the priority ranking of structures/ components discussed in Section 5.2.2.

4.1 Aging Effects on Random Failures

In past internal-event PRA studies of the effects of age-related degradation on the calculated plant risk, the main focus was on the degradation of active components, such as generators and pumps, and the optimization of maintenance/surveillance programs for such components (e.g., Vesely June 1990, NUREG/CR-6415, and NUREG/CR-6157). Although these research findings are not directly applicable to the evaluation of the age-related degradation of structures and passive components, some analysis methodologies for risk quantification can be utilized in future efforts on the application of seismic PRA approaches.

In past seismic PRA studies, random failures (non-seismic failures) of active components, such as diesel generators and service water (SW) pumps, were often identified as dominant risk contributors. Therefore, the available information on the risk significance of the age-degradation of active components, such as those discussed below, may need to be incorporated in a seismic PRA study on the aging effects. By accounting for both random and seismic failures, it may be possible to quantify the relative significance of the effects of aging of structures and passive components with respect to those of random failures of active components.

To quantify the effects of age-related degradation on active components, the so-called linear aging model has been used extensively in the past PRA studies (e.g., NUREG/CR-6415). In this model, the failure rate, or hazard function, of a component, $\lambda(t)$, is expressed as a sum of two independent failure rates, one associated with random failure, λ_o, and the other associated with failures due to aging, αt, as,

$$\lambda(t) = \lambda_o + \alpha t \qquad (1)$$

Typical values for the above constant, α, are 1.0 E-5 to 1.0 E-7 per hour per year for pumps (NUREG/CR-6415). The underlying assumptions for the above linear aging model are:

(1) The component failure rate is proportional to the amount of deterioration, D,

$$\lambda(t) = \kappa D \qquad (2)$$

(2) Both the occurrence time and the severity of deterioration are considered to be random.

(3) The occurrence of deterioration is described by a stationary Poisson process.

A direct application of the above linear aging model to structures and passive components under seismic loads may not be possible because the linear damage accumulation, in Eq. (2), does not lead to a linear increase of failure rates (NUREG/CR-6157). Modifications to the above model may need to be considered, such as by accounting for a possible nonlinear relationship between the degree of deterioration and the failure rate (or component fragility), when an application to structures and passive components is considered.

The methodologies of sensitivity analyses, used to quantify the impact of aging of each component, are briefly outlined below. Although the described methodologies have been used primarily for internal event PRAs, the basic formulations may also be applicable to a seismic PRA. First, the CDF value, C, is expressed as a function of component failure rates, as,

$$C = f_1(q_i) \tag{3}$$

Similarly, the change in the CDF, ΔC, is also expressed as a function of the changes in the failure rates due to aging, Δq_i,

$$\Delta C = f_2(\Delta q_i) \tag{4}$$

The standard Taylor expansion of ΔC produces the following:

$$\Delta C = \sum_i S_i \Delta q_i + \sum_{i>j} S_{ij} \Delta q_i \Delta q_j + \sum_{i>j>k} S_{ijk} \Delta q_i \Delta q_j \Delta q_k + \cdots + S_{123\ldots n} \Delta q_1 \Delta q_2 \Delta q_n \tag{5}$$

in which, the sensitivity coefficients, S_i, S_{ij}, are defined as,

S_i = the change in risk per unit change in q_i due to individual aging effects in component *i*

S_{ij} = the change in risk per unit change in $q_i q_j$ due to multiple aging effects in components *i, j*

S_{ijk} = the change in risk per unit change in $q_i q_j q_k$ due to multiple aging effects in components *i, j, k*

$S_{12\ldots n}$ = the change in risk per unit change in $q_1 \; q_2 \cdots q_n$ due to multiple aging effects in 1,2...n

The above first order coefficients, S_i, are also called the Birnbaum importance measure. When only the first-order terms are considered in Eq. 5, and also by assuming the foregoing linear aging model, the change in CDF due to aging of the *i*-th component is expressed as,

$$\Delta C = S_i * \alpha \frac{L^2}{2} \tag{6}$$

in which, L is the overhaul interval. Based on four internal event PRAs, the risk contributions of component aging were estimated as listed in Table 4-1 (NUREG/CR-5248). In this table, the components with a rank of 5 (in the last column) represent the highest contribution to the increase in CDF-value due to their aging; and those with a rank of 1 represent the lowest contribution. This type of information could be useful to evaluate the risk significance of the effects of aging on seismic failures relative to random failures. For example, the contribution of the seismic failure of a structural component to the increase in CDF value due to aging may be calculated by formulations similar to those described above, and compared with the contributions by random failures such as those listed in Table 4-1. Such a comparison may be used to prioritize maintenance/repair programs on risk-informed bases.

4.2 Aging Effects on Seismic Failures

It appears that the work by Ellingwood, et al (NUREG/CR-6425) is the only PRA study that could be identified, which directly addressed the impact of aging of structures and passive components. The analysis, which was based on the past seismic PRA study on the Zion NPP, utilized some assumptions/judgements to assess the effects of aging on the structural seismic fragilities. However, key elements necessary to address the aging effects on plant risk were described in detail.

In the described seismic PRA, a simplified Boolean equation, which consists of eleven seismic failures as the basic events, was directly used for the risk quantification for the Zion NPP (NUREG/CR-6425). Non-seismic failures were not included in the analysis. To account for the age-related structural deterioration, reductions in the median fragility of 10% to 47% were estimated for shear walls and roofs based on a worst case scenario, such as rebars becoming completely ineffective due to corrosion. The calculated CDF increased by a factor of about 2.0 due to aging. The Vesely-Fussell and Birnbaum importance measures were used in a sensitivity analysis. Also, the time-dependent changes in plant risk were evaluated by assuming a single component failure (auxiliary building shear wall failure).

Since no other studies are currently available in this area, a number of technical issues appear to remain unresolved regarding the risk significance of aging of structures and passive components, including:

(1) The effects of age-related degradation on seismic response are not well understood at this point for various structures/components. The potential changes in fragility values need to be quantified in a format that can be used in risk quantification.

(2) A more complete plant logic model needs to be used in a seismic PRA to account for both random failures and seismic failures. From a viewpoint of prioritizing the maintenance/overhaul program, the relative risk significance of the aging effects between passive and active components may need to be quantified.

(3) The seismic demands for equipment may be altered because the age-related cracking in buildings could affect the building stiffness, and therefore the floor responses. Past studies (e.g., NUREG/CR-5407) indicate that such effects may not be negligible.

(4) Due to aging, some components that had been neglected in the original plant risk model may become a non-negligible risk contributor, such as major passive components (NUREG/CR-6157).

4.3 Relative Risk Significance of Structures and Components

In a seismic PRA, the relative risk significance of structures and components are quantified by various importance measures. Some of the frequently used measures are outlined below.

<u>Birnbaum Importance</u>.....As described earlier, the Birnbaum importance measure is the first order sensitivity coefficient of Eq. 5, S_i.

$$I^B = \frac{\partial C}{\partial p_i} \quad (7)$$

Although some software programs may automatically calculate this measure, it is considered to be a poor indicator of relative risk contribution because highly important, but highly robust, passive components will have a high Birnbaum importance.

Vesely-Fussell Importance..... The ratio of the CDF-value of the sum of all the cutsets containing a component to the total CDF. This importance measure has been used most frequently in the past seismic PRA's.

Risk Achievement Worth Measure..... The ratio of the increase in CDF by setting the component capacity to zero (failure ratio of one) to the original CDF value.

Risk Reduction Worth Measure..... The ratio of the decrease in CDF by setting the component capacity to infinity (failure ratio of zero) to the original CDF value. This measure is a poor indicator for most components and equipment, except for components that represent a single cutset such as major building collapse.

In the past two decades, seismic PRA studies have been carried out on a large number of NPP's, including the most recent studies as part of the independent plant examination of external events (IPEEE) (NUREG-1407). Surveys on the seismic fragility values used in the past seismic PRA's are also available in numerous publications (e.g., NUREG/CR-4334, Kipp 1988, Cambell 1988, NUREG/CR-3558 1985, and Park December 1998). As an example of such surveys, fragility values and dominant failure modes are tabulated in Table 4-2 for various components. This type of information is useful to identify the components with a relatively low seismic capacity (and therefore, a potential risk contributor).

Typically, 5~6 components are singled out as the dominant risk contributors as a result of a seismic PRA study. Based on a survey of a large number of past seismic PRA's (including those of IPEEE), structures and components identified as dominant risk contributors are listed in Table 4-3 (Park, 1997). Based on this listing and the data in Table 4-2, the following types of structures and passive components may be considered to be the most frequently observed weak links:

- Anchorage and supports of equipment
- Flat-bottom storage tanks
- Critical reinforced concrete members
- Concrete block walls
- Interconnecting pipes (e.g. buried piping)
- Cable trays
- Dams

The information described in Tables 4-2 and 4-3 contributed to the determination of the priority ranking of structures/components, which is described in Section 5.2 of this report.

Table 4-1 Factor Values and Final Ranking of Structures/Components (NUREG/CR-5248)

Component	S_i (C D/yr.)	α (hr^{-1} yr^{-1})	L (Mo.)	Risk Increase Δ C (C D/yr.)	Final Rank
Small other safety pipe	1.0E-3	3.0E-7	60.0	2.1E-3	5
Cables	1.1E-1	2.7E-9	60.0	2.1E-3	5
Containment (BWR)	1.0E-0	1.0E-7	18.0	1.9E-3	5
Connectors	2.0E-2	2.7E-8	60.0	1.8E-3	5
S/G tube	3.0E-4	5.0E-6	36.0	9.5E-4	5
Turbine pump	9.3E-3	2.7E-6	12.0	5.4E-4	4
Relay	4.8E-2	2.5E-7	6.0	4.0E-4	4
Diesel	2.0E-2	3.6E-6	3.0	2.7E-4	4
RX Internals	1.0E-1	2.0E-9	18.0	2.4E-4	4
Breaker	7.2E-2	1.6E-8	18.0	5.6E-5	3
Motor operated valve	2.2E-2	3.6E-6	3.0	5.6E-5	3
BWR pipe (small LOCA)	1.0E-3	3.0E-8	36.0	3.7E-5	3
Motor pump	6.7E-3	2.2E-7	12.0	3.2E-5	3
Large other safety pipe	6.4E-3	3.0E-9	18.0	2.3E-5	3
Thermostat	6.0E-3	1.5E-7	18.0	2.2E-5	3
Chillers	6.0E-4	1.5E-6	18.0	2.2E-5	3
RPV	1.0E-0	2.0E-12	120.0	1.4E-5	3
Battery	2.0E-2	3.4E-7	6.0	1.1E-5	3
Compressor (instr. air)	5.0E-4	5.0E-7	6.0	8.4E-6	3
Air operated valve	3.2E-4	4.0E-7	18.0	6.2E-6	2
DC bus	1.1E-1	1.1E-9	18.0	5.9E-6	2
CRDM (BWR)	1.0E-1	3.0E-9	18.0	4.5E-6	2
Check valve	8.0E-4	3.8E-9	18.0	3.7E-6	2
Fan	6.0E-4	2.1E-7	18.0	3.1E-6	2
Heat exchanger	6.4E-3	1.4E-8	3.0	3.9E-6	2
Bolts	1.0E-4	5.1E-7	18.0	2.5E-6	2
AC bus	4.4E-2	1.1E-9	18.0	2.3E-6	2
Safety/relief valve	1.0E-4	6.7E-7	18.0	1.0E-6	2
Containment (other)	1.0E-0	1.0E-13	60.0	7.0E-7	2
Other concrete structures	1.0E-0	1.0E-13	60.0	7.0E-7	2
Transformer	1.2E-2	1.7E-9	18.0	5.1E-7	1
Inverter	4.7E-6	4.9E-6	12.0	5.0E-7	1
Transfer switch	4.7E-6	2.3E-7	18.0	3.3E-7	1
Snubbers	1.1E-6	5.1E-6	18.0	8.4E-8	1
Hydraulic valve	1.0E-5	1.3E-7	18.0	6.3E-8	1
Turbine	1.0E-6	1.0E-7	60.0	5.4E-8	1
Bistable	1.2E-5	1.4E-7	18.0	4.2E-8	1
Manual valve	1.0E-5	2.2E-9	60.0	2.7E-8	1
Battery charger	1.1E-4	3.5E-8	12.0	2.6E-8	1
Tank (atmos. pres.)	2.5E-2	2.0E-12	12.0	2.2E-8	1
Rectifier	4.7E-6	8.7E-8	12.0	8.9E-9	1
Tank (medium pres.)	6.0E-3	1.0E-12	12.0	5.4E-9	1
CRDM (PWR)	1.0E-3	3.0E-11	18.0	1.5E-9	1

Table 4-2 Summary of Fragility Database (Park 1998)

	Category Name	Dominant Failure Mode	Median Fragility Range (g)			
			Past PRA[a]		Other[b]	
			N[c]	Range	N[d]	Range
1	Concrete containment	Shear failure	8	2.50-9.20	-	-
2	Steel containment	Shell wall buckling	1	9.00-9.00	-	-
3	Reactor pressure vessel	Anchor bolt	13	1.04-5.70	3	3.83-3.83
4	Steam generator	Support	9	1.70-6.80	4	2.45-2.45
5	Reactor coolant pump	Support	8	0.90-4.60	3	2.64-2.64
6	Recirculation pump	Bracket	6	0.90-2.20	-	-
7	Core assembly	CRD housing	21	0.60-6.71	5	2.06-2.06
8	Pressurizer	Lateral support	1	5.73-5.73	1	2.00-2.00
9	Piping	Support	14	2.50-13.6	-	-
10	Valves	Yoke support	35	0.80-13.7	22	4.83-20.5
11	Heat exchanger	Anchor bolt	23	0.30-13.0	4	1.00-1.18
12	Flat bottom tank	Shell wall buckling	17	0.20-1.00	8	0.45-2.01
13	Other tanks and vessels	Anchor bolt	28	1.00-46.0	13	1.07-3.91
14	Batteries and racks	Battery cases and plates	16	0.90-5.95	54	0.80-7.30
15	Motor control center	Chattering	17	0.06-4.20	70	0.30-7.63
16	Switchgears	Chattering	26	0.40-6.90	60	2.33-8.50
17	Diesel generator	Anchor bolt	14	0.70-3.89	9	0.65-1.00
18	Large vertical pumps	-	9	0.80-7.50	2	2.21-2.21
19	HVAC	Deflection of fan	14	1.10-5.58	5	2.24-6.90
20	Cable tray	Support	9	1.10-5.80	3	2.23-2.23
21	Other pumps	Anchor bolt	8	2.10-5.47	3	2.80-3.19
22	Motors	-	-	-	2	12.1-12.1
23	Transformers	Support	4	0.30-5.80	5	2.78-8.80
24	Instruments	Accuracy	1	4.46-4.46	52	1.15-16.3
25	Electric panels	Chattering	11	2.77-7.60	116	1.66-11.5
26	Light fixtures	-	-	-	1	9.20-9.20
27	Communication equipment	-	-	-	1	5.00-5.00
28	Inverters	Function failure	1	3.41-3.41	5	2.00-15.6
29	Circuit breakers	Trip	6	0.37-4.20	2	7.63-7.63
30	Ceramic insulators	-	1	0.20-0.20	9	0.20-
31	Pipe supports	-	-	-	1	1.46-1.46
32	Offsite power	Ceramic insulator	29	0.20-0.62	-	-
33	Electric cabinets	Chattering	2	1.10-3.88	7	7.50-7.63
34	Ducts	Support	1	4.89-4.89	5	3.97-3.97
35	Electrical penetration	Pressure loss	1	3.69-3.69	4	12.0-12.0

[a] Ground motion PGA.
[b] Local floor ZPA/Average Spectral Acceleration values.
[c] Number of fragility values.
[d] Number of Original data (e.g. test data).

Table 4-3 Dominant Risk Contributors Identified in Past Seismic PRA's (Park 1997)

<u>Civil Structures</u> • Shear wall failure • Roof/slab failure • Soil liquefaction • Unreinforced masonry walls • Ceiling failure in control room • Turbine building collapse • Impact between buildings • Stack failure <u>Diesel Generator</u> • Fuel oil (day tank) • Random failure of diesel generator • Oil Cooler <u>Emergency Feedwater</u> • E.F. pump • Random failure of E.F. • Condensate storage tank <u>Piping</u> • Interconnecting pipes	<u>Power Supply</u> • Offsite power loss • 125 v DC batteries/racks • 125 v DC distribution panels • 125 v DC fuse box • 250 v DC motor control center (MCC) • 4 kv switchgear • 4kv busses • Transformers • Cable trays <u>Reactor Coolant System</u> • Pressurizer supports • Control system drive system/housing • Excessive deflection of core/core shroud • Reactor coolant pump support • Random failure of Pressurizer SRV • Seal failure of RCP <u>Service Water</u> • SW pump • Dam failure (ultimate heat sink)

5 SCOPING STUDY – ASSESSMENT OF TECHNOLOGY NEEDS; DEVELOPMENT OF PHASE II PROGRAM SCOPE

One of the objectives of the Phase I effort is to perform a scoping study to identify the important structures and passive components most susceptible to age-related degradation and to define the scope of any additional research that may be needed. Sections 5.1 through 5.3 below describe the scoping study performed to identify the technology needs and the selection process for identifying which structures and passive components need to be reviewed in greater detail. A description of the recommendations for performing further research is presented in Section 6.

5.1 Site Visits – Review of License Renewal Inspection Reports

The NRC staff has performed inspections at the Oconee and Calvert Cliffs nuclear power plants to verify compliance with the requirements of 10 CFR Part 54 for license renewal. The inspection results or findings provide valuable first hand observations of the effects of age-related degradation and assessments of the adequacy of licensee programs to monitor and manage these effects. Although the inspections were intended to verify compliance with the rule and to ensure that the applicant's license renewal program is consistent with the license renewal application, the inspections did include assessments of some structures and passive components, pertinent to this study.

5.1.1 Calvert Cliffs

Inspections were performed at the Calvert Cliffs Nuclear Power Station during the period April 5 – 16, 1999 and are documented in NRC Inspection Report Nos. 50-317/99-04 and 50-318/99-04. The inspections concentrated on potential and plausible aging effects and the management of those effects. The inspection team reviewed maintenance records, performed system walkdowns and reviewed design, function, documentation, and aging management programs for selected systems, structures and related commodity groups and for selected aging effects. The structures and passive components reviewed pertinent to this study included the control room and diesel generator building HVAC systems, component supports, water intake structure, primary containment building, auxiliary building, and safety-related diesel generator building structure. The aging effects/mechanisms reviewed include corrosion (crevice, pitting, microbiologically induced corrosion (MIC) and general); aggressive chemical attack of concrete; wear of components; weathering of structures; stress corrosion cracking of bolting; dynamic loading; and the degradation of Boraflex and Carborundum.

The inspection report does not provide a comprehensive description of all instances of aging degradation observed. Instead only those aging effects which were observed but not properly addressed in the licensee's aging management programs were reported.

With that qualification the following observations were made:

<u>Containment</u>

Concrete cracks were observed at two containment buttresses of Unit 1. If not addressed, such cracks may lead to corrosion of embedded steel reinforcement. Minor leaching of calcium hydroxide was noted on the

Unit 2 concrete containment dome in about five locations.

Minor chips and flaking of the containment liner paint were observed. The expansion joint material between the floor slab and the cylinder liner plate was previously replaced because the old joint had degraded.

Auxiliary Building

Diagonals through wall cracks were observed on the concrete wall in the fan room in the auxiliary building. Also, in a licensee report addressing the operability of a support, a crack running diagonally under the support was evaluated and determined to be caused by initial settlement of the auxiliary building.

Component Supports

Based on a review of various licensee documents it was determined that repeated water hammer and thermal expansion transient events had occurred at the plant and that the licensee's attempts to control these events have not been successful. This was evident by the repeated damage to the Unit 1 low pressure safety injection support R-16. It was therefore determined that the License Renewal Application should be revised to consider the effects of repeated water hammer and thermal expansion transients on piping supports as plausible aging effects for these components. The concern is that systems repeatedly subjected to these events can accumulate aging effects such as bending of hangers and damage to the piping system

Water Intake Structure

The visual inspection of the fluid retaining walls and slabs of the Unit 2 water intake cavities showed normal concrete aging.

Buried Piping

In response to NRC Generic Letter GL 89-13, the licensee committed to inspect and repair its underground piping to address the aging effects caused by saltwater in the Service Water System piping. The underground piping is cast iron with an internal mortar lining. The outside of the underground piping is insulated and protected from the soil by layers of wrap, enamel coating, and cathodic protection. More than 600 areas of defective mortar were found ranging in size from a quarter to a couple of square feet. Although graphitic corrosion and other aging effects were occurring, ultrasonic testing confirmed that the pipe wall thickness was still above minimum design wall thickness requirements.

5.1.2 Oconee

Inspections were performed at the Oconee Nuclear Station, Units 1, 2, & 3, during the period June 2 through July 30, 1999 and are documented in NRC Inspection Report Nos. 50-269/99-12, 50-270/99-12 and 50-287/99-12. In the inspections, the aging/degradation of five categories of structures and passive components pertinent to this study were assessed. These included; safety-related reinforced concrete buildings and water-control structures, foundations, anchorage's, masonry walls and buried piping. The areas walked down were the exterior walls of the containment, auxiliary and turbine buildings, intake structure, tendon gallery, switchyards and cable trenches. All structures were determined to be in acceptable condition. The documents reviewed included: the License Renewal Application, technical basis and inspection program documents, Problem Investigation Process (PIPs) and Repair Work Orders as well as previous NRC and licensee inspection reports.

Although all structures were determined to be in acceptable condition, the following evidence of aging degradation pertinent to this study were noted by the NRC Structural Inspector:

<u>Concrete</u>

Hairline cracks in the range of 0.254 mm (0.01 in.) to 1.016 mm (0.04 in.) were observed on concrete surfaces. Very few cracks more than 1.016 mm (0.04 in.) in size, not already noted in licensee inspection reports, were noted. These were already identified in their 5-year civil/structural inspection program (1997/1999). The licensee evaluated the cracks and a PIP was initiated or a Repair Work Order was issued.

Leaching was noted in a few places. Small levels of leaching are monitored by various periodic programs and significant leaching is repaired.

<u>Structural Steel</u>

Corrosion was noted in a few places for anchorage/embedments (exposed surfaces) for which Repair Work Orders were issued.

<u>Masonry Walls</u>

All masonry walls observed appeared to be in good condition. The masonry walls at Oconee are reinforced.

5.2 Technology Needs and Priority Ranking of Structures and Passive Components

5.2.1 Technology Needs

In order to gain an understanding as to the technology needs and which structures and components require further assessments, a review was conducted of what NRC and industry programs exist and how well they are addressing aging degradation. The programs reviewed covered NRC and industry requirements, as well as NRC and industry research related to aging degradation of structures and components at NPPs.

To facilitate this review and presentation of the results, a table was developed in matrix form for each category such as anchorages, tanks, and reinforced concrete structures. From the original eighteen categories shown in Table 2-1, eight structures and passive components were selected for this assessment of technology needs. The other ten categories were eliminated because there were either relatively few degradation occurrences identified in the Degradation Occurrence Database (DOD) or it is well known that there are existing programs that adequately address aging concerns for these items.

The eight types of structures and passive components that were assessed are presented in Table 5-1. For each of the structures and components, the NRC and/or Industry program that relates or addresses aging concerns are tabulated along with a summary of whether the programs adequately address aging. While this table may not list every single program, it does capture the major requirements, research programs, and industry programs that address aging.

To identify whether the NRC and industry programs adequately address aging, the programs are classified in Table 5-1 by the designation yes, no, partially, or uncertain along with a summary explanation. The term uncertain is used if based on the available information at this time, it is not clear whether the program does or does not adequately address all of the aspects of age-related degradation of the particular structure/component.

5.2.2 Priority Ranking of Structures/Components

To identify which structures and components warrant further review in subsequent phases of the research program, it was decided to rank or prioritize them. The process of ranking the eight structures and passive components considered four key parameters: seismic risk significance, degradation occurrences, importance to current licensing basis/license renewal, and adequacy of existing NRC and industry programs. Then a final ranking was developed based on a compilation of all the information from these four key parameters.

Table 5-2 presents the prioritization of the eight structures and passive components based on the four key parameters identified above. The approach used to rank the different categories using these four parameters is discussed below.

1. <u>Seismic Risk Significance</u>

The contribution of risk significance and in particular seismic risk significance was discussed previously in Section 4 of this report. Based on the information presented in Tables 4-2 and 4-3 of Section 4.3, it was concluded that the following types of structures and passive components may be considered to be the most frequently observed weak links in terms of seismic risk assessments:

- Anchorage and supports of equipment
- Flat-bottom storage tanks
- Critical reinforced concrete members
- Concrete block walls
- Interconnecting pipes (e.g. buried piping)
- Cable trays
- Dams

The above list of structures and components is consistent with the eight structures and components identified in Table 5-2 with the exception of dams. Based on the limited number of degradation occurrences identified for dams in the DOD and the existence of Regulatory Guide 1.127, Rev. 1 for the inspection of water-control structures, dams were excluded from the evaluation reported in Table 5-2. Regarding cable trays, the supports for cable trays are included in Table 5-2 under the heading "supports for equipment and systems."

To rank the eight structures and components listed in Table 5-2 in terms of seismic risk significance, three sources of information were examined. The three sources consisted of seismic HCLPF (high confidence - low probability of failure) values, IPEEE results pertaining to seismic PRAs, and IPEEE results pertaining to SMAs (seismic margins assessments). The results of this assessment are shown in Table 5-3. This table presents the seismic HCLPF values (median, minimum, and maximum) for the eight structures and components based on past industry reports (Park 1998, NUREG/CR-4334, and EPRI NP-4101-SR). To rank the structures in terms of contributors to CDF or weaknesses in seismic PRAs, the IPEEE results presented in the NRC memo "Preliminary IPEEE Insights

Report," 1998 were reviewed. The NRC memo on IPEEE results was also utilized to rank the structures in terms of controlling components in SMAs.

Seismic risk significance is used as a factor in ranking structures and passive components because seismic events have been found to be an important contributor to core damage frequency. This is exemplified by the information presented in Table 5-4.

The last column in Table 5-3 presents the overall seismic risk significance based on a compilation of the other three columns. The overall seismic risk significance for each of the structures and components was rated using qualitative measures in view of the limited quantitative data that were available. The level of seismic risk significance was identified as very high, high, moderate, low, or insufficient data. Masonry walls and flat bottom tanks were rated as very high and anchorages, buried piping, and supports for equipment and systems, were rated as high.

2. <u>Degradation Occurrences</u>

The second key parameter used to rank the eight structures and components is the number of degradation occurrences. This information was obtained from the DOD. Actual number of degradation occurrences for the original eighteen categories are shown in Table 2-1. However for purposes of ranking the eight structures and components in Table 5-2, the use of quantitative data may be misleading and so qualitative measures such as very high, high, moderate, low, or unknown are used. Using the actual number of occurrences might be misleading because for some categories of structures and passive components such as concrete there are many concrete members/structures in a plant while for other categories such as tanks there are much fewer. While no category received a very high rating for degradation occurrences, masonry walls, flat bottom tanks, anchorages, concrete containments, and steel containments were rated as high. These ratings were assigned based on the number of occurrences for a given category considering the number of such structures or components that typically exist at NPPs.

3. <u>Importance to Current Licensing Basis/License Renewal</u>

The third key parameter used to rank the eight structures and components listed in Table 5-2 is the importance to current licensing basis and license renewal. This was based on discussions with NRC staff in NRR and RES as well as BNL's past and current experience and participation in related NRC activities. Once again qualitative measures were assigned using very high, high, moderate, or low. Very high rating was given to concrete because of the need to validate existing criteria and where necessary develop additional acceptance criteria for concrete, particularly cracks in concrete.

4. <u>Adequacy of Existing NRC and Industry Programs</u>

The fourth key parameter used to rank the structures and passive components in Table 5-2 is the adequacy of existing NRC and industry programs in addressing the concerns related to degradation. The process to evaluate the adequacy of these programs was described earlier in Section 5.2.1 under the heading "Technology Needs." The results of this evaluation for each program was combined to develop a single measure for each of the eight structures/components. The final rating, Yes, No, or Uncertain, was then included in Table 5-2.

5.2.3 Final Ranking of Structures and Passive Components

Table 5-2 presents the ranking for the eight structures and passive components using the four key parameters described above. The last column entitled, "Priority Ranking for Further Study" represents the compilation from all of the other parameters. Masonry walls (particularly unreinforced walls) and flat bottom steel tanks were rated as very high followed by anchorages, concrete, and buried piping which were rated as high. Supports for equipment and systems were rated as moderate and concrete and steel containments were rated as low. It should be noted that a rating of low for example does not mean the structure or component is not important or does not experience age-related degradation but rather, relative to the other structures and components, it is not ranked as high. This occurs because several of the key parameters such as seismic risk significance or adequacy of existing NRC/industry programs result in its lower ranking.

5.3 Preliminary Study of Concrete Degradation

The effects of aging-related degradation on seismic performance of reinforced concrete members are not well understood at this point because the past experimental work in this area is rather limited. To understand the mechanism of the interaction between aging degradation and seismic responses, as well as to possibly quantify the effects of aging degradation, the application of nonlinear finite element (FE) analysis to degraded concrete members was attempted. The results of the analyses are described in detail in Appendix C to this report.

To model the existing pre-cracks, such as those caused by corrosion of rebars and alkali-silica reaction, the use of smeared crack models is considered to be inappropriate because arbitrary external forces need to be applied on the analysis model to cause cracking. A discrete crack model is used to simulate existing cracks. The new cracks induced by the seismic loading are then superimposed onto the existing cracks using the smeared crack model.

One of the artificially degraded shear walls tested by the Ryukyu University was analyzed to study the feasibility of reproducing the seismic responses of degraded reinforced concrete (RC) structures by nonlinear FE analysis.

The comparison of analyses and test results indicated that the analysis assumptions did not fully reflect the actual degradation conditions. Further efforts seem to be necessary, such as a more accurate characterization of the changes in material properties for a better correlation.

Based on the observations of the preliminary application of nonlinear FE analysis, the following areas are singled out for further efforts:

(1) The discrete crack model needs to be better defined and calibrated to account for the nonlinear behavior of aggregate interlocking, dowel action, and bond-slippage under cyclic loading reversals.

(2) Properly accounting for the bond mechanism seems to be a key to successfully reproduce the observed complex phenomena of degraded RC components under seismic loads (e.g., the increase in stiffness in shear wall tests and the seismic performance of columns with significantly corroded reinforcement). The modeling of bond mechanism needs to be improved.

(3) The changes in material properties of degraded RC components need to be quantified, including the compressive/tensile strength and modulus of elasticity of concrete.

Table 5-1 Adequacy of Aging Programs – Unreinforced Masonry Walls

NRC PROGRAM	NRC GUIDANCE / ADDL. REQMTS.	INDUSTRY PROGRAMS	ADEQUATELY ADDRESSES AGING YES/NO
IE Bull. 80-11 Masonry Wall Design	Appendix A to SRP Section 3.8.4 Interim Criteria for Safety-Related Masonry Wall Evaluation NRC technical positions on post-cracking evaluation methods	Initially, Industry Group proposed acceptable evaluation methods. Individual plant submittals in response to IE Bull. 80-11	**No**: Did not address continued aging effects. No follow-up required after 80-11 resolution. Over 10 years since completion of program.
IN 87-67 Lessons Learned From Regional Inspections of Licensee Actions in Response to IE Bull. 80-11			**Uncertain**: Information Notice identified cracks in unreinforced masonry walls that were not accounted for. No specific action or written response was required. Note: For License Renewal, a masonry wall aging management program which follows the recommendations of IN 87-67 for maintaining the IE Bulletin 80-11 structural qualification basis is currently being evaluated for acceptability.
10 CFR 50.65 Maintenance Rule	R.G. 1.160, Rev. 2 NUREG-1526 Insp. Proc. 62706 Insp. Proc. 62002 NRC Letter "NRC Comments on NEI 96-03, Rev. D, ..." Oct. 1, 1996 SECY-97-055	NUMARC 93-01, Rev. 2 NEI 96-03, Rev. D Individual plant responses to Rule	**Uncertain:** Scope includes masonry walls but NRC Letter has comments on NEI 96-03 & per SECY, NRC "inspectors did not assess or were unable to conclude whether the licensee's program for monitoring structures complied with the Rule." Specific criteria are established by each licensee – not submitted for technical review. NEI 96-03 identifies cracks in masonry walls need to be considered but does not provide any inspection, acceptance, or evaluation criteria. Note: For License Renewal, Maintenance Rule structures monitoring of masonry walls which is conducted in accordance with 10 CFR 50, Appendix B – Quality Assurance; follows the

Table 5-1 Adequacy of Aging Programs – Unreinforced Masonry Walls (Continued)

			guidance in R.G. 1.160, Rev. 2 and NUMARC 93-01, Rev. 2; and utilizes the acceptance criteria of IE Bulletin 80-11 is currently being evaluated for acceptability.
IPE/IPEEE	GL 88-20, IPE For Severe Accident Vulnerabilities GL 88-20, Supplement No. 4, IPEEE For Severe Accident Vulnerabilities NUREG-1407	Individual plant submittals	**No:** Limited to success path(s) components only. Does not cover all safety related items. Does not address aging degradation in guidance documents or licensee IPEEE submittals.
		ASCE 11-90 Std., Guideline for Structural Assessment of Existing Buildings	**Partially:** Primarily provides guidance for assessing (identifying, quantifying using tests, & reporting) the physical and material condition. Makes reference to other standards & documents. Does not provide guidelines on analysis of aged condition, acceptance criteria, and applicability to NPPs.
NUREG-1522	N/A	N/A	**Partially:** Limited review of older plants. Scope was to identify aging degradation of structures and civil engineering features.

Table 5-1 Adequacy of Aging Programs – Tanks

NRC PROGRAM	NRC GUIDANCE / ADDL. REQMTS.	INDUSTRY PROGRAMS	ADEQUATELY ADDRESSES AGING YES/NO
USI A-46 "Seismic Qualification of Equipment in Operating Plants"	NRC GL 87-02, Supplement No. 1 Transmitting SSER No. 2 NUREG-1030 NUREG-1211	GIP-2 & EPRI reports referred to within	**No**: Limited to USI A-46 plants. Limited to one "safe shutdown" path & consideration of a single equipment failure. Only a one-time assessment. No specific requirement to consider aging.
10 CFR 50.65 Maintenance Rule	R.G. 1.160, Rev. 2 NUREG-1526 Insp. Proc. 62706 Insp. Proc. 62002 NRC Letter "NRC Comments on NEI 96-03, Rev. D, …" Oct. 1, 1996 SECY-97-055	NUMARC 93-01, Rev. 2 NEI 96-03, Rev. D Individual plant responses to Rule	**Uncertain:** Scope includes tanks but NRC Letter has comments on industry program and it is not clear how or whether the licensee's program for tanks comply with the Rule. Specific criteria are established by each licensee – not submitted for technical review. Note: For License Renewal, Maintenance Rule structures monitoring of tanks which is conducted in accordance with 10 CFR 50, Appendix B – Quality Assurance and follows the guidance in R.G. 1.160, Rev. 2 and NUMARC 93-01, Rev. 2 is currently being evaluated for acceptability.
IPE/IPEEE	GL 88-20, IPE For Severe Accident Vulnerabilities GL 88-20, Supplement No. 4, IPEEE For Severe Accident Vulnerabilities NUREG-1407	Individual plant submittals	**No:** Limited to success path(s) components only. Does not cover all safety related items. Does not address aging degradation in guidance documents or licensee IPEEE submittals.
NUREG-1522	N/A	N/A	**Partially:** Limited review of older plants. Scope was to identify aging degradation of structures and civil engineering features.

Table 5-1 Adequacy of Aging Programs – Concrete Structures/Members (Continued)

Structural Aging (SAG) Program	NUREG/CR-6424	N/A	**Partially:** The program has substantial and very useful information. Certain areas should be expanded (e.g. technical basis for crack acceptance criteria applicable to NPPs, analysis methods for degraded concrete members, guidance for inaccessible areas).
NUREG-1522	N/A	N/A	**Partially:** Limited review of older plants. Scope was to identify aging degradation of structures and civil engineering features.
NRC sponsored risk assessment studies	NUREG/CR-5407 Assessment of the Impact of Degraded Shear Wall Stiffness on Seismic Plant Risk and Seismic Design Loads; and a number of other studies (see Section 4 of this report)		**Partially:** Scope covered evaluating how the existing seismic PRAs are affected by stiffness degradation due to concrete cracking associated with a seismic event. Results for 3 plants evaluated indicate increases in core damage frequencies for seismic initiated events vary from 0 to 30 percent.

Table 5-1 Adequacy of Aging Programs - Anchorages

NRC PROGRAM	NRC GUIDANCE / ADDL. REQMTS.	INDUSTRY PROGRAMS	ADEQUATELY ADDRESSES AGING YES/NO
USI A-46 "Seismic Qualification of Equipment in Operating Plants"	NRC GL 87-02, Supplement No. 1 Transmitting SSER No. 2 NUREG-1030 NUREG-1211	GIP-2 & EPRI reports referred to within	**No:** limited to USI A-46 plants. Only for mechanical & electrical equipment. Limited to one "safe shutdown" path & consideration of a single equipment failure. Only a one-time assessment. Only for accessible anchorages. Excludes NSSS equipment inside containment. Only aging effect included is size of concrete cracks near anchors.
10 CFR 50.65 Maintenance Rule	R.G. 1.160, Rev. 2 NUREG-1526 Insp. Proc. 62706 Insp. Proc. 62002 NRC Letter "NRC Comments on NEI 96-03, Rev. D, …" Oct. 1, 1996 SECY-97-055	NUMARC 93-01, Rev. 2 NEI 96-03, Rev. D Individual plant responses to Rule ACI 349.3R-96	**Uncertain:** Scope includes anchorages but NRC Letter has comments on NEI 96-03 & per SECY, NRC "inspectors did not assess or were unable to conclude whether the licensee's program for monitoring structures complied with the Rule." Specific criteria are established by each licensee – not submitted for technical review. NEI 96-03 references ACI 349.3R as an additional source which may be used (i.e. adherence not required). ACI 349.3R lacks complete criteria for anchorage degradation. Adequacy of ACI 349.3R not formally endorsed at this time by NRC. Note: For License Renewal, Maintenance Rule structures monitoring of anchorages which is conducted in accordance with 10 CFR 50, Appendix B – Quality Assurance and follows the guidance in R.G. 1.160, Rev. 2 and NUMARC 93-01, Rev. 2 is currently being evaluated for acceptability.

