Full Paper Transactions, SMiRT 28
Toronto, Canada, August 10-15, 2025

STRUCTURAL PERFORMANCE OF IN-SERVICE IRRADIATED CONCRETE BIOLOGICAL SHIELD AND REACTOR SUPPORT SYSTEM DURING A LOSS OF COOLING ACCIDENT

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ABSTRACT

This study investigates the structural performance of reactor pressure vessel (RPV) support systems under loss-of-coolant accident (LOCA) conditions with a focus on the effects of long-term neutron irradiation on the surrounding concrete biological shield (CBS). The CBS, a reinforced concrete structure enclosing the RPV, is subject to fast neutron fluence that can lead to radiation-induced volumetric expansion (RIVE) and degradation of mechanical properties. Projections indicate that neutron fluence in US pressurized water reactors (PWRs) will exceed damage thresholds by 80 years of operation. A structural analysis was conducted on shoe-type support systems to evaluate failure modes such as wedge failure, concrete crushing, and pry-out in unreinforced concrete. The analysis was performed using both analytical and numerical methods. Results show reduced safety margins when irradiation damage compromises reinforcement effectiveness, particularly at the depth of hoop reinforcement. The findings highlight the need for further analysis incorporating realistic reinforcement details to accurately assess the load-bearing capacity and safety of irradiated support systems during accidental loading scenarios.

FOREWORD

The analysis presented in this report assumes concrete properties obtained under accelerated irradiation conditions in test reactors. Although in-service irradiated concrete properties have yet to be obtained through a harvesting program, preliminary data from Japan and the United States indicate a reduction in the radiation-induced volumetric expansion as the fast neutron flux decreases toward in-service irradiation in pressurized water reactors. The analysis results presented herein will require re-evaluation in the future if significant differences in irradiated concrete properties between in-service and accelerated irradiation conditions are found.

INTRODUCTION

The primary function of the concrete biological shield (CBS) in light-water reactors (LWRs) is to protect equipment and personnel from neutron and gamma radiation escaping the reactor pressure vessel (RPV). Beyond this radioprotection role, the CBS in many pressurized water reactors (PWRs) also provides

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structural support to both the RPV and the nuclear steam supply system (NSSS). To ensure the safe and viable long-term operation of LWRs, it is crucial to understand and characterize the effects of radiation on concrete and its constituents and to develop predictive tools to assess the structural performance of the CBS during operation and accidental conditions. The main degradation mechanism is referred to as the *neutron radiation-induced volumetric expansion* (RIVE) of aggregate-forming minerals. This volumetric change is associated with degradation of the mechanical properties of irradiated concrete, including Young's modulus, as well as tensile and compressive strength at fluences greater than 10^{19} n/cm² (E > 0.1 MeV). Such fluence levels are predicted to be reached for all PWRs operating in the United States after 80 years of operation. The effects of in-service irradiation on the formation of structural damage using different modeling strategies are reported here and in the companion conference paper entitled "Effects of Mineral Composition on the Structural Performance of In-Service Irradiated Concrete Biological Shield." In addition, an initial study of the effect of irradiation on the structural performance of an RPV support system focusing on concrete degradation is presented. The companion paper also describes assessment of structural performance of the CBS subjected to in-service irradiation. The present analysis focuses on the performance assessment of the remaining structural capacity of the portion of the CBS directly located under the RPV support when subject to loss-of-coolant accident (LOCA) conditions occurring after prolonged irradiation exposure.

METHODS

Several designs of RPV support systems exist in PWRs operating in the United States. Appendix D of NUREG/CR-7280 provides design details for the reactor support systems of several US plants (Biwer et al., 2021). These systems can be categorized into vertical column support systems, neutron shield tanks, support skirts, cantilever beam supports, shoe-type supports, and ring girder supports. The support systems for reactors can be categorized based on the significance of irradiation-induced damage to the concrete around the system, as detailed in the following sections.

