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DURABILITY AND STRUCTURAL INTEGRITY OF NUCLEAR POWER PLANT CONCRETE: A CRITICAL REVIEW OF MATERIALS, AGEING, AND ASSESSMENT

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Abstract

Multiple components in nuclear power plants are made of concrete, which provide structural support, confinement, and radiation shielding over multi-decadal service. This critical review synthesizes evidence on material selection, irradiation and internal swelling, thermal and chemical degradation, mechanical-property evolution, non-destructive assessment, structural health monitoring, and long-term operation frameworks, with particular attention to evidence transferability and coupled ageing. Stiffness and tensile-related properties generally respond to distributed cracking earlier than compressive strength. This pattern is evident across historical irradiation datasets, experimental degradation studies, operating experience, and structural analyses. However, structural significance depends strongly on aggregate mineralogy, temperature and moisture history, restraint, prestress, exposure gradients, and the credited safety function. The available evidence does not support a single universal irradiation or degradation threshold that can be applied without regard to aggregate mineralogy, exposure conditions and structural restraint. This review introduces a coupled-ageing convergence framework that maps diverse mechanisms (irradiation, alkali-silica reaction (ASR), delayed ettringite formation (DEF), thermal, chemical, corrosion) onto six structural state variables (dimensional change, cracking, stiffness/tensile degradation, transport, prestress evolution, embedded-steel deterioration) and demonstrates, through integrated evidence synthesis, that compressive strength alone is an incomplete condition indicator for long-term operation (LTO) assessment. A defensible long-term operation strategy therefore combines traceable material and exposure records, targeted multi-method inspection, selective destructive verification, uncertainty-aware structural analysis, and function-specific acceptance criteria.

Keywords: concrete biological shield; containment; confinement; radiation-induced volumetric expansion; alkali-silica reaction; prestress loss; non-destructive testing; long-term operation.

1. Introduction

Concrete is a principal nuclear construction and safety material because it combines load-bearing capacity, fire resistance, radiation attenuation, durability, and constructability. In light-water reactor (LWR) facilities, it is used in basemats, containments, reactor buildings, concrete biological shields (CBSs), equipment supports, pools, cooling-water and waste-management structures [1–4]. Many of these components are massive, highly reinforced, difficult to replace, and must therefore retain their credited functions for decades. Material selection is therefore inseparable from structural function. Ordinary concrete remains common in existing CBSs and containments, while heavyweight or high-performance formulations are best viewed as targeted solutions rather than universal upgrades. The relevant design question is whether a material can simultaneously satisfy structural, shielding,

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thermal, durability and inspection requirements over the intended operating period [1, 4, 5]. Nuclear concrete is consequently a multifunctional safety material rather than only a structural composite. The CBS illustrates this coupling: representative thicknesses are about 1.5–2.2 m in pressurized water reactors (PWRs) and 0.6–1.2 m in boiling water reactors (BWRs), and the concrete must attenuate the same neutron–gamma field which can progressively alter its constituents [5]. Advanced heavyweight concrete (HWC), self-compacting concrete (SCC), and ultra-high-performance concrete (UHPC) may improve shielding, placement, or durability, but their nuclear use still requires qualification of long-term dimensional, thermal and irradiation performance [4].

Three functions dominate safety significance: confinement, shielding, and structural support [1–3]. Concrete containments must resist normal, seismic and accident actions while controlling cracking, deformation, residual prestress, liner strain, joints, penetrations and leakage. Shielding depends on thickness, density and elemental composition for gamma attenuation and on hydrogen-bearing phases and absorbers for neutron moderation and capture [6, 7]. These functions can also conflict. Higher density may improve gamma attenuation but complicate placement. Retained water assists neutron moderation but also affects shrinkage, transport, and internal reactions. Prestress improves crack control while introducing long-term sensitivity to creep, shrinkage, and steel relaxation. Nuclear concrete design must therefore preserve functional compatibility rather than optimize any single property in isolation [1, 3, 7]. Fast-neutron exposure can disorder susceptible aggregate minerals, causing radiation-induced volumetric expansion (RIVE), microcracking, and reductions in stiffness and strength [5, 6, 8, 9]. The available irradiation databases identify an order-of-magnitude fluence region in which such changes become increasingly evident; its interpretation and limitations are discussed in Section 4.1.a. Structural support functions further introduce sustained, seismic, thermal, and accident loads and time-dependent prestress losses. Ageing should therefore be judged by whether its magnitude, location, and progression challenge a credited safety function, not simply by the presence of visible damage.

Despite extensive individual-mechanism reviews (e.g., NUREG/CR-6927 [2], IAEA NP-T-3.5 [1]), prior syntheses have not systematically addressed how mechanisms with diverse origins converge structurally through a limited set of state variables, nor have they integrated evidence transferability, coupled ageing, and function-specific acceptance into a unified long-term operation (LTO) framework. This review addresses that gap by synthesizing material, exposure, degradation, property-evolution, and assessment evidence into a decision-oriented structure. Many earlier nuclear power plants (NPPs) were developed around 30–40-year operating periods, whereas operation for 60 years or longer is increasingly relevant [1, 10]. A further difficulty is that several critical regions are inaccessible during operation. Ageing management must consequently combine indirect evidence with uncertainty-aware structural models and traceable construction/operating records. This need for data continuity underlies modern ageing-management programme (AMP) and plant-information approaches and explains why missing baseline material or exposure information can become a major LTO uncertainty decades after construction [10, 11]. Key mechanisms, including RIVE, alkali–silica reaction (ASR) [12], delayed ettringite formation (DEF) [13], thermal and chemical degradation, corrosion, creep, shrinkage, and prestress relaxation, interact through temperature, moisture, cracking, and restraint [10, 14]. High-fluence laboratory tests also differ from commercial reactors in flux, temperature and moisture, complicating extrapolation [6, 7, 9]. Reliable LTO therefore depends on plant records, exposure reconstruction, inspection and modelling [10, 11]. This review follows an integrated sequence in which material characteristics and exposure conditions are linked to degradation mechanisms, subsequent property evolution, structural consequences, assessment, and ultimately

decisions on safety functions and long-term operation.

This study adopts a structured critical-review approach rather than a formal meta-analysis. The literature search was conducted primarily between 2020 and early 2026, covering publications from approximately 1990 to 2025, with retention of key foundational reports published earlier (e.g., selected NUREG documents). Primary sources included Scopus, Web of Science, ScienceDirect and Google Scholar, supplemented by the official portals of the International Atomic Energy Agency (IAEA), the U.S. Nuclear Regulatory Commission (USNRC), ASME, the American Concrete Institute (ACI), the Canadian Standards Association (CSA), the Japan Electric Association (JEA), the Korea Electric Association (KEA) and relevant national standards bodies. Search strings combined terms such as "nuclear concrete", "biological shield", "radiation-induced volumetric expansion", "alkali-silica reaction nuclear", "containment prestress", "NPP ageing management" and related keywords; backward and forward citation chasing from major reviews and the ACES project deliverables [10, 14] was also employed. Approximately 150 documents were screened; approximately 100 were retained for detailed synthesis after application of relevance and transferability criteria, in which 61 references were cited in this review.

This review synthesizes findings across ACES work packages and integrates them with the broader international literature to provide a unified, decision-oriented framework for LTO assessment, thereby extending the ACES corpus for readers who require cross-cutting guidance rather than mechanism-specific technical detail. Sources were retained when they provided directly relevant information on NPP concrete materials, exposure conditions, degradation mechanisms, mechanical-property evolution, inspection or monitoring, structural assessment, or long-term operation. Duplicate sources, studies without transferable relevance to nuclear concrete were excluded from quantitative synthesis. This review was designed as a critical evidence synthesis and did not claim exhaustive retrieval of all publications in the field.

Evidence was interpreted according to its directness to NPP conditions, completeness of material and exposure characterization, specimen or structural scale, and consistency with independent studies. For example, accelerated high-flux irradiation studies [6, 9] were judged to have high mechanistic relevance but limited quantitative transferability to LWR service because of differences in flux, temperature, moisture redistribution, and available creep relaxation (Section 4.1.a). In contrast, the Seabrook ASR case [15] and Forsmark prestress measurements [16] were judged to have high transferability because they report in-service conditions, material provenance, and multi-method validation. Because irradiation, ASR/DEF, thermal, and chemical degradation datasets differ substantially in aggregate mineralogy, exposure definition, temperature, moisture, geometry, loading, and test duration, formal statistical pooling was not undertaken. Numerical values are therefore identified as single study observations, observed literature ranges, or order of magnitude screening ranges rather than confidence intervals or design reduction factors. The quantitative screening envelopes assembled in Table 3 were derived by assigning each numerical entry to one of three provenance classes - (i) historical dataset / order-of-magnitude screening range, (ii) multi-study observed envelope, or (iii) single-study observation -after exclusion of tests lacking adequate mineralogical or exposure characterisation, inconsistent energy cut-offs, uncontrolled moisture/temperature, or free-expansion conditions not representative of restrained NPP structures (see Section 5 and the note to Table 3 for the full selection rules). They are presented solely as indicative triggers for plant-specific assessment and are explicitly distinguished from design reduction factors, code-allowable residuals, or universal safety thresholds. Contradictory findings were evaluated by identifying likely moderating variables and experimental confounders instead of averaging physically dissimilar datasets.

Relative to existing reviews and major technical reports on nuclear concrete, this work makes three specific contributions. First, it integrates dispersed data on irradiation, internal swelling, thermal and chemical degradation, and corrosion into a unified, mechanism based set of structural state variables, showing how dimensional change, cracking, stiffness and tensile degradation, transport, prestress evolution, and deterioration of embedded steel together govern functional performance. Second, it develops quantitative yet explicitly uncertainty aware screening envelopes for key mechanical property changes (e.g., for RIVE and ASR) and embeds them within a through life assessment logic, emphasizing how they should and should not be used in plant specific structural evaluations. Third, it links this mechanistic and quantitative synthesis to practical assessment and regulatory decision making, by proposing an evidence based ageing assessment workflow, clarifying the role of inspection, NDT, SHM and structural analysis, and situating these elements within current international codes, standards, and LTO guidance. ACES deliverables form an important part of the evidence base and conceptual foundation; the integrated state-variable framework, the screening envelopes with transparent provenance, and the end-to-end LTO workflow constitute the novel contributions of the present review. In this way, the review is intended not only to summarize the state of knowledge, but to provide a structured framework that end users can apply directly in developing and defending NPP concrete ageing and LTO cases.

The remainder of this review proceeds from materials to mechanisms to structural consequences and assessment. Sections 2 and 3 summarize nuclear specific material and mixture design requirements. Section 4 reviews key ageing and degradation mechanisms and their interactions, followed in Section 5 by the resulting mechanical property changes relevant to structural assessment. Section 6 examines inspection, monitoring and structural health methodologies, Section 7 situates these findings within international codes and regulatory frameworks, and Section 8 synthesizes recommendations for lifetime extension and new build projects. Section 9 then draws the main cross cutting conclusions and explicitly restates the novel contributions of this review for NPP concrete LTO.