Table 5-1 Adequacy of Aging Programs - Anchorages (Continued)

IPE/IPEEE	GL 88-20, IPE For Severe Accident Vulnerabilities GL 88-20, Supplement No. 4, IPEEE For Severe Accident Vulnerabilities NUREG-1407	Individual plant submittals	**No:** Limited to success path(s) components only. Does not cover all safety related items. Does not address aging degradation in guidance documents or licensee IPEEE submittals.
IE Bull. 79-14 Seismic Analysis For As-built Safety-related Piping Systems		Individual plant responses to IE 79-14	**No:** Only for piping systems. Only a one-time assessment. No specific requirements to inspect for degradation. Inaccessible areas – case by case basis. Conducted ≅ 1980
IE Bull. 79-02 Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts		Individual plant responses to IE Bull. 79-02	**No:** Only for piping systems. Only a one-time assessment. No specific requirements to inspect for degradation. Sampling techniques may have been used. Conducted ≅ 1980.
		ASCE 11-90 Std., Guideline for Structural Assessment of Existing Buildings	**No:** Does not explicitly address aging of anchorages
NUREG-1522	N/A	N/A	**Partially:** Limited review of older plants. Scope was to identify aging degradation of structures and civil engineering features.
NRC/RES Program: "Anchor Bolt Behavior and Strength During Earthquakes"			**Partially:** Addresses effect of dynamic vs. static loading and effect of cracked concrete on anchor failure load. Does not address degradation of grout and steel anchors.

Table 5-1 Adequacy of Aging Programs – Concrete Structures/Members (Continued)

NRC Inspection Requirements for Water-Control Structures	R.G. 1.127		**Partially:** Addresses inspection of water-control structures such as dams, channels, intake structures, & embankments. Does not apply to concrete members in non-water control structures (e.g. roofs, floors, walls in other structures). Note: For License Renewal, an aging management program for water-control structures based on R.G. 1.127 is currently being evaluated for acceptability.
NRC Sponsored Seismic Category I Structures Program (tests on shear walls performed in 1980's)	See NUREG/CR-5407 and ASCE report (described below) for list of applicable tests and reports		**No:** Does not address aging. Results indicated that measured stiffness values from tests were lower than calculated values by as much as a factor of 4 or more.
		ASCE 11-90 Std., Guideline for Structural Assessment of Existing Buildings	**Partially:** Primarily provides guidance for assessing (identifying, quantifying using tests, & reporting) the physical and material condition. Makes reference to other standards & documents. Does not provide guidelines on analysis of aged condition, acceptance criteria, and applicability to NPPs.
		ASCE Report –Stiffness of Low Rise Reinforced Concrete Shear Walls, 1994	**No:** Does not address aging. Acknowledges that stiffnesses from tests under OBE/SSE loads could be substantially lower than computed values based on uncracked properties. Provides recommendation to use two concrete in-plane stiffness estimates – upperbound based on 1.25 x E & G and lowerbound based on .75 x E & G.

Table 5-1 Adequacy of Aging Programs – Concrete Structures/Members

NRC PROGRAM	NRC GUIDANCE / ADDL. REQMTS.	INDUSTRY PROGRAMS	ADEQUATELY ADDRESSES AGING YES/NO
10 CFR 50.65 Maintenance Rule	R.G. 1.160, Rev. 2 NUREG-1526 Insp. Proc. 62706 Insp. Proc. 62002 NRC Letter "NRC Comments on NEI 96-03, Rev. D, …" Oct. 1, 1996 SECY-97-055	NUMARC 93-01, Rev. 2 NEI 96-03, Rev. D Individual plant responses to Rule ACI 349.3R-96	**Uncertain:** Scope includes concrete members but NRC Letter has comments on NEI 96-03 & per SECY, NRC "inspectors did not assess or were unable to conclude whether the licensee's program for monitoring structures complied with the Rule." Specific criteria are established by each licensee – not submitted for technical review. NEI 96-03 references ACI 349.3R as an additional source which may be used (i.e. adherence not required). ACI 349.3R lacks complete criteria for cracks (e.g. length, depth, orientation, number of cracks & criteria as a function of environmental conditions). Adequacy of ACI 349.3R not formally endorsed at this time by NRC. Note: For License Renewal, Maintenance Rule structures monitoring of concrete structures/members which is conducted in accordance with 10 CFR 50, Appendix B – Quality Assurance and follows the guidance in R.G. 1.160, Rev. 2 and NUMARC 93-01, Rev. 2 is currently being evaluated for acceptability.
IPE/IPEEE	GL 88-20, IPE For Severe Accident Vulnerabilities GL 88-20, Supplement No. 4, IPEEE For Severe Accident Vulnerabilities NUREG-1407	Individual plant submittals	**No:** Limited to success path(s) components only. Does not cover all safety related items. Does not address aging degradation in guidance documents or licensee IPEEE submittals.

Table 5-1 Adequacy of Aging Programs – Buried Piping

NRC PROGRAM	NRC GUIDANCE / ADDL. REQMTS.	INDUSTRY PROGRAMS	ADEQUATELY ADDRESSES AGING YES/NO
10 CFR 50.65 Maintenance Rule	R.G. 1.160, Rev. 2 NUREG-1526 Insp. Proc. 62706 Insp. Proc. 62002 NRC Letter "NRC Comments on NEI 96-03, Rev. D, ..." Oct. 1, 1996 SECY-97-055	NUMARC 93-01, Rev. 2 NEI 96-03, Rev. D Individual plant responses to Rule ACI 349.3R-96	**Uncertain:** Included in the scope of the Maintenance Rule but NRC Letter has comments on NEI 96-03 and it is not clear how or whether the licensee's program for buried piping comply with the Rule. Specific criteria are established by each licensee – not submitted for technical review. Note: For License Renewal, Maintenance Rule structures monitoring of buried piping which is conducted in accordance with 10 CFR 50, Appendix B – Quality Assurance and follows the guidance in R.G. 1.160, Rev. 2 and NUMARC 93-01, Rev. 2 may be applicable. It has not been evaluated for acceptability.
IPE/IPEEE	GL 88-20, IPE For Severe Accident Vulnerabilities GL 88-20, Supplement No. 4, IPEEE For Severe Accident Vulnerabilities NUREG-1407	Individual plant submittals	**No:** Limited to success path(s) components only. Does not cover all safety related items. Does not address aging degradation in guidance documents or licensee IPEEE submittals.
NUREG-1522	N/A	N/A	**Partially:** Limited review of older plants. Scope was to identify aging degradation of structures and civil engineering features.

Table 5-1 Adequacy of Aging Programs – Supports for Equipment and Systems
Including steel members, bolted and friction (Unistrut type, bolted, & clamp) connections, and welded connections

NRC PROGRAM	NRC GUIDANCE / ADDL. REQMTS.	INDUSTRY PROGRAMS	ADEQUATELY ADDRESSES AGING YES/NO
USI A-46 "Seismic Qualification of Equipment in Operating Plants"	NRC GL 87-02, Supplement No. 1 Transmitting SSER No. 2 NUREG-1030 NUREG-1211	GIP-2 & EPRI reports referred to within	**No**: Limited to USI A-46 plants. Limited to one "safe shutdown" path & consideration of a single equipment failure. Only a one-time assessment. No specific requirement to consider aging.
10 CFR 50.65 Maintenance Rule	R.G. 1.160, Rev. 2 NUREG-1526 Insp. Proc. 62706 Insp. Proc. 62002 NRC Letter "NRC Comments on NEI 96-03, Rev. D, …" Oct. 1, 1996 SECY-97-055	NUMARC 93-01, Rev. 2 NEI 96-03, Rev. D Individual plant responses to Rule	**Uncertain:** Scope includes supports but NRC Letter has comments on industry program and it is not clear how or whether the licensee's program for supports comply with the Rule. Specific criteria are established by each licensee – not submitted for technical review. Note: For License Renewal, Maintenance Rule structures monitoring of supports not covered by ASME Section XI which is conducted in accordance with 10 CFR 50, Appendix B – Quality Assurance and follows the guidance in R.G. 1.160, Rev. 2 and NUMARC 93-01, Rev. 2 is currently being evaluated for acceptability.
10 CFR 50.55a Conditions of Construction Permit – Codes and Standards		ASME Section XI	**Partially:** Applicable only to Class 1, 2, & 3 pressure retaining ASME components and their supports. 10 CFR 50.55a, Par. (g) states for old plants prior to 1971, must meet requirements to the "extent practical." Exempts - small diameter lines, components operating at $\leq$275 psig & at temps. $\leq$200 °F, and inaccessible supports (embedded in concrete, underground, or encapsulated by guard pipe).

Table 5-1 Adequacy of Aging Programs – Supports for Equipment and Systems (Continued)
Including steel members, bolted and friction (Unistrut type, bolted, & clamp) connections, and welded connections

IPE/IPEEE	GL 88-20, IPE For Severe Accident Vulnerabilities GL 88-20, Supplement No. 4, IPEEE For Severe Accident Vulnerabilities NUREG-1407	Individual plant submittals	**No:** Limited to success path(s) components only. Does not cover all safety related items. Does not address aging degradation in guidance documents or licensee IPEEE submittals.
		ASCE 11-90 Std., Guideline for Structural Assessment of Existing Buildings	**Partially:** Primarily provides guidance for assessing (identifying, quantifying using tests, & reporting) the physical and material condition. Makes reference to other standards & documents. Does not provide guidelines on analysis of aged condition, acceptance criteria, and applicability to NPPs. Although the scope and intent of the Standard is for buildings, the guidelines could also be applied to supports.
NUREG-1522	N/A	N/A	**Partially:** Limited review of older plants. Scope was to identify aging degradation of structures and civil engineering features.

Table 5-1 Adequacy of Aging Programs – Concrete Containments

NRC PROGRAM	NRC GUIDANCE / ADDL. REQMTS.	INDUSTRY PROGRAMS	ADEQUATELY ADDRESSES AGING YES/NO
10 CFR 50.65 Maintenance Rule	R.G. 1.160, Rev. 2 NUREG-1526 Insp. Proc. 62706 Insp. Proc. 62002 Insp. Proc. 62003 NRC Letter "NRC Comments on NEI 96-03, Rev. D, ..." Oct. 1, 1996 SECY-97-055	NUMARC 93-01, Rev. 2 NEI 96-03, Rev. D Individual plant responses to Rule	**Uncertain:** Scope includes containments but NRC Letter has comments on industry program and it is not clear how or whether the licensee's program for items within the containment system comply with the Rule. Specific criteria are established by each licensee – not submitted for technical review. NEI 96-03 does not provide specific guidance or criteria for containments but lists general references such as ASME, Section XI and ACI 349.3R as additional sources which may be used. ACI 349.3R does not apply to containments. Note: For License Renewal, Maintenance Rule structures monitoring of containments is not accepted as a substitute for ASME Section XI, Subsection IWE and IWL.
10 CFR 50.55a Conditions of Construction Permit – Codes and Standards		ASME Section XI, Subsections IWE and IWL	**Yes:** Applicable to ASME Class MC (steel containment) & metallic liners of Class CC (concrete containment) and Class CC concrete containment. 10 CFR 50.55a, Par. (g) (4) states that Class MC and Class CC must meet requirements .... to the extent practical. ASME exempts inaccessible areas. However, 10 CFR 50.55a has requirements for inaccessible areas. No specific guidance or acceptance criteria are provided though.

Table 5-1 Adequacy of Aging Programs – Concrete Containments (Continued)

IPE/IPEEE	GL 88-20, IPE For Severe Accident Vulnerabilities GL 88-20, Supplement No. 4, IPEEE For Severe Accident Vulnerabilities NUREG-1407	Individual plant submittals	**No:** Limited to success path(s) components only. Does not cover all safety related items. Does not address aging degradation in guidance documents or licensee IPEEE submittals.
NUREG-1611 Aging Management of NPP Containments for License Renewal			**Yes:** Report provides technical information and agreements resulting from the NUMARC Inspection Report reviews and the inservice inspection requirements of ASME, Subsection IWE and IWL as promulgated in 10 CFR 50.55a. Specific exceptions are identified and additional evaluations and augmented inspection activities for license renewal are recommended. Note: For License Renewal, NUREG-1611 defines the basis for an acceptable aging management program for containments.
NUREG-1522	N/A	N/A	**Partially:** Limited review of older plants. Scope was to identify aging degradation of structures and civil engineering features. Identified potential aging concerns related to detrimental environment in tendon gallery and grease leakage.
Research Program on the Effect of Grease Leakage Into Concrete	NUREG/CR-6598		**Yes:** Investigated the extent of tendon sheathing filler leakage into the concrete and effects on the concrete properties (tensile and compressive).
NRC/RES Program: "Inspection of Aged/Degraded Containments"			**Uncertain:** Program currently in progress.

Table 5-1 Adequacy of Aging Programs – Concrete Containments (Continued)

NRC/RES Program: "Capacity of Aged/Degraded Containments"			**Uncertain:** Program currently in progress.

Table 5-1 Adequacy of Aging Programs – Steel Containments

NRC PROGRAM	NRC GUIDANCE / ADDL. REQMTS.	INDUSTRY PROGRAMS	ADEQUATELY ADDRESSES AGING YES/NO
10 CFR 50.65 Maintenance Rule	R.G. 1.160, Rev. 2 NUREG-1526 Insp. Proc. 62706 Insp. Proc. 62002 Insp. Proc. 62003 NRC Letter "NRC Comments on NEI 96-03, Rev. D, ..." Oct. 1, 1996 SECY-97-055	NUMARC 93-01, Rev. 2 NEI 96-03, Rev. D Individual plant responses to Rule	**Uncertain:** Scope includes steel containment, but NRC Letter has comments on Industry Program and it is not clear how or whether the licensee's program for these items comply with the Rule. Specific criteria are established by each licensee – not submitted for technical review. NEI 96-03 does not provide specific guidance or criteria for containments but lists general references such as ASME, Section XI which may be used. Note: For License Renewal, Maintenance Rule structures monitoring of containments is not accepted as a substitute for ASME Section XI, Subsection IWE and IWL.
10 CFR 50.55a Conditions of Construction Permit – Codes and Standards		ASME Section XI	**Yes:** Applicable to ASME Class MC (steel containment) & metallic liners of Class CC (concrete containment), and Class CC concrete containment. 10 CFR 50.55a, Par. (g) (4) states that Class MC and Class CC must meet requirements .... to the extent practical. ASME exempts inaccessible areas. However, 10 CFR 50.55a has requirements for inaccessible areas. No specific guidance or acceptance criteria provided though.

Table 5-1 Adequacy of Aging Programs – Steel Containments (Continued)

IPE/IPEEE	GL 88-20, IPE For Severe Accident Vulnerabilities GL 88-20, Supplement No. 4, IPEEE For Severe Accident Vulnerabilities NUREG-1407	Individual plant submittals	**No:** Limited to success path(s) components only. Does not cover all safety related items. Does not address aging degradation in guidance documents or licensee IPEEE submittals.
NUREG-1611 Aging Management of NPP Containments for License Renewal			**Yes:** Report provides technical information and agreements resulting from the NUMARC Inspection Report reviews and the inservice inspection requirements of ASME, Subsection IWE and IWL as promulgated in 10 CFR 50.55a. Specific exceptions are identified and additional evaluations and augmented inspection activities for license renewal are recommended. Note: For License Renewal, NUREG-1611 defines the basis for an acceptable aging management program for containments.
NUREG-1522	N/A	N/A	**Partially:** Limited review of older plants. Scope was to identify aging degradation of structures and civil engineering features.
NRC efforts related to corrosion of two BWR Mark I steel containments	GL 87-05 IN 88-82 NUREG-1540		**Partially:** Corrosion of steel shell was identified at two BWR Mark I steel containments. GL required licensees to perform inspections and report results. IN informed licensees of potential corrosion problems in torus. NUREG-1540 describes regulatory actions taken and describes a study by BNL on the performance of a degraded containment under severe accident conditions.

Table 5-1 Adequacy of Aging Programs – Steel Containments (Continued)

NRC/RES Program: "Inspection of Aged/Degraded Containments"			**Uncertain:** Program currently in progress.
NRC/RES Program: "Capacity of Aged/Degraded Containments"			**Uncertain:** Program currently in progress.

Table 5-2 Prioritization of Structures/Components for Further Evaluation

NO.	STRUCTURE/COMPONENT	RANK CONSIDERING:			ARE EXISTING PROGRAMS ADEQUATE? (See Table 5-1)	PRIORITY RANKING FOR FURTHER STUDY
		SEISMIC RISK SIGNIFICANCE (See Table 5-3)	DEGRADATION OCCURRENCES	IMPORTANCE TO CURRENT LICENSING BASIS/LICENSE RENEWAL		
1	**MASONRY WALLS**	VERY HIGH	HIGH	HIGH	UNCERTAIN	VERY HIGH
2	**TANKS:**					
	FLAT BOTTOM	VERY HIGH	HIGH	HIGH	UNCERTAIN	VERY HIGH
	OTHERS	LOW	LOW	LOW	UNCERTAIN	LOW
3	**ANCHORAGES**	HIGH	HIGH	HIGH	UNCERTAIN	HIGH
4	**CONCRETE STRUCTURES** (Other than Containments)	MODERATE	MODERATE	VERY HIGH	UNCERTAIN	HIGH
5	**BURIED PIPING**	HIGH	UNKNOWN	HIGH	UNCERTAIN	HIGH
6	**SUPPORTS FOR EQUIPMENT AND SYSTEMS**	HIGH	MODERATE	MODERATE	UNCERTAIN	MODERATE
7	**CONCRETE CONTAINMENTS**	LOW	HIGH	HIGH	YES	LOW
8	**STEEL CONTAINMENTS**	INSUFFICIENT DATA	HIGH	HIGH	YES	LOW

Table 5-3 Summary of Evaluation for Ranking Seismic Risk Significance

NO.	STRUCTURE/COMPONENT	SEISMIC RISK SIGNIFICANCE*					OVERALL SEISMIC RISK SIGNIFICANCE
		SEISMIC COMPONENT HCLPF VALUE (g) BASED ON PAST INDUSTRY REPORTS			IPEEE RESULTS Based on NRC Memo 1/20/98, "Preliminary IPEEE Insights Report"		
		MIN	MAX	MEDIAN	CONTRIBUTORS TO CDF/WEAKNESSES IN SEISMIC PRA (Tables 3.3/3.4 of Ref. Memo)	CONTROLLING COMPONENTS IN SMA (Table 3.7 of Ref. Memo)	
1	MASONRY WALLS	.10	.50	.20	VERY HIGH/ VERY HIGH	NOT IDENTIFIED	VERY HIGH
2	TANKS:						
	FLAT BOTTOM	.079	.42	.26	HIGH/ NOT IDENTIFIED	HIGH	VERY HIGH
	OTHERS	.30	15.0	.90			LOW
3	ANCHORAGES	.22	16.8	1.26	NOT REPORTED SEPARATELY/ VERY HIGH	VERY HIGH	HIGH
4	CONCRETE STRUCTURES (Other than Containments)	.20	2.70	.70	HIGH/ NOT IDENTIFIED	NOT IDENTIFIED	MODERATE
5	BURIED PIPING	.27	.39	.39	NOT IDENTIFIED/ NOT IDENTIFIED	NOT IDENTIFIED	HIGH
6	SUPPORTS FOR EQUIPMENT AND SYSTEMS	.06	3.50	.79	NOT REPORTED SEPARATELY/HIGH	HIGH	HIGH
7	CONCRETE CONTAINMENTS	.80	1.20	1.0	NOT IDENTIFIED/ NOT IDENTIFIED	NOT IDENTIFIED	LOW
8	STEEL CONTAINMENTS	Insufficient data	Insufficient data	Insufficient data	NOT IDENTIFIED/ NOT IDENTIFIED	NOT IDENTIFIED	INSUFFICIENT DATA

Table 5-4 Core Damage Frequencies (Per Reactor Year)

PROGRAM	EVENT	RANGE OF CORE DAMAGE FREQUENCIES (Per Reactor Year)
IPE* (INTERNAL EVENTS)	STATION BLACKOUT, LOCA, ISLOCA/ STEAM GEN. TUBE RUPTURE, ATWS, INTERNAL FLOODS, TRANSIENTS	7.5×10^{-8} TO 4.0×10^{-4}
IPEEE** (EXTERNAL EVENTS)	SEISMIC	2.2×10^{-7} TO 2.2×10^{-4}
	FIRE	1.0×10^{-9} TO 5.3×10^{-3}
	HFO (HIGH WINDS, FLOODS, & OTHER EXTERNAL INITIATING EVENTS) HIGH WINDS FLOODING LIGHTNING*** SNOW & ICE*** AIRCRAFT CRASH***	 3.7×10^{-7} TO 5.7×10^{-5} 2.1×10^{-8} TO 1.0×10^{-5} 8.0×10^{-6} 6.7×10^{-6} 5.7×10^{-7}

* NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor and Plant Performance," December 1997
** NRC Memo, "Preliminary IPEEE Insights Report," from L. Joseph Callan to S. Jackson & the Commissioners, January 20, 1998
*** Based on data from one plant

6 CONCLUSIONS AND RECOMMENDATIONS

6.1 Conclusions

Based on the results of the Phase I activities discussed in the previous sections of this report and summarized in Table 5-2, it has been concluded that Phase II of this program should be continued for the following structures and passive components (SCs): masonry walls, flat bottom tanks, anchorages, reinforced concrete structures (other than containments since they are being addressed in other NRC research programs), and buried piping.

The focus of the research should be on improving and developing methods to assess the effects of age-related degradation on the seismic performance of SCs, including the fragility evaluations for PRA/SMA studies. The methodologies that will be developed to determine seismic performance could then be used to quantify the impact of age-related degradation of SCs on overall plant risk. This would lead to greater confidence in the use of risk assessment as a tool in making risk informed decisions for age-degraded structures. The research will also establish the technical bases for resolving specific issues related to degradation of the selected SCs.

The Phase II efforts should include the following activities:

- Evaluation and expansion, if necessary, of existing degradation condition assessment techniques and the collection and review of available U.S. and foreign test results on naturally degraded or artificially degraded SCs.

- Performance of analytical structural evaluations of degraded SCs utilizing methods such as linear or nonlinear finite element methods.

- Development of fragility curves for degraded SCs based on results of analytical structural evaluations or tests of degraded SCs. The reduction in fragility curves should be evaluated for their effect on overall plant risk.

- Development of degradation acceptance criteria for SCs based on the above activities, existing codes, standards, and other NRC or industry reports.

The results of the Phase II efforts should establish the technical bases for the formulation of recommendations during Phase III for regulatory guidance on the assessment of age-degraded structures. During Phase III, the recommendations that have been developed could be applied to an actual plant to test the methodologies and make any necessary refinements.

6.2 Recommendations for Phase II and Phase III Program Scope

The following sections describe the specific research activities recommended for reinforced concrete structures. This component was selected to initiate Phase II activities since it has broad application throughout all nuclear power plants and elements of the work will have immediate use in current license renewal activities. Although the specific activities have not been developed for the other SCs identified above, it is envisioned that the same approach could be used once the effort on reinforced concrete structures is completed. As the research program progresses the focus and/or scope of these activities may need to be adjusted.

The sections that follow describe the approach recommended for developing probability based degradation acceptance limits for reinforced concrete components. The results of this effort could be used to evaluate whether degraded conditions, that may be identified during walkdowns or a condition assessment performed by plant personnel, have a significant effect on overall plant risk.

6.2.1 Condition Assessment and Quantification of Concrete Degradation

In order to assess the effects of age-related degradation of reinforced concrete structures, the condition of the degraded structures must be known. Therefore, a structural condition assessment of concrete structures is performed. A structural condition assessment would include activities such as visual inspections, physical measurements of degradations, nondestructive testing (NDT), and destructive testing.

Industry standards such as ACI 207.3 – 79 (revised 1985) describe methods available for evaluating physical properties of concrete in existing structures. The methods described include cracking surveys, surface mapping, core drilling and testing, and nondestructive testing methods. The cracking survey is an examination of the concrete for the purpose of locating, marking, and identifying cracks. Surface mapping consists of detailed drawings identifying cracks, spalling, scaling, popouts, honeycombing, distortions, condition of joints, corrosion of reinforcement if exposed, and soundness of surface concrete. NDT is used to determine the various properties of the concrete such as strength, modulus of elasticity, homogeneity, and integrity.

The scope of the research under this task should consist of identifying current and newly developed methods that can be used in identifying the extent of concrete degradation at nuclear power plants. Acceptable condition assessment techniques are needed in order to quantify the extent of degradation for comparison against the degradation acceptance limits which will be developed as part of the Phase II research effort.

The important structural properties of interest for quantifying degradation of the concrete components are reductions in:

- Concrete compressive strength
- Bond strength
- Concrete "area" (due to cracking and/or spalling)
- Reinforcing steel area

It is desirable to identify and describe condition assessment techniques for each of these properties in a manner that can be performed by an experienced structural engineer without destructive examination methods. Visual inspection, which may be supplemented by nondestructive examination methods, if required, is preferred even if some judgement is necessary.

The degradation acceptance limits for the four structural properties listed above should be developed separately as described in Sections 6.2.2, 6.2.3, and 6.2.4 below. These degradation limits should be determined based on the calculated reduction in fragility's and an assessment of the effect on overall plant risk.

In addition to the above, a review of available information is needed to relate observable levels of degradations to quantifiable reductions in reinforced concrete basic properties -concrete compressive strength, concrete area, steel reinforcement area, and bond strength. This could be obtained from available U.S. and foreign test data on naturally and artificially degraded concrete components.

6.2.2 Structural Performance Evaluation of Degraded Concrete Components

The effects of degradation on the performance of structural concrete components should be evaluated. The characteristics of the structural elements that affect its performance are their: strength, stiffness, and ductility. The strength of the element is the more important characteristic and most influences the element's seismic capacity. The element stiffness influences the distribution of forces between parallel elements and the fundamental frequency of the building. The element ductility allows for redistribution of loads between elements and results in lower effective loads from seismic input motions. Once these characteristics are identified for individual elements, the effects of degradation on the entire structure can be evaluated using standard structural analysis methods. The focus should be placed on reinforced concrete shear walls and beams. These have been selected because they are frequently found in nuclear power stations and data exist describing their response to degradation.

The structural characteristics of interest depend on: concrete compressive strength (it is being assumed that shear strength, and modulus of elasticity can be related to the compressive strength), bond strength, concrete "area", and reinforcing steel area. Concrete strength can be reduced by the following degradation mechanisms: leaching and efflorescence, sulfate attack, alkali-aggregate interaction, and acidic attack. Bond strength is reduced as the concrete strength is reduced, and can also be affected as a result of rebar corrosion resulting in cracking or spalling of the concrete. Both cracking and spalling can reduce the concrete area. The reinforcing steel area can be reduced by corrosion.

Relationships between the element's structural characteristics (strength, stiffness, and ductility) and the basic properties (concrete compressive strength, bond strength, concrete area, and reinforcing steel area) should be developed. This can be achieved utilizing closed form solutions where appropriate or finite element methods. In either case, the analytical methodology should be benchmarked against known results to confirm that the approach being utilized can predict the response of the concrete elements being investigated.

6.2.3 Fragility and Risk Evaluation of Degraded Concrete Components

Once the analytical methodology has been benchmarked, fragility curves should be developed for both undegraded and degraded concrete components. The fragility analysis should assess, in probabilistic terms, the capability of the concrete components to withstand a specified loading. The fragility modeling process will provide a median-centered (or most likely) estimate of system performance and an estimate of the variability or uncertainty in performance.

All-important sources of uncertainty should be included in the fragility analysis to predict the likely variability in performance of the structure in service. Available statistical data, which provide the strength of reinforced concrete flexural members (beams and slabs) and short concrete shear walls, need to be collected. This would include parameters such as concrete compressive strength and tensile strength, steel yield strength and modulus of elasticity, and placement of reinforcement.

An appropriate sample (set of varying parameters) should be developed for each case to be analyzed. The analytical methods described above should be used to obtain the response of the concrete components for each sample set. Fragility curves can then be developed from the results. In addition to performing the fragility analysis for the undegraded case, fragility analyses should preferably be performed for degradation of each structural property (concrete compressive stress, concrete area, steel area, and bond) separately and in combination with one another.

The degraded concrete component fragility curves should be used to assess the impact of degradation on the overall seismic risk to the plant. This will form the basis for identifying acceptance limits to be used for comparison with degradation identified in a condition assessment. The results from previous seismic risk assessments performed on plants can be used to generate the required data.

6.2.4 Application of Methodologies to an Actual Plant

To test the methodologies developed in this research program, it is recommended that these methods be applied to an actual plant. This trial case would also be useful to refine the methods if deemed necessary. The trial case could consist of selecting a representative nuclear plant, which currently has known cases of concrete degradation. This could be either an operating plant or decommissioned plant where accessibility is easier and limited destructive testing could be performed if needed. Alternatively, it may be possible to get information from actual known cases of degradation, which have occurred in the past. To test the methodologies and results, some or preferably all of the approaches developed in Sections 6.2.1 through 6.2.3 above should be implemented.

The application of the methodologies could be done in a cost-effective way. For example, the proposed fragility evaluation methodology could be applied to a representative plant for which a PRA study has already been performed in the past. Therefore, most of the existing analysis results may be utilized to calculate the effect of degradation on the reduction in fragility's and the effect on the overall plant risk.

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APPENDIX A

DEGRADATION OCCURRENCE DATABASE

COMPONENT	SUBCOMPONENT	SYSTEM	AGING EFFECTS	AGING MECHANIS	PLANT	MO	DA	YR	HOW IDENTI	EVALN MET	REPAIR METH	DKT	REF DO	REF NO
ANCHORAGES	ANCHOR BOLTS - STRAINER	SERVICE WATER	DETERIORATION	N. A.	ROBINSON 2			95*	VISUAL	N. A.	N. A.	261	NUREG	1522
ANCHORAGES	EXPANSION ANCHOR	ERCSW	FAILURE	CORROSION	MILLSTONE 2	4	29	86	VISUAL	VISUAL	N. A.	336	LER	860100
ANCHORAGES	EXPANSION ANCHOR - NUTS	ERCSW	LOOSENING	VIBRATION	QUAD CITIES 1	5	8	87	N. A.	VISUAL	REPLACEMENT	254	LER	870801
ANCHORAGES	GROUT & BASEPLATES	N. A.	DETERIORATION	CORROSION	BEAVER VALLEY 1			95*	VISUAL	N. A.	N. A.	334	NUREG	1522
ANCHORAGES	GROUT- EQUIPMT. SUPPORT	PUMPHOUSE	CRACKING DETERIORATION	MOISTURE	POINT BEACH 2			95*	VISUAL	N. A.	N. A.	301	NUREG	1522
ANCHORAGES	GROUT- EQUIPMT. SUPPORT	PUMPHOUSE	CRACKING DETERIORATION	MOISTURE	POINT BEACH 1			95*	VISUAL	N. A.	N. A.	266	NUREG	1522
ANCHORAGES	N. A.	SEVERAL	DETERIORATION	N. A.	COOPER			95*	VISUAL	N. A.	N. A.	298	NUREG	1522
ANCHORAGES	STUDS - EMBEDDED	N. A.	FAILURE	MECHANICAL LOADS	INDIAN POINT 2	2	12	91	VISUAL	VISUAL	REPAIR	247	LER	910401
CABLE TRAY SYSTEM	ELECTRICAL CABLE TRAY-SEAL	CBEAF	DETERIORATION	N. A.	BRUNSWICK 1	10	25	95	TEST	N. A.	REPAIR	325	LER	952001
CONCRETE	CEILING	FHB	CRACKING DETERIORATION	MOISTURE	TURKEY POINT 3			95*	N. A.	N. A.	N. A.	250	NUREG	1522
CONCRETE	CEILING/FOUNDATION	SECONDARY CONTMT.	DETERIORATION	N. A.	DRESDEN 2	4	7	89	TEST	N. A.	REPAIR	237	LER	891400
CONCRETE	FLOORS, WALLS, FOUNDATION	VARIOUS STRUCTURES	CRACKING SPALLING	N. A.	COOPER			95*	VISUAL	N. A.	N. A.	298	NUREG	1522
CONCRETE	INTAKE STRUCT. - BEAMS	CIRCULAT. WATER	CRACKING	CORROSION - EMBED. STL.	TURKEY POINT 4			89	N. A.	N. A.	REPAIR	251	NUREG	1522
CONCRETE	INTAKE STRUCT. - BEAMS	CIRCULAT. WATER	CRACKING	CORROSION - EMBED. STL.	TURKEY POINT 3			89	N. A.	N. A.	REPAIR	250	NUREG	1522
CONCRETE	INTAKE STRUCT. - BEAMS & WALLS	SERVICE WATER	CRACKING	CORROSION - EMBED. STL.	SAN ONOFRE 1			84	INSPECTION	N. A.	REPAIR	206	NUREG	1522
CONCRETE	INTAKE STRUCTURE	SERVICE WATER	CRACKING	N. A.	ROBINSON 2			95*	VISUAL	N. A.	N. A.	261	NUREG	1522
CONCRETE	INTAKE STRUCTURE	RAW WATER - INTAKE STR.	CRACKING	N. A.	BEAVER VALLEY 1			95*	VISUAL	N. A.	N. A.	334	NUREG	1522
CONCRETE	MASONRY WALL	N. A.	CRACKING	N. A.	TURKEY POINT 3			95*	N. A.	N. A.	N. A.	250	NUREG	1522
CONCRETE	MASONRY WALL	N. A.	CRACKING	N. A.	TURKEY POINT 4			95*	N. A.	N. A.	N. A.	251	NUREG	1522
CONCRETE	MASONRY WALL	FAN HOUSE	CRACKING	N. A.	INDIAN POINT 2	9	16	85	INSPECTION	VISUAL	REPAIR	247	NRC IN	87-67-1
CONCRETE	MASONRY WALL	N. A.	CRACKING	N. A.	OYSTER CREEK	5	5	86	INSPECTION	VISUAL	REPAIR	219	NRC IN	87-67-1
CONCRETE	MASONRY WALL	N. A.	CRACKING	N. A.	YANKEE ROWE	1	26	87	INSPECTION	VISUAL	REPAIR	29	NRC IN	87-67-1

COMPONENT	SUBCOMPONENT	SYSTEM	AGING EFFECTS	AGING MECHANIS	PLANT	MO	DA	YR	HOW IDENTI	EVALN MET	REPAIR METH	DKT	REF DO	REF NO
CONCRETE	MASONRY WALL	N. A.	CRACKING	N. A.	POINT BEACH 2			95*	VISUAL	N. A.	N. A.	301	NUREG	1522
CONCRETE	MASONRY WALL	N. A.	CRACKING	N. A.	SALEM 1	4	7	87	INSPECTION	VISUAL	REPAIR	272	NRC IN	87-67-1
CONCRETE	MASONRY WALL	N. A.	CRACKING	N. A.	PEACH BOTTOM 2	6	15	87	INSPECTION	VISUAL	REPAIR	277	NRC IN	87-67-1
CONCRETE	MASONRY WALL	N. A.	CRACKING	N. A.	ROBINSON 2			95*	VISUAL	N. A.	N. A.	261	NUREG	1522
CONCRETE	MASONRY WALL	N. A.	CRACKING	N. A.	PEACH BOTTOM 3	6	15	87	INSPECTION	VISUAL	REPAIR	278	NRC IN	87-67-1
CONCRETE	MASONRY WALL	N. A.	CRACKING	N. A.	TROJAN			95*	VISUAL	N. A.	N. A.	344	NUREG	1522
CONCRETE	MASONRY WALL	N. A.	CRACKING	N. A.	SALEM 2	4	7	87	INSPECTION	VISUAL	REPAIR	311	NRC IN	87-67-1
CONCRETE	MASONRY WALL	N. A.	CRACKING	N. A.	POINT BEACH 1			95*	VISUAL	N. A.	N. A.	266	NUREG	1522
CONCRETE	MASONRY WALL	N. A.	CRACKING	N. A.	BEAVER VALLEY 1			95*	VISUAL	N. A.	N. A.	334	NUREG	1522
CONCRETE	MAT	CONTAINMENT	LOSS OF MATERIAL	EROSION	MILLSTONE 3	3	21	97*	N. A.	N. A.	N. A.	423	NRC IN	97-11
CONCRETE	PEDESTAL PUMP	SWS	CRACKING	MOISTURE	MILLSTONE 3	6	21	96	INSPECTION	ENGIN. JUGM.	REPAIR	423	LER	961800
CONCRETE	ROOF	SECONDARY CONTMT.	DETERIORATION	N. A.	QUAD CITIES 1	8	13	91	TEST	TEST	REPAIR	254	LER	911100
CONCRETE	WALLS & ROOF	PUMPHOUSE	CRACKING	N. A.	POINT BEACH 1			95*	VISUAL	N. A.	N. A.	266	NUREG	1522
CONCRETE	WALLS & ROOF	PUMPHOUSE	CRACKING	N. A.	POINT BEACH 2			95*	VISUAL	N. A.	N. A.	301	NUREG	1522
CONCRETE	WALLS, CEILINGS, BASEMAT	VARIOUS STRUCTURES	CRACKING SPALLING	N. A.	ROBINSON 2			95*	VISUAL	N. A.	N. A.	261	NUREG	1522
CONCRETE	WALLS, CEILINGS, FOUNDATION	VARIOUS STRUCTURES	CRACKING SPALLING	N. A.	BEAVER VALLEY 1			95*	VISUAL	N. A.	N. A.	334	NUREG	1522
CONCRETE	WALLS, FLOORS, CEILINGS	VARIOUS STRUCTURES	CRACKING	N. A.	POINT BEACH 1			95*	VISUAL	N. A.	N. A.	266	NUREG	1522
CONCRETE	WALLS, FLOORS, CEILINGS	VARIOUS STRUCTURES	CRACKING	N. A.	TROJAN			95*	VISUAL	N. A.	N. A.	344	NUREG	1522
CONCRETE	WALLS, FLOORS, CEILINGS	VARIOUS STRUCTURES	CRACKING	N. A.	POINT BEACH 2			95*	VISUAL	N. A.	N. A.	301	NUREG	1522
CONDUIT SYSTEM	ELECTRICAL CONDUIT	SWITCHYARD	DETERIORATION	MOISTURE	INDIAN POINT 3	9	5	86	TRIP	N. A.	REPLACEMENT	286	LER	861000
CONDUIT SYSTEM	ELECTRICAL CONDUIT	N. A.	RUPTURE	MOISTURE, SALT	PILGRIM 1	6	28	91	LEAKING	ENGIN. JUGM.	N. A.	293	LER	911500
CONDUIT SYSTEM	ELECTRICAL CONDUIT	RCIC	LOSS OF MATERIAL	CORROSION	COOPER	7	15	92	INSPECTION	ENGIN. JUGM.	REPLACEMENT	298	LER	921200

