FPL

L-2018-187
10 CFR 54.17

October 5, 2018

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555-0001

Re: Florida Power & Light Company
Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Turkey Point Units 3 and 4 Subsequent License Renewal Application
Revision to SLRA Section 3.5.2.2.2.6, Reduction of Strength and Mechanical Properties of Concrete Due to Irradiation

References:

1. FPL Letter L-2018-004 to NRC dated January 30, 2018, Turkey Point Units 3 and 4 Subsequent License Renewal Application (ADAMS Accession No. ML18037A812)
2. FPL Letter L-2018-082 to NRC dated April 10, 2018, Turkey Point Units 3 and 4 Subsequent License Renewal Application – Revision 1 (ADAMS Accession No. ML18113A134)
3. Turkey Point Nuclear Generating Units 3 and 4 - Plan for the Irradiated Concrete Technical Issue Regulatory Audit Regarding the Subsequent License Renewal Application Review (EPID NO. L-2018-RNW-0002) (ADAMS Accession No. ML18173A087)

Florida Power & Light Company (FPL) submitted a subsequent license renewal application (SLRA) for Turkey Point Units 3 and 4 to the NRC on January 30, 2018 (Reference 1) and SLRA Revision 1 on April 10, 2018 (Reference 2).

The purpose of this letter is to provide, as an attachment to this letter, a revision to SLRA Section 3.5.2.2.2.6, Reduction of Strength and Mechanical Properties of Concrete Due to Irradiation. In accordance with Reference 3, the NRC conducted an audit on SLRA Section 3.5.2.2.2.6, Reduction of Strength and Mechanical Properties of Concrete Due to Irradiation. The audit took place from July 17 through 19, 2018, and from September 18 through 20, 2018. The staff performed extensive reviews of the irradiation analysis and supporting technical documentation. As a result of these reviews, a revision to SLRA Section 3.5.2.2.2.6 has been prepared and attached. This section supersedes the current Section 3.5.2.2.2.6 in SLRA, Revision 1 in its entirety.

A084
NRR

If you have any questions, or need additional information, please contact me at 561-691-2294.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on October 5, 2018.

Sincerely,

William Maher
Senior Licensing Director
Florida Power & Light Company

WDM/SF

Attachment: Revision to SLRA Section 3.5.2.2.2.6

cc:

Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, Turkey Point Plant
Project Manager, USNRC, Turkey Point Nuclear
Project Manager, USNRC, SLRA
Project Manager, USNRC, SLRA Environmental
Ms. Cindy Becker, Florida Department of Health

Revision to SLRA Section 3.5.2.2.2.6

Discussion

As noted in NUREG-2191 and NUREG-2192, reduction of strength, loss of mechanical properties, and cracking due to irradiation could occur in PWR and BWR Group 4 concrete structures that are exposed to high levels of neutron and gamma radiation. These structures include the reactor (primary/biological) shield wall, the sacrificial shield wall, and the reactor vessel support/pedestal structure. Data related to the effects and significance of neutron and gamma radiation on concrete mechanical and physical properties is limited, especially for conditions (dose, temperature, etc.) representative of light-water reactor (LWR) plants. However, based on literature review of existing research, radiation fluence limits of 1×10^{19} neutrons/cm^2 neutron radiation and 1×10^8 Gy (1×10^{10} rad) gamma dose are considered conservative radiation exposure levels beyond which concrete material properties may begin to degrade markedly. NUREG-2191 recommends further evaluation of the need for a plant-specific program to manage aging effects of irradiation if the estimated (calculated) fluence levels or irradiation dose received by any portion of the concrete from neutron (fluence cutoff energy $E > 0.1$ MeV) or gamma radiation exceeds the respective threshold level during the subsequent period of extended operation or if plant-specific OE of concrete irradiation degradation exists that may impact intended functions. Based on the projected neutron fluence and gamma dose on the Turkey Point primary shield wall, the thresholds above are expected to be exceeded. As a result, an evaluation was performed on the irradiation effects on the primary shield wall at Turkey Point and included in Section 3.5.2.2.2.6 of the Turkey Point SLRA.

In accordance with Reference 1, the NRC conducted an audit on SLRA Section 3.5.2.2.2.6, Reduction of Strength and Mechanical Properties of Concrete Due to Irradiation. The audit took place from July 17 through 19, 2018 and from September 18 through 20, 2018. The staff performed extensive reviews of the irradiation analysis and supporting technical documentation. As a result of these reviews, a revision to SLRA Section 3.5.2.2.2.6 has been prepared and is included below. This section supersedes the current Section 3.5.2.2.2.6 in SLRA, Revision 1 in its entirety.

Discussion Reference

1. Turkey Point Nuclear Generating Units 3 and 4 - Plan for the Irradiated Concrete Technical Issue Regulatory Audit Regarding the Subsequent License Renewal Application Review (EPID NO. L-2018-RNW-0002) ADAMS Accession No. ML18173A087

Revised SLRA Section 3.5.2.2.2.6

Calculations performed to determine the projected peak neutron fluence and gamma dose within the PTN Unit 3 and Unit 4 reactor cavity for 80 years of plant operation have shown they are above the radiation exposure thresholds of 1×10^{19} n/cm^2 and 1×10^{10} rad, respectively, noted above for degradation of concrete (reduction of strength or mechanical properties) at the end of the SPEO. The evaluation and supporting calculation summaries of the irradiation effects on the primary shield wall, including embedded reinforcements, and reactor vessel support structures are described below and indicate the primary shield wall and RV supports will be able to perform their component intended functions for the SPEO.

Neutron fluence and gamma dose incident on the primary shield wall at the end of plant life were determined as follows. Westinghouse performed PTN plant specific analyses in 2009 to determine neutron and gamma fluxes incident on the primary shield wall before and after the extended power uprate (EPU) which was implemented in 2012. The flux values incident on the primary shield wall were calculated for energy groups corresponding to the energy groups provided in the original PTN design basis. The predicted flux values on the primary shield wall are derived from the reactor vessel fluence evaluation performed for the EPU. The EPU fluence evaluation satisfied the requirements set forth in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence". The methodology used for the EPU fluence evaluation was approved by the U.S. Nuclear Regulatory Commission (NRC) and is described in detail in WCAP-14040-A (Ref. 1) and WCAP-16083-NP-A (Ref. 2). The primary shield wall fluxes were calculated by taking the two-dimensional 2D planar flux solution for the radial and azimuthal (r Θ) geometry of the reactor vessel fluence evaluation with an axial factor, based on the peak axial relative power for the cycle, of 1.288 was applied. The cited flux value represents an average across the 0 to 45° azimuthal span of the 2D planar model. Note that deriving neutron fluence on the primary shield wall using the fluences calculated for reactor vessel embrittlement presented in SLRA Section 4.2 would result in significantly higher values than those calculated by Westinghouse for the primary shield wall. This is due to a number of factors including the application of a 20% bias for fuel assemblies on the periphery of the core, r Θ averaging, attenuation through the reactor vessel wall, and air attenuation in the reactor cavity.