Low Significance

Information collected on *base skirt supports* was limited. Most Babcock and Wilcox PWRs in commercial operation incorporate a support skirt design similar to that used for boiling water reactors (BWRs). It appears that all in-service and accidental loads are directly transferred to the concrete foundation through the base skirt assembly. This area, which is located directly under the RPV, receives irradiation doses lower than the estimated damage threshold (Framatone, 2021). When a neutron shield tank is present, it is expected that the fast neutron flux will be dramatically reduced, thus preventing any significant irradiation-induced damage to the CBS (Bruck, 2018).

Moderate Significance

Column support–based systems rely exclusively on steel columns to transfer vertical loads to the foundation. The potential irradiation-induced degradation of the concrete surrounding the columns does not appear to affect their function. Lateral loads are transferred to the CBS wall through embedded horizontal brackets or anchors. Although this region is subject to lower fast neutron fluence, the anchor bond properties and the resistance of surrounding concrete may be affected by irradiation-induced damage.

High Significance

Support systems directly located under the nozzles include shoes resting on the CBS wall or a cantilever system embedded in the concrete. Some systems also include a ring girder located below the nozzle pads. The role of the ring girder in the distribution of vertical and horizontal loads from the RPV to the CBS has yet to be studied. Nevertheless, shoe-type and cantilever beam supports are essential for transferring both vertical and horizontal loads from the RPV to the CBS. The possible effects of irradiation on these systems require further study.

INTEGRITY OF SHOE-TYPE RPV SUPPORT SYSTEM

This section analyzes the bearing capacity of a generic RPV shoe-type support system with a focus on the resistance of the anchor group system embedded in concrete and subjected to in-service irradiation. A *shoe* or *saddle* is used under the RPV's inlet/outlet nozzles or between nozzle load brackets to restrict vertical and tangential movement while allowing for radial thermal expansion. The shoe is anchored to the biological shield wall and is typically seated directly on its top surface or set in a recess. To minimize heat transfer to the concrete, some designs incorporate cooling systems or taller vertical plates for better air cooling, although this places the weldment's load-bearing baseplate closer to radiation fields. In this study, a generic shoe support design was analyzed that includes six steel supports under the RPV nozzles, one for each hot and cold leg nozzle. Each nozzle support consists of an upper part attached to the nozzle and a lower supporting part that is anchored into the concrete's primary shield wall. This allows for thermal expansion while transferring loads to the concrete.

Loading

The maximum loads on PWR vessel supports are caused by the combined effects of postulated pipe ruptures (LOCAs), seismic events (safe shutdown earthquake [SSE]), and other license-based loads. These effects involve complex dynamic loading and structural analysis to address thermohydraulic, fluid-structure, and soil-structure interactions. Thermohydraulic and structural models are used to derive the loading conditions exerted on RPV support structures as informed by reactor components like the RPV, steam generators, and reactor coolant pumps, as well as the concrete structural components.

The selected tool for the thermohydraulic analysis of the system, including reactor cavity over-pressurization, is the Idaho National Laboratory RELAP5-3D code which can simulate transients in LWR systems such as loss of coolant and operational events like loss of feedwater and station blackout. For this project, a key feature of RELAP5-3D is its capability to model 3D thermohydraulics through dedicated 3D hydraulic components. In the event of a postulated instantaneous double-ended break of an RPV nozzle or a hot/cold leg, asymmetric mass flows, temperature, and pressure distributions occur in the containment compartments (Hosford et al., 1981), thus requiring 3D thermohydraulic modeling. RELAP5-3D enables the division of the system into control volumes connected by zero-dimensional junctions, creating a nodalization of the system. The code solves two-phase, two-component (water and steam/noncondensable gases) equations for mass, momentum, and energy, calculating thermohydraulic parameters for each volume and junction over time. A 3D model of the reactor cavity was created using RELAP5-3D's multid hydraulic component (Fig. 1).