COMPONENT	SUBCOMPONENT	SYSTEM	AGING EFFECTS	AGING MECHANIS	PLANT	MO	DA	YR	HOW IDENTI	EVALN MET	REPAIR METH	DKT	REF DO	REF NO
CONDUIT SYSTEM	ELECTRICAL CONDUIT SEAL	CBEAF	FAILURE	MECHANICAL WEAR	BRUNSWICK 1	10	25	95	LEAKING	N. A.	REPAIR	325	LER	952001
CONDUIT SYSTEM	JUNCTION BOX/CONDUIT	N. A.	FAILURE	N.A.	BROWNS FERRY 2	6	6	86	FUSE FAILURE	VISUAL	N. A.	260	LER	860800
CONDUIT SYSTEM (UG.)	ELECTRICAL CONDUIT	SAFETY INJECTION	LOSS OF MATERIAL	MOISTURE	SAN ONOFRE 1	1	20	90	GROUND	N. A.	N. A.	206	LER	900100
CONTAINMENT	CONCRETE SHELL	CONTAINMENT	N.A.	GREASE LEAKAGE	POINT BEACH 2			95*	VISUAL	N. A.	N. A.	301	NUREG	1522
CONTAINMENT	CONCRETE SHELL	CONTAINMENT	CRACKING SPALLING	N. A.	BEAVER VALLEY 1			95*	ILRT	N. A.	N. A.	334	NUREG	1522
CONTAINMENT	CONCRETE SHELL & TEND. GALLERY	CONTAINMENT	N.A.	GREASE LEAKAGE	TROJAN			95*	N. A.	N. A.	NONE	344	NUREG	1522
CONTAINMENT	CONCRETE SHELL - BUTTRESS	CONTAINMENT	CRACKING SPALLING	N. A.	POINT BEACH 1			95*	VISUAL	N. A.	N. A.	266	NUREG	1522
CONTAINMENT	LINER	CONTAINMENT	EXCESSIVE DEFORMN.	N. A.	BEAVER VALLEY 1			95*	INSPECTION	VISUAL	MONITORING	334	NUREG	1522
CONTAINMENT	LINER	CONTAINMENT	EXCESSIVE DEFORMN.	N. A.	TURKEY POINT 4			95*	VISUAL	N. A.	N. A.	251	NUREG	1522
CONTAINMENT	LINER	CONTAINMENT	EXCESSIVE DEFORMN.	N. A.	POINT BEACH 2			95*	VISUAL	N. A.	N. A.	301	NUREG	1522
CONTAINMENT	LINER	CONTAINMENT	LOSS OF MATERIAL	N. A.	THREE MILE ISLAND 1	5	10	93	INSPECTION	VISUAL	N. A.	289	LER	930500
CONTAINMENT	LINER	CONTAINMENT	LOSS OF MATERIAL	MOISTURE	ROBINSON 2	4		92	N. A.	VISUAL	N. A.	261	NRC IN	97-10
CONTAINMENT	LINER	CONTAINMENT	LOSS OF MATERIAL	N. A.	TURKEY POINT 4			95*	VISUAL	N. A.	N. A.	251	NUREG	1522
CONTAINMENT	LINER	CONTAINMENT	LOSS OF MATERIAL	MOISTURE	BEAVER VALLEY 1	6		92	INSPECTION	VISUAL	N. A.	334	NRC IN	97-10
CONTAINMENT	LINER	CONTAINMENT	LOSS OF MATERIAL	MOISTURE	SALEM 2			93	N. A.	VISUAL	N. A.	311	NRC IN	97-10
CONTAINMENT	LINER	CONTAINMENT	EXCESSIVE DEFORMN.	N. A.	TROJAN			95*	N. A.	N. A.	N. A.	344	NUREG	1522
CONTAINMENT	LINER	CONTAINMENT	LOSS OF MATERIAL	MOISTURE	BRUNSWICK 2	1		93	INSPECTION	VISUAL	N. A.	324	NRC IN	97-10
CONTAINMENT	LINER - COATING	CONTAINMENT	PEELING/MISSING	N. A.	TROJAN			95*	N. A.	N. A.	N. A.	344	NUREG	1522
CONTAINMENT	LINER - COATING	CONTAINMENT	PEELING	N. A.	BEAVER VALLEY 1			95*	VISUAL	N. A.	N. A.	334	NUREG	1522
CONTAINMENT	LINER - COATING	CONTAINMENT	PEELING	N. A.	TURKEY POINT 4			95*	VISUAL	N. A.	N. A.	251	NUREG	1522
CONTAINMENT	LINER - COATING	CONTAINMENT	PEELING	N. A.	POINT BEACH 2			95*	VISUAL	N. A.	N. A.	301	NUREG	1522
CONTAINMENT	LINER - COATING	CONTAINMENT	DETERIORATION	N. A.	CLINTON	7		97*	N. A.	N. A.	STRIP & RECOAT	461	CORRESP.	WWW.NRC.GOV

COMPONENT	SUBCOMPONENT	SYSTEM	AGING EFFECTS	AGING MECHANIS	PLANT	MO	DA	YR	HOW IDENTI	EVALN MET	REPAIR METH	DKT	REF DO	REF NO
CONTAINMENT	LINER-COATING	CONTAINMENT	LOSS OF MAT./PEELING	N. A.	ROBINSON 2			95*	VISUAL	N. A.	N. A.	261	NUREG	1522
CONTAINMENT	PENETRATION	CONTAINMENT	LOSS OF MATERIAL	MOISTURE	BEAVER VALLEY 1			95*	VISUAL	N. A.	N. A.	334	NUREG	1522
CONTAINMENT	PENETRATION AIR LOCK	CONTAINMENT	DETERIORATION	MECHANICAL WEAR	MILLSTONE 1	3	10	91	TEST	N. A.	REPAIR	245	LER	910500
CONTAINMENT	PENETRATION AIR LOCK	CONTAINMENT	LOSS OF MATERIAL	MECHANICAL WEAR	SHEARON HARRIS 1	12	6	90	TEST	N. A.	N. A.	400	LER	902500
CONTAINMENT	PENETRATION BELLOWS	CONTAINMENT	CRACKING	N. A.	DRESDEN 2	9	23	90	INSPECTION	NA	NA	237	LER	900902
CONTAINMENT	PENETRATION SEAL	CONTAINMENT	DETERIORATION	MECHANICAL WEAR	ROBINSON 2	3	19	90	ALARM	N. A.	REPAIR	261	LER	900601
CONTAINMENT	PENETRATION SEAL	CONTAINMENT	DETERIORATION	N. A.	DRESDEN 2	7	14	95	LEAKING	VISUAL	TIGHTENING	237	LER	951700
CONTAINMENT	PRESTRESS. SYS.	CONTAINMENT	LOSS OF PRELOAD	STRESS RELAXATION	TURKEY POINT 4	11	17	92	N. A.	TEST	N. A.	251	LER	920900
CONTAINMENT	PRESTRESS. SYS.	CONTAINMENT	N.A.- WATER ACCUMULN.	N.A.	TURKEY POINT 3			95*	N. A.	N. A.	N. A.	250	NUREG	1522
CONTAINMENT	PRESTRESS. SYS. ANCHOR HEAD	CONTAINMENT	FAILURE	HSC	BELLEFONTE 2	3	8	85*	INSPECTION	ULTRAS. TEST	N. A.	439	IE IN	85-10
CONTAINMENT	PRESTRESS. SYS. ANCHOR HEAD	CONTAINMENT	FAILURE	HSC	FARLEY 2	2	26	85	INSPECTION	ULTRAS. TEST	REPLACEMENT	364	IE IN	85-10
CONTAINMENT	PRESTRESS. SYS. ANCHOR HEAD	CONTAINMENT	CRACKING	N. A.	FARLEY 2	1	27	85	INSPECTION	VISUAL	REPAIR	364	LER	850501
CONTAINMENT	PRESTRESS. SYS. ANCHOR HEAD	CONTAINMENT	FAILURE	HSC	BELLEFONTE 1	3	8	85*	INSPECTION	ULTRAS. TEST	N. A.	438	IE IN	85-10
CONTAINMENT	PRESTRESS. SYS. ANCHOR HEAD	CONTAINMENT	FAILURE	HSC	BYRON 1	3	8	85*	INSPECTION	ULTRAS. TEST	N. A.	454	IE IN	85-10
CONTAINMENT	PRESTRESS. SYS. ANCHOR HEAD	CONTAINMENT	FAILURE	HSC	BYRON 2	3	8	85*	INSPECTION	ULTRAS. TEST	N. A.	455	IE IN	85-10
CONTAINMENT	PRESTRESS. SYS. ANCHOR. CONC.	CONTAINMENT	CRACKING	N. A.	TURKEY POINT 3			95*	VISUAL	N. A.	N. A.	250	NUREG	1522
CONTAINMENT	PRESTRESS. SYS. ANCHOR.-STEEL	CONTAINMENT	LOSS OF MATERIAL	MOISTURE	TURKEY POINT 3			95*	VISUAL	N. A.	N. A.	250	NUREG	1522
CONTAINMENT	PRESTRESS. SYS. PLATES & CAPS	CONTAINMENT	LOSS OF MATERIAL	CORROSION GREASE	POINT BEACH 2			95*	VISUAL	N. A.	N. A.	301	NUREG	1522
CONTAINMENT	PRESTRESS. SYS. PLATES & CAPS	CONTAINMENT	LOSS OF MATERIAL	CORROSION GREASE	POINT BEACH 1			95*	VISUAL	N. A.	N. A.	266	NUREG	1522
CONTAINMENT	STEEL SHELL	CONTAINMENT	LOSS OF MATERIAL	CORROSION CHEM. ATTACK	CATAWBA 1	9	21	89	INSPECTION	VISUAL	N. A.	413	NRC IN	89-79
CONTAINMENT	STEEL SHELL	CONTAINMENT	LOSS OF MATERIAL	MOISTURE	MCGUIRE 1	2	26	90	INSPECTION	N. A.	REPAIR	369	LER	900600
CONTAINMENT	STEEL SHELL	CONTAINMENT	LOSS OF MATERIAL	CORROSION CHEM. ATTACK	MCGUIRE 2	8	24	89	INSPECTION	VISUAL	N. A.	370	NRC IN	89-79

COMPONENT	SUBCOMPONENT	SYSTEM	AGING EFFECTS	AGING MECHANIS	PLANT	MO	DA	YR	HOW IDENTI	EVALN MET	REPAIR METH	DKT	REF DO	REF NO
CONTAINMENT	STEEL SHELL	CONTAINMENT	LOSS OF MATERIAL	MOISTURE CH. ATTACK	MCGUIRE 1	7	27	89	INSPECTION	VISUAL	N. A.	369	LER	892000
CONTAINMENT	STEEL SHELL	CONTAINMENT	LOSS OF MATERIAL	MOISTURE CH. ATTACK	CATAWBA 1	9	21	89	INSPECTION	VISUAL	N. A.	413	LER	892000
CONTAINMENT	STEEL SHELL	CONTAINMENT	LOSS OF MATERIAL	CORROSION CHEM. ATTACK	CATAWBA 2	9	21	89	INSPECTION	VISUAL	N. A.	414	NRC IN	89-79
CONTAINMENT	STEEL SHELL DRYW. SEAL	CONTAINMENT	DETERIORATION	N. A.	QUAD CITIES 1	9	6	82	LEAKING	LEAK RATE TEST	N. A.	254	LER	822602
CONTAINMENT	STEEL SHELL DRYWELL	CONTAINMENT	LOSS OF MATERIAL	MOISTURE	OYSTER CREEK	0	0	80	LEAKING	ULTRAS. TEST	N. A.	219	GL	87-05
CONTAINMENT	TEST CHANNEL PLUGS	CONTAINMENT	LOSS OF MATERIAL	N. A.	BEAVER VALLEY 1	1	22	91	INSPECTION	VISUAL	PLUGGED	334	LER	910400
CONTAINMENT	TORUS	CONTAINMENT	LOSS OF MATERIAL	N. A.	NINE MILE POINT 1	9		88	INSERVICE INSPECT.	ULTRAS. TEST	N. A.	220	NRC IN	88-82
CONTAINMENT	TORUS	CONTAINMENT	LOSS OF MATERIAL	N. A.	COOPER			95*	VISUAL	N. A.	N. A.	298	NUREG	1522
CONTAINMENT	TORUS	CONTAINMENT	COATING DETERIORATION	COATING DEGRAD.	FITZPATRICK			95*	N. A.	N. A.	N. A.	333	NUREG	1522
CONTAINMENT	TORUS	CONTAINMENT	CRACKING	SUBFREEZ. TEMPERATURE	HATCH 2	2	3	84	INSPECTION	ENGIN. JUGM.	N. A.	366	IE IN	85-99
CONTAINMENT	TORUS - SUPPORTS	CONTAINMENT	CRACKING	MOISTURE	VERMONT YANKEE	6	28	78	VISUAL	VISUAL	N. A.	271	IE BUL	78-11
COOLING TOWER	CELL	CIRCULAT. WATER	FAILURE	N. A.	HATCH 2	9	2	95	VISUAL	N. A.	REPAIR	366	LER	950300
ELECTRICAL CONDUCT.(UG)	CABLE	ERCSW	FAILURE	CHEMICAL ATTACK	DIABLO CANYON 1	10	29	89	FAILURE	N. A.	REPLACEMENT	275	LER	930501
ELECTRICAL CONDUCTOR	BUS DUCT INSULATION	SAT	DETERIORATION	MOISTURE	BYRON 1	5	23	96	FAILURE	N. A.	REPAIR	454	LER	960700
ELECTRICAL CONDUCTOR	CABLE	SWITCHYARD	FAILURE	MOISTURE	CRYSTAL RIVER 3	3	29	93	SHORT	N. A.	REPLACEMENT	302	LER	930203
ELECTRICAL CONDUCTOR	CABLE IGNITER	CCGC	N.A.	ELEVATED TEMPERATURE	RIVER BEND 1	8	17	93	INSPECTION	ENGIN. EVALUAT.	REPLACEMENT	458	LER	931900
ELECTRICAL CONDUCTOR	CABLE INSULATION	SAFETY INJECTION	DETERIORATION	MOISTURE	SAN ONOFRE 1	1	20	90	GROUND	N. A.	N. A.	206	LER	900100
ELECTRICAL CONDUCTOR	CABLE INSULATION	N. A.	DETERIORATION	N. A.	HATCH 1	10	8	96	N. A.	N. A.	REPLACEMENT	321	LER	961200
ELECTRICAL CONDUCTOR	CABLE INSULATION	120 V HEATER CIRCUIT	DETERIORATION	ELEV. TEMP. IRRADIAT.	BIG ROCK POINT	1	14	92	VISUAL	N. A.	REPLACEMENT	155	LER	920400
ELECTRICAL CONDUCTOR	CABLE INSULATION	SAFETY INJECTION	DETERIORATION	CHEMICAL ATTACK	QUAD CITIES 2	10	9	75	INSPECTION	N. A.	N. A.	265	IE CIRC	77-06
ELECTRICAL CONDUCTOR	CABLE INSULATION	N. A.	DETERIORATION	MOISTURE	PALISADES	1	16	96	LOSS OF POWER	TEST	REPLACEMENT	255	LER	960201
EXCHANGER	HEAT EXCHANGER	N. A.	FOULING	ORGANISMS	SURRY 1	10	6	95	INSPECTION	ENGIN. JUGM.	N. A.	280	LER	951000

COMPONENT	SUBCOMPONENT	SYSTEM	AGING EFFECTS	AGING MECHANIS	PLANT	MO	DA	YR	HOW IDENTI	EVALN MET	REPAIR METH	DKT	REF DO	REF NO
EXCHANGER	HEAT EXCHANGER	RSHX	FOULING	ORGANISMS	BEAVER VALLEY 1	1	9	94	LOW FLOW	VISUAL	CLEANING	334	LER	940100
EXCHANGER	HEAT EXCHANGER	CCW	FOULING	ORGANISMS	WATERFORD 3	3	7	94	N. A.	NON DEST. EXAMIN	CLEANING	382	LER	940400
EXCHANGER	HEAT EXCHANGER	ERCSW	FOULING	ORGANISMS	BEAVER VALLEY 2	4	27	89	LOW FLOW	VISUAL	N. A.	412	LER	891300
EXCHANGER	HEAT EXCHANGER	RBCLC	CRACKING	N. A.	NINE MILE POINT 1	11	21	86	TEST	N. A.	N. A.	220	LER	863399
EXCHANGER	HEAT EXCHANGER	ERCSW	FOULING	ORGANISMS	OYSTER CREEK	7	5	94	EXCEED. ALLOW. LIM.	VISUAL	CLEANING	219	LER	941000
EXCHANGER	HEAT EXCHANGER	RHR	FOULING	ORGANISMS	WNP- 2	5	8	92	LOW FLOW	N. A.	N. A.	397	LER	921700
EXCHANGER	HEAT EXCHANGER	ERCSW	FOULING	ORGANISMS	CALVERT CLIFFS 2	4	25	85	LOW FLOW	VISUAL	CLEANING	318	LER	850100
EXCHANGER	HEAT EXCHANGER	ERCSW	FOULING	ORGANISMS	MILLSTONE 3	7	25	96	INSPECTION	VISUAL	N. A.	423	LER	962500
EXCHANGER	HEAT EXCHANGER	ERCSW/CS	PLUGGING	ORGANISMS	CALVERT CLIFFS 2	10	15	85	INSPECTION	VISUAL	CLEANING	318	LER	850900
EXCHANGER	HEAT EXCHANGER	ERCSW	FOULING	ORGANISMS CORROSION	BEAVER VALLEY 1	1	9	94	TEST	N. A.	CLEANING	334	LER	940100
EXCHANGER	HEAT EXCHANGER	RBCCW	CRACKING	FATIGUE	PILGRIM 1	9	18	96	LEAKING	N. A.	N. A.	293	LER	960800
EXCHANGER	HEAT EXCHANGER	RSS	FOULING	ORGANISMS	MILLSTONE 3	5	15	95	N. A.	VISUAL	CLEANING	423	LER	951100
EXCHANGER	HEAT EXCHANGER	ERCSW	FOULING	ORGANISMS	CRYSTAL RIVER 3	9	13	94	LOW FLOW	VISUAL	N. A.	302	LER	941300
EXCHANGER	HEAT EXCHANGER TUBING	CCW	WALL THINNING	EROSION	TURKEY POINT 3	1	3	91	LEAKING	ENGIN. JUGM.	N. A.	250	LER	910100
EXCHANGER	HEAT EXCHANGER TUBING	ERCSW/CS	PLUGGING	ORGANISMS	BEAVER VALLEY 2	4	27	89	TEST	VISUAL	CLEANING	412	LER	851800
EXCHANGER	HEAT EXCHANGER TUBING	ERCSW/CS	PLUGGING	FOREIGN OBJECTS	OYSTER CREEK	7	22	85	TEST	N. A.	N. A.	219	LER	851800
EXCHANGER	HEAT EXCHANGER TUBING	ERCSW/CS	PLUGGING	ORGANISMS	NORTH ANNA 2	5	13	88	N. A.	VISUAL	N. A.	339	LER	881601
EXCHANGER	ICE CONDENSER	CIC	PLUGGING	N. A.	D.C. COOK 2	9	2	87	INSPECTION	VISUAL	DEFROST	316	LER	871000
EXCHANGER	ICE CONDENSER	CIC	PLUGGING	N. A.	D.C. COOK 2	3	5	87	INSPECTION	VISUAL	DEFROST	316	LER	870200
EXCHANGER	ISOLATION CONDENS. TUBES	N. A.	CRACKING	N. A.	MILLSTONE 1	2	12	76	INSPECTION	METAL. TEST	N. A.	245	IE BUL	76-01
EXCHANGER	STEAM GEN. SUPPORTS-BOLTS	RCS	CRACKING	SCC	PRAIRIE ISLAND 1	9	4	80	FAILURE	ULTRAS. TEST	REPLACEMENT	282	IE IN	80-36
EXCHANGER	STEAM GEN. SUPPORTS- BOLTS	RCS	CRACKING	SCC	HADDAM NECK	3	22	74*	INSPECTION	ULTRAS. TEST	REPLACEMENT	213	RO BUL	74-03

COMPONENT	SUBCOMPONENT	SYSTEM	AGING EFFECTS	AGING MECHANIS	PLANT	MO	DA	YR	HOW IDENT	EVALN MET	REPAIR METH	DKT	REF DO	REF NO
EXCHANGER	STEAM GEN. TUBE PLUGS	RCS	CRACKING	PWSCC	SUMMER 1	9	8	89*	INSPECTION	N. A.	N. A.	395	NRC IN	89-65
EXCHANGER	STEAM GEN. TUBE PLUGS	RCS	CRACKING	PWSCC	NORTH ANNA 2	9	14	90*	LEAKING	NON DEST. EXAMIN	N. A.	339	NRC BUL	89-01
EXCHANGER	STEAM GEN. TUBE PLUGS	RCS	CRACKING	PWSCC	ST. LUCIE 1	11		94	LEAKING	METAL. TEST	N. A.	335	NRC IN	94-87
EXCHANGER	STEAM GEN. TUBE PLUGS	RCS	CRACKING	PWSCC	NORTH ANNA 1	9	14	90*	LEAKING	NON DEST. EXAMIN	N. A.	338	NRC BUL	89-01
EXCHANGER	STEAM GEN. TUBE PLUGS	RCS	CRACKING	IGSCC	NORTH ANNA 1	2	25	89	LEAKING	NON DEST. EXAMIN	N. A.	338	NRC IN	89-33
EXCHANGER	STEAM GEN. TUBE PLUGS	RCS	CRACKING	PWSCC	OCONEE 1	9	8	89*	INSPECTION	N. A.	N. A.	269	NRC IN	89-65
EXCHANGER	STEAM GEN. TUBE PLUGS	RCS	CRACKING	N. A.	SEQUOYAH 1	9	14	90*	LEAKING	NON DEST. EXAMIN	N. A.	327	NRC BUL	89-01
EXCHANGER	STEAM GEN. TUBE PLUGS	RCS	CRACKING	PWSCC	MCQUIRE 1, 2	7	0	89	INSPECTION	ED. CUR. TEST	REPLACEMENT	370	NRC IN	89-65
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	CORROSION	FARLEY 1	4	30	91	INSPECTION	ED. CUR. TEST	N. A.	348	LER	910300
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	CORROSION	NORTH ANNA 2	9	21	93	INSPECTION	ED. CUR. TEST	PLUGGED	339	LER	930600
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	IGSCC	FT. CALHOUN 1	3		84	LEAKING	ED. CUR. TEST	PLUGGED	285	IE IN	84-49
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	CORROSION	NORTH ANNA 1	1	10	92	INSPECTION	ED. CUR. TEST	N. A.	338	LER	900400
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	CORROSION	NORTH ANNA 1	1	30	91	INSPECTION	ED. CUR. TEST	N. A.	338	LER	910300
EXCHANGER	STEAM GEN. TUBING	RCS	RUPTURE	FOREIGN OBJECTS	PRAIRIE ISLAND 1	10	2	79	LEAKING	N. A.	N. A.	282	IE IN	79-27
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	CORROSION	KEWAUNEE	4	22	94	N. A.	NON DEST. EXAMIN	PLUGGED	305	LER	940400
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	IGA, IGSCC	KEWAUNEE	3	18	93	INSPECTION	ED. CUR. TEST	PLUGGED	305	LER	930400
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	ODSCC, IGA	TROJAN	8	13	91	PREVENT. MAINTEN.	METAL. TEST	N. A.	344	LER	912701
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	IGA, IGSCC	KEWAUNEE	10	18	96	INSERVICE INSPECT.	NON DEST. EXAMIN	PLUGGED	305	LER	960600
EXCHANGER	STEAM GEN. TUBING	RCS	RUPTURE	IGA, IGSCC	PALO VERDE 2	3	14	93	LEAKING	N. A.	N. A.	529	LER	930102
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	ODSCC	BRAIDWOOD 1	10	24	96	INSERVICE INSPECT.	N. A.	N. A.	456	LER	961200
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	ODSCC	BRAIDWOOD 1	10	24	95	INSERVICE INSPECT.	N. A.	N. A.	456	LER	951500
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	ODSCC	BRAIDWOOD 1	4	7	94	INSERVICE INSPECT.	N. A.	N. A.	456	LER	950300

COMPONENT	SUBCOMPONENT	SYSTEM	AGING EFFECTS	AGING MECHANIS	PLANT	MO	DA	YR	HOW IDENTI	EVALN MET	REPAIR METH	DKT	REF DO	REF NO
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	ODSCC	BRAIDWOOD 1	4	7	94	INSERVICE INSPECT.	N. A.	N. A.	456	LER	940700
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	ODSCC	BYRON 1	4	17	96	INSPECTION	N. A.	N. A.	454	LER	960300
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	ODSCC	BYRON 1	11	7	95	INSPECTION	ED. CUR. TEST	N. A.	454	LER	951101
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	PWSCC	SUMMER 1	4	10	90	INSPECTION	ED. CUR. TEST	PLUGGED	395	LER	900500
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	PWSCC , ODSCC	SUMMER 1	3	18	93	INSPECTION	ED. CUR. TEST	PLUGGED	395	LER	930201
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	CORROSION	FARLEY 1	11	12	92	INSPECTION	ED. CUR. TEST	N. A.	348	LER	920500
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	PWSCC	SUMMER 1	10	4	91	INSPECTION	ED. CUR. TEST	PLUGGED	395	LER	910800
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	PWSCC	DIABLO CANYON 2	4	26	96	INSPECTION	ED. CUR. TEST	N. A.	323	LER	960300
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	CORROSION	FARLEY 2	10	30	96	INSPECTION	ED. CUR. TEST	N. A.	364	LER	960300
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	CORROSION	FARLEY 2	4	5	95	INSPECTION	ED. CUR. TEST	N. A.	364	LER	950100
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	CORROSION	FARLEY 2	11	12	93	INSPECTION	ED. CUR. TEST	PLUGGED	364	LER	930300
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	CORROSION	FARLEY 2	12	16	90	INSPECTION	ED. CUR. TEST	PLUGGED	364	LER	900501
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	CORROSION	FARLEY 1	10	8	95	INSPECTION	ED. CUR. TEST	PLUGGED	348	LER	950900
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	CORROSION	FARLEY 1	3	26	94	INSPECTION	ED. CUR. TEST	PLUGGED	348	LER	940200
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	ODSCC	BYRON 1	10	6	94	INSPECTION	N. A.	N. A.	454	LER	941200
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	PWSCC	HADDAM NECK	11	10	91	PREVENT. MAINTEN.	ED. CUR. TEST	N. A.	213	LER	912301
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	CORROSION	MAINE YANKEE	3	4	95	INSPECTION	ED. CUR. TEST	PLUGGED	309	LER	950400
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	IGSCC, PWSCC	GINNA	4	14	91	PREVENT. MAINTEN.	ED. CUR. TEST	N. A.	244	LER	910500
EXCHANGER	STEAM GEN. TUBING	RCS	WALL THINNING	EROSION	MAINE YANKEE	3	14	92	INSPECTION	ENGIN. JUGM.	N. A.	309	LER	921300
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	CORROSION	HADDAM NECK	2	22	95	PREVENT. MAINTEN.	ED. CUR. TEST	N. A.	213	LER	950700
EXCHANGER	STEAM GEN. TUBING	RCS	LOSS OF MATERIAL	PWSCC	ZION 2	0	0	96	INSPECTION	ED. CUR. TEST	PLUGGED	304	NRC IN	97-26
EXCHANGER	STEAM GEN. TUBING	RCS	LOSS OF MATERIAL	PWSCC	BRAIDWOOD 2	0	0	96	INSPECTION	ED. CUR. TEST	PLUGGED	457	NRC IN	97-26

COMPONENT	SUBCOMPONENT	SYSTEM	AGING EFFECTS	AGING MECHANIS	PLANT	MO	DA	YR	HOW IDENTI	EVALN MET	REPAIR METH	DKT	REF DO	REF NO
EXCHANGER	STEAM GEN. TUBING	RCS	LOSS OF MATERIAL	PWSCC	ST. LUCIE 1	5	19	97*	INSPECTION	ED. CUR. TEST	PLUGGED	335	NRC IN	97-26
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	IGSCC, PWSCC	GINNA	4	4	93	PREVENT. MAINTEN.	ED. CUR. TEST	N. A.	244	LER	930200
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	PWSCC	HADDAM NECK	6	13	93	PREVENT. MAINTEN.	ED. CUR. TEST	N. A.	213	LER	930800
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	IGSCC, PWSCC	GINNA	4	20	92	PREVENT. MAINTEN.	ED. CUR. TEST	N. A.	244	LER	920500
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	IGSCC	FT. CALHOUN 1	5	16	84	LEAKING	ED. CUR. TEST	PLUGGED	285	IE IN	84-49
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	PWSCC	MCGUIRE 1	8	22	93	LEAKING	METAL. TEST	N. A.	369	NRC IN	94-05
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	N. A.	INDIAN POINT 3	10	19	88	LEAKING	N. A.	N. A.	286	NRC IN	94-62
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	N. A.	MCGUIRE 1	1	16	92	LEAKING	N. A.	N. A.	369	NRC IN	94-62
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	N. A.	BRAIDWOOD 1	10	23	93	LEAKING	N. A.	N. A.	456	NRC IN	94-62
EXCHANGER	STEAM GEN. TUBING	RCS	RUPTURE	SCC	DOEL2 BELGIUM	6	25	79	LEAKING	N. A.	N. A.		IE IN	79-27
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	FATIGUE	NORTH ANNA 1	7	15	87	LEAKING	N. A.	N. A.	338	NRC BUL	88-02
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	N. A.	MAINE YANKEE	7	15	94	LEAKING	DESTR. EXAMIN.	N. A.	309	NRC GL	95-03
EXCHANGER	STEAM GEN. TUBING	RCS	LOSS OF MATERIAL	PWSCC	SEQUOYAH 2	0	0	95	INSPECTION	ED. CUR. TEST	PLUGGED	328	NRC IN	97-26
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	CORROSION	POINT BEACH 2	10	14	91	INSPECTION	ED. CUR. TEST	N. A.	301	LER	910200
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	IGA, IGSCC	KEWAUNEE	5	1	95	INSERVICE INSPECT.	NON DEST. EXAMIN	PLUGGED	305	LER	950100
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	IGA, IGSCC	KEWAUNEE	3	18	92	INSPECTION	ED. CUR. TEST	PLUGGED	305	LER	920600
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	IGA, IGSCC	KEWAUNEE	4	28	91	INSPECTION	ED. CUR. TEST	PLUGGED	305	LER	910500
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	IGA, IGSCC	KEWAUNEE	4	4	90	INSPECTION	ED. CUR. TEST	PLUGGED	305	LER	900500
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	SCC	ZION 2	10	11	96	INSPECTION	ED. CUR. TEST	N. A.	304	LER	960800
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	IGSCC, PWSCC	GINNA	4	16	90	PREVENT. MAINTEN.	ED. CUR. TEST	N. A.	244	LER	900400
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	CORROSION	MAINE YANKEE	9	3	93	INSPECTION	ED. CUR. TEST	PLUGGED	309	LER	931800
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	PWSCC	MAINE YANKEE	7	27	94	LEAKING	ED. CUR. TEST	PLUGGED	309	LER	941201

COMPONENT	SUBCOMPONENT	SYSTEM	AGING EFFECTS	AGING MECHANIS	PLANT	MO	DA	YR	HOW IDENTI	EVALN MET	REPAIR METH	DKT	REF DO	REF NO
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	CORROSION	MAINE YANKEE	12	17	90	LEAKING	NON DEST. EXAMIN	PLUGGED	309	LER	901200
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	CORROSION	POINT BEACH 2	10	22	92	INSPECTION	ED. CUR. TEST	N. A.	301	LER	920500
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	CORROSION	POINT BEACH 2	10	23	90	INSPECTION	ED. CUR. TEST	PLUGGED	301	LER	900300
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	PWSCC	DIABLO CANYON 1	10	22	95	PREVENT. MAINTEN.	ED. CUR. TEST	N. A.	275	LER	951801
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	PWSCC	DIABLO CANYON 1	4	3	94	PREVENT. MAINTEN.	ED. CUR. TEST	N. A.	275	LER	942100
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	CORROSION	POINT BEACH 2	10	21	86	EDDY CURR. TEST	N. A.	PLUGGED	301	LER	860700
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	IGSCC, PWSCC	GINNA	4	7	95	PREVENT. MAINTEN.	ED. CUR. TEST	N. A.	244	LER	950400
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	CORROSION	POINT BEACH 2	10	10	94	INSPECTION	ED. CUR. TEST	N. A.	301	LER	940300
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	IGSCC, PWSCC	GINNA	3	23	94	PREVENT. MAINTEN.	ED. CUR. TEST	N. A.	244	LER	940600
EXCHANGER	STEAM GEN. TUBING	RCS	CRACKING	PWSCC , ODSCC	ZION 1	10	5	95	INSPECTION	ED. CUR. TEST	N. A.	295	LER	952000
EXCHANGER	STEAM GEN. UPPER SHELL	RCS	CRACKING	THERMAL FATIGUE & OTHER	INDIAN POINT 2	0	0	89	INSERVICE INSPECT.	ULTRAS. TEST	GRINDING	247	NRC IN	90-04
EXCHANGER	STEAM GEN. UPPER SHELL	RCS	CRACKING	THERMAL FATIGUE	ZION 1	0	0	89	INSERVICE INSPECT.	ULTRAS. TEST	GRINDING	295	NRC IN	90-04
EXCHANGER	STEAM GEN. UPPER SHELL	RCS	CRACKING	THERMAL FATIGUE & OTHER	INDIAN POINT 2	0	0	87	INSERVICE INSPECT.	ULTRAS. TEST	GRINDING	247	NRC IN	90-04
EXCHANGER	STEAM GEN. UPPER SHELL	RCS	CRACKING	THERMAL FATIGUE	INDIAN POINT 3	0	0	82	LEAKING	N. A.	N. A.	286	IE IN	82-37
EXCHANGER	STEAM GENER. STUDS	RCS	CRACKING	SCC	ARKANSAS 1	3	12	82*	VISUAL	ULTRAS. TEST	N. A.	313	IE IN	82-06
EXCHANGER	STEAM GENER. STUDS	RCS	CRACKING	SCC	OCONEE	3	12	82*	VISUAL	ULTRAS. TEST	N. A.		IE IN	82-06
EXCHANGER	STEAM GENER. STUDS	RCS	CRACKING	CHEMICAL ATTACK	MAINE YANKEE	3	12	82*	VISUAL	ULTRAS. TEST	REPLACEMENT	309	IE IN	82-06
EXCHANGER	STEAM GENERATOR	RCS	CRACKING	CORROSION FATIGUE	INDIAN POINT 3			82	LEAKING	ULTRAS. TEST	REPAIR	286	IE IN	85-65
EXCHANGER	STEAM GENERATOR	RCS	CRACKING	SCC FATIGUE	SURRY 2			83	INSPECTION	ULTRAS. TEST	GRINDING	281	IE IN	85-65
FILTER	CHARCOAL	ABVS	FAILURE	AGING/ END OF LIFE	DIABLO CANYON 2	8	8	94	TEST	ANALYS.	REPLACEMENT	323	LER	940500
FILTER	CHARCOAL	ABVS	FAILURE	CHEMICAL ATTACK	SURRY 2	12	5	94	ODOR	TEST	REPLACEMENT	281	NRC IN	95-41
FILTER	CHARCOAL	ABVS	FAILURE	AGING/ END OF LIFE	DIABLO CANYON 2	7	7	94	LOW FLOW	ANALYS.	REPLACEMENT	323	LER	940301

COMPONENT	SUBCOMPONENT	SYSTEM	AGING EFFECTS	AGING MECHANIS	PLANT	MO	DA	YR	HOW IDENTI	EVALN MET	REPAIR METH	DKT	REF DO	REF NO
FILTER	CHARCOAL	SBGT	FAILURE	CHEMICAL ATTACK	DRESDEN 2	1	17	91	TEST	TEST	REPLACEMENT REPAIR	237	LER	910902
FILTER	CHARCOAL	ABVS	FAILURE	AGING/ END OF LIFE	FERMI 2	5	19	91	N. A.	ANALYS.	REPLACEMENT	341	LER	911000
FILTER	CHARCOAL	FHB	FAILURE	AGING/ END OF LIFE	DIABLO CANYON 2	8	18	94	TEST	ANALYS.	REPLACEMENT	323	LER	940500
FILTER	CHARCOAL	SBGT	FAILURE	AGING/ END OF LIFE	QUAD CITIES 1	7	14	93	LOW FLOW	INSPECTION	REPLACEMENT	254	LER	930900
FILTER	HOUSING	ERCSW	LOSS OF MATERIAL	EROSION	HADDAM NECK	8	16	93	PREVENT. MAINTEN.	ENGIN. EVALUAT.	N. A.	213	LER	931500
FILTER	PROCESS FILTER	ERCSW	PLUGGING	FOREIGN OB. SILT/DEBRIS	HADDAM NECK	4	25	92	EXCEED. ALLOW. LIM.	N. A.	N. A.	213	LER	921200
FILTER	PROCESS FILTER	ERCSW	PLUGGING	FOREIGN OBJECTS SILT/DEBRIS	HADDAM NECK	6	23	92	N. A.	N. A.	CLEANING	213	LER	921500
FILTER	PROCESS FILTER	ABVS	FAILURE	AGING/ END OF LIFE	SURRY 2	4	19	92	LOW FLOW	ENGIN. JUGM.	REPLACEMENT	281	LER	920600
FILTER	SCREEN	ERCSW	CRACKING	AGING/ END OF LIFE	SOUTH TEXAS 2	5	26	93	TEST	VISUAL	REPLACEMENT	499	LER	931000
FILTER	SCREEN	CIRCULAT. WATER	PLUGGING	FOREIGN OB. DEBRIS	DRESDEN 3	10	29	95	LOW FLOW	VISUAL	CLEANING	249	LER	951900
FILTER	SCREEN	CIRCULAT. WATER	EXCESSIVE DEFORMN.	MECHANICAL LOADS	MILLSTONE 1	10	4	90	N. A.	VISUAL	CLEANING	245	LER	901601
FILTER	SCREEN	CIRCULAT. WATER	PLUGGING	FOREIGN OB. DEBRIS	MILLSTONE 1	10	4	90	N. A.	VISUAL	CLEANING	245	LER	901601
FILTER	SCREEN	CIRCULAT. WATER	PLUGGING	FOREIGN OB. DEBRIS	MILLSTONE 3	3	30	90	N. A.	VISUAL	CLEANING	423	LER	901100
FILTER	SCREEN	CIRCULAT. WATER	PLUGGING	FOREIGN OB. DEBRIS	PERRY 1	6	9	94	N. A.	VISUAL	N. A.	440	LER	941501
FILTER	SCREEN	CIRCULAT. WATER	PLUGGING	FOREIGN OB. DEBRIS	MILLSTONE 3	4	5	92	N. A.	VISUAL	CLEANING	423	LER	921100
FILTER	SCREEN	EDG	PLUGGING	PARTICLES RUST	LIMERICK 1	8	20	96	LOW FLOW	VISUAL	CLEANING	352	LER	961700
FILTER	SCREEN	CIRCULAT. WATER	FOULING	ORGANISMS	DIABLO CANYON 2	12	19	94	N. A.	VISUAL	ALARM INSTAL.	323	LER	941200
FILTER	SCREEN	CIRCULAT. WATER	FAILURE	ORGANISMS	WOLF CREEK 1	10	7	85	N. A.	VISUAL	REPAIR	482	LER	856900
FILTER	SCREEN	CIRCULAT. WATER	PLUGGING	FOREIGN OB. DEBRIS	FITZPATRICK	10	19	90	LOW FLOW	N. A.	N. A.	333	LER	902300
FILTER	STRAINER	EGF	FOULING	ORGANISMS	ARKANSAS 2	6	27	86	INSPECTION	TEST	REMOVAL	368	LER	861401
FILTER	STRAINER	RCIC	PLUGGING	PARTICLES RUST	COOPER	3	25	92	LEAKING	ANALYS.	CLEANING	298	LER	920500
FILTER	STRAINER	RHR	FOULING	ORGANISMS	QUAD CITIES 1	8	8	96	PREVENT. MAINTEN.	INSPECTION	CLEANING	254	LER	961300