Neutron fluences and gamma doses were then determined from the fluxes described above based on plant operation pre-EPU through 2012 and 2013 (40 years), and post-EPU after 2012 and 2013 (40 years) for Unit 3 and 4 respectively. For the pre-EPU period, the better plant performance of Unit 4 of 77.5% was utilized. For the post-EPU period, a best achievable capacity factor of 95% was assumed. Based on these calculations the neutron fluence and gamma dose incident on the primary shield wall are 3.57×10^{19} neutrons/cm^2 and 1.9×10^{10} rads, respectively at the end of the SPEO.

The main constituents of concrete are cement, large aggregate (gravel), small aggregate (sand), and water. Concrete consists of aggregates bound together by a cementitious material. Different types of cement could have been used in concrete for reactor radiation shielding (typically referenced as reactor cavity concrete, reactor shield concrete, biological shield concrete), including Portland cement, high-alumina cement, specialty cements, and others. Concrete from plants with US geographical diversity was evaluated based on information contained in NUREG/CR-3474. This reference presented data on concrete from currently operating plants as well as plants that have been decommissioned or had started construction but were not completed. They are considered to be representative of the currently operating plants in the United States. This information was supplemented with a review of concrete information from additional reactor sites, together with data from ORNL, the Crystal River plant, and the Trojan plant (Ref. 3). Collectively the reviewed data presented in this reference provides good geographical representation to supplement information of NUREG/CR-3474.

For US plants based on the characterization presented in Reference 3, Portland cements were used, with the majority being Type II. Aggregates can be categorized as containing high silica for certain plants to limestone (PTN), with a range of elements represented. Compression strength ranged from 3,000 psi (PTN) to 5,500 psi. Per Section 5.1.6.2 of the UFSAR, the concrete ingredients for the PTN primary shield wall are as follows:

Ingredients	Applicable Specification
Cement	ASTM C-150-64 Florida Type II
Air Entraining Agent	ASTM C-260-63T (Neutralized Vinsol Resin Airecon)
Water Reducing Agent	ASTM C-494-62T Type D (Retardwell, Union Carbide)
Aggregate	ASTM C-33-64 (Fine and Coarse Aggregate, Miami Oolite (limestone))

Note that water-to-cement ratios, including for PTN, were not generally available for internal structural concrete. However, the specified cement for PTN is a standard Portland cement. Type II is a general use specification with moderate sulfate resistance. The cement specified to ASTM C-150 contains Portland cement clinker, water or calcium sulfate or both and limestone not to exceed more than 5 percent by mass. The Miami Oolite, now referred to as Miami Limestone, is composed mainly of ooids (small spheroidal sedimentary grains) with some quartz sand and small mollusk

fossils. Because of the use of the air entrainment agent, air-entrained concrete usually contains up to 10 percent less water than non-air-entrained concrete of equal slump. The water reducing agent is typically used for water reduction and retardation and is generally used in small doses, so the water added to the mixture in the form of the admixture itself can be ignored. Type D water-reducing and accelerating admixtures ordinarily reduce mixing-water requirements in the 5 to 8 percent range. From ACI 211.1, an estimate for water-to-cement ratio (w/c ratio) for air-entrained concrete with 28 day compressive strength of 3,000 psi can be made. This would result in a w/c ratio of 0.59. With consideration of the Type D water reducing agent, the resulting w/c ratio would be estimated as 0.54 to 0.56. The expected PTN w/c ratio is relatively high in relation to higher strength concretes. Additionally, the use of the steel liner at the inside diameter of the primary shield wall will assist in reducing evaporative dehydration.

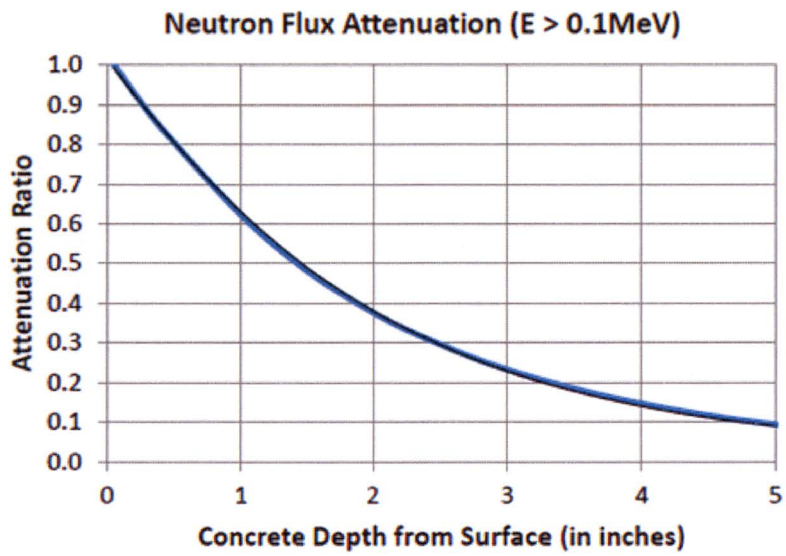
Concrete design codes utilize the specified compressive strength. Design specifications state this value at a defined age, often 28-days from time of mixing. This value is verified by laboratory testing of mix samples. The concrete strength must meet specified values, allowing a finite number of failures per number of tests. Concrete strength generally increases with age. This increase in strength is commonly rapid in the early years and gradually decreases with time. Strength can often be 20% to 30% higher than the 28-day strength in an aged concrete structure. This increased strength, while typically not credited for design assessment, is a condition in an aged structure that will provide additional compensation for reduced concrete properties as a result of long-term irradiation or gamma heating. For the PTN primary shield wall concrete, strength is specified as 3000 psi at 28 days, and 7500 psi at 90 days. These values were confirmed as part of the original concrete test program. However, 3000 psi is used in the analysis (Section 5.1.6.2 of the UFSAR).

Neutron Attenuation

The attenuation through the concrete is important for the evaluation of the structural capacity of the primary shield wall. In order to allow for parametric study of radiation in concrete for attenuation, thermal heating, and the effect of rebar, EPRI prepared models for 2-loop and 3-loop pressurized water reactors that focused on radiation in the concrete. These were prepared in collaboration with Professor Benoit Forget from MIT. These models are described in EPRI report 3002002676 (Ref. 3). The MCNP5 code (Ref. 4) was used and the concrete was typical of concrete constructed with Portland cement. The concrete properties were taken from PNNL-15870 entitled "Compendium of Material Composition Data for Radiation transport Modeling" (Ref. 5). Concrete for two reference types in PNNL-15870 were considered, Concrete, Ordinary (NIST) and Concrete, Portland. The bounding attenuation of these material types was considered. Based on the use of Portland cement at PTN, the EPRI attenuation was used in the PTN calculations.