2. Overview of concrete requirements for NPPs

NPP concrete requirements arise from the intersection of safety classification, credited function and exposure. Qualification must therefore address capacity, serviceability, durability, shielding, accident loads, QA/QC, inspectability and ageing rather than compressive strength alone [1, 10].

2.1. Classification of structures: Category I (safety-significant) vs. non-safety structures

In historical U.S. practice, Seismic Category I identifies structures required to perform during and after the Safe Shutdown Earthquake, but this term should not be treated as equivalent to the broader international concepts safety-related or important to safety [2, 3, 11]. Safety significance is function-based and can extend to nominally non-safety structures whose failure could disable a credited structure, system, or component (SSC).

Typical safety-significant concrete SSCs include containments, basemats, reactor supports, CBS walls, spent-fuel pools and structures that protect safety systems. Their importance is graded rather than binary: flooding, falling loads, fire or seismic interaction can make an otherwise conventional concrete element relevant to nuclear safety [2, 3]. This consequence-informed view is more useful for inspection prioritization than relying on labels alone.

The code architecture reflects this grading: ACI 318 provides the general concrete basis; ACI 349 covers nuclear safety-related concrete other than containments; and concrete containments are treated through ACI 359/ASME Section III, Division 2 [17–19]. For existing plants, the applicable edition is licensing-basis-specific. Classification should therefore remain traceably linked to function, exposure, inspection and acceptance criteria throughout life [11, 20].

2.2. Functional requirements for containment buildings and biological shields

Containments and CBSs illustrate different but coupled performance requirements. Containment assessment must consider strength, deformation, crack control, prestress, liner/penetration integrity and leakage; deterioration at a liner interface can be functionally significant before global concrete capacity is exhausted [2, 3].

Prestressed containments add a time-dependent requirement because residual compressive prestress controls crack opening and deformation. Creep, shrinkage, steel relaxation, tendon condition and anchorage behavior must therefore be evaluated together with liner integrity. In CBSs, by contrast, attenuation and irradiation stability become dominant, illustrating why different safety-related concrete structures require different material-performance indicators [1, 5, 16].

CBS performance combines attenuation and, in some designs, structural support. Gamma shielding is controlled mainly by thickness, density and elemental composition, whereas neutron attenuation depends strongly on hydrogen-bearing phases and absorbers [7]. Exposure is highly non-uniform; bounding surface fluence for representative PWR cases can reach the order of 6×10^{19} n/cm^2 ($E > 0.1$ MeV) over extended operation while most of the wall receives less [5, 6]. Thus irradiation must be evaluated spatially, and exceedance of a screening fluence should trigger plant-specific material and structural assessment rather than automatic rejection [5, 6, 8]. Because neutron attenuation profiles depend on reactor geometry, source term, concrete composition, shield thickness, and the adopted transport model, a generic PWR or BWR depth profile is not proposed here; plant-specific neutron-transport calculations should be used for quantitative LTO assessment.

2.3. Environmental loading conditions: Radiation fields, moisture, and thermal gradients

Nuclear exposure is inherently coupled [21]. Fast neutrons cause displacement damage in aggregate minerals, gamma radiation contributes ionization, heating and water radiolysis, moisture affects neutron moderation as well as ASR/DEF, corrosion and shrinkage, and temperature modifies transport, creep and reaction kinetics [6, 7, 9, 10]. Historical CBS design contexts often used about 65 °C for general concrete and about 93 °C locally, but these are contextual design values rather than universal degradation limits [5].

Moisture is especially important because its role is bidirectional: hydrogen-bearing water contributes to neutron moderation, but persistent moisture also facilitates ionic transport, ASR/DEF and corrosion, whereas drying contributes shrinkage and can reduce neutron shielding. The engineering state variable is therefore the evolving internal moisture field, which can remain highly non-uniform across metre-thick walls for decades [7, 10, 14].

Massive walls can retain high internal moisture while exposed surfaces dry, producing long-lived hygral and thermal gradients. Consequently, highly exposed structures are best represented by spatially resolved radiation–hygro–thermal–mechanical analyses linked to actual operating history and monitoring [6, 7, 14]. The resulting requirement is performance-based: the material–structure system must preserve confinement, shielding and support under evolving exposure.

3. Concrete design and material selection for NPPs

Nuclear mixture design is a multi-objective problem balancing strength, heat generation, dimensional stability, transport resistance, constructability and radiation attenuation. Table 1 condenses the principal material strategies and trade-offs.

3.1. Fundamental constituents: Cement, water, and aggregate

Historical U.S. NPP concretes commonly used Type II Portland cement and local aggregates, with reported specified compressive strengths range from 13 to 55 MPa and approximately 35 MPa being typical of many safety-related concretes [2, 3]. High strength was not the sole objective because massive sections are often controlled by shielding, stiffness, crack control and thermal effects.

Conventional constituent limits remain necessary for quality control, but they should not substitute for performance evidence. Increasing binder content merely to raise 28-day strength can increase hydration heat, autogenous shrinkage and cracking risk in massive sections. Conversely, lower-permeability mixtures may be beneficial only if they can be reliably mixed, pumped, consolidated and cured around dense nuclear reinforcement [1, 2].

Aggregate occupies about 60–75% of concrete volume and controls stiffness, thermal expansion, ASR/ACR (alkali-carbonate reaction) susceptibility and much of RIVE response [1, 2, 5, 6, 9, 22]. Highly irradiated zones therefore require petrographic/mineralogical qualification in addition to conventional grading and strength tests. Water content must likewise balance transport resistance and constructability while recognizing the contribution of retained hydrogen to neutron moderation [7].

3.2. Heavyweight concrete for radiation shielding

HWC uses dense aggregates such as magnetite, hematite, barite, ilmenite and limonite; reported shielding-concrete densities range from 2900 to 6000 kg/m^3 compared with 2200–2450 kg/m^3 for ordinary concrete [23–25]. Higher density generally improves gamma attenuation, but neutron shielding also depends on hydrogen and absorbers, so density alone is insufficient [25].

Magnetite and hematite provide high iron content and generally strong gamma attenuation, whereas barite offers high photon attenuation through barium. Comparative studies show that the preferred aggregate depends on the incident spectrum and required thickness, and that combinations of dense constituents with hydrogen- or boron-bearing phases may be more effective for mixed neutron–gamma fields [23–25]. This spectrum-specific behavior limits simple density-based ranking.

Heavy aggregates can increase segregation, pumping difficulty and thermal incompatibility, and irradiation stability remains source/mineral-specific. HWC should therefore be qualified simultaneously for attenuation, mechanical compatibility, rheology, thermal response and long-term mineral stability rather than by density alone.

3.3. Advanced formulations: UHPC and SCC

UHPC combines a very low water-to-binder ratio (w/b), optimized packing and often fibres, producing very high strength and low permeability. RS-UHPC studies report compressive strengths around 150–180 MPa, but shielding is governed mainly by heavyweight constituents rather than strength itself [4]. Accelerated durability evidence is promising, yet long-term irradiation, autogenous shrinkage, heat and fire response remain insufficiently established [4, 26].

RS-UHPC formulations also demonstrate the trade-off between performance and qualification. Reported cement contents can reach several hundred kilograms per cubic metre, with silica fume and fibres used to densify the matrix and control cracking [4]. Such compositions may reduce transport but can increase heat evolution, cost and autogenous shrinkage. Their use in massive safety-related components therefore requires stronger long-term evidence than is currently available [4, 26].

SCC addresses a different nuclear need: reliable placement around dense reinforcement, liners, anchors and penetrations. Its benefit is reduced consolidation defects rather than inherent resistance to irradiation or ASR. Nuclear SCC, especially with heavyweight aggregates, should be validated at representative scale.

3.4. Supplementary cementitious materials: Fly ash, GGBS, and silica fume

SCMs modify heat evolution, pore connectivity, chemical resistance and ASR susceptibility. Fly ash can reduce heat and improve workability [27]; GGBS reduces heat generation during hydration, chloride transport and permeability; silica fume densifies the matrix and is central to UHPC [1, 4, 24, 28]. Properly selected SCMs can mitigate ASR but do not eliminate ACR or aggregate-controlled RIVE.

SCM benefits are exposure-specific. Fly ash and GGBS are attractive for mass-concrete heat control, GGBS for chloride resistance, and silica fume for very low permeability and UHPC; however, high replacement levels can alter carbonation resistance, early strength and curing sensitivity [28]. Supply continuity and source chemistry also matter in nuclear projects with long construction periods, so constituent provenance should form part of lifetime material documentation.

The main trade-offs are slower early strength, carbonation sensitivity at high replacement or poor curing, autogenous shrinkage in very low-w/b systems, and source variability. Nuclear binder qualification should therefore be performance-based and documented by constituent source, heat release, transport, dimensional stability, chemical resistance and, where relevant, shielding and irradiation response.

Overall, NPPs require function-specific concrete families: conventional moderate-heat mixtures remain a robust baseline, HWC serves shielding, SCC improves placement, UHPC is a selective high-performance option, and SCMs tailor thermal and transport behaviour. Aggregate mineralogy remains the common control variable for ASR and RIVE.

Table 1. Materials and mixture-design options for NPP concrete

Material/ strategy	Typical characteristics	Mechanical performance	Thermal	Durability	ASR/RIVE implications	Radiation shielding	Nuclear advantages	Main limitations
Conventional moderate-heat concrete	Normal density; historical compressive strength (f_c) $\approx$ 13–55 MPa [2]	Established strength/ stiffness/ creep	Moderate heat when properly specified	Proven if well placed/cured	Controlled mainly by aggregate mineralogy	Adequate bulk shielding in many SSCs	Mature, economical, robust	Source/curing/ cracking sensitive
Heavyweight concrete (magnetite/ hematite/ barite)	$\rho \approx$ 2900–6000 kg/ m^3 [23–25]	Mixture/ source dependent	Thermal compatibility must be checked	Segregation control essential	No generic RIVE immunity	High gamma attenuation; neutron response composition-dependent	Reduced shielding thickness	Workability, pumping, source-specific irradiation data
H-/ B-bearing shielding additions	Hydrogen-bearing phases and/or B absorbers	Depends on dosage/ dispersion	Bound-water retention may be T-sensitive	Long-term chemical stability required	Adjunct to aggregate qualification	Improved neutron moderation/ capture [7, 25]	Tailored mixed-field shielding	May affect setting, strength and rheology
SCC	High flow and passing ability	Strength set by binder system	As corresponding mix	Reduces consolidation defects	No intrinsic ASR/RIVE benefit	Composition controls shielding	Congested reinforcement /liners /anchors	Segregation/ blocking; full-scale validation
UHPC / RS-UHPC	Very low w/b; fibres; $f_c \sim$ 150–180 MPa [4]	Very high strength/ toughness	High binder heat; fire response needs qualification	Very low permeability [26]	Irradiation/ RIVE evidence limited	Heavy constituents, not strength, govern shielding	Thin/high-demand / localized components	Cost, autogenous shrinkage, limited nuclear codification

Material/ strategy	Typical characteristics	Mechanical performance	Thermal	Durability	ASR/RIVE implications	Radiation shielding	Nuclear advantages	Main limitations
SCM-rich binders (FA/ GGBS/SF)	Heat/transport/ reactivity tailored by SCM type	Later-age performance can improve	FA/GGBS reduce heat; SF less so	Lower permeability/ chloride ingress [28]	Can mitigate ASR; not ACR/RIVE	Mostly indirect via density/ moisture	Mass concrete, aggressive exposure, UHPC	Source variability, carbonation/ shrinkage trade-offs

Note. HWC = heavyweight concrete; RS-UHPC = radiation-shielding ultra-high-performance concrete; SCC = self-compacting concrete; SCM = supplementary cementitious material; ASR = alkali-silica reaction; RIVE = radiation-induced volumetric expansion. Values are literature ranges, not design limits.