COMPONENT	SUBCOMPONENT	SYSTEM	AGING EFFECTS	AGING MECHANIS	PLANT	MO	DA	YR	HOW IDENTI	EVALN MET	REPAIR METH	DKT	REF DO	REF NO
FILTER	STRAINER	ERCSW	FOULING	ORGANISMS	MILLSTONE 3	10	24	96	LEAKING	VISUAL	REPLACEMENT	423	LER	964100
FILTER	STRAINER	ERCSW	PLUGGING	FOREIGN OB. SILT/DEBRIS	HADDAM NECK	8	23	91	EXCEED. ALLOW.LIM.	N. A.	CLEANING	213	LER	911700
FILTER	STRAINER	ERCSW	PLUGGING	ORGANISMS	SURRY 1	7	12	92	LOW FLOW	VISUAL	REPLACEMENT	280	LER	920900
FILTER	STRAINER	ERCSW	PLUGGING	FOREIGN OB. DEBRIS	HADDAM NECK	10	11	95	PREVENT. MAINTEN.	N. A.	CLEANING	213	LER	951900
FILTER	STRAINER	ERCSW	PLUGGING	FOREIGN OB. DEBRIS	HADDAM NECK	11	1	93	PREVENT. MAINTEN.	N. A.	CLEANING	213	LER	931700
FILTER	STRAINER	ERCSW	PLUGGING	FOREIGN OBJECTS	TURKEY POINT 3	1	31	96	LOW FLOW	VISUAL	CLEANING	250	LER	960100
FILTER	STRAINER	ERCSW	PLUGGING	FOREIGN OBJECTS AQ. GRASS	TURKEY POINT 3	1	3	96	LOW FLOW	VISUAL	CLEANING	250	LER	950300
HVAC DUCT	DUCT	CRDM	LOSS OF MATERIAL	CHEMICAL ATTACK	TURKEY POINT 4	3	13	87	INSPECTION	NON DEST. EXAMIN	N. A.	251	NRC IN	86-108-1
HVAC DUCT	DUCT	CBEAF	FAILURE	WEAR	BRUNSWICK 1	10	25	95	LEAKING	N. A.	REPAIR	325	LER	952001
HVAC DUCT (UG)	DUCT COUPLING SEAL	SGT	FAILURE	MOISTURE RAIN	BROWNS FERRY 2	5	14	91	OSCILLATIN G FLOW	N. A.	REPAIR/MODIF.	260	LER	911100
INSULATION	CERAMIC INSULATORS	4160/480 V TRANSFORM.	CRACKING	N. A.	DIABLO CANYON 1, 2	8	26	92*	VISUAL	N. A.	REPLACEMENT	275	NRC IN	92-63
INSULATION	CERAMIC INSULATORS	4160/480 V TRANSFORM.	CRACKING	N. A.	WNP-3	8	26	92*	VISUAL	N. A.	N. A.	508	NRC IN	92-63
PIPING SYSTEM	PENETRATION SEAL	CCSW	DETERIORATION	MECHANICAL WEAR	DRESDEN 2	7	14	95	TEST	TEST	TIGHTENING	237	LER	951700
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	TURKEY POINT 3	4	22	88*	NRC NOTIFIC.	ULTRAS. TEST	N. A.	250	NRC IN	88-17
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	FORT ST. VRAIN	4	22	88*	NRC NOTIFIC.	ULTRAS. TEST	N. A.	267	NRC IN	88-17
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	TROJAN	4	22	88*	NRC NOTIFIC.	ULTRAS. TEST	N. A.	344	NRC IN	88-17
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	SEQUOYAH 1	4	22	88*	NRC NOTIFIC.	ULTRAS. TEST	N. A.	327	NRC IN	88-17
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	SURRY 1	4	22	88*	NRC NOTIFIC.	ULTRAS. TEST	N. A.	280	NRC IN	88-17
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	RANCHO SECO	4	22	88*	NRC NOTIFIC.	ULTRAS. TEST	N. A.	312	NRC IN	88-17
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	SALEM 1	4	22	88*	NRC NOTIFIC.	ULTRAS. TEST	N. A.	272	NRC IN	88-17
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	SAN ONOFRE 1	4	22	88*	NRC NOTIFIC.	ULTRAS. TEST	N. A.	206	NRC IN	88-17
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	ROBINSON 2	4	22	88*	NRC NOTIFIC.	ULTRAS. TEST	N. A.	261	NRC IN	88-17

COMPONENT	SUBCOMPONENT	SYSTEM	AGING EFFECTS	AGING MECHANIS	PLANT	MO	DA	YR	HOW IDENTI	EVALN MET	REPAIR METH	DKT	REF DO	REF NO
PIPING SYSTEM	PIPING	RCS	WALL THINNING	EROSION	SALEM 1	6	11	90	LEAKING	ENGIN. JUGM.	N. A.	272	LER	902604
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	NORTH ANNA 1	4	22	88*	NRC NOTIFIC.	ULTRAS. TEST	N. A.	338	NRC IN	88-17
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	SHEARON HARRIS 1	4	22	88*	NRC NOTIFIC.	ULTRAS. TEST	N. A.	400	NRC IN	88-17
PIPING SYSTEM	PIPING	RS	CRACKING	FATIGUE	BROWNS FERRY 2	3		83*	NRC NOTIFIC.	METAL. TEST	N. A.	260	IE BUL	83-02
PIPING SYSTEM	PIPING	RCS	WALL THINNING	IMPROPER DESIGN	SAN ONOFRE 3	5	10	90	INSPECTION	NON DEST. EXAMIN	REPLACEMENT	362	NRC IN	91-19
PIPING SYSTEM	PIPING	BORIC ACID TANK	CRACKING	SCC	PRAIRIE ISLAND 1	1	29	83	LEAKING	METAL. TEST	N. A.	282	IE IN	84-18
PIPING SYSTEM	PIPING	MCL	CRACKING	THERMAL FATIGUE	OCONEE 3	3	31	82*	INSPECTION	RADIOG. TEST	N. A.	287	IE IN	82-09
PIPING SYSTEM	PIPING	MCL	CRACKING	THERMAL FATIGUE	OCONEE 2	3	31	82*	LEAKING	RADIOG. TEST	N. A.	270	IE IN	82-09
PIPING SYSTEM	PIPING	FEEDWATER	CRACKING	CORROSION FATIGUE	MAINE YANKEE	10	16	79*	NRC NOTIFIC.	RADIOG. TEST	N. A.	309	IE BUL	79-13-2
PIPING SYSTEM	PIPING	FEEDWATER	CRACKING	CORROSION FATIGUE	MILLSTONE 2	10	16	79*	NRC NOTIFIC.	RADIOG. TEST	N. A.	336	IE BUL	79-13-2
PIPING SYSTEM	PIPING	FEEDWATER	CRACKING	CORROSION FATIGUE	GINNA	10	16	79*	NRC NOTIFIC.	RADIOG. TEST	N. A.	244	IE BUL	79-13-2
PIPING SYSTEM	PIPING	FEEDWATER	CRACKING	CORROSION FATIGUE	SURRY 1	10	16	79*	NRC NOTIFIC.	RADIOG. TEST	N. A.	280	IE BUL	79-13-2
PIPING SYSTEM	PIPING	FEEDWATER	CRACKING	CORROSION FATIGUE	POINT BEACH 2	10	16	79*	NRC NOTIFIC.	RADIOG. TEST	N. A.	301	IE BUL	79-13-2
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	SAN ONOFRE 2	4	22	88*	NRC NOTIFIC.	ULTRAS. TEST	N. A.	361	NRC IN	88-17
PIPING SYSTEM	PIPING	FEEDWATER	CRACKING	CORROSION FATIGUE	D.C. COOK 1	10	16	79*	NRC NOTIFIC.	RADIOG. TEST	N. A.	315	IE BUL	79-13-2
PIPING SYSTEM	PIPING	RSHX	FOULING	ORGANISMS	SURRY 1	10	23	90	INSPECTION	ENGIN. JUGM.	N. A.	280	LER	901401
PIPING SYSTEM	PIPING	RWCS	CRACKING	SCC	NINE MILE POINT 1	3	22	76	LEAKING	NON DEST. EXAMIN	N. A.	220	IE BUL	76-04
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	SALEM 2	4	22	88*	NRC NOTIFIC.	ULTRAS. TEST	N. A.	311	NRC IN	88-17
PIPING SYSTEM	PIPING	FEEDWATER	FAILURE	EROSION	SURRY 2	12	9	86	NRC NOTIFIC.	ULTRAS. TEST	N. A.	281	NRC IN	88-17
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	PERRY 1	4	22	88*	NRC NOTIFIC.	ULTRAS. TEST	N. A.	440	NRC IN	88-17
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	SAN ONOFRE 3	4	22	88*	NRC NOTIFIC.	ULTRAS. TEST	N. A.	362	NRC IN	88-17
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	MILLSTONE 2	4	22	88*	NRC NOTIFIC.	ULTRAS. TEST	N. A.	336	NRC IN	88-17

COMPONENT	SUBCOMPONENT	SYSTEM	AGING EFFECTS	AGING MECHANIS	PLANT	MO	DA	YR	HOW IDENTI	EVALN MET	REPAIR METH	DKT	REF DO	REF NO
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	DIABLO CANYON 2	4	22	88*	NRC NOTIFIC.	ULTRAS. TEST	N. A.	323	NRC IN	88-17
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	CALVERT CLIFFS 2	4	22	88*	NRC NOTIFIC.	ULTRAS. TEST	N. A.	318	NRC IN	88-17
PIPING SYSTEM	PIPING	CIRCULAT. WATER	PLUGGING FOULING	ORGANISMS	CALVERT CLIFFS 1	5	5	90	LEAKING	ENGIN. JUGM.	N. A.	317	LER	901700
PIPING SYSTEM	PIPING	FEEDWATER	CRACKING	FATIGUE	KEWAUNEE	10	16	79*	NRC NOTIFIC.	RADIOG. TEST	N. A.	305	IE BUL	79-13-2
PIPING SYSTEM	PIPING	RS	CRACKING	IGSCC	MONTICELLO	3	4	83*	NRC NOTIFIC.	ULTRAS. TEST	N. A.	263	IE BUL	83-02
PIPING SYSTEM	PIPING	MCL	CRACKING	THERMAL FATIGUE	CRYSTAL RIVER 3	1	21	82	LEAKING	METAL. TEST	N. A.	302	IE IN	82-09
PIPING SYSTEM	PIPING	SIPSL	CRACKING	IGSCC	GINNA	10	8	76	N. A.	METAL. TEST	N. A.	244	IE IN	79-19
PIPING SYSTEM	PIPING	RS	CRACKING	IGSCC	SURRY 2	10	7	76	INSPECTION	METAL. TEST	N. A.	281	IE IN	79-19
PIPING SYSTEM	PIPING	SFCS	CRACKING	IGSCC	THREE MILE ISLAND 1	5	16	79	INSPECTION	METAL. TEST	N. A.	289	IE IN	79-19
PIPING SYSTEM	PIPING	FEEDWATER	CRACKING	N. A.	D.C. COOK 1	5	25	79	LEAKING	N. A.	N. A.	315	GL	79020
PIPING SYSTEM	PIPING	RBS & BWMS	CRACKING	IGSCC	ARKANSAS 1	11	7	74	INSPECTION	NON DEST. EXAMIN	N. A.	313	IE CIRC	76-06
PIPING SYSTEM	PIPING	RCS	CRACKING	THERMAL FATIGUE	TIHANGE 1 BELGIUM	12	9	86	LEAKING	METAL. TEST	N. A.		NRC BUL	88-08 -1
PIPING SYSTEM	PIPING	RCS	CRACKING	THERMAL FATIGUE	FARLEY 2	12	9	87	LEAKING	METAL. TEST	N. A.	364	NRC BUL	88-08
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	ARKANSAS 2	4	22	88*	NRC NOTIFIC.	N. A.	N. A.	368	NRC IN	88-17
PIPING SYSTEM	PIPING	RS , RHR	CRACKING	IGSCC	HATCH 1	3	4	83*	NRC NOTIFIC.	ULTRAS. TEST	N. A.	321	IE BUL	83-02
PIPING SYSTEM	PIPING	FEEDWATER	LOSS OF MATERIAL	N. A.	SUSQUEHANNA 1	12	16	81	INSPECTION	NON DEST. EXAMIN	N. A.	387	NRC IN	92-35
PIPING SYSTEM	PIPING	RS	CRACKING	IGSCC	NINE MILE POINT 1	3		82	HYDROTEST	ULTRAS. TEST	N. A.	220	IE BUL	82-03
PIPING SYSTEM	PIPING	FEEDWATER	CRACKING	CORROSION FATIGUE	DIABLO CANYON 1	3	17	77	NRC NOTIFIC.	RADIOG. TEST	N. A.	275	IE BUL	79-13-2
PIPING SYSTEM	PIPING	FEEDWATER	CRACKING	CORROSION FATIGUE	SALEM 1	7	20	79	NRC NOTIFIC.	RADIOG. TEST	N. A.	272	IE BUL	79-13-2
PIPING SYSTEM	PIPING	FEEDWATER	CRACKING	CORROSION FATIGUE	BEAVER VALLEY 1	7	18	79	NRC NOTIFIC.	RADIOG. TEST	N. A.	334	IE BUL	79-13-2
PIPING SYSTEM	PIPING	FEEDWATER	CRACKING	FATIGUE	ROBINSON 2	7	15	79	NRC NOTIFIC.	RADIOG. TEST	N. A.	261	IE BUL	79-13-2
PIPING SYSTEM	PIPING	FEEDWATER	CRACKING	SCC	SAN ONOFRE 1	6	5	79	NRC NOTIFIC.	RADIOG. TEST	N. A.	206	IE BUL	79-13-2

COMPONENT	SUBCOMPONENT	SYSTEM	AGING EFFECTS	AGING MECHANIS	PLANT	MO	DA	YR	HOW IDENTI	EVALN MET	REPAIR METH	DKT	REF DO	REF NO
PIPING SYSTEM	PIPING	FEEDWATER	CRACKING	CORROSION FATIGUE	D.C. COOK 2	5	20	79	LEAKING	RADIOG. TEST	N. A.	316	IE BUL	79-13-2
PIPING SYSTEM	PIPING	RCS	CRACKING	N. A.	QUAD CITIES 2	9	15	74	N. A.	N. A.	N. A.	265	RO BUL	74-10
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	SURRY 2	12	9	82	FAILURE	NON DEST. EXAMIN	N. A.	281	NRC BUL	87-01
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	DUANE ARNOLD	4	22	88*	NRC NOTIFIC.	ULTRAS. TEST	N. A.	331	NRC IN	88-17
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	D.C. COOK 2	4	22	88*	NRC NOTIFIC.	ULTRAS. TEST	N. A.	316	NRC IN	88-17
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	DIABLO CANYON 1	4	22	88*	NRC NOTIFIC.	ULTRAS. TEST	N. A.	275	NRC IN	88-17
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	CALLAWAY	4	22	88*	NRC NOTIFIC.	ULTRAS. TEST	N. A.	483	NRC IN	88-17
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	CALVERT CLIFFS 1	4	22	88*	NRC NOTIFIC.	ULTRAS. TEST	N. A.	317	NRC IN	88-17
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	SEQUOYAH 1	11	29	94	LEAKING	NON DEST. EXAMIN	N. A.	327	NRC IN	95-11
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	ARKANSAS 1	4	22	88*	NRC NOTIFIC.	ULTRAS. TEST	N. A.	313	NRC IN	88-17
PIPING SYSTEM	PIPING	RCS	LOSS OF MATRL.-HOLE	CORROSION	DIABLO CANYON 2	10	1	94	N. A.	ENGIN. JUGM.	WELDING	323	LER	940600
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	RIVER BEND 1	4	22	88*	NRC NOTIFIC.	ULTRAS. TEST	N. A.	458	NRC IN	88-17
PIPING SYSTEM	PIPING	RCS	LOSS OF MATERIAL	CORROSION CHEM. ATTACK	ARKANSAS 1	10		86	INSPECTION	VISUAL	GRINDING	313	IE IN	86-108
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	PILGRIM 1	4	22	88*	NRC NOTIFIC.	ULTRAS. TEST	N. A.	293	NRC IN	88-17
PIPING SYSTEM	PIPING	RCS	WALL THINNING	IMPROPER DESIGN	SAN ONOFRE 2	3	12	91	INSPECTION	NON DEST. EXAMIN	REPLACEMENT	361	NRC IN	91-19
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	DRESDEN 2	4	22	88*	NRC NOTIFIC.	N. A.	N. A.	237	NRC IN	88-17
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	LASALLE 1	12	10	87	LEAKING	ULTRAS. TEST	N. A.	373	NRC IN	88-17
PIPING SYSTEM	PIPING	RS	CRACKING	IGSCC	BRUNSWICK 1			88	INSPECTION	METAL. TEST	N. A.	325	NRC IN	90-30
PIPING SYSTEM	PIPING	MSR	WALL THINNING	EROSION	MILLSTONE 2	10	16	91	RUPTURE	NON DEST. EXAMIN	N. A.	336	IE IN	91-18
PIPING SYSTEM	PIPING	JPIL	CRACKING	IGSCC	BROWNS FERRY 3	8		84	INSERVICE INSPECT.	ULTRAS. TEST	N. A.	296	IE IN	84-41
PIPING SYSTEM	PIPING	PRPS	CRACKING	IGSCC	MONTICELLO	2	3	84	INSERVICE INSPECT.	METAL. TEST	N. A.	263	IE IN	84-41
PIPING SYSTEM	PIPING	CSI	CRACKING	IGSCC	QUAD CITIES 2	3	8	80	LEAKING	METAL. TEST	N. A.	265	IE IN	80-15

COMPONENT	SUBCOMPONENT	SYSTEM	AGING EFFECTS	AGING MECHANIS	PLANT	MO	DA	YR	HOW IDENTI	EVALN MET	REPAIR METH	DKT	REF DO	REF NO
PIPING SYSTEM	PIPING	FEEDWATER	CRACKING	THERMAL FATIGUE	DIABLO CANYON 1			92	VISUAL	ULTRAS. TEST	N. A.	275	NRC IN	93-20
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	FT. CALHOUN 1	4	22	88*	NRC NOTIFIC.	ULTRAS. TEST	N. A.	285	NRC IN	88-17
PIPING SYSTEM	PIPING	FEEDWATER	WALL THINNING	EROSION	OYSTER CREEK	4	22	88*	NRC NOTIFIC.	ULTRAS. TEST	N. A.	219	NRC IN	88-17
PIPING SYSTEM	PIPING	EGF	LOSS OF MATERIAL	CORROSION MOISTURE	DIABLO CANYON 1	7	2	92	TEST	ULTRAS. TEST	N. A.	275	LER	920601
PIPING SYSTEM	PIPING	ERCSW	FOULING	CORROSION ORGANISMS	WOLF CREEK 1	5	1	90	TEST	ENGIN. JUGM.	N. A.	482	LER	931400
PIPING SYSTEM	PIPING	PRPS	CRACKING	IGSCC	PILGRIM 1	7	1	84*	INSERVICE INSPECT.	METAL. TEST	N. A.	293	IE IN	84-41
PIPING SYSTEM	PIPING	ERCSW	FOULING	ORGANISMS	MILLSTONE 3	7	25	91	LOW FLOW	N. A.	N. A.	423	LER	911901
PIPING SYSTEM	PIPING	RCS	CRACKING	VIBRATION	VERMONT YANKEE	10	13	94	LEAKING	N. A.	REPLACEMENT	271	LER	941301
PIPING SYSTEM	PIPING	DIESEL FUEL OIL	EXCESSIVE DEFORMN.	CORROSION	DIABLO CANYON 1	7	2	92	TEST	N. A.	REPAIR	275	LER	920601
PIPING SYSTEM	PIPING	RHR	FAILURE	VIBRATION	CATAWBA 2	1	31	93	N. A.	N. A.	N. A.	414	LER	930400
PIPING SYSTEM	PIPING	RHR	CRACKING	N. A.	MILLSTONE 3	12	2	96	LEAKING	N. A.	N. A.	423	LER	952001
PIPING SYSTEM	PIPING	RCS	CRACKING	N. A.	DRESDEN 2	9	13	74	LEAKING	METAL. TEST	N. A.	237	RO BUL	74-10
PIPING SYSTEM	PIPING	MSR	WALL THINNING	EROSION	MILLSTONE 3	3	12	91	RUPTURE	ANALYS.	N. A.	423	IE IN	91-18
PIPING SYSTEM	SUPPORT	RAW WATER - INTAKE STR.	LOSS OF MATERIAL	N. A.	BEAVER VALLEY 1			95*	VISUAL	N. A.	N. A.	334	NUREG	1522
PIPING SYSTEM	SUPPORT	SEVERAL	LOSS OF MATERIAL	CORROSION MOISTURE	COOPER			95*	VISUAL	N. A.	N. A.	298	NUREG	1522
PIPING SYSTEM	SUPPORT	ECCS	FAILURE	VIBRATION	QUAD CITIES 2	6	27	72*	FAILURE	ENGIN. JUG.	N. A.	265	IE BUL	72-01
PIPING SYSTEM	SUPPORT	N. A.	LOSS OF MATERIAL	N. A.	TROJAN			95*	VISUAL	N. A.	N. A.	344	NUREG	1522
PIPING SYSTEM	SUPPORT	FEEDWATER	LOSS OF MATERIAL	CORROSION MOISTURE	ROBINSON 2			95*	VISUAL	N. A.	N. A.	261	NUREG	1522
PIPING SYSTEM	SUPPORT	CVCS	EXCESSIVE DEFORMN.	N. A.	D.C. COOK 2	1	16	89	INSPECTION	VISUAL	N. A.	316	LER	890201
PIPING SYSTEM	SUPPORT	CVCS	LOSS OF MATERIAL	CORROSION CHEM. ATTACK	MILLSTONE 2	5	16	95	TEST	METAL. TEST	N. A.	336	LER	952300
PIPING SYSTEM	SUPPORT - HANGER	RHR	LOOSENING	MECHANICAL LOADS	CRYSTAL RIVER 3	2	2	86	INSPECTION	VISUAL	REPAIR	302	LER	860302
PIPING SYSTEM	SUPPORT - HANGER	RHR	FAILURE	VIBRATION	BROWNS FERRY 3	7	16	85	VISUAL	N. A.	REPAIR	296	LER	851701

COMPONENT	SUBCOMPONENT	SYSTEM	AGING EFFECTS	AGING MECHANIS	PLANT	MO	DA	YR	HOW IDENT	EVALN MET	REPAIR METH	DKT	REF DO	REF NO
PIPING SYSTEM	SUPPORT - HANGER	RHR	FAILURE	VIBRATION	BROWNS FERRY 3	6	28	86	INSPECTION	METAL. TEST	REPAIR	296	LER	860700
PIPING SYSTEM	SUPPORT - HANGER	RHR	CRACKING	FATIGUE	BROWNS FERRY 3	6	28	86	INSPECTION	METAL. TEST	REPAIR	296	LER	851701
PIPING SYSTEM	SUPPORT - HANGER	RHR	CRACKING	VIBRATION	BROWNS FERRY 3	9	8	86	INSPECTION	VISUAL	REPAIR	296	LER	860900
PIPING SYSTEM	SUPPORT - HANGER	SERVICE WATER	LOSS OF MATERIAL	CORROSION MOISTURE	ROBINSON 2			95*	VISUAL	N. A.	N. A.	261	NUREG	1522
PIPING SYSTEM	SUPPORT - SNUBBER	N. A.	LOSS OF MATER. & OTHER	VIBRATION & OTHER	SAN ONOFRE 3	9	27	85	VISUAL	VISUAL	REPLACEMENT	362	LER	852901
PIPING SYSTEM	SUPPORT - SNUBBER - PIN	CCW	FAILURE	VIBRATION	SAN ONOFRE 2	5	23	89	VISUAL	VISUAL	REPLACEMENT	361	LER	890900
PIPING SYSTEM	TUBING	IASL	FAILURE	VIBRATION	LIMERICK 2	7	13	90	FAILURE	ENGIN. JUGM.	REPAIR	353	LER	901101
RPV	CORE SHROUD	RCS	CRACKING	IGSCC	KUOSHENG 2 TAIWAN	12	31	96*	N. A.	N. A.	N. A.		INER	72217
RPV	CORE SHROUD	RCS	CRACKING	IGSCC	HATCH 2	3		96*	INSPECTION	ULTRAS. TEST	REPAIR	366	NUREG	1544
RPV	CORE SHROUD	RCS	CRACKING	IGSCC	BROWNS FERRY 3			94	INSPECTION	ULTRAS. TEST	N. A.	296	BWRVIP	97-366
RPV	CORE SHROUD	RCS	CRACKING	IGSCC	FUKUSHIMA DAIICHI 2	12	31	96*	N. A.	N. A.	N. A.		INER	T2217
RPV	CORE SHROUD	RCS	CRACKING	IGSCC	MUEHLEBERG SWITZERLAND	12	31	96*	N. A.	N. A.	N. A.		INER	T2217
RPV	CORE SHROUD	RCS	CRACKING	IGSCC	SPAIN	11	10	94*	N. A.	N. A.	N. A.		SECY	94-276
RPV	CORE SHROUD	RCS	CRACKING	IGSCC	KKM SWITZERLAND	11	10	94	NOT AVAIL.	N. A.	N. A.		SECY	94-276
RPV	CORE SHROUD	RCS	CRACKING	IGSCC	NINE MILE POINT 1			95	INSPECTION	VISUAL	REPAIR	220	NRC IN	97-17
RPV	CORE SHROUD	RCS	CRACKING	IGSCC	SUSQUEHANNA 1			93	INSPECTION	VISUAL, ULTRAS.	N. A.	387	BWRVIP	97-366
RPV	CORE SHROUD	RCS	CRACKING	IGSCC	VERMONT YANKEE			95	INSPECTION	ULTRAS. TEST	REPAIR	271	BWRVIP	97-366
RPV	CORE SHROUD	RCS	CRACKING	IGSCC	MILLSTONE 1			91	INSPECTION	N. A.	N. A.	245	BWRVIP	97-366
RPV	CORE SHROUD	RCS	CRACKING	IGSCC	BROWNS FERRY 2			94	INSPECTION	ULTRAS. TEST	N. A.	260	BWRVIP	97-366
RPV	CORE SHROUD	RCS	CRACKING	IGSCC	OYSTER CREEK			94	INSPECTION	VISUAL, ULTRAS.	REPAIR	219	BWRVIP	97-366
RPV	CORE SHROUD	RCS	CRACKING	IGSCC	BRUNSWICK 2			96	INSPECTION	ULTRAS. TEST	N. A.	324	BWRVIP	97-366
RPV	CORE SHROUD	RCS	CRACKING	IGSCC	BRUNSWICK 1			93	INSPECTION	ULTRAS. TEST	N. A.	325	BWRVIP	97-366

COMPONENT	SUBCOMPONENT	SYSTEM	AGING EFFECTS	AGING MECHANIS	PLANT	MO	DA	YR	HOW IDENTI	EVALN MET	REPAIR METH	DKT	REF DO	REF NO
RPV	CORE SHROUD	RCS	CRACKING	IGSCC	QUAD CITIES 1			94	INSPECTION	ULTRAS. TEST	N. A.	254	BWRVIP	97-366
RPV	CORE SHROUD	RCS	CRACKING	IGSCC	DRESDEN 3			94	INSPECTION	VISUAL, ULTRAS.	N. A.	249	BWRVIP	97-366
RPV	CORE SHROUD	RCS	CRACKING	IGSCC	PEACH BOTTOM 2			94	INSPECTION	ULTRAS. TEST	N. A.	277	BWRVIP	97-366
RPV	CORE SHROUD	RCS	CRACKING	IGSCC	CHISHAN 1 TAIWAN	12	31	96*	N. A.	N. A.	N. A.		INER	T2217
RPV	CORE SHROUD	RCS	CRACKING	IGSCC	MONTICELLO			94	INSPECTION	VISUAL, ULTRAS.	N. A.	263	BWRVI R	97-366
RPV	CORE SHROUD	RCS	CRACKING	IGSCC	WUERGRASSE N GERMANY	12	31	96*	N. A.	N. A.	N. A.		INER	T2217
RPV	CORE SHROUD	RCS	CRACKING	IGSCC	PEACH BOTTOM 3			93	INSPECTION	VISUAL	N. A.	278	BWRVIP	97-366
RPV	CORE SHROUD	RCS	CRACKING	IGSCC	OSCARSHAMN 1 JAPAN	12	31	96*	N. A.	N. A.	N. A.		INER	T2217
RPV	CORE SHROUD	RCS	CRACKING	IGSCC	CHINSHAN 2 TAIWAN	12	31	96*	N. A.	N. A.	N. A.		INER	T2217
RPV	CORE SPRAY PIPING	RCS	CRACKING	N. A.	VERMONT YANKEE			96	INSPECTION	VISUAL, ULTRAS.	REPAIR	271	BWRVIP	97-366
RPV	CORE SPRAY PIPING	RCS	CRACKING	N. A.	PILGRIM 1			80	INSPECTION	VISUAL, ULTRAS.	N. A.	293	BWRVIP	97-366
RPV	CORE SPRAY PIPING	RCS	CRACKING	N. A.	BROWNS FERRY 3			94	INSPECTION	VISUAL	REPAIR	296	BWRVIP	97-366
RPV	CORE SPRAY PIPING	RCS	CRACKING	N. A.	QUAD CITIES 1			80	INSPECTION	VISUAL	REPAIR	254	BWRVIP	97-366
RPV	CORE SPRAY PIPING	RCS	CRACKING	N. A.	PEACH BOTTOM 2			96	INSPECTION	VISUAL	N. A.	277	BWRVIP	97-366
RPV	CORE SPRAY PIPING	RCS	CRACKING	N. A.	FITZPATRICK			87	INSPECTION	VISUAL	REPAIR	333	BWRVIP	97-366
RPV	CORE SPRAY PIPING	RCS	CRACKING	N. A.	DRESDEN 3			80	INSPECTION	VISUAL	REPAIR	249	BWRVIP	97-366
RPV	CORE SPRAY PIPING	RCS	CRACKING	N. A.	DRESDEN 2			80	INSPECTION	VISUAL, ULTRAS.	N. A.	237	BWRVIP	97-366
RPV	CORE SPRAY PIPING	RCS	CRACKING	N. A.	BRUNSWICK 2			80	INSPECTION	VISUAL	REPAIR	324	BWRVIP	97-366
RPV	CORE SPRAY SPARGER	RCS	CRACKING	N. A.	PEACH BOTTOM 2			82	INSPECTION	VISUAL	REPAIR	277	BWRVIP	97-366
RPV	CORE SPRAY SPARGER	RCS	CRACKING	N. A.	OYSTER CREEK			78	INSPECTION	VISUAL	REPAIR	219	BWRVIP	97-366
RPV	CORE SPRAY SPARGER	RCS	CRACKING	N. A.	MONTICELLO			93	INSPECTION	VISUAL	REPAIR	263	BWRVIP	97-366
RPV	CORE SPRAY SPARGER	RCS	CRACKING	N. A.	BRUNSWICK 2			80	INSPECTION	VISUAL	REPAIR	324	BWRVIP	97-366

COMPONENT	SUBCOMPONENT	SYSTEM	AGING EFFECTS	AGING MECHANIS	PLANT	MO	DA	YR	HOW IDENTI	EVALN MET	REPAIR METH	DKT	REF DO	REF NO
RPV	CORE SPRAY SPARGER	RCS	CRACKING	N. A.	VERMONT YANKEE			80	INSPECTION	VISUAL	REPAIR	271	BWRVIP	97-366
RPV	CORE SPRAY SPARGER	RCS	CRACKING	N. A.	SUSQUEHANN A 2			96	INSPECTION	VISUAL	N. A.	388	BWRVIP	97-366
RPV	CORE SPRAY SPARGER	RCS	CRACKING	N. A.	PEACH BOTTOM 3			85	INSPECTION	VISUAL	REPAIR	278	BWRVIP	97-366
RPV	CRD	RCS	CRACKING	CORROSION	MONTICELLO	4	22	91	N. A.	TEST	N. A.	263	LER	910801
RPV	CRD GUIDE TUBE PINS	RCS	FAILURE	SCC	MIHAMA 3 JAPAN	7	23	82*	N. A.	ULTRAS. TEST	N. A.		IE IN	82-29
RPV	CRD GUIDE TUBE PINS	RCS	FAILURE	SCC	FESSENHAIM 1 FRANCE	7	23	82*	NOT AVAIL.	ULTRAS. TEST	N. A.		IE IN	82-29
RPV	CRD GUIDE TUBE PINS	RCS	FAILURE	SCC	NORTH ANNA 1		5	82	N. A.	ULTRAS. TEST	N. A.	338	IE IN	82-29
RPV	CRD PINS	RCS	FAILURE	N. A.	D.C. COOK 2	1	3	85	INSPECTION	VISUAL	REPLACEMENT	316	LER	850800
RPV	CRD ROD CONTR. ASSEMBLY	RCS	CRACKING	N. A.	HADDAM NECK	3	19	86	TEST	N. A.	N. A.	213	LER	861500
RPV	CRD STUB TUBE	RCS	CRACKING	N. A.	NINE MILE POINT 1			84	INSPECTION	VISUAL, ULTRAS.	REPAIR	220	BWRVI R	97-366
RPV	CRD TUBES	RCS	LOSS OF MATERIAL	N. A.	LACROSSE	5	6	86	N. A.	N. A.	N. A.	409	LER	861500
RPV	DRY TUBE	RCS	CRACKING	N. A.	BROWNS FERRY 2			94	INSPECTION	VISUAL	REPLACEMENT	260	BWRVIP	97-366
RPV	DRY TUBE	RCS	CRACKING	N. A.	DRESDEN 3	4		94	INSPECTION	VISUAL	REPLACEMENT	249	BWRVIP	97-366
RPV	FASTENERS	RCS	LOSS OF MATERIAL	CHEMICAL ATTACK	CALVERT CLIFFS 1	2	21	94	VISUAL	N. A.	REPAIR	317	LER	940401
RPV	HEAD	RCS	LOSS OF MATERIAL	CHEMICAL ATTACK	TURKEY POINT 4	3	13	87	INSPECTION	NON DEST. EXAMIN	N. A.	251	NRC IN	86-108
RPV	HEAD	RCS	CRACKING	N. A.	FITZPATRICK	4	1	90	INSERVICE INSPECT.	N. A.	N. A.	333	NRC IN	90-32
RPV	HEAD	RCS	LOSS OF MATERIAL	CHEMICAL ATTACK	SALEM 2	3	17	88*	LEAKING	NON DEST. EXAMIN	N. A.	311	GL	88-05
RPV	HEAD	RCS	CRACKING	TGSCC	QUAD CITIES 2	4	23	90	INSERVICE INSPECT.	METAL. TEST	N. A.	265	NRC IN	90-32,90-29
RPV	HEAD & BOLTS	RCS	LOSS OF MATERIAL	CHEMICAL ATTACK	TURKEY POINT 4	3	17	88*	LEAKING	NON DEST. EXAMIN	N. A.	251	GL	88-05
RPV	IN-CORE HOUSING	RCS	CRACKING	N. A.	NINE MILE POINT 1			97	INSPECTION	VISUAL	REPLACEMENT	220	BWRVIP	97-366
RPV	JET PUMP ASSEMBLY	RCS	CRACKING	N. A.	QUAD CITIES 1			94	INSPECTION	VISUAL	REPAIR	254	BWRVIP	97-366
RPV	JET PUMP ASSEMBLY	RCS	CRACKING	N. A.	MONTICELLO			89	INSPECTION	VISUAL	REPAIR	263	BWRVIP	97-366

COMPONENT	SUBCOMPONENT	SYSTEM	AGING EFFECTS	AGING MECHANIS	PLANT	MO	DA	YR	HOW IDENTI	EVALN MET	REPAIR METH	DKT	REF DO	REF NO
RPV	JET PUMP ASSEMBLY	RCS	CRACKING	N. A.	WNP-2			85	INSPECTION	VISUAL, ULTRAS.	REPAIR	397	BWRVIP	97-366
RPV	JET PUMP ASSEMBLY	RCS	CRACKING	N. A.	LASALLE 1	9		90	INSPECTION	N. A.	REPAIR	373	BWRVIP	97-366
RPV	JET PUMP ASSEMBLY	RCS	CRACKING	N. A.	BROWNS FERRY 3			94	INSPECTION	VISUAL	REPAIR	296	BWRVIP	97-366
RPV	JET PUMP ASSEMBLY	RCS	CRACKING	N. A.	LASALLE 2	9		90	INSPECTION	N. A.	REPAIR	374	BWRVIP	97-366
RPV	JET PUMP ASSEMBLY-BEAM	RCS	CRACKING	N. A.	PEACH BOTTOM 3			81	INSPECTION	VISUAL, ULTRAS.	REPLACEMENT	277	BWRVIP	97-366
RPV	JET PUMP ASSEMBLY-BEAM	RCS	CRACKING	N. A.	DRESDEN 2	8		95	INSPECTION	ULTRAS. TEST	REPLACEMENT	237	BWRVIP	97-366
RPV	JET PUMP ASSEMBLY-BEAM	RCS	CRACKING	N. A.	GRAND GULF 1			96	INSPECTION	VISUAL, ULTRAS.	REPLACEMENT	416	BWRVIP	97-366
RPV	JET PUMP ASSEMBLY-BEAMS	RCS	CRACKING	N. A.	FITZPATRICK			92	INSPECTION	VISUAL, ULTRAS.	REPLACEMENT	333	BWRVIP	97-366
RPV	JET PUMP ASSEMBLY-BEAMS	RCS	CRACKING	N. A.	SUSQUEHANN A 2			93	INSPECTION	VISUAL	REPLACEMENT	388	BWRVIP	97-366
RPV	JET PUMP ASSEMBLY-	RCS	CRACKING	N. A.	DRESDEN 3	4		94	INSPECTION	VISUAL	REPAIR	249	BWRVIP	97-366
RPV	JET PUMP ASSEMBLY-SCREW	RCS	N. A.	N. A.	DUANE ARNOLD			96	INSPECTION	VISUAL	REPAIR	331	BWRVIP	97-366
RPV	JET PUMP ASSEMBLY-SCREW	RCS	CRACKING	N. A.	HOPE CREEK			94	INSPECTION	VISUAL	N. A.	354	BWRVIP	97-366
RPV	SHROUD SUPPORT	RCS	CRACKING	IGSCC	PEACH BOTTOM 3	1	21	88	INSPECTION	ULTRAS. TEST	REPAIR	278	NRC IN	92-57
RPV	SHROUD SUPPORT	RCS	CRACKING	IGSCC	DRESDEN 2			95	INSPECTION	VISUAL	REPAIR	237	BWRVIP	97-366
RPV	SHROUD SUPPORT	RCS	CRACKING	N. A.	QUAD CITIES 1			92	INSPECTION	VISUAL	REPAIR	254	BWRVIP	97-366
RPV	SHROUD SUPPORT	RCS	CRACKING	N. A.	QUAD CITIES 2			91	INSPECTION	VISUAL	REPAIR	265	BWRVIP	97-366
RPV	STEAM DRYER-SCREWS	RCS	CRACKING	N. A.	PILGRIM 1			97*	INSPECTION	VISUAL	N. A.	293	BWRVIP	97-366
RPV	STUDS	RCS	CRACKING	SCC	DRESDEN 2	1	16	91	N. A.	ULTRAS. TEST	STUDS REPLACEMENT	237	LER	910201
RPV	THERMAL SHIELD - BOLTS	RCS	FAILURE	VIBRATION	SAN ONOFRE 1	1	8	89	INSPECTION	VISUAL	N. A.	206	LER	890101
RPV	THIMBLE TUBE	RCS	WALL THINNING	EROSION	BEAVER VALLEY 1	3	28	88*	INSPECTION	VISUAL	N. A.	334	NRC IN	87-44-1
RPV	THIMBLE TUBE	RCS	WALL THINNING	EROSION	MILLSTONE 3	3	28	88*	INSPECTION	VISUAL	N. A.	423	NRC IN	87-44-1
RPV	THIMBLE TUBE	RCS	WALL THINNING	EROSION	NORTH ANNA 1	9	16	87*	INSPECTION	ED. CUR. TEST	N. A.	338	NRC IN	87-44