The jet impingement force was calculated according to ANSI/ANS 58.2 guidelines using the following formula:

$$F_b = G_e^2 A_e \varrho_e g + A_e(P_e - P_{amb}), \tag{1}$$

where G_e is the mass flux, A_e is the break area, ϱ_e is the fluid density, g is gravitational constant, P_e is the pressure at the break plane, and P_{amb} is the compartment pressure. Figure 2 shows the mass flow and the jet impingement force for a double-ended guillotine break (DEGB). The maximum calculated forces are 6.03 MN, 6.03 MN, 3.01 MN and 1.51 MN for the DEGB 2A (0.766 m^2), Reference 1A (0.383 m^2), 0.5 A (0.192 m^2), and 0.25 A (0.0958 m^2), respectively.

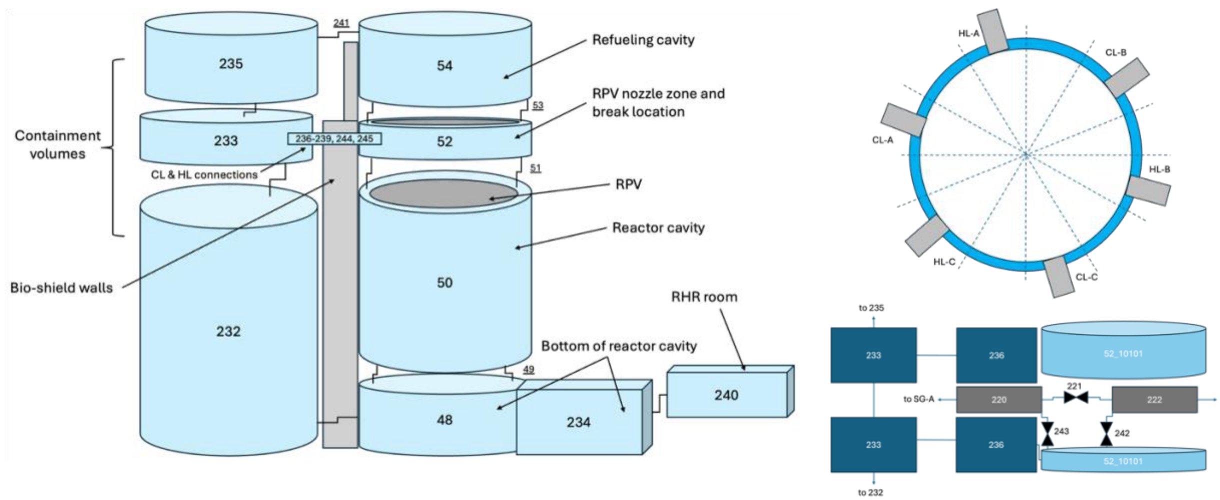


Figure 1. RELAP5-3D reactor cavity and containment nodalization (left), azimuthal discretization of the reactor cavity (top right), and cold leg A break nodalization (bottom right).

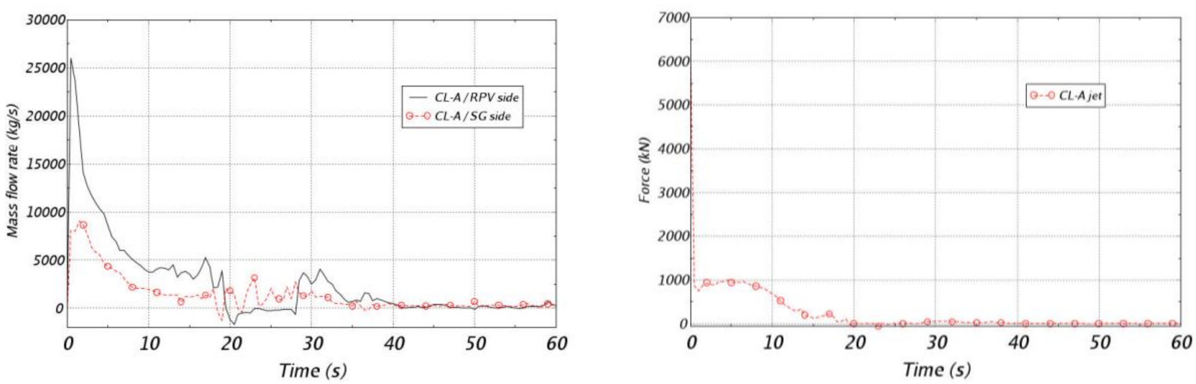


Figure 2. 2A DEGB case: mass flow at the breaks (left, and jet impingement force (right).

