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Regulatory Improvements for Power Reactors Transitioning to Decommissioning

Comment On: NRC-2015-0070-0178

Regulatory Improvements for Power Reactors Transitioning to Decommissioning; Request for Comment on Draft Regulatory Basis

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General Comment

Paul Gunter, Director of the Reactor Oversight Project for Beyond Nuclear enters the attached comments in Proposed Rulemaking.

Paul Gunter, Director
Reactor Oversight Project
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Attachments

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BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

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In the Matter of)	
)	
Comment on Draft Regulatory Basis)	
)	Docket ID NRC-2015-0070
Regulatory Improvements for Power)	
Reactors Transitioning to Decommissioning)	June 13, 2017
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**Comments of Beyond Nuclear from Paul Gunter,
Director of the Reactor Oversight Project**

**The role of decommissioning in aging and fatigue management:
An “autopsy” should be performed at decommissioned nuclear power stations to
inform the analysis of residual safety margins at operating reactors under license
renewal and subsequent relicensing**

As posted in the Federal Register (Vol. 82, No. 49, March 15, 2017, pp. 13778-13781), the U.S. Nuclear Regulatory Commission (NRC) is requesting public comments on **Docket ID NRC-2015-0070** relating to the draft regulatory basis to support a rulemaking to amend NRC's regulations for the decommissioning of nuclear power reactors. The NRC's stated goals in amending these regulations is to 1) provide for an efficient decommissioning process; 2) reduce the need for exemptions from existing regulations; 3) address other decommissioning issues deemed relevant by the NRC staff; and, 4) support the principles of good regulation, including openness, clarity, and reliability.

As pertains to “an efficient decommissioning process”, the Federal Register Notice on the Proposed Rule identifies “the NRC staff has determined that additional stakeholder input is needed prior to finalizing recommendations” related to number of topics that

specifically includes “aging management and fatigue management.” The NRC asks “3. Are there any additional options that the NRC should consider during development of the proposed rule?”

Beyond Nuclear submits the following comments to <http://www.regulations.gov> in support of the requested stakeholder input on the issue of aging management and fatigue management.

Decommissioning has an important role to contribute to understanding aging/fatigue management for license extensions for operating reactors

Post de-fueled activities conducted during decommissioning to investigate and assess the effects of aging and fatigue on nuclear power have an increasingly vital role to play in protecting public safety as relates to the regulator-industry present effort to extend the operating licenses of nuclear power stations longer and longer.

Due to number of factors including the increasing high and uncertain cost-of-completion of nuclear power plant construction, the equally unpredictable time-to-completion of new construction and other financial, market and technical disincentives, “The Bridge to the Future” for nuclear power generation in the United States now relies upon the operating license extension of existing aging reactors after their original 40-year license expires. New reactor licensing and construction has not materialized as once anticipated by government and industry to replace or expand nuclear capacity in many countries including the United States. The NRC is now long engaged in a license extension application process to extend the original 40-year operating licenses to 60-years. Of the United States’ current total of 99 operational units, the NRC has granted 88 units a 20-

year license renewal although 4 of those units have since permanently closed with 2 additional units that plan early closure in 2019 (Oyster Creek and Pilgrim) far short of their granted extension period. An additional 8 units are in license renewal review although 4 of those units have announced that they will instead close (Diablo Canyon 1 & 2 before entering into the license renewal period and Indian Point 2 & 3 as early as 2020 & 2021 respectively with possibly two 2-year incremental extensions if agreed by the State of New York). The NRC is presently anticipating the future submittals from 5 more units (River Bend 1, Perry 1, Clinton 1 and Comanche Peak 1 & 2).

The NRC and industry have additionally launched a pilot program for a “Subsequent License Renewal” process that is said to be “on track” to file an application in 2018 for the Peach Bottom Nuclear Power Station (Units 2 & 3) and a 2019 application for the Surry Nuclear Power Station (Units 1 & 2) in request of additional extensions from 60 to 80-years. A 2016 Nuclear Energy Institute industry poll stated that an additional 20 units have expressed interest in the Subsequent License Renewal application process.

At the same time, more and different material degradation mechanisms are being identified as age-related degradation phenomenon as well as the discovery of weakening “anomalies” in the manufacturing process affecting nuclear safety margins that were not identified and/or captured at the origin of these structures and components’ construction and fabrication.