COMPONENT	SUBCOMPONENT	SYSTEM	AGING EFFECTS	AGING MECHANIS	PLANT	MO	DA	YR	HOW IDENTI	EVALN MET	REPAIR METH	DKT	REF DO	REF NO
RPV	TOP GUIDE	RCS	CRACKING	N. A.	OYSTER CREEK			91	INSPECTION	VISUAL	N. A.	219	BWRVIP	97-366
RPV	TOP GUIDE & CORE PLATE	RCS	CRACKING	CORROSION	WUERGASSEN GERMANY	3	17	88	INSERVICE INSPECT.	NON DEST. EXAMIN	N. A.		NRC IN	95-17
SEAL	CONTAINMENT LINER/FLOOR SEAL	CONTAINMENT	DETERIORATION	N. A.	TROJAN			95*	N. A.	N. A.	N. A.	344	NUREG	1522
SEAL	FLOOD/PROTECTION SEALS	FLOOD PROTECTION	DETERIORATION	N. A.	BEAVER VALLEY 2	1	11	97	ENGINEER. EVALUAT.	N. A.	REPAIR	412	LER	970200
STR. STEEL /CONCRETE	FLOOR - ICE CONDENSER	ICE CONDENSER	EXCESSIVE DEFORMN.	MOISTURE ICE	SEQUOYAH 1	3	16	92	VISUAL	N. A.	REPAIR	327	LER	911100
STRUCTURAL STEEL	CHANNEL WELD	N. A.	FAILURE	N. A.	ZION 1	9	10	83	TEST	N. A.	NONE	295	LER	833600
STRUCTURAL STEEL	COVER	FIRE PROTECT.	LOSS OF MATRL.- HOLE	N. A.	MCGUIRE 2	2	4	85	VISUAL	VISUAL	REPAIR	370	LER	850400
STRUCTURAL STEEL	DOOR	FIRE PROTECT.	LOSS OF MATERIAL	MECHANICAL WEAR	DAVIS-BESSE 1	6	25	86	PREVENT. MAINTEN.	VISUAL	N. A.	346	LER	862701
STRUCTURAL STEEL	DOOR	RCIC	HOLE	N. A.	QUAD CITIES 2	5	24	92	N. A.	VISUAL	REPAIR	265	LER	921700
STRUCTURAL STEEL	DOOR	FIRE PROTECT.	CRACKING	N. A.	FARLEY 1	4	4	89	VISUAL	VISUAL	N. A.	348	LER	890100
STRUCTURAL STEEL	DOOR	FIRE PROTECT.	CRACKING	N. A.	FARLEY 1	5	5	89	N. A.	VISUAL	REPAIR	348	LER	890200
STRUCTURAL STEEL	DOOR LATCH	CONTROL ROOM	FAILURE	N. A.	COMANCHE PEAK 1	5	10	94	N. A.	N. A.	REPAIR	445	LER	940200
STRUCTURAL STEEL	DOOR LATCH	CONTROL BUILDING	FAILURE	MECHANICAL WEAR	DIABLO CANYON 1	6	5	87	N. A.	VISUAL	REPLACEMENT	275	LER	870901
STRUCTURAL STEEL	DOOR LATCH	TURBINE GEN. BLDG.	FAILURE	MECHANICAL WEAR	TROJAN	8	31	88	N. A.	VISUAL	REPLACEMENT	344	LER	883700
STRUCTURAL STEEL	DOOR LATCH	CCSW	LOSS OF MATERIAL	MECHANICAL WEAR	DRESDEN 3	3	19	92	LEAK TEST	N. A.	REPLACEMENT	249	LER	920800
STRUCTURAL STEEL	DOOR LATCH	CONTROL BUILDING	FAILURE	MECHANICAL WEAR	SUSQUEHANNA 1	11	16	87	N. A.	VISUAL	REPAIR	387	LER	873301
STRUCTURAL STEEL	DOOR SEAL	CCSW	DETERIORATION	MECHANICAL WEAR	DRESDEN 3	5	16	94	LEAK TEST	N. A.	TIGHTENING	249	LER	941500
STRUCTURAL STEEL	ELEVATED RELEASE TOWER	N. A.	LOSS OF MATERIAL	MOISTURE	COOPER			95*	VISUAL	N. A.	N. A.	298	NUREG	1522
STRUCTURAL STEEL	EQUIPMENT SUPPORT	SEVERAL LOCATIONS	LOSS OF MATERIAL	N. A.	BEAVER VALLEY 1			95*	VISUAL	N. A.	N. A.	334	NUREG	1522
STRUCTURAL STEEL	FRAME/SUPPORTS/ BASEPLATE	N. A.	LOSS OF MATERIAL	N. A.	TROJAN			95*	VISUAL	N. A.	N. A.	344	NUREG	1522
STRUCTURAL STEEL	LINER	REACTOR CAVITY	CRACKING	TGSCC	INDIAN POINT 2	2	7	93	VISUAL	N. A.	REPAIR	247	LER	930102
STRUCTURAL STEEL	LINER	SPENT FUEL POOL	N.A.-LEAKING	N. A.	RANCHO SECO	11	16	86	N. A.	N. A.	N. A.	312	LER	862501

COMPONENT	SUBCOMPONENT	SYSTEM	AGING EFFECTS	AGING MECHANIS	PLANT	MO	DA	YR	HOW IDENTI	EVALN MET	REPAIR METH	DKT	REF DO	REF NO
STRUCTURAL STEEL	LINER	SPENT FUEL POOL	N.A.-LEAKING	N. A.	TROJAN			95*	VISUAL	N. A.	N. A.	344	NUREG	1522
STRUCTURAL STEEL	ROOF	TURBINE GEN. BLDG.	DETERIORATION	MOISTURE RAIN	SOUTH TEXAS 2	2	24	92	TURBINE TRIP	N. A.	REPAIR	499	LER	920300
STRUCTURAL STEEL	SLUICE GATE	ERCSW	FOULING	ORGANISMS	ARKANSAS 1	9	22	96	TEST	VISUAL	CLEANING	313	LER	960800
STRUCTURAL STEEL	SPENT FUEL RACK BORAFLEX	SPENT FUEL POOL	DETERIORATION	IRRADIATION EMBRITL.	SOUTH TEXAS 1	1	24	95	TEST	N. A.	N. A.	498	LER	950200
STRUCTURAL STEEL	SPENT FUEL RACK BORAFLEX	SPENT FUEL POOL	DETERIORATION	IMPROPER DESIGN	PALISADES	8	17	93	INSPECTION	VISUAL	N. A.	255	LER	930702
STRUCTURAL STEEL	SPENT FUEL RACK BORAFLEX	SPENT FUEL POOL	DETERIORATION	IRRADIATION EMBRITL.	PRAIRIE ISLAND 1	11	27	96	N. A.	ANALYS.	N. A.	282	LER	961900
STRUCTURAL STEEL	STEEL MEMBERS	N. A.	LOSS OF MATERIAL	MOISTURE	ROBINSON 2			95*	VISUAL	N. A.	N. A.	261	NUREG	1522
TANK	BOOSTER PUMP GLAND WATER	SERVICE WATER	LOSS OF MATERIAL	MOISTURE	COOPER			95*	VISUAL	N. A.	N. A.	298	NUREG	1522
TANK	COATING	EGF	DETERIORATION	CHEMICAL ATTACK	ARKANSAS 2	6	27	86	INSPECTION	N. A.	CLEANING	368	LER	861401
TANK	CVC HOLD UP	N. A.	EXCESSIVE DEFORMN.	N. A.	ROBINSON 2			95*	VISUAL	N. A.	N. A.	261	NUREG	1522
TANK	FUEL OIL TANK (COATING)	EGF	DETERIORATION	N. A.	PERRY 1	8	8	91	VISUAL	N. A.	N. A.	440	LER	900501
TANK	FUEL OIL TANK (COATING)	EGF	DETERIORATION	N. A.	MAINE YANKEE	8	8	91	VISUAL	TEST	FILTERS REPLACEMENT	309	LER	910800
TANK	PRIMARY WATER STORAGE	PRIMARY WATER	LOSS OF MATERIAL	N. A.	ROBINSON 2			95*	VISUAL	N. A.	N. A.	261	NUREG	1522
TANK	RADWASTE TANK	AWCT	RUPTURE	CORROSION	MILLSTONE 1	11		77*	FAILURE	NON DEST. EXAMIN	N. A.	245	IE IN	79-07
TANK	REFUELING WATER STORAGE	REFUELING WATER	LOSS OF MATERIAL	N. A.	ROBINSON 2			95*	VISUAL	N. A.	N. A.	261	NUREG	1522
TANK	SHELL & ANCHORAGE	CONDENSATE STORAGE	LOSS OF MATERIAL	N. A.	TURKEY POINT 3			95*	N. A.	N. A.	N. A.	250	NUREG	1522
TANK	SHELL - BOTTOM	REFUELING WATER STOR.	N. A.	N. A.	HADDAM NECK	6	0	91	LEAKING	N. A.	WELDING/PLATES	213	NUREG	1522
TANK	STEAM GEN. DRAIN TANK	N. A.	CRACKING DETERIORATION	N. A.	BEAVER VALLEY 1			95*	VISUAL	N. A.	N. A.	334	NUREG	1522
VESSEL	PRESSURIZER	RCS	LOSS OF MATERIAL	CHEMICAL ATTACK	ARKANSAS 2	4	24	87	LEAKING	VISUAL	REPAIR	368	LER	870301
VESSEL	PRESSURIZER	RCS	CRACKING	PWSCC	SAN ONOFRE 3	2	27	89	LEAKING	METAL. TEST	N. A.	362	NRC IN	90-10
VESSEL	PRESSURIZER	RCS	CRACKING	PWSCC	PALISADES	10	9	93	LEAKING	NON DEST. EXAMIN.	N. A.	255	LER	931100
VESSEL	PRESSURIZER	RCS	CRACKING	TGSCC, IGSCC	SURRY 1	9	12	95	INSPECTION	METAL. TEST	N. A.	280	LER	950701

COMPONENT	SUBCOMPONENT	SYSTEM	AGING EFFECTS	AGING MECHANIS	PLANT	MO	DA	YR	HOW IDENTI	EVALN MET	REPAIR METH	DKT	REF DO	REF NO
VESSEL	PRESSURIZER	RCS	CRACKING	PWSCC	ARKANSAS 1	12	22	90	LEAKING	ENGIN. JUGM.	N. A.	313	LER	902101
VESSEL	PRESSURIZER	RCS	CRACKING	PWSCC	CALVERT CLIFFS 1	3	21	94	LEAKING	ENGIN. JUGM.	N. A.	317	LER	940300
VESSEL	PRESSURIZER	RCS	CRACKING	PWSCC	SAN ONOFRE 2	2	18	92	INSPECTION	ENGIN. JUGM.	N. A.	361	LER	920401
VESSEL	PRESSURIZER	RCS	CRACKING	PWSCC	ST. LUCIE 2	3	2	93	LEAKING	ED. CUR. TEST	N. A.	389	LER	930400
VESSEL	PRESSURIZER	RCS	CRACKING	PWSCC	PALO VERDE 1	1	2	92	LEAKING	ENGIN. JUGM.	N. A.	528	LER	920100
VESSEL	PRESSURIZER	RCS	CRACKING	PWSCC	CALVERT CLIFFS 2			89	LEAKING	N. A.	N. A.	318	NRC IN	90-10
WATER-CONROL STR.	DAM - CONCRETE	SERVICE WATER	SPALLING	N. A.	ROBINSON 2			95*	VISUAL	N. A.	N. A.	261	NUREG	1522
WATER-CONTROL STR.	DAM - STEEL GATES	SERVICE WATER	LOSS OF MATERIAL	MOISTURE	ROBINSON 2			95*	VISUAL	N. A.	N. A.	261	NUREG	1522

APPENDIX B

DEGRADATION REFERENCE DATABASE

TYPE	ID	TITLE	MO	YR	AUTH/ORG	DESCRIPTION	TYPES OF COMPONENTS	POTENTIAL AGING ISSUES
CODE	ASME Code, Section V	Nondestructive Examination		1992	ASME	Contains requirements and methods for nondestructive examination which are Code requirements when and to the extent they are specifically referenced and required by other Code Sections.	All pressure retaining components specifically referenced by other Code Sections.	All types of degradation for the covered components.
CODE	ASME Code, Section XI	Rules for Inservice Inspection of Nuclear Power Plant Components		1992	ASME	Specify rules for examination, testing, and inspection of components and systems in a nuclear power plant.	Class 1, 2, & 3 Components; Class MC and Metallic Liners of Class CC (Concrete Containment) Components; Component Supports; and Class CC Concrete Components.	All types of degradation for the covered components.
INDUSTRY STANDARD/ GUIDELINE		Structural Materials Handbook, Vol. 1-3			ORNL	The Structural Materials Handbook and the complimentary Structural Material Electronics Data Base have been developed at ORNL as a part of the NRC structural aging (SAG) Program. Handbook contains concrete and other material properties data that have application to the resolution of issues that might arise during nuclear power plant continued service reviews.	None identified	None identified
INDUSTRY STANDARD/ GUIDELINE	ACI 201.1R-68 (Revised 1984)	Guide for Making a Condition Survey of Concrete in Service		1984	ACI	This guide provides a system for reporting on the condition of concrete in service. It includes a check list of the many details to be considered in making a report, and provides standard definitions of 40 terms associated with the durability of concrete.	Concrete	None identified
INDUSTRY STANDARD/ GUIDELINE	ACI 201.2R-77	Guide for Durable Concrete		1977	ACI	Discusses important causes of concrete degradation and gives recommendations on how to prevent such damage. Topics covered include freezing and thawing, aggressive chemical exposure, abrasion, corrosion of steel and other materials embedded in concrete, chemical reaction of aggregates, repair of concrete, and the use of coatings.	Concrete structures	Concrete degradation

TYPE	ID	TITLE	MO	YR	AUTH/ORG	DESCRIPTION	TYPES OF COMPONENTS	POTENTIAL AGING ISSUES
INDUSTRY STANDARD/ GUIDELINE	ACI 207.3R-79, Revised 1985	Practices for Evaluation of Concrete in Existing Massive Structures for Service Conditions		1985	ACI	Current methods available for evaluating physical properties of concrete in existing structures to determine its capability of performing satisfactorily are identified and discussed.	Concrete in existing massive structures such as buildings and reactor foundations, hydraulic structures, and dams.	All types of degradation for the covered components.
INDUSTRY STANDARD/ GUIDELINE	ACI 222R-89	Corrosion of Metals in Concrete		1989	ACI	Discusses the factors that cause and control corrosion of steel in concrete, techniques for detecting corrosion in structures in service, and remedial procedures.	Concrete structures	Corrosion of metals in concrete
INDUSTRY STANDARD/ GUIDELINE	ACI 224.1R-93	Causes, Evaluation, and Repair of Cracks in Concrete Structures		1993	ACI	Summarizes the causes of cracks in concrete and the means for their control. The report also describes evaluation procedures and methods for crack repair such as epoxy injection, routing (enlarging the cracks and sealing), stitching U-shaped metal units, additional reinforcement, and grouting.	Concrete structures	Concrete structure degradation
INDUSTRY STANDARD/ GUIDELINE	ACI 224R-90	Control of Cracking in Concrete Structures		1990	ACI	This document presents the principal causes of cracking and a discussion of crack control procedures. The control of cracking due to drying shrinkage and crack control for flexural members, layered systems and mass concrete are covered in detail.	Concrete	Cracking in concrete structures
INDUSTRY STANDARD/ GUIDELINE	ACI 311.4R-88	Guide for Concrete Inspection		1988	ACI	The guide discusses the need for inspection of concrete construction, the types of inspection activities involved and the responsibilities of individuals and organizations involved in those activities. Recommended minimum levels of inspections are given.	Concrete structures	None identified

TYPE	ID	TITLE	MO	YR	AUTH/ORG	DESCRIPTION	TYPES OF COMPONENTS	POTENTIAL AGING ISSUES
INDUSTRY STANDARD/ GUIDELINE	ACI 349.3R-96	Evaluation of Existing Nuclear Safety-Related Concrete Structures	3	1996	ACI	The purpose of this report is to provide the plant owner and engineering staff with an appropriate procedure and background for examining the performance of facility structures and taking appropriate actions based on observed conditions.	Concrete structures	Assessment of concrete, reinforcing steel, structural steel, and prestressing steel performance
INDUSTRY STANDARD/ GUIDELINE	ASCE 11-90	Guideline for Structural Condition Assessment of Existing Buildings		1991	ASCE	The intent of this standard is to provide guidelines and methodology for assessing the structural conditions of existing buildings constructed from combinations of material including concrete, metals, masonry, and wood. The standard establishes an assessment procedure including the investigation, testing methods, and the format of the report of the condition assessment.	Concrete, metal, masonry, and wood structures	Assessment of concrete, metal, masonry, and wood structures
INDUSTRY STANDARD/ GUIDELINE	GIP-2	Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment, Rev. 2	2	1992	SQUG	The GIP provides the detailed technical approach, generic procedures and documentation guidance which can be used by USI A-46 licensees to verify the seismic adequacy of mechanical and electrical safe shutdown equipment.	Mechanical and electrical equipment needed to bring the plant to a safe shutdown condition during and following an SSE.	All types of degradation for the covered components
INDUSTRY STANDARD/ GUIDELINE	NEI 95-10 Rev. 0	Industry Guideline For Implementing The Requirements of 10 CFR Part 54 - The License Renewal Rule	3	1996	NEI	This guideline provides an approach for implementing the requirements of 10 CFR Part 54, the license renewal rule.	Structures, Systems and Components (SSCs) covered by the License Renewal Rule	Assessment of SSCs
INDUSTRY STANDARD/ GUIDELINE	NEI 96-03, Rev. D	Guideline for Monitoring the Condition of Structures at Nuclear Power Plants	7	1996	NEI	The document provides guidance for monitoring structures at nuclear power plants. The guidelines are intended to meet the regulatory requirements of the maintenance rule and the license renewal rule when used in conjunction with existing programs and the guidance documents, as appropriate, for either the maintenance rule (NUMARC 93-01) or the license renewal rule (NEI 95-10).	1) Structures that are relied upon to remain functional during and following design basis events, 2) that are used in EOPs, 3) whose failure could prevent safety related SSCs from fulfilling their intended function, 4)whose failure could cause a scram or actuation of a safety-related system, and 5) that are relied on for compliance with the Commission's regulation for fire protection, environmental qualification…	Degradation of concrete & steel structures and components, coatings, painted surfaces, expansion joints, seals, glazing, flashing, earthen structures/dams, etc.

TYPE	ID	TITLE	MO	YR	AUTH/ORG	DESCRIPTION	TYPES OF COMPONENTS	POTENTIAL AGING ISSUES
INDUSTRY STANDARD/ GUIDELINE	NUMARC 90-01, Rev. 1	Pressurized Water Reactor, Containment Structures License Renewal Industry Report	9	1991	NUMARC	This guideline provides the technical basis for license renewal of PWR containment structures. The scope of the report includes steel-lined reinforced concrete (including prestressed) and free-standing steel PWR containment structures. Containment internal structures are excluded from the scope of this document.	PWR containment structures	Assessment of concrete, reinforcing steel, prestressing system, liner, and free-standing steel containment
INDUSTRY STANDARD/ GUIDELINE	NUMARC 90-02, Rev. 1	Boiling Water Reactor, Reactor Pressure Vessel License Renewal Industry Report	8	1992	NUMARC	This guideline provides the technical basis for license renewal of BWR reactor pressure vessels. The age related degradation mechanisms were identified from a review/evaluation of nuclear power plant operating experience, relevant laboratory data, and related experience in other industries.	BWR reactor pressure vessel	Assessment of attachment welds, closure studs, nozzles and safe ends, penetrations, vessel shell and flanges, top and bottom heads, and the vessel support skirt
INDUSTRY STANDARD/ GUIDELINE	NUMARC 90-03	Boiling Water Reactor, Vessel Internals License Renewal Industry Report	6	1992	NUMARC	This guideline provides the technical basis for license renewal of BWR reactor pressure vessel internals. The age related degradation mechanisms were identified from a review/evaluation of nuclear power plants operating experience, relevant laboratory data, and related experience in other industries.	BWR reactor pressure vessel internals	Assessment of control blades, core shroud, core plate, core spray sparger, CRDM housing, top guide, dry tubes, access hole cover and other components
INDUSTRY STANDARD/ GUIDELINE	NUMARC 90-04	Pressure Water Reactor, Vessel License Renewal Industry Report	9	1992	NUMARC	This guideline provides the technical basis for license renewal of PWR reactor pressure vessels. The age related degradation mechanisms were identified from a review/evaluation of nuclear power plant operating experience, relevant laboratory data, and related experience in other industries.	PWR reactor pressure vessel	Assessment of closure head dome, closure stud assemblies, vessel flange, CRDM, shell, CRDM housing, shroud support ring, and other components
INDUSTRY STANDARD/ GUIDELINE	NUMARC 90-05, Rev. 1	Pressurized Water Reactor, Reactor Pressure Vessel Internals License Renewal Industry Report	12	1992	NUMARC	This guideline provides the technical basis for license renewal of PWR reactor pressure vessel internals. The age related degradation mechanisms were identified from a review of vendor-specific evaluations, nuclear power plants operating experience, relevant laboratory data, and related experience in other industries.	PWR reactor pressure vessel internals	Assessment of upper support plate, guide tube assembly, upper core plate, core shroud, core barrel flange, core support barrel, and other components

TYPE	ID	TITLE	MO	YR	AUTH/ORG	DESCRIPTION	TYPES OF COMPONENTS	POTENTIAL AGING ISSUES
INDUSTRY STANDARD/ GUIDELINE	NUMARC 90-06 Rev. 1	Class I Structures License Renewal Industry Report	12	1991	NUMARC	This guideline provides the technical basis for license renewal for U.S. nuclear power plant Class I structures. The age related degradation mechanisms were identified from a review/evaluation of nuclear power plants operating experience, relevant laboratory data, and related experience in other industries.	U.S. nuclear power plant Class I structures	Assessment of concrete, reinforcing steel, piles, structural steel, stainless steel liner plate, and miscellaneous components.
INDUSTRY STANDARD/ GUIDELINE	NUMARC 90-07	PWR Reactor Coolant System License Renewal Industry Report	10	1990	NUMARC	This guideline identifies specific license renewal requirements for pressurized water reactor coolant system components. The potential for significant aging degradation is determined by examining the components current design basis, its performance history, and the extend to which it is covered by existing maintenance and refurbishment programs.	PWR reactor coolant system components	Assessment of pressurizers, and component integral supports
INDUSTRY STANDARD/ GUIDELINE	NUMARC 90-09	BWR Primary Coolant Pressure Boundary License Renewal Industry Report	4	1992	NUMARC	This guideline provides the technical basis for license renewal for U.S. boiling water reactor primary coolant pressure boundaries. The age related degradation mechanisms were identified from a review/evaluation of nuclear power plants operating experience, relevant laboratory data, and related experience in other industries.	BWR primary coolant pressure boundaries	Assessment of heat exchangers, and supports
INDUSTRY STANDARD/ GUIDELINE	NUMARC 90-10, Rev. 1	BWR Containments License Renewal Industry Report	12	1991	NUMARC	This guideline provides the technical basis for license renewal of BWR containments. The age related degradation mechanisms were identified from a review/evaluation of nuclear power plants operating experience, relevant laboratory data, and related experience in other industries.	BWR	Assessment of concrete, reinforcing steel, steel liner, prestressing, containment shell, anchors, and other component
INDUSTRY STANDARD/ GUIDELINE	NUMARC 93-01	Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	5	1993	NUMARC	This guideline describes an acceptable approach to meet the NRC Maintenance Rule. The guideline includes: selecting the SSC's within the scope of the Rule and establishing and applying risk significant criteria and performance criteria.	Safety-related (SR): SSCs relied upon to remain functional during and after DBE's. Non-safety related: SSC's that are relied upon to mitigate accidents or transients or are used in plant EOPs, whose failure could prevent SR SSCs from fulfilling their SR function, or whose failure could cause a reactor scram or actuation of a SR system.	None identified

TYPE	ID	TITLE	MO	YR	AUTH/ORG	DESCRIPTION	TYPES OF COMPONENTS	POTENTIAL AGING ISSUES
NUREG	NUREG-1144 Rev. 2	Nuclear Plant Aging Research (NPAR) Program Plan, Status and Accomplishments	6	1991	NRC	This report identifies aging mechanisms and effects that could cause degradation. It includes methods of inspection, surveillance, condition monitoring, and maintenance as means of managing and mitigating aging effects that may affect safe plant operation.	SSCs	Degradation of RPVs, containments, reactor internals, piping, and steam generator tubes
NUREG	NUREG-1339	Resolution of Generic Safety Issue: Bolting Degradation or Failure in Nuclear Power Plants	7	1990	Johnson, R.E./ NRC	This report presents NRC's review and evaluation of the document and the conclusion that this document, together with other information from industry and NRC, provides the bases for resolving bolting degradation issues.	Bolting/fasteners	Degradation and failure of pressure boundary bolting, supports and embedded bolting
NUREG	NUREG-1377 Rev. 4	NRC Research Program on Plant Aging: Listing and Summaries of Reports Issued Through September 1993	12	1993	Vora, J. P. /NRC	This document contains a listing of reports generated in the Nuclear Plant Aging Research Program. Each summary describes the elements of the research covered in the report and outlines the significant results.	SSCs	SSCs
NUREG	NUREG-1522	Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures	6	1995	Ashar, H., Bagchi, G./ NRC	This report presents information on the condition of structures and civil engineering features at operating NPPs in the U.S. Much of the data comes from walkdowns conducted at six old plants.	Structures including containments, buildings, buried piping/tunnels, dams/cooling canals, tanks, anchorages, supports, etc.	Degradation occurrences of containment structures, intake structures, prestressed concrete structures, masonry walls, buried piping, storage tanks, anchorages, supports, and inaccessible structures
NUREG	NUREG-1526	Lessons Learned from Early Implementation of the Maintenance Rule at Nine Nuclear Power Plants	6	1995	Petrone, C. D. /NRC	This report summarizes the lessons learned from the nine pilot site visits that were performed to review early implementation of the maintenance rule using the draft NRC Maintenance Inspection Procedure. Licensees followed NUMARC 93-01.	SSCs covered by the Maintenance Rule	NRC recognizes, that in certain cases the performance or condition of SSCs could be effectively controlled by doing adequate preventive maintenance rather than by monitoring against goals.

TYPE	ID	TITLE	MO	YR	AUTH/ORG	DESCRIPTION	TYPES OF COMPONENTS	POTENTIAL AGING ISSUES
NUREG	NUREG-1540	BWR Steel Containment Corrosion	4	1996	Tan, C. P. Bagchi, G/ NRC	This report describes regulatory actions taken after corrosion was discovered in the drywell at the Oyster Creek Plant and in the torus at the Nine Mile Pt. 1 Plant. The report also describes the causes of corrosion, requirements for monitoring, and measures taken.	Steel containments	Corrosion in inaccessible areas of containments
NUREG	NUREG-1557	Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal	10	1996	Regan, C. et al.	This report provides a brief summary of the technical information and NUMARC/NRC agreements from the 10 Industry Reports (IRs) except for the cable license renewal IR.	PWR & BWR vessels, PWR & BWR containments, PWR RCS, BWR primary coolant pressure boundary, PWR & BWR vessel internals, Class I structures	Assessment of RPV (shell, nozzles, CRD housing, etc.), containment (concrete & steel shell and components), pressurizer, integral supports, RPV internals, and structures (concrete & steel)
NUREG	NUREG-1568	License Renewal Demonstration Program: NRC Observations and Lessons Learned	12	1996	Prato, R. J., Kuo, P. T., Newberry,S.F NRC	The observation and lessons learned discussed in this report will be used to identify additional guidance and/or clarifications that need to be added to NEI 95-10, Revision 0, for an acceptable implementation of LR requirements under 10 CFR Part 54.	SSCs	None identified
NUREG	NUREG-1611	Aging Management of Nuclear Power Plant Containments for License Renewal, Draft	9	1997	Liu, W. C, Kuo, P. T, Lee, S. S./ NRC	The purpose of this report is to reconcile the technical information and agreements resulting from the NUMARC IR reviews and the inservice inspection requirements of Subsection IWE and IWL as promulgated in 50.55a for license renewal consideration.	Containments	Assessment of concrete, structural steel, liner, reinforcing steel, and prestressing systems
NUREG	NUREG/CP-0036	Proceedings of the Workshop on Nuclear Plant Aging, August 4-5, 1982, Bethesda, Maryland	11	1982	Bader, B. E., Hanchey,L.A NRC/ SANDIA	The objective of the workshop was to facilitate an exchange of thoughts between the NRC and industry on time-related degradation and its influence on reactor safety.	SSCs	Aging degradation of steam generator tubing and components, piping, and insulation

TYPE	ID	TITLE	MO	YR	AUTH/ORG	DESCRIPTION	TYPES OF COMPONENTS	POTENTIAL AGING ISSUES
NUREG	NUREG/CP-0100	Proceedings of the International Nuclear Power Plant Aging Symposium	3	1989	NRC	This report contains 48 papers on various topics relating to aging including: aging research programs, aging of SSCs, reliability, and role of maintenance in aging management.	SSCs	Aging degradation of RPVs, containment, piping systems, steam generator tubes, concrete structures, electrical cables, and supports
NUREG	NUREG/CP-0105, Vols. 1,2 & 3	Proceedings of the U.S. Nuclear Regulatory Commission Seventeenth Water Reactor Safety Information Meeting	3	1990	Weiss, A. J./ NRC/BNL	This report contains 84 papers out of the 111 that were presented at the meeting. Vol. 3 contains papers on Plant Aging. The generic content and structure of programs for addressing degradation due to aging are discussed in this NUREG.	Systems and components	Aging degradation of piping systems, heat exchangers, RPVs, electrical cables, and steam generator tubing
NUREG	NUREG/CP-0114, Vol. 1	Probabilistic Methods for Condition Assessment and Life Prediction of Concrete Structures in Nuclear Power Plants	10	1990	Ellingwood, B, Mory, Y.	The goal of this research is to develop a methodology to facilitate quantitative assessments of current and future structural reliability and the performance of concrete structures in NPPs. This methodology takes in to account the stochastic nature of past and future loads due to operating conditions, the environment, randomness in strength and degradation processes.	Concrete Structures	Assessment and life prediction of concrete structures
NUREG	NUREG/CP-0120	Aging of Concrete Containment Structures in Nuclear Power Plants (Proceedings of the U.S. NRC Fifth Workshop on Containment Integrity)	5	92	Naus D. J. et al.	This paper discusses current inservice inspection requirements for concrete containments. Pertinent concrete structures are described in term of their importance, design considerations, and materials of construction. Degradation factors which can potentially impact the ability of these structures to meet their functional and performance requirements are identified.	Category I concrete structures.	Degradation of concrete, steel reinforcement, prestressed concrete structures, and liner/structural steel
NUREG	NUREG/CP-0122, Vol. 1 & 2	Proceedings of the Aging Research Information Conference	9	1992	Beranek, A./ NRC/BNL	This report includes papers regarding the results of research in the area of nuclear plant aging from programs sponsored by the Office of Nuclear Regulatory Research, NRC.	SSCs	Aging degradation of concrete structures, heat exchangers, piping systems, filters, electrical cables, RPVs, CRDs, supports, and pressurizers

TYPE	ID	TITLE	MO	YR	AUTH/ORG	DESCRIPTION	TYPES OF COMPONENTS	POTENTIAL AGING ISSUES
NUREG	NUREG/CP-0133 Vol. 3	Aging Management of Light Water Reactor Concrete Containment (Proceedings of U.S. NRC Twenty-First Water Reactor Safety Information Meeting)	10	93	Shah V. N., Hookham C. J.	This paper evaluates aging of light water reactor concrete containments and identifies three degradation mechanisms that have the potential to cause widespread aging damage after years of satisfactory experience: alkali-silica reaction, corrosion of reinforced concrete, and sulfate attack. Techniques to detect and mitigate these long- term aging effects are discussed.	Concrete structures	Aging degradation of concrete containments, foundations, reinforcing steel, and prestressing steel
NUREG	NUREG/CP-0157	Aging of the Containment Pressure Boundary in Light-Water Reactor Plants. (Twenty Fourth Water Reactor Safety Information Meeting)	1	1997	Naus D. J. et al.	The objectives of this work are to (1) understand the significant factors relating occurrence of corrosion, and the structural capacity reduction of steel containment and liners of concrete containments; (2) provide information about the structural capacity margins for steel containments, and concrete containments; (3) provide recommendations by assessing the seriousness of reported incidences of containment degradations.	Containment pressure boundary	Aging degradation of containment pressure boundary components
NUREG	NUREG/CR-2641	The In-Plant Reliability Data Base for Nuclear Power Plant Components: Data Collection and Methodology Report	7	1982	Drago, J. P., Borkowski, R. J., Pike, D. H. / ORNL	The development of a components reliability data base for use in Nuclear Power Plant probabilistic risk assessments and reliability studies is presented in this report. The source of the data are the in-plant maintenance work request records from a sample of Nuclear Power Plants.	Systems and components	Aging degradation of steam generators, heat exchangers, and snubbers
NUREG	NUREG/CR-3543	Survey of Operating Experiences from LERs to Identify Aging Trends	1	1984	Murphy, G. A./ ORNL	This report includes data on the systems, components, subparts, age-related failure mechanisms, severity, and the method of detection of failures.	Systems and components	Age related degradation of heat exchangers, and steam generators
NUREG	NUREG/CR-3818	Report of Results of Nuclear Power Plant Aging Workshop	8	1984	Clark, N. H., Berry, D. L./ SANDIA	Report of the results of the Nuclear Power Plant Aging Workshop	SSCs	Aging degradation of steam generators, concrete/anchors, tendons, and heat exchangers

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NUREG	NUREG/CR-3819	Survey of Aged Power Plant Facilities	6	1985	Rose, J.A. et al. / INEL	Age related failure information gathered from various documents was analyzed for reoccurring failure patterns. The results of this survey are to be used to implement a research program that will identify nuclear power plant facility aging effects.	SSCs	Aging of nuclear power plant facilities
NUREG	NUREG/CR-4273,IS-4878	Crack Propagation in High Strain Regions of Sequoyah Containment	3	1993	Greimann L., Fanous, F., Bluhm, D. / Iowa State University	The objective of this work is to predict the extend of crack propagation which will occur from a postulated small cracks in the high strain regions of the Sequoyah containment.	Containment	Crack propagation in the high strain region of the Sequoyah containment
NUREG	NUREG/CR-4279	Aging and Service Wear of Hydraulic and Mechanical Snubbers Used on Safety-Related Piping and Components of Nuclear Power Plants, Vol. 1	2	1986	Bush. S. H., Heasler, P.G., Dodge, R. E./ PNL	The primary purpose of this report is to asses the effect of various aging mechanisms on snubbers operation.	Hydraulic and mechanical snubbers	Aging mechanisms in snubbers
NUREG	NUREG/CR-4329	Reliability Evaluation of Containments Including Soil-Structure Interaction	12	1985	Pires, J., Hwang, H., Reich, M./ BNL	The probability-based method for the reliability analysis of structures developed in BNL has been extended to include soil-structure interaction in the analysis. An advantage of the direct transformation method used in this work is that it does not require the generation of artificial earthquake time histories.	Reinforced concrete containment	None identified
NUREG	NUREG/CR-4652	Concrete Component Aging and its Significance Relative to Life Extension of Nuclear Power Plants	9	1986	Naus, D. J./ ORNL	The objectives of this study are (1) to identify aging and service wear effects that could cause degradation of SSCs; (2) to identify methods of inspecting monitoring and evaluating residual life of SSCs; (3) to evaluate the effectiveness of storage, maintenance, repair, and replacement practices in mitigating the rate and extent of degradation caused by aging.	Concrete components	Aging and service degradation of prestressed concrete containments and reactor vessels, and miscellaneous reactor concrete structures

TYPE	ID	TITLE	MO	YR	AUTH/ORG	DESCRIPTION	TYPES OF COMPONENTS	POTENTIAL AGING ISSUES
NUREG	NUREG/CR-4731	Residual Life Assessment of Major Light Water Reactor Components, Vol. 1	6	1987	Shah, V. N. / INEL,EGG	The report presents an assessment of the aging of selected major light water reactor components and structures. Unresolved technical issues related to understanding and managing the aging of these components are identified.	SSCs	Aging of SSCs
NUREG	NUREG/CR-4731	Residual Life Assessment of Major Light Water Reactor Components, Vol. 2	11	1989	Amar, A.S. et al. / INEL,EGG	The report presents an assessment of the aging of selected major light water reactor components and structures. Unresolved technical issues related to understanding and managing the aging of these components are identified.	SSCs	Aging of ping systems, CRDs, steam generator tubing, BWR containments, electrical cables, and pressurizers
NUREG	NUREG/CR-4977	SHAG Test Series: Seismic Research on Aged United States Gate Valve and on a Piping System in the Decommissioned Heissdampfreactor(HDR):Summary. Vol. 1, 2	8	1989	Steele, R., Arendts, J. G./ INEL	The objectives of the HDR seismic program were: 1) to measure the effects of hydraulic and dynamic loads on gate valve operability, 2) to obtain valve response to multiaxial , in situ seismic loads, and 3) to obtain piping system response data.	Gate valves, piping systems, supports	Damaging effects of hydraulic and dynamic loads
NUREG	NUREG/CR-5248	Prioritization of TIRGALEX-Recommended Components for Further Aging Research	11	1988	Levy, I.S. et al. / PNL	The report identified safety-related structures and components that should be prioritized for evaluation in the NRC NPAR program.	SSCs	Assessment of aging effects for RPVs, concrete structures, piping systems, supports, steam generators, heat exchangers, CRDs, and pressurizers
NUREG	NUREG/CR-5314, Vol. 3	Life Assessment Procedures for Major LWR Components, Cast Stainless Steel Components	10	1990	Jaske, C. E., Shah, V. N./ INEL	The report presents a procedure for estimating the current condition and residual life of safety-related cast stainless steel components in light water reactors. The procedure accounts for loss of fracture toughness caused by thermal embrittlement.	Core internals, recirculation piping, control rod drive mechanism, core internals.	Life assessments for core internals, recirculation piping, control rod drive mechanisms, and core internals

