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OVERVIEW OF THE AGE-RELATED DEGRADATION OF NUCLEAR POWER PLANT STRUCTURES

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ABSTRACT

License renewal of nuclear power plants is an issue of increasing interest to the U.S. nuclear industry and the U.S. NRC. This paper presents and evaluates the plausible age-related degradation mechanisms that may affect the concrete and steel containment structures and other Class I structures to continue to perform their safety functions. Preventive and/or mitigative options are outlined for managing degradation mechanisms that could significantly affect plant performance during the license renewal period. The provided technical information and the degradation management options may be used as references for comparison with plant specific conditions to ensure that age-related degradation is controlled during the license renewal term.

Plausible degradation mechanisms described and analyzed as they may affect the concrete, reinforcing steel, containment steel shell, prestressed-tendon, steel liner and other structural components typically used in Class I structures. The significance of these age-related degradation mechanisms to the structural components are evaluated, giving consideration to the design basis and quality of construction; typical service conditions; operating and maintenance history; and current test, inspection and refurbishment practices for containment and Class I structures.

Degradation mechanisms which cannot be generically dispositioned on the basis of the two-step approach: (1) they will not cause significant degradation, or (2) any potential degradation will be bounded by current test, inspection, analytical evaluation, and/or refurbishment programs are identified.

Aging degradation management measures are recommended to address the remaining age-related degradation mechanisms. A three-phase approach for the management of the containment and Class I structures is introduced. Various techniques, testing tools and the acceptable criteria for each step of the evaluation of the structures status are provided. The preventive and mitigative measures to limit further degradation are also provided.

1.0 BACKGROUND

A number of Industry Reports, co-sponsored by the Electric Power Research Institute (EPRI) and the United States Department of Energy (DOE), have been prepared by nuclear industry in the U.S. These reports provide the generic technical information for use by the utilities in the license renewal process. Bechtel has been an active participant in this effort, especially in the evaluation of nuclear power plant safety-related structures.

This paper provides a brief discussion on evaluation of the age-related degradation mechanisms (ARDM) which may affect the performance of the structural component for its intended design function. The structures or the structural components may serve a single or multiple safety functions as specified in NRC's criteria of important to license renewal (ITLR). The structural components which constitute the safety-related structures including containment structure and Class I structures, are categorized based on the materials, environment and/or designated function. For each of the component-degradation mechanism combination, evaluation is performed and generic conclusion as to the significance of the plausible ARDM is presented. Preventive and/or mitigative options, for managing degradation mechanisms that could significantly affect plant operation during the license renewal period are outlined. The provided technical information and the degradation management options may be used as references for comparison with plant specific conditions to ensure that the relevant ARDMs are controlled during the license renewal term.

Components such as grout and sealant, which may form an integral part of pressure boundaries, are treated as architectural components and are not discussed in this paper.

2.0 AGE-RELATED DEGRADATION MECHANISMS

The following set of ARDMs relevant to structural components of safety-related structures are derived from a review/evaluation of component service experience, relevant laboratory data, and related experience from other industries.

Concrete

- Freeze-thaw
- Leaching of calcium hydroxide
- Aggressive chemical
- Reaction with aggregates
- Elevated temperature
- Irradiation
- Creep
- Shrinkage
- Abrasion and cavitation
- Cracking of masonry block walls

Reinforcing Steel/Embedded Steel

- Corrosion
- Irradiation

Structural Steel

- Corrosion
- Irradiation

Steel Liner/Free-Standing Steel Containment

- Strain aging
- Corrosion

Prestressing System

- Corrosion
- Prestressing losses

Miscellaneous

- Settlement

For each of the ARDM-structural component combinations listed in the matrix, assessment of the significance must be made by applicants for license renewal by comparison of the design conditions and the environment pertaining to their specific plants. The potential ARDMs are considered significant when, if allowed to continue without any monitoring program, additional prevention or mitigation measures, it could not be shown that the structural component could continue to perform its designated safety function.