Anchorage Failure Modes

The shoe support system depends on a steel plate to transfer the vertical downward force exerted by the RPV and nozzle system, along with seven long anchors (diameter: 2¾ in. [~7 cm], length: ~2 ft [~60 cm]) to transfer shear force to the CBS wall. The anchor centers are positioned at a distance of approximately 23¾ in., which is also close to 60 cm. The height of the support box for the nozzle is 23 in. The horizontal force creates a bending moment at the interface between the bottom steel plate and the supporting concrete.

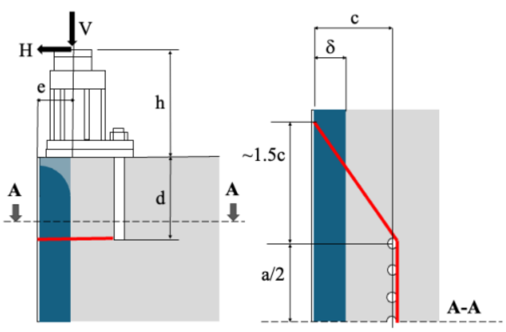
Unfortunately, the embedded reinforcement details were not publicly available in the literature at the time of this writing. The presence of reinforcement through the concrete thickness beneath the support plate plays a significant role in the shear strength of an anchor group near a concrete edge. Different failure modes are possible (see Figure 3 for details):

1. **Concrete edge failure** develops from the external anchors toward the edge of the concrete. The angle between the concrete edge and the fracture is approximately 30 degrees. In the absence of reinforcement, the shear capacity is controlled by the geometry of the fracture surface and the shear strength of plain concrete. If stirrups are present, then they can significantly increase the shear capacity of the anchored system.

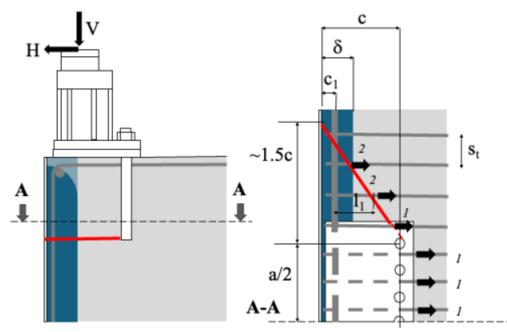
2. **Stirrup failure (yielding)** may occur if the tensile stress in the reinforcing bars exceeds the yield strength.

3. **Concrete crushing** occurs beneath the bottom steel plate of the support box. The effect is worsened by the bending moment caused by the horizontal force applied at the top of the shoe box. In plain concrete, compression stresses under the plate near the concrete edge may lead to a wedge-type fracture toward the cavity. Concrete failure by crushing requires reinforcement to tie the hypothetical wedge.

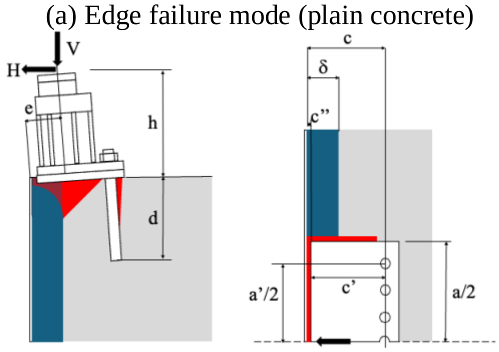
4. **Pry-out failure** occurs as a result of the rotation of the anchor.