While the replacement of large heavy components to extend the operation life of nuclear power stations is an option for some safety-related reactor systems, structures and components (i.e. steam generators, reactor pressure vessel heads, pressurizers) it is not a universal. For some vital safety-related structures and components, replacement

is simply not structurally possible even if it were deemed economically affordable. Nor is non-destructive inspection of these large components and structures viable given the lack of access to examination. Destructive examination in operating reactors is often ruled out for these safety-related structures and components.

Beyond Nuclear contends that in these cases the decommissioning process can provide regulators, industry and the public with important, informed insights and analyses on the robustness of residual safety margins for inaccessible and irreplaceable structures for more reliable confidence in the safe operations of facilities that intend to extend operations. The results of destructive examinations and material testing from decommissioned facilities can not only meaningfully inform the license extension process and analysis but it can serve to validate and verify computer modelling and the testing of simulations.

The decommissioning rule therefore needs to incorporate at minimum a formal guidance process and arguably a set of regulatory requirements for the post-defueled destructive examination and material testing of specified priority safety-related systems, structures and components in order to analyze the effect of age-related and fatigue-related degradation as well as confirm the reliability of original manufacturing practices (i.e. forging). In so doing, regulator, industry and public can have a better understanding of the residual safety margins in those same components and structures in operating power reactors.

In fact, the NRC has already documented the need for the decommissioning process to be viewed as a “Strategic Approach for Obtaining Material and Component Aging

Information.”¹ The NRC Office of Research states that “To date, harvesting opportunities have been limited due to few decommissioning plants.”²

The NRC staff identifies that present decommissioning activity “provides a unique opportunity to plan harvesting to address the highest priority technical and regulatory issues.”³ The NRC further identifies, “As decommissioning plants announce their plans, there is a clear list of SSC (systems, structures, components) and their characteristics (metallurgy, temperature, fluence, etc.) that would be desired to address the data need.”⁴

Beyond Nuclear, however, contends that destructive examination and material testing during the post-defueled decommissioning process need to be more than just an option or opportunity. As the nation’s reactor fleet ages, public safety deserves it to be a formal requirement to reasonably inform the analysis of residual safety margins for those reactors granted 40 to 60-year extensions and more reliably inform the licensing review process for the 60 to 80-year subsequent license extensions now being contemplated.

Examples where aging/fatigue degradation processes can be better understood by conducting an “autopsy” through destructive examine and material testing

There are a host of age-related degradation mechanisms that can be analyzed in permanently closed, de-fueled reactors to gain insight into remaining safety margins at

¹ “Strategic Approach for Obtaining Material and Component Aging Information,” U.S. Nuclear Regulatory Commission, Nuclear Reactor Research, Industry/NRC Materials Programs Technical Information Exchange Meeting, Nuclear Regulatory Commission Headquarters June 2-4, 2015, http://www.beyondnuclear.org/storage/decommissioning/decom_nrc_harvesting_opportunities_2015%20.pdf

² Ibid, NRC, Slide 4

³ Ibid, NRC, Slide 4

⁴ Ibid, NRC, Slide 17

operating reactors and their license extensions. The degradation mechanisms attacking large steel components and structures (i.e. reactor pressure vessel, containment and containment liner) that weaken Defense-In-Depth at nuclear power stations include embrittlement (thermal aging and irradiation assisted), fatigue and corrosion. Large safety-related concrete structures including the containment, the irradiated fuel storage pool and the basemat foundations can also be weakened by alkali silica reaction (ASR) and irradiation.

In these cases, as nuclear power stations are decommissioned, an “autopsy” can be undertaken through destructive examination and material testing of samples harvested from affected structures and components.

An example list of non-replaceable systems, structures and components include the following.

The Reactor Pressure Vessel

The reactor pressure vessel cannot be replaced to extend reactor operations. While annealing of an existing pressure vessel has been suggested as a “last resort” here in the United States, the only demonstration project for the Palisades nuclear power station was cancelled due to revised fluence estimates and concern over public hearings if annealing were authorized.⁵

⁵ “Degradation of primary components of pressurized water cooled nuclear reactors: Current Issues and Future Challenges; Thermal annealing of reactor pressure vessels is a needed mitigation option,” IAEA, November 8, 2013, Slides 29 and 36, https://www.iaea.org/NuclearPower/Downloadable/Meetings/2013/2013-11-05-11-08-TM-NPE/38.Server_USA.pdf