The results of these calculations are shown in the figure below (Ref. 7). The MCNP-5 analyses of generic 2 and 3 loop biological shields shown in EPRI Report 30020011710

indicated that the attenuation was slightly more rapid in the 3-loop model, thus the 2-loop model was more conservative. The more conservative 2-loop results are shown below (Ref. 7) and were used for the PTN evaluation. Note that the flux is reduced by one order of magnitude in 5 inches.



Flux attenuation in Portland concrete in first 5 inches of concrete (E > 0.1 MeV)

The curve above can be fit with a polynomial trend line for use in calculations. The equation for the Attenuation Ratio AR (Ref. 3) is (d is the depth (inches) into the concrete from the surface):

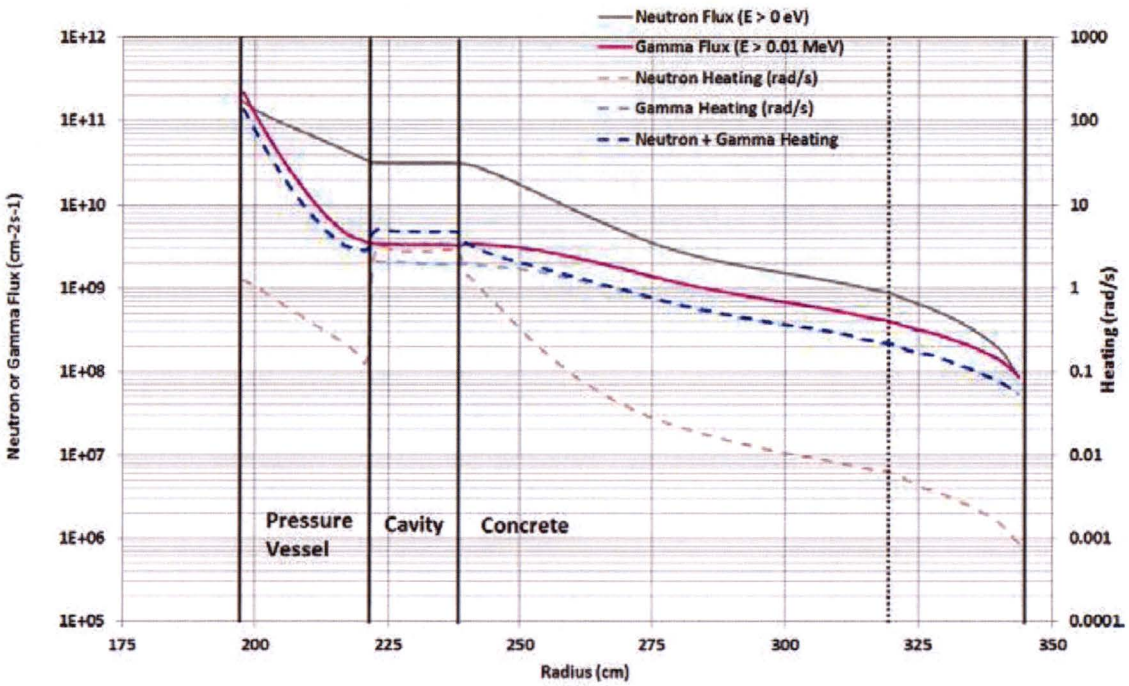
$$AR = -9.5577E\text{-}06d^5 + 4.8793E\text{-}04d^4 - 9.6891E\text{-}03d^3 + 9.4840E\text{-}02d^2 - 0.47235d + 1.0165$$

For PTN SLR, the AR to reach the damage threshold would be = 1/3.57 = 0.28

As noted above, the expected PTN w/c ratio is relatively high in relation to higher strength concretes. Additionally, the use of the steel liner at the inside diameter of the primary shield wall will assist in reducing evaporative dehydration. As stated by Maruyama (Ref. 8), water loss should relate to the reduction of both heat capacity and shielding performance. Water, which is the main source of the hydrogen atom in the concrete, is considered to have a large impact on the shielding performance of concrete. As such, the concrete attenuation model used for PTN based on the EPRI model above is considered reasonable and appropriate. Using neutron attenuation data and the 80 year neutron fluence information discussed above, the PTN neutron fluence falls below the NUREG-2191 and NUREG-2192 concrete irradiation damage threshold for neutrons (1.0×10^{19} n/cm^2) at a depth of 2.6" into the primary shield wall.

Gamma Attenuation and Heating

The same report addressing neutron fluence on the shield wall concrete also addresses gamma dose, with additional work contained in Reference 9. Using gamma attenuation data from Figure 1 in Reference 9 for gamma energies > 0.01 MeV, data regarding the gamma dose in the concrete shield wall is provided as a function of distance as shown below.



Neutron and gamma fluxes and dose through the shield wall

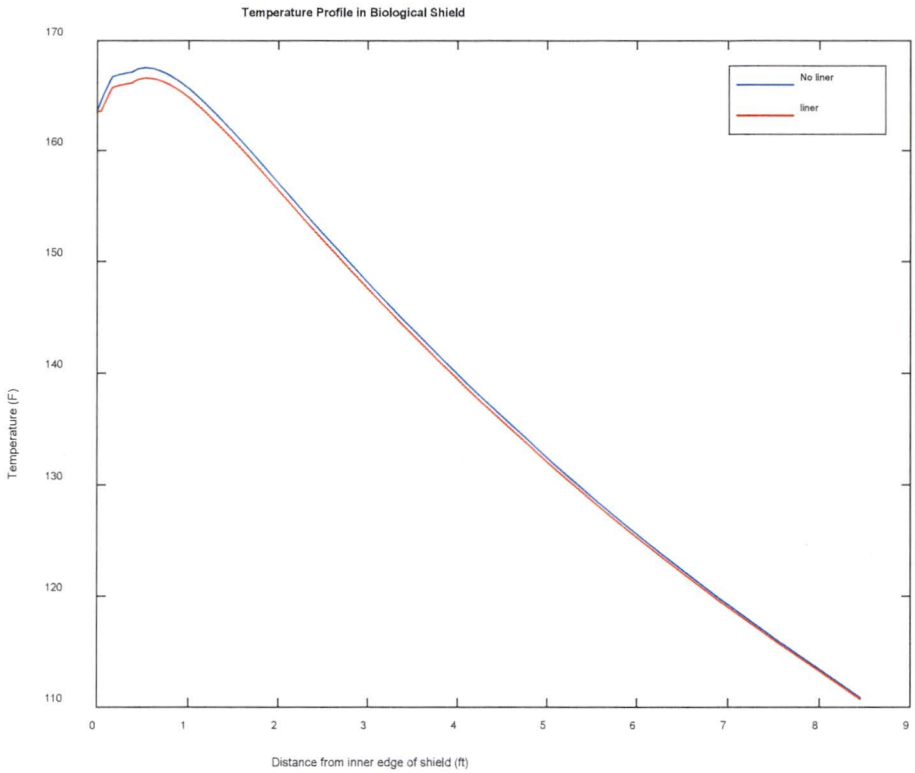
The data presented above for the gamma dose can be utilized to determine the relative values of total gamma dose as a function of depth into the primary shield wall at PTN. The use of this data is reasonable to determine the change in dose as a function of depth in the concrete. The data presented in the table below includes the data extracted from the **Error! Reference source not found.**figure and the equivalent data for PTN, which is developed using the ratio of the attenuated dose rate to reactor cavity dose rate from above. Provided in the table are the 80 year attenuated doses based on the PTN integrated dose of 1.9×10^{10} rads incident on the inner surface of the primary shield wall. Additionally, a row is provided with linear interpolation of the depth at which the integrated gamma heating dose is equal to 1×10^{10} rads.