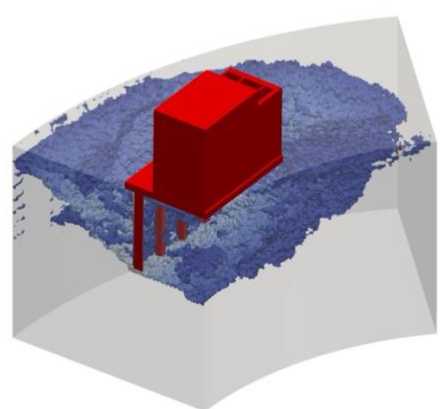
(a) Edge failure mode (plain concrete)



(a) Edge failure mode (reinforced concrete)



(c) Concrete crushing mode



(d) Pry-out mode

Figure 3. Hypothetical failure modes of the (reinforced) concrete below the nozzle shoe support.

Irradiation-induced damage may affect the portion of the concrete located directly under the support plate near the reactor cavity. The effects of irradiation on the concrete located below the support system depend on its elevation relative to the fuel at mid-core.

Plain concrete edge failure: The shear failure mode for plain concrete only accounts for the effects of the horizontal load H. The failure is assumed to occur along three planes: one horizontal plane located at the elevation of the bottom of the anchors, and two lateral outward planes. The three shear planes form a trapezoidal prism sloping toward the reactor cavity. The lateral planes meet the concrete edge at an estimated distance of approximately $1.5c$ (Sharma et al., 2016b). Assuming *a shear strength of approximately 3 MPa*, and neglecting the contribution of tension resistance, the edge failure shear resistance for the anchored system is approximately 8.75 MN when considering only the plain concrete contribution. The presence of irradiation-induced damage leads to a linear reduction in shear capacity: $\Delta V/V = -\delta/c$, where δ is the irradiation-induced depth, and c is the horizontal distance to the concrete edge.

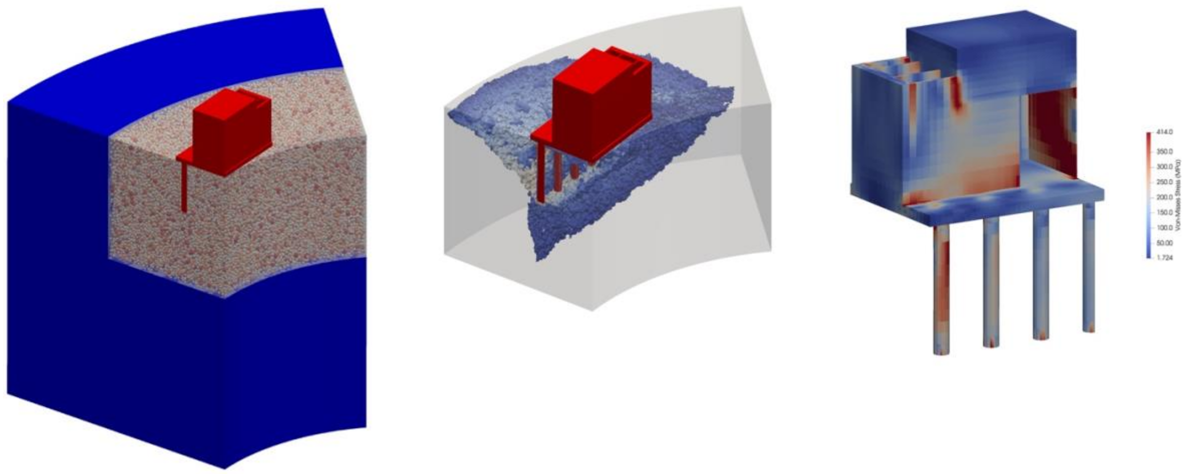
Reinforced concrete edge failure: Reinforcements in the concrete provide resistance to the wedge failure by contributing the hook and bond capacities of each bar crossing the failure plane. The hook capacity depends on the location of the failure crack relative to the hook specifically relative to the concrete cover of the vertical reinforcement, denoted as c_1. According to Sharma et al. (2016b), any stirrup not intercepted by the crack or with an anchorage length in the assumed breakout body $\leq \phi_s$ does not contribute to the anchorage load capacity. The total anchor group capacity is the sum of the hook and bond resistances of the contributing bars. Assuming values for material properties, the total capacity is found to be approximately 9.3 MN for each layer of reinforcement. The effect of irradiation-induced damage on the anchor group capacity depends on the damage depth. If the damage exceeds the location of the hook, then the hook capacity contribution is nullified, thus reducing the total anchor capacity by approximately 40% if the damage reaches approximately 7 cm. Similarly, the bond capacity is affected by the damage, with reductions of 8%, 22%, and 35% for damage depths of 10 cm, 15 cm, and 20 cm, respectively.