The destructive examination and material testing can begin once the vessel is defueled, excavating or cutting samples directly from the reactor pressure vessel beltline to analyze for the loss of ductility and embrittlement of vessel base metal and weld material. However, both NRC and the industry have been reluctant to harvest and analyze material samples. At present, the Nuclear Energy Institute (NEI) argues that there is sufficient data from the very limited decommissioning analysis to suggest that there are no significant “obstacles” to subsequent license renewal.⁶ NEI identifies that it is informing its analyses for the large irreplaceable reactor pressure vessel for extending reactor operations from 60 to 80 years on sample analyses taken from just two reactors; Spain’s 37-year old Jose Cabrera nuclear power station and the 27-year old Zion nuclear power station in Illinois.⁷ Rather than supplement this limited sample set through simulations and modelling, older service irradiated samples should continue to be harvested for material testing to broaden in the in-field observed consequences.

The Yankee Rowe nuclear power station in Western Massachusetts is a classic example of the NRC and industry position that runs contrary to the agency claim that decommissioning opportunities have to date been limited by a lack of reactor sites to harvest samples for material testing. In fact, Yankee Rowe was the pilot applicant for the NRC and industry’s original 20-year license extension program. The NRC public record demonstrated unequivocal evidence that the Yankee Rowe reactor pressure vessel was embrittled and weakened by neutron radiation making it vulnerable to failure during extended operation. The large reactor pressure vessel has not redundancy or

⁶ “Research finds few obstacles to long-term operation of nuclear power plants,” NEI, July 23, 2015, <https://www.nei.org/News-Media/News/News-Archives/Research-Finds-Few-Obstacles-to-Long-Term-Operatio>

⁷ Ibid, NEI

backup feature. The NRC was aware of Yankee violations of two critical technical specifications but had acquiesced for more than a year. A subsequent petition filed by the Union of Concerned Scientists forced the disclosure of the significant uncertainty for the residual safety margins and severity of embrittlement of the irreplaceable reactor pressure vessel and the reactor was permanently shut down to begin the decommissioning process. In 1996, Citizen Awareness Network, Nuclear Information Resource Service and nine other citizens organizations around the United States challenged (10 CFR 2.206) the “possession only” license for Yankee Rowe and three other permanently closed reactors requesting that the license be modified for emergency enforcement actions requiring the owners to essentially conduct a collaborative autopsy on their reactor pressure vessels by excavating “boat samples” or cuttings of the pressure vessel wall base material and beltline welds.⁸

The eleven petitioning organizations requested emergency enforcement action for the possession-only licenses of the Yankee Nuclear Power Station (or Yankee Rowe), Rancho Seco Nuclear Generating Station, Trojan Nuclear Plant, and San Onofre Nuclear Generating Station Unit 1, licensed respectively to the Yankee Atomic Electric Company (YAEC), Sacramento Municipal Utility District, Portland General Electric Company, and Southern California Edison Company (Licensees). The Petitioners requested that the NRC require the collaborative effort by the licensees of the four nuclear power plants to document and research radiation embrittlement of reactor

⁸ Yankee Atomic Electric Co., Sacramento Municipal Utility District, Portland General Electric Co., and Southern California Edison Co.; Receipt of a Petition for, and Issuance of, a Director’s Decision under 10 CFR 2.206, The Federal Register, Vol. 61, No. 121, June 21, 1996, pp.31964-31966, <https://www.gpo.gov/fdsys/pkg/FR-1996-06-21/pdf/96-15838.pdf>

pressure vessels as an age-related deterioration phenomenon. Specifically, among several actions, the petitioners requested that the NRC require the owners of the four nuclear power plants to present substantial base metal and weld specimens from their respective RPVs to the NRC for analysis in order to study and materially archive the radiation embrittlement phenomenon. The petitioners identified that (1) the four permanently closed reactors constitute a valuable asset for evaluating RPV embrittlement,(2) “boat” or scoop samples from the RPV could be retrieved with minimal occupational radiation exposure, (3) data from boat samples could be used to verify the veracity of simulated embrittlement in research reactors, and (4) the boat samples could be subjected to annealing or reheating processes to analyze the results for restoring ductility of the material and for determining the durability of an annealing process. As documented in The Federal Register Notice of June 21, 1996, the NRC denied the petitioners request for confirmatory action by destructive examination and material testing for reactor pressure vessel embrittlement at all four of the permanently closed reactors. The best of our knowledge and affirmed by NEI documentation,⁹ none of the identified units’ reactor pressure vessels have ever been harvested for samples and materially tested, including the Trojan nuclear power station as inferred by the NRC Director’s Decision, ever happened.¹⁰

Additionally, the irreplaceable reactor pressure vessel and its components can also incur adverse “anomalies” incorporated during the manufacturing and fabrication like

⁹ Ibid., NEI, July 23, 2015

¹⁰ Ibid., Yankee Rowe, et al.

