CONF-9010185--3

CONF-9010185--3

DE91 001632

MANAGEMENT OF THE AGING OF CRITICAL SAFETY-RELATED CONCRETE STRUCTURES IN LIGHT-WATER REACTOR PLANTS*

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*Research sponsored by the Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission under Interagency Agreement 1886-8084-5B with the U.S. Department of Energy under Contract DE-AC05-84OR21400 with Martin Marietta Energy Systems, Inc.

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ABSTRACT

The Structural Aging Program has the overall objective of providing the USNRC with an improved basis for evaluating nuclear power plant safety-related structures for continued service. The program consists of a management task and three technical tasks: materials property data base, structural component assessment/repair technology, and quantitative methodology for continued-service determinations. Objectives, accomplishments, and planned activities under each of these tasks are presented. Major program accomplishments include development of a materials property data base for structural materials as well as an aging assessment methodology for concrete structures in nuclear power plants. Furthermore, a review and assessment of inservice inspection techniques for concrete materials and structures has been completed, and work on development of a methodology which can be used for performing current as as well as reliability-based future condition assessments of concrete structures is well under way.

1. INTRODUCTION

Aging of nuclear power plant structures, systems, and components occurs with the passage of time and has the potential, if its effects are not controlled, to increase the risk to public health and safety. Many factors complicate the contribution of aging effects to the residual life of the various safety-significant plant structures, systems, and components. Uncertainties arise due to the following:[1] (1) differences in design codes and standards for components of different vintage; (2) lack of past measurements and records; (3) limitations in the applicability of time-dependent models for quantifying the contribution of aging to overall structure, system, or component failure; and (4) inadequacy of detection, inspection, surveillance, and maintenance methods or programs.

2. BACKGROUND

Within the nuclear power industry, the aging of plant structures, systems, and components has become the subject of significant research in the last few years.[2-4] This interest is prompted by the need to quantify the effects of aging in terms of potential loss of component integrity or function and to support current or future condition assessments of critical components. Since certain concrete structures (Category I) play a vital role in the safe operation of nuclear power plants,[5-8] guidelines and criteria for use in evaluating the remaining integrity (residual life) of each structure are needed. Standardized review guidelines for near-term evaluation of operating license renewal applications may be required as early as 1991-1992 when utilities are planning to submit initial requests.

3. CATEGORY I CONCRETE STRUCTURES

3.1 General Design Requirements

Category I structures are those essential to the function of the safety-class systems and components, or that house, support, or protect safety-class systems or components, and whose failure could lead to loss of function of safety-class systems and components housed, supported, or protected. In addition, these structures may serve as barriers to the release of radioactive material and/or as biological shields. The basic laws that regulate the design (and construction) of nuclear power plants are contained in Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR 50)[9], which is clarified by Regulatory Guides, Standard Review Plans, NUREG reports, etc. "General Design Criteria" of Appendix A to 10 CFR 50 requires that structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. "General Design Criteria 2" requires that the structures important to safety be designed to withstand the effects of natural phenomena (e.g., earthquakes, tsunamis, hurricanes, floods, seiches, and tornados) without loss of capability to perform their safety function. "General Design Criteria 4" requires that structures important to safety be able to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents including LOCAs. Furthermore, these structures must be appropriately protected against dynamic effects including the effects of missiles, pipe whip, and flooding that may result from equipment failures and from events and conditions outside the nuclear power facility.

3.2 Materials of Construction

The Category I concrete structures are composed of several constituents which, in concert, perform more than one function, i.e., load-carrying capacity, radiation shielding, and leak tightness Primarily, they include the following material systems: concrete, mild steel reinforcement, prestressing steel, and steel liner plate.

The concrete typically used in nuclear safety-related structures consists of Type II portland cement, fine aggregates, water, various admixtures for improving properties or performance of the concrete, and either normal-weight or heavyweight coarse aggregate. Type II portland cement has been used because of its improved sulfate resistance and reduced heat of hydration relative to the general purpose or Type I portland cement. Coarse aggregate consists of gravel, crushed gravel, or crushed stone. For those concrete structures in nuclear power plants which provide primary (biological) radiation shielding, heavyweight or dense aggregate materials (e.g., barites, limonites, magnetites, ilmenites, etc.) may have been used to reduce the section thickness requirements needed for attenuation. The hardened concrete typically provides the compressive load capacity for a structure. Design 28-day compressive strengths for the concrete materials utilized in nuclear power plant structures have typically ranged from 21 to 41 MPa depending on the application.

Most of the mild, or conventional, reinforcing steels used in nuclear power plants to provide primary tensile and shear load resistance/transfer consist of plain carbon steel bar stock with deformations (lugs or protrusions) on the surface. The minimum yield strength of this material ranges from about 270 to 415 MPa, with the 415 MPa material being most common. Conventional reinforcing steel also encompasses welded wire fabric, deformed wire, bar and rod mats, and all accessory steel components used in positioning/placing the reinforcement, e.g., seats, ties, etc.

A post-tensioning system consists of prestressing tendons which are installed and tensioned using jacks and other devices and then anchored to hardened concrete. A number of containment structures utilize steel prestressing tendons to provide primary resistance to tensile loadings. Three major categories of prestressing systems exist depending on the type of tendon utilized: wire, strand, or bar. These materials typically have minimum ultimate tensile strengths ranging from 1035 to 1860 MPa. The tendons are installed within preplaced ducts (conduits) in the containment structure and are post-tensioned from one or both ends after the concrete has achieved sufficient strength. After tensioning, the tendons are anchored by buttonheads, wedge anchors, or nuts, depending on the prestressing system utilized. Corrosion protection is provided by filling the ducts with corrosion-inhibiting grease (unbonded) or portland cement grout (bonded). With the exception of Robinson 2 (bar tendons) and Three Mile Island 2 (strand tendons), plants that have post-tensioned containments utilize unbonded tendons. A few plants have used bonded rock anchor tendons, e.g., Ginna and Bellefonte.

Leak tightness of reinforced and post-tensioned concrete containment structures is provided by a liner system. A typical liner system is composed of steel plate stock less than 13-mm thick, joined by welding, and anchored to the concrete by studs, structural steel shapes, or other steel products. The pressurized-water reactor containments and the "dry well" portions of boiling-water reactor containments are typically lined with carbon steel plate. The liner of the "wet well" of boiling-water reactor containments, as well as that of the light-water reactor (LWR) fuel pool structures, typically consists of stainless steel plates. Certain LWR facilities have used carbon steel plates clad with stainless steel for liner members. Although the liner's primary function is to provide a leaktight barrier, it also acts as part of the formwork during concrete placement and is used for supporting internal piping and equipment.

3.3 Description of Category I Concrete Structures

A myriad of concrete structures are contained as a part of an LWR facility. Table 1 provides a general listing of safety-related concrete structures in LWR plants. The names and configurations of these structures vary somewhat from plant to plant depending on the nuclear steam supply system vendor, architect-engineer firm and owner preference. However, major Category I structures can be grouped into three general categories for purposes of discussion, i.e., those characteristic to BWR plants, those characteristic to PWR plants, and those common to BWR and PWR plants. A more detailed description of structures relative to that presented below is provided in Refs. 7, 8, and 10.

Table 1. Representative LWR Safety-Related Concrete Structures[a]

Primary Containment/Basemat	Intake Structure
BWR Reactor Building	Cooling Tower
PWR Shield Building	Spray Ponds
Containment Internal Structures	Utility or Piping Tunnels
Auxiliary Building	Part of Turbine Building (Category I Components)
Control Room/Control Building	Auxiliary Feedwater Pump House
Diesel Generator Building	Switchgear Room
Fuel Storage Facility	Unit Vent Stack
Tanks and Tank Foundation	Radwaste Building

[a]Source: "Class I Structures License Renewal Industry Report," NUMARC 90-06, Nuclear Management and Resources Council, Washington, D.C., June 1990 (draft).

3.3.1 Boiling-Water Reactor Plant Structures

BWR Primary Containments. Although the majority of BWR plants utilize steel primary containments, there are several which utilize either reinforced concrete (Mark I, Mark II, and Mark III) or post-tensioned concrete (Mark II) primary containments. Leak tightness of the concrete containments is provided by a steel liner attached to the inside surface of the containment structure. Boiling-water reactor containments, because of provisions for pressure suppression, typically have "normally dry" sections (dry well) and "flooded" sections (wet well) which are interconnected by piping or vents.

<u>BWR Containment Internal Structures</u>. The principal BWR containment internal structures are constructed of reinforced concrete or steel and: (1) provide support for the operating and intermediate floors, piping, and equipment; (2) mitigate the consequences of a LOCA by protecting the engineered safety features; and (3) provide radiation shielding.

For Mark I plants (also Mark II and Mark III plants), a primary radiation (biological) shield wall surrounds the reactor pressure vessel and is supported on the reactor pedestal. Its purpose is to attenuate radiation emanating from the reactor core and, as a result of this service, is subjected to a thermal gradient across the wall. The primary shield wall is typically a composite structure constructed of concrete for shielding purposes, and may be lined on both surfaces with steel plates which act as the main structural components of the wall.

In addition to the primary shield wall, the Mark II containment encloses the suppression pool, which is covered by a diaphragm floor as well as several other intermediate floors. The diaphragm floor is supported by the wet-well walls and other supports within the suppression pool area. The intermediate floors are constructed of reinforced concrete, structural steel, or a combination.

Mark III plant primary internal structures include the primary shield wall, the dry well, the weir wall, the operating floor, and floors located inside the dry well and the annulus between the dry well and the containment. The primary shield wall and interior floors are similar to those of the Mark II plants. The dry well in the Mark III plant is a cylindrical structure with a flat roof slab. It completely encloses the reactor vessel and reactor coolant system and functions as a pressure boundary. The lower portion of the dry well is surrounded by the suppression pool and is equipped with horizontal vents connecting it to the suppression pool. The upper section of the dry-well structure is reinforced concrete. The weir wall forms the inner boundary of the suppression pool. It is located inside the dry well and completely surrounds the lower portion of the reactor coolant system. The weir wall is either a reinforced concrete or composite structure, having its surface lined with stainless steel to provide leak tightness. The Mark III operating floor provides laydown space for refueling operations. The floor is supported by the containment walls and the refueling pool, and is usually a combination of structural steel framing and reinforced concrete.

<u>BWR Reactor Buildings</u>. The primary function of the reactor building is to provide a medium leakage enclosure around the primary containment that can be maintained at a slightly negative pressure. Depending on the primary containment design, the reactor building structure may or may not be designed to pro-

vide shielding and/or protect the primary containment structure from external missiles or other adverse atmospheric events.

The reactor building for Mark I and Mark II plants is a multi-level reinforced-concrete structure that completely encloses the dry well and suppression pool structures. The reactor building may be a reinforced-concrete structure up to and including the roof, or it may be reinforced concrete only up to the refueling floor with the remainder being a steel-framed structure with metal siding and roof panels. The reactor building occupies the same reinforced-concrete basemat as the primary containment. Floors in the reactor building are constructed of reinforced concrete and supported by concrete or steel beam and column framing systems. Exterior load bearing walls are fabricated of reinforced concrete and are designed as shear walls to resist lateral loads. The interior walls may also be designed as shear/load bearing walls. Where required, the walls and floors are designed to provide radiation shielding.

The secondary containment for a Mark III plant is a reinforced-concrete structure that completely encloses a free-standing steel primary containment vessel. Its design provides for: (1) radiation (biological) shielding, (2) controlled release of the annulus atmosphere under accident conditions, and (3) environmental protection for the containment vessel. The secondary containment is designed to maintain leaktight integrity and to suffer no loss of function due to tornado or design-basis earthquake. In some Mark III plants the secondary containment function is accomplished, in part, by a steel frame structure having metal siding and roof decking. The ability to control radioactive releases is provided by maintaining the annular region between the secondary and primary containments at a slightly negative pressure relative to ambient conditions.

<u>BWR Fuel-Storage Facility</u>. The fuel-storage facility provides for receiving, shielding, shipping, and handling new and spent fuel. The spent-fuel pool has reinforced concrete walls for shielding and a stainless steel liner for leak tightness. The bearing walls, which support the fuel pool, are also designed as shear walls. Floors and roof are of reinforced concrete supported by steel beams. Interior columns are of structural steel or composite construction. Interior walls are concrete-block masonry or reinforced concrete. The fuel-handling floor surrounds the top of the fuel-storage pool. In some plants, the exterior walls of the fuel-storage facility are steel framed with concrete panels, concrete block, or metal siding. The roofs of these buildings are composite design. In the Mark I and Mark II plants, the fuel storage facility is an integral part of the reactor building. For Mark III plants, it is a separate structure.

3.3.2 <u>Pressurized-Water Reactor Plants</u>

<u>PWR Primary Containments</u>. The majority of pressurized-water reactor plants utilize either reinforced concrete (large dry, ice condenser, and sub-

atmospheric) or prestressed concrete (large dry) primary containments. The primary differences between the containment designs relates to volume requirements, basemat configurations, and containment internal-structures layout. The PWR-containment structures generally consist of a concrete basemat foundation, vertical cylinder walls, and dome. The basemat may consist of a simple mat foundation on fill, natural cut or bedrock, or a pile/pile cap arrangement. Interior containment surfaces are lined with a thin carbon steel liner to prevent leakage. Two of the PWR plants (Bellefonte and Ginna) have rock anchor systems to which the post-tensioning systems are attached.

<u>PWR Containment Internal Structures</u>. The containment internal structures in PWR plants tend to be more massive in nature than the internal structures in BWR plants because they are typically required to support the reactor-pressure vessel, steam generators, and other large equipment and tanks. In addition, these structures provide shielding of radiation emitted by the nuclear steam supply system. PWR containment internal structures can be described in terms of whether they are of the "dry" type (large dry or subatmospheric) or "ice condenser" type. The "dry" containments are designed to accept the initial full volume of steam released by a LOCA without pressure suppression, or operate at subatmospheric pressure, thus reducing the design LOCA pressure. The "ice condenser" containments have an internal divider barrier that channels the postulated LOCA releases into an ice condenser where condensation of steam will reduce potential pressure buildup.

The PWR "dry" containment internal structures include the primary shield wall, secondary shield wall, and operating and intermediate floors. The primary shield wall is a reinforced-concrete structure that completely surrounds the reactor vessel to form the reactor cavity. It provides biological shielding and can provide support for the reactor pressure vessel. As such, the primary shield wall is subjected to a thermal gradient across its thickness generated by the attenuation heat of gamma and neutron irradiation from the core. The secondary shield walls are of reinforced-concrete construction and surround the primary loop forming the steam-generator compartments and protecting the containment and reactor cooling system from design-basis-accident events resulting from a pipe rupture. The secondary shield walls may also provide anchorage for major piping and support intermediate and operating floors. The operating and intermediate containment internal floors are constructed of reinforced concrete or steel grating supported by structural steel.

The PWR "ice condenser" containment internal structures, except for the divider barrier and ice condenser features, are similar in function and structural design to the "dry" containment internal structures. A divider-barrier structure separates the reactor-coolant system from the upper containment, and in the event of a LOCA, contains the steam released from the reactor-coolant system and channels it through vent doors into the ice condenser. The operating floor serves as part of the divider barrier.

PWR Shield Building. Pressurized-water reactors with steel reactor pressure vessels have a shield building structure that is similar to the BWR Mark III secondary containment building. The PWR shield building is typically a reinforced-concrete structure with a shallow dome roof and is designed to provide essentially the same functions as the Mark III containment building.

PWR Fuel-Storage Facility. New and spent fuel is transferred between the reactor vessel and the fuel transfer penetration through the refueling canal. The refueling canal may also be used for temporary storage of reactor vessel internals or fuel during maintenance or refueling operations. The refueling canal is of reinforced-concrete construction lined with stainless steel to prevent leakage.

The new- and spent-fuel storage pools for PWR plants are typically located in an auxiliary building proximate to the containment. The fuel storage pools are fabricated of reinforced-concrete wall and slab components. These structures are generally massive in cross-section to support a large pool of water and the fuel elements and are lined with stainless steel. The pools are connected to the inside of the containment via the refueling canal described above.

3.3.3 Structures Common to BWR/PWR Plants

Safety-related structures, common to BWR/PWR plants, which are fabricated of reinforced concrete or of composite concrete and steel construction include: auxiliary building; radwaste building; turbine building; control room/control building; diesel generator building; tanks and tank foundations; intake structures; cooling towers; cooling pond, spray pond and spray canal; utility and piping tunnels; auxiliary feedwater pump house; switchgear room; and unit vent stack. Reference 8 provides a description of these structures.

