Response to Public Comments on Draft Regulatory Guide (DG)-1373
"Fresh and Spent Fuel Pool Criticality Analyses"
Proposed New Regulatory Guide (RG) 1.240

On September 8th, 2020, the U.S Nuclear Regulatory Commission (NRC) published a notice in the *Federal Register* (85 FR 55522) that Draft Regulatory Guide, DG-1373, (Proposed New Regulatory Guide (RG) 1.240), was available for public comment. The Public Comment period ended on October 23rd, 2020. The NRC received comments from the organizations and people listed below. The NRC has combined the comments and NRC staff responses in the following table.

1.Charles Rombough CTR Technical Services, Inc. ADAMS Accession No. ML20281A530	2. Dale Lancaster Nuclear Conultants.com ADAMS Accession No. ML20295A200	3. Anonymous ADAMS Accession No. ML20296A546
4. Benjamin Holtzman, Senior Project Manager Fuel and Radiation Safety 1201 F Street, NW, Suite 1100 Washington, DC 20004 ADAMS Accession No. ML20297A267	5. Phil Couture, Manager Fleet Licensing Programs Entergy Operations, Inc. and Entergy Nuclear Operations, Inc. 1340 Echelon Parkway Jackson, MS 39213 ADAMS Accession No. ML20302A006	6.Gary Peters, Director Licensing & Regulatory Affairs Framatome Inc. 3315 Old Forest Road Lynchburg, VA 24501 ADAMS Accession No. ML20302A007
7. Gary Peters, Director Licensing & Regulatory Affairs Framatome Inc. 3315 Old Forest Road Lynchburg, VA 24501 ADAMS Accession No. ML20302A434		

Commenter	Section of DG-1373	Specific Comments (These are the full comments as provided in each submission)	NRC Resolution
1. Charles Rombough	Section C 1.k	The standard practice for depletion parameters has been to use a limiting "burnup averaged" value where the averaging is across the whole burnup range of the assembly. This has been acceptable in the past. For example, if an assembly experiences cycle average soluble borons of 1400, 800, and 800, and a continuous burn at 1000 ppm is shown to be conservative, then using a value of 1000 across the whole burnup range of the assembly is acceptable. Has the NRC changed its position? If an assembly experiences a cycle average boron of 1400 ppm for only one cycle, would the depletion have to be done at 1400 ppm through the whole burnup of the assembly? If so, this is unreasonable. Please provide justification for why this new position is reasonable.	The NRC disagrees with the comment. The comment states in part, "The standard practice for depletion parameters has been to use a limiting "burnup averaged" value where the averaging is across the whole burnup range of the assembly." This is not true because the standard practice and current NRC position has been to use a limiting cycle-average soluble boron consistent with John Wagner, "Impact of Soluble Boron for PWR Burnup Credit Criticality Safety Analysis," Trans. Am. Nucl. Soc., 89, November 2003. The Wagner paper is the only published work on the use of a constant soluble boron concentration during the depletion portion of a spent nuclear fuel criticality safety analysis and has been the industry touchstone on the topic since it was published. The Wagner paper provides the basis for NEI 12-16 Revision 4 (ADAMS Accession No. ML19269E069), section 4.2.1, sub-section <u>Soluble Boron during Depletion.</u> Wording in NEI 12-16 Revision 4, <u>Soluble Boron during Depletion</u>, clearly states a cycle-average soluble boron is to be used in the analysis but wording in NEI 12-16 Revision 4, section 9.4 is not as clear. However, several licensee applications have used a constant soluble boron in the depletion portion of the analysis with basis other than cycle-average and have been reviewed on an application specific basis. The clarification in C.1.k was included to address those attempts. Therefore the C.1.k clarification is not a new NRC position, but