Gamma dose as a function of distance through the shield wall

Radius (cm)	**Depth (in)**	Error! Reference source not found. **Dose (rad/sec)**	**Ratio**	**Realistic PTN Integrated Dose (rad)**
238	0	2	1	1.90E+10
250	4.7	1.9	.95	1.79E+10
264	**10.1**	**n/a**	**.53**	**1.00E+10**
267	11.4	1	.5	9.40E+9
275	14.6	0.75	.375	-
281	16.9	0.6	.3	-
287	19.3	0.5	.25	-
294	22.0	0.4	.2	-

Based on the 80 year gamma dose information presented above, the gamma dose falls below the NUREG-2191 and NUREG-2192 concrete irradiation damage threshold for gamma radiation (1.0×10^{10} Rads) at a depth of 10.1" into the primary shield wall.

Elevated temperatures inside and outside the primary shield wall have the potential to degrade concrete structures by accelerating the depletion, or vaporization, of the moisture content in the concrete. Elevated concrete temperatures can result both from the environmental temperatures outside the concrete and from the thermal effects of radiation. At PTN, environmental temperatures are controlled by containment ventilation cooling systems. Reference 7 provides an analysis with what are considered limiting conditions for thermal conductivity, radiation levels, rebar location, air gap (between the reactor vessel and the primary shield wall inner surface) temperature, air gap flow, and outside wall temperature. For a 150°F air temperature in the air gap, the calculated maximum temperature in the concrete is 168°F at a depth of approximately 6 in. from the inside surface of the concrete. The conclusion is that the temperature increase is nominal and is expected to be below the American Concrete Institute (ACI) code requirements. (See figure below from Ref. 7). Note that per Reference 7, the presence of a liner and variation in rebar depth has minimal effect on concrete temperatures.



Gamma Heating Effects in Primary Shield Wall

For PTN, a calculation was performed as part of the EPU which established a design air gap temperature of 132.4°F. Thus, the resulting temperature of gamma heating for PTN, whose primary shield wall includes a ¼" steel liner, would be approximately 150°F at a depth of 6 in. from the inside surface of the concrete. Accordingly, gamma heating is not considered an issue for the PTN primary shield wall concrete. The above is provided for information only. Gamma heating was considered in the original CLB calculation for PTN and is bounding since the original design basis gamma flux values used in the calculation are greater than post-EPU values.

Irradiation Effects on the Primary Shield Wall

With regard to reduction in concrete strength, the reference that has been used by many as the definition of acceptable exposure is a paper by Hilsdorf. In 1978, H. K. Hilsdorf et al. (Ref. 11) published a paper which compiled and summarized previously published experimental data on the effects of nuclear radiation on the mechanical properties of concrete. The data used by Hilsdorf combined tests with widely varying temperatures and energy levels. Based on these plots, Hilsdorf concluded that neutron radiation with a fluence of greater than 1.0×10^{19} n/cm^2 may have a detrimental effect on concrete strength and modulus of elasticity (Ref. 11).

Since 2014, significant work has been performed and published on the effects of radiation in concrete. A large database of neutron data was collected and presented by Field, et. al. (Ref. 12). Figure 3 of that reference reproduced below is a plot of concrete strength versus neutron fluence. This presentation of neutron fluence data indicates that the compressive strength appears to begin to decrease at a fluence of approximately 1 x 10^{19} n/cm^2. In this plot, though, the energy spectrum and specimen temperature vary between experiments.