3.0 CONCRETE DEGRADATION MECHANISMS

Freeze-Thaw

Repeated freezing and thawings could cause deterioration of concrete properties. Factors which provide immunity to this effect include adequate entrained air content, low permeability, proper gradation of aggregates with good physical qualities and sufficient curing period of concrete before exposure to freeze-thaw attack. Concrete construction following guidelines addressed in ACI 201.2R (Ref. 1) is generally immune to degradation due to cyclic freezing and thawing. A systematic condition survey by Ontario Hydro on 16 hydraulic structures (Ref. 2) suggests that it would be highly unlikely that the strengths of safety-related structures in a nuclear power plant would be compromised by freeze-thaw action to warrant concern for its structural integrity. Accordingly, freeze-thaw degradation is not considered a significant ARDM to safety-related concrete structures.

Leaching of Calcium-Hydroxide

Leaching of calcium hydroxide is caused by dissolving the calcium-containing products in the concrete and carrying it through thickness of concrete by water. The leaching action of the water can only occur if the water passes through the concrete. Water that merely passes over the surface will not cause significant leaching. Leaching over long periods increases the porosity of concrete, making it more susceptible to other forms of aggressive attack, and reducing its strength.

Dense concrete with a suitable cement content that is well cured ensures concrete of low permeability thus increasing resistance to leaching. Low water-to-cement ratio, smaller coarse aggregate, long curing periods, entrained air, and thorough consolidation all contribute to water tightness. Safety-related concrete structures of a nuclear power plant generally follow the guidelines addressed in ACI 201.2R (Ref. 1) which ensures a concrete of low permeability. Limited cases of leaching were observed due to inadequately prepared construction joints. However, they were discovered during normal plant surveillance and well in advance of it becoming significant. Accordingly leaching of calcium hydroxide is not considered a significant ARDM to safety-related concrete structures.

Aggressive Chemicals

Chemicals with $pH < 5.5$ (Ref. 1) or concentrations above the threshold limits of 500 ppm chlorides (Ref. 3), 1500 ppm sulfates (Ref. 4) are considered aggressive to concrete. Acid attack can increase porosity and permeability of concrete, reduce its alkaline nature at the surface of attack and around rebar, reduce strength and render the concrete subject to further deterioration. Sulfate attack can produce significant expansive stresses within the concrete, leading to cracking, spalling, and strength loss.

Use of adequate cement content, low water-to-cement ratio, and thorough consolidation and curing contribute to low permeability and provide effective protection against aggressive chemicals attack. If the safety-related concrete structures are exposed to the aggressive chemicals for intermittent periods only, then degradation caused by aggressive chemicals will not be significant. If they are exposed to the aggressive chemicals for extended periods, then the degradation caused by aggressive chemicals is potentially significant. The potential areas are the below grade portion of concrete structures depending on the composition of the soil/groundwater. If the concrete structures are exposed to the aggressive groundwater or soil for extended periods, the existing condition of the concrete cannot be easily assessed. This degradation mechanism cannot be dispositioned generically. A phased testing program and ARDM management program are recommended to assess the condition of this portion of concrete structures. The first step in a prudent aging management program would be testing of the chemical quality of the groundwater and the bearing soil media. Only where this testing indicates that the groundwater or soil media is aggressive, will further degradation evaluation be necessary. Follow-on evaluations may include local excavation and examination of the concrete by nondestructive testing methods. Further measures, if degradation is found to be significant, will be plant specific.

Reactive Aggregate

Three basic types of chemical reactions between aggregates and alkalis have been experienced in some concrete structures: alkali-aggregate reaction, cement-aggregate reaction and expansive alkali-carbonate reaction (Ref. 1). They may cause expansion and severe cracking of concrete structures. This phenomenon has been widely known and long understood. Maps and data have been developed identifying certain geographic regions known to yield reactive aggregates. Almost without exception, the design and construction specifications of every nuclear power plant have specified requirements of aggregate selection criteria and testing procedures in order to avoid reactive aggregates. Therefore, although in some structures where safety is not critical, such as highway pavements, have exhibited aggregate reaction degradation, there is no evidence of degradation due to reactive aggregates in safety related structures of a nuclear power plant.