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			rather reinforcing the correct interpretation of the current position. The NRC staff made no change to DG 1373 as a result of this comment.
2. Dale Lancaster	Section C.1.b	Historically Dominion has identified regulatory margin and margin reserved for future changes. Item b of the exceptions mentions "maintain(ing) any excess safety margins being used to justify assumptions or simplifications." Please clarify item b such that identification of regulatory margin is sufficient for maintaining any excess safety margins being used to justify assumptions or simplifications and that a licensee can still reserve margin for its own use (i.e., future changes).	The NRC disagrees with the comment. NEI 12 16, Rev. 4, Section 1.6 states, in part, "Use of engineering judgment and assumptions may incorporate risk insights as part of a "graded" licensing approach and is acceptable as long as the assessments consider relevant safety margins and defense-in-depth attributes. For example, a criticality analysis that demonstrates a maximum k_{eff} with a relatively large margin to the regulatory k_{eff} limit, may be permitted to make more assumptions about results or uncertainties than a criticality analysis that demonstrates a maximum k_{eff} with a relatively small margin to the regulatory k_{eff} limit." DG 1373 item C.1.b merely finishes the concept by acknowledging that assumptions and/or simplifications made in a nuclear criticality safety analysis with a large margin regulatory k_{eff} limit may not be appropriate as changes, whether a single change or an aggregate of changes, are made that reduce that margin. The NRC staff made no change to DG 1373 as a result of this comment.
Dale Lancaster	Section C.1.e	This exception is excellent but there may still be some confusion on definitions. NEI-12-16 Section 4.2.3 says, "it covers all uncertainties associated with depletion, such as uncertainty in computation of the isotopic inventory by the depletion code,	The NRC disagrees with this comment in part. Specifically, the comment states in part that "The historically used 1.5% of the worth of fission products and actinides will no longer be

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		uncertainty in cross-sections (both actinides and fission products), etc." Historically, the NRC has interpreted depletion uncertainty to only cover the uncertainty in isotopic content. The industry position is it includes the uncertainty in cross sections. It is good news that the NRC and industry appear to agree now that much more data has been provided to the NRC. The historically used 1.5% of the worth of fission products and actinides will no longer be needed. If this understanding is correct, no changes to Section C.1.e are needed.	needed. If this understanding is correct, no changes to Section C.1.e are needed." The comment appears to confuse the status of validation of the depletion code used in SFP nuclear criticality safety analysis and the validation of the computer code used to calculate keff in the SFP. These are two different codes and each must be validated seperately. The NRC staff made no change to DG 1373 as a result of this comment.
Dale Lancaster	Section C.1.k	Item k states, the "NRC does not endorse other interpretations of the phrase "burnup averaged," such as averaging across the whole burnup range for a given fuel assembly." The average over the whole burnup range has been implemented successfully in two recent license applications. In the response to an NRC question on use of a burnup averaged soluble boron for Indian Point Unit 2 (ML19157A309 from June 6, 2019), a significant number of calculations were performed that showed that using the average soluble boron over the entire burnup range of interest produced the same reactive fuel as if a boron letdown curve were used. Figure 1.2 from that RAI response is provided here. (See figure in ADAMS Accession No. ML20295A200) In the SER for Indian Point Unit 2 given in September 2019 (ML19209C966) the NRC said, *"The licensee's analysis cites the above-mentioned paper by J. C. Wagner to justify its use of a constant soluble boron concentration rather than a timedependent soluble boron letdown curve for its depletion calculations. However, the licensee is not using the constant soluble boron concentration in a manner consistent with*	This comment covers the same topic as Submission number 1 Comment number 1. See the resolution to Submission number 1 Comment number 1 accordingly. The NRC staff made no change to DG 1373 as a result of this comment.