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NUREG	NUREG/CR-5314, Vol. 5	Insights for Aging Management of Light Water Reactor Components, Metal Containments	1	1994	Shah, V. N., Sinha, U. P. / INEL, Smith, S. K./ OEES	The report evaluates available technical information and field experience related to the management of aging damage to light water reactor metal containments. A generic aging management approach is suggested for the effective aging management of metal containments.	Metal containments	Aging management of metal containments
NUREG	NUREG/CR-5379, Vol. 1	Nuclear Plant Service Water System Aging Degradation Assessment, Phase 1	6	1989	Jarrel, D. B./ PNL	This report assesses information: 1) to identify the principal aging mechanism, 2) to examine the current surveillance specifications, 3) to produce an inspection plan, and 4) to use the information to resolve related generic issues for developing regulatory criteria on aging and life extension.	Service water system (SWS)	Corrosion, compounded by biologic and inorganic accumulation of the SWS
NUREG	NUREG/CR-5379, Vol. 2	Nuclear Plant Service Water System Aging Degradation Assessment, Phase 2	10	1992	Jarrel, D. B./ PNL	The goals of the Service Water System (SWS) aging degradation assessment task was to advance the understanding and management of the technical safety issues related to the aging of SWS in operating commercial power plants.	Intake structure, pump galley and structures, piping distribution network from the pumps to heat exchangers, all discharge piping from exchangers to outlet or discharge structure, discharge structure.	Aging degradation caused by corrosion, fouling (biological and inorganic), and wear
NUREG	NUREG/CR-5407	Assessment of the Impact of Degraded Shear Wall Stiffnesses on Seismic Plant Risk and Seismic Design Loads	2	1994	Klamerus, M. P. et al./ SNL, EQE	The USNRC has sponsored Sandia National Laboratory to perform an evaluation of the effects of reduced natural frequencies (caused by concrete cracks) on several existing seismic Probabilistic Risk Assessments in order to determine seismic risk implication. This report presents results for the reevaluations of the seismic risk for three nuclear power plants.	Nuclear power plant buildings	None identified
NUREG	NUREG/CR-5490, Vol. 1	Regulatory Instrument Review: Management of Aging of LWR Major Safety-Related Components	10	1990	Werry, E. V./ PNL	The focus of the review was on 26 NPAR-defined safety-related aging issues, including examination, inspection, maintenance and repair, excessive/harsh testing, and irradiation embrittlement.	Light water reactor pressure vessels, steam generators, reactor coolant piping, and pressurizers.	Aging degradation of major LWR safety-related components

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NUREG	NUREG/CR-5491	Shippingport Station Aging Evaluation	1	1990	Allen, R. P., Johnson,A.B. / PNL	The report presents information on the Shippingport station and its decommissioning, a discussion of the selection and relevancy of naturally aged components and the lessons learned from the studies.	SSCs	Aging of seam generators, and piping systems for the Shippingport Station
NUREG	NUREG/CR-5507	Results from the Nuclear Plant Aging Research Program: Their Use in Inspection Activities	9	1990	Gunther, W., Taylor, J./ BNL	This report provides recommendations for communicating pertinent information to NRC inspectors which are based on a detailed assessment of the NRC's Inspection Program, and feedback from resident and regional inspectors.	Component cooling water (PWRs) and residual heat removal (BWRs)	A summary of the research, coupled with an aging inspection guide will provide the inspectors with important insights into the aging degradation of various equipment and systems.
NUREG	NUREG/CR-5612	Degradation Modeling with Application to Aging and Maintenance Effectiveness Evaluation	3	1991	Samanta, P. K. et al./ BNL	This report presents a modeling approach to analyze component degradation and failure data to understand the aging process of components. Reported results are an important step in showing that degradation can be modeled to identify aging effects.	None identified	None identified
NUREG	NUREG/CR-5643	Insights Gained from Aging Research	3	1992	Blahnik, D. E. et al./ BNL	This program has identified components and systems that have a propensity for age-related degradation and has evaluated methods for detecting and mitigating aging effects.	Components and systems	Age related degradation of heat exchangers, CRDs, and pipe supports
NUREG	NUREG/CR-5700	Aging Assessment of Reactor Instrumentation and Protection Systems Components, Phase 1	7	1992	Gehl, A.C., Hagen,E.W./ ORNL	This study examined the effects of aging on equipment performance and normal service life for instrumentation and protection systems.	Instrumentation and protection systems.	The effect of aging on instrumentation and protection systems

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NUREG	NUREG/CR-5754	Boiling-Water Reactor Internals Aging Degradation Study Phase 1	9	1993	Luk, K. H./ORNL	The report documents the results of a study on the effects of aging on 25 selected BWR internal components. A data base is established using data from LER's. Two major age-related degradation mechanisms were identified: stress corrosion cracking and fatigue.	BWR internal components	The effects of aging of 22 BWR internal components including core shroud, shroud head, core plate, core spray sparger, CRD housing, control blade, etc
NUREG	NUREG/CR-6048	Pressurized-Water Reactor Internals Aging Degradation Study Phase I	9	1993	Luk, K. H./ ORNL	The main objective of this study is to assess the effects of aging degradations on PWR internal components. The assessment includes an evaluation of the effectiveness of the plant in-service inspection program in detecting failures.	PWR internal components	Aging degradation of PWR internals including thermal shield support bolts, core support barrel, hold-down ring, control rod guide tube support pins, etc
NUREG	NUREG/CR-6052	Methodology for Reliability Based Conditions Assessment, Application to Concrete Structures in Nuclear Plants	6	1993	Mory, Y, Ellingwood,B /John Hopkins University	This research is a part of the Structural Aging Program (SAG). The goal of this report is to develop a set of probability-based tools to facilitate the quantitative assessment of current and future structural reliability and performance of concrete structures.	Concrete structures	Assessment of concrete structures
NUREG	NUREG/CR-6424 ORNL/TM-13148	Report on Aging of Nuclear Power Plant Reinforced Concrete Structures	3	1996	Naus, D. J./ ORNL	This report is a summary of the Structural Aging (SAG) Program. Included are information on longevity of NPP reinforced concrete structures, a structural materials information center, in-service inspection and condition assessment techniques, repair methods and materials, and reliability-based methodology for concrete assessment.	Reinforced concrete structures	Assessment of concrete, steel reinforcement, prestressing steel, liner plate, and embedded steel
NUREG	NUREG/CR-6425 ORNL/TM-13149	Impact of Structural Aging on Seismic Risk Assessment of Reinforced Concrete Structures in Nuclear Power Plants	3	1996	Ellingwood, B. R., Song, J.	This report examines the role played by structural degradation on plant risk through the vehicle of a seismic PRA of an operating PWR. It seeks to determine whether changes in certain critical structural component or system capacities due to reinforcement corrosion or concrete deterioration from aggressive environmental influences have a statistically significant impact on the probability of core damage or plant damage states.	Concrete structures and systems	Aging degradation of concrete structures

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NUREG	NUREG/CR-6490 , Vol. 1 & 2	Nuclear Power Plant Generic Aging Lessons Learned (GALL)	11	1996	Kasza, K.E.	The purpose of this report is to provide a systematic review of plant aging information in order to asses material and component aging issues related to the continued operation and license renewal of operating reactors.	Mechanical and structural components and systems	Assessment of reactor internals, closure studs, RPV upper head, CRD housing, core shroud, shielding wall concrete, etc
NUREG	NUREG/CR-6598	An Investigation of Tendon Sheathing Filler Migration into Concrete	3	1998	Naus, D. J., Oland, C. B. / ORNL	The objective of this report was to provide an indication if leakage of the tendon sheathing filler into the concrete shell of prestressed concrete containments affect the concrete properties to the extent that the containment structural capacity could be affected.	Concrete containment	Degradation of concrete properties
NUREG	NUREG/CR-6631	Fragility Modeling of Aging Containment Metallic Pressure Boundaries	8	1999	Ellingwood B., Cherry, J.	The report presents a general framework for probabilistic modeling of containment structural performance, with emphasis on steel containment and reinforced concrete containments with steel liners subjected to corrosion. The analytical modeling presumes the availability of an advanced nonlinear finite element code. The report concludes with a discussion of insights and perspectives that might be drawn from such fragility analysis.	Containments	Corrosion of steel liners and steel containments
PAPER		Effect of Aging Degradation on Seismic Performance of Reinforced Concrete Structures: Summary of Japanese Literature in Related Areas	9	1998	Park, J./ BNL	This report presents a summary of a literature survey of available Japanese publications.	Reinforced concrete	Degradation of reinforced concrete
PAPER	ACI Material Journal, Vol. 87, No. 5	Effect of Rusting Reinforcing Steel on its Mechanical Properties and Bond with Concrete	10	1990	Maslehuddin, M. et al.	The investigation was carried out to determine the mechanical properties and bond strength of reinforcing steel exposed to natural environments for periods up to 16 months. Developed data will clarify the doubts often expressed by field engineers in the use of reinforcing steel.	Reinforced concrete	Degradation of steel reinforcement

TYPE	ID	TITLE	MO	YR	AUTH/ORG	DESCRIPTION	TYPES OF COMPONENTS	POTENTIAL AGING ISSUES
PAPER	ACI Structural Journal, Vol. 96 No. 3	Corrosion Influence on Bond between Steel and Concrete	5	1999	Amleh, L., Mirza, S.	The primary objective of this research was to simulate severe local corrosion conditions under a very aggressive environment. The results of this research leads to a better understanding of the corrosion problem, and its influence on bond between the reinforcing steel and the concrete.	Concrete structures	Effect of reinforcement slip on reinforced concrete members
PAPER	ACI Structural Journal, Vol. 96, No. 3	Effects of Reinforcement Slip on Hysteretic Behavior of Reinforced Concrete Frame Members	5	1999	Filippou, C., D'Ambrisi, A., Issa, A.	A new approach in describing the nonlinear histeretic behavior of reinforced concrete frame elements is proposed in this paper. This approach consists of isolating the mechanisms that control the hysteretic behavior of girders and columns into individual subelements that are connected in a series to form the complete girder or column element. The analytical results show agreement with experimental data.	Concrete structures	None identified
PAPER	ACI Structural Journal, Vol. 96, No. 3	How to Treat Shear in Structural Concrete	5	1999	Marty, P.	Starting from some remarks on structural analysis, this paper reviews the recent development of the strut-and-tie model, compression field and limit analysis approaches, and attempts to show how the different methods supplement each other, and how they can be used in the design of a new structures, and in the evaluation of existing structures.	Concrete structures	None identified
PAPER	ASME, NDE, Vol. 5	Nondestructive Evaluation NDE Planning and Application, PVP Conference, Honolulu, July, 1989		1989	ASME	This volume attempts to bridge the technical and nontechnical aspects of NDE with the goal of bringing NDE technology to the users, the designers and analysts and the governing bodies of the pressure vessel and piping community.	SSCs	NDE assessment of containment, RPV, and piping
PAPER	ASME,PVP-Vol.171	Life assessment and Life Extension of Power Plant Components - 1989, PVP Conference, Honolulu, July, 1989		1989	ASME	The objective of this symposium volume is to discuss the technical, and economic issues related to aging, remaining life assessment, and life extension.	RPV, piping, cables, tubing	Aging and life assessment of RPV, piping, and tubings

TYPE	ID	TITLE	MO	YR	AUTH/ORG	DESCRIPTION	TYPES OF COMPONENTS	POTENTIAL AGING ISSUES
PAPER	Concrete Durability, SP170-40	A Rational Model to Predict the Service Life of RC Structures in Marine Environment			Yokozeki, K. et al.	A model for predicting the crack development period after the onset of corrosion in reinforcing bars due to chloride-induced deterioration and a rational service life prediction model of reinforced concrete structures are proposed in this paper. The validity of this proposal is discussed.	Concrete structures	Corrosion and cracks development in concrete structures
PAPER	Concrete International	Prestressed Concrete in U.S. Nuclear Power Plants, Part 1 and 2	5	94	Ashar, H. et al./ NRC	The paper discusses different prestressing systems, the role of corrosion and hydrogen embrittlement in containment aging and degradation, alternative methods of prestressed concrete containment in-service inspection, and reports several cases of aging and degradation of prestressed concrete containments.	Prestressed concrete contaimnents	There are several locations of higher stress concentration where stress could reach well above the yield strength of material. These conditions, triggered by a conducive environment, are ideal for hydrogen assisted cracking
PAPER	Concrete Repair Bulletin, September/ October 1997	Power Plant Life Extended by Repair, Maintenance Program	9	1997		The holistic system approach to repair concrete is identified to be a critical key factor in controlling further corrosion deterioration. This paper provides information about the holistic system approach and a comprehensive repair and maintenance program.	Concrete	Assessment of concrete degradation
PAPER	Concrete under Severe Conditions: Environment and Loading, Vol. 1	Proposed Durability Design for Marine Structures		1995	Yamamota, A. et al.	This paper presents research to establish a durability design method for reinforced concrete marine structures considering chloride-induced deteriorating mechanisms, such as chloride diffusion and chloride permeability.	Concrete structures	Chloride-induced deterioration of concrete structures
PAPER	Darmstadt Concrete, Annual Journal on Concrete and Concrete Structures, Vol. 11	Estimation of Crack Depth in Concrete Utilising Ultrasonic Impulse-Echo-Technique		1996	Jansohn, R., Kroggel, O., Ratmann, M.	This paper discuses the assessment of corrosion of reinforcement using ultrasonic back scatter techniques and estimations of crack depth in concrete utilizing the ultrasonic impulse-echo technique.	Concrete and concrete structures	Corrosion of concrete and concrete structures

TYPE	ID	TITLE	MO	YR	AUTH/ORG	DESCRIPTION	TYPES OF COMPONENTS	POTENTIAL AGING ISSUES
PAPER	Durability of Building Materials and Components, Vol. 1, London	Effects of Rebar Corrosion on the Structural Performance of Singly Reinforced Beams		1996	Lee, H., Noguchi, T., Tomosawa, F.	The purpose of this study was to investigate the relation between the degree of rebar corrosion and the strength of reinforced beams by the finite element method. Tension test and pull-out bond tests on rebars were conducted to obtain the constitutive laws for rebar elements and bond elements. The analysis results were then verified by static test on the beams.	Reinforced concrete beams	Rebar corrosion in reinforced beams
PAPER	Engineering Structures, Vol. 20	Pros and Cons of Pushover Analysis of Seismic Performance Evaluation	0	1998	Kravinker, H., Seneviratna, G.	The purpose of this paper is to summarize basic concepts on which the pushover analysis can be based, assess the accuracy of pushover predictions, identify conditions under which pushover will provide adequate information, and identify cases in which the pushover predictions will be inadequate.	Steel and concrete structures	None identified
PAPER	International Conference on Structural Faults and Repair, London	Load Carrying Capacity of Concrete Structures With Corroded Reinforcement	7	1995	Rodriquez, J, et al.	Corrosion of reinforcing bars is one of the main causes which induces an early deterioration of concrete structures. This paper summarizes research work in order to relate the level of steel corrosion to load carrying capacity and serviceability of concrete beams.	Concrete structures	Corrosion of steel
PAPER	International Congress on Creating with Concrete, Dundee, 1999	Design of Concrete Structures in the 21st Century		1999	Sakai, K. et al.	In this paper a new framework for the design of concrete structures is proposed and a numerical calculation example is shown on the basis of performance-based evaluations of a reinforced concrete beam exposed to a chloride-laden atmosphere.	Concrete structures	The effect of chloride-laden atmosphere on concrete structures
PAPER	Journal of Materials in Civil Engineering, Vol. 4, No. 4	Corrosion Cracking in Relation to Bar Diameters, Cover, and Concrete Quality	11	1992	Rasheeduzzafar, et al.	This paper attempts to quantify the effect of three parameters: concrete cover, concrete quality, and bar size in providing corrosion protection to reinforcing steel. In view of the importance of the c/d ratio, clear cover specification without consideration of the bar size leads to inadequate and misleading design for corrosion protection.	Reinforced concrete	Corrosion of rebars in reinforced concrete structures

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PAPER	Nagaoka University of Technology, Nagaoka, Japan	Degradation Model for Reinforced Concrete Structures under Salt Attack Environment			Maruyama, K., Shimomura, T Hamada, H.	This study aims to make a framework to predict the time-dependent changes of structural capacity of reinforced concrete structures. Combining experimental studies on the deterioration of structures due to salt attack with the studies on penetration and dispersion of corrosion inducers into concrete, the authors proposed a simulation on how concrete is deteriorating under salt attack.	Concrete structures	Degradation of concrete due to salt attack.
PAPER	Nagaoka University of Technology, Nagaoka, Japan	Effect of Rebar Corrosion on the Structural Capacity of Concrete Structures			Maruyama, K Shimomura, T	The paper discusses the mechanical deterioration of reinforced concrete beams under a salt attack environment. Taking the amount of stirrups (varying the stirrup spacing) as a parameter, the flexural capacity of beams was experimentally examined under static and fatigue loading.	Concrete structures	Degradation of concrete due to salt attack
PAPER	Nagaoka University of Technology, Nagaoka, Japan	Residual Capacity of Concrete Beams Damaged by Salt			Kawamura, A et al.	For the maintenance and repair of existing concrete structures it is necessary to determine the residual capacity of structures damaged by salt attack. This paper discusses the reduction of flexural capacity of RC beams when the longitudinal reinforcing bars corrode.	Concrete beams	Degradation of concrete beams due to salt attack
PAPER	Nagaoka University of Technology, Nagaoka, Japan	Simulation of Time-dependent Performance Change of RC Structures Subjected to Salt Attack		1999	Shimomura, T	This paper presents an approach to predict the time-dependent performance change of structures by means of numerical simulations based on a concept of performance-based integrated design and durability of concrete structures. A comprehensive computational system composed of mathematical models was demonstrated with a case study for a RC T-beam exposed to a chloride-laden atmosphere.	Concrete structures	Degradation of concrete structures
PAPER	Nuclear Engineering and Design 166	Aging Management of Containment Structures in Nuclear Power Plants		1996	Naus, D. J. et al.	This paper discusses degradation factors important to aging management, evaluation of non-destructive techniques, assessments of repair practices for concrete, review of the parameters affecting corrosion of metals embedded in concrete and service life predictions of new or existing reinforced concrete structures.	Containment structures	Aging management of concrete and steel containments

TYPE	ID	TITLE	MO	YR	AUTH/ORG	DESCRIPTION	TYPES OF COMPONENTS	POTENTIAL AGING ISSUES
PAPER	OECD-NEA Workshop on FE Analysis of Degraded Concrete Structures	FE Analysis of Degraded Concrete Structures: Current Knowledge and Prospects for the Future	10	1998	Park, Y.	This paper presents an overview of past experimental studies on the seismic response of degraded RC components based on a recent literature survey of Japanese documents. The state-of-the art of the application of nonlinear FE analysis to RC structures under severe earthquake loading are described.	Concrete structures	Degradation of concrete components.
PAPER	Proceedings of the Fifth International Conference, Brighton, UK, November 1990	Prediction of Service Lives of Reinforced Concrete Buildings Based on the Corrosion Rate of Reinforcing Steel	11	1990	Morinaga, S./ Shimizu Corp, Japan	The subject of this paper was an investigation of the life of reinforced concrete structures and to find methods to predict service life.	Reinforced concrete structures.	Corrosion of reinforcing steel.
PAPER	Proceedings of the International Conference on Concrete under Severe Conditions, Vol. 2	An Experimental Study on Deterioration of Aseismatic Behavior of R/C Structural Walls Damaged by Electrolytic Corrosion Testing Method		1995	Yamakawa,T	The study is a trial to investigate the influence of steel corrosion on the aseismatic behavior of RC structural walls under chloride attack. Loading tests for the 6 tests specimens were conducted under constant gravity load and repeated lateral forces. Experimental results and a discussion of these test specimens are reported in this paper.	Reinforced concrete walls.	Corrosion of rebar in reinforced concrete structures
PAPER	Proceedings of the Second Conference on Concrete under Severe Conditions Vol. 2, Norway, June 1993	An Experimental Study on Damage Affecting Aseismatic Behavior of Structural Walls under Chloride Attack Envir. Of the Semitrop. Region	6	1993	Yamakawa,T et al.	This paper discusses damage affecting the seismic behavior of reinforced concrete structural members due to corrosion of steel reinforcing bars through experiments under constant axial load and alternately repeated lateral loads.	Structural walls	Corrosion of rebar in reinforced concrete structures
PAPER	Proceedings of the Second Conference on Concrete under Severe Conditions, Vol. 2, Norway, June 1993	Seismic Behavior of R/C Columns Damaged under Exposure Test	6	1993	Yamakawa,T	Test results for RC column speciments damaged by three years of exposure were provided. At the same time a none-corroded RC column specimen was also tested under combination with cyclic lateral forces and a constant axial compression load. The test results are compared and a discussed in this paper with respect to the relationship between performance and corrosion of the RC columns.	RC columns.	Weathering of RC columns

TYPE	ID	TITLE	MO	YR	AUTH/ORG	DESCRIPTION	TYPES OF COMPONENTS	POTENTIAL AGING ISSUES
PAPER	Proceedings of the Second International Conference on Concrete under Severe Conditions, Norway, June 1993	FEM Analysis for Structural Performance of Deteriorated RC Structures due to Rebar Corrosion	6	1993	Lee, H., Noguchi, T., Tomosawa,F	This study provides an analytical determination of the structural performance of reinforced concrete beams with corroded rebars by FEM. Tension test and pull-out bond tests on rebars were conducted to obtain the constitutive laws for rebar elements and bond elements. A parametric analysis of constitutive laws for each material was then conducted by the FEM, to examine the mechanism of the structural performance reduction of corroded concrete beams.	Reinforced concrete structures	Corrosion of rebar in reinforced concrete structures
PAPER	Proceedings of the Sixth Symposium at North Carolina State University	Degradation of Passive Components	12	1996	Ashar, H. , Jeng, D./ NRC	This paper presents failures and degradation of passive structural components at NPPs.	Passive structures and components	Failure and degradation of piping supports, anchorages, and water-control structures
PAPER	Proceedings of the Sixth Symposium at North Carolina State University	Seismic Responses and Resistance of Age Degraded Structures and Components	12	1996	Shao, L.C. et al./ NRC	This paper discusses failures and the degradation of safety-related structures and components and certain nonsafety-related passive structures and components whose failure could prevent the safety-related structures and components from performing their safety function. The paper discusses the aging experience of these in NPPs, the aging issues, past and current research programs, and potential areas for research.	Passive structures and components	Effect of aging on passive structures and components
PAPER	Proceedings of the Sixth Symposium at North Carolina State University	Structural Condition Monitoring in a Life Cycle Management Program	12	1996	Esselman T.C, Eissa M. A., McBrine W. J. / Altran Corp., Boston	This paper describes an aging assessment program that would be performed as part of a comprehensive Life Cycle Management program in nuclear power plants. The program is degradation-based and relies on the recognition and monitoring of real degradation in the plant.	SSCs	None identified
PAPER	Proceedings of the SMIRT 14, Lyon, France, August 17-22, 1997	Nuclear Power Plant Generic Aging Lessons Learned (GALL)	8	1997	Regan, C./ NRC	The purpose of the generic aging lessons learned paper was to provide a systematic review of plant aging information to assess material and component aging issues related to the operation and license renewal of nuclear power plants. The results reveal that all significant aging issues are being addressed by the USNRC regulatory process.	SSCs	Assessment of material and component aging issues for SSCs

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PAPER	Special Report of Institute of Technology, Shimizu Corporation, No. 23	Prediction of Service Lives of Reinforced Concrete Buildings Based on the Rate of Corrosion of Reinforcing Steel	6	1988	Morinaga, S.	The corrosion of reinforcing steel due to carbonation of concrete and chloride included in concrete were investigated in this paper. A method for predicting the life of reinforced concrete buildings determined by the corrosion of the reinforcing steel was established.	Reinforced concrete	Corrosion of rebar in reinforced concrete structures
PAPER	University of Transport and Communication, Zilina, CSFR	The Influence of the Reinforcement Corrosion on the Load-Bearing Capacity of Reinforced Concrete Structures			Kapasny, L., Zemco, S.	The paper deals with the causes of steel reinforcement corrosion in concrete structures, methods of investigation, influence of corroded reinforcement on concrete and the load-bearing capacity of the concrete structural elements.	Concrete structures	Corrosion of rebar in reinforced concrete structures
PAPER	www.itd.anl.gov/highlights/index.html	Argonne and General Electric More Accurately Predict Reactor Vessel Aging			NRC	Argonne has demonstrated that high-energy gamma-rays are considerably more important than previously suspected in degrading the reactor pressure vessel in certain commercial nuclear plants.	RPV	Gamma ray induced degradation of the RPV
PAPER	www.nrc.gov/glimpsel/mfs/o2/3791450	Steam Generator Tube Repair Limits			Ward, D. A./ NRC	The sudden rupture of steam generator tubes due to a transient such as a steam line break or a seismic event needs to be precluded. Paper discusses issues of steam generator tube repair limits.	Exchangers-Steam Generator-Tubes	Corrosion of rebar in reinforced concrete structures.
PAPER	www.nrc.gov/OPA/gmo/tip	Reactor License Renewal			NRC	This paper discusses issues of regulation and guidance for reactor license renewal. NRC's task is to establish a reasonable process and safety standards so that licensees can make timely decisions whether to seek license renewal or not.	SSCs	Age-related degradation of "passive" and "long lived" structures, systems, and components and updating of time-limited aging analyses.

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PAPER	www.nrc.gov/OPA/gmo/tip/tip07.htm	Reactor Pressure Vessel Embrittlement and Annealing			NRC	An elevated transition temperature makes RPV beltline materials more susceptible to rapid crack growth under conditions such as those which can occur from pressurized thermal shock (PTS). This document reviewed data relevant to the PTS evaluations of several plants.	RPV	Embrittlement induced failure due to thermal shock
PAPER	www.nrc.gov/OPA/gmo/tip/tip27/htm	Steam Generator Tube Issues			NRC	This paper describes different kinds of steam generator tube degradations and methods of performing inservice inspection of the tubes.	Exchangers - Steam Generator - Tubes	Degradation of exchangers, steam generators and tubes due to transients
PAPER	www.nrc.gov/OPA/gmo/tip/tip29.htm	BWR Reactor Internals			NRC	This paper presents information about cracking in the core shroud in several nuclear power stations inside and outside of U.S.A and highlights BWR internals cracking issues.	RPV	Cracking of the RPV-Core Shroud
PAPER	www.sandia.gov/labnews/ln03-01-96/anneal.html	Annealing Process Reverses Long-term Effect of Radiation Bombardment in Reactors	3	1996	SANDIA	Sandia National Laboratory is working with researchers from Russia to determine whether it is possible to reverse the aging process that causes steel pressure vessels to lose some of their ductility over time. One approach to managing this embrittlement is thermal annealing - heating the vessel and then cooling it to ambient temperature in a specific, controlled way.	RPV	Aging degradation of RPV
PRESENTATION	PLIM + PLEX 95 , NICE	Detecting and Monitoring the Aging of Civil Engineering Structures in Nuclear Facilities	11	1995	Heep, W. / Nok Engineering, Baden, Swiss	This paper focuses on the Swiss approach to cope with the aging management of systems, structures and components of nuclear power plants.	SSCs	Not discussed

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PRESENTATION	PLIM + PLEX 95, NICE	Aging Surveillance and Remaining Lifetime Evaluation Based on Intelligent Inspection Planning and Manager System	11	95	Aguado, M., Cuerto, C./ Tecnatom,S,A	This paper briefly describes the methodology used for aging surveillance and its application to the reactor vessel of a PWR plant, along with its implementation in a computer system allowing management of the remaining lifetime of the equipment to be optimized.	RPV	The surveillance of aging in the RPV, and CRDM
PRESENTATION	PLIM + PLEX 95, NICE	STUDSVIC's Work on Irradiation Effects in Materials	11	95	Grounes, M./ Studsvik Nuclear	This paper presents irradiation-effects studies which have been in progress at Studsvik since the late 1950's. One of the main areas of investigation is failure analysis of failed reactor components.	Systems and components	The effect of irradiation on the RPV
PRESENTATION	www.nrc.gov/OPA/gmo/news96.htm	Current Regulatory Challenges	7	96	Jackson, S. A./ NRC	The presentation discusses NRC safety philosophy, vision, and goals. Aging and degradation effects on nuclear power plants, license renewal, maintenance rules are addressed in this presentation.	SSCs	Aging degradation of NPPs
PRESENTATION	www.nrc.gov/OPA/gmo/tip/tip01.htm	Challenges of Change	4	1996	Jackson, S. A./ NRC	This speech before the Japan Atomic Industrial Forum describes changes in economics, government, industry aging of reactors, life extension, waste storage and disposal, and new reactor design.	SSCs	Aging and life extension of reactors
REGULATION	10 CFR 50.65	Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	1	1996	US Govt.	This document establishes rules for monitoring the performance or condition of SSCs, against licensee-established goals, in a manner sufficient to provide reasonable assurance that such SSCs are capable of fulfilling their intended functions.	SSCs (safety-related and certain nonsafety-related)	None identified

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REGULATION	10 CFR 54	Requirements for Renewal of Operating Licenses for Nuclear Power Plants	1	1997	US Govt.	This document governs the issuance of renewed operating licenses for nuclear power plants. Requirements relating to aging degradation are described. This includes an integrated plant assessment (IPA) and aging management requirements.	SSCs (safety-related and certain nonsafety-related)	Not discussed
REGULATION	IE IN 85-24	Failures of Protective Coating in Pipes and Heat Exchangers	3	1985	NRC	This information notice is provided to alert addressees about significant problems pertaining to the selection and application of protective coatings for safety-related use, especially painting interior surfaces of pipes and tubing.	Pipes, Heat exchangers	Protective coating for pipes, and heat exchangers
REGULATION	INSPN. PROC. 62002	NRC INSPECTION MANUAL	12	1996	NRC - ECGB	This procedure describes inspection requirements for the assessment of licensee-developed maintenance programs for structures, passive components, and civil engineering features within the scope of 10CFR50.65 "Requirements for Monitoring the Effectiveness of Maintenance at NPPs."	Passive SSC's	Maintenance programs for intake structures, masonry walls, steel structures, dams, supports and anchorages.
REGULATION	INSPN. PROC. 62003	NRC INSPECTION MANUAL (Inspection of Steel and Concrete Containment Structures at Nuclear Power Plants)	6	1997	NRC - ECGB	This procedure describes inspection requirements for the assessment of the effectiveness of licensees inspection programs within the scope of 10CFR50.55	Steel and concrete containment structures	Inspection programs for steel and concrete containment structures
REGULATION	Letter to NEI	NRC Comments on NEI 96-03, Rev. D, "Guideline for Monitoring the Condition of Structures at Nuclear Power Plants"	10	1996	Martin T.T. / NRC	This document presents comments on the current version of NEI 96-03. NRC concludes that the NEI guideline is not acceptable for use under the license renewal rule because the document lacks specific details in areas applicable to license renewal.	Structures: concrete, masonry walls, structural steel, roof systems, siding, windows/doors, earthen structures/dams, prestressing steel, steel liner plate, rebars, embedment steel	Monitoring of degradation mechanisms associated with components

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REGULATION	NRC Inspection Procedure 62706	NRC Inspection Manual Maintenance Rule	8	1995	NRC	This document discusses the implementation of the Maintenance Rule (10CFR50.65), the Station Blackout Rule, and Generic Letter 94-01 following the guidance in Reg. Guide 1.160 and NUMARC 93-01.	SSCs	None identified
REGULATION	NRC BL 88-09	Thimble Tube Thinning in Westinghouse Reactors	7	1988	NRC	The purpose of this bulletin is to request that addressees establish and implement an inspection program to periodically confirm incore neutron monitoring system thimble tube integrity. Wear of the thimble tubes results in degradation of the RCS pressure boundary and also create potentially non-insoluble leak of reactor coolant.	RPV	Wear of the RPV-Thimble Tube
REGULATION	NRC GL 88-01	NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping	1	1988	NRC	The NRC staff continues to believe that replacement with IGSCC resistant materials will provide the greatest degree of assurance against future cracking problems. Considering that each piping system has many weldments and each plant has many piping systems, the entire problem must be evaluated in an integrated way.	Piping	IGSCC in BWR piping
REGULATION	NRC GL 88-11	NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations	7	1988	NRC	Purpose of this letter is to call attention to Regulatory Guide 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials ." Licensees should use the methods described in this document to predict the effect of neutron radiation on reactor vessel materials.	Vessels-RPV	Radiation embrittlement of reactor pressure vessel
REGULATION	NRC GL 92-01 Rev. 1	Reactor Vessel Structural Integrity, 10 CFR 50. 54 (f) (Generic Letter 92-01, Revision 1)		1992	NRC	This letter replaces Generic Letter 92-01. NRC issued this generic letter to obtain information needed to assess compliance with requirements and commitments regarding reactor vessel integrity in view of certain concerns raised in the staff's review of reactor vessel integrity for the Yankee NPS.	Reactor pressure vessels	Assessment of reactor presure vessel

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REGULATION	NRC IE Bulletin 80-11	Masonry Wall Design	5	1980	NRC	IN 79-28 describes a problems with structural integrity of concrete masonry walls. IE Bulletin 80-11 addresses action to be taken by all power reactor facilities with an Operating License to resolve these problems.	Concrete masonry walls	None identified
REGULATION	NRC IN 85-10	Post-tensioned Containment Anchor Head Failure	2	1985	NRC	The objective of this notice is: (1) to present methods available to evaluate the capability of mass concrete to meet design criteria, and (2) to detect the retrogression in physical properties of concrete which could affect the capability of the concrete.	Concrete structures	Failure of containment tendon anchor.
REGULATION	NRC IN 86-99	Degradation of Steel Containments	12	1986	NRC	This notice is to provide recipients with information of a potentially significant safety problem regarding the degradation of a steel containment resulting from corrosion.	Steel containment	Corrosion of steel containments
REGULATION	NRC IN 87-67	Lessons Learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11	12	87	NRC	This information notice is provided to inform addressees of lessons learned from NRC inspection of activities related to the reevaluation work conducted and plant modifications made in response to IE Bulletin 80-11, Masonry Wall Design. NRC inspectors observed that mechanisms did not exist at certain facilities to ensure that the physical condition of masonry walls remain as previously analyzed.	Masonry walls	Degradation/changes in conditions of masonry walls
REGULATION	NRC IN 88-82	Torus Shells with Corrosion and Degradation of Coating	10	1988	NRC	This notice is provided to alert addressees to the discovery of suppression pool steel shells with corrosion and degraded coatings in BWR containments.	Torus shell	Degradation of coating and corrosion of BWR containments

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REGULATION	NRC IN 89-79	Degraded Coatings and Corrosion of Steel Containment Vessel	12	1989	NRC	This notice is intended to provide information about severely degraded coatings and the corrosion of steel containment vessels that are caused by boric acid and collected condensation in the annular space between the steel shell and concrete shield building.	Steel containment vessels	Degradation and corrosion of steel containments
REGULATION	NRC IN 91-18	High-Energy Piping Failure Caused by Wall Thinning	3	1991	NRC	This information notice is intended to alert addressees to continuing erosion/corrosion problems affecting the integrity of piping systems. Despite implementation of long-term monitoring programs pursuant to Generic Letter 89-08, "Erosion/Corrosion Pipe Wall Thinning", piping failures caused by wall thinning continue to occur in operating plants.	Piping	Erosion and corrosion of piping
REGULATION	NRC IN 93-21	Summary of NRC Staff Observation Compiled During Engineering Audits of Inspection of Licensees Erosion/Corrosion Programs	3	1993	NRC	This summary states that most of the problems in implementing erosion/corrosion programs pertain to weaknesses or errors in (1) using predictive models, (2) calculating minimum wall thickness acceptance criteria of the code, (3) analyzing the results of UT test, (4) self assessment of erosion/corrosion programs activities, (5) dispositioning components after reviewing the results of inspection analyses or (6) repairing or replacing components that failed to meet minimum wall thickness criteria.	Piping, Tubing	Corrosion of piping and tubing
REGULATION	NRC IN 96-09	Damage in Foreign Steam Generator Internals	2	1996	NRC	This information notice is intended to alert addressees to recent findings of damage to steam generator internals, namely support plates and wrapper, at foreign PWR facilities.	Steam generator internals	Damage of steam generator-support plate and wrapper
REGULATION	NRC IN 96-14	Degradation of Radwaste Facility Equipment at Millstone Nuclear Power Station, Unit 1	3	1996	NRC	This information notice is intended to alert addressees to occurrences of degradation of vessels and piping in the radwaste facility. A lack of continuing and preventive maintenance appeared to allowed several systems and components to significantly degrade.	Radwaste tank and piping	Degradation of piping, and vessels

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REGULATION	NRC IN 97-10	Liner Plate Corrosion on Concrete Containments	3	1997	NRC	This information notice is intended to alert addressees to occurrences of corrosion in the liner plates of reinforced and pre-stressed concrete containments.	Concrete containment	Corrosion in liner plates
REGULATION	NRC IN 97-11	Cement Erosion from Containment Subfoundations at Nuclear Power Plants	3	1997	NRC	This information notice is intended to alert addressees to information regarding the possible erosion of cement from porous concrete subfoundation below the reactor building basemats at some reactor sites.	Containment subfoundations	Erosion of containment subfoundation
REGULATION	NRC IN 97-13	Deficient Conditions Associated with Protective Coating at Nuclear Power Plants	3	1997	NRC	This information notice is intended to alert addressees about several instances in which protective coatings were not properly applied, maintained, or qualified for their intended use and have jeopardized the operability of safety-related equipment.	Tanks, containment liners, piping	Deficient conditions of protective coatings for piping, containment liners and tanks
REGULATION	NRC INSP. REP. 50-220/96-12	Maintenance Rule Inspection at Nine Mile Point 1	1	1997	NRC	This inspection included a review of the licensee's implementation of CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Scope included maintenance. Overall, the team judged the maintenance rule program to be weak.	SSCs	None identified
REGULATION	NRC INSP. REP. 50-2440-96014(DRS)	Maintenance Rule Inspection at Perry Nuclear Power Plant	1	1997	NRC	This inspection included a review of the licensee's implementation of CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Scope included operations, maintenance, QA, and engineering . One violation was issued.	SSCs	None identified