4. Ageing and degradation mechanisms in NPP Concrete

Ageing results from interacting irradiation, thermal, hygral, chemical and mechanical processes. Although triggers differ, most mechanisms converge toward expansion/shrinkage, microcracking, increased transport, stiffness/tensile loss, corrosion and stress redistribution. Table 2 summarizes the main evidence.

4.1. Irradiation-induced degradation

a. Neutron fluence and radiation-induced damage

Fast-neutron damage is spatially localized and strongly dependent on spectrum, mineralogy, flux, temperature, and moisture. The historical database assembled by Field et al. [9] includes 307 compressive-strength, 62 tensile-strength, 138 modulus, and 114 expansion measurements. Across these heterogeneous data, changes in expansion and mechanical properties become increasingly evident at fast-neutron fluences of the order of 10^{19} n/cm^2 for $E > 0.1$ MeV [6–9]. This value is used here solely as an order-of-magnitude screening region for further assessment; it is neither a detection limit nor a serviceability or safety threshold. The literature does not support a unique property change that consistently defines its onset because response depends strongly on aggregate mineralogy, neutron spectrum and flux, temperature, moisture, and restraint. Energy cutoff must be reported consistently, and high test-reactor fluxes further complicate direct extrapolation to multi-decadal LWR service. Consequently, 10^{19} n/cm^2 should be treated as a trigger for plant-specific material and structural assessment rather than as a universal acceptability limit.

b. Radiation-induced volumetric expansion (RIVE)

RIVE links neutron exposure to cracking: displacement damage disorders susceptible minerals, aggregate expansion loads the surrounding paste, and restraint converts free swelling into multiaxial stress [5, 6, 22]. Mineral-scale expansion varies widely: quartz can reach approximately 17.8% in the compiled irradiation data, whereas changes in calcite are substantially smaller- but bulk concrete expansion is constrained by mixed mineralogy, paste, cracking and creep [22]. Concrete-scale expansion of order 1% can be structurally significant, and analyses show that RIVE and modulus may control CBS response more strongly than residual compressive strength [5, 29]. Plant assessment should therefore couple neutron transport, mineralogy, expansion/property evolution and structural analysis.

Structural restraint makes RIVE fundamentally different from a free-expansion material test. Reinforcement, geometry and pre-existing loads create compressive and tensile zones, while creep progressively relaxes part of the swelling-induced stress [5, 29]. Consequently, the same aggregate expansion can produce different crack patterns and safety margins in different structures. This is one reason harvested material and full structural analysis are more informative than mineral expansion data alone.

c. Gamma irradiation and water radiolysis

Gamma irradiation acts mainly through ionization, heating and water radiolysis [6, 7]. A commonly cited NPP-concrete experiment reported roughly 10% compressive and 5% tensile/flexural strength loss at ~ 5–6 × 10^5 Gy, but gamma-only evidence is limited and often confounded by neutron exposure, heating or drying [2, 7]. Accordingly, gamma dose should be interpreted together with dose rate, temperature and moisture rather than as a universal damage threshold.

Gamma effects are particularly difficult to isolate because dose, dose rate, temperature and moisture evolve together. Radiolysis products may alter the aqueous phase, while energy deposition accelerates drying and thermal gradients. Future qualification should therefore use matched thermal/moisture controls and report specimen sealing and simultaneous neutron exposure; otherwise apparent gamma-induced property changes may partly reflect non-radiation processes [2, 7].

4.2. Internal swelling reactions

a. Alkali–silica reaction (ASR)

ASR requires reactive silica, alkalis and sufficient moisture; slow-reacting aggregates may remain latent for 15–25 years or more [2, 15]. At advanced expansions of ~ 0.5–1.5%, historical material-level compilations report compressive, tensile and modulus losses of roughly 40–60%, 65–80% and 60–80%, respectively [2]. These are broad envelopes, not design factors: reinforcement restraint can reduce free expansion and preserve capacity while serviceability, joint movement and stiffness are still affected. Assessment should combine crack/deformation trending, petrography, cores and structural analysis [15].

The interaction between damage and restraint is particularly important in reinforced or prestressed NPP structures. Chemical prestressing can preserve or even increase some global stiffness measures despite a marked reduction in concrete modulus, while local deformation may still threaten joints, liners or equipment alignment [15, 30]. ASR evaluation should therefore distinguish material deterioration from structural performance and avoid direct conversion of free-expansion tests into member-capacity loss.

b. Delayed ettringite formation (DEF)

DEF is associated with unfavorable early-age thermal history, sulfate/aluminate chemistry and later moisture. Susceptibility is commonly associated with early temperatures around or above 65–70 °C, but no universal threshold exists [14, 31]. Ettringite growth can widen pre-existing cracks and increase permeability; however, the reviewed corpus does not support transferable residual-strength factors. Prevention through early-age thermal control and binder qualification is therefore more reliable than post-damage intervention.

Diagnosis also requires caution because ettringite can precipitate in cracks produced by other mechanisms. The presence of ettringite is not by itself proof that DEF initiated the damage. Early-age temperature records, binder chemistry, moisture history and petrographic evidence should be interpreted together, particularly where ASR and DEF may coexist [14, 31].

4.3. High-temperature effects

Sustained operating temperatures primarily alter moisture, creep and reaction kinetics, whereas accident/fire heating adds steep gradients, vapor pressure, dehydration and possible spalling. Historical nuclear contexts often used ~ 65 °C generally and ~ 93 °C locally [2, 5]. Mechanical changes are often modest or non-monotonic below ~ 200 °C, while more pronounced deterioration commonly develops above ~ 350 °C; peak temperature alone is insufficient because duration, heating rate, moisture, aggregate and cooling control residual response [32].

Thermal behavior also depends on sealing and moisture. Unsealed specimens may lose water and strength differently from the relatively moist interior of massive NPP walls, while rapid accident heating can generate pore pressure and spalling that are absent during slow operational heating [32]. For this reason, normal-operation thermal ageing and accident residual-capacity assessment require different test protocols and constitutive models.

Thermal damage arises from progressive water loss and incompatible paste–aggregate deformation, which concentrates stress at interfaces before severe hydrate decomposition [32]. Because aggregate mineralogy also governs irradiation response, thermal history and mineralogy should be retained as lifetime state variables in both new-build records and LTO analyses.

4.4. Chemical and environmental degradation

a. Calcium leaching and carbonation

Calcium leaching dissolves portlandite and progressively decalcifies C–S–H, increasing porosity and transport; accelerated tests can raise diffusivity by one to two orders of magnitude, so laboratory time cannot be converted directly to service life [10, 14]. Carbonation lowers pore-solution pH and can depassivate reinforcement once the front reaches steel. Both mechanisms are spatial and should be assessed by degradation depth, moisture, cracking and cover rather than bulk strength alone.

Leaching and carbonation also differ in structural consequence. Leaching progressively removes calcium-bearing solids and increases porosity, whereas carbonation may locally refine pores while lowering alkalinity and increasing corrosion susceptibility. Depth profiles of mineralogy, pH and transport properties are therefore more informative than bulk strength measurements for diagnosing these mechanisms [10, 14].

Chemical attack in NPPs is often highly localized. Leakage at penetrations, pool interfaces, joints or liners can create concentration and moisture conditions far more severe than plant-average chemistry. Conversely, massive interior concrete may remain largely unaffected. Inspection and modelling should therefore resolve transport paths and local exposure history instead of assigning a uniform chemical environment to an entire structure [1, 28, 33, 34].

b. Acid and sulfate attack

Acid and sulfate attack dissolve or transform cement hydrates and can cause softening, expansion and cracking [2, 10, 28]. Boric-acid leakage is a nuclear-specific example: under an extreme accelerated condition of 70 °C and 180,000 ppm, concrete lost ~ 27.8% compressive strength after 180 days, whereas lower concentrations in the same study caused much smaller changes [33]. Separate experiments at ~ 2400 ppm confirmed diffusion-controlled calcium leaching [34]. Chemical severity therefore depends on local concentration, temperature, transport and duration, not nominal system chemistry alone.

Sulfate deterioration is likewise binder- and transport-dependent: gypsum and ettringite formation can generate expansion where reaction products are restrained, but low-permeability SCM-rich systems may substantially reduce ingress [26, 28]. Laboratory concentration, solution renewal and exposed surface area must be stated because they strongly affect apparent degradation rate and limit direct service-life extrapolation.

c. Corrosion of reinforcement, prestressing steel, and liners

Carbonation, chlorides, leaching, cracks and persistent moisture can depassivate reinforcement; corrosion then causes section loss, bond degradation, cracking and spalling [1, 2, 28]. Prestressing steel and inaccessible steel liners are especially safety-significant because localized damage can reduce prestress or leak-tightness while global concrete strength remains high. Assessment should

combine environmental /electrochemical indicators, liner or tendon surveillance and structural evaluation.

Embedded steel introduces a further mismatch between local and global indicators. A small through-wall liner pit can be important to confinement even when the concrete shell remains strong, while superficial rust staining does not necessarily imply major section loss. Acceptance criteria should therefore be tied to the function of the affected steel component and to the feasibility of inspection or repair [1, 28].

Across mechanisms, the engineering question is whether the current extent, spatial distribution and progression threaten confinement, shielding or structural support. Coupled effects are important because RIVE/ASR cracking accelerates transport, temperature changes moisture and kinetics, creep relaxes swelling stresses, and corrosion-generated cracks further increase ingress.