“carbon macrosegregation” from the forging process.¹¹ The lack of quality control and quality assurance during the original metal forging process introduces small but significant impurities of excess carbon into the forged component like the reactor pressure vessel. The portion with excess carbon from the forging process if not adequately cropped off and subsequently rolled into the final component makes that component vulnerable to accelerated aging, embrittlement and rapid tearing during operations.

Excavating pressure vessel samples harvested from affected components for material testing by optical emission spectrometry (OES) can identify material impurities from the manufacturing process to verify and validate quality control and quality assurance practices of the manufacturers that supplied those same components to reactors still in service under license extension.¹²

In this particularly example, harvesting base metal samples from the permanently closed Crystal River Unit 3 reactor’s pressure vessel and pressure vessel head in Florida for a material test for carbon macrosegregation would be valuable in verifying and validating the controversial AREVA Le Creusot Forge manufacturing process in France. Seventeen U.S. nuclear power stations remain in operation with at-risk large steel components provided by the controversial Le Creusot’s Forge process. “For its Crystal River 3 plant in Florida, Progress Energy Corp. (Raleigh, N.C.) contracted

¹¹ Petition for Emergency Enforcement Action per Chapter 10 Code of Federal Regulation Part 2.206 (10 CFR 2.206) at Listed U.S. Reactors with Forged Components and Parts Manufactured at France’s AREVA-Le Creusot Forge and Japan Casting and Forging Corporation, Beyond Nuclear et. al, January 24, 2017, http://www.beyondnuclear.org/storage/areva-le-creusot/2206_le-creusot_us_01242017_fnl.pdf

¹² Optical Emission Spectrometry (OES), Gamal Abdel Hamid, Education, August 6, 2016, <https://www.slideshare.net/GamalAbdulHamid/optical-emission-spectrometry-oes>

Framatom ANP (U.S. office: Lynchburg, VA) to manufacture a new vessel head. Framatom has also been awarded a follow-on contract to full installation services at Crystal River. The new vessel head and associated control rod drive mechanism forgings have been completed, and the next six steps in the manufacturing process are underway at Framatom's Charlon-St. Marcel manufacturing facility. The components are scheduled for delivery to Crystal River this summer, to be installed during the plant's next scheduled refueling outage."¹³ AREVA has its roots in Framatom along with which came the transfer of the one and the same Le Creusot Forge Charlon-St. Marcel industrial facility. The Crystal River 3 pressure vessel head is therefore a prime candidate for harvesting, carbon macrosegregation material testing and quality control/assurance verification at operating reactors from a decommissioned facility.

Large safety-related concrete structures credited for safety

The destructive examination and material testing should additionally include large irreplaceable safety-related concrete structures like the containment, the irradiated nuclear fuel pool and the reactor building foundation which can also be weakened radiation in close proximity to the high levels of radiation and by alkali silica reaction (ASR) all of which attack the material's compression and seismic strength. The ASR issue is a good example where the industry has demonstrated resistance to harvesting and materially testing concrete samples to assess aging.

¹³ "Bill for pressure vessel heads could top \$1 billion," Power, S&P Global Platts, February 3, 2003, https://online.platts.com/PPS/P=m&s=1029337384756.1478827&e=1096494853343.-2672017605169417981/?artnum=2PU004072A0N1mUD12N226_2

Seabrook Units 1 and 2 illustrate the example. Unit 1 is operating and under review for a 20-year license extension. Unit 2 construction was cancelled in 1989. The concrete pours were identical. Seabrook Unit 1 concrete structure is identified to have lost significant compression strength due to the ASR degradation process as identified in an emergency enforcement petition filed by Citizens Within the Ten Mile Radius (C-10).¹⁴

C-10 has identified that “water has been leaking through the containment structure at Seabrook since its initial construction. Significant water infiltration extends from 80 feet below to 6 feet above ground level. Due to ASR degradation Seabrook nuclear power plant is in violation of their current license. Aggressive water has been leaking through their containment since 1990. The Nuclear Regulatory Commission (NRC) Mortar Bar Test revealed that the ASR in Seabrook’s affected and unaffected concrete was not “self limited”. The reactor’s ASR is active and progressive, it continues to erode unabated with no way to be repaired.”¹⁵