Elevated Temperature

The strength and modulus of elasticity of concrete are reduced when it is subjected to prolonged exposure to elevated temperatures (Ref. 5). Studies have indicated that the degradation begins to occur in the range of 180 to 200 F. Except in a few local areas, concrete of a safety related structure does not normally experience temperatures higher than 120 to 150 F during normal operation. For local areas, such as at high energy pipe penetrations, either high temperature concrete is used or some design means are used to maintain the temperature below the limit of 200 F. Therefore, elevated temperature is not a significant degradation mechanism for concrete used in safety related structures.

Irradiation

Concrete properties may change from prolonged exposure to neutron and/or gamma radiation (Ref. 6, 7, 8, 9). The property changes are caused by aggregate growth, decomposition of water, and thermal warming causing water migration. Concrete can experience decrease in its strength and modulus of elasticity, and reduction of shielding characteristics if there is a significant loss of water through evaporation.

In a typical 1000 MWe nuclear power plant, only primary shield wall of a PWR plant is potentially subjected to a sufficient amount of cumulative neutron flux (5×10^{19} neutrons/cm^2), to reduce the structural capacity after 80 years of plant operation. Comparing the strength demand with the remaining strength capacity of the primary shield wall, there is sufficient safety margin against the most critical loading condition. For both BWR and PWR plants, the maximum integrated gamma doses to the primary shield in 80 years of operation are in the range at which degradation may begin to be measurable (1×10^{10} rads). It has been evaluated that the nominal loss in strength will not compromise the primary shield wall's structural integrity (Ref. 10). Therefore, irradiation is not a significant age-related degradation mechanism.

Creep

Creep in concrete structures occurs under sustained loading. It can result from progressive cracking at the aggregate-cement paste interface, from moisture exchange with the atmosphere, and from moisture movement within the concrete. The magnitude of creep deformation is generally determined by the water-to-cement ratio, aggregate-to-cement paste ratio, stress level, temperature and age of concrete at application of load. Prestressed concrete structures may be subject to more pronounced creep and relaxation effects. Creep deformation is attenuated exponentially with time. Creep-induced concrete cracks are small and are not sufficiently large to result in concrete deterioration or to expose the reinforcing steel to environmental stressors. In light of the low actual compressive stresses experienced by safety-related structures, except for prestressed concrete containment, creep degradation will not affect the continued safety function performance during the license renewal term. The creep deformation of a prestressed concrete containment is represented in prestressing loss of the tendon which will be addressed later.

Shrinkage

Shrinkage of concrete occurs as the result of water leaving the concrete. Above 80% of the ultimate shrinkage occurs during the first six months, 90% in the first year and about 98% in the first five years. Excessive shrinkage may cause cracking of concrete surfaces. The significance of the cracking as a potential contribution to degradation is in providing aggressive elements access to the reinforcing steel, promoting the possibility of corrosion. In the absence of aggressive chemicals in conjunction with shrinkage cracks, no further degradation will take place. The corrosion of rebar will be addressed later.

Abrasion and Cavitation

Abrasion and cavitation are limited to structures exposed to flowing water because of its ability to transport materials which can abrade the concrete and in removal of concrete materials under vacuum created at the water/air-to-concrete interface. Cavitation damage is not common at velocities less than 40 fps. However, for closed conduits, degradation due to cavitation can occur at velocities as low as 25 fps at abrupt changes in slope or curvature. They could be potentially significant degradation mechanisms for intake structures, cooling towers and/or spray ponds. However, these structural components may be inspected periodically in accordance with NRC Regulatory Guide 1.127, "Inspection of Water-Control Structures Associated with Nuclear Power Plants" (Ref. 11). According to this Regulatory Guide, the inspection of these structures is performed at periodic intervals not to exceed 5 years and includes engineering data compilation and on-site inspection programs. Based on the guidelines, timely identification of any significant degradation on these structures are assumed.