4.5. Coupled degradation, evidence quality, and knowledge gaps

Ageing mechanisms in NPP concrete rarely develop independently because radiation, temperature, moisture, chemical exposure, mechanical loading, and time act simultaneously on the material–structure system. As illustrated in Fig. 1, these exposure fields initiate or modify degradation mechanisms that subsequently produce dimensional change, microcracking, transport enhancement, stiffness and strength degradation, and deterioration of embedded steel. Their structural significance is then governed by restraint, prestress, geometry, and load redistribution, ultimately affecting the credited functions of confinement, shielding, and structural support. Inspection and monitoring, data management, uncertainty analysis, and the applicable regulatory framework therefore need to operate across the entire ageing-assessment process rather than at isolated stages.

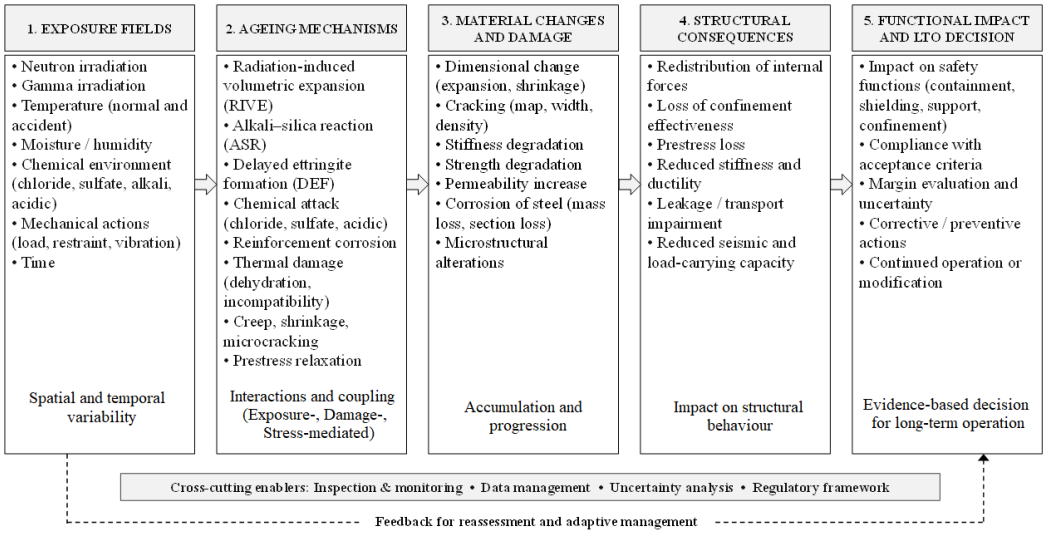


Figure 1. Coupled ageing and structural-consequence framework for concrete in nuclear power plants

Coupled ageing processes can be classified into three principal interaction types. Exposure-mediated coupling occurs when environmental variables such as temperature or moisture modify irradiation response, ASR/DEF kinetics, corrosion, creep, or transport. Damage-mediated coupling arises when degradation caused by one mechanism facilitates another; for example, cracking associated with RIVE, ASR, thermal incompatibility, or corrosion can increase permeability and promote further ingress of water or aggressive species. Stress-mediated coupling occurs when reinforcement

restraint, prestress, sustained loading, creep, or shrinkage modifies stresses generated by internal expansion and cracking [5, 14, 15, 29]. Representative interactions and their current evidence status are summarized in Table 2.

Table 2. Examples of coupled interactions among ageing mechanisms in NPP concrete

Coupling type	Interaction example	Evidence status
Exposure-mediated	Temperature increase accelerates chemical attack	Well documented
	Moisture uptake increases corrosion risk	Well documented
	Irradiation may modify chemical reaction rates	Limited / emerging
Damage-mediated	RIVE or ASR cracking increases permeability, promoting corrosion	Well documented
	Corrosion-induced cracking facilitates ingress of chloride or sulfate	Well documented
	Microcracking enhances diffusion of aggressive species	Well documented
Stress-mediated	Restraint amplifies DEF expansion	Well documented
	Load interacts with creep, shrinkage, and microcracking	Well documented
	Prestress loss combined with creep and irradiation	Limited / emerging

The strength and transferability of the available evidence differ substantially among these interactions. In Table 2, interactions are graded as "Well documented" when multiple independent experimental or operating-experience studies provide consistent qualitative and at least semi-quantitative relationships under conditions relevant to NPP concrete, and as "Limited/Emerging" when the interaction is mechanistically plausible and supported by isolated studies or modelling, but quantitative transferability to commercial LWR conditions remains insufficiently established. Moisture-controlled ASR, DEF, and corrosion, crack-enhanced transport, and the interaction of sustained loading with creep and shrinkage are comparatively well established. By contrast, quantitative relationships for irradiation-modified chemical kinetics, irradiation-assisted ASR, and combined irradiation–creep–prestress effects remain limited or emerging. Similar differences exist among individual degradation mechanisms. The qualitative relationship between fast-neutron exposure, susceptible aggregate mineralogy, RIVE, and cracking is well established, but extrapolation to commercial LWR conditions remains affected by differences in neutron spectrum and flux, temperature, moisture, and mineralogical differences identified in Section 4.1.a [6, 22, 29]. Gamma-only mechanical effects have lower confidence because many historical experiments are confounded by simultaneous neutron exposure, heating, or drying [6, 7]. ASR mechanisms and material-level property degradation are comparatively well documented, whereas translation from free expansion to restrained structural response remains strongly structure-specific [15]. DEF is mechanistically established, but the reviewed evidence does not support a transferable normalized residual-property relationship [14, 31].

These limitations have direct implications for coupled modelling and LTO assessment, and they define which interactions can be represented quantitatively and which require bounding or sensitivity approaches. Experimentally supported interactions should be represented explicitly where appropriate, whereas poorly quantified couplings should be treated through sensitivity analyses, bounding scenarios, or uncertainty propagation rather than through unverified multiplicative degradation factors. They also motivate a focused research agenda: the highest-priority research needs are service-rate irradiation data, controlled gamma-only experiments, quantitative characterization of irradiation–moisture–creep–prestress interactions, improved understanding of irradiation-assisted internal swelling reactions, and validation of coupled models against service-aged or harvested NPP con-

crete [6, 35, 36]. Addressing these gaps is essential for moving from mechanism-specific empirical assessment toward a more defensible through-life prediction of structural and functional performance.

5. Mechanical property alterations under ageing

Section 4 identifies the mechanisms and their interactions; this section concentrates exclusively on the resulting mechanical-property changes and structural implications needed for assessment. Strength, stiffness, creep, shrinkage, and prestress evolve together, while the integrated quantitative evidence and its transferability are summarized in Table 3. The distinction is intentional: Section 4 remains focused on physical/chemical mechanisms and coupling, whereas Section 5 addresses the consequences relevant to structural evaluation and LTO decision-making.

Table 3. Integrated evidence on major ageing mechanisms, mechanical-property changes, and structural implications

Mechanism	Exposure/ condition	f_c/f_{c0}	f_t/f_{t0}	E/E_0	Expansion / physical indicator	Structural implication	Evidence basis / key limitation / sources
Neutron irradiation / RIVE	High-fluence historical data; effects increasingly evident around 10^{19} n/cm^2, $E > 0.1$ MeV	to ~ 0.5	to ~ 0.25	to ~ 0.5	Concrete expansion order ~ 1% at significant exposure	Aggregate swelling, microcracking, CBS stress redistribution and stiffness loss	PROVENANCE: Historical screening range (Class i). SELECTION: Lower envelope of the Field et al. multi-source compilation (307 f_c, 62 f_t, 138 E, 114 expansion measurements) and related NUREG/CR datasets [5–9, 22]; retained only data with identifiable mineralogy and consistent $E > 0.1$ MeV cut-off; excluded tests with uncontrolled moisture/temperature or non-paired legacy pairs. The ratios (f_c/f_{c0} to ~ 0.5; f_t/f_{t0} to ~ 0.25; E/E_0 to ~ 0.5) are order-of-magnitude triggers at fluences $\gtrsim 10^{19}$ n/cm^2, NOT design reduction factors or acceptance limits; strongly controlled by mineralogy, spectrum, flux, T, moisture and restraint.
Gamma irradiation	Selected gamma-only experiment ~ 6×10^5 Gy	~ 0.90	~ 0.95	NR	Radiolysis/ drying; no transferable strain	Heating/ moisture-related dimensional and property changes remain uncertain	PROVENANCE: Single-study observation (Class iii). SELECTION: One commonly cited gamma-only experiment reporting ~ 10% f_c and ~ 5% f_t/flexural loss at ~ 5–6 $\times 10^5$ Gy [2, 7]; retained solely as magnitude illustration. Many other 'gamma' studies are confounded by simultaneous neutron exposure, dose rate, heating or drying [2, 6, 7]. NOT a transferable engineering limit; gamma dose must be interpreted with dose rate, T and moisture.
ASR	Advanced material-level expansion ~ 0.5–1.5%	0.40–0.60	0.20–0.35	0.20–0.40	Expansion ~ 0.5–1.5%	Cracking and stiffness/ serviceability loss; capacity depends strongly on restraint	PROVENANCE: Multi-study observed envelope (Class ii). SELECTION: Span of residual ratios reported in historical material-level compilations at advanced free expansions of ~ 0.5–1.5% (f_c loss ~ 40–60%, f_t ~ 65–80%, E ~ 60–80% → residual f_c/f_{c0} 0.40–0.60, f_t/f_{t0} 0.20–0.35, E/E_0 0.20–0.40) [2, 15]. Extreme outliers from atypical aggregates or purely unrestrained tests were discarded. Envelope is NOT a design reduction factor; restraint, alkalis, moisture and T govern structural transferability.