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REGULATION	NRC INSP. REP. 50-255/97003 (DRS)	Maintenance Rule Inspection at Palisades Nuclear Generating Plant	4	1997	NRC	This inspection included a review of the licensee's implementation of CFR 50.65 , "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Scope included operations, maintenance, engineering, and QA. No violations were identified.	SSCs	None identified
REGULATION	NRC INSP. REP. 50-277/96-07 and 50-278/96-07	Maintenance Rule Inspection at Peach Bottom Atomic Power Station Units 1 and 2	10	1996	NRC	This inspection included a review of the licensee's implementation of CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Scope included maintenance . An apparent violation was identified .	SSCs	None identified
REGULATION	NRC INSP. REP. 50-280/97-01 and 50-281/97-01	Maintenance Rule Inspection at Surry 1 and 2	2	1997	NRC	This inspection included a review of the licensee's implementation of CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Scope included operations, maintenance, and engineering . Based on the results of this inspection, seven apparent violations were identified and are being considered for escalated enforcement action.	SSCs	Maintenance of concrete-water intake structures
REGULATION	NRC INSP. REP. 50-282/306-96012	Maintenance Rule Inspection at Prairie Island Nuclear Generation Plant	1	1997	NRC	This inspection included a review of the licensee's implementation of CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Scope included operations, maintenance , QA,and engineering. Three violations were issued.	SSCs	None identified
REGULATION	NRC INSP. REP. 50-286/96-80	Maintenance Rule Inspection at Indian Point 3 Nuclear Power Plant	2	1997	NRC	This inspection included a review of the licensee's implementation of CFR 50.65 , "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Scope included maintenance. One violation was issued.	SSCs	None identified

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REGULATION	NRC INSP. REP. 50-298/96-12	Maintenance Rule Inspection at Cooper Nuclear Station	10	1996	NRC	This inspection included a review of the licensee's implementation of CFR 50.65, , "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Scope included operations, maintenance, and engineering. Six violations were issued.	SSCs	None identified
REGULATION	NRC INSP. REP. 50-315/96009 (DRS) and 50-316/96009 (DRS)	Maintenance Rule Inspection at DC Cook Nuclear Station, Units 1 and 2	11	1996	NRC	This inspection included a review of the licensee's implementation of CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Scope included operations, maintenance, QA, and engineering. Two violations were issued.	SSCs	None identified
REGULATION	NRC INSP. REP. 50-321/96-12 and 50-366/96-12	Maintenance Rule Inspection at Hatch Nuclear Plant, Units 1 and 2	11	1996	NRC	This inspection included a review of the licensee's implementation of CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Scope included operations, and maintenance. Four violations were issued.	SSCs	Maintenance of filters-screens, and strainers
REGULATION	NRC INSP. REP. 50-3227/96-12 and 50-328/96-12	Maintenance Rule Inspection at Sequoyah, 1 and 2	1	1997	NRC	This inspection included a review of the licensee's implementation of CFR 50.65 , "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Scope included operations, maintenance, and engineering. One violation was issued.	SSCs	None identified
REGULATION	NRC INSP. REP. 50-335/96-13 and 50-389/96-13	Maintenance Rule Inspection at St. Lucie, 1 and 2	10	1996	NRC	This inspection included a review of the licensee's implementation of CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Scope included operations, and engineering. Three violations were issued.	SSCs	None identified

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REGULATION	NRC INSP. REP. 50-346/97002 (DRS)	Maintenance Rule Inspection at Davis-Besse Nuclear Power Station	3	1997	NRC	This inspection included a review of the licensee's implementation of CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Scope included operations, maintenance, engineering, and QA. One violation was issued.	SSCs	Maintenance of heat exchangers, and screens
REGULATION	NRC INSP. REP. 50-382/97-01	Maintenance Rule Inspection at Waterford Steam Electric Station, Unit 3	3	1997	NRC	This inspection included a review of the licensee's implementation of CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Scope included operations, maintenance, and engineering. Two violations were issued.	SSCs	None identified
REGULATION	NRC INSP. REP. 50-397/96-18	Maintenance Rule Inspection at Washington Nuclear Project 2	1	1997	NRC	This inspection included a review of the licensee's implementation of CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Scope included operations, maintenance, and engineering. Two violations of NRC requirements were issued.	SSCs	None identified
REGULATION	NRC INSP. REP. 50-413/97-01 and 50-414/97/01	Maintenance Rule Inspection at Catawba Nuclear Station, Units 1 and 2	3	1997	NRC	This inspection included a review of the licensee's implementation of CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Scope included operations, maintenance , and engineering. Two violations were issued.	SSCs	Maintenance of filters-charcoal-HVAC
REGULATION	NRC INSP. REP. 50-416/97-01	Maintenance Rule Inspection at Grand Gulf Nuclear Station	4	1997	NRC	This inspection included a review of the licensee's implementation of CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Scope included operations, maintenance, and engineering. One violation of NRC requirements was issued.	SSCs	None identified

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REGULATION	NRC INSP. REP. 50-443/97-80	Maintenance Rule Inspection at Seabrook Station	3	1997	NRC	This inspection included a review of the licensee's implementation of CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Scope included maintenance. Two violations were issued.	SSCs	None identified
REGULATION	NRC INSP. REP. 50-518/96-09, 50-529/96-09 and 50-530/96-09	Maintenance Rule Inspection at Palo Verde Nuclear Generation Station, Units 1 , 2 and 3	8	1996	NRC	This inspection included a review of the licensee's implementation of CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Scope included operations, maintenance, and engineering. One unresolved item was identified.	SSCs	Maintenance of exchangers-steam generator-tubes
REGULATION	NRC Regulatory Guide 1.127	Inspection of Water-Control Structures Associated with Nuclear Power Plants	3	1978	NRC	This guide describes a basis acceptable to the NRC staff for developing an appropriate inservice inspection and surveillance program	Water-control structures	Inspection and surveillance of dams, canals, and intake structures
REGULATION	NRC Regulatory Guide 1.160, Rev. 1	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	1	1995	NRC	This guide discusses information provided in 10 CFR 50.65 to insure that safety-related and certain nonsafety-related structures, systems, and components are capable of performing their intended functions.	Safety and certain non-safety related SSCs	None identified
REGULATION	NRC Regulatory Guide 1.174	An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis	7	1998	NRC	This guide provides guidance on the use of PRA findings and risk insights in support of licensee requests for changes to a plant's licensing basis, as in requests for license amendments and technical specification changes under Section 50.90-92 of 10CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."	SSCs	None identified

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REGULATION	NRC Regulatory Guide 1.35, Rev. 3	Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments	7	1990	NRC	This guide describes a basis acceptable to the NRC staff for developing an appropriate inservice inspection and surveillance program for ungrouted tendons in prestressed concrete containment structures of light-water-cooled reactors.	Prestressed tendons used in concrete containments	Degradation of tendons
REGULATION	NRC Regulatory Guide 1.35.1	Determining Prestressing Forces for Inspection of Prestressed Concrete Containments	7	1990	NRC	This guide expands and clarifies the NRC staff position of determining prestressing forces to be used for inservice inspection of prestressed concrete containment structures.	Prestressed tendons used in concrete containments	Degradation of tendons
REGULATION	NRC SECY-97-055	Maintenance Rule Status, Results, and Lessons Learned	3	1997	Callan, L. J. / NRC	Since the effective date of the maintenance rule in July 1996, the NRC staff has completed 18 maintenance rule baseline inspections and revised the applicable regulatory guide. This document discusses results and lessons learned from these inspections.	SSCs	None identified
REGULATION	WORKING DRAFT	Standard Review Plan for the Review of License Renewal Application for Nuclear Power Plants	9	1997	NRC	The Standard Review Plan for License Renewal is a part of a continuing regulatory framework development activity that documents current methods of review and provides a basis for orderly modifications of the review process in the future.	SSCs	Assessment of SSCs for license renewal
REPORT	ASCE Publication	Stiffness of Low Rise Reinforced Concrete Shear Walls		1994	ASCE	The purpose of this paper is to review the methods currently used to compute the in-plane stiffness of low aspect ratio reinforced concrete shear walls used in nuclear power plant structures and to recommend a position.	Concrete structures	None identified

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REPORT	BNL Technical Report A-3270-12-86	Aging and Life Extension Assessment Program (ALEAP) System Level Plan	12	1986	Fullwood, R. et al / BNL	This program presents the BNL structured approach to assessing the effects of the aging of nuclear power plant components and systems on safe operation and the extension of plant operation beyond the originally planned plant life.	Components and systems	Assessment of CRD, filters, heat exchangers, supports, and RPV for extended operation
REPORT	NISTIR 4712	Predicting the Remaining Service Life of Concrete	11	1991	Clifton, J. R. / NIST	The study presented in this report consists of two mayor activities: 1. The evaluation of models which can be used for predicting the remaining service life of concrete, and 2) The evaluation of accelerated aging techniques and test which provide data for service life models or can be used to predict the remaining service life of existing concrete.	Concrete structures	Assessment of remaining life of concrete structures
REPORT	ORNL/NRC/ LTR-90/17	Structural Aging Assessment Methodology for Concrete Structures in Nuclear Power Plants	3	1991	Hookham, C. J.	An aging assessment methodology for concrete structures was developed which consists of a procedure for categorizing and ranking the safety-related concrete structures in terms of their safety significance, environmental exposure, and subelement function.	Concrete structures	Aging assessment of concrete, steel reinforcing, prestressing, and liner plate/structural steel
REPORT	ORNL/NRC/ LTR-90/29	In-service Inspection and Structural Integrity Assessment Methods for Nuclear Power Plant Concrete Structures	9	1991	Refai, T. M., Lim, M. K. / Construction Technology Laboratories, Inc.	This document has the objective of reviewing and assessing nondestructive evaluation , sampling, and structural integrity testing techniques.	Concrete structures	The assessment of the structural integrity of concrete, steel reinforcement, and prestressing systems
REPORT	ORNL/NRC/ LTR-92/3	Structural Aging Program Technical Progress Report for Period January 1,1991 to December 31 1991	2	1992	Naus, D. J., Oland, C. B./ ORNL	The Structural Aging Program has the 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.	Concrete and concrete related Category 1 structures	Assessment of concrete, steel reinforcing, prestressing, liner plate/structural steel for continued service

TYPE	ID	TITLE	MO	YR	AUTH/ORG	DESCRIPTION	TYPES OF COMPONENTS	POTENTIAL AGING ISSUES
REPORT	ORNL/NRC/ LTR-92/4	Condition Assessment and Reliability-Based Life Prediction of Concrete Structures in Nuclear Plants	1	1992	Ellingwood, B. R., Mori, Y./ Johns Hopkins University	The research described in this report is in support of the Structural Aging Program (SAG). The goal of this task is to develop a methodology to facilitate quantitative assessment of current and future reliability and performance of concrete structures in nuclear plants.	Concrete structures	Assessment of concrete structures
REPORT	ORNL/NRC/ LTR-92/8	The Structural Materials Information Center and its Potential Applications	5	1992	Oland, C. B./ ORNL	This report presents an overview of the Structural Material Information Center where material properties are being collected and assembled into a data base. Also provided are examples of how the data base could be used to assist in performing service assessment of reinforced concrete structures or in determining structural reliability of nuclear power plant structures.	None identified	None identified
REPORT	ORNL/NRC/ LTR-93/28	Repair Materials and Techniques for Concrete Structures in Power Plants	3	1994	Krauss, P. D./ ORNL	This report discusses deterioration and repair of concrete structures in nuclear power facilities.	Concrete structures	Concrete structures
REPORT	ORNL/NRC/ LTR-94/22	Summary of Materials Contained in the Structural Materials Information Center	11	1994	Oland, C. B., Naus, D. J. / ORNL	Material properties, data, and information for 144 portland cement concrete, metallic reinforcement, prestressing tendon, structural steel, and rubber materials were collected at the Structural Material Information Center. The Structural Material Handbook is a four-volume reference document that contains the complete data base for each material. The report contains a summary of the environment-dependent property for each material.	None identified	None identified
REPORT	ORNL/NRC/ LTR-94/6	Reliability Assessment of Degrading Concrete Shear Walls	4	1994	Mory, Y., Ellingwood, B. R./ Johns Hopkins University	The evaluation of the (random) residual strength of a shear wall requires that the cumulative effect of defects in a cross section be considered. This paper discusses methods for performing this evaluation and their application to a reinforced concrete wall is illustrated.	Concrete walls	None identified

TYPE	ID	TITLE	MO	YR	AUTH/ORG	DESCRIPTION	TYPES OF COMPONENTS	POTENTIAL AGING ISSUES
REPORT	ORNL/NRC/ LTR-95/14	In-service Inspection Guidelines for Concrete Structures in Nuclear Power Plants	12	1995	Hookham, C Naus, D. J./ ORNL	This report examines available inspection methods, their applications to nuclear plant concrete structures, and guidelines for establishing acceptance criteria, inspection schedules, and inspection qualifications.	Concrete structures	Degradation of concrete, steel reinforcing, prestressing, and liner plate/structural steel
REPORT	ORNL/NRC/ LTR/13	Concrete Containment Posttensioning System Aging Study	7	1995	Hill, H.T.	This report documents a study of concrete containment posttensioning system mechanisms, examination methodology and examination results.	Concrete containment posttensioning system	Concrete containment
REPORT	PNL-SA-18407	Understanding and Managing Corrosion in Nuclear Power Plants	8	1990	Johnson, A. B. / PNL	This report defines the concept of understanding and managing corrosion and focuses on an overview of how the concept is being applied, drawing on results from the NPAR program.	SSCs	SSCs
REPORT	TR-4082-2	Aging Characteristics of Nuclear Power Plant Components	11	1993	Taylor. J. et al. / BNL	This report provides information to understand the stressors and mechanisms that can cause aging degradation. Additionally, aging management techniques, such as monitoring and maintenance practices are provided.	Piping, flanges, bolts/fasteners, supports, snubbers, heat exchangers, RPVs, concrete structures, structural steel and liners.	Aging degradation of RPV, heat exchangers, fasteners, supports, concrete structures, containments, structural steel and liners
REPORT	TR-96-07	Probabilistic Qualification of Nuclear Concrete Containment and Safety-related Structures	3	1996	Orisamo, I.R./ Martec Limited, Canada	This report presents a review of the literature pertaining to the reliability and qualification of existing nuclear containments and safety-related structures. Omissions and deficiencies of current methodologies and practices are identified.	Containments and safety-related concrete structures	Reliability assessment and qualification of containments and safety-related concrete structures

TYPE	ID	TITLE	MO	YR	AUTH/ORG	DESCRIPTION	TYPES OF COMPONENTS	POTENTIAL AGING ISSUES
REPORT	WCAP 14422, Rev. 2	License Renewal Evaluation: Aging Management for Reactor Coolant System Supports	2	1997	Westinghouse Electric Corp.	This report evaluates aging of reactor coolant system supports to ensure that their intended functions can be maintained during an extended period of operation.	Reactor coolant system supports	SCC, corrosion, neutron and thermal embrittlement, fatigue, wear, creep and concrete degradation in RCC supports.

APPENDIX C

FE ANALYSIS OF DEGRADED CONCRETE STRUCTURES:

CURRENT KNOWLEDGE AND PROSPECTS FOR THE FUTURE

OECD-NEA Workshop on
FE Analysis of Degraded Concrete Structures
October 29-30, 1998

FE Analysis of Degraded Concrete Structures :

Current Knowledge and Prospects for the Future

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Brookhaven National Laboratory
Upton, New York 11973

ABSTRACT

This paper discusses the prospects for the practical application of nonlinear FE methods to age-degraded reinforced concrete (RC) structures in the area of seismic response analysis. First, findings obtained from past field surveys and laboratory tests on age-degradation of RC structures and its impact on the seismic performance are summarized. Second, the state-of-the-art of the application of nonlinear FE analysis to RC structures under severe earthquake loadings are described with a few demonstration examples. And lastly, based on preliminary analysis examples, technical difficulties and future research needs are discussed on the application of FE analyses to degraded concrete structures.

INTRODUCTION

In the US, a large number of old nuclear power plants (NPP) exist which were designed mainly in the 1960's and 1970's. Problems associated with age-related structural degradation were reported on various structures and components (e.g., Ref. 1). The phenomena of age-related degradation of concrete structures were studied extensively in the past, including degraded structures in operating NPP's (e.g., Ref. 2). Concerns were raised in the technical community regarding the potential effects of age-related degradation on the seismic performance of concrete structures in such old NPP's. Although the results of phenomenological studies on the aging process of RC structures, such as the effects of carbonation and chemical attacks on the progress of corrosion of reinforcement, are extensively available in open publication, the effects of observed degradation on the seismic performance of RC structures are much less understood.

This paper presents an overview of the past experimental studies on the seismic response of degraded RC components based on the recent literature survey of Japanese documents in the related area.

Next, the recent rapid progress in application of nonlinear FE analysis to RC structures to simulate the complex hysteretic responses under earthquake loading is described for a possible application to degraded RC structures. The attempts to apply nonlinear FE analysis to RC structures started in the early 1970's. Significant progress was made during the first two decades, but the application to dynamic/cyclic problems was seriously limited due mainly to the lack of robust analysis tools. The recent rapid progress in this area may be attributed largely to the improvement in computer hardware. This paper discusses key elements of the recent development in the area of material constitutive models with application examples of shear walls.

Lastly, the technical problems associated with the application of nonlinear FE analysis to degraded RC structures are discussed using an analysis example of degraded shear walls. Attempts were made to model the cracks caused by the corrosion of rebars by using a discrete crack model. The cracks induced by the lateral seismic loads are modeled by the rotating sheared crack model, and then superimposed on the existing cracks. The presented results are considered to be partially successful. The areas for further improvement are singled out based on the application example.

MECHANISM OF AGING DEGRADATION PROCESS

The mechanism of a typical aging degradation process of RC structures is conceptually illustrated in Fig.1. The degradation process is divided into four stages, i.e., the incubation period (Stage I), the corrosion progress period (Stage II), the crack propagation period (Stage III), and the structural deterioration period (Stage IV). During the incubation period, chloride penetration and/or carbonation of concrete progress; the length of this period is largely a function of the thickness of the cover concrete, the types of finishing material and the chloride diffusion rate.

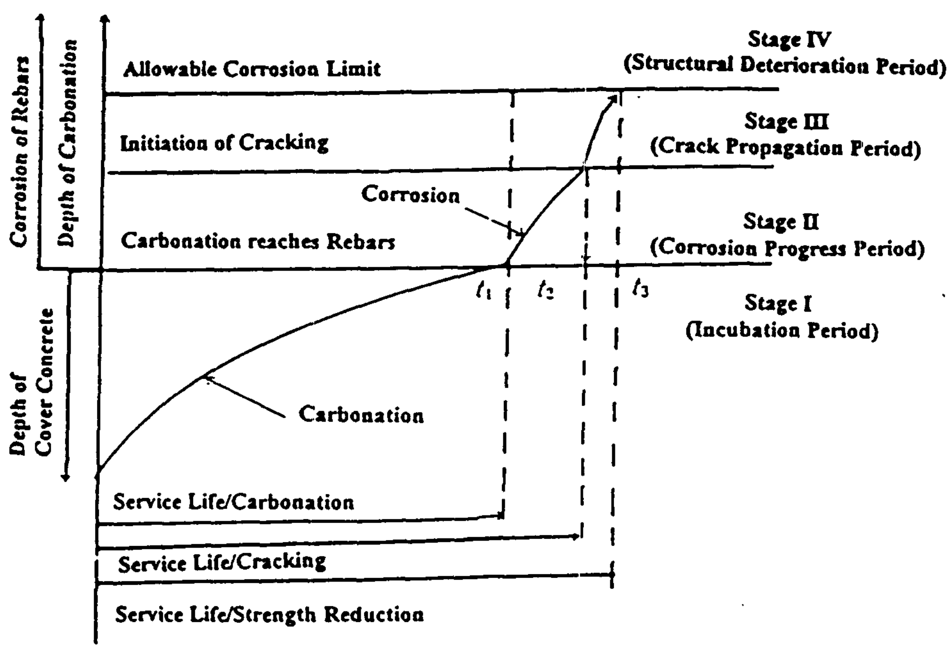


Fig. 1: Mechanism of Typical Aging Degradation Process of RC Structures (Ref. 3)

As the front of the chloride penetration/carbonation reaches the embedded rebars, the corrosion of rebars starts to progress (Stage II), given sufficient supply of water and oxygen. Fig. 2 shows the relationship between the observed corrosion grades in existing buildings and the distance of the carbonation front from the rebar surface (Ref. 4). It can be observed that the corrosion of embedded reinforcement would start before the carbonation front reaches the rebar surface. Also, there is a clear difference in the growth of corrosion between exterior and interior surfaces. At an interior surface, corrosion tends to grow slowly even after the carbonation front has reached the rebar surface; whereas at an exterior surface the corrosion growth is accelerated once the front has reached the rebar.

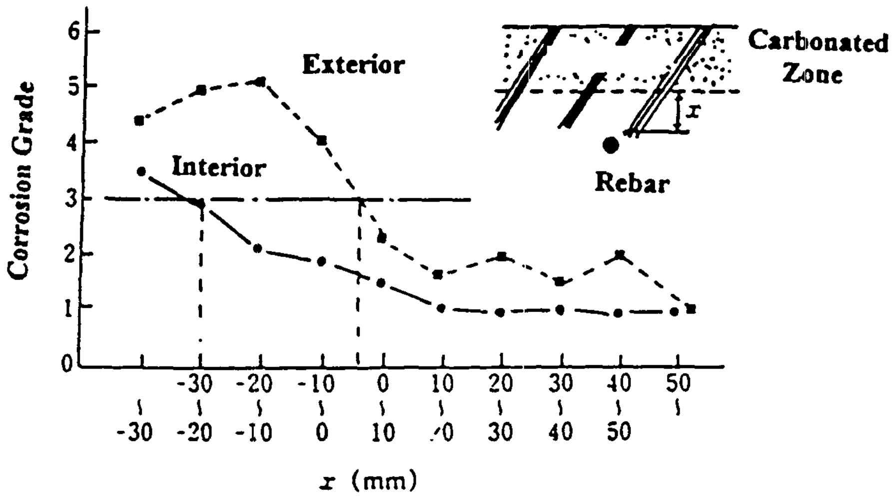


Fig. 2: Distance Between Carbonation Front from Rebar vs. Corrosion Grade (Ref.4)

While the corrosion of embedded reinforcement would cause cracking of surrounding concrete (Stage III), existing cracks (e.g., due to shrinkage) tend to accelerate the growth of corrosion of reinforcement. According to past surveys of old RC buildings and laboratory tests, there exists a clear correlation between the three degradation quantities, i.e., the surface crack width, carbonation depth and corrosion of reinforcement. Such examples are shown in Figs. 3 and 4.

As the aging degradation progresses further, serious structural deterioration may start to develop, such as spalling of cover concrete and delamination of components if proper repair work is not applied (Stage IV). In most laboratory loading tests on degraded RC components conducted in the past, specimens were typically degraded (either naturally or artificially) up to the foregoing Stage III, although the boundaries between the four degradation stages are rather fuzzy.

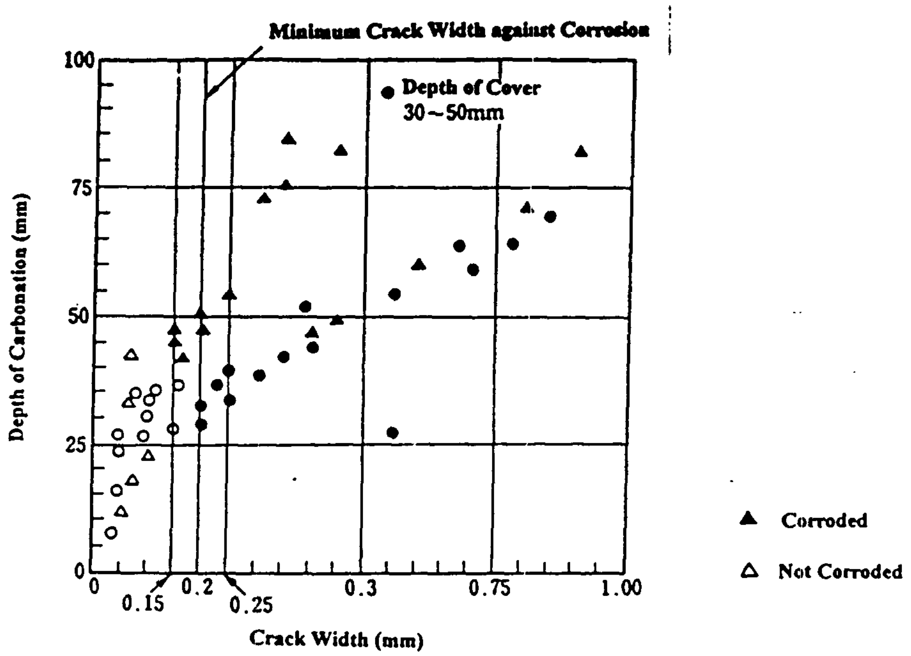


Fig. 3: Observed relationship Between Crack Width, Carbonation Depth and Corrosion of Rebars in 60 Year Old Building (Ref.5)

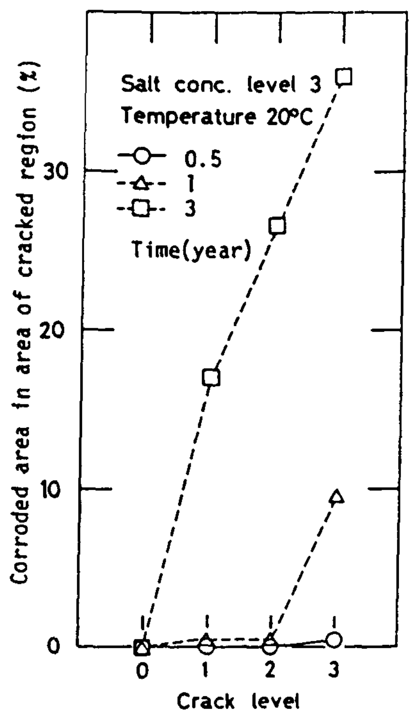


Fig. 4: Effects of Crack Width (Level 0 = no crack; 1 = micro crack; 2 = 0.03 mm crack; 3 = 0.06 mm crack) on Corrosion Progress Observed in Laboratory Tests (Ref.6)

SEISMIC PERFORMANCE OF DEGRADED RC COMPONENTS

In this paper, the degradation associated with excessive cracks caused primarily by corrosion of rebars and alkali-silica reaction is considered with regard to its effects on seismic performance. Other forms of degradation, such as leaching and abrasion, are not within the scope of the study described in this paper. In typical laboratory loading tests of degraded RC components, test specimens are either naturally or artificially degraded prior to the application of cyclic loading. The methods for accelerated artificial degradation include the accelerated corrosion of rebars by spraying salt-water and/or by means of electrolytic corrosion; and the pre-cracking of the concrete by applying low-level loading, inserting a cracking agent into drilled holes, and by the use of alkali-silica reaction. It appears that most of the past seismic loading tests available in open publication were conducted in Japan (Ref. 3). The following is the summary of key findings on the effects of degradation on seismic performance.

Effects of Corrosion..........The corrosion of reinforcing steel is considered to affect the seismic performance in the following three different ways:

(1) Reduction of load-carrying capacity due to the loss of cross-section of reinforcement;

(2) Degradation of bond capacity; and

(3) By causing cracks in the surrounding concrete.

Regarding the above first item (1), attempts were made in the past to predict the strength reduction as a function of the amount of the loss of cross section of rebars based on empirical strength equations or simplified structural models. Such an example is shown in Fig. 5. It should be noted that such calculations have not been experimentally verified, and may represent an overly conservative prediction of the effects of degradation.

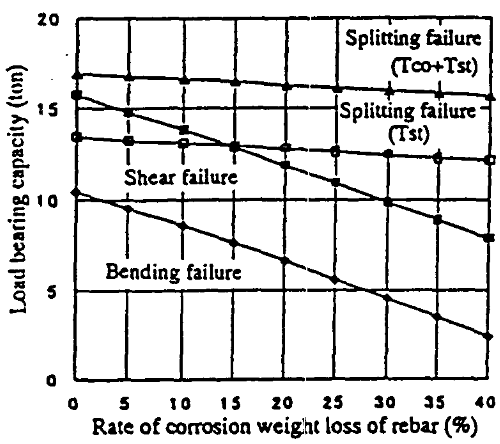


Fig. 5: Example of Calculation Result for the Losses in the Load-Carrying Capacity of RC Beam (Ref.7)

According to past pullout tests of corroded rebars (e.g., Ref. 8, 9), the bond-slippage relationship tends to be improved when the degree of corrosion is moderate. Fig. 6 summarizes the bending test results of slender beams reinforced with rebars corroded at various degrees (Ref. 7). The bending strength is normalized by the calculated value, and plotted against the corroded area ratio (the ratio of corroded surface area divided by the total surface area). As the amount of corrosion increases, the bending capacity tends to increase; this tendency is peaked at about 20% of the corroded area ratio. When the degree of corrosion exceeds this value, the bond is deteriorated and cover concrete is eventually spalled off. The bending capacity of a RC component, however, is not significantly affected by this type of degradation. The loss of bond would reduce the shear strength capacity, but tend to increase the ductility (e.g., Ref. 11). The degradation of bond would also cause a "pinching" in hysteresis loops (Ref. 12).

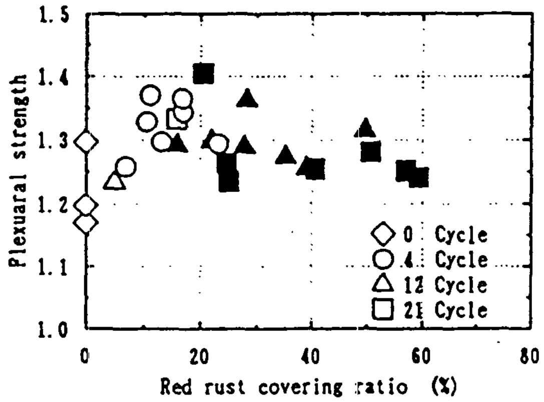


Fig. 6: Normalized Bending Strength (Normalized by Calculated Strength) as a Function of Amount of Corrosion (Ref.10)

Effects of Alkali-Silica Reaction..........The alkali-silica reaction is known to cause a significant degradation of concrete. The excessive cracking caused by the alkali-silica reaction is the major concern with respect to the seismic capacity of RC structures. A large number of seismic loading tests were performed mainly on beams to study the effects of such cracking (e.g., Refs. 11, 13, 14 and 15). The observed effects on the seismic performance indicate the significance of the location/orientation of cracks with respect to the applied seismic stresses. The following are some key findings from the past loading tests of pre-cracked beams:

(1) Vertical cracks (normal to member axis) significantly reduce the bending stiffness. But the reduction in bending strength due to existing vertical cracks is either negligible or up to 10-20%.

(2) Vertical cracks, in general, do not affect shear strength, except when they are located at the compression failure zone. A vertical crack at the compression failure zone would cause a sliding shear failure, which tends to reduce both strength and ductility.

(3) **Horizontal cracks (along component axis) affect the shear strength more than the bending strength. They may cause a horizontal sliding shear failure, which may significantly reduce the loading capacity.**

(4) Cracks would affect the shear capacity if the crack pattern coincides with that of the cracks caused by applied seismic loads or when they alter the failure mode (e.g., by connecting diagonal cracks caused by seismic loads).

<u>Degraded Shear Walls and Columns</u>..........A large number of degraded RC shear walls and columns are being tested at the Ryukyu University in Japan (e.g., Refs. 16, 17, 18 and 19). Recent test results were also provided by Prof. Yamakawa of the Ryukyu University to the writer. Specimens were degraded by mixing salt in the concrete, and then the corrosion of the reinforcement was further accelerated by using the electrolytic corrosion method or by exposing specimens to oceanic environment for several years. Fig. 7 shows examples of the comparisons of seismic responses between non-degraded and degraded RC components. Some key findings observed so far from the series of tests are briefly described below:

(1) In all the test cases of shear walls and columns, the ultimate strength of the degraded specimens were not significantly reduced in comparison with the non-degraded specimens, as well as the calculated (by fiber model) results.

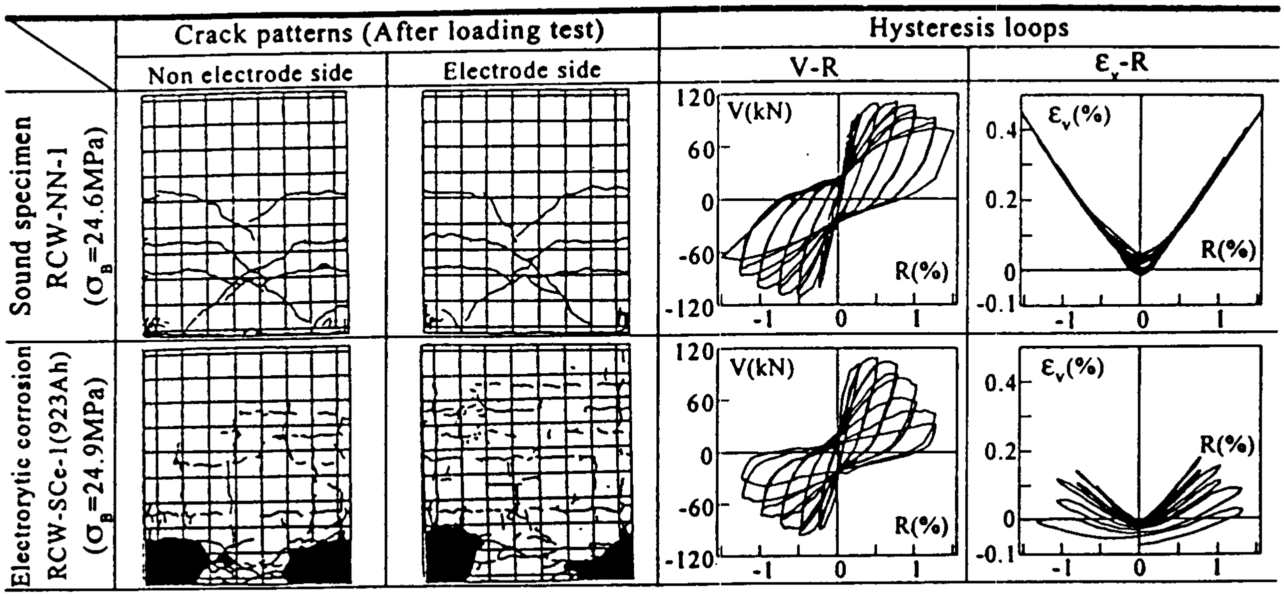
(2) In the shear wall tests (e.g., Fig. 7-a), the stiffness of the degraded specimens were higher than those of non-degraded specimens due to the improvement of bond capacity caused by corrosion. The ductility of the degraded specimens at a large plastic deformation, however, tends to be reduced due to spalling of cover concrete.

(3) A large number of cracks (surface crack width was 0.15~0.8 mm) were formed in the degraded shear walls due to the corrosion of rebars. The crack pattern of the degraded specimens due to the applied seismic forces was significantly different from that of the non-degraded specimens (see Fig. 7-a) because the critical loading paths were altered by the existing cracks.

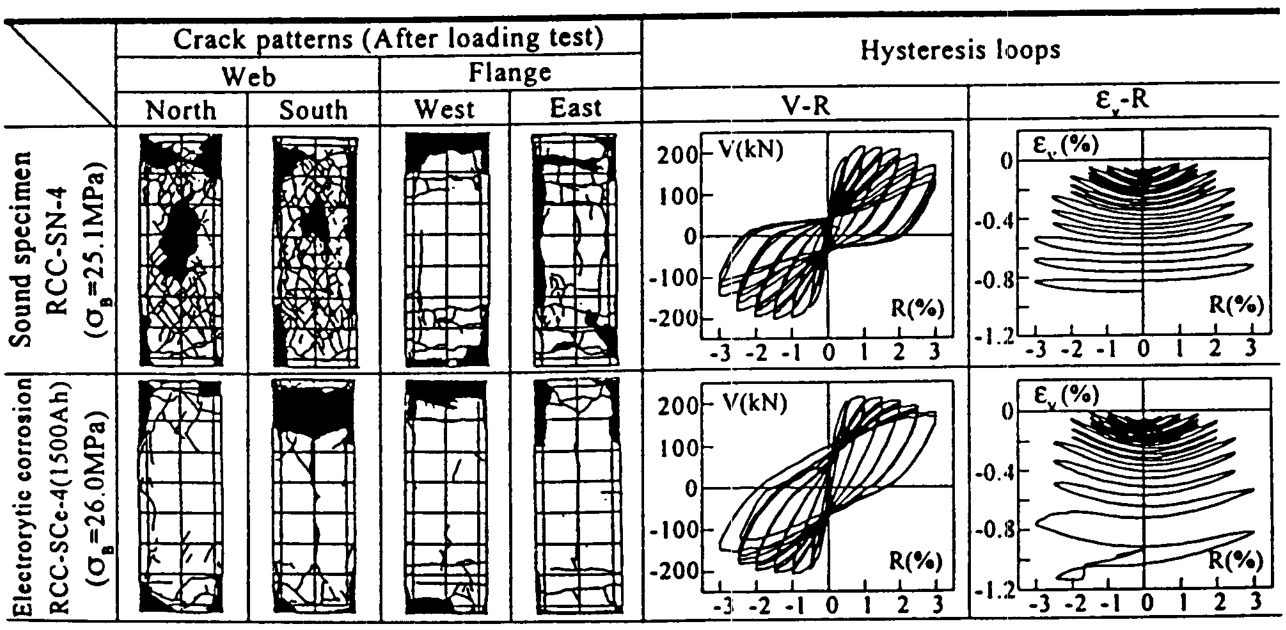
(4) In the degraded column tests by using the electrolytic corrosion method, the equivalent yield stress of the main rebars was reduced by 7~24%, and the total weight by 6~11% due to corrosion. However, the observed seismic performance was either not affected or rather improved by the degradation. The enhanced bond capacity and the increase in the confinement pressure both caused by the corrosion of rebars, are attributed to the observed phenomena (Ref. 19).

(5) In the degraded column tests by exposing the specimens to oceanic environment for several years, the equivalent yield stress of the main rebars was reduced by 9~11%, and the total weight by about 3% due to corrosion. The crack widths were larger than those of the foregoing electrolytic cases, although the degree of corrosion was

lower. Regarding the seismic performance, similar trends as described above were observed. However, the specimens tend to fail in fracture of main reinforcements at the reduced cross-section locations at large deformation, whereas the buckling of main reinforcements was the observed failure mode of the non-degraded cases.



a: Shear Walls (Top - Non-Degraded; Bottom - Degraded)



b: Columns (Top - Non-Degraded; Bottom - Degraded)

Fig. 7: Comparison of Seismic Responses between Non-Degraded and Degraded RC Components (Ref.18, Courtesy of Prof. Yamakawa of Ryukyu Univ.)

NONLINEAR FEM ANALYSIS OF RC STRUCTURES

Significant progress has been made in the last several years on the application of nonlinear FE analysis to RC structures to simulate complex hysteretic behavior under severe earthquake loading. The recent progress in this area was described in some detail by the writer (Ref.20 and 21). In this paper, key elements of constitutive models are discussed based on an application example to a typical shear wall test (non-degraded). The described analysis methods are utilized in the application to degraded RC components described later in this paper. The overall analysis approach is briefly outlined below:

- The nonlinearity of concrete is modeled based on the orthotropic plasticity theory with the equivalent uniaxial assumption. The hysteretic model for the uniaxial behavior of concrete is illustrated in Figs. 8 and 9.
- The double-rotating smeared crack model is used for the 2-D problems. The two principal stress-strain directions are rotated independently until cracks are formed.
- To solve the nonlinear problems, the modified Newton-Raphson and initial stiffness methods were combined in an iteration scheme to minimize the unbalanced forces (Ref. 21). The nonlinear FEM code, ISSAC (Ref.22), developed by the writer, was used for all the analysis examples described in this paper.