Cracking of Masonry Block Walls

Masonry block walls are considered ITLR components because they can be designed as bearing walls, shear walls, and piping or equipment supporting walls. Masonry block wall cells may or may not contain reinforcing steel and grout to provide the structural strength of the walls. Basically, the degradation mechanisms for the reinforced concrete walls are similar to those of masonry block walls except that the latter tend to crack at the mortar joint between the blocks due to carbonation and drying of the units. Although the cracking of masonry block wall is an ongoing processes throughout plant life, most cracking occurs in the early stages of plant operation.

Nuclear Regulatory Commission IE Bulletin No. 80-11 "Masonry Wall Design" (Ref. 12) required licensees: 1. to identify the masonry walls which are in close proximity to or have attachments from safety-related piping or equipment, and 2. to re-evaluate the design adequacy and construction practices. Upon comparing licensees responses to IE Bulletin No. 80-11 with the observations during NRC's walkdown inspections, NRC issued information Notice No. 87-67, "Lessons Learned from Regional Inspections of Licensee Actions in Response to IE Bulletin No. 80-11" (Ref. 13) which NRC proposed some plant-specific actions to be taken to correct problems. For some plants, NRC proposed a periodic surveillance to ascertain that the level of structural adequacy to which the licensee committed is maintained. For some plants, NRC proposed that the licensee perform analyses for the reason of cracking, structural adequacy, or document the repair effort, and provide countermeasures to prevent recurring of similar cracks.

On this basis, the degradation of the masonry block walls which took place in the early stages of plant operations were identified and mitigative programs were established.

4.0 REINFORCING STEEL/EMBEDDED STEEL DEGRADATION MECHANISMS

Corrosion

Concrete's high alkalinity (pH > 12.5) provides an environment that protects rebar and embedded steel from corrosion. If the pH is lowered (e.g. pH 10 or less) corrosion may occur. However, the corrosion rate is still insignificant until it reaches a pH of 4.0. When moisture and a supply of oxygen are present, the presence of water-soluble chloride ions, above threshold levels of 0.2 percent (0.4 percent calcium chloride) by mass of Portland cement can accelerate corrosion under many circumstances (Ref. 1). Reduction in pH could be caused by the leaching of alkaline products through cracks, entry of acidic materials, or carbonation. Chlorides could be present in constituent materials of the original concrete mix, or they may be introduced environmentally. The degree to which concrete will provide satisfactory protection for rebar/embedded steel is, in most instances, dependent upon the quality of the concrete and the depth of concrete cover over the steel. The permeability of concrete is also a major factor affecting corrosion resistance.

Typical safety-related concrete structures in a nuclear power plant are generally designed and constructed following ACI 318 Code with its relevant ACI Standards and ASTM Specifications which assure concrete of low permeability and rebar/embedded steel with adequate concrete cover to prohibit exposure to the corrosive environment. The primary area where corrosion could occur is on the below grade exterior surface of a concrete structure, especially in the zone of fluctuating groundwater levels. A phased testing program and ARDM management program similar to those of concrete ARDM due to aggressive chemicals are recommended to assess the condition of the rebar/embedded steel in this portion of concrete structures.

Irradiation

The effect of irradiation on steel material is to increase the yield strength, decrease the ultimate tensile ductility and increase the ductile to brittle transition temperature (Ref. 5). The threshold of cumulative neutron irradiation on the change of steel properties is 10^{18} neutrons/cm^2. But only after exposure of 1×10^{19} neutrons/cm^2, are appreciable changes noticed. The only concrete structure that may be exposed to enough radiation to potentially affect the mechanical properties of the rebar/embedded steel is the primary shield wall. An evaluation has been performed based on the fluence level after 80 years of operation of a typical 1000 MWe LWR (Ref. 10) which is exposed to approximately 1×10^{18} to 1×10^{19} neutrons/cm^2 at the primary shield wall. It was concluded that no significant degradation is expected in the shield wall's ability to perform its safety function. This conclusion is based on the following 3 reasons:

1. The typical fluence level for reinforcing steel in the primary shield wall is below the level expected to cause degradation.

2. Increase in yield strength is a positive factor for shield wall integrity.

3. The ductility reduction of steel is insignificant at the calculated fluence level. The irradiated rebar have ample safety margin to satisfy the ductility ratio requirement imposed by even the extreme loading condition.