3.4 Potential Degradation Factors

The longevity, or long-term performance, of Category I concrete structures is primarily a function of the durability or propensity of these structures to withstand potential degradation effects. Over the life of a nuclear power plant, changes in the properties of the structure's constituent materials will in all likelihood occur as a result of aging and environmental stressor effects. These changes in properties, however, do not have to be detrimental to the point that the structure has deteriorated and is unable to meet its functional and performance requirements. In fact, it has been noted that when specifications covering concrete's production are correct and are followed, concrete will not deteriorate.[11] Concrete in many structures, however, can suffer undesirable degrees of change with time because of improper specifications, a violation of specification, or environmental stressor or aging factor effects. Table 2 summarizes primary mechanisms (factors) which can produce premature deterioration of concrete structures. A brief description of the mechanisms (factors) which can degrade the concrete, mild steel reinforcement, prestressing steel, and liner/structural steel materials follows. A more detailed discussion of the mechanisms is provided in Refs. 8 and 10.

Table 2. Degradation Factors That Can Impact the Performance of Category I Concrete Structures

Material System	Degradation Factor	Primary Manifestation
Concrete	Chemical attack	
	Efflorescence and leaching	Increased porosity
	Salt crystallization	Cracking
	Alkali-aggregate reactions[a]	Volume change/cracking
	Sulfate attack	Volume change/cracking
	Bases and acids	Increased porosity/erosion
	Physical attack	
	Freeze/thaw cycling	Cracking/spalling
	Thermal exposure/thermal cycling	Cracking/spalling
	Irradiation	Volume change/cracking
	Abrasion/erosion/cavitation	Section loss
	Fatigue/vibration	Cracking
Mild Steel Reinforcement	Corrosion	Concrete cracking/spalling
	Elevated temperature	Decreased yield strength
	Irradiation	Reduced ductility
	Fatigue	Bond loss
Prestressing	Corrosion	Reduced section
	Elevated temperature	Reduced strength
	Irradiation	Reduced ductility
	Stress relaxation	Prestress force loss
Liner/Structural Steel	Corrosion	Section loss

[a]Includes reactions of cement aggregate and carbonate aggregate.

3.4.1 Concrete

If concrete is properly designed for the environment to which it will be exposed, produced using good quality control principles, and placed using proper construction and curing methods, it is capable of essentially maintenance-free performance for decades. However, concrete is potentially vulnerable to attack under a variety of exposures, i.e., chemical or physical attack.

Chemical Attack. Chemical attack is the alteration of concrete through chemical reaction with either the cement paste or the coarse aggregate. Generally, the attack occurs at the exposed surface region of the concrete, but with the presence of cracks or prolonged exposure, chemical attack can affect entire structural elements. Deterioration of concrete by chemical reactions manifests itself in different ways, i.e., increase in porosity and permeability, decreases in strength, and cracking and spalling.[12] The rate of chemical attack on concrete is a function of the pH of the aggressive fluid and the concrete permeability, alkalinity, and reactivity. Chemical attack may occur as efflorescence and leaching, sulfate attack, bases and acids, salt crystallization, and alkali-aggregate reactions.

Efflorescence occurs on the surface of concrete. Pure water from the condensation of fog or water vapor, and soft water from rain or melting snow and ice, may contain few or no calcium salts. In contact with concrete, the water will tend to hydrolyze or dissolve the calcium-containing concrete products until equilibrium is reached. Frequently, the leachate interacts with CO_2 from the atmosphere to form a precipitate of white calcium carbonate (efflorescence) on the surface. However, in the case of flowing water or seepage under pressure, conditions exist for continued hydrolysis (leaching). Theoretically, hydrolysis of cement paste can continue until most of the calcium hydroxide (most susceptible component) has been leached away, exposing other cementitious constituents to chemical decomposition and eventually leaving behind silica and alumina gels having little or no strength.

Most soils contain sulfates in the form of gypsum in amounts from 0.01% to 0.05% (expressed as SO_4) which are harmless to concrete. However, where higher concentrations (>0.1% soluble sulfate) of sodium, potassium, or magnesium sulfates are present in alkali soils or waters, significant deterioration can occur. When the cement paste comes into contact with sulfate ions, the alumina-containing hydrates in the presence of calcium hydroxide are converted to ettringite, which results in an expansive reaction leading to cracking. In the case of magnesium sulfate attack, the conversion of calcium hydroxide to gypsum can also cause expansion by the formation of relatively insoluble and weakly alkaline magnesium hydroxide, i.e., stability of the calcium-silicate hydrate system in the paste is reduced making it more susceptible to sulfate attack.

Hydrated cement paste has a high pH value (12.5 to 13.5) due to the presence of large concentrations of Na^+, K^+ and OH^- ions. Any environment having a pH less than 12.5 is potentially harmful and can lead to destabilization of the cementitious hydration products. In addition to the pH, the permeability of the concrete affects the rate of chemical attack. Most industrial and natural waters, because of their relatively low pH, can be considered aggressive to concrete. The cation exchange between acidic solutions and the constituents in portland cement paste gives rise to soluble salts of calcium which are removed by leaching and increase the porosity and permeability of the concrete. Reference 13 presents a listing of reactivity with concrete of various chemicals.

Salts can cause damage to concrete through the development of crystal growth pressures that arise through physical causes. Deterioration of this type occurs when concrete is in contact with water containing large quantities of dissolved solids, e.g., $CaSO_4$, $NaCl$, Na_2SO_4. As the water permeates into the concrete, the salts crystallize in open pores due to evaporation. Repeated evaporation can cause the salt deposits to build to the point that the stresses generated will crack the concrete. Structures in contact with fluctuating water levels (e.g., water-management structures) or in contact with ground waters containing salts are susceptible to this type of deterioration.

Chemical reactions involving alkali ions (portland cement), hydroxyl ions, and certain silica (or carbonate) constituents present in aggregate materials lead to an expansive reaction and cracking. This form of chemical attack is significant in that it can occur throughout the entire section thickness. Pop-outs and exudation of viscous alkali-silica fluid are other manifestations of this phenomenon. Although alkali-aggregate reactions (AAR) typically occur within 10 years of construction, deterioration has not occurred in some structures until 15 to 20 years following construction, and some structures have not exhibited early signs of deterioration until ages of 20 to 25 years. The delay in structures exhibiting deterioration due to AAR indicates that there may be less reactive forms of silica which, although considered innocuous during aggregate qualification testing, may eventually cause deterioration.[14]

Physical Attack. Physical attack is the second major cause of deterioration of concrete. Although it is often difficult to separate physical attack from chemical attack of concrete, for purposes of discussion physical attack will include degradation factors that result from environmental or mechanical effects, i.e., freeze/thaw cycling, thermal exposure/thermal cycling, irradiation, abrasion/erosion/cavitation, and fatigue/vibration.

Concrete, when in a saturated or near-saturated condition, can be susceptible to damage during freezing and thawing cycles caused by hydraulic pressure generated in the capillary cavities of the cement paste as the water freezes. Damage to concrete can take several forms: scaling, spalling, and pattern cracking (e.g., D-cracking). The most common damage is scaling or spalling due to expansion of the cement paste matrix. Factors controlling the resistance of concrete to freeze/thaw cycling include air entrainment (size and spacing of air bubbles), water-cement ratio (lower water-cement ratios result in less water being available for freezing), and curing (concrete should be properly cured prior to frost exposure). Nuclear power plant structures that may be affected by freeze/thaw damage include those that are utilized in the intake/conveyance/management of cooling water and those structures having exterior surface locations that promote the "ponding" of rain or snow.

Elevated temperature and thermal gradients are potentially degrading to concrete structures in that they affect concrete's strength (load-carrying ability) and stiffness (deformations and loads at constraints). The mechanical property variations result largely because of changes in the moisture content of the concrete constituents and progressive deterioration of the cement-paste matrix which can lead to cracking and spalling. Concrete exposed to temperatures of 90°C may lose up to 10% of its room-temperature strength and modulus of elasticity values.[15] Significant strength losses occur above 450°C.[16] Thermal cycling, even at relatively low temperatures (<65°C), can also have deleterious effects on concrete's mechanical properties. At higher temperatures (200 to 300°C), the first thermal cycle causes the largest percentage of damage.[17] Although Category I concrete structures are generally limited to maximum temperatures of 65°C by technical specifications, local areas may be heated to temperatures approaching that of the nuclear steam-supply-system coolant, 345°C, at local piping penetrations, improperly ventilated areas, etc.

Irradiation in the form of either fast and thermal neutrons emitted by the reactor core or gamma rays produced as a result of capture of neutrons by members (particularly steel) in concrete can affect the concrete. The fast neutrons produce considerable growth of certain aggregate materials, e.g., flint. Gamma rays produce radiolysis of water in cement paste which can affect concrete's creep and shrinkage behavior to a limited extent and also result in evolution of gas. Prolonged exposure of concrete to irradiation can result in decreases in tensile and compressive strengths and modulus of elasticity. Limited research studies[18] have reported threshold levels for measurable damage to concrete of 1×10^{19} n/cm^2 for neutron fluence and 10^{10} rads of dose for gamma radiation. Table 3, derived from Reference 19, provides data for radiation environments at outside surface of LWR pressure vessels for a 1000 MW(e) plant with an 80% capacity factor. Values in the table indicate that radiation levels approaching those which may damage concrete in the primary shield wall may occur after 40 years of operation. The radiation levels presented are somewhat conservative, however, because further attenuation due to the presence of air gaps, insulation, etc., has not been included.

Table 3. Radiation Environments at Outside Surface of LWR Reactor Pressure Vessels [1000 MW(e) with 80% plant capacity factor]

	BWR			PWR		
	40 Year (32 EFPY)	60 Year (48 EFPY)	80 Year (64 EFPY)	40 Year (32 EFPY)	60 Year (48 EFPY)	80 Year (64 EFPY)
Neutron fluence (n/cm2)						
Slow (E < 1.0 MeV)	3.7×10^{18}	5.6×10^{18}	7.5×10^{18}	2.0×10^{19}	3.0×10^{19}	4.0×10^{19}
Fast (E > 1.0 MeV)	5.1×10^{17}	7.7×10^{17}	1.0×10^{18}	1.0×10^{18}	1.5×10^{18}	2.0×10^{18}
Gamma total integrated dose (rads)	1.6×10^{10}	2.4×10^{10}	3.2×10^{10}	4.7×10^{9}	7.0×10^{9}	9.3×10^{9}

Progressive loss of material at the concrete surface can occur due to abrasion, erosion, or cavitation. Resistance of concrete to these effects is dependent on the quality of the concrete, i.e., porosity, strength, aggregate particle durability. Category I structures which provide water intake, conveyance, and management are most susceptible to abrasion, erosion, and cavitation effects.

Concrete structures subjected to fluctuations in loading, temperature, or moisture content can be damaged by fatigue. Fatigue damage initiates at microcracks in the cement paste matrix, proximate to large aggregate particles, reinforcing steel, or other stress raisers, e.g., defects. Upon continued-load fluctuations, the microcracks can coalesce and propagate to form structurally significant cracks which can expose the concrete and reinforcing steel to hostile environments or produce increased deflections. Within LWR facilities, supports for the nuclear steam-supply-system components and operating pumps/turbines may suffer from the effects of vibration.

3.4.2 Mild Steel Reinforcing

Deterioration of mild steel reinforcing materials can occur as a result of corrosion, elevated temperature, irradiation, and fatigue.

Corrosion. Although deterioration of concrete structures may be attributable to the combined effects of more than one cause, corrosion of embedded metal is one of the principal causes. Corrosion of steel is an electrochemical process. Electrochemical potentials which form the corrosion cells may be generated as: (1) composition cells resulting when two dissimilar metals are embedded in concrete, or when significant variations exist in surface characteristics of the steel; and (2) concentration cells occurring due to differences in concentration of dissolved ions in the vicinity of steel, such as alkalies, chlorides, and oxygen.[20] As a result, one of two metals (or different parts of the metal when only one metal is present) becomes anodic and the other cathodic. Other potential causes of corrosion include stray electrical currents or galvanic action with an embedded steel of different metallurgy. The transformation of metallic iron to ferric oxide (rust) is accompanied by an increase in volume which, depending on the state of oxidation, may be as large as 600 percent of the original metal. The volume increase can cause cracking and spalling of the concrete.

In good-quality, well-compacted concretes, reinforcing steel with adequate cover should not be susceptible to corrosion because the highly alkaline conditions present within the concrete (pH $>$ 12) causes a passive iron oxide film to form on the iron surface, i.e., metallic iron will not be available for anodic activity. However, when the concrete pH falls below 11, a porous oxide layer (rust) can form on the reinforcing steel due to corrosion. Carbonation and the presence of chloride ions can destroy the passive iron oxide film. The penetration of CO_2 from the environment can be accelerated due to the concrete being porous (poor quality) or the presence of microcracks. The penetration of chloride ions can also destroy the passive oxide film on the reinforcing steel, even at high alkalinities (pH $>$ 11.5). For typical concrete mixes, the threshold chloride content to initiate steel corrosion is about 0.6 to 1.2 kg Cl^-/m^3 (Ref. 14). Furthermore, when large amounts of chloride are present, concrete tends to hold more moisture which increases the risk of steel corrosion by lowering concrete's electrical resistivity. Once the passivity of the steel is destroyed, concrete's electrical resistivity and oxygen availability control the corrosion rate.

Because of the potential damage to concrete structures caused by the corrosion of reinforcing steel, this factor is potentially significant for all Category I concrete structures.

Elevated Temperature. The properties of mild steel reinforcement used in design are generally a function of the yield strength which is affected by exposure to elevated temperatures. References 21 and 22 indicate that for temperatures up to about 200°C, the yield strength of rebars is reduced less than 10% and that the loss of bond properties to concrete does not occur until 300°C or above. Typical LWR thermal exposures that the mild steel reinforcing would experience are below the temperatures at which properties are affected.

Irradiation. Neutron radiation produces changes in the mechanical properties of carbon steel, e.g., increased yield strength and rise in ductile/brittle transition temperature. The reduced ductility increases the possibility of brittle fracture. References 23 and 24 suggest that a threshold level of neutron fluence for alteration of reinforcing steel properties is 1×10^{18} n/cm^2. In LWR facilities, reinforcing steel potentially affected would be in the primary shield wall adjacent to the reactor pressure vessel.

Fatigue. Fatigue of the mild steel reinforcing system would be coupled with that of the surrounding concrete. The most likely effect of fatigue loadings would be a loss of bond between the steel reinforcement and concrete. Because of the typically low normal-stress levels in the reinforcing steel elements in Category I structures, fatigue failure is not likely to occur.

3.4.3 Prestressing Steel

Potential sources of deterioration of the prestressing steel would be corrosion, elevated temperature, irradiation and loss of prestressing force.

Corrosion. Corrosion of prestressing systems can be highly localized or uniform. Most corrosion-related failures have been the result of localized attack produced by pitting, stress corrosion, hydrogen embrittlement, or a combination. Failure of prestressing tendons can also occur as a result of microbiological-induced corrosion. Protection of the prestressing system is provided by filling the tendon ducts with organic corrosion inhibitors (nongrouted tendons) or portland cement grout (grouted tendons). Due to the importance of the post-tensioning system to the overall structural integrity of the containments and the stress state in the tendons, the tolerance to corrosion attack is not as great as for the mild steel reinforcement. The prestressing systems, however, are one of the most highly monitored systems in an LWR plant.

Elevated Temperature. Reference 22 indicates that thermal exposures up to 200°C do not significantly reduce the tensile strength of prestressing wires or strand. Elevated temperatures however, affect the relaxation and creep properties of the tendons. Reference 25 indicates that losses in a 15.2-mm-diam. strand initially stressed to 75% its guaranteed ultimate tensile strength at 40°C will be 5 to 6.4% after 30 years. As temperature levels experienced by the post-tensioning systems in LWR plants are below 200°C, thermal damage to the prestressing steels under normal operating conditions is unlikely.