Mechanism	Exposure/ condition	f_c/f_{c0}	f_t/f_{t0}	E/E_0	Expansion / physical indicator	Structural implication	Evidence basis / key limitation / sources
ASR – Seabrook	In-service affected cores	≈ control	NR	~ 0.47 of expected	Plant-specific deformation/ cracking	Illustrates that modulus may degrade before compression	PROVENANCE: Single-study (plant) observation (Class iii). SELECTION: In-service Seabrook cores [15]; modulus ~ 0.47 of an 'expected' value that was not a measured construction-age baseline. Retained only to illustrate that modulus may degrade before compressive strength under restrained ASR. NOT a transferable residual-property factor; plant-specific materials, exposure and restraint apply.
DEF	High early-age T + sulfate/ aluminate chemistry + later moisture	NR	NR	NR	Progressive expansion	Ettringite-associated cracking, deformation and permeability increase	PROVENANCE: Mechanistic evidence only — no residual-property envelope. SELECTION: No transferable normalized $f_c/f_t/E$ relationship was found in the reviewed corpus [10, 14, 31]; hence NR entries. Diagnosis requires early-age T records, binder chemistry, moisture history and petrography together. No screening ratio is proposed.
Thermal exposure	Moderate service to severe accident/ fire heating	0.98 at 75 °C/ 8 mo; 0.23 at 600 °C/ 1 mo	0.94; 0.38	NR	Drying/ thermal strain; severe cracking at high T	Prestress, deformation and residual-capacity effects	PROVENANCE: Single-study examples within broad literature (Class iii). SELECTION: Representative residual ratios from documented thermal series (e.g. 0.98 at 75 °C/8 mo; 0.23 at 600 °C/1 mo) [32]; chosen to show the contrast between modest service-temperature change and severe accident/fire loss. Heating rate, duration, moisture, aggregate, load and cooling are major moderators; values are NOT generic residual-capacity factors.
Sulfate / boric attack	180-d accelerated chemical exposures	UHPC/ NSC: 0.893/ 0.477; extreme boric: ~ 0.72	UHPC/ NSC: 0.830/ 0.446	UHPC/ NSC: 0.806/ 0.672	Mass loss or expansion, mechanism-specific	Local strength loss and increased exposure of embedded steel	PROVENANCE: Single-study comparative observations (Class iii). SELECTION: Residual ratios from documented accelerated exposures (UHPC vs NSC; extreme boric acid ~ 180 000 ppm, 70 °C, 180 d →~ 27.8% f_c loss) [26, 28, 33, 34]. Retained to illustrate concentration-, binder- and transport-dependence. Laboratory concentration, solution renewal and exposed surface area limit direct service-life extrapolation; NOT transferable design residuals.
Leaching / carbonation	Low-Ca water or CO_2 ingress	NR	NR	NR	Porosity/ depth changes; usually not intrinsic swelling	Transport increase and reinforcement-corrosion risk	PROVENANCE: Mechanistic evidence only — no bulk residual-property envelope. SELECTION: Accelerated tests raise diffusivity by 1–2 orders of magnitude but cannot be converted directly to service age [10, 14]; hence NR. Assessment should use degradation depth, moisture, cracking and cover rather than bulk strength. No screening ratio is proposed.

Mechanism	Exposure/ condition	f_c/f_{c0}	f_t/f_{t0}	E/E_0	Expansion / physical indicator	Structural implication	Evidence basis / key limitation / sources
Reinforcement / liner corrosion	Carbonation, chlorides, leaks, moisture traps	–	–	–	Corrosion-product expansion / local wall loss	Steel area/ bond loss, prestress reduction or loss of leak-tightness	PROVENANCE: Operating experience + corrosion science — function-based, not bulk-ratio. SELECTION: No bulk $f_c/f_t/E$ screening ratios apply; localized liner/tendon damage can govern confinement or prestress while global concrete strength remains high [1, 2, 28]. Acceptance must be tied to the function of the affected steel component and to inspectability, not to a concrete residual-strength factor.

Note: Property ratios are residual values relative to a reference condition (f_c/f_{c0}, f_t/f_{t0}, E/E_0). NR = not reliably reported or transferable. Numerical entries are assigned to one of three provenance classes, stated in the rightmost column and derived as follows. (1) Historical screening range — lower envelope of a multi-source historical compilation after exclusion of tests lacking mineralogy, consistent energy cut-off, or controlled moisture/temperature (applied to neutron/RIVE). (2) Multi-study observed envelope — span of residual ratios from independent experimental programmes under comparable exposure definitions, after discarding atypical-aggregate or free-expansion outliers (applied to ASR material-level data). (3) Single-study observation — residual ratio from one adequately documented experiment or plant case, retained only for illustration of magnitude and mechanism (applied to gamma-only, Seabrook ASR, thermal and chemical rows). Selection therefore rests on source type, completeness of material/exposure characterisation, and exclusion of non-transferable or confounded data; formal statistical pooling was not used. These values are not confidence intervals, design reduction factors, code-allowable residuals, or universal safety thresholds; they are order-of-magnitude indicative ranges intended solely as triggers for plant-specific material and structural assessment and must not be applied without regard to aggregate mineralogy, exposure conditions, restraint and the credited safety function. Symbols: f_c = compressive strength; f_t = tensile strength; E = elastic modulus; subscript 0 denotes the reference (unaged) condition.

Derivation of the quantitative screening envelopes in Table 3 followed a structured filtering of the compiled evidence into three provenance classes. (i) Historical dataset / order-of-magnitude screening range: used where a heterogeneous multi-source compilation (e.g., the Field et al. irradiation database [9] and NUREG/CR compilations [2]) shows property changes becoming increasingly evident beyond a characteristic exposure level; the residual ratios reported (e.g., f_c/f_{c0} to $\sim$ 0.5 for high-fluence RIVE) represent the lower envelope of the compiled observations after exclusion of tests lacking mineralogical identification, inconsistent energy cut-offs, or uncontrolled moisture and temperature, and serve only as order-of-magnitude triggers for plant-specific assessment. (ii) Multi-study observed envelope: used where several independent experimental programmes report residual-property ranges under comparable exposure definitions (e.g., ASR expansion of $\sim$ 0.5–1.5% with $f_c/f_{c0} \approx$ 0.40–0.60 [2, 15]); the envelope spans the central body of reported results after discarding extreme outliers associated with atypical aggregate mineralogy, unrestrained free-expansion tests, or accelerated conditions far outside NPP service envelopes. (iii) Single-study observation: used where only one adequately documented experiment or plant case is available (e.g., selected gamma-only test at $\sim 6 \times 10^5$ Gy [2, 7]; Seabrook ASR cores [15]); individual thermal or boric-acid series [32–34]); these entries are explicitly labelled as single-study so that they cannot be mistaken for transferable engineering limits. In all three classes, values were retained only when material and exposure characterisation was sufficient to judge transferability to NPP concrete. Formal statistical pooling was not performed because aggregate mineralogy, spectrum/flux, temperature, moisture, geometry, restraint and test duration differ systematically across the source studies. Consequently, the envelopes in Table 3 must not be interpreted as design reduction factors, code-allowable residuals, or universal safety thresholds; they are structured starting points for plant-specific triage and uncertainty-aware analysis.

5.1. Reduction of compressive and tensile strengths

Compressive strength is often a lagging indicator. Neutron irradiation and ASR can substantially reduce tensile resistance and modulus before severe compressive loss develops [2, 6, 7, 9]. Historical

high-fluence datasets include lower-envelope residual values corresponding to compressive-strength losses approaching 50% and tensile-strength losses approaching 75%. However, these observations arise from heterogeneous, non-paired experiments and should not be interpreted as universal reduction factors (Table 3). For ASR, the broad 0.5–1.5% expansion envelope corresponds to material-level losses of ~ 40–60% in compression and ~ 65–80% in tension [2].

These observations also affect sampling strategy. Core compressive strength is valuable for verifying material condition, but it should be complemented by splitting/direct tensile or fracture-related tests where cracking mechanisms dominate. Comparison with archive material or unaffected concrete from the same original mixture is preferable to comparison only with nominal design strength, especially when ageing changes stiffness more than peak compression [10, 15].

Plant evidence reinforces this hierarchy: at Seabrook, core compressive strengths from affected and comparison regions were ultimately similar, whereas measured modulus was only about 47% of the expected value [15]. Moderate thermal exposure may leave strength nearly unchanged, while severe heating can cause large losses [32]. Aged concrete should therefore be characterized by a property set—at minimum compression, tensile-related response and modulus—rather than f_c alone, because reliance on compressive strength alone can materially underestimate stiffness loss and the structural implications of distributed cracking.

5.2. Evolution of elastic modulus and structural stiffness

Elastic modulus responds directly to distributed cracking and is frequently more sensitive than compressive strength under RIVE, ASR, thermal incompatibility and decalcification. Material modulus, however, is not identical to structural stiffness: restraint, reinforcement and chemical prestressing (by adding special additives) can alter global response, and ACES analyses show that ASR-induced prestressing can partially offset material softening in some structural configurations [30]. Static core modulus, dynamic non-destructive testing response and global structural measurements should therefore be interpreted together.

Static and dynamic stiffness indicators are not interchangeable. Ultrasonic or resonance measurements operate at small strain and are sensitive to distributed cracking and moisture, whereas static modulus from cores represents the constitutive response over a larger strain range. Their greatest value is obtained when NDT trends are calibrated against physical samples and then propagated into structural models rather than interpreted as stand-alone material properties [37].

5.3. Creep and relaxation: Influence of moisture and irradiation

Creep, shrinkage and steel relaxation govern long-term prestress and stress redistribution rather than abrupt strength loss. Moisture is central because drying adds deformation beyond basic creep. Forsmark prestressing tests measured ~ 1.4% relaxation after 1000 h at 20 °C and ~ 2.7% at 50 °C, while containment studies showed that measured moisture histories improve prestress-loss predictions [16]. LTO should therefore trend measured tendon force against the minimum required value rather than rely on a single deterministic creep model.

Containment monitoring shows why environmental measurements matter. Temperature and humidity vary through wall thickness and elevation, so a single generic value can bias creep and relaxation predictions. Long-term tendon-force measurements provide a direct check on combined elastic, creep, shrinkage, relaxation and other losses and should therefore be used to update rather than merely confirm deterministic design estimates [1, 16].

Irradiation–creep interaction remains uncertain but structurally important: creep can relax RIVE-generated stresses, so accelerated high-flux tests may overpredict cracking if expansion develops much faster than service-time relaxation [7, 29].

5.4. Drying shrinkage and autogenous volume changes

Drying shrinkage creates long-lived gradients in massive walls because surfaces dry while the core remains moist; autogenous shrinkage becomes more important in very low w/b UHPC. Temperature adds immediate and delayed deformation, while RIVE, ASR and DEF may impose competing expansion. For LTO, moisture transport, shrinkage, creep, thermal strain and internal swelling should therefore be treated as a coupled time-dependent state rather than independent final strains [14, 16].

Long-term moisture gradients also couple structural and shielding performance. Surface drying increases restrained shrinkage while internal water is retained much longer; very low-w/b concretes can additionally develop significant self-desiccation before external drying becomes dominant. These scale effects should be represented explicitly when transferring laboratory shrinkage data to metre-thick containment or CBS walls [14].

Overall, the most sensitive ageing indicators are commonly modulus, tensile/fracture response and dimensional change; structural significance depends on restraint, prestress and load redistribution. This distinction guides the inspection strategy in Section 6.

6. Assessment and monitoring of concrete structures

Assessment of NPP concrete should establish the current condition, spatial extent, progression, and safety significance of degradation under the practical constraints of thick sections, dense reinforcement, liners, tendons, and restricted access. A defensible assessment therefore requires a graded combination of historical records and baseline information, visual inspection, targeted non-destructive testing, structural health monitoring (SHM) or other functional monitoring, selective destructive verification, and structural analysis [1, 10, 37, 38]. As illustrated in Fig. 2, these activities form an evidence-based decision process rather than a rigid linear sequence: findings at any stage may require additional inspection, sampling, model updating, or reassessment before an LTO decision can be supported.

The assessment strategy should be governed by the credited safety function, the suspected degradation mechanism, accessibility, and the decision that the evidence must support. Screening methods are appropriate for broad condition mapping, whereas diagnosis and structural acceptance require progressively stronger and more independent evidence. Selective coring or local opening should therefore be used when non-destructive evidence remains ambiguous, when competing mechanisms must be distinguished, or when measured material properties are required for calibrated structural analysis.