5.0 STRUCTURAL STEEL DEGRADATION MECHANISMS

Corrosion

Steel corrodes in the presence of moisture and oxygen and the rate of corrosion increases by presence of chlorides. If steel is in contact with other metal that is more noble in the galvanic series, corrosion ratio is also increased. Corrosion products consisting of hydrated oxides of iron form an exposed, unprotected surface on the steel. The affected surface may be wasted away such that visible perforation may occur. Corrosion of steel may cause the protective coatings to lose their ability to adhere to the corroding surface. Damage to the coatings can be detected visually well in advance of significant degradation of the steel. In most of the nuclear power plants, housekeeping and maintenance programs include routine visual inspections on coating degradation and general corrosion of accessible structural steel surface and preventive measures are taken so that significant degradation is precluded. The susceptible locations for steel corrosion are those at inaccessible areas such as column base plates and anchor bolts, metal decking and steel brackets in areas of high humidity or areas potentially affected by leakage. A phased inspection program including visual inspection, radiographic testing, magnetic particle testing and liquid penetration testing can identify the extent of corrosion. If the inspection and testing does not provide conclusive results that properly bound the identified degradation then destructive testing, including cutting samples for chemical analysis, metallurgical evaluation, etc., may be required.

Irradiation

The effects of irradiation on structural steel are the same as those for reinforcing steel. Most structural steel is exposed to lower fluence than reinforcing steel except steel plates used in some BWR plants on the inner and outer surfaces of the radiation shield wall directly surrounding the reactor vessel. Evaluation similar to the irradiation effects on rebar has been performed for the liner plate of a typical 950 MWe BWR plant with a lined concrete primary shield wall (Ref. 10). It is concluded that no significant degradation is expected in the steel's ability to perform its designed safety function after 80 years of plant operation.

6.0 PRESTRESSING SYSTEM DEGRADATION MECHANISMS

Corrosion

Most corrosion-related failures of prestressing tendons are attributed to pitting, stress corrosion, hydrogen embrittlement, or some combination of these, (Ref. 14). Corrosion of tendon wires causes cracking or a reduction in wire cross sectional area, which may render reduction of the prestressing forces applied to the concrete. Tendon ducts are usually filled with petroleum-based grease product in order to minimize the possibility of tendon corrosion. However, the potential of corrosion on tendons and anchor heads does exist. Regulatory Guide 1.35 and ASME Section XI, Subsection IWL contain provisions for managing the effects of such degradation. Both programs require visual examination and testing of the tendon samples to detect evidence of corrosion or other damage and provide acceptance criteria. Samples of the corrosion protective medium are taken for testing of alkalinity, water content, aggressive ions, and pH. The visual inspection and testings are performed periodically, then potentially significant degradation caused by corrosion of prestressing tendons and anchor heads is managed effectively.

Prestressing Loss

Pretensioned load on tendon tends to reduce overtime. It can be caused by:

- Stress relaxation of prestressing wire
- Shrinkage, creep of concrete
- Anchorage seating losses
- Tendon friction
- Reduction in wire cross section due to corrosion

These losses are anticipated and calculated in the design process. If the losses were to exceed those considered in the design, the design margin could be reduced. Prestressing losses are presently monitored as part of the inservice inspection programs, by periodic liftoff tests as described in Regulatory Guide 1.35 and the ASME Code Section XI, Subsection IWL. If losses are greater than expected, the tendons are retensioned or replaced. These inspection and surveillance programs will be continued throughout the license renewal period.