Irradiation. Results from studies[23] in which 2.5-mm-diam. prestressing wires were stressed to 70% of their ultimate tensile strength and irradiated to a total dose of 4×10^{16} n/cm^2 (flux of 2×10^{10} n·cm^2·s) showed that for exposures up to this level, the relaxation behavior of irradiated and unirradiated materials was similar. These radiation levels are likely to be higher than those experienced by the prestressing systems in LWR containment structures.

Loss of Prestressing Force. Primary contributors to the loss of original force level that was applied to the prestressing tendons include: friction, end effects, anchorage deflections (take up and slip), elastic shortening, tendon relaxation, and concrete creep/shrinkage. Of these factors, the last two are time dependent and thus aging related. Guidelines for developing surveillance programs acceptable to the USNRC, as well as estimating loss in prestressing force with time, are provided in Regulatory Guides.[26,27]

3.4.4 Liner/Structural Steel

The primary degradation factor for the liner plate and structural steel (both embedded sections and those within the containment) is corrosion. Typically, exposed steel surfaces are coated with a primer, or primer-finish coat system. For liner plates, local attack is of greatest importance because of its possible affect on leak tightness. Local attack may result at locations where there has been loss-of-coating integrity, impact, failure of adjoining floor sealant, etc. The local attack mechanisms include galvanic corrosion, pitting, crevicing, and corrosion caused by stray electric currents or microbiological effects. Structural steel embedments are normally protected by the surrounding concrete. However, when the concrete becomes cracked or porous or has its pH lowered, it makes the embedments susceptible to corrosion.

3.5 Performance History of Category I Concrete Structures

In general, the performance of concrete materials and structures in nuclear power plants has been good. This to a large degree can be attributed to the effectiveness of the quality control/quality assurance programs in detecting potential problems (and subsequent remedial measures) prior to plant operation.[28] However, there have been several instances in nuclear power plants where the capability of concrete structures to meet future functional/ performance requirements has been challenged due to problems arising from improper material selection, construction/design deficiencies, or environmental effects. Examples of these instances include anchorhead failures (Farley, Byron, Bellefonte), dome delaminations (Crystal River 3, Turkey Point 3), and corrosion of steel tendons and rebars (Fort St. Vrain, San Onofre). Other problems such as the presence of voids in concrete, concrete cracking, materials out of specification, misplaced rebar, etc., are identified.[7,8,10] Although many of the documented problems are not due to environmental stressors of aging factors, if not discovered they could potentially compromise integrity of the structures during an extreme event or exhibit synergistic effects with any environmental stressors or aging factors present.

4. STRUCTURAL AGING PROGRAM

Results of a study[10] conducted under the NRC Nuclear Plant Aging Research (NPAR) Program[29] were utilized to help formulate the Structural Aging (SAG) Program[30] which was initiated in 1988. The SAG Program has the overall objective of preparing a handbook or report which will provide the NRC license reviewers and licensees with the following: (1) identification and evaluation of the structural degradation processes; (2) issues to be addressed under nuclear power plant continued-service reviews, as well as criteria, and their bases, for resolution of these issues; (3) identification and evaluation of relevant inservice inspection or structural assessment programs in use, or needed; and (4) methodologies required to perform current assessments and reliability-based life-predictions of safety-related concrete structures. To accomplish this objective, the SAG Program is addressing the sources of uncertainty identified earlier with respect to determination of the residual life of safety-related components or structures. Structural Aging Program activities are conducted under a management task and three major technical task areas: (1) materials property data base, (2) structural component assessment/ repair technologies, and (3) quantitative methodology for continued service determinations.

4.1 Program Management

The overall objective of the program management task is to effectively manage the technical tasks undertaken to address priority structural safety issues related to nuclear power plant continued-service applications. Primary management activities include: (1) program planning and resource allocation, (2) program monitoring and control, and (3) documentation and technology transfer. Under the first of these activities, a five-year plan was prepared,[30] and subcontracts related to meeting objectives of the technical task areas have been implemented with six organizations. The program monitoring and control activity primarily addresses the preparation of management reports, annual technical progress reports,[31] and participation in NRC information meetings. Documentation and technology transfer includes program coordination with other government agencies-related activities (e.g., NRC Low-Level Radioactive Waste Program), participation in technology working groups (e.g., ASME Section XI Working Group on Concrete Pressure Components), coordination with foreign technologies,[32] and technology exchange through participation in national[33,34] and international conferences.[35] Reference 31 summarizes activities conducted under the program management task.

4.2 Materials Property Data Base

The objective of the materials property data base task is to develop a reference source which contains data and information on the time variation of material properties under the influence of pertinent environmental stressors and aging factors. This source will be used to assist in the prediction of long-term deterioration of critical structural components in nuclear power plants and to establish limits on hostile environmental exposure, i.e., establish component service life or improve probability of a component surviving an extreme event. Primary activities under this task include the development of the Structural Materials Information Center, assemblage of materials property data, and formulation of material behavior models.

4.2.1 Structural Materials Information Center (SMIC)

A review and assessment of materials property data bases, hardware, and software has been completed.[36] Conclusions derived from the investigation were that no data bases existed which met the needs of the SAG Program, personal computers provide the most economical approach in setting up the desired data base, and a "canned" commercial data base, which can be formatted to meet program requirements, should be utilized. Reference 37 presents the plan utilized in the development of the SMIC which consists of the *Structural Materials Handbook* and the *Structural Materials Electronic Data Base*.

The *Structural Materials Handbook*[38] is an expandable, hard-copy reference document that contains complete sets of data and information for each material in the data base and serves as the information source for the *Structural Materials Electronic Data Base*. The handbook consists of four volumes and is provided in a loose-leaf format so that each volume can be easily revised and updated. Volume 1 contains design and analysis information useful for structural assessments and structural margins evaluations. This volume contains design values for mechanical, thermal, physical, and other properties presented as tables, graphs, and mathematical equations. Volume 2 reflects the supporting documentation and includes test results and data used to develop the design values presented in Volume 1. Material data sheets are provided in Volume 3. These sheets include general information and baseline data as well as material composition and constituent material properties. Volume 4 contains appendices describing the handbook organization as well as updating and revision procedures.

The *Structural Materials Electronic Data Base* is an electronically accessible version of the handbook. The data base was developed on an IBM-compatible personal computer using a data base management system designed specifically for maintaining and displaying properties of engineering materials.[39,40] Each material record in the data base may contain up to nine categories of data and information, i.e., designations, specifications, composition, notes, forms, graphs, properties, classes, and rankings. The user may search an entire data base file to locate materials with similar material properties. During the search, each material record is screened for selected tabular data and certain property values based on comparison indicators, i.e., =, >, <, > =, < =, and < >. The user may elect to perform property searches using either the International System of Units (SI) or customary units. Due to current software limitations, all the data and information reported in the handbook are not included in the data base. The data base, however, provides an efficient means for searching various data base files. More details on the data base are available in Appendix F to Volume 4 of Ref. 38.

Each material system is presented as a separate chapter in the handbook and as a separate data file in the data base. A unique seven-character material code, consistent in both the handbook and data base, is assigned to each material system. The material code consists of four identifying parameters (i.e., 01 C B 001), each of which can be used to sift information in the data base. The chapter index, 01, is used to represent the various material systems such as concretes, structural steels, etc. The group index, C, is used to organize groups of materials with common compositional traits. The class index, B, is used to organize groups of materials with common compositional traits into subsets having similar compositional makeup or chemistry. The identifier, 001, is used to differentiate structural materials having the same chapter, group, and class indices according to a specific concrete mix, ASTM standard specification, etc. Since a wide variety of descriptive information and materials property data are contained in the data base as tables, notes, and graphs, each entry is assigned a unique four-digit property code (i.e., 3021 represents compressive strength value used for structural design) selected from an established set of material property codes provided in Ref. 38.

4.2.2 Data Assemblage

One of the findings of Ref. 10 was that material property data for concrete over an extended period of time are practically nonexistent, especially for concretes which have been subjected to aging factors or environmental stressors characteristic of those associated with nuclear power plants. For most concrete structures that have been in service for the time period of interest (30 to 100 years), either detailed information about constituent materials, plastic concrete properties, environmental exposure, or the time variation of material properties is unknown. Since these types of data and information are not readily available, three approaches are being used to supplement and expand the data base: (1) exchange of technology, (2) development of properties using prototypical material samples obtained from existing concrete structures, and (3) accelerated aging tests. To date, the first two of these approaches have produced eleven material property data base files which have been incorporated into SMIC. A detailed description of the information sources utilized to develop these data base files is provided in Ref. 38.

4.2.3 Material Behavior Modeling

Prediction or explanation of the complex interrelationships that occur between concrete's constituents and between concrete and its environment requires the development of mathematical models based on scientific and engineering principles. Such models play a vital role in the development of techniques for reliability-based life predictions of concrete structures in nuclear power plants. Models being developed address the aging factors (cement hydration, alkali-aggregate reaction, etc.) and environmental stressors (temperature, irradiation, freeze-thaw, etc.) which can impact the Category I concrete structures as well as any synergistic effects that result when more than one degradation factor is present.

4.3 Structural Component Assessment/Repair Technology

The objectives of this task are to: (1) develop a systematic methodology which can be used to make quantitative assessments of the presence, magnitude, and significance of any environmental stressors or aging factors which adversely impact the durability of safety-related concrete structures in nuclear power plants; and (2) provide recommended inservice inspection or sampling procedures which can be utilized to develop the data required both for evaluating the current condition of concrete structures as well as trending the performance of these components. Primary activities under this task include development of a structural aging assessment methodology for concrete structures in nuclear power plants, review and evaluation of inservice inspection and structural integrity assessment methods for detection and quantification of potential deterioration phenomena in concrete structures, and evaluation of remedial/preventative measures considerations for concrete structures.

4.3.1 Structural Aging Assessment Methodology

The structural aging assessment methodology is founded on several criteria: relation of subelements to overall importance of the parent safety-related concrete structures, safety significance of the structure as a whole, influence of applied environment, and possibility of occurrence as well as end result of degradation. Application of the structural aging assessment methodology involves seven primary activities: (1) identification of Category I concrete structures and their subelements; (2) rating of the importance, I, of each subelement to its parent structure based on its structural contribution; (3) evaluation and assigning a safety significance ranking value, SS, to each Category I structure (value is common to all subelements of the parent structure); (4) evaluation and assigning an environmental exposure severity rating value, EE, to each subelement; (5) identification of key degradation factors for each subelement, assignment of a degradation factor grading value, DFG, to each factor, and calculation of a degradation factor significance value, DFS, for each subelement by summing the degradation factor grading values and dividing by the number of degradation factors; (6) computation of the rank of each subelement, SR, in terms of importance to aging using

$$SR = w_1 I + w_2 SS + w_3[(EE + DFS)/2] \quad (1)$$

where w_1, w_2, and w_3 are weighting factors permitting certain components of the equation to be prioritized; and (7) calculation of the cumulative rank for each Category I structure by summing the subelement ranks and dividing by the number of subelements. Application of this methodology provides a listing of safety-related concrete structures and subelements, ranked in order of importance to aging and longevity of the nuclear power plant. Reference 41 presents more details on the methodology as well as its application to three light-water reactor plants.

4.3.2 NDE/Sampling Inspection Technology

Basic activities under this subtask are related to evaluation of inservice inspection and structural integrity assessment methods for detection and quantification of potential deterioration phenomena in nuclear power plant concrete structures. Section 3.4 provided a listing of the various degradation factors that can impact the performance of Category I concrete structures in terms of the materials of construction, i.e., concrete, mild steel reinforcing, steel prestressing, and liner/structural steel.

Methods used to detect degradation of concrete materials are grouped into two categories: direct and indirect. Direct techniques generally involve a visual inspection of the structure, removal/testing/analysis of material, or a combination. Periodic visual examinations of exposed concrete provides a rapid and effective means for identifying and defining areas of distress, e.g., cracking, spalling, and volume change. In areas exhibiting extensive deterioration, or where more quantitative results are desired, core samples can be removed for strength testing and petrographic examination. The indirect techniques measure some property of concrete from which an estimate of concrete strength, elastic behavior, or extent of concrete degradation can be made through existing correlations. Several potential nondestructive techniques for evaluating concrete materials and structures include: (1) audio, (2) electric, (3) impulse radar, (4) infrared thermography, (5) magnetic, (6) microscopic refraction, (7) modal analysis, (8) nuclear, (9) radiography, (10) rebound hammer, (11) ultrasonic, and (12) pulse echo. In addition to core sampling, potential destructive testing techniques that can be used to evaluate concrete materials include: (1) air permeability, (2) break-off, (3) chemical, (4) probe penetration, and (5) pull out. A description of each of these test methods, as well as their capabilities and limitations, has been completed and the results are presented in Ref. 42.

4.3.3 Remedial/Preventative Measures Considerations

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

4.4 Quantitative Methodology for Continued-Service Determinations

The overall objective of this task is to develop a methodology which can be used for performing condition assessments and making reliability-based life predictions of critical safety-related concrete structures in nuclear power plants. The methodology will integrate information on degradation and damage accumulation, environmental factors, and load history into a decision tool that will enable a quantitative measure of structural reliability and performance under projected future service conditions based on an assessment of the existing structure. When completed, the methodology will take into account the stochastic nature of past and future loads due to operating conditions and the environment, randomness in those physical processes and environmental stressors that may lead to degradation in strength, and uncertainty in nondestructive evaluation techniques. Activities associated with this task include: (1) identification and appraisal of existing condition assessment methods and damage prediction models, (2) assembly of pertinent data for use in the predictive models, (3) development of reliability-based condition assessment methodologies for the analysis of current and future reliability, and (4) validation of condition assessment using laboratory or prototypical structures data. Results to date are discussed below and include the development of probabilistic models and identification of degradation models. More details on these results are available in Ref. 43.

Probabilistic models have been developed to assess time-dependent reliability and deterioration of concrete structures subjected to stochastic loads. The changes in engineering properties of steel and concrete materials over an extended service life are taken into account. Degradation mechanisms related to corrosion of reinforcing steel, detensioning of prestressing tendons, and loss of concrete strength potentially impact numerous concrete structures in nuclear power plants.

Degradation models and load process statistics necessary to illustrate the methodology have been identified. Reliability functions also have been developed to illustrate the evolution in structural reliability over time. Such functions can be used as a basis for selecting appropriate plant-license extension periods or to determine required intervals of inspection and maintenance necessary to maintain reliability at an acceptable level.

5. APPLICATION OF STRUCTURAL AGING PROGRAM RESULTS

When completed, the results of this program will provide an improved basis for the USNRC staff to permit continued operation near, at, or beyond the nominal 40-year design life of a nuclear power plant. More specifically, potential regulatory applications of this research include: (1) improved predictions of long-term material and structural performance and available safety margins at future times, (2) establishment of limits on exposure to environmental stressors, (3) reduction in total reliance by licensing on inspection and surveillance through development of a methodology which will enable the integrity of structures to be assessed (either pre- or post-accident), and (4) improvements in damage inspection methodology through potential incorporation of results into national standards which could be referenced by standard review plans. Although activities under this program address civil structures in nuclear power plants, many of the techniques and methodologies developed will be equally applicable to general civil engineering structures, e.g., factories, warehouses, office buildings, etc.

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MANAGEMENT OF THE AGING OF CRITICAL SAFETY-RELATED CONCRETE STRUCTURES IN LIGHT-WATER REACTOR PLANTS*

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PRESENTATION TO:
18th WATER REACTOR SAFETY INFORMATION MEETING
ROCKVILLE, MD
OCTOBER 22 - 24, 1990

*Research sponsored by the Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission under Interagency Agreement 1886-8084-5B with the U.S. Department of Energy under Contract DE-AC05-84OR21400 with Martin Marietta Energy Systems, Inc.

The submitted manuscript has been authored by a contractor of the U.S. Government under Contract No. DE-AC05-84OR21400. Accordingly, the U.S. Government retains a nonexclusive, royalty-free license to publish or reproduce the published form of this contribution, or allow others to do so, for U.S. Government purposes.