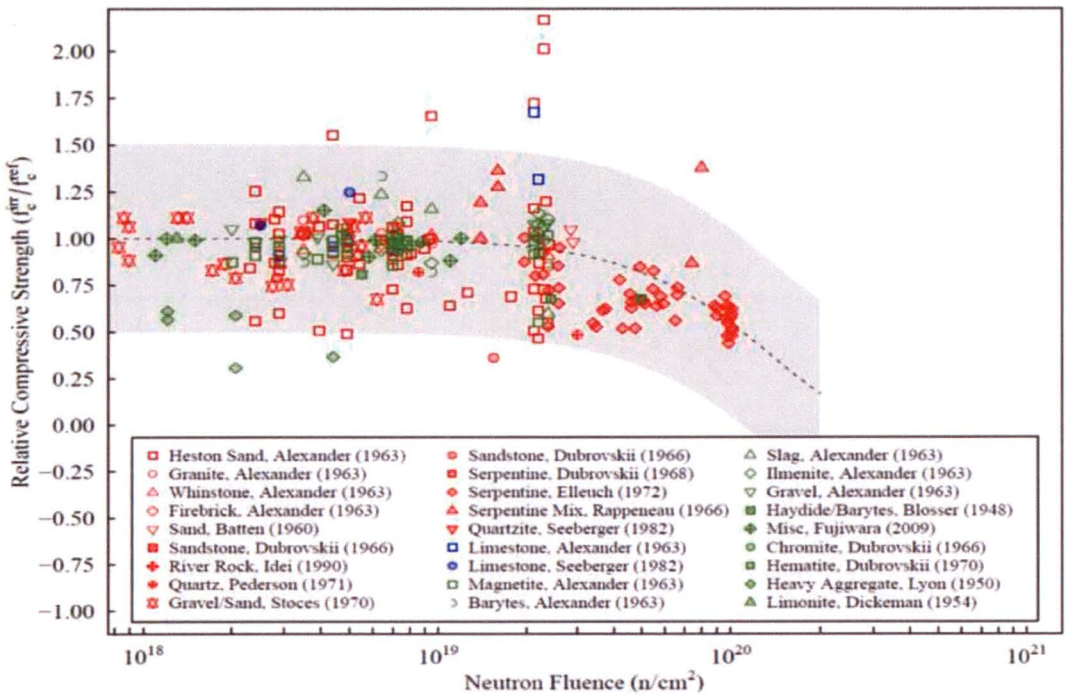
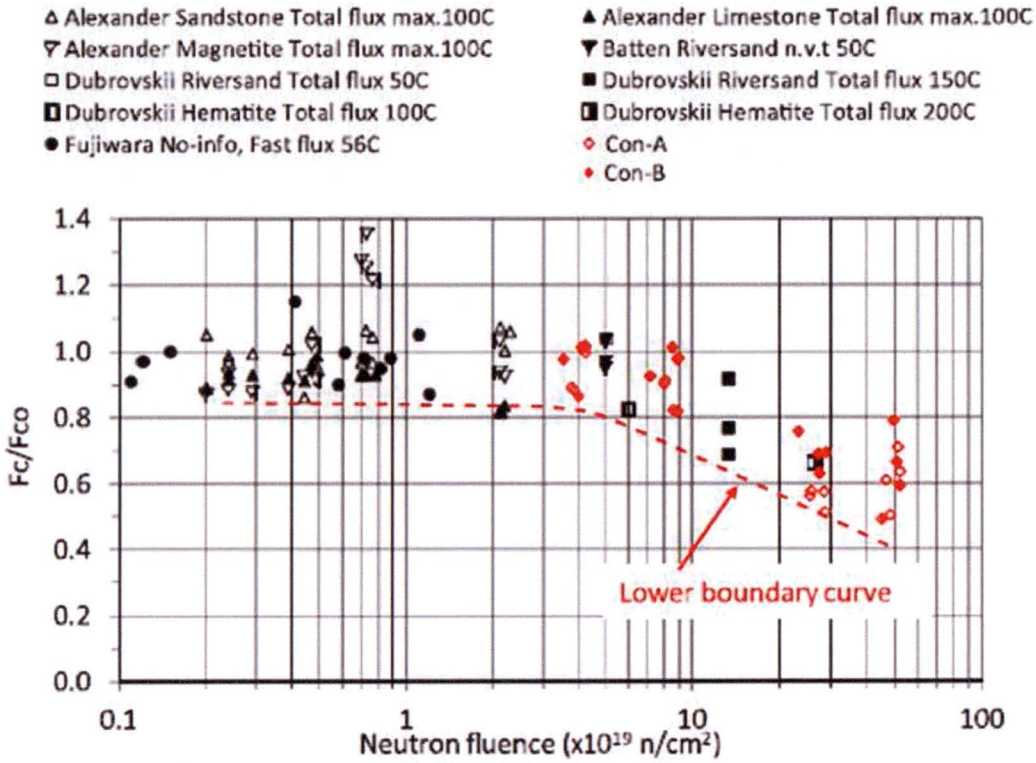


Figure 3: Relative compressive strength of concrete and mortar specimens versus neutron fluence in the range of 1x10^{18} n/cm^2 to 1x10^{21} n/cm^2 from Figure 2. The neutron spectrum and specimen temperature vary between experiments. Siliceous concrete is depicted with red symbols, calcareous with blue, and miscellaneous concretes with green. Filled symbols indicate experiments conducted above 100 °C, open symbols indicate experiments conduced below 100 °C. Mix design can be determined by cross referencing with Table 3. A decrease in compressive strength above 2x10^{19} n/cm^2 is suggested.

Neutron fluence energy levels greater than 0.1 MeV have been shown to cause more than 95% of all atom displacements in aggregates (Ref. 13). For this reason, neutron fluence with energy greater than 0.1 MeV is the most relevant for the assessment of concrete capacity. The temperatures of the concrete biological shields for light water reactors (LWRs) are all expected to be less than 200°F (93°C). Test results with much higher temperatures are not considered as representative.

Maruyama, et. al. (Ref. 14) presented consistent data with E > 0.1 MeV and with test temperatures less than 212°F (100°C) along with new test results. The concrete for these tests was considered as typical concrete that would be utilized for commercial nuclear reactor operation. The data is summarized in the figure below. It shows that a

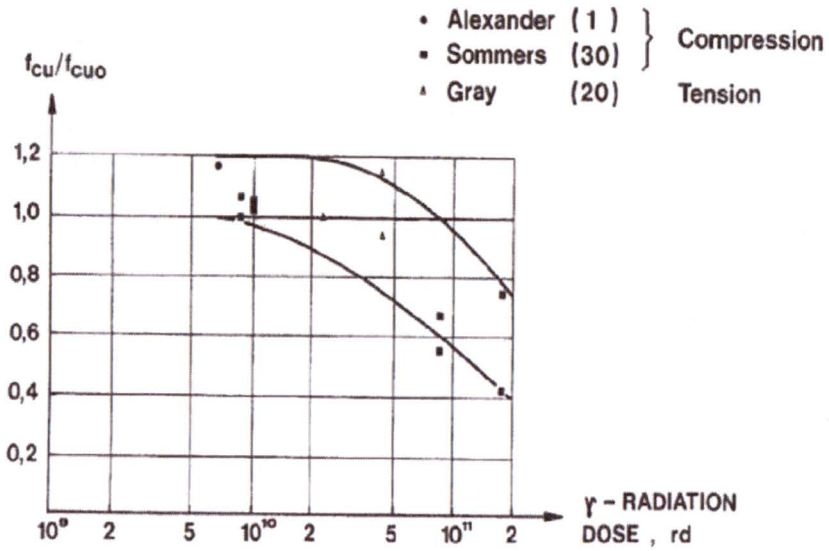
strength reduction begins at a neutron fluence of 1.0 x 10^{19} n/cm^2. This finding is consistent with the original recommendation of Hilsdorf but with additional testing to higher fluence levels and clarity in the test conditions.