In this paper, the results of a parametric study are presented for (1) negative slope of the concrete stress-strain curve, (2) compressive strength reduction factor for cracked concrete, and (3) modeling of shear transfer across the cracks. Fig. 10 shows the empirical factor for cracked concrete proposed by researchers (Ref. 21). In this figure, the equation "envelope" was determined to envelope all the empirical curves, and used in the "baseline case" analysis described below.

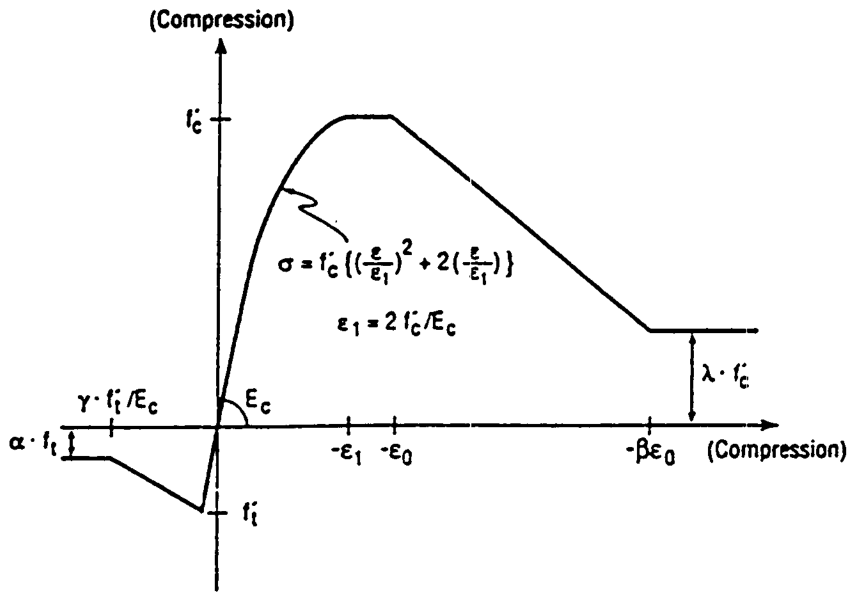


Fig. 8: Envelope Curve for Concrete

Fig. 11 illustrates the hysteretic model for the shear transfer across the cracks. The shear strength is continuously reduced as the tension strain normal to the crack surface, ϵ_t, increases as indicated in Fig. 11. Two other simpler methods, that are used frequently in the past studies, are also examined in this paper.

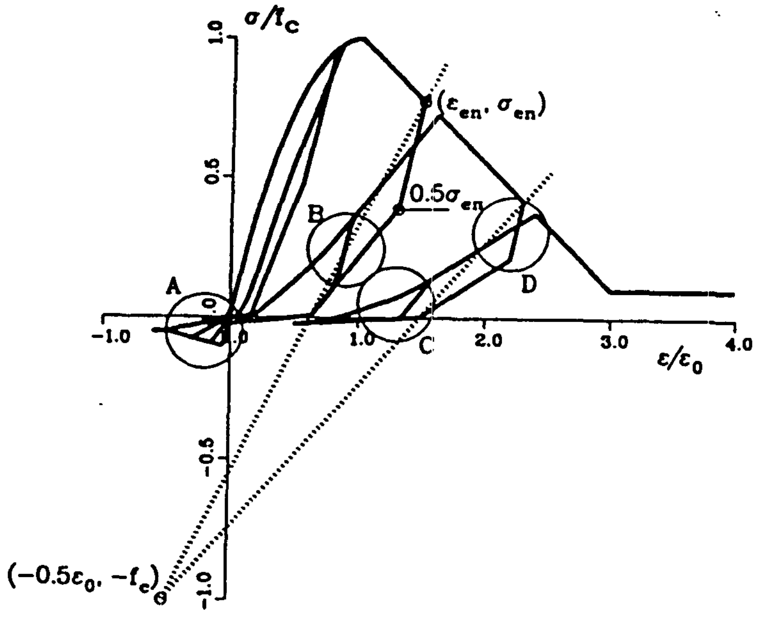


Fig. 9: Uniaxial Hysteretic Model for Concrete

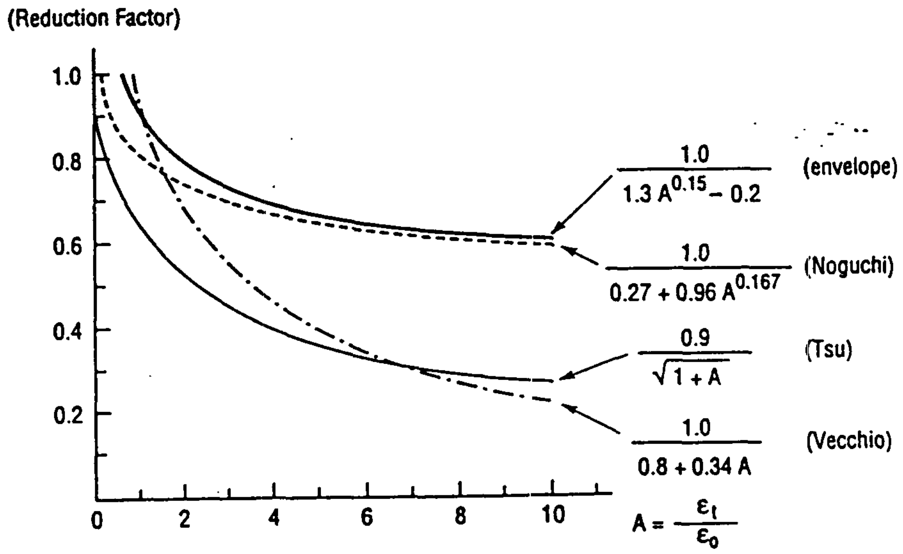


Fig. 10: Compressive Strength Reduction Factors for Cracked Concrete

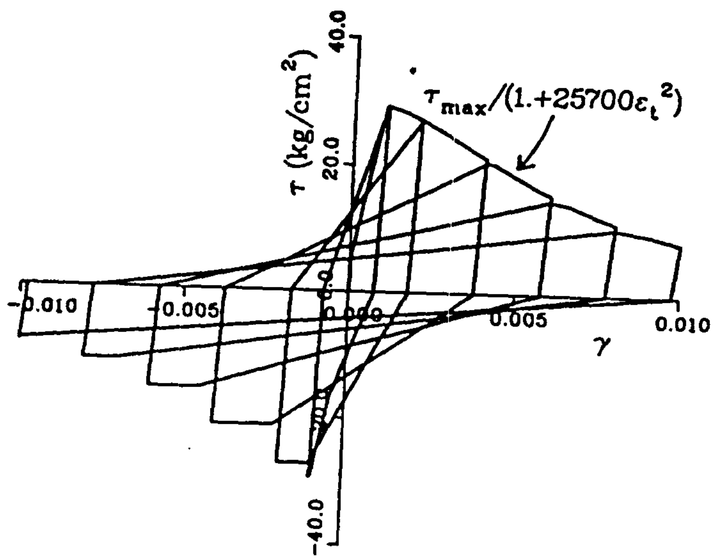


Fig. 11: Hysteretic Model for Shear Transfer Across the Cracks

Among simpler models, the following Yamada-Aoyagi model is one of the most popular approaches due to the ease in programming implementation. The shear modulus, G, is expressed as a function of the elastic shear modulus G_0, and the maximum tensile strain ϵ_{max}, as

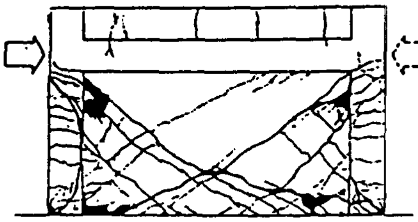
$$G(kg/cm^2) = \frac{1}{\frac{1}{G_0} + \frac{\varepsilon_{max}}{36}} \qquad (1)$$

Another popular simplified approach is to express the inelastic shear modulus as a function of the equivalent uniaxial moduli of concrete, E_i, and Poisson's ratio v, based on the orthotropic plasticity theory. For a biaxial stress condition the shear modulus may be expressed as,

$$G = \frac{1}{4(1+\upsilon)}(E_1 + E_2) \qquad (2)$$

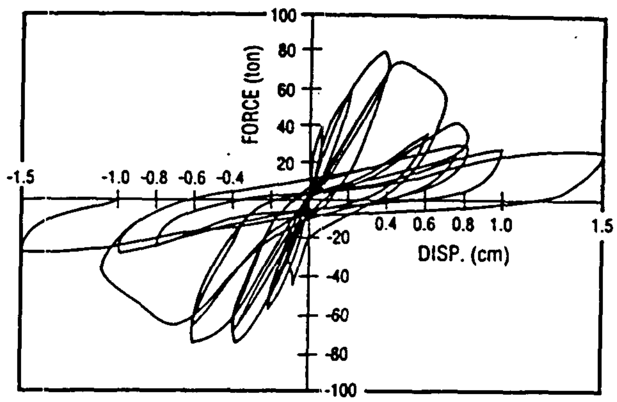
Fig. 12 shows the shear wall (Ref.23) used for the parametric study. The test specimen (non-degraded) is an in-fill wall surrounded by edge columns and a beam. As indicated in Fig.12-b, the specimen failed in a very brittle diagonal shear failure soon after reaching the maximum strength point.

An FE model was made using 2-D 8-node isoparametric elements (4-Gauss points) as illustrated in Fig.13-a. The calculated load-deformation relationship for the "baseline case" is shown in Fig. 13-b, where, a negative slope factor of β = 4.0 was assumed (see Fig.8) to define the stress-strain relationship of concrete.



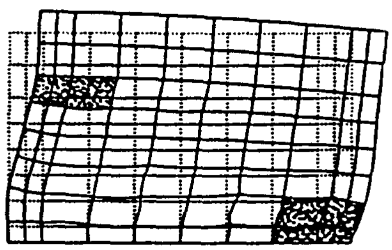
Hazama Wall

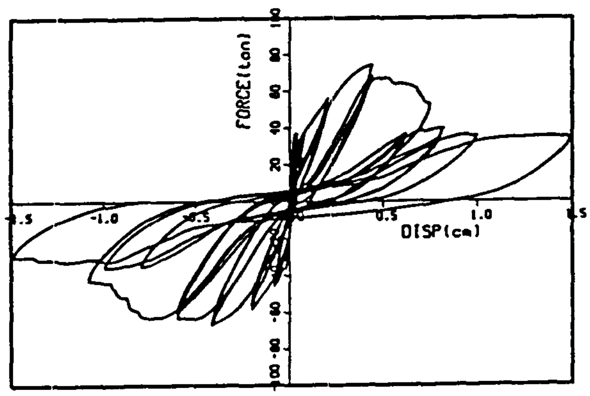
$f'_c = 344\ kg/cm^2$
$\delta_n/f'_c = 0.09$
$\rho = 0.2\%$ (*wall*)
2.0% (*columns*)



(a) Final Failure Mode (b) Load-Deformation Relationship

Fig. 12: Example of Shear Wall Test Used for Parametric Study (Ref.23)





(a) Analysis Model and Failure Mode
(shaded elements represent crushed concrete)

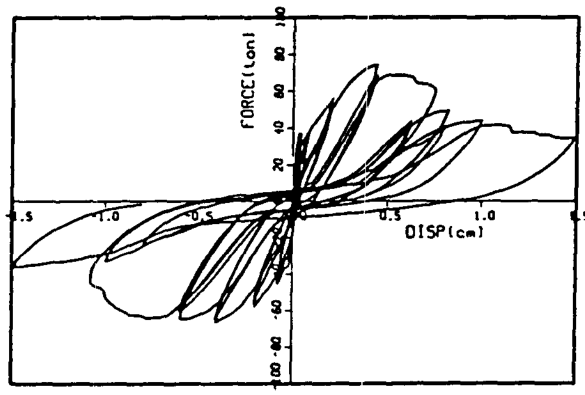
(b) Calculated Load-Deformation Relationship

Fig. 13: Analysis Results of "Baseline Case"

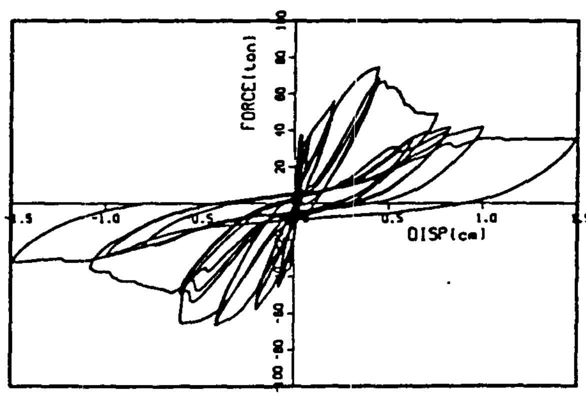
The assumed negative slope factor, β, appears to affect the post-peak behavior as illustrated in Fig.14. By assuming β to be 2.0 (i.e., the crushing strain is only two times larger than the maximum point strain ϵ_0), the calculated solution became somehow unstable due to the significantly steep negative stiffness.

The effects of using different equations for the compressive strength reduction for cracked concrete are illustrated in Fig.15. Based on the calculated results, it seems most existing empirical equations tend to overestimate the strength reduction for this particular case.

The calculated load-deformation relationships, by using the foregoing simpler models for the shear transfer across the cracks, are given in Fig.16. The calculated example may indicate that the reduction in shear transfer across the crack surface (see Fig.11) is a significant factor in order to accurately predict the brittle shear failure under large plastic deformation reversals.

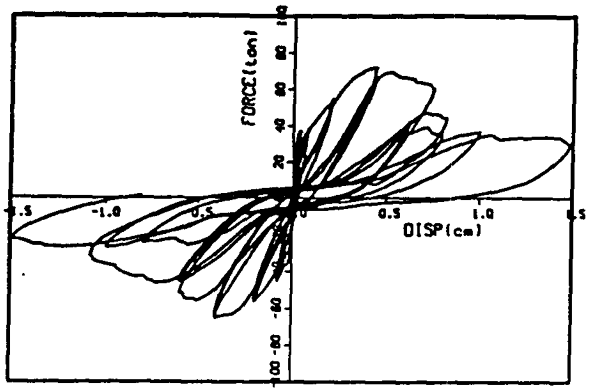


(a) Moderate Negative Slope (β = 6)

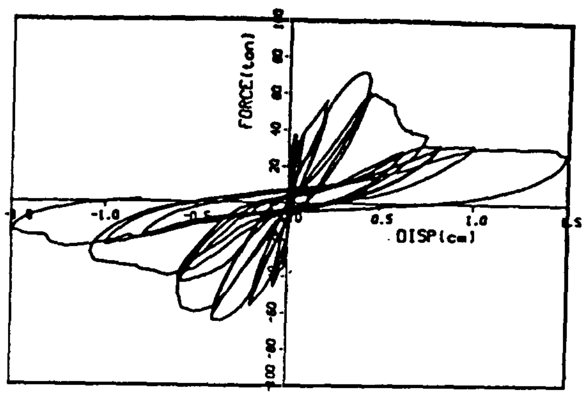


(b) Steep Negative Slope (β = 2)

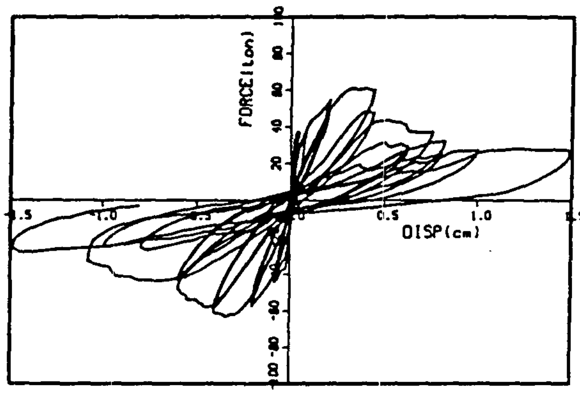
Fig. 14: Effects of Negative Slope of Stress-Strain Curve



(a) Noguchi's Equation

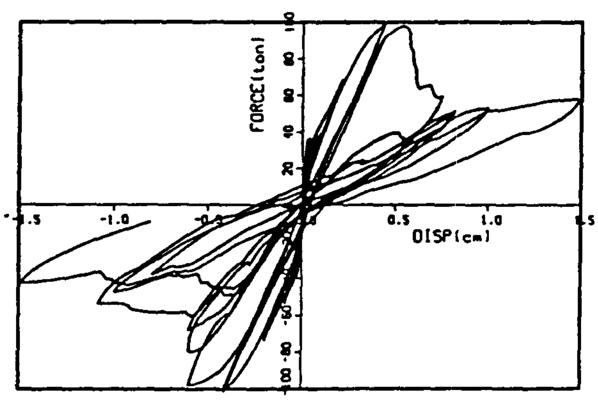


(b) Vecchio's Equation

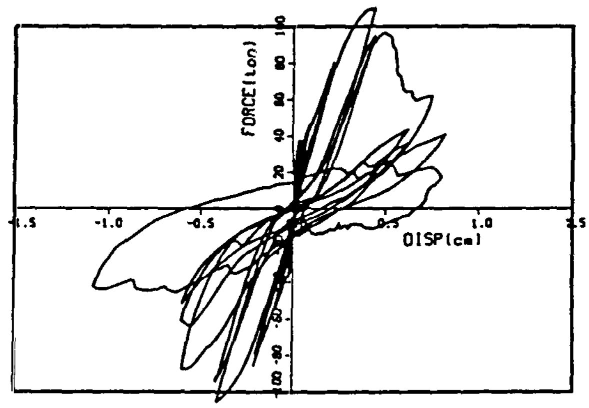


(c) Tsu's Equation

Fig. 15: Effects of Different Equations for Compressive Reduction Factor



(a) Yamada-Aoyagi Model



(b) $G = \dfrac{1}{4(1+\upsilon)}(E_1 + E_2)$

Fig. 16: Effects of Different Models for Shear Transfer Across Cracks

PRELIMINARY ANALYSIS OF DEGRADED SHEAR WALL

The effects of the aging-related degradation on the seismic performance are not well understood at this point because the past experimental work in this area is rather limited. To understand the mechanism of the interaction between the aging degradation and seismic responses, as well as to possibly quantify the effects of aging degradation, the application of the foregoing nonlinear FEM approach is attempted. Based on the observation of the available test results described earlier, the following two features are considered to be important elements of degradation, and addressed in this paper:

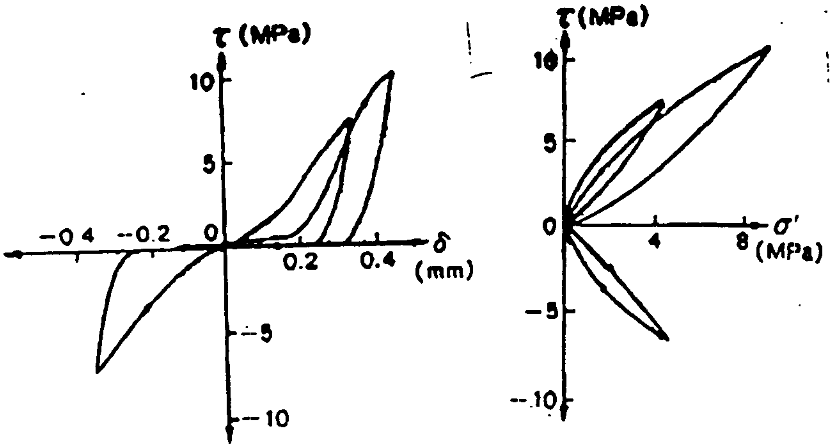
- age-related cracks; and
- changes in bond-slippage relationship.

To model the existing pre-cracks, such as those caused by corrosion of rebars and alkali-silica reaction, the use of the smeared crack models is considered to be inappropriate because arbitrary external forces need to be applied on the analysis model to cause cracking. A discrete crack model is used to simulate existing cracks. The new cracks induced by the seismic loading are then superimposed onto the existing cracks using the smeared crack model.

Fig. 17 shows a typical test result to simulate the aggregate interlocking behavior (Ref.24). In this test, a concrete block was split in half to produce a natural crack surface, and lateral (τ) as well as normal (σ) stresses were applied on the crack surface to cause cyclic shear deformation (δ) under a constant crack width (w). According to Li and Maekawa (Ref.24), the skeleton curve for the shear stress, τ, is determined as,

$$\tau\,(MPa) = 3.83\, f_c'^{1/3}\, \frac{(\delta/w)^2}{1+(\delta/w)^2} \qquad (3)$$

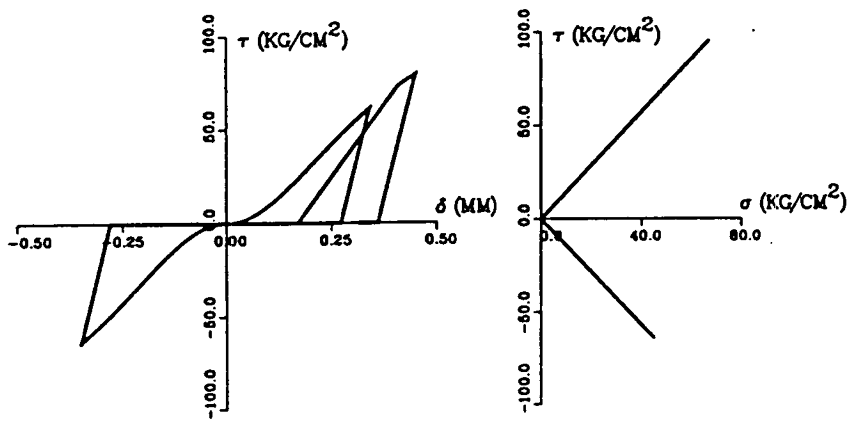
To reproduce the above nonlinear behavior in an FEM format, the aggregate interlocking is represented by a pair of truss elements with a 35° angle to the crack surface. In this modeling, the ratio of the compressive normal stress to the shear stress is kept constant at tan35° (= 0.70). The calculated responses using this model are given in Fig. 18.



(a) Shear stress (τ) vs. Shear Deformation (δ) Relationship

(b) Shear Stress (τ) vs. Normal Stress (σ) Relationship

Fig. 17: Shear Stress due to Aggregate Interlocking under Constant Crack Width, w = 0.3mm (Ref.24)



(a) Shear Stress vs. Shear Deformation Relationship

(b) Shear Stress vs. Normal Stress Relationship

Fig. 18: Calculated Response for Test Results of Fig. 17.

Next, to model the additional resistance/deformation of the crack surface due to the bond/slippage of reinforcement, the writer's earlier study (Ref.25) is utilized. Fig. 19 shows a re-plotting of existing pullout test results in terms of the stress of rebars, σ, and the normalized slippage at the point of pullout, S_n. In which, the normalized slippage is the ratio of slippage, S, divided by the diameter of rebar, D (i.e., $S_n = S/D$). Based on similar plots of many other pullout test data, the following empirical relationship was obtained (Ref.25):

$$\sigma(ksi) = 771 \cdot \tau_m^{1/2} \cdot S_n^{2/3} \qquad (4)$$

The bond strength, τ_m, was assumed to be 1.5 *ksi* for the bottom rebars (strong bond) and 0.9 *ksi* for the top rebars (weak bond). This relationship is used to determine the additional nonlinear spring for the bond-slippage relationship across the cracks.

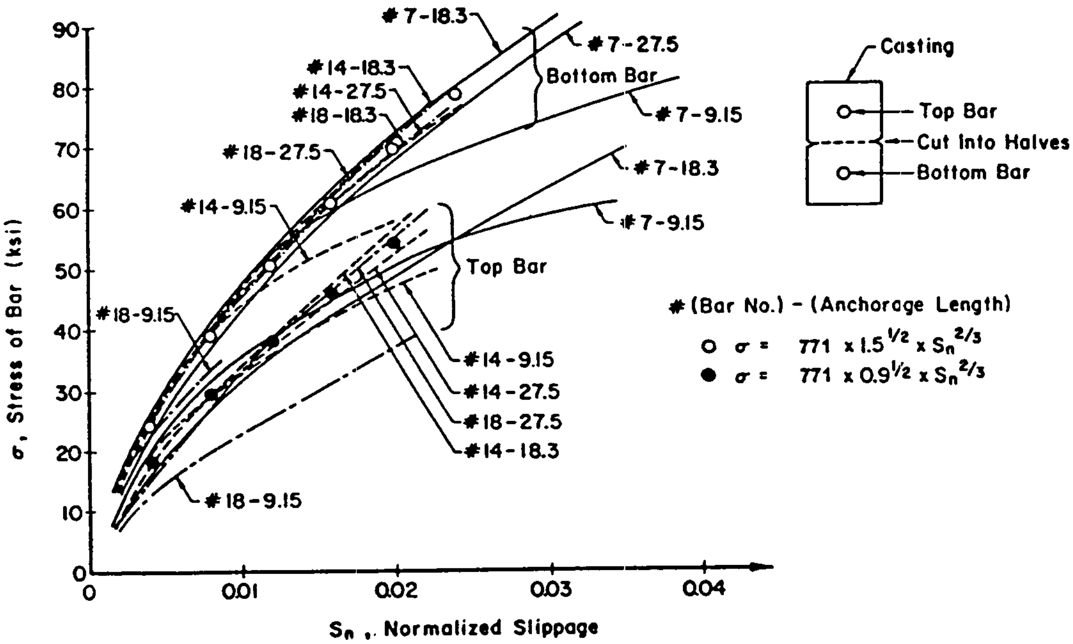


Fig. 19: Normalized Bond-Slippage Relationship

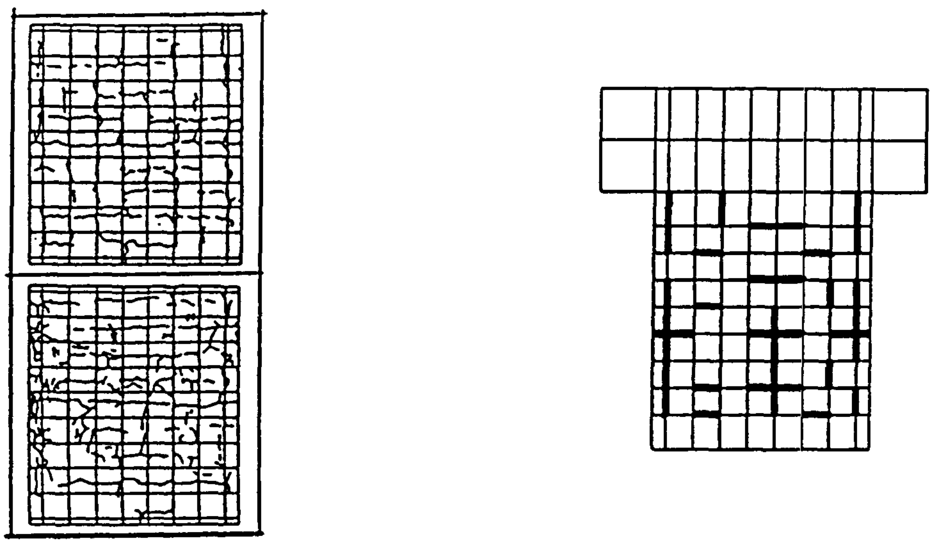
To complete the modeling of the discrete crack behavior, the additional shear resistance due to the dowel action needs to be considered. Based on the analyses of the available test data (Ref.26), the stiffness and the shear strength of the dowel action were determined as,

$$k = \frac{\rho E_c}{2(1+\upsilon)D} \qquad (5)$$

$$\tau_{max} = 5.5\,\rho f_c' \qquad (6)$$

in which, ρ is the steel ratio; v is the Poisson's ratio; D is the diameter of rebars; and f_c' is the concrete strength. It should be pointed out that for typical shear walls, the above additional shear resistance due to the dowel action is negligible in comparison with the shear stress due to the aggregate interlocking.

One of the degraded shear walls, SCe-2, tested by the Ryukyu University (Ref.16), is analyzed here. This shear wall is very similar to the one shown in Fig.7-a, except the degree of the degradation due to corrosion is slightly higher than the case shown in Fig. 7-a. Fig.20 shows the observed surface cracks due to corrosion, and the assumed locations of discrete crack elements. Although the measured crack widths ranged between 0.15mm to 0.8mm at the surface, a constant crack width of 0.1mm was assumed for the analysis model because the described discrete crack model can account for only through-wall cracks. Later, a parametric study is conducted to see the effects of varying assumed crack widths. For the bond-slippage relationship, a bond strength of τ_m = 0.9 ksi was assumed for the non-degradation case, and a 50% higher value for degraded cases to account for the improvement of bond capacity due to corrosion.



(a) Observed Cracks on Both Sides due to Degradation (Specimen SCe-2, Ref.16)

(b) Assumed Locations of Discrete Cracks in FEM

Fig. 20: Modeling of Cracks due to Corrosion by using Discrete Crack Model

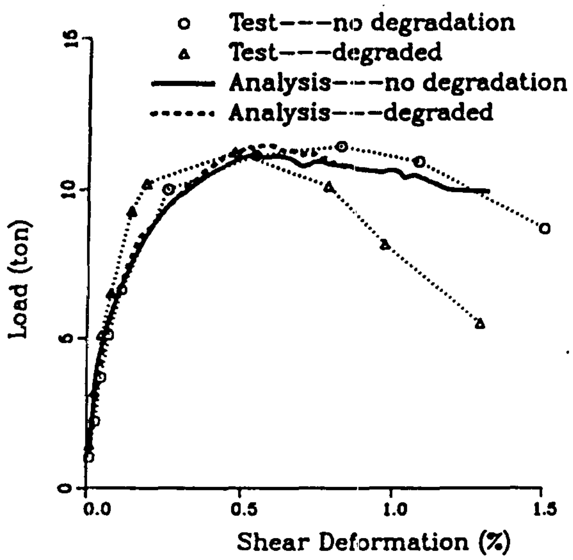
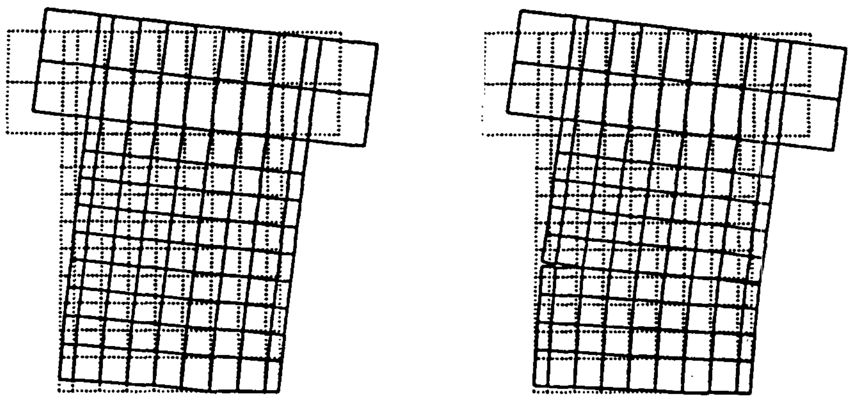
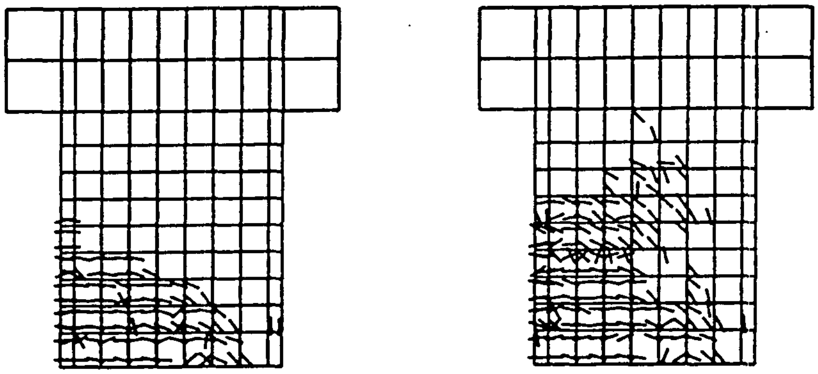


Fig. 21: Comparison of Load-Deformation Relationships



a: Without Degradation Cracks b: With Degradation Cracks

Fig. 22: Calculated Deformation at the Maximum Point



(a) Without Degradation Cracks (b) With Degradation Cracks

Fig. 23: Calculated Cracks by Smeared Crack Models due to Lateral Loads

Figs. 21 through 23 show the analysis results both for non-degraded and degraded cases. It appears that the uplifting due to the bond-slippage at the base significantly contributes to the deformation characteristics of this shear wall as illustrated in Fig.22-a. For the non-degraded case, both the load-deformation relationship and the crack pattern correlate well with the test results (see also Fig.7-a). For the degraded case, the observed test results indicate a higher stiffness than the non-degraded specimen up to the maximum point despite a large number of cracks due to corrosion (see broken lines in Fig.21). The calculated load-deformation relationship for the degraded case, however, is almost identical with that of the non-degraded case. The significant difference in the crack pattern between the degraded and non-degraded cases, as shown in Fig.7-a, seems to be reproduced by the analyses as shown in Fig.23.

Figs. 24 and 25 show the results of additional analyses by varying the assumed crack width. It appears the calculated load-deformation relationships are not sensitive to the assumed crack width as indicated in Fig.24.

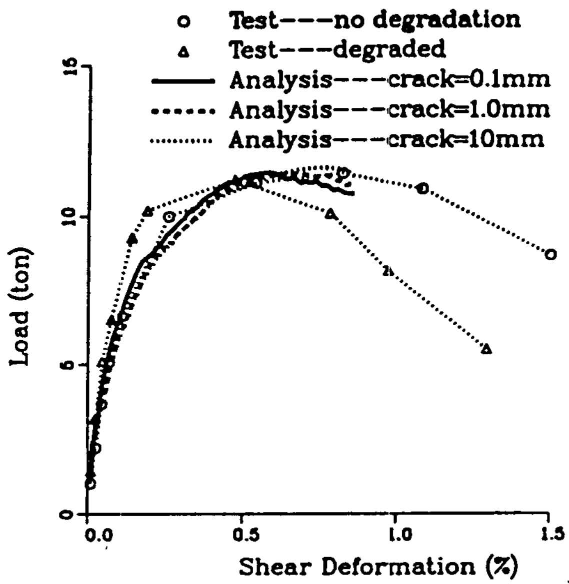
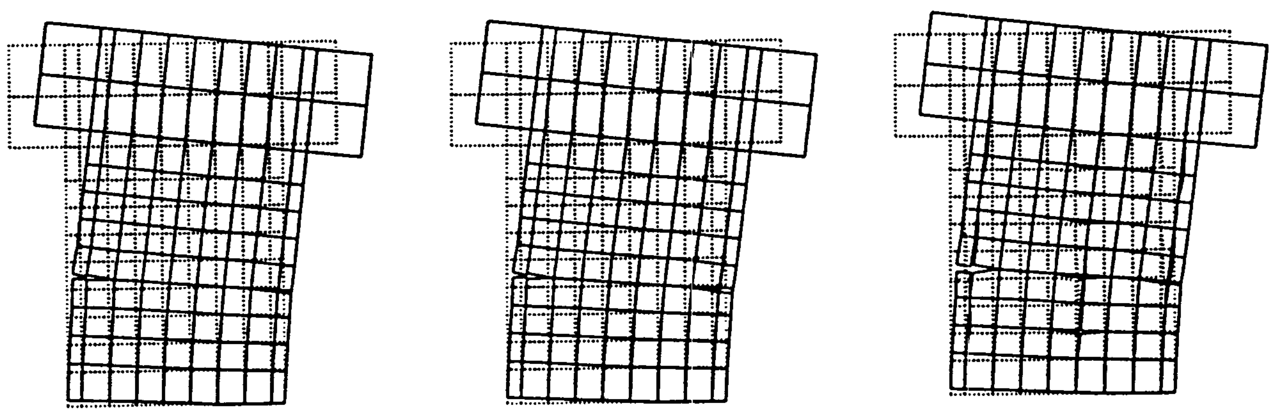


Fig. 24: Comparison of Load-Deformation Relationships by varying Crack Width



(a) w = 0.01mm (b) w = 1.0mm c: w = 10mm

Fig. 25: Comparison of Calculated Deformation by Varying Crack Width

The possible reasons for the observed differences between the test and analysis for the degraded case are:

(1) During the accelerated degradation process, the properties of the concrete may have been changed, which is not properly accounted for in the analysis;

(2) The increase in the bond capacity due to corrosion may be more than assumed in the analysis;

(3) In the analysis, all the discrete cracks are assumed to be through-wall. The actual cracks in the test specimen, however, may have been only surface cracks, although the observed crack pattern (Fig.20-a) seems to be extensive.

CONCLUSIONS

By utilizing the currently available analysis techniques, an attempt was made to reproduce the seismic behavior of degraded RC components. A set of assumptions were made to simulate the observed degradation condition. Although some aspects of the observed effects of degradation, such as the significant changes in the crack pattern due to the shift of critical loading paths, were successfully reproduced in the analysis, it appears that the analysis assumptions did not fully reflect the actual degradation conditions. Further efforts seem to be necessary, such as a more accurate characterization of the changes in material properties for a better correlation.

Based on the observations of the past seismic loading tests of degraded RC components, as well as the above preliminary application of nonlinear FE analysis, the following areas are singled out for further efforts:

(1) The discrete crack model needs to be refined and calibrated to account for the nonlinear behavior of aggregate interlocking, dowel action, and bond-slippage under cyclic loading reversals.

(2) Properly accounting for the bond mechanism seems to be a key to successfully reproduce the observed complex phenomena of degraded RC components under seismic loads (e.g., the increase in stiffness in shear wall tests and the seismic performance of columns with significantly corroded reinforcement). The modeling of bond mechanism needs to be improved.

(3) The changes in material properties of degraded RC components need to be quantified, including the compressive/tensile strength and modulus of elasticity of concrete.

As our understanding of the fundamental mechanism of the effects of age-related degradation on seismic performance progresses and the FE analysis techniques are further improved, nonlinear FE analysis will become a powerful analysis tool for the structural evaluation of degraded concrete structures.

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10. SUPPLEMENTARY NOTES

T.Y. Chang, Project Manager

11. ABSTRACT *(200 words or less)*

This report describes the results of the first phase of a multi-year research program to assess age-related degradation of structures and passive components for U.S. nuclear power plants. The purpose of this research program is to develop the technical basis for the validation and improvement of analytical methods and acceptance criteria which can be used to make risk-informed decisions and to address technical issues related to degradation of structures and passive components. The Phase I assessment of age-related degradation of structures and passive components at nuclear power plants has been completed. This assessment consisted of (1) collection of degradation occurrences, development of a computerized database, and trending analyses; (2) review of technical information related to aging degradation to identify the significant aging issues for those structures and passive components which would have the greatest impact on plant risk; and (3) performance of a scoping study to identify those structures and passive components that warrant further detailed evaluation in Phase II of this program. The scoping study concluded that the structures and passive components that warrant further detailed evaluation are masonry walls, flat bottom tanks, anchorages, concrete structures (other than containments), and buried piping.

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