6.1. Visual inspection and condition assessment

Visual examination remains the primary screening method for cracking, spalling, leakage, staining, corrosion products and joint movement, but its value lies in repeatable mapping and trending. Surface appearance alone cannot identify mechanism or residual capacity; progressive or safety-significant anomalies should trigger targeted NDT, material verification and re-analysis [1, 14].

Condition classification should consider both activity and consequence. Stable cracking that is understood and bounded may justify routine surveillance, whereas progressive deformation, leakage, liner distress or unexplained cracking should trigger enhanced NDT and structural evaluation [14, 38]. This avoids both under-reaction to functionally significant local damage and over-reaction to visually severe but structurally restrained cracking.

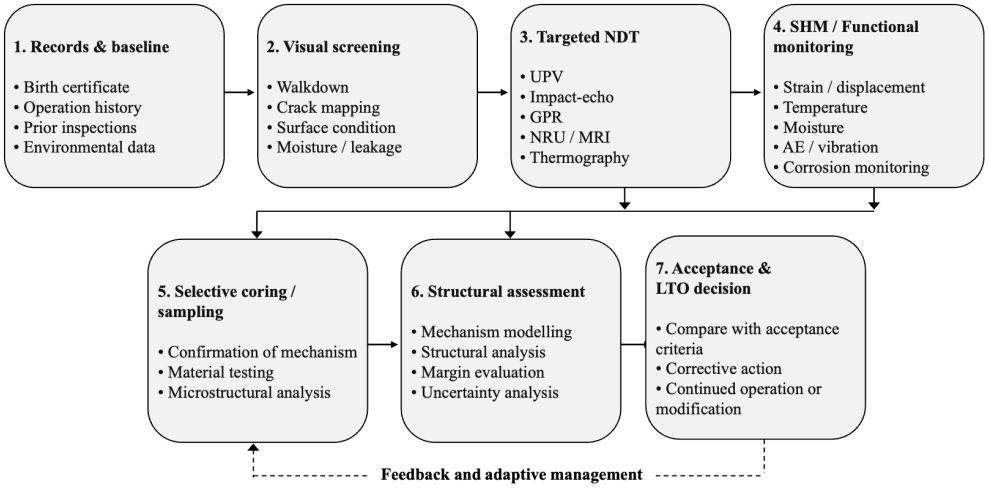


Figure 2. Evidence-based assessment and long-term-operation decision framework: GPR - Ground-Penetrating Radar; NRU/MRI - Nonlinear Resonant Ultrasound/Magnetic Resonance Imaging; AE - Acoustic Emission

6.2. Non-destructive testing techniques

NDT extends condition assessment into regions that cannot be characterized adequately by visual inspection alone, but nuclear concrete structures present particular challenges because of their thickness, aggregate heterogeneity, multiple reinforcement layers, liners, and restricted access [37]. Method selection should therefore be based on the target defect, accessible surfaces, required depth, and the decision to be supported rather than on nominal penetration capability alone.

For thick NPP structures, acoustic methods are generally better suited to detecting internal cracking, delamination, voids, and thickness variations, electromagnetic methods to reinforcement, duct, and interface mapping, and electrochemical methods to corrosion-related assessment. Because each technique responds differently to reinforcement, moisture, geometry, and defect orientation, combining independent physical principles generally provides more reliable evidence than reliance on a single method [37].

In practice, the inspection programme should answer four questions: which credited safety function may be affected; which degradation mechanism or defect is suspected; what surfaces and internal regions are accessible; and whether the examination is intended for screening, diagnosis, trending, property calibration, or structural acceptance. For suspected ASR, crack and deformation mapping may be supplemented by acoustic assessment and petrographic or core verification. For highly irradiated CBS regions, exposure reconstruction and aggregate characterization should inform targeted ultrasonic examination, monitoring, and selective material sampling. For reinforcement or liner corrosion, geometry mapping should be combined with electrochemical, thickness, or local verification methods as required by safety significance. The principal capabilities, limitations, access requirements, and recommended applications of the available methods are summarized in Table 4.

a. Ultrasonic methods and hammer sounding

UPV is useful for relative mapping of uniformity and cracking, but moisture, aggregate and steel affect velocity; conversion to mechanical properties requires plant-specific calibration. Pulse-echo arrays, tomography and impact-echo provide one-sided internal imaging and can detect thickness, voids or duct anomalies at substantial depth under favorable conditions. Hammer sounding is best retained as rapid near-surface screening [37].

b. Acoustic emission and ground-penetrating radar

AE detects active crack growth or wire breakage and is valuable for proof tests or known progressive damage, whereas GPR rapidly maps reinforcement, ducts and interfaces but dense steel can mask deeper targets [37]. The strongest strategy is multi-method data fusion followed, where justified, by targeted cores or local opening.

c. Nonlinear resonant ultrasound and magnetic resonance methods

Nonlinear resonant ultrasound (NRU) and magnetic resonance imaging (MRI) -based approaches extend conventional NDT by probing microcrack density, distributed damage and internal moisture that linear ultrasonic or electromagnetic methods may miss. NRU exploits the nonlinear elastic response of damaged concrete: microcracking and contact-type defects shift resonance frequencies and increase harmonic generation under increasing drive amplitude, providing a sensitive indicator of early-stage or distributed damage from mechanisms such as ASR, radiation-induced microcracking and thermal distress. MRI and related magnetic-resonance methods map internal moisture distribution, porosity and, in some configurations, organic or chemical phase changes; in the NPP context they are most useful for diagnosing moisture-driven processes (ASR, DEF, leaching, corrosion risk) in thick sections where hydrogen-bearing phases also affect neutron moderation. Practical limitations include equipment access and magnetic environment constraints in operating plants, specimen size limits for high-resolution MRI, and the need for baseline or reference measurements. Consequently, NRU/MRI techniques are best deployed as complementary diagnostic tools within a multi-method programme—typically on cores, mock-ups or selected accessible surfaces - rather than as standalone acceptance methods, and their findings should be interpreted together with petrography, conventional ultrasonics and structural monitoring when used to support LTO decisions.

d. Infrared thermography

Infrared (IR) thermography maps surface and near-surface temperature fields and is used in NPP concrete assessment to detect delamination, voids, moisture anomalies, and areas of preferential heat flow associated with cracking or varying cover. Passive thermography exploits natural or operational temperature gradients (solar loading, diurnal cycling, or process heat), whereas active thermography applies controlled heating or cooling to enhance contrast. In massive nuclear structures, thermography is most effective for accessible exterior surfaces, liners, and attachments where delamination or moisture ingress produces measurable thermal signatures; it does not replace through-thickness acoustic or radiographic methods for deep internal defects. Interpretation must account for surface emissivity, coatings, reinforcement density, ambient conditions and transient thermal history, so results are best treated as qualitative or semi-quantitative screening that guides targeted ultrasonic or intrusive verification. When combined with visual inspection and other NDT, thermography supports efficient prioritisation of large concrete surfaces during outages and can contribute trend data if surveys are repeated under comparable thermal boundary conditions. As with other advanced methods, thermographic findings should be linked to the credited safety function and to structural analysis before they are used in LTO acceptance decisions.

6.3. Continuous structural health monitoring

SHM provides trends between inspections and is most valuable for strain, temperature, moisture, displacement, prestress and active damage. Baselines, temperature compensation, redundancy and linkage to structural models are essential because sensor drift and environmental strains can exceed the ageing signal [1, 38].

SHM is most useful when sensor trends are interpreted against predicted environmental and structural response. Redundancy is important because individual embedded sensors can fail over multi-decadal service, and baseline readings must be established early enough to distinguish ageing from construction or commissioning effects [1, 38].

a. Embedded instrumentation: Strain, temperature, and pore-pressure monitoring

Vibrating-wire gauges and thermocouples have multi-decadal NPP experience; fibre-optic systems can add distributed strain/temperature measurements. Hydrogeological or pore-pressure monitoring should be used where below-grade conditions justify it rather than as a universal requirement. New plants should place sensors concurrently with thermal, moisture, radiation and structural analyses.

Table 4. Comparison of NDT and SHM methods for NPP concrete structures

Method	Principle	Target	Access	Quantitative use	Rebar sensitivity	Advantages	Limitations	Recommended use
Visual + hammer	Optical / impact response	Cracks, spalling, leakage, near-surface delamination	Surface/ shallow	Low–moderate	Low	Fast, inexpensive, trendable	Surface-limited; operator dependent	Routine screening and anomaly mapping
UPV	Through-transmission elastic waves	Uniformity, cracking, relative stiffness	Medium–deep; preferably two-sided	Moderate with calibration	High	Mature and repeatable	Moisture, aggregate, steel, crack orientation affect velocity	Relative condition/ stiffness mapping
Ultrasonic echo / tomography + impact echo	Reflected/ resonant stress waves	Voids, cracks, thickness, ducts	One-sided; depth material/ geometry dependent	Moderate–high for geometry	High	Internal imaging	Backscatter and reinforcement complicate interpretation	Thick walls, ducts, targeted anomaly characterization [37]
GPR	Electro-magnetic reflection	Rebar, ducts, interfaces, cavities	One-sided	High for geometry	Very high	Fast as-built mapping	Dense steel masks deeper targets	Rebar/duct mapping before acoustic NDT/ coring
Acoustic emission	Passive transient waves	Active cracking, debonding, wire breaks	Structural-scale array	High for activity/ location	Moderate	Detects active damage	Noise/source classification; not residual capacity	Progressive cracking, proof tests, tendons
Electrochemical methods	Potential/ resistivity/ polarization	Corrosion probability/ activity	Near reinforcement	Moderate	Moderate	Direct corrosion relevance	Moisture-sensitive; not structural capacity	Carbonated/ chloride-affected concrete
Embedded / fibre-optic SHM	Strain/ temperature/ moisture sensing	Creep, shrinkage, thermal strain, swelling	Local or distributed	High where instrumented	Low	Long-term trends	Drift, failure, retrofit constraints	Containments, basemats, new-build critical zones
Tendon force + leakage tests	Mechanical force / pressure-leak measurement	Prestress and confinement function	Accessible tendon / whole containment	High for credited function	N/A	Direct safety-function evidence	May not localize hidden damage; bonded tendons difficult	Prestressed-containment and leak-tightness surveillance [16]

Method	Principle	Target	Access	Quantitative use	Rebar sensitivity	Advantages	Limitations	Recommended use
NRU / MRI	Nonlinear resonance / magnetic resonance	Microcrack density, distributed damage, internal moisture	Cores, lab specimens, selected accessible surfaces	Moderate–high for relative damage; limited absolute strength conversion	Low–moderate	Sensitive to early/distributed damage; moisture mapping (MRI)	Access/magnetic constraints; geometry-specific calibration; not standalone acceptance	Complementary diagnosis of ASR, RIVE microcracking, moisture-driven mechanisms
IR thermography	Infrared surface temperature mapping (passive/active)	Delamination, voids, moisture anomalies, near-surface defects	Accessible exterior surfaces, liners, attachments	Qualitative to semi-quantitative screening	Low	Rapid large-area survey; outage-friendly prioritisation	Surface-limited; emissivity/ambient sensitive; not deep internal imaging	Screening of accessible surfaces; guide targeted NDT/coring

Note: Penetration and detection capability depend strongly on concrete heterogeneity, reinforcement, defect geometry, frequency, surface condition and equipment; nuclear-representative validation is required [37].

b. Containment leak-rate and tendon-prestress monitoring

Containment leakage tests and tendon-force measurements are especially valuable because they probe credited function directly. Global leakage testing does not locate every hidden liner defect, and bonded tendons are difficult to measure directly; these tests must therefore complement local inspection, NDT and time-dependent modelling [16]. No single method demonstrates integrity: confidence comes from mutually consistent evidence across material, component and structural scales.