“In 2010, NextEra and the NRC reported ASR concrete degradation was confirmed in Seabrook’s control building. The data revealed a moderate to severe reduction of the mechanical properties through lab certified petrographic testing. The NRC has repeatedly stated that ASR is confirmed only through petrographic examination in accordance the ASTM code. IN NRC report (ML13151A328) “The first core samples in April and May 2010. This area was selected because, qualitatively, it had the most significant groundwater intrusion, and the walls show the most extensive pattern

¹⁴ Additional Comments of Citizens Within the Ten-Mile Radius (C-10), Emergency Enforcement Petition (10 CFR 2.206), February 15, 2016

http://www.beyondnuclear.org/storage/decommissioning/asr_seab_02152016_prb_ML16047A021.pdf

¹⁵ Ibid, C-10 2.206, p.3 of 6

cracking and secondary deposits. The initial examination of the core samples was positive – the core samples displayed the visual characteristics of high quality, competent concrete and proper concrete placement procedures. However, subsequent quantitative testing revealed a reduction in concrete strength and elasticity modulus (Young’s modulus).”¹⁶

Rather harvest core samples of concrete from the adjacent cancelled Unit 2 facility, NextEra has opted to create a simulation of the Unit 1 concrete to research the ASR phenomenon on structural safety margins. “NextEra has further eroded confidence in their research by creating their own test specimens in Ferguson, Texas and then applying the study data to evaluate the current and future impact of ASR on Seabrook Station concrete structures. This study does not test any actual samples from Seabrook’s primary containment nor can the unique confluence of Northeast weather, storm surge etc. be adequately replicated. The study is not valid. NextEra’s intent to apply the study results to evaluate the current and future impact of ASR at Seabrook Station is unsound. This illogical method fails to provide any measure of confidence to the citizens impacted by the level of safety at Seabrook Station. Seabrook station has been in violation of their current license for seven years because of ASR. It has been discovered in a number of Seabrook structures, but the primary containment building still has not been tested to confirm ASR or the extent of active and progressive concrete degradation. The NRC has not required that NextEra change their position based on their false assumption that ‘reinforcing steel and other restraints limits ASR.’ They have not required NextEra to reverse their current position to test only ‘replica’ cores in their

¹⁶ Ibid, C-10 2.206, p. 4 of 6

Ferguson, Texas study rather than also testing actual cores from Seabrook Station's primary containment structures. NRC stated, 'NextEra's off-site research test program must represent the actual in-situ conditions of Seabrook's primary containment.' The NRC statement is frankly, absurd. NextEra's ASR current position is scientifically unfounded. Their 'replica' research study has not been peer reviewed. Repeatedly, experts send commentaries to the NRC and NextEra to state the scientific truth but NextEra continues to resist. The NRC does not correct NextEra's assumption and position. It is morally outrageous and scientifically unsupportable to not test Seabrook's actual concrete in primary containment with ASTM code testing given the safety risk. Frankly, NextEra has not fooled the public. We know Texas's concrete study cannot accurately represent Seabrook Station's ASR concrete degradation. No one will know the degree, extent and rate of Seabrook's ASR concrete degradation under their current license unless the actual in-site primary containment data in New Hampshire is tested though the petrographic examination in accordance the ASTM code, as represented in C-10's 2.206 petition."¹⁷

Additional information supporting the need to harvest concrete core samples from the cancelled Unit 2 to research and evaluate the in-field residual safety margins for Unit 1 compression strength is contained in the C-10 Petition to Intervene in the Seabrook relicensing proceeding.¹⁸

¹⁷ Ibid, C-10 2.206, p. 5 of 6

¹⁸ Citizens Within the Ten-Mile Radius (C-10) Petition for leave to intervene: Nuclear Regulatory Commission Docket No. 50-443 NextEra License Amendment Request for Seabrook Station, <http://www.beyondnuclear.org/storage/decommissioning/NRC%20Docket%2050-443%20C-10%20petition%20to%20intervene%20-%204.10.17-1.pdf>

In addition to ASR, the effect of radiation on concrete structures in high fields of radiation can be researched and evaluation from core samples gathered from decommissioned nuclear power plants as a way of measuring residual safety margins in operating nuclear power stations.

Thank you for the opportunity to comment,

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Paul Gunter, Director

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