Concrete crushing: In pure flexure, only the effect of the horizontal load on the bending moment is considered. The resisting bending moment is given by a classical code expression assuming a linear strain field under the support plate. Using the value of 0.35% for the ultimate concrete strain leads to estimating the ultimate horizontal force at the top of the support box around 6.2 MN.

Pry-out failure: The pry-out failure mode is analyzed using the lattice discrete particle model (LDPM) method. The methods and results of this analysis are provided in the next section. Because of the lack of detailed reinforcement and dimensional data, the simulation is based on the following simplifying assumptions:

1. CBS reinforcement is not represented.
2. Symmetry boundary conditions are assumed, so only one half of the support is simulated.
3. The support block and anchors are modeled using hexahedral finite elements with elastic steel properties.
4. The LDPM method is used to simulate the concrete around the support block, extending 91.44 cm radially behind the block and 152.4 cm below the support plate.
5. The effects of irradiation are simulated by neglecting the contribution of the innermost LDPM elements within 10, 15, and 20 cm from the concrete surface facing the reactor cavity.
6. The concrete mix is assumed to have a maximum aggregate size of 50 mm and a minimum aggregate size cutoff at 25 mm.

The model geometry is shown in Fig. 4(a). The support block and anchors are shown in red, with the surrounding elastic concrete in blue. The failure surface is presented in Fig. 4(b) and the von Mises strain in the support block is depicted in Fig. 4(c) showing no yielding of the anchors.



(a) (b) (c)

Figure 4. (a) 45° wedge model of the CBS showing LDPM (opaque gray) with internal aggregate, along with the support block (red) and the surrounding elastic concrete (blue); (b) CBS wedge crack surface at a peak horizontal load of 4.25 MN with applied vertical loads; (c) Von Mises stress distribution within the support block showing no yielding of the anchors at a peak horizontal load of 3.26 MN without applied vertical loads.

Figure 5 summarizes the results of analytical calculations and simulations using the LDPM method for different failure modes, including plain concrete crushing, wedge failure, and pry-out failure. White diamonds represent damage without vertical load, and black diamonds account for vertical load. The red lines show the jet thrust force from RELAP5-3D for a full break area (1A) and a partial break area (0.5A). The horizontal load applied to the support system depends on the break area, the reactor system's dynamic response during LOCA, and design details allowing nozzle deformation. A jet thrust–induced horizontal load of approximately 6 MN at full break is considered an upper bound. The structural safety margin for unirradiated concrete is minimal in the crushing mode, although reinforcement (e.g., stirrups) can enhance the bearing capacity. The safety factor for wedge failure in unirradiated reinforced concrete is approximately 1.5. The bearing capacity for pry-out failure with plain concrete is lower than the design load, but reinforcement would significantly improve capacity. Further studies are needed.