A plot of concrete compressive strength ratio (F_c/F_{co}) versus total neutron Fluence (Fc tested strength after irradiation; Fco original strength, not irradiated)

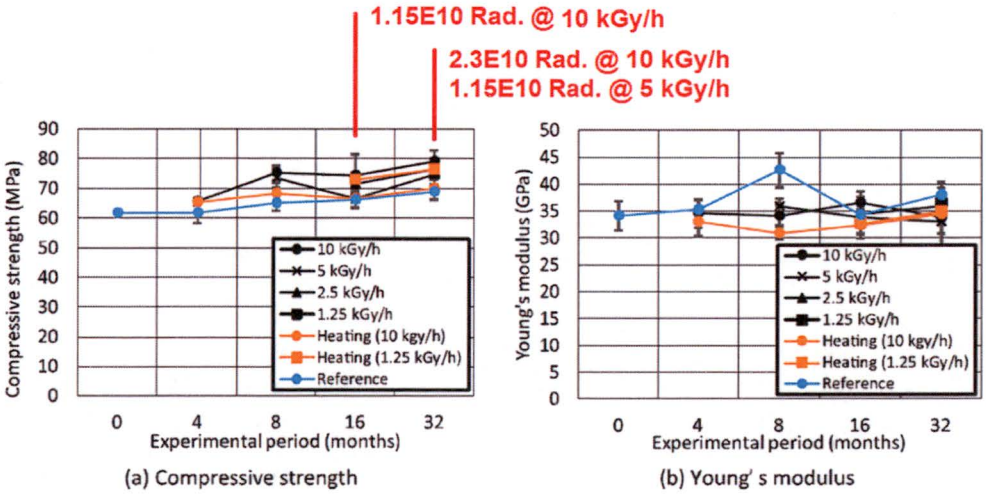
Applying the above figure, and utilizing the PTN calculated neutron fluence incident on the primary shield wall of 3.57 x 10^{19} and the calculated attenuation, the reduction in concrete strength as a result of neutron fluence would be 10% to a depth of 2.6 inches into the primary shield wall. Note that the embedded columns associated with the reactor vessel supports are outside of this region.

Hilsdorf (Ref. 11) also presented the change in compressive strength versus gamma dose for a limited amount of data. This is shown in the figure below. The mean interpolated value of the trend of this data would indicate a decrease in compressive strength for a dose between 2.0 x 10^{10} rads to 3.0 x 10^{10} rads. However, the data that was used to derive the plot is varied and not considered as fully representative of commercial reactor conditions.



Change in Concrete Tensile and Compressive Strength versus Gamma Dose

Maruyama also performed gamma tests and summarized data (Ref. 14). Representative data for compressive strength and elastic modulus is shown in the figures below. These figures are annotated with the gamma radiation at different exposure rates. A tendency for the compressive strength of the specimens to increase the longer that they were irradiated was noted.



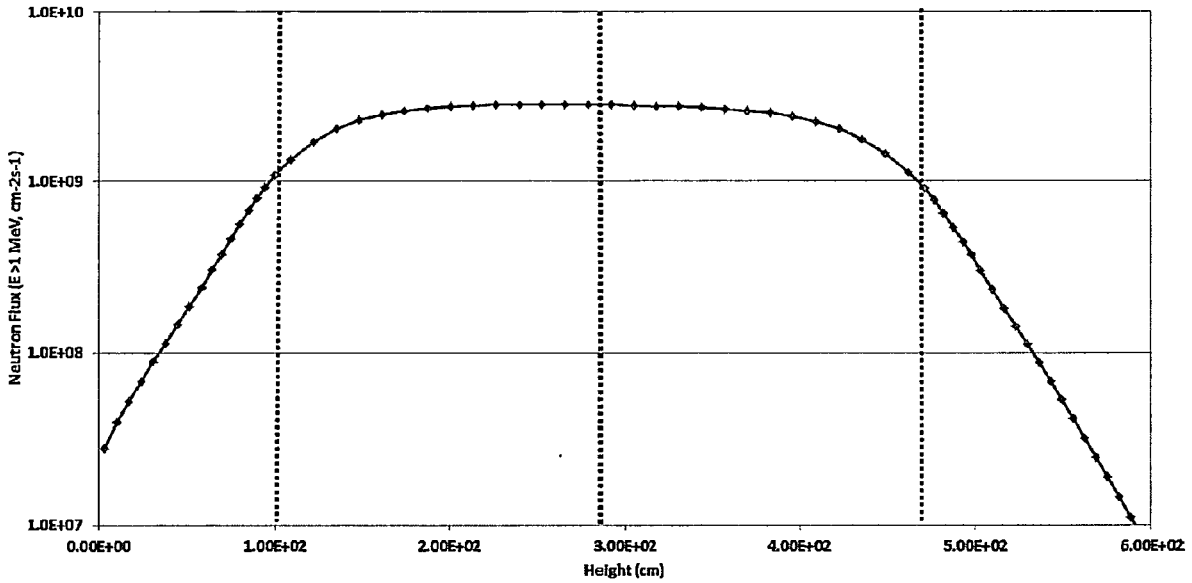
Change in Concrete Compressive Strength and Elastic Modulus versus Gamma Dose

The Maruyama paper suggested that either the threshold reference value for gamma exposure be raised to a high level or abandoned entirely. With consideration of the prior Hillsdorf data and the available test data presented by Maruyama, the test data indicates gamma irradiation up to and beyond a threshold of 2.3×10^{10} Rads has no effect on material properties.

Based on the above discussion, and considering the 80 year gamma dose incident on the primary shield wall at PTN is 1.9×10^{10} Rads, there will be no degradation of the primary shield concrete at PTN due gamma radiation. However, FPL will continue to follow EPRI and industry efforts to better define the effects of gamma radiation on concrete, and will update this evaluation and implement a plant specific AMP if necessary as noted in Commitment Number 53 in Table 17-3.

From Reference 7, the structural capacity of the primary shield wall will be potentially affected by the depth of concrete where the neutron fluence exceeds the damage threshold of 1.0×10^{19} n/cm^2 (E > 0.1 MeV). Where the neutron fluence exceeds 1.0×10^{19} n/cm^2 (E > 0.1MeV), there could also be an expansion of the concrete called radiation-induced volumetric expansion (RIVE) with a corresponding reduction in concrete properties. The extent of these changes will be limited to the inner inches of concrete. The effect of RIVE is addressed below.

As presented above, the primary shield wall concrete retains a substantial portion of its compressive strength for the radiation values expected to be experienced through the SPEO, and is shown to retain 90% or more of its compressive strength for neutron fluence and gamma dose values well above the values even on the inner surface of the concrete shield wall. Note that the radiation evaluations discussed above are based on the neutron flux and gamma doses that occur along the axial centerline of the active fuel region. As such, these are expected to be peak values. Lower values are expected along the wall at points further away from the axial centerline of the active fuel region, as shown in the figure below (Ref. 6). Thus, no concrete will be exposed to gamma radiation or neutron fluence values above the NUREG-2191 and NUREG-2192 concrete degradation thresholds of 1×10^{10} rad and 1×10^{19} n/cm^2, respectively, above and below the active fuel region heights along the primary shield wall.