PRESENTATION WILL ADDRESS FOUR TOPICS

- INTRODUCTION
- BACKGROUND
- STRUCTURAL AGING PROGRAM DESCRIPTION AND STATUS
- SUMMARY

PRESENTATION WILL ADDRESS FOUR TOPICS

- INTRODUCTION
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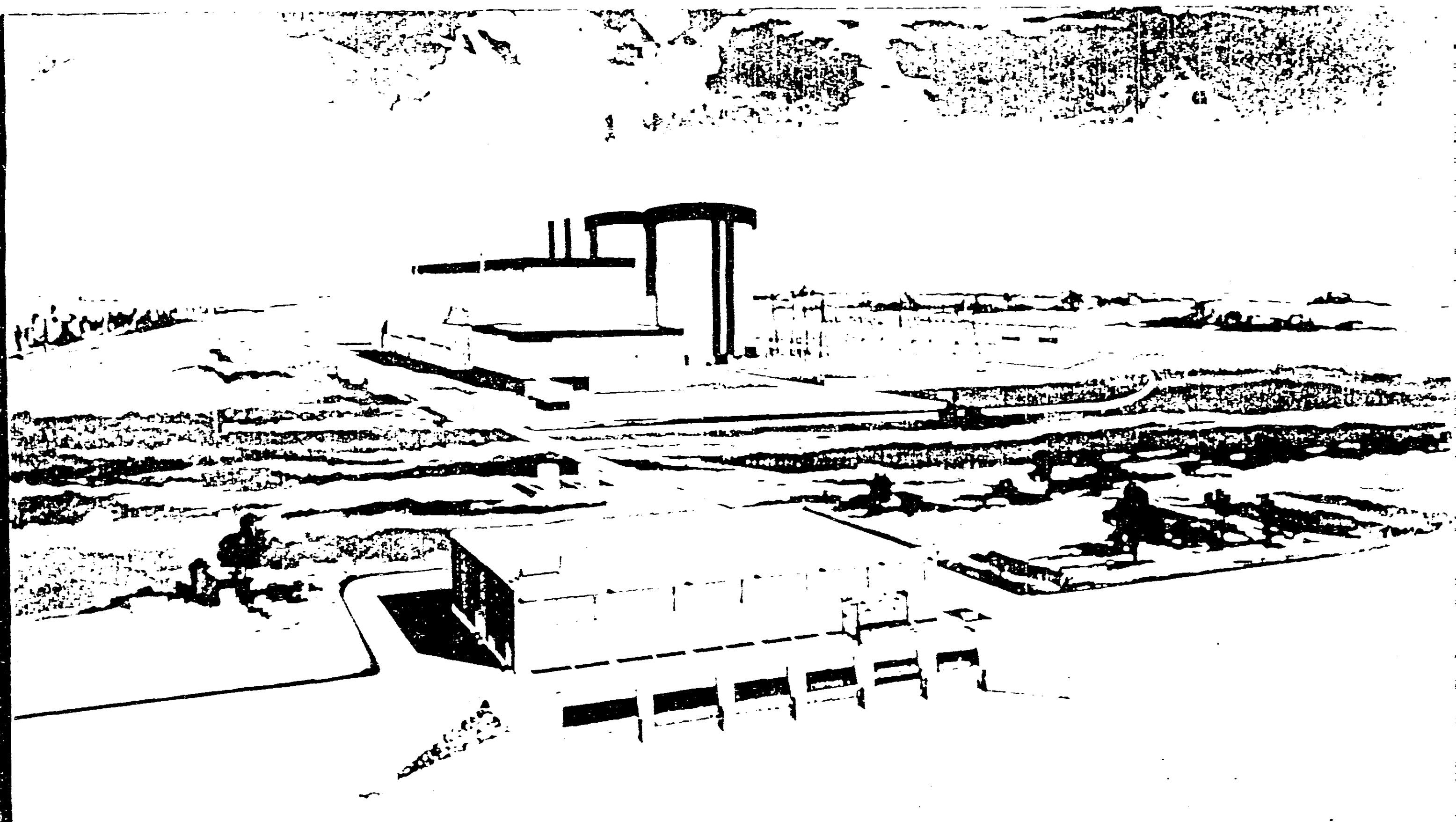
A MYRIAD OF CONCRETE—BASED STRUCTURES IS CONTAINED AS PART OF A LWR PLANT



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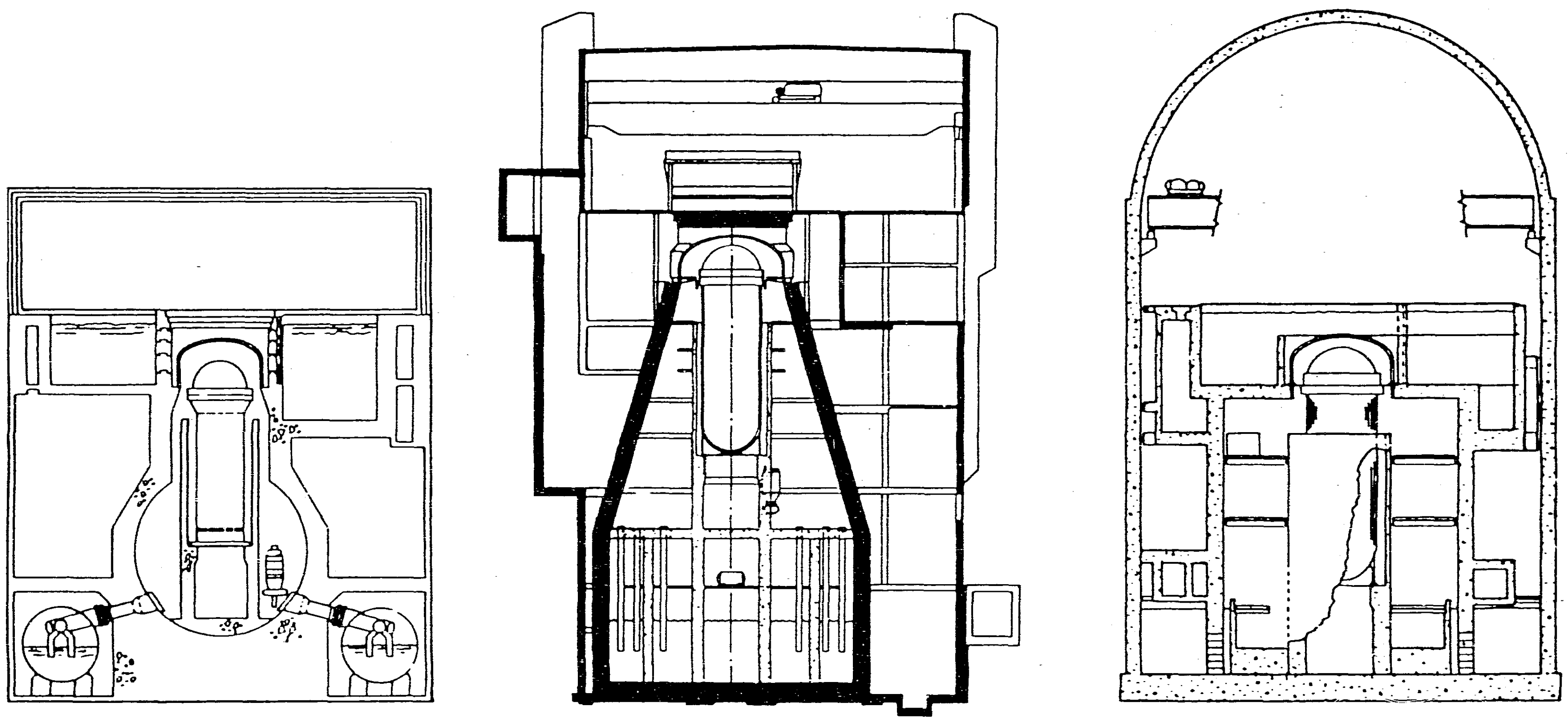
MARTIN MARIETTA

MARTIN MARIETTA ENERGY SYSTEMS, INC.

ORNL-DWG 88-4488 ETD

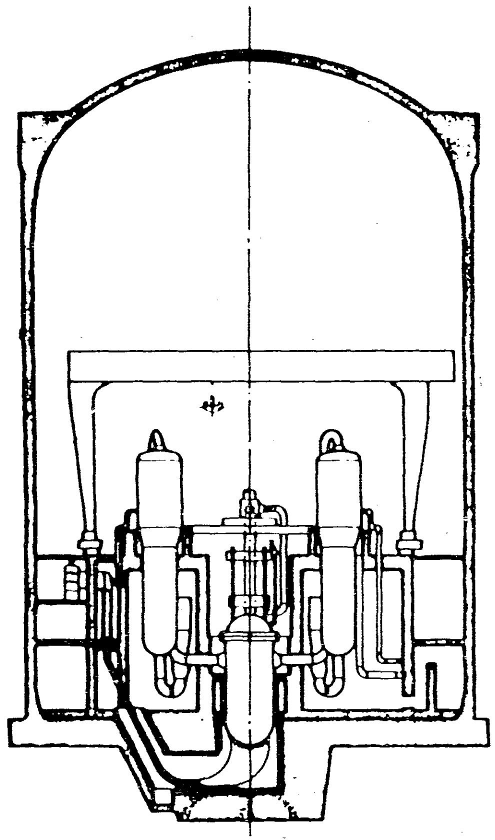
WITH RESPECT TO PUBLIC SAFETY (FISSION PRODUCT RELEASE), THE CONTAINMENT IN BWR PLANTS HAS BEEN IDENTIFIED AS THE MOST CRITICAL COMPONENT



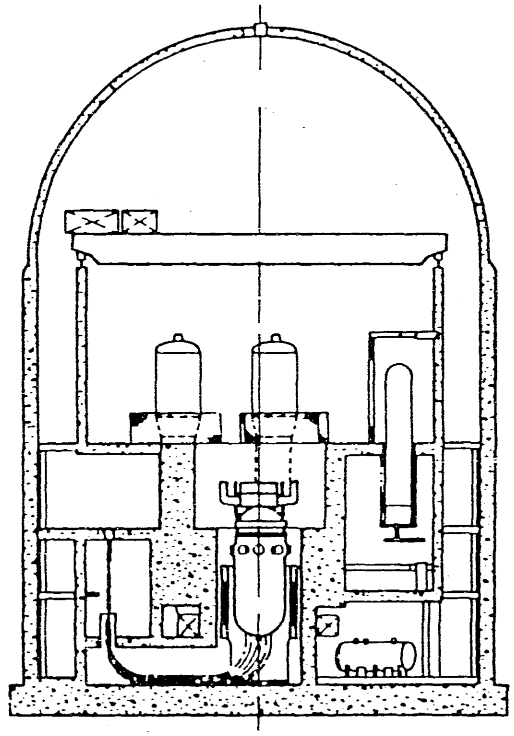
BWR MK I　　BWR MK II　　BWR MK III

ORNL-DWG 88-4489 ETD

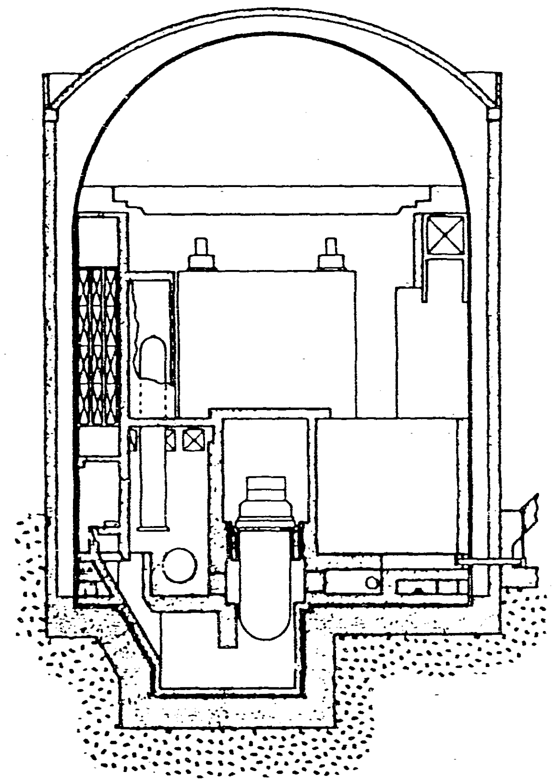
WITH RESPECT TO PUBLIC SAFETY (FISSION PRODUCT RELEASE), THE CONTAINMENT IN PWR PLANTS HAS BEEN IDENTIFIED AS THE SECOND MOST CRITICAL COMPONENT