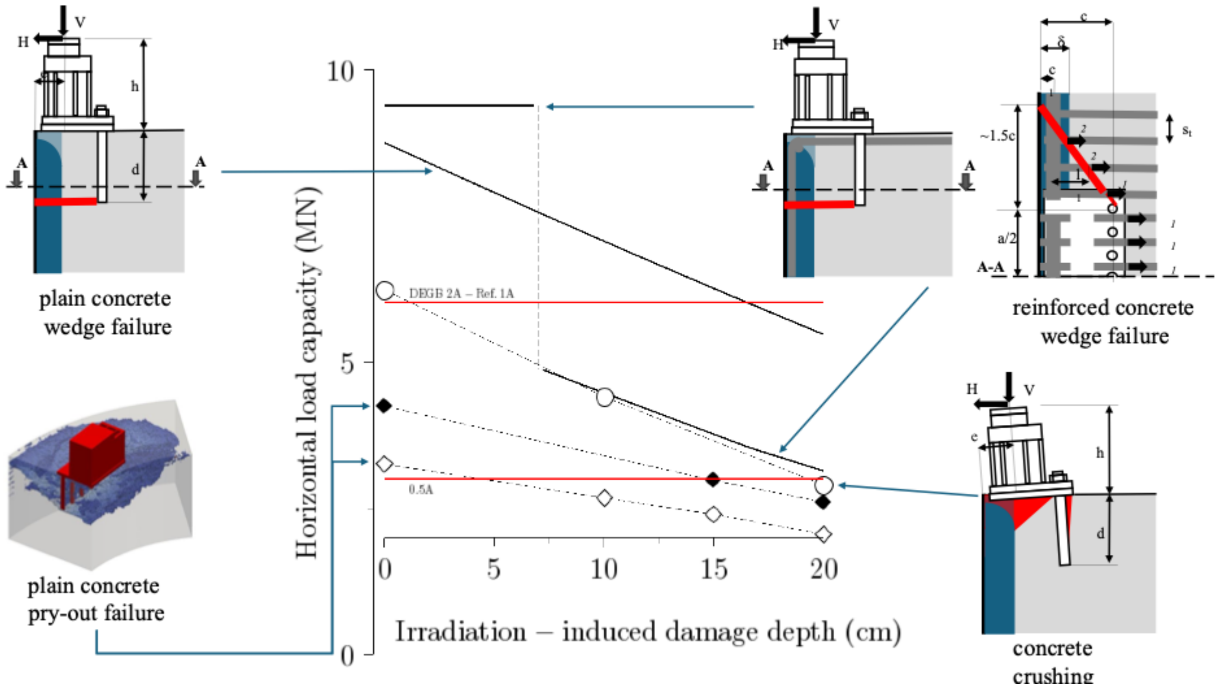


Figure 5. Evolution of the horizontal load capacity with the irradiation-induced damage depth.

CONCLUSION

The temperature and pressure conditions resulting from a hypothetical loss-of-coolant accident (LOCA) were studied using the RELAP5-3D code. When a nozzle break occurs, steam is released into the reactor cavity, gradually filling the cavity and connected volumes, including the containment building. After approximately 20 seconds from the initial start of the LOCA, the pressure stabilizes to a steady value of approximately 4 bars in both the cavity and the containment building. Similarly, the temperature in the cavity drops to a near-steady value of approximately 120°C. After one minute, the temperatures in the reactor cavity, refueling cavity, and containment building converge to similar values. The peak pressure and temperature values are influenced by the location of the nozzle break. As the break location moves further from the reactor pressure vessel (RPV), the steam flow is partially diverted to adjacent volumes, resulting in a reduction in pressure and temperature within the cavity. Furthermore, the steam jet at the nozzle break generates a thrust, applying an estimated horizontal load of approximately 6 MN in the worst-case scenario.

A structural analysis of the shoe support system for a generic PWR was conducted with a focus on horizontal loads during accidental conditions because reinforcement details were unknown. Several failure modes were analyzed, including horizontal wedge formation, concrete crushing, and pry-out failure. The analysis showed that the concrete crushing failure mode is more detrimental than the wedge failure mode for unirradiated concrete, whereas pry-out failure leads to the lowest load capacity but does not account for reinforcement. Irradiation-induced damage reduces the support system's structural capacity Edge failure analysis showed a structural capacity approximately 1.5 times higher than the jet thrust estimated by RELAP5-3D for the maximum break area. However, once irradiation damage reaches the hoop reinforcement location in the CBS, the structural capacity drops significantly. This loss can be partly offset by plain concrete's shear resistance, although the safety factor reduces to approximately 1.25. Numerical simulations of unreinforced concrete showed pry-out failure at lower loads. Further studies on the role of reinforcement are ongoing.

ACKNOWLEDGMENTS

This research was sponsored by the US Department of Energy (DOE) Office of Nuclear Energy Light Water Reactor Sustainability Program Materials Research Pathway under contract DE-AC05-00OR22725 with UT Battelle LLC/Oak Ridge National Laboratory (ORNL).