SLR Primary Shield Wall Structural Evaluation

The primary shield wall at PTN Unit 3 and Unit 4 is 7'-0" (84") thick, as described in UFSAR Sections 5.1.8.1 and 5.1.8.3, and is part of the RV support structure, as described in UFSAR Section 5.1.9.3. The RV supports consist of six (6) individual supports, one of which is placed under each of the three hot leg and three cold leg reactor coolant system pipe nozzles. A majority of each RV support is embedded in the primary shield wall. The primary shield wall also provides structural support for a portion of the containment internal structure that includes the slab at El. 58'-0", walls above 32'-0", radial walls above 32'- 0" and the slab at 30'-6". The primary shield wall is 21 feet in height, surrounds the RV (that is approximately 14.5 feet in diameter), and extends from just below the matting flange with the RV head at the refueling canal floor down to where it connects with the containment building foundation. The wall is a donut shaped reinforced concrete (3000 psi) structure that has an inside diameter of 19 feet and an outside octagonal sided wall with a minimum outside diameter of 33 feet.

As noted above, concrete depths required to reduce the neutron fluence levels to below the NUREG-2191 and NUREG-2192 threshold value is 2.6 inches. Based on the discussion above, the worst case reduction in the strength of the concrete could occur along the circumference of the inner face of the primary shield wall up to 2.6 inches in depth at the active fuel core midline elevation, with the effects decreasing with concrete depth and vertical distance from the fuel centerline to the ends of the active fuel core. No radiation levels in excess of the threshold value is expected above and below the active fuel region elevation along the interior face of the primary shield wall. Reinforcing steel (rebar) across from the area encompassing the effective height of active fuel is located a distance of 19 inches from the inner edge of the wall, and is not affected by increased neutron fluence or gamma dose.

A review of the impact of the possible irradiation effects on the concrete and the ability of the concrete to continue to perform its component intended functions through the SPEO was performed. The conclusions are as follows:

Radiation effects such as neutron fluence and Radiation-Induced-Volumetric-Expansion (RIVE) effects were determined. The existing primary shield wall was evaluated for the CLB loading with the radiation effects by using the same original design analysis approach as the recently updated CLB calculation. Due to the RIVE effect, the excessive compressive stress was calculated and the inner side of the concrete (up to 3.14 inches) is considered as yielded (cracked). Accordingly, the design stresses were re-calculated for the reduced concrete section due to the crack under the CLB loading and considering the reduced strengths and modulus of the irradiated concrete. Comparing with the un-irradiated concrete (where the maximum interaction ratio (IR) is calculated as 0.74), the maximum IR for the irradiated concrete (including the cracking discussed above) was calculated as 0.82, which has increased but is still less than 1.0. Therefore, the existing primary shield wall including the radiation effects is qualified for the CLB loading based on the evaluation results.

An additional evaluation was performed demonstrating that the primary shield wall can allow a cracking depth of up to 7 inches and still be able to perform its component intended function. Thus much higher radiation exposures can be accommodated by the primary shield wall. Please note that leak-before-break (LBB) of reactor coolant system auxiliary lines has been submitted as part of the SLRA (Section 4.7.4 and Enclosure 4, Attachment 12). Upon NRC approval, the loads on the reactor vessel supports and primary shield wall concrete will be significantly reduced. For the primary shield wall, implementation of auxiliary line LBB will result in the IR being reduced to 0.41 (tension). The governing load case would be Normal (IR = 0.41 for tension) and Emergency (IR = 0.32 for compression). Considering the IR increasing ratios (i.e., 10.8% for tension and 10.2% for the maximum compression), the maximum IRs are approximated as 0.45 (= 0.41 x 1.108) for tension and 0.35 (= 0.32 x 1.102) for the maximum compression.

In addition, there has been no site-specific OE relative to radiation effects on concrete intended functions. Also, the Turkey Point reactor vessels are nozzle supported as described above, and there is no support pedestal. The RV supports are structural steel with most of the support embedded in the primary shield wall. The RV supports are not exposed to the same levels of neutron and gamma radiation as the horizontal exposed supports are at the RV nozzle level, above the effective height of the active fuel, and the embedded portions are shielded by the concrete wall.

SLR Reactor Vessel Support Evaluation

The aging effect of irradiation of structural steel is limited to reduction in fracture toughness due to embrittlement from exposure to neutron fluence. The NRC previously identified radiation embrittlement of the RV supports as a generic safety issue in GSI-15. The NRC resolved the issue, as documented in NUREG-1509 on the basis of a risk-informed evaluation, without imposing new requirements on licensees. Additionally,

NUREG-2191 and NUREG-2192 do not specifically identify the reactor vessel support steel for aging effect reduction in fracture toughness as a result of irradiation embrittlement.

However, during the SLR NRC concrete irradiation audit conducted from July 17-20, 2018 at Turkey Point, the NRC identified reduction in fracture toughness due to irradiation embrittlement of the reactor vessel (RV) support steel as a potential aging effect for consideration. The NRC noted NUREG-1509 as a resource for addressing the issue for PTN SLR. Accordingly, an evaluation of the PTN RV supports for reduction in fracture toughness for SLR is provided below.

The RV support structure for each PTN Unit consists of six (6) individual supports, one of which is placed under each of the three hot leg and three cold leg Reactor Coolant System pipe nozzles at elevation (EL) 25'-7 ½". A majority of each RV support is embedded in the primary shield wall. The primary shield wall is 21 feet in height, surrounds the RV (that is approximately 14.5 feet in diameter) and has an inside diameter of 19 feet. The reactor vessel active fuel region is 12 feet in height, is located between the upper and lower core plates and extends approximately from EL 8'-10" to EL 20'-10'. The approximate active fuel core midline is at elevation 14'-10". The RV support structure includes vertical columns, cantilever beams, horizontal (cross) beams and roller assembly. The columns and portion of the cantilever beams are located inside the primary shield wall, with the centerline of the cantilever beams at a height approximately equal to the top of the active fuel, and the inboard edge of the innermost column ~ 5 inches from the inside surface of the primary shield wall. Note that the bottom horizontal beams do not perform a structural function and are not included in the RV support analysis. As such, and because the slightly higher fluence incident on the lower beams than on the cantilever beams would not embrittle the lower beams to the point of gross failure, the lower horizontal beams are not considered in the evaluation.