LARGE DRY

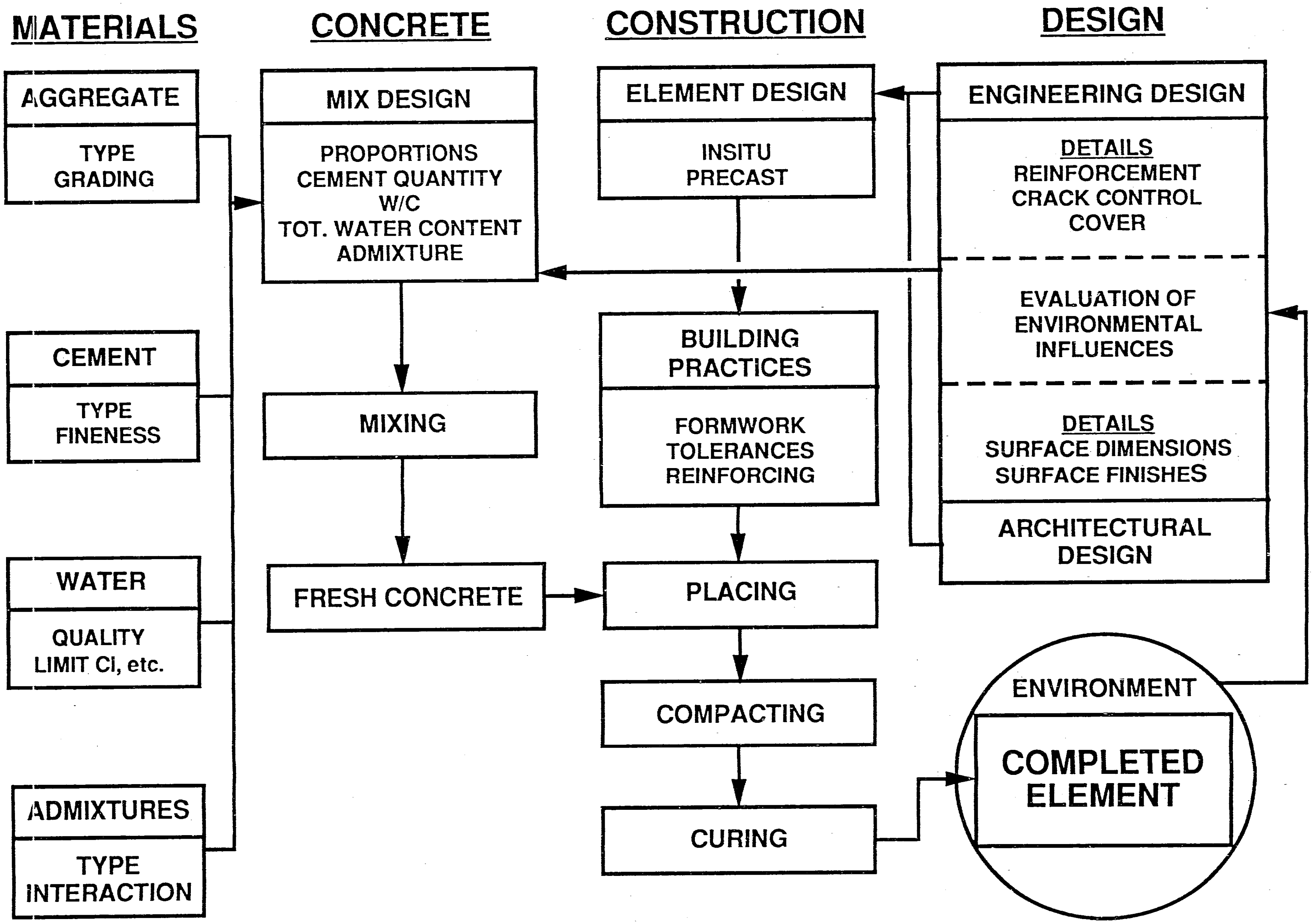


SUB-ATMOSPHERIC



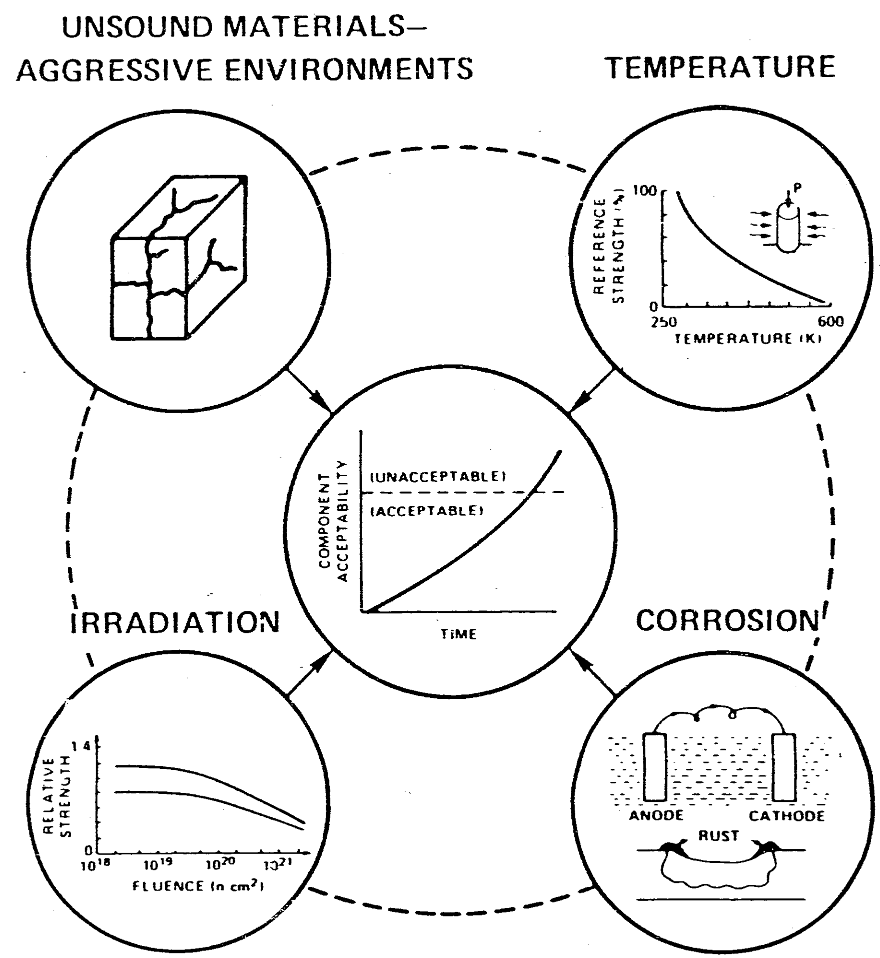
ICE CONDENSER

FACTORS WHICH AFFECT THE DURABILITY OF CONCRETE STRUCTURES ARE MANY, VARIED AND INTERACTIVE



ORNL–DWG 86–4662 ETD

DETERIORATING INFLUENCES CAN IMPACT THE ABILITY OF CONCRETE COMPONENTS TO MEET THEIR FUNCTIONAL AND PERFORMANCE REQUIREMENTS



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MARTIN MARIETTA

MARTIN MARIETTA ENERGY SYSTEMS, INC

SEVERAL POTENTIAL DEGRADATION FACTORS CAN COMPROMISE THE ABILITY OF CONCRETE STRUCTURES IN NPPs TO MEET THEIR FUNCTIONAL AND PERFORMANCE REQUIREMENTS

CONCRETE DEGRADATION MODES

MATERIAL INTERACTIONS

- Alkali-Silicate Reactions
- Alkali-Carbonate Reactions

SHORT-TERM EXPOSURES

- Fire Damage
- Temperature Shock
- Chemical Spills
- Earthquakes
- Tornados, Hurricanes, Tsunamis
- Missile Loads
- Design Basis Accidents
- Mechanical Overloading

AGGRESSIVE ENVIRONMENTS

- Freeze/Thaw
- Wetting and Drying
- Reinforcement Corrosion
- Abrasion
- Concrete Erosion
- Cavitation
- Acid Rain
- Temperature Changes
- Irradiation
- Seepage
- Fatigue
- Chemical Attack

ENVIRONMENT-STRESS INTERACTIONS

- Alkali-Aggregate Reactions
- Leaching of Calcium Hydroxide
- Freeze/Thaw
- Wetting and Drying
- Reinforcement Corrosion

SEVERAL INSTANCES OF STRUCTURAL DEGRADATION HAVE OCCURRED AT NPPS

CORROSION

TENDONS:	FORT ST. VRAIN
ANCHORHEADS:	FARLEY, BYRON, BELLEFONTE
STEEL CONTAINMENT:	OYSTER CREEK, MONTICELLO, MCGUIRE 2, NINE MILE POINT 2
STEEL EQUIPMENT SUPPORTS:	SAN ONOFRE 1
REBAR:	SAN ONOFRE INTAKE STRUCTURE

OTHER TENDON PROBLEMS

EXCESS RELAXATION:	GINNA
WATER IN TENDON DUCTS:	MILLSTONE

CONCRETE PROBLEMS

SWELLING:	WOLSUNG 1
CRACKING:	GENTILLY 2
DESIGN / CONSTRUCTION QUALITY:	MARBLE HILL, WATTS BAR, CRYSTAL RIVER 3, TURKEY POINT 3, CALVERT CLIFFS, OTHERS

PRESENTATION WILL ADDRESS FOUR TOPICS

- INTRODUCTION
- **BACKGROUND**
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STRUCTURAL AGING PROGRAM HAS EVOLVED FROM TWO PRIOR PROGRAMS AT ORNL

- EPRI NP-4208, "*THE LONGEVITY OF NUCLEAR POWER SYSTEMS*"
    - ◊ APPENDIX A- "*REACTOR PRESSURE VESSEL LIFE EXTENSION STUDY*"
    - ◊ APPENDIX B- "*CONCRETE MATERIAL SYSTEMS IN NUCLEAR SAFETY RELATED STRUCTURES- A REVIEW OF FACTORS RELATED TO THEIR DURABILITY, DEGRADATION DETECTION AND EVALUATION, AND REMEDIAL MEASURES FOR AREAS OF DISTRESS*"
- NRC NUCLEAR PLANT AGING RESEARCH (NPAR) PROGRAM
    - ◊ NPAR PROGRAM STRATEGY WAS APPLIED TO SAFETY-RELATED CONCRETE COMPONENTS
    - ◊ NUREG/CR-4652, "*CONCRETE COMPONENT AGING AND ITS SIGNIFICANCE RELATIVE TO LIFE EXTENSION OF NUCLEAR POWER PLANTS*"

SEVERAL BARRIERS EXIST RELATIVE TO BEING ABLE TO PREDICT FUTURE PERFORMANCE OF CRITICAL CONCRETE STRUCTURES

- A SYSTEMATIC APPROACH OR METHODOLOGY FOR TREATING THE PROBLEM
- AN EFFECTIVE MECHANISM FOR OBTAINING AND REPORTING DATA ON THE ACTUAL INSERVICE PERFORMANCE OF MATERIALS
- KNOWLEDGE OF THE MECHANISMS OF DEGRADATION
- KNOWLEDGE OF THE ENVIRONMENTAL FACTORS CAUSING DEGRADATION
- THE ABILITY TO SIMULATE OR ACCOUNT FOR THE SYNERGISM BETWEEN DEGRADATION FACTORS
- AVAILABILITY OF MATHEMATICAL MODELS DESCRIBING MATERIAL BEHAVIOR IN SPECIFIC ENVIRONMENTS OR APPLICATIONS

PRESENTATION WILL ADDRESS FOUR TOPICS

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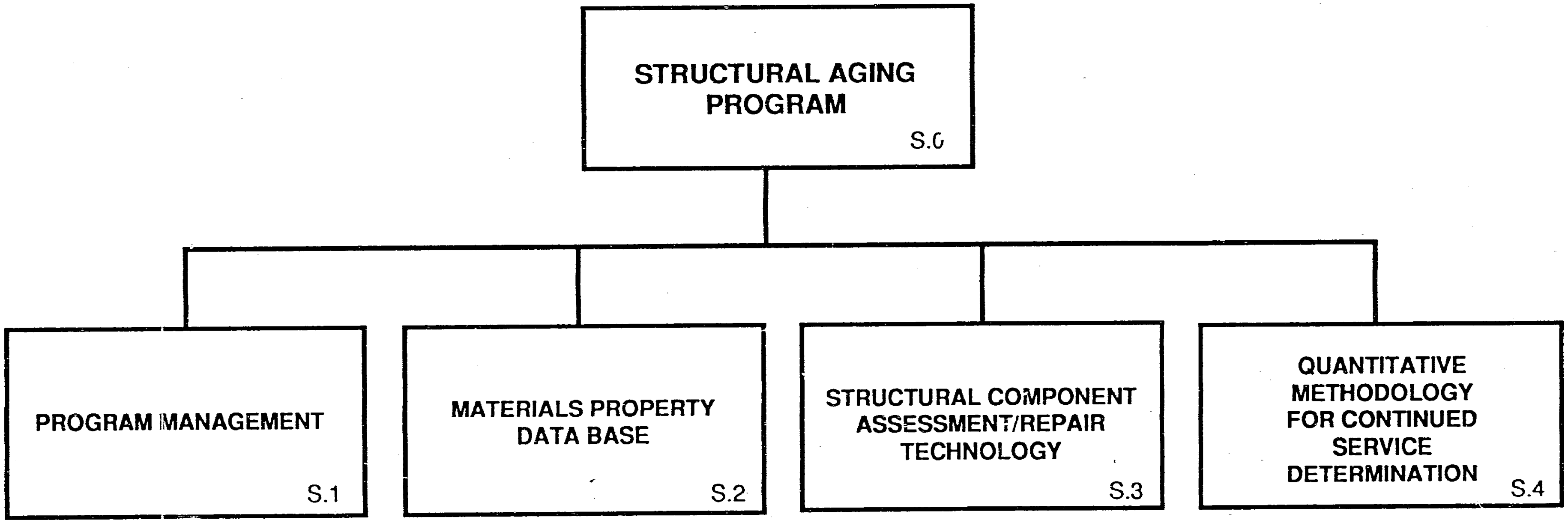
OVERALL OBJECTIVE OF THE PROGRAM IS TO PROVIDE NRC WITH STRUCTURAL SAFETY ISSUES AND ACCEPTANCE CRITERIA FOR USE IN NPP EVALUATIONS FOR CONTINUED SERVICE

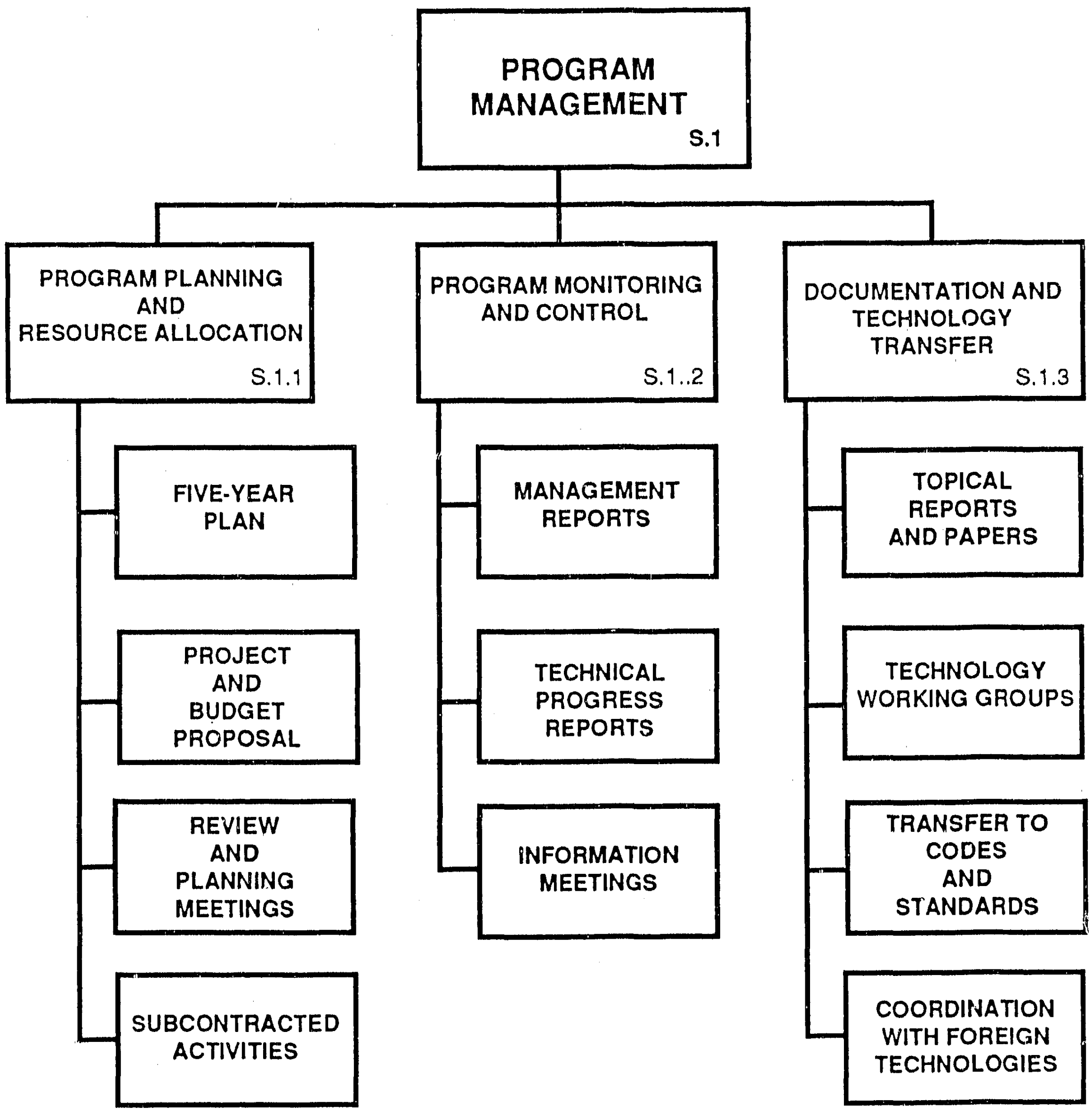
FINAL PRODUCT WILL BE AN EXPANDABLE HANDBOOK (OR REPORT) WHICH WILL PROVIDE NRC LICENSE REVIEWERS AND LICENSEES WITH:

- IDENTIFICATION AND EVALUATION OF THE DEGRADATION PROCESSES THAT AFFECT THE PERFORMANCE OF STRUCTURAL COMPONENTS
- ISSUES TO BE ADDRESSED UNDER NUCLEAR POWER PLANT CONTINUED SERVICE REVIEWS, AS WELL AS CRITERIA, AND THEIR BASES, FOR RESOLUTION OF THESE ISSUES
- IDENTIFICATION AND EVALUATION OF RELEVANT INSERVICE INSPECTION OR STRUCTURAL ASSESSMENT PROGRAMS IN USE, OR NEEDED
- METHODOLOGIES REQUIRED TO PERFORM CURRENT ASSESSMENTS AND RELIABILITY-BASED LIFE PREDICTIONS OF SAFETY-RELATED CONCRETE STRUCTURES