REFERENCES

ANSI/ANS 58.2. (1988). "Design Basis for Protection of Light Water Nuclear Power Plant Against the Effect of Postulated Pipe Rupture." Technical Report ANSI/ANS 58.2. American Nuclear Society.

Biwer, B., Ma, D., Xi, J. and Jing, Y. (2021). "Review of Radiation-Induced Concrete Degradation and Potential Implications for Structures Exposed to High Long-Term Radiation Levels in Nuclear Power Plants," NUREG/CR-7280, ANL/EVS-20/8 *U.S. Nuclear Regulatory Commission*.

Bruck, P. (2018). "Guidance for Aging Management of Concrete Biological Shields that Utilize a Neutron Shield Tank, Basis for Concrete Biological Shield Wall for Aging Management." Electric Power Research Institute, 3002013051.

Cheniour, A., Davidson, E., Le Pape, Y., Pandya, T., Collins, B., Spencer, B., Godfrey, A. and Asgari, M. (2023). "A Structural Model of the Long-Term Degradation of the Concrete Biological Shield." *Nucl. Eng. Des.*, 405:112217.

Denisov, A.V., Dubrovskii, V.B. and Solovyov, V.N. (2012). "Radiation Resistance of Mineral and Polymer Construction Materials." ZAO MEI Publishing House. in Russian.

Field, K., Remec, I. and Le Pape, Y. (2015). "Radiation Effects on Concrete for Nuclear Power Plants – Part I: Quantification of Radiation Exposure and Radiation Effects." *Nucl. Eng. Des.*, 282:126–143.

Framatome (2021). "Framatome Reactor Vessel and RCP TLAA and Aging Management Review Input to the ONS SLRA," ANP-3898NP, Revision 0.

Hosford, S.B., Mattu, R., Meyer, R.O., Throm, E.D. and Tinkler, C.G. (1981). "Asymmetric Blowdown Loads on PWR Primary Systems – Resolution of Generic Task Action Plan A-2." NUREG-0609. US Nuclear Regulatory Commission.

Le Pape, Y., Field, K.G. and Remec, I. (2015). "Radiation Effects in Concrete for Nuclear Power Plants – Part II: Perspective from Micromechanical Modelling." *Nucl. Eng. Des.*, 282:144–157.

Le Pape. Y. (2015b). "Structural Effects of Radiation-Induced Volumetric Expansion on Unreinforced Concrete Biological Shields." *Nucl. Eng. Des.*, 295:534–548, 2015.

Le Pape, Y., Alsaid, M.H.F. and Giorla, A.B. (2018). "Rock-Forming Minerals Radiation-Induced Volumetric Expansion – Revisiting the Literature Data." *J. Adv. Con. Tech.*, 16:191–209.

Le Pape, Y., Sanahuja, J. and Alsaid, M.H.F. (2020). "Irradiation-Induced Damage in Concrete-Forming Aggregates – Revisiting Literature Data through Micromechanics." *Mat. Struc.*, 53:35.

Le Pape, Y., Cheniour, A., Alnaggar, M., Davidson, E., Choi, Y.-J. and Sabatino, S. "Effects of Neutron Irradiation on the Structural Performance of the Concrete Biological Shield in Light Water Reactors." *Transactions, SMiRT 27*, Yokohama, Japan, March 3-8, 2024.

Maruyama, I., O. Kontani, M. Takizawa, S. Sawada, S. Ishikawa, J. Yasukouchi, O. Sato, J. Etoh, and T. Igari. (2017). "Development of the Soundness Assessment Procedure for Concrete Members Affected by Neutron and Gamma-Irradiation." *J. Adv. Con. Techn.*, 15:440–523.

Sharma, A., Eligehausen, R. and Asmus, J. (2017). "Experimental Investigation of Concrete Edge Failure of Multiple-Row Anchorages with Supplementary Reinforcement." *Struct. Concr.*, 18 (1):153–163.