7.0 FREE-STANDING STEEL CONTAINMENT AND STEEL LINER DEGRADATION MECHANISMS

Strain Aging

There are two types of strain aging: static strain aging, which occurs after the material has been deformed; and dynamic aging, which occurs during plastic straining. Strain aging results in higher yield strength, higher ultimate tensile strength, lower notch toughness, and reduced ductility. Dynamic strain aging is not expected in the carbon steel components of free-standing steel containment or liner plate during their service life, since the strains associated with the design service loads are below the elastic limit of the material. Static strain aging is possible in the carbon steel plates of steel containment and liner plate, which are cold formed during construction with free nitrogen present. At ambient temperatures, static strain aging can result in substantial property changes within two to three years after the material is worked and it can be accelerated with an increase in temperature. Generally, the steel containment and liner plate are made from low carbon steel plate which are cold formed. But the plates are normalized, or stress relieved or both after forming with minimal (<5%) subsequent cold working. Therefore, static strain aging will not affect the continued safety function performance of the steel containment and liner plate.

Corrosion

The types of corrosion applicable to steel containments and liner plates are general corrosion, galvanic corrosion, and stress corrosion cracking (SCC). General corrosion can take place when steel is exposed to oxygen and moisture. Galvanic corrosion occurs when the electrical potential difference between dissimilar metal, placed in contact with each other, results in the flow of electrons between them. SCC results from the combined action of corrosion and tensile stresses. The stresses may be either applied or residual and must be at or near the materials' yield point.

The corrosion of inaccessible or below grade regions of steel liners or free-standing containment of a PWR plant are identified as a plant-specific ARDM which can not be shown to be adequately controlled for the extended license term by established procedures (Ref. 15). Plant-specific evaluations are necessary for the following components:

Reinforced and Prestressed Concrete Containments
1. Containment liner below grade exterior surface
2. Basemat liner exterior surface
3. Liner anchors below grade

Free-Standing Steel Containment with Flat Bottom and an Ice-Condenser
1. Embedded shell region
2. Basemat liner
3. Liner anchors

For a BWR containment, the following components are identified as having a corrosion potential for which effective programs do not currently manage the degradation (Ref. 16):

Mark II and Mark III Concrete Containments
1. Containment liner below grade exterior surface (Mark III only)
2. Basemat liner exterior surface
3. Liner anchors below grade

Mark I, II and III Free-Standing Steel Containments
1. Embedded shell region
2. Sand pocket region (exclude Mark III)
3. Support skirt (exclude Mark III)

For these inaccessible areas, a phased inspection and management program as described in Section 4.0 for management of rebar corrosion is recommended. For other areas of steel containments and concrete containment liner plates, corrosion can be controlled through the use of coatings. Inspection procedures of Section XI, Subsection IWE of ASME Code and Appendix J to 10CFR50 integrated leak rate test can provide detection and management of general or localized corrosion degradation in accessible areas.

8.0 MISCELLANEOUS MECHANISMS

Settlement

The most pronounced settlement of a structure occurs in the first several months after construction which is readily evident early in the life of a structure. Following completion of construction, settlement rates decline, and long-term settlement is generally small. Possible exceptions would include sites with soft soil and/or significant changes in underground water conditions. This construction-related degradation is typically monitored throughout the construction program and compared with the settlement design allowance. Various safety-related structures may be surveyed at different intervals. As the plant goes from construction into the operating phase, the settlement is predicted to be significant, which may warrant continuing settlement monitoring during operation.

While continuing settlement monitoring after plant start-up is usually unnecessary, for plants where long-term settlement is being monitored on an ongoing basis, early identification of significant differential settlement allows appropriate measures to be taken to ensure that the structural integrity and functionality of the plant is maintained. In order to manage the effects of structural settlement, the license renewal applicant should review and evaluate the plant specific features of the site soil conditions.

9.0 SUMMARY

This paper presents a list of generic plausible degradation mechanisms for which the safety-related structures of a typical nuclear power plant may be exposed. Structural component-degradation mechanisms are evaluated on generic basis. Most of the structural component-degradation mechanism combinations can be categorized as insignificant on the basis of design specification and criteria, and construction measures taken by the nuclear industry. Significant aging degradation is a potential for some structural component-degradation mechanisms combinations,. For these cases, phased surveillance, inspection and/or testing programs are recommended to identify whether and where the degradation may exist. Once confirmed, additional inspections and testing are needed to predict the trending of the degradation. In some cases, degradation management programs may be needed, in order to assure that the structures can maintain their designed functions during the license renewal period.

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