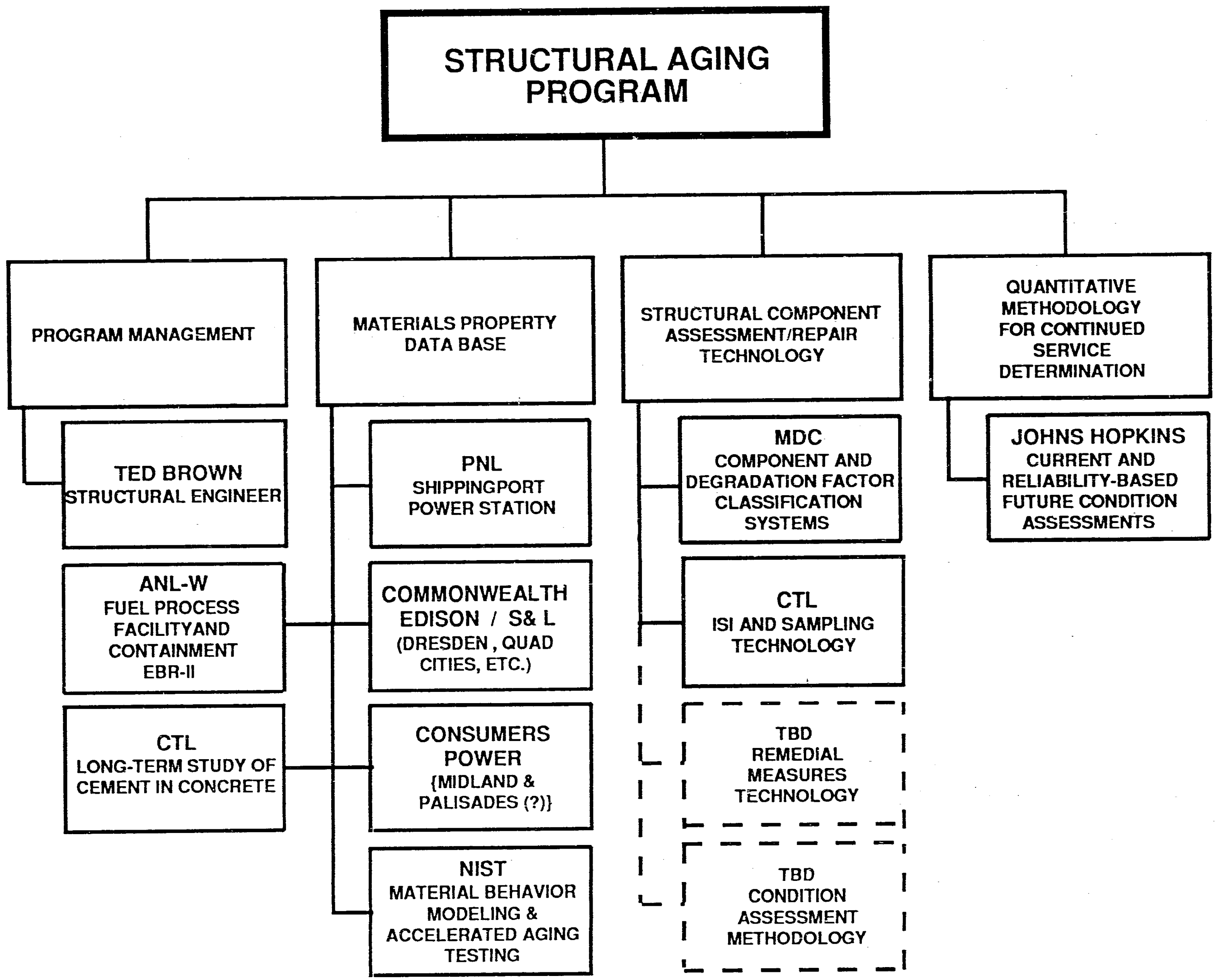
STRUCTURAL AGING PROGRAM CONSISTS OF THREE TECHNICAL TASKS AND A MANAGEMENT TASK



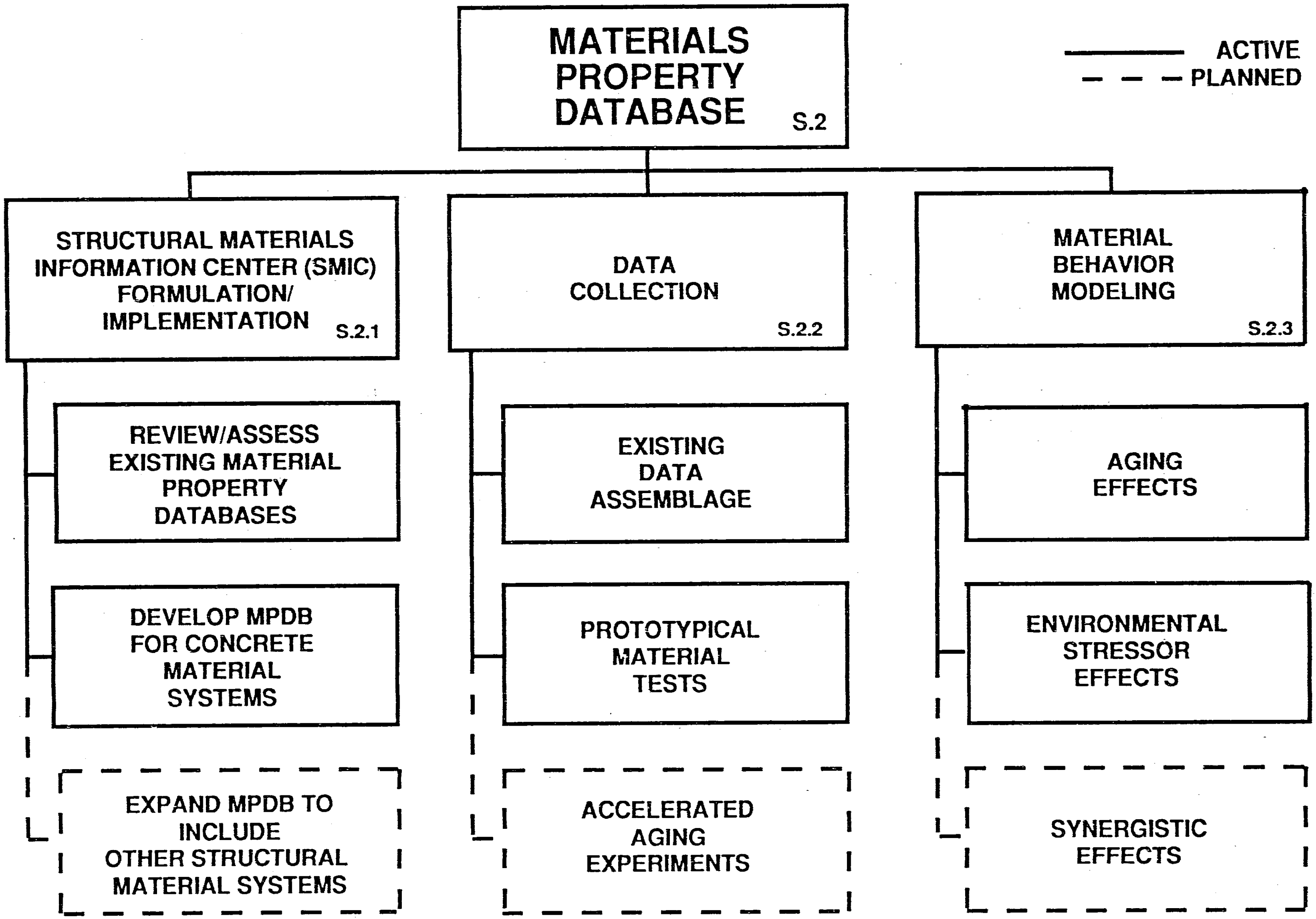
OBJECTIVE OF TASK S.1 IS TO EFFECTIVELY MANAGE THE TECHNICAL TASKS UNDERTAKEN TO ADDRESS PRIORITY STRUCTURAL SAFETY ISSUES RELATED TO NPP CONTINUED SERVICE APPLICATIONS



STRUCTURAL AGING PROGRAM RELIES HEAVILY ON SUBCONTRACTED ACTIVITIES TO TAKE ADVANTAGE OF EXPERTISE AVAILABLE AT OTHER LOCATIONS



OBJECTIVE OF TASK S.2 IS TO DEVELOP A COMPUTER-BASED STRUCTURAL MPDB WHICH WILL CONTAIN INFORMATION ON THE TIME VARIATION OF MATERIAL PROPERTIES UNDER THE INFLUENCE OF PERTINENT ENVIRONMENTAL STRESSORS AND AGING FACTORS



A DATA BASE FOR USE IN DEVELOPMENT OF INTEGRATED MATERIALS KNOWLEDGE SYSTEMS TO PREDICT CHEMICAL, PHYSICAL AND MECHANICAL RESPONSE OF MATERIALS OR STRUCTURES TO VARIOUS STIMULI IS BEING DEVELOPED UNDER TASK S.2

- SMIC FORMULATION/IMPLEMENTATION
  - ◊ COMPLETED EVALUATION OF PERTINENT DATA BASES, HARDWARE AND SOFTWARE
    - UNITED KINGDOM (CEGB, TWC, AERE, BCA)
    - FEDERAL REPUBLIC OF GERMANY (BAM, TUM, AACHEN)
    - FRANCE (EDF, CEN)
    - JAPAN (JAPAN POWER ENGINEERING AND INSPECTION)
    - ITALY (ENEL)
    - REPUBLIC OF CHINA (TAIWAN POWER CO.)
    - KOREA (KOREA POWER ENGINEERING CO.)
    - CANADA (UNIVERSITY OF TORONTO, ONTARIO HYDRO)
    - UNITED STATES (NIST, WES, MRL, BUREAU OF MINES, PCA, ETC.)

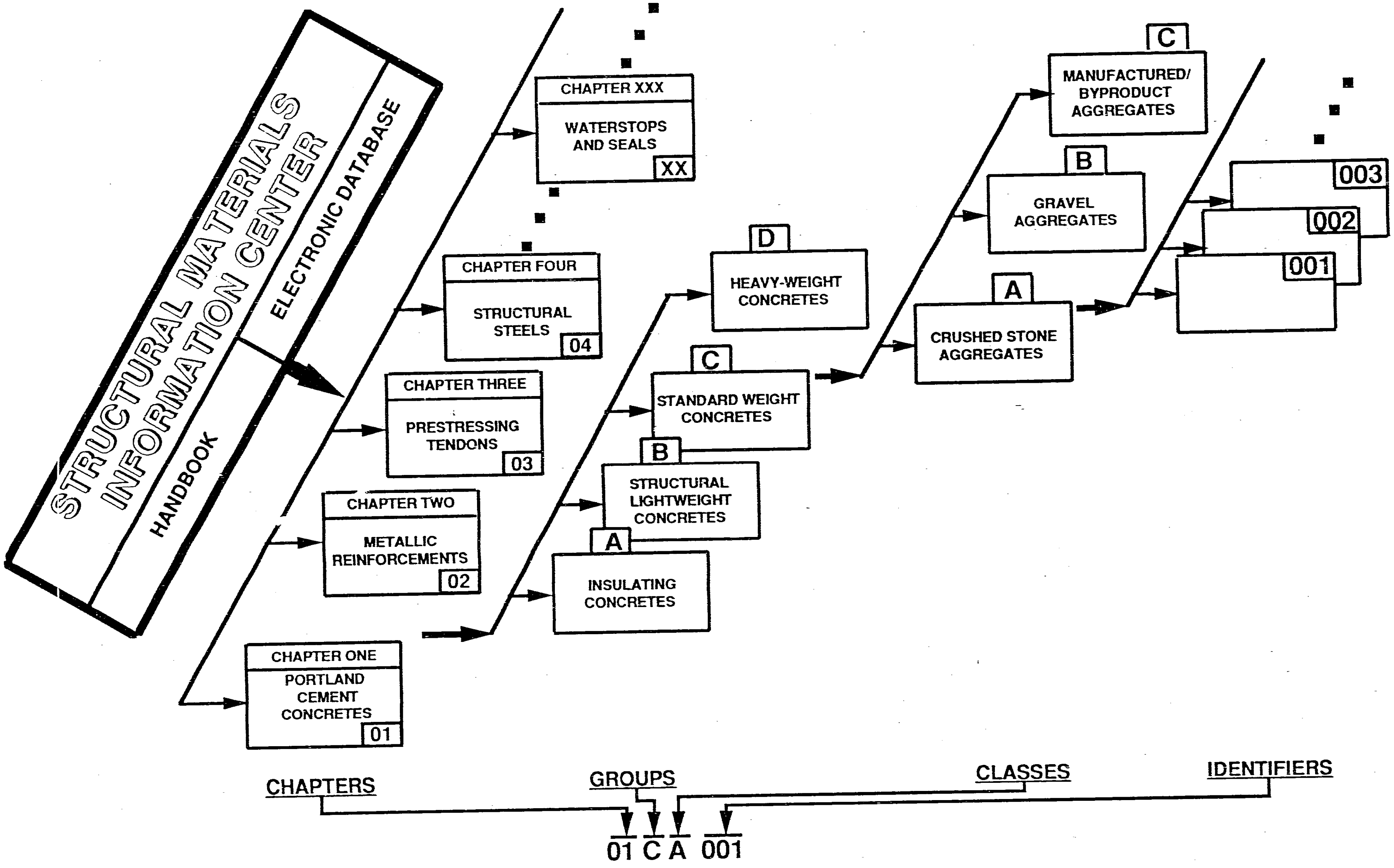
A DATA BASE FOR USE IN DEVELOPMENT OF INTEGRATED MATERIALS KNOWLEDGE SYSTEMS TO PREDICT CHEMICAL, PHYSICAL AND MECHANICAL RESPONSE OF MATERIALS OR STRUCTURES TO VARIOUS STIMULI IS BEING DEVELOPED UNDER TASK S.2 (CONT.)

◊ COMPLETED REPORT "A REVIEW AND ASSESSMENT OF MATERIALS PROPERTY DATABASES WITH PARTICULAR REFERENCE TO CONCRETE MATERIAL SYSTEMS (ORNL/NRC/LTR - 89/3)"

◊ PROCURED MatDB/EnPlot PACKAGE AND COMPLETED EVALUATION OF ITS CAPABILITY FOR HANDLING CONCRETE DATA

◊ COMPLETED REPORT "PLAN FOR USE IN DEVELOPMENT OF THE STRUCTURAL MATERIALS INFORMATION CENTER (ORNL/NRC/LTR - 89/8)"

MATERIAL PROPERTIES DATABASE IS BEING FORMULATED SO THAT INFORMATION CAN BE SELECTED BASED ON SEVERAL PARAMETERS

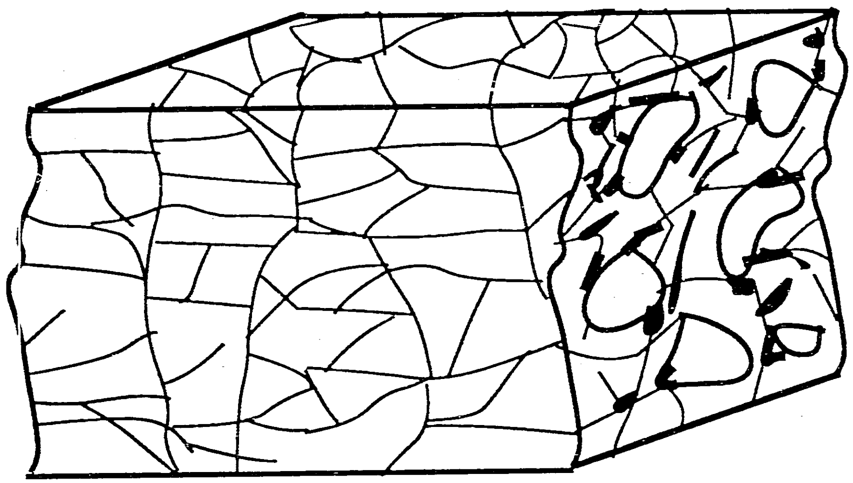


THREE PRIMARY APPROACHES ARE BEING UTILIZED TO DEVELOP "REPRESENTATIVE" MATERIALS PROPERTY DATA FOR INPUT INTO THE SMIC

- "OPEN LITERATURE" REPORTS
- PROCUREMENT AND TESTING OF SAMPLES FROM AGED CONCRETE STRUCTURES
- ACCELERATED AGING TECHNIQUES

PREDICTION OR EXPLANATION OF THE COMPLEX INTERRELATIONSHIPS THAT OCCUR BETWEEN CONCRETE'S CONSTITUENTS AND BETWEEN CONCRETE AND ITS ENVIRONMENT REQUIRES THE DEVELOPMENT OF MATHEMATICAL MODELS