7. International design codes, standards, and regulatory frameworks

NPP concrete is governed by a layered system of structural codes, containment rules, in-service inspection, ageing-management guidance and national licensing requirements. These layers must remain clearly differentiated: (i) design and construction codes establish the initial structural and material basis; (ii) in-service inspection provisions (e.g., ASME Section XI IWL/IWE, KEPIC MI) govern periodic examination; (iii) ageing-management guidance (IAEA SSG-48, IGALL, national AMP frameworks) structures the evaluation of time-dependent degradation; and (iv) the plant licensing basis determines which specific edition of each document is legally controlling. The controlling basis for an existing plant is the edition incorporated into its licence, not automatically the latest publisher edition. For review purposes, older editions are retained where they document the design basis or historical practice, while current publisher and regulatory documents are cited separately to avoid implying that the newest code is automatically enforceable for an operating plant [17–19, 39–46].

7.1. ACI Framework: ACI 318, ACI 349, and ACI 359

ACI 318 provides the general structural-concrete foundation, ACI 349 adds nuclear safety-related requirements, and ACI 359 forms the concrete-containment basis with ASME Section III, Division 2 [17, 18, 39]. Current publisher editions include ACI CODE-318-25 and ACI CODE-349-23 [19, 41], but these should be distinguished from historical or regulator-endorsed editions.

7.2. ASME construction and in-service inspection

ASME Section III, Division 2 addresses construction of concrete containments, while Section XI IWL/IWE provides complementary in-service examination of concrete/prestress and metallic containment/liner systems. ASME currently publishes a 2025 Division 2 edition [40], whereas national regulators may adopt earlier editions with conditions. Structural design, in-service examination, prestress surveillance and leakage testing are therefore complementary layers.

7.3. IAEA ageing management and IGALL

IAEA SSG-48 frames ageing management across the life cycle, and IGALL provides generic ageing reviews, AMPs and TLAAs [43, 44]. The historical uploaded IGALL report [47] remains useful context, while ACES extends the same logic toward more quantitative monitoring and assessment [38].

7.4. USNRC and international regulatory perspectives

In the United States, RG 1.142 Rev. 3 addresses safety-related concrete other than containments [42], while 10 CFR Part 54 and the updated GALL-SLR/SRP-SLR documents support subsequent licence renewal [45, 46]. Canada uses the CSA N287.2 series for CANDU concrete containments [20]. Table 5 summarizes scope and relationships. Across jurisdictions, design-code compliance establishes the initial basis, whereas LTO depends on condition evidence, ageing management and demonstration of continuing functional margin.

7.5. East-Asian frameworks: China, Japan and Korea

Beyond the ACI–ASME–IAEA–USNRC corpus already reviewed, three major East-Asian nuclear programmes maintain parallel code and regulatory systems that are increasingly relevant for both domestic LTO and technology export. Although each system retains strong historical links to ASME Section III Division 2, ACI 349 and IAEA guidance, they have developed distinct durability, seismic and ageing-management provisions that affect concrete SSCs.

China. National and energy-industry standards now form a coherent hierarchy for nuclear concrete. The overarching technical standard is GB/T 51390-2019 Technical standard for concrete structures of nuclear power plants, which covers design, construction and acceptance of safety-related reinforced and prestressed concrete structures [48]. Prestressed concrete containments are treated in the energy-industry standards NB/T 20303-2024 (design) and NB/T 20332-2024 (construction) [49, 50]. Durability design for nuclear safety-related concrete is addressed specifically by NB/T 20549-2019 [51], while GB/T 51323-2018 provides the first national standard on maintenance and reliability appraisal of NPP buildings and structures [52]. These documents are applied under the regulatory oversight of the National Nuclear Safety Administration (NNSA). Chinese practice emphasises early-age thermal control, durability classification by exposure environment, and progressive incorporation of 60-year design life requirements for new plants.

Japan. Japanese requirements combine Japan Electric Association (JEA) codes, Atomic Energy Society of Japan (AESJ) standards and Architectural Institute of Japan (AIJ) guidelines, all applied under the post-Fukushima regulatory framework of the Nuclear Regulation Authority (NRA). Seismic design of buildings, civil structures and equipment is governed by JEAC 4601-2021 (technical code) and the companion JEAG 4601-2021 (guidelines) [53, 54]. Ageing management is codified in AESJ-SC-P005:2021 Code on Implementation and Review of Nuclear Power Plant Ageing Management Programs [55], which requires technical evaluations at 10-year intervals and supports the formal Plant Life Management (PLM) process. For concrete structures specifically, the AIJ Guidelines for Maintenance and Management of Structures in Nuclear Facilities (latest revision 2024) provide performance-based inspection, soundness evaluation and intervention criteria throughout service life [56]. Maintenance management itself is covered by JEAC 4209 [57]. The Japanese system therefore places strong emphasis on seismic capacity, periodic technical evaluation of ageing, and structure-specific maintenance guidelines rather than a single containment construction code equivalent to ASME Section III Division 2.

Republic of Korea. Korean practice is formalised in the Korea Electric Power Industry Code (KEPIC). Concrete containments are addressed by KEPIC SNB (based on ASME Section III Division 2 / ACI 359) [58], while other nuclear safety-related concrete structures follow KEPIC SNC (based on ACI 349) [59]. In-service inspection of concrete and metallic containment components is covered by the KEPIC MI series [60] (aligned with ASME Section XI IWL/IWE). Regulatory oversight is exercised by the Nuclear Safety and Security Commission (NSSC) with technical support from the Korea Institute of Nuclear Safety (KINS). Ageing is evaluated primarily through the Periodic Safety Review (PSR) process required under the Nuclear Safety Act; plant-specific ageing management programmes have been implemented fleet-wide since 2016. Prestress monitoring, structural integrity testing and ultimate-pressure-capacity evaluations follow KEPIC SNB together with regulatory guides that parallel USNRC RG 1.35 and RG 1.216.

Collectively, the Chinese, Japanese and Korean frameworks confirm the international consensus that design-code compliance establishes the initial licensing basis, while continued functional performance of concrete SSCs depends on ageing-management programmes, periodic safety reviews, and plant-specific condition assessment. They also illustrate that durability design, seismic detailing and prestress surveillance receive particular national emphasis, reinforcing the need for function-specific acceptance criteria already advocated in Sections 5–6 of this review.

7.6. From ageing evidence to an LTO safety case

For an operating plant, ageing evidence enters the LTO safety case through scope setting, ageing-management review, ageing-management programmes, and, where applicable, time-limited ageing analyses [43–46]. Inspection and monitoring are judged against plant-specific licensing-basis criteria and credited safety functions. Stable, bounded degradation may remain under surveillance; adverse trends or reduced analytical margin require additional characterization, re-analysis, corrective action, or programme revision. The evidence chain begins with establishing the current condition and identifying the governing degradation mechanism, followed by evaluation of its progression and projected evolution, assessment of the resulting functional consequences, comparison with applicable acceptance criteria, and selection of the appropriate management or corrective action. Universal limits are not proposed because acceptance criteria depend on the specific plant, structure, safety function, and licensing basis.

8. Recommendations for lifetime extension and new builds

The evidence supports through-life durability management rather than reactive repair. Existing plants must demonstrate that actual ageing remains bounded; new plants should reduce future uncertainty through material qualification, QA/QC, inspectability and lifetime records [10, 11, 38, 61].

8.1. Optimizing mix designs for long-term durability

Future mixtures should be qualified by function-specific performance: heat release, transport, creep/shrinkage, thermal expansion, ASR/ACR, corrosion resistance and shielding where relevant. Aggregate petrography/mineralogy is particularly important because it controls both internal reactivity and RIVE [6, 29]. Mass concrete should prioritize thermal control; SCC should be validated for congested placement; UHPC should remain selective until long-term irradiation and dimensional behaviour are better established [4, 26].

Performance-based qualification should include representative full-scale construction tests where rheology is critical. A theoretically durable mixture that segregates, traps voids around liner anchors or cannot be adequately cured in a massive placement is not a durable nuclear concrete in practice. Constructability and QA/QC are therefore integral material properties at the project scale [10, 18].

8.2. Prevention, mitigation, and intervention for ageing concrete

The first priority is to remove or reduce the driving exposure where feasible—for example, by controlling moisture ingress or persistent leakage—and to determine whether degradation is active or stable. Prevention remains preferable for ASR, DEF, corrosion, and irradiation-sensitive aggregate selection, but existing plants require a graded intervention strategy. Local repair or crack treatment may restore surface protection or leak resistance but should not be interpreted as arresting an active internal swelling reaction [1, 15, 31].