As noted in the figure above, the variation of radiation fields with elevation is small in the region close to the core mid-plane. At an elevation corresponding to the top of the fuel, neutron fluence is approximately 40% of the fluence incident on the primary shield wall at the core mid-plane. At two feet above the top of the fuel, the neutron fluence ($< 1 \times 10^{17}$ n/cm^2) is well below the fluence incident on the shield wall. Based on these fluence values, the upper portions of the RV supports (roller assembly, tangential brackets, side shims, machine screws that attach to the RV nozzle pads, etc.) are greater than two feet above the top of the fuel. Thus, they are not exposed to neutron fluences that would cause reduction in fracture toughness due to neutron embrittlement and no further evaluation is required. The cantilever beams and top cross beams, which are located at an elevation corresponding to the top of the fuel, will see an end-of-life fluence of (3.57×10^{19} n/cm^2 x 0.40) ~1.43×10^{19} n/cm^2. The end-of-life fluence on the RV support columns embedded in the concrete would be less than 10% of the fluence incident on the primary shield wall at the core mid-plane height.

The remaining RV support components to be evaluated for reduction in fracture toughness due to irradiation embrittlement consist of the embedded columns, the

cantilever beams, the cross beams, and associated bolting. The columns and beams are 10WF89 and 14WF342 composed of ASTM A588 Type B steel. The bolting is 2¼" – 4.5 UNC-2A x 9" long with 5" of thread composed of ASTM A354, Grade BC alloy steel. Certified mill test reports (CMTRs) for the beams and bolting were reviewed to confirm the material composition and properties.

Based on review of NUREG-1509 and the design documentation of the PTN RV support bolting, no further evaluation for reduction in fracture toughness for the bolting is required.

Per NUREG-1509, the reduction in fracture toughness assessment of RV support steel can be based on a transition temperature analysis, wherein a demonstration is made that there is adequate margin between the minimum operating temperature and the ductile-to-brittle fracture mode transition temperature (commonly known as the NDT temperature) for end-of-life (EOL) conditions. The transition temperature approach is based on the proposition that catastrophic failure by brittle fracture can be avoided by maintaining the normal operating RV support service temperature above the NDT temperature of the steel. When using the transition temperature to evaluate the support integrity, the NDT temperature at EOL should include the irradiation induced shift. Uncertainties related to NDT determinations should also include a margin of safety between the lowest service temperature (LST) and the NDT temperature. Demonstration that the RV supports are in conformance with this relationship is necessary and sufficient to preclude their failure by brittle fracture.

The NDT temperature of steel is one of the essential parameters in brittle fracture analysis. Following the ASME Code, the NDT temperature may be defined as the highest temperature for fracture of a standard drop-weight specimen when tested to ASTM Standard Test Method E 208-87a. The NDT also can be based on the temperature at which Charpy V-notch specimens absorb a specified amount of energy. To determine the EOL NDT temperature, the initial, material-dependent NDT and the anticipated shift (increase) as a function of radiation exposure needs to be known.

For the columns and beams, from Table 4-1 of NUREG-1509, the normalized initial NDT (mean plus 1.3 standard deviation) for the ASTM A588 Type B material is -27°F.

NUREG-1509 provides a method for approximating the NDT shift by determining exposure in terms of displacements per atom (dpa), and then using Figure 3-1 of that reference to establish the ΔNDT. To calculate the approximate ΔNDT for the beam material, the PTN specific neutron flux must first be converted to a dpa rate and then total dpa for 80 years calculated. For PTN, the total dpa through the end of the SPEO was calculated to be 4.89×10^{-3}.

Using the fitted curve from Figure 3-1 of NUREG-1509, the ΔRDT would be ~70 °C, or 126 °F. The fitted curve was utilized because the test data points associated with the dpa being evaluated (in the 1 to 5×10^{-3} range) are all below the fitted data curve. Additionally, the upper-bound curve in the region of interest included points that combine neutron and gamma exposure and are based on only 2 worst case data points.

The end-of-life NDT for the beam material would be ~99°F which is below the normal operating temperature (Modes 1 through 4) in the reactor cavity of ~120°F. This provides an NDT margin of 21°F at the end of the SPEO. Also note that the initial NDT is based on mean plus 1.3 standard deviation. This could provide a further margin of 23°F based on the actual material properties from the CMTRs.

Additional conservatisms associated with stress analysis of RV support steel are described below.

(1) Per the CMTRs for the beams, the yield strength of the beam is reported as 58.16 ksi, which is about 20% larger than 48.75 ksi (used in the determination of the maximum IR of the RV support steel).

(2) The yield strength of 48.75 ksi was calculated based on the operating temperature of 150 °F, which is higher temperature than the calculated ambient temperature of 120 °F. Lower temperature results in more capacity for the RPV support steel.

(3) Based on the span depth ratio of the cantilever portion of the RPV support steel, the beam is considered as a deep beam where the shear is typically governing. Per the stress analysis, the shear capacity of the beam was calculated by considering only the web area. The beam is reinforced by using 1" thick stiffener plates. Thus, the shear stress flows not only in the web but also in the stiffeners and the top and bottom flanges of the beam. The effective area of the beam for the shear should be larger than the web-only area.

Also, upon implementation of the auxiliary line LBB analysis (excluding LOCA loads), and removing some of the conservatism used in the stress analysis, the maximum shear stress of the beam is approximated as 5.41 ksi, which is less than the NUREG-1509 threshold of 6 ksi.

Finally, visual inspections performed under the ASME Section XI, Subsection IWF aging management program (AMP), described in SLRA Sections 17.2.2.32 and B.2.3.32, have not identified dimensional shifts or changes in the RV support steel. Future inspections will provide early indication of issues associated with the supports.

Accordingly, there is reasonable assurance that loss (or reduction) of fracture toughness due to irradiation embrittlement will not affect the ability of the RV structural steel to perform its component intended functions through the SPEO.

Conclusion

Based on the above, a plant-specific program to manage the effects of concrete irradiation is not expected to be necessary to ensure the components perform their intended function consistent with the CLB through the subsequent period of extended operation. However, FPL will continue to follow the on-going industry efforts that are clarifying the effects of irradiation of concrete and corresponding aging management recommendations as noted in Commitment Number 53 in Table 17-3, and will:

a) ensure their applicability to the PTN Unit 3 and Unit 4 primary shield wall and associated reactor vessel supports;
b) update design calculations, as appropriate; and
c) develop an informed plant-specific program, if needed.

With regard to concrete inside containment other than the primary shield wall, the total integrated gamma dose received by SSCs during 80 years of normal operation based on environmental qualification (EQ) documentation information is 5.02×10^7 rads. Neutron fluence from the reactor vessel/core will attenuate to well below the NUREG-2191 and NUREG-2192 fluence threshold for degradation in or before the 84 inch thickness of the primary shield wall and is, therefore, not a concern outside the reactor cavity, consistent with the existing CLB.

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