AGING EFFECTS

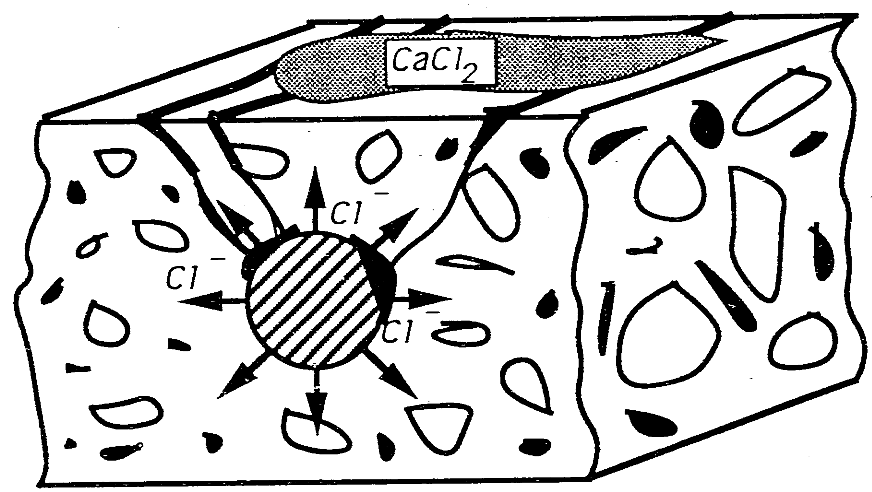


ALKALI - AGGREGATE REACTION

$$x = t_0' + kt_1^a$$

$(a > 1)$

ENVIRONMENTAL EFFECTS



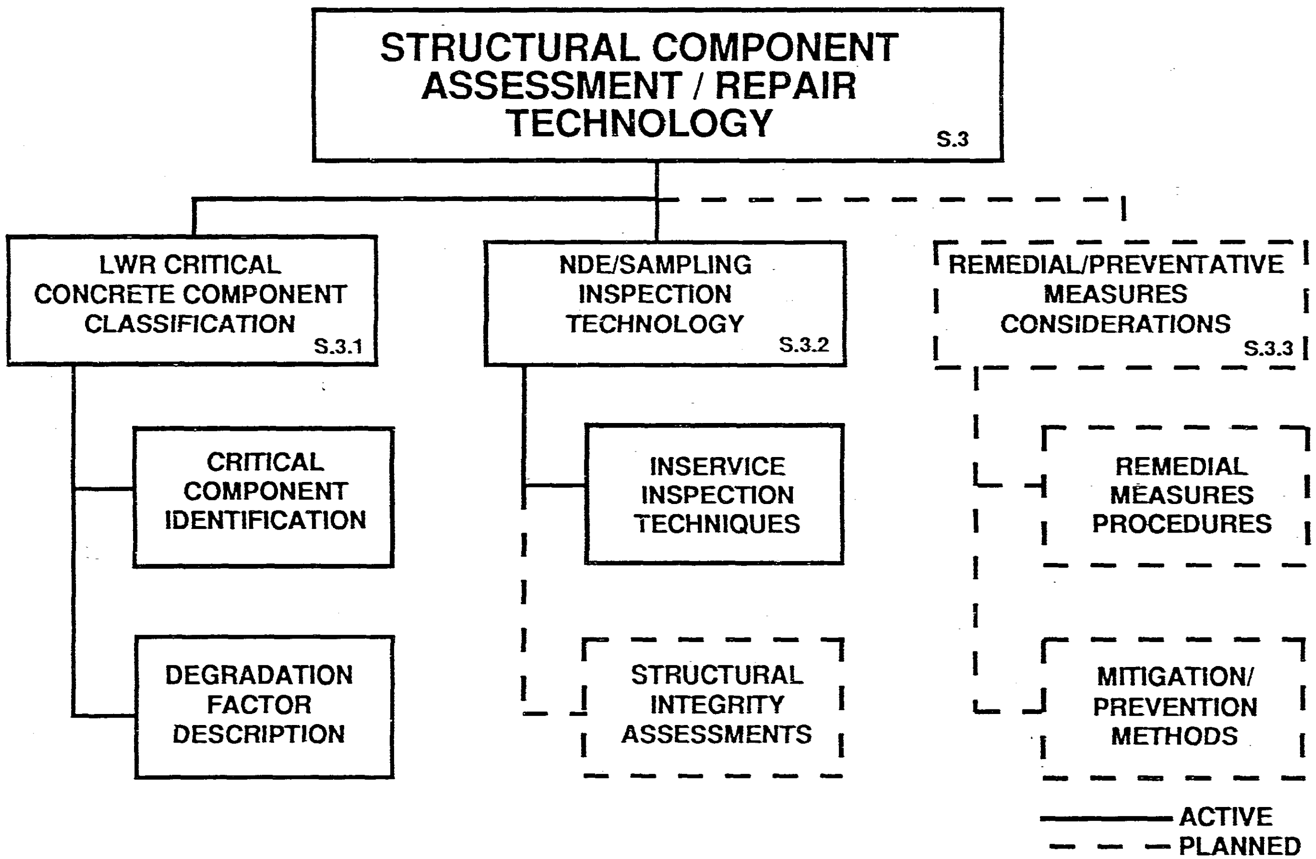
REBAR CORROSION

$$x = kt^{1/2} + kt_1^a$$

($a = 0.5$, diffusion control)
($a = 1.0$, reaction control)

OBJECTIVES OF TASK S.3 ARE TO DEVELOP PROCEDURES TO:

- QUANTITATIVELY ASSESS PRESENCE, MAGNITUDE AND SIGNIFICANCE OF ANY DEGRADATION FACTORS THAT CAN IMPACT DURABILITY
- PROVIDE DATA (ISI OR SAMPLING TECHNIQUES) FOR USE IN CURRENT OR FUTURE STRUCTURAL CONDITION ASSESSMENTS



IN ADDITION, TECHNIQUES WILL BE ESTABLISHED FOR:

- MITIGATION OF ENVIRONMENTAL STRESSOR OR AGING FACTOR DETRIMENTAL EFFECTS
- REPAIR, REPLACEMENT OR RETROFITTING OF DEGRADED CONCRETE COMPONENTS

PERTINENT CONCRETE STRUCTURAL COMPONENTS AND THE FACTORS WHICH CAN IMPACT THEIR PERFORMANCE ARE BEING ESTABLISHED UNDER TASK S.3

- LWR CRITICAL CONCRETE COMPONENT CLASSIFICATION

  ◊ CONCRETE COMPONENTS CLASSIFICATION SYSTEM

    - IDENTIFY CRITICAL SAFETY-RELATED STRUCTURAL COMPONENTS IN LWR FACILITIES

    - DESCRIBE COMPONENTS AND LIST THEIR FUNCTIONAL AND PERFORMANCE REQUIREMENTS

    - CHARACTERIZE COMPONENTS IN TERMS OF THEIR RELATIVE IMPORTANCE TO SAFETY (i.e., RATING SYSTEM)

PERTINENT CONCRETE STRUCTURAL COMPONENTS AND THE FACTORS WHICH CAN IMPACT THEIR PERFORMANCE ARE BEING ESTABLISHED UNDER TASK S.3 (CONT.)

- LWR CRITICAL CONCRETE COMPONENT CLASSIFICATION (CONT.)

  ◊ DEGRADATION FACTOR DESCRIPTION

    - IDENTIFY EXPECTED TYPE AND RANGE OF DEGRADATION FACTORS (AGING, ENVIRONMENTAL STRESSOR, OPERATION, TESTING, etc.) THAT CAN IMPACT EACH COMPONENT

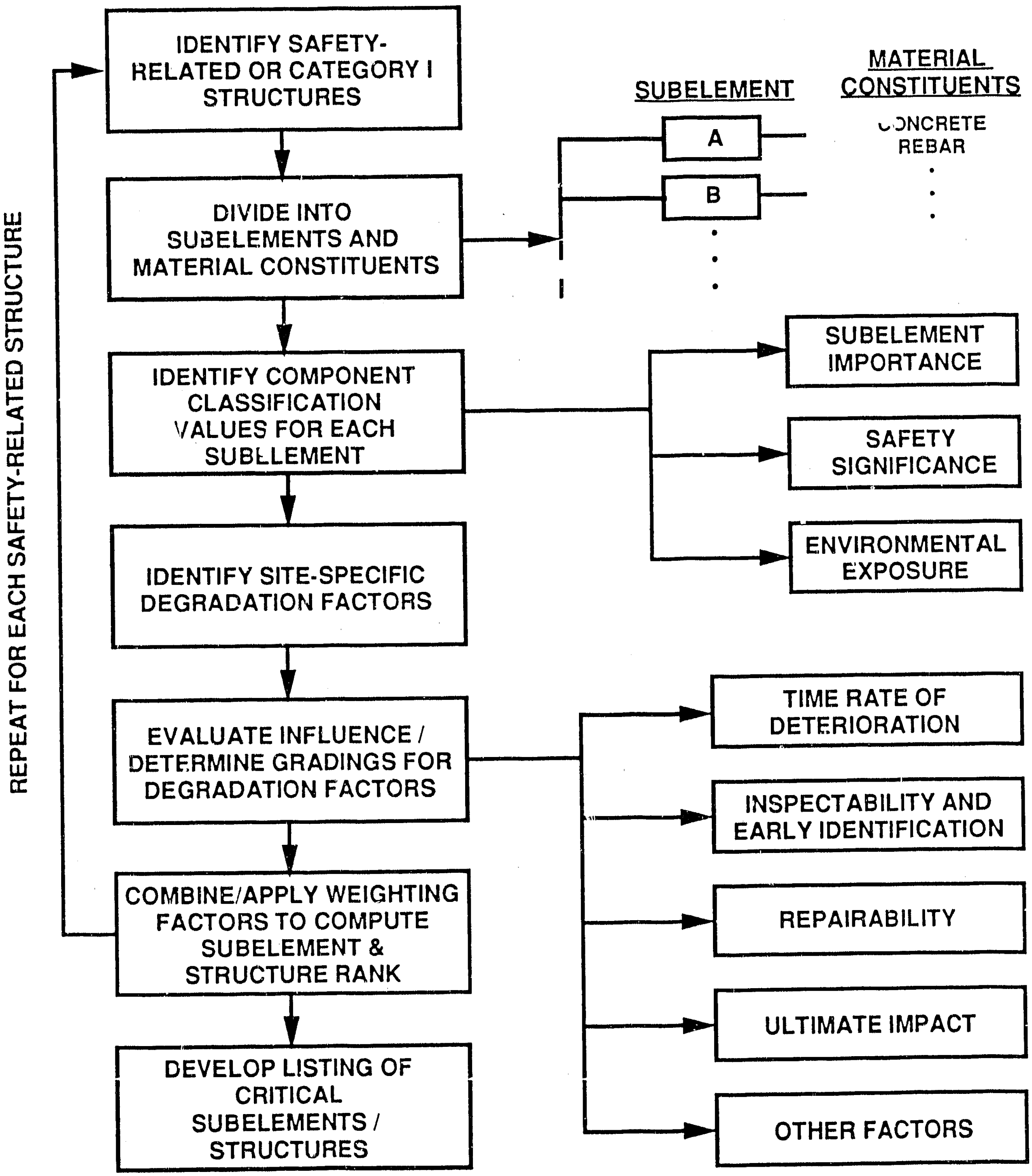
    - IDENTIFY CRITICAL PERFORMANCE CHARACTERISTICS AND PROPERTIES WHICH CAN SERVE AS INDICATORS OF DEGRADATION SEVERITY

    - QUANTIFY DEGRADATION FACTOR SIGNIFICANCE THROUGH DEVELOPMENT OF A GRADING SYSTEM BASED ON INTENSITY OR DEGREE (RATE) OF DETERIORATION

STRUCTURAL COMPONENT SIGNIFICANCE ASSESSMENT (RANKING) METHODOLOGY UTILIZES FOUR CRITERIA TO INCORPORATE INTERACTION OF AGING EFFECTS

- FUNCTIONAL IMPORTANCE OF SUBELEMENTS
- SAFETY SIGNIFICANCE
- ENVIRONMENTAL EXPOSURE
- DEGRADATION FACTOR SIGNIFICANCE

STRUCTURES AND SUBELEMENTS MOST IMPORTANT TO AGING ARE IDENTIFIED THROUGH CALCULATION OF SUBELEMENT RANK AND CUMULATIVE STRUCTURE RANK



A MATRIX FORMAT IS USED TO PRESENT STRUCTURAL COMPONENT SIGNIFICANCE METHODOLGY RESULTS

PRIMARY STRUCTURE ---------- CUMULATIVE RANK ------ FUNCTION ------------------------------

SUBELEMENT -------------------- IMPORTANCE ------- ENVIRONMENT EXPOSURE ------- SAFETY SIGNIFICANCE ------- SUBELEMENT RANK -------

COMPOSITION --------------------

KEY DEGRADATION FACTORS --------------- RATING -----

SUBELEMENT -------------------- IMPORTANCE ------- ENVIRONMENT EXPOSURE ------- SAFETY SIGNIFICANCE ------- SUBELEMENT RANK -------

COMPOSITION --------------------

KEY DEGRADATION FACTORS --------------- RATING -----

STRUCTURAL COMPONENT SIGNIFICANCE METHODOLOGY HAS BEEN APPLIED TO A PWR PLANT WITH A LARGE-DRY METAL CONTAINMENT LOCATED IN THE MIDWEST

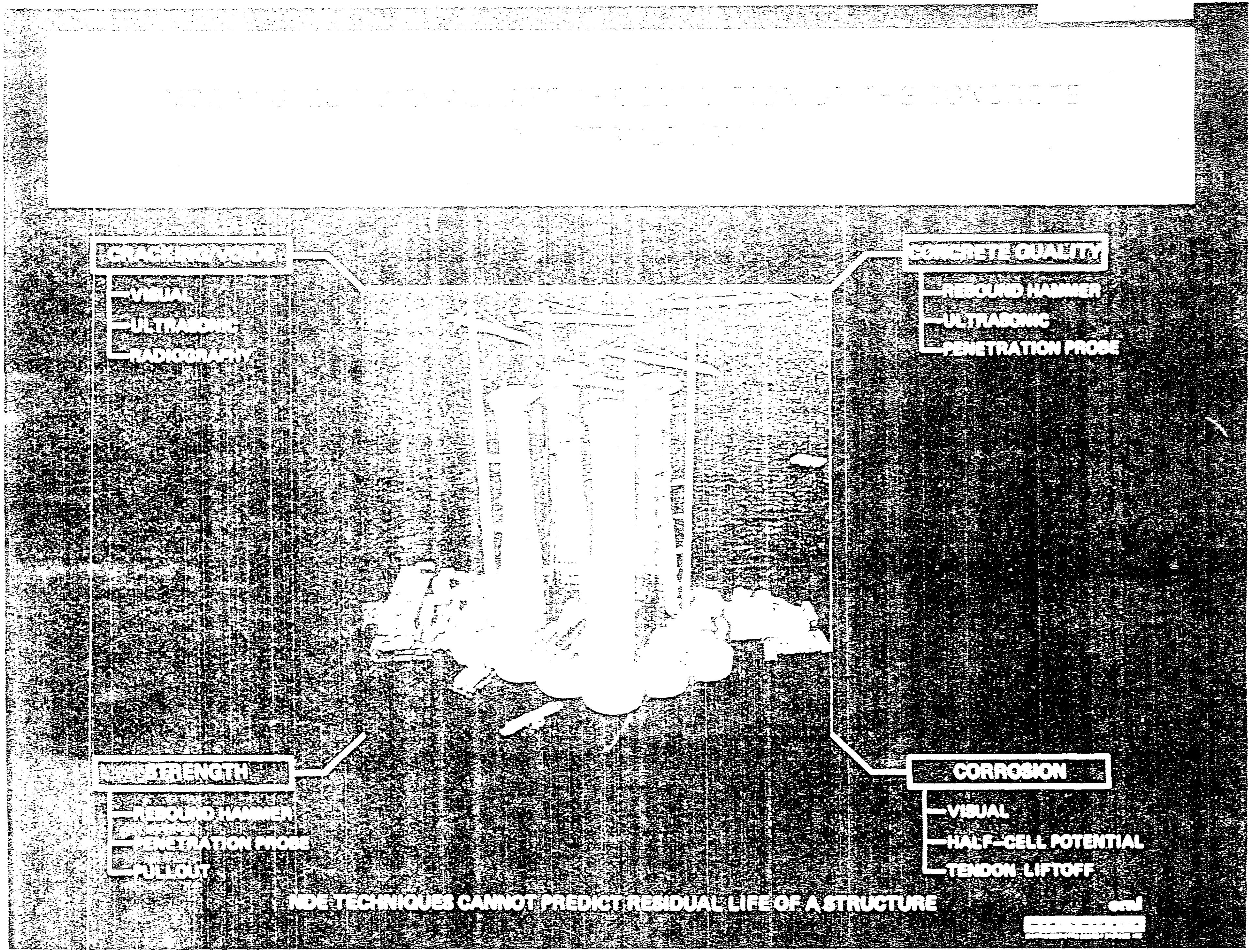
PRIMARY STRUCTURE	SUBELEMENT	SUB-ELEMENT RANK	CUMU-LATIVE RANK
CONTAINMENT - INTERNAL STRUCTURES			157
	REACTOR CAVITY WALLS	180	
	BOTTOM SLAB	171	
	REACTOR COOLANT PUMP VAULT WALLS	158	
	•		
	•		
SHIELD BUILDING			158
	FOUNDATION	172	
	SHIELD WALLS (VERTICAL)	164	
	DOME		156
	•		
	•		