Table 5. Comparison of major international codes, standards, and regulatory frameworks for NPP concrete

Framework	Scope (document type)	Structure/ application	Design philosophy	Confinement/ shielding relevance	ISI / ageing-management / LTO role	Document role & licensing status
ACI 318	Design code - general structural concrete	General/ non-nuclear basis	Strength/ serviceability/ durability	No nuclear-specific confinement	None specific — general structural evaluation only (design code, not ISI/AMP)	ROLE: General design code (not nuclear-specific). LICENSING: Current publisher edition ACI CODE-318-25 [19] is not automatically the plant licensing basis; the governing edition is the one incorporated in the facility licence.
ACI 349	Design code - nuclear safety-related concrete excluding containments	Safety-related buildings/ supports/ pools	Nuclear loads, reliability, QA	Support/ shielding; not containment leak-tightness	None specific — mainly design/construction requirements (not ISI/ AMP)	ROLE: Design/construction code for nuclear safety-related concrete excluding containments. LICENSING: Current publisher edition ACI CODE-349-23 [41]; the licensing-basis edition may differ and is plant-specific.
ACI 359 / ASME III.2	Design/ construction code - concrete containments	RC/PC shell, liner, tendons, penetrations	Integrated structural/ confinement design	Direct containment function	None — construction/ design basis only (ISI is ASME XI, not III.2)	ROLE: Design/construction code for concrete containments (not in-service inspection). LICENSING: Historical design-basis edition [17] must be distinguished from the current publisher edition [40]; only the licensed edition is controlling for an operating plant.
ASME XI IWL/IWE	In-service inspection code - concrete/ prestress and metallic containment	Concrete/ prestress and metallic containment/ liners	Condition-based integrity verification	Direct containment relevance	Primary ISI provisions for concrete/prestress and metallic containment/liners; periodic examination and corrective action	ROLE: In-service inspection (ISI) code - not a design code and not ageing-management guidance. LICENSING: Edition and any conditions of use are set by the national regulator; not automatically the latest ASME publication year.
IAEA SSG-48 / IGALL	Ageing-management guidance - through-life AMP/LTO (not a design or ISI code)	Safety-significant SSCs	AMP/AMR/ TLAA, operating experience	Function-based	Primary international AMP/ AMR/TLAA guidance for LTO; not a substitute for national ISI codes	ROLE: International ageing-management / LTO guidance (AMP, AMR, TLAA). LICENSING: Guidance only - not national law and not a plant licensing basis by itself [43, 44, 47]; national regulators may adopt or adapt elements.

Framework	Scope (document type)	Structure/ application	Design philosophy	Confinement/ shielding relevance	ISI / ageing-management / LTO role	Document role & licensing status
USNRC	National licensing framework - U.S. legal/ regulatory basis	U.S. NPP SSCs	Licensing basis + consensus standards + AMP	Concrete/ containment via RGs, ASME, leakage requirements	Licence renewal / subsequent licence renewal framework (10 CFR 54; GALL-SLR / SRP-SLR) — licensing process, not a design code	ROLE: National legal/licensing framework (10 CFR, RGs, GALL-SLR/SRP-SLR). LICENSING: Controls which consensus standards and editions are endorsed for U.S. plants; latest industry code is not automatically endorsed [42, 45, 46].
CSA N287	Design + material + examination standard - CANDU concrete containment	CANDU containment	Integrated material-design-construction-testing	Direct containment/ shielding	Includes pre-operational and in-service examination requirements within the CSA N287 series	ROLE: Design, material, construction and examination standard for CANDU concrete containments. LICENSING: Edition is jurisdiction- and plant-licence specific (currently CSA N287 series, e.g. N287.2-17) [20]
China GB/ T 51390 / NB/T 20303 & 20332	National design/ construction/ durability standards - safety-related concrete & PCC	Design, construction, acceptance	Limit-state + durability classification	Direct containment (NB/T series)	Ageing/maintenance via NB/T 20549 (durability) and GB/T 51323 (reliability appraisal); separate from design codes GB/T 51390 & NB/T 20303/20332	ROLE: National/energy-industry design, construction and durability standards (under NNSA oversight). LICENSING: GB/T and NB/T documents form the Chinese regulatory basis for new and operating plants; 60-year design-life emphasis for new builds; not interchangeable with ACI/ASME editions.
Japan JEAC/ JEAG 4601 + AESJ-SC-P005 + AIJ Guidelines	Design code - general structural concrete	Buildings, civil structures, equipment	Seismic importance classification + performance-based maintenance	Containment via NRA technical standards	Ageing management: AESJ-SC-P005 (10-year evaluations); maintenance: AIJ Guidelines and JEAC 4209 — distinct from seismic design codes JEAC/ JEAG 4601	ROLE: Mixed - seismic design codes (JEAC/JEAG 4601), ageing-management code (AESJ-SC-P005), and maintenance guidelines (AIJ; JEAC 4209). LICENSING: Applied under NRA regulatory framework; no single ASME III.2-equivalent containment construction code; 10-year ageing evaluations are a regulatory PLM requirement, not optional guidance.
Korea KEPIC SNB / SNC / MI	Design code - nuclear safety-related concrete excluding containments	Design, construction, ISI	ASME-III.2 / ACI 349 based	Direct (SNB)	ISI: KEPIC MI (IWL/IWE equivalent); ageing management via PSR + fleet AMP — distinct from design codes KEPIC SNB/ SNC	ROLE: Design/construction (KEPIC SNB/SNC, ASME/ ACI-based) + ISI (KEPIC MI = IWL/IWE equivalent). LICENSING: Under NSSC/ KINS oversight; plant-specific licensing basis selects the applicable KEPIC edition; PSR and fleet AMP implement ageing management separately from the design codes.

Where degradation has reduced structural margin, plant-specific corrective actions may include

local strengthening, modification of load transfer, prestress management, repair or replacement of accessible corroded components, or enhanced surveillance supported by structural re-analysis. These measures should be regarded as categories of intervention rather than universally qualified solutions and require structure- and licensing-specific verification. RIVE in deeply irradiated concrete has no demonstrated reversal treatment; management therefore relies primarily on exposure characterization, monitoring, reassessment of functional margin, and structural intervention where technically feasible [5, 6, 30, 35]. Repair compatibility in stiffness, shrinkage, thermal expansion, permeability, and radiation environment should be verified rather than selecting repair materials on strength alone.

8.3. "Birth certificates" and through-life documentation

The concrete "birth certificate" is a comprehensive quality record and baseline data document that tracks the exact specifications, material origins, placement conditions, and initial inspection results of structural concrete from day one. A practical concrete "birth certificate" need not capture every possible variable. For new safety-significant concrete, based on the reviewed evidence, this review proposes that the minimum high-value dataset should include constituent provenance and mixture proportions, aggregate petrography/mineralogy, placement and curing records, early-age temperature history, baseline mechanical and transport properties, as-built reinforcement/liner/tendon information, non-conformances, and the spatial location of test or archive specimens [10, 11, 14]. Operating records should subsequently link relevant radiation, temperature, moisture, leakage, inspection, monitoring, and repair histories to the affected structural region.

Legacy plants will inevitably contain incomplete records. Missing information should be reconstructed hierarchically from original construction documentation, retained or archived specimens, targeted petrography/mineralogy and core testing, verified data from genuinely equivalent sister units, and finally conservative bounding assumptions subjected to sensitivity analysis. The objective is not to recreate unavailable historical information with false precision, but to identify which missing variables materially control the LTO decision and reduce those uncertainties selectively [10, 11].

8.4. Future research: Coupled modelling and harvested concrete

The main research gap is validation of coupled ageing over realistic timescales. Accelerated irradiation differs from LWR service in flux, temperature, moisture redistribution and available creep relaxation [6, 35]. Service-aged concrete harvested from decommissioned NPPs offers uniquely valuable validation material, especially when combined with controlled re-irradiation, mineral/microstructural characterization, mechanical testing and multiscale modelling [35, 36]. ACES irradiation and post-irradiation programmes provide an important experimental basis for this strategy [36].

Harvested cores are not automatically representative; their value depends on known location, fluence, temperature, moisture history and aggregate mineralogy. Coordinated harvesting programmes should therefore map structures before decommissioning and preserve both highly exposed and comparison material. This would provide benchmark datasets for irradiation–creep interaction, irradiation-assisted ASR, bond, fracture and advanced-concrete qualification [6, 35, 36].

Uncertainty should be separated into four categories: material variability, exposure reconstruction, measurement uncertainty, and model-form uncertainty. The appropriate treatment is graded. Bounding analysis and sensitivity studies are suitable where defensible parameter distributions are available, probabilistic analysis can complement deterministic bounding and sensitivity studies, while sequential updating methods may incorporate inspection and monitoring evidence as it accumulates. For safety-significant structures, uncertainty should be propagated to the functional quantity governing the decision- such as residual prestress, deformation, leakage-related integrity, or structural

capacity- rather than represented by a generic material safety factor. A universal reliability index is not proposed because regulatory acceptance targets and consequence classifications are plant- and jurisdiction-specific.

For existing NPPs, the practical approach begins with reconstruction of the baseline condition and exposure history, followed by identification of active degradation mechanisms, continued monitoring, projection of structural response, and evaluation of the remaining functional margin. For new NPPs, the objective is to design the concrete, instrumentation, archive specimens and documentation so those questions can be answered directly decades later.

9. Summary, conclusions, and novel contributions

This review synthesizes evidence across irradiation, ASR, DEF, thermal, chemical, and corrosion mechanisms to establish three cross-cutting conclusions that, while individually recognized in mechanism-specific literature, have not previously been integrated into a unified LTO framework. First, ageing mechanisms converge structurally through a limited set of state variables (dimensional change, cracking, stiffness/tensile degradation, transport, prestress evolution, and embedded-steel deterioration). Second, material damage and structural consequence are decoupled by restraint, prestress, and function (Sections 5 and 6). Third, the largest LTO uncertainty is transfer from accelerated or local material evidence to spatially heterogeneous, multi-decadal structural response (Sections 4.1.a, 4.5, and 8.4).

These findings support a graded LTO framework in which plant-specific exposure reconstruction and material provenance define the problem; inspection, NDT, SHM, and selective sampling constrain its current state; and calibrated structural analysis translates degradation into functional margin. Where evidence is heterogeneous in aggregate mineralogy, exposure definition, temperature, moisture, geometry, or loading - as is the case for irradiation, ASR, and thermal datasets reviewed here - bounding and sensitivity analysis are preferable to statistical pooling because the latter assumes exchangeability of observations, which is not satisfied. Deterministic thresholds become justified only when supported by plant-specific material characterization, exposure reconstruction, and structural analysis, as demonstrated in the Seabrook ASR case (Section 5.1) and the ACES irradiation programme (Section 8.4).

Consistent with the scope and methodology set out in Section 1, this review provides three specific novel contributions: (i) the explicit mapping of diverse degradation mechanisms onto a common set of six structural state variables; (ii) the assembly of quantitative, uncertainty-aware screening envelopes with transparent provenance and explicit transferability caveats; and (iii) an end-to-end, function-specific ageing-assessment workflow that links these elements to inspection, analysis and LTO decision-making. ACES deliverables, IAEA guidance and national regulatory documents remain the principal evidence base; the integrated framework and workflow are the original synthesis offered here.

The highest-priority research needs are service-rate irradiation evidence, controlled gamma-only studies, quantitative validation of coupled irradiation–moisture–creep–swelling interactions, and systematic characterization of concrete harvested from decommissioned NPPs [6, 35, 36]. For future plants, mineralogical qualification, early-age thermal records, inspectability, archived materials, and spatially linked lifetime data should be treated as part of structural-integrity design rather than as optional documentation. The resulting objective is not merely durable concrete, but a concrete safety system whose condition and residual margin can still be demonstrated credibly several decades after construction.

Declaration of generative AI and AI-assisted technologies

During the preparation of this manuscript, the author used Paperpal to improve the language, consistency and readability of the manuscript. The author reviewed and edited the output and take full responsibility for the accuracy, originality, and integrity of the manuscript.

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