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		the reference. Instead, the licensee is using a fuel assembly's lifetime burnup averaged soluble boron. Based on similar concerns discussed in Section 3.4.3.3.4 above, the NRC staff requested the licensee to justify the use of a constant average soluble boron concentration in its depletion calculation in order to address any potential nonconservatisms. In its June 6, 2019, letter, the licensee responded to the NRC staff's RAI (RAI 1) regarding its use of a fuel assembly's lifetime burnup averaged soluble boron. The licensee explicitly addressed legacy fuel, and where necessary, determined the margin between fuel assembly actual burnup and the minimum burnup requirements. The NRC staff finds the licensee's use of a fuel assembly's lifetime.burnup averaged soluble boron acceptable for its legacy fuel. For current and future cycles, the licensee confirmed that the cycle average soluble boron will be less than the fuel assembly's lifetime burnup averaged soluble boron that was assumed in the application. Since the fuel assembly's lifetime burnup averaged soluble boron (assumed in the application) exceeds the cycle average soluble boron of every cycle the fuel assembly has experienced, the use of the fuel assembly's lifetime burnup averaged soluble boron is considered acceptable by the NRC staff.” It is understood that the NRC acceptance for the Indian Point Unit 2 application is not a generic approval for all future applications but the method of the analysis done for Indian Point would apply to all PWRs. Millstone 3 recently addressed the same issue on burnup averaged soluble boron (ML19092A332, March 27, 2019 plus revision ML19135A067, May 7, 2019). The NRC response in the SER (ML19126A000, May 28, 2019) was: *“The use of multi-cycle averages or fuel assembly lifetime soluble boron average could be non-conservative, especially if the higher soluble boron cycle occurred just prior to the fuel being placed in*	

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		the SFP. This possibility has not been sufficiently vetted for the practice to be considered generally acceptable." The Millstone 3 approach to solving the issue included redoing the Wagner analysis to confirm the understanding of the paper. Once the Wagner paper was confirmed, they showed how it does not apply when using a burnup averaged boron. This was done by analysis at the three most limiting burnups. They found the maximum non-conservatism using the Wagner defined cycles to be 23 pcm. This effort was followed by analysis of a limiting Millstone three cycles out to 60 GWd/T including the effect of burnable absorbers. For this case, the most limiting analysis point showed a maximum non-conservatism of 43 pcm. This Millstone 3 analysis was done using CASMO-5. To confirm that these CASMO-5 results apply to KENO, confirmation runs were made that showed a 37 pcm non-conservatism. The Millstone 3 RAI response only analyzed 4 burnup points that were determined to be the most limiting. The Indian Point analysis covered 44 burnups. The Indian Point analysis used Monte Carlo runs where each run had a one sigma uncertainty of 12 pcm. Indian Point analysis had one point (38 GWd/T burnup) with a 21 pcm non-conservatism but this is less than 2 sigma of nonconservatism so it could not establish any non-conservatism. Further, Dominion analysis of that same burnup point showed an 18 pcm non-conservatism which is well within the uncertainty. In review of the analysis, it may be possible for some non-conservatism when assuming a burnup averaged soluble boron but it is small and comparable to normal uncertainties in the Monte Carlo analysis. Typical criticality analysis allows 1000 pcm for regulatory discretion, so the possible tens of pcm from this effect should be contained in this regulatory discretion. If a short cycle occurs at a power plant (which has happed many times), if burnup averaging of the soluble boron is not allowed, the assemblies in the core at the time would not be allowed to credit burnup and would be required to be stored in	

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		Region 1. A remedy would be to establish a burnup adder for short cycles. This can be done but adding this complexity for a complete core of assemblies is undesirable. If burnup averaged soluble boron is used, the number of assemblies that do not meet the soluble boron requirements will be few. It is recommended to remove exception k to NEI-12-16.	
3. Anonymous	General	NOT APPLICABLE: This comment is outside the scope of this regulatory guide.	No NRC response is necessary.
4. Benjamin Holtzman	Section C.1.a	The use of the term "burnable absorber" is typically used for in-reactor neutron absorbing material such as gadolinium.	The NRC agrees with the comment and changed "burnable absorber" to "neutron absorber" in Section C.1.a of DG-1373.
Benjamin Holtzman	Section C.1.a	NEI 12-16, Section 1.4 discusses the double contingency principle. However, the example provided in DG-1373 "Exception A" is not related to double contingency principle. With respect to the specific example provided under "Exception A," in many cases, the neutron absorber panels are not yet installed at the time the initial criticality safety analyses are performed. Either the racks have not been manufactured, or the absorber inserts are used only together with assemblies that are inserted in the racks. In both situations, no documents exist to show panels are correctly installed at the time of the criticality analysis. However, this should not lead to the conclusion that because of the absence of such documents, panels cannot be assumed to be correctly installed. Specifically, in these cases, racks would be manufactured, or inserts inserted with assemblies, under a nuclear quality assurance (QA) program with the appropriate controls. Therefore, an assumption of incorrect installation would be inappropriate at the time the analyses are performed. While a licensee or applicant may consider certain unlikely conditions as part of the off-nominal condition, such as the	The NRC disagrees with the comment as follows: The comment takes exception to the example used in DG-1373 C.1.a. DG-1373 C.1.a. expands on the guidance in NEI 12 16 Rev. 2 regarding the Double Contingency Principle as it applies to SFP and New Fuel Storage Vault (NFSV) nuclear criticality safety analysis. NEI 12-16 Rev. 4 uses an example of items controlled by licensee's Technical Specification. DG-1373 C.1.a. expands on the guidance in NEI 12 16 by elucidating the idea that aspects outside the licensee's Technical Specification can come into consideration when applying the Double Contingency Principle and uses an example of an incorrect installation of a neutron absorbing material, stating in part, "…such as the possibility that a burnable [neutron] absorber panel may not have been correctly installed.

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		possibility that a neutron absorber panel may not have been correctly installed, this should not be part of the normal condition assumptions. Neutron absorber panels are installed under QA programs and any known deviations are captured in the utility's corrective action program for resolution. It would be an unnecessary administrative burden to require utilities to produce records regarding the status of the long-standing spent fuel pool racks when other processes are in place.	However, if no controls or documents exist to preclude such a condition, then the licensee or applicant should treat it as part of the normal condition." The commentor takes exception to the use of an incorrect installation of a neutron absorber panel as an example because licenses have controls to preclude or identify the incorrect installation of a neutron absorber panel. The NRC acknowledges that licensee's have procedures and programs that should preclude or identify the incorrect installation of a neutron absorber panel. However, NRC does not believe that invalidates the example as that means that there are "…controls or documents exist to preclude such a condition…" and the incorrect installation of a neutron absorber would not need to be treated as part of the normal condition. The comment then asks that if the example is retained that the NEI 12-16 section referenced be changed from 1.4 to 5.2.2. The NRC disagrees as the point of DG-1373 C.1.a is expanding on the guidance in NEI 12 16 Rev. 2 regarding the Double Contingency Principle as it applies to SFP and NFSV nuclear criticality safety analysis and that changing from the section of NEI 12-16 that discusses the Double Contingency Principle to one that does not discuss the Double Contingency Principle would dilute the exception. The comment goes on to ask that if the example is retained that the sentence, "However, if no

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			controls or documents exist to preclude such a condition, then the licensee or applicant should treat it as part of the normal condition" be deleted. The NRC disagrees as removing that sentence would indicate that all conditions would have to be considered concurrent, even if there are controls to prevent it. The NRC staff made no change to DG 1373 as a result of this comment.
Benjamin Holtzman	Section C.1.e	The intent of this section is unclear. The clarification/exception refers first to PWR requirements (Section 4.2.3 of NEI 12-16), then BWR requirements (Section 4.3.1 of NEI 12-16), and then again to PWR requirements (Section 4.2.3 of NEI 12-16).	The NRC agrees with the comment as an editorial error. The intent of DG-1373 Section C.1.e is to reference just the PWR requirements in NEI 12-16, Rev. 4, Section 4.2.3 and not BWR requirements in Section 4.3.1. The NRC staff fixed the error.
5. Phil Couture	General	Entergy has been an active participant in the NRC and industry meetings regarding this topic and endorses the industry comments provided by the Nuclear Energy Institute (NEI)	This comment endorses the comments made in submission number 4, see the resolution of the comments for submission number 4. The NRC staff made no additional changes to DG-1373 as a result of this comment.
6. Gary Peters	Section C. 1	NEI 12-16, Revision 4 discusses the use of nuclide depletion for SFP evaluations. DG-1373 does not address the acceptability of the "fresh fuel equivalence" method as an alternative to nuclide depletion.	The NRC disagrees with this comments as follows: This comment requests that "fresh fuel equivalence" be included as