STRUCTURAL COMPONENT SIGNIFICANCE METHODOLOGY HAS BEEN APPLIED TO A BWR PLANT WITH A MARK II REINFORCED CONCRETE CONTAINMENT LOCATED IN THE EAST

PRIMARY STRUCTURE	SUBELEMENT	SUB-ELEMENT RANK	CUMU-LATIVE RANK
CONTAINMENT VESSEL			178
	BASE SLAB	182	
	VERTICAL AND INCLINED WALLS (SUPPRESSION CHAMBER & DRYWELL)	174	
CONTAINMENT - INTERNAL STRUCTURES			162
	DIAPHRAGM SLAB	156	
	RPV PEDESTAL	171	
	PRIMARY SHIELD WALL	159	
• • •			

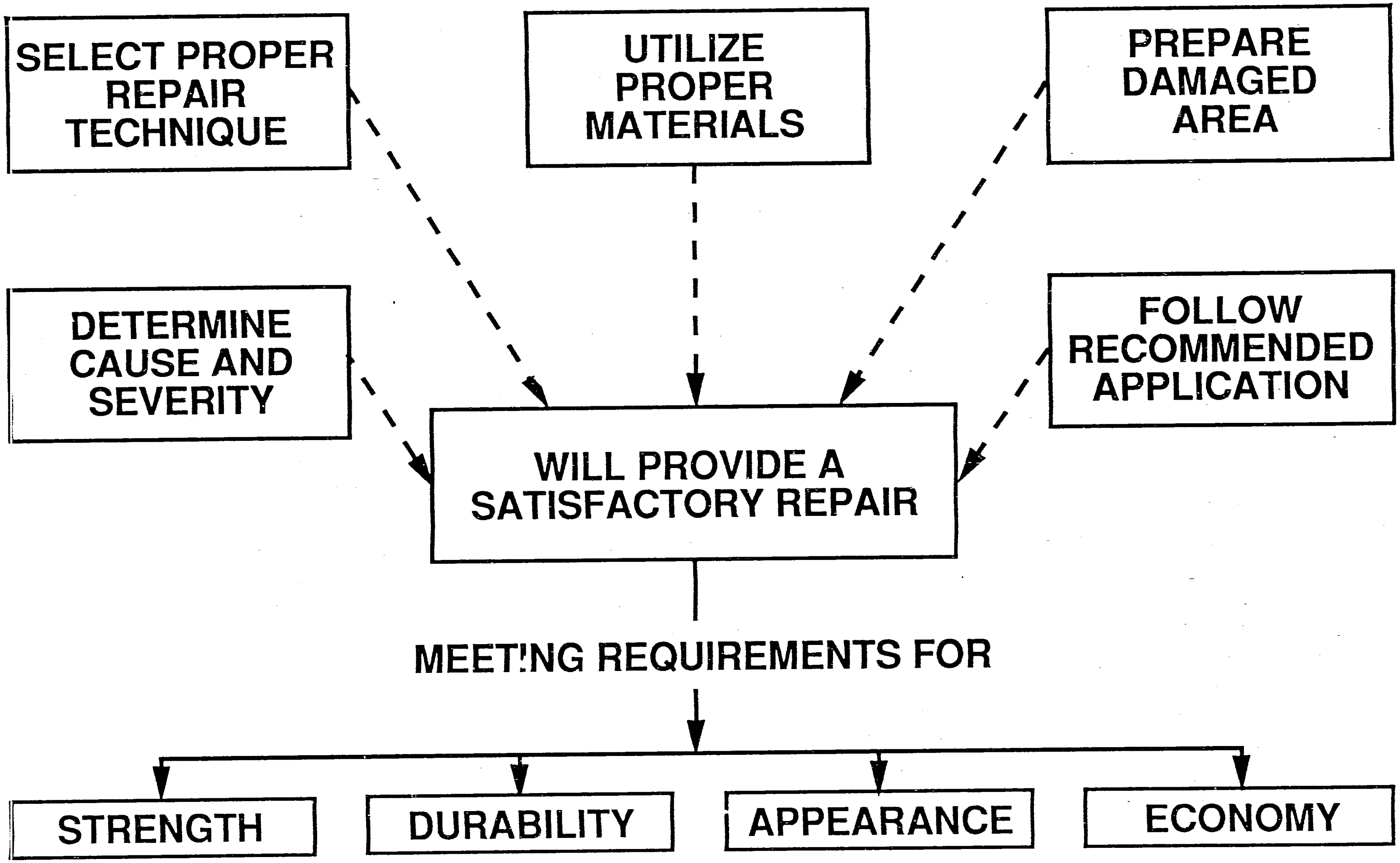
STRUCTURAL COMPONENT SIGNIFICANCE METHODOLOGY HAS BEEN APPLIED TO A PWR PLANT WITH A LARGE-DRY PRESTRESSED CONCRETE CONTAINMENT LOCATED IN THE MIDWEST

PRIMARY STRUCTURE	SUBELEMENT	SUB-ELEMENT RANK	CUMU-LATIVE RANK
CONTAINMENT VESSEL			172
	DOME	178	
	VERTICAL WALLS (INC. BUTTRESSES)	182	
	MAT FOUNDATION	200	
	•		
	•		
CONTAINMENT - INTERNAL STRUCTURES			153
	BOTTOM SLAB (ABOVE LINER PLATE)	151	
	POLAR CRANE SUPPORT WALL	144	
	FLOOR SLABS	131	
	•		
	•		

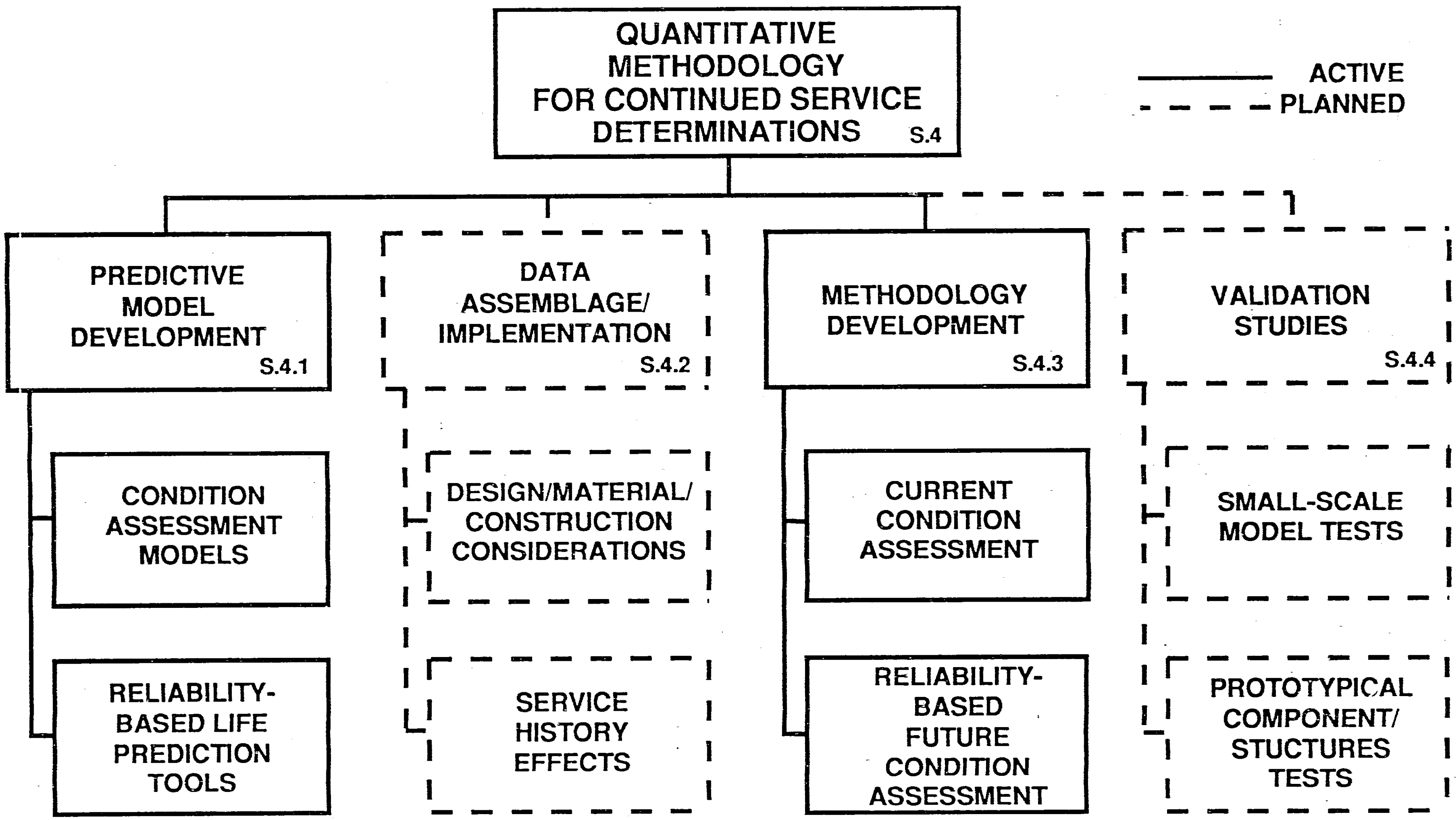

CRACKING/VOIDS
VISUAL
ULTRASONIC
RADIOGRAPHY
CONCRETE QUALITY
REBOUND HAMMER
ULTRASONIC
PENETRATION PROBE
STRENGTH
REBOUND HAMMER
PENETRATION PROBE
PULLOUT
CORROSION
VISUAL
HALF-CELL POTENTIAL
TENDON LIFTOFF
NDE TECHNIQUES CANNOT PREDICT RESIDUAL LIFE OF A STRUCTURE
ornl

OBJECTIVES OF REMEDIAL WORK INCLUDE:

- RESTORE COMPONENT'S STRUCTURAL INTEGRITY
- ARREST MECHANISM(S) PRODUCING DISTRESS
- ENSURE (AS FAR AS POSSIBLE) THAT CAUSE OF DISTRESS WILL NOT REOCCUR

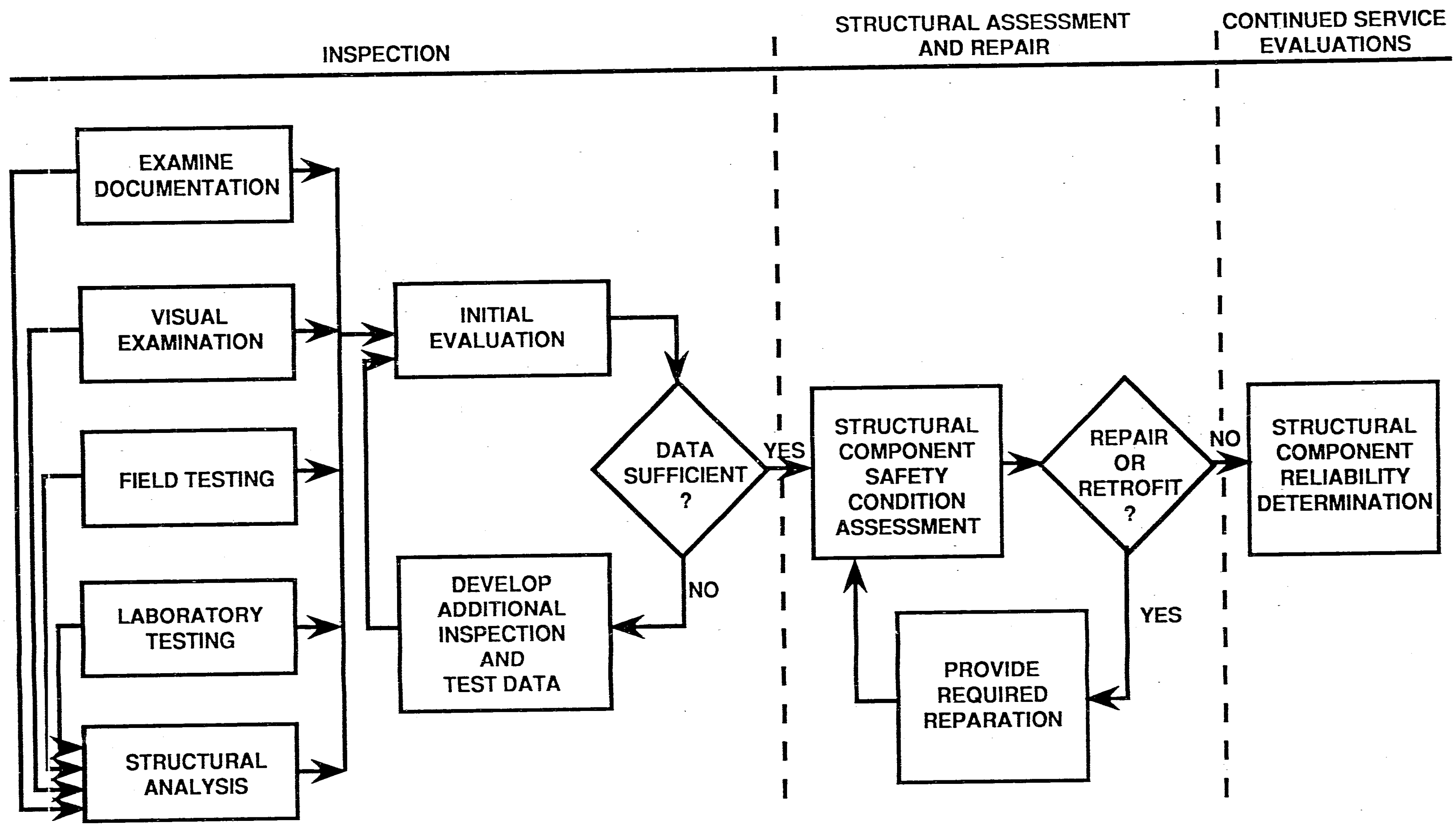


OBJECTIVE OF TASK S.4 IS TO DEVELOP A METHODOLOGY WHICH CAN BE USED FOR PERFORMING CONDITION ASSESSMENTS AND MAKING RELIABILITY-BASED LIFE PREDICTIONS OF CRITICAL CONCRETE STRUCTURES



EVIDENCE WILL BE PROVIDED WHETHER STRUCTURES IN THEIR CURRENT CONDITION WILL BE ABLE TO WITHSTAND POTENTIAL FUTURE DESIGN EVENTS WITH A LEVEL OF RELIABILITY ADEQUATE TO MEET REQUIREMENTS FOR PROTECTING PUBLIC HEALTH AND SAFETY

INFORMATION ON DEGRADATION AND DAMAGE ACCUMULATION, ENVIRONMENTAL FACTORS, AND LOAD HISTORY WILL BE INTEGRATED INTO A DECISION TOOL



METHODOLOGY WILL PROVIDE A QUANTITATIVE MEASURE OF STRUCTURAL RELIABILITY AND PERFORMANCE UNDER PROJECTED FUTURE SERVICE REQUIREMENTS BASED ON THE CONDITION OF THE EXISTING STRUCTURE

PRESENTATION WILL ADDRESS FOUR TOPICS

- INTRODUCTION
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SEVERAL PRODUCTS USEFUL TO BOTH THE NUCLEAR AND GENERAL CIVIL ENGINEERING COMMUNITIES WILL RESULT FROM THE STRUCTURAL AGING PROGRAM

- FORMULATION AND IMPLEMENTATION OF A "PC-BASED" STRUCTURAL MATERIALS INFORMATION CENTER
- DEVELOPMENT OF CRITERIA FOR ISI/STRUCTURAL ASSESSMENT PROCEDURE(S) TO PROVIDE DATA NEEDED TO EVALUATE THE CURRENT CONDITION AND TO TREND PERFORMANCE OF CRITICAL CONCRETE COMPONENTS
- DEVELOPMENT OF TECHNIQUES TO IDENTIFY, ASSESS, AND PREDICT RATE EFFECTS OF ENVIRONMENTAL STRESSORS AND AGING FACTORS THAT CAN LEAD TO CONCRETE DETERIORATION, AS WELL AS PROCEDURES TO MITIGATE THESE EFFECTS

- END -

DATE FILMED

11 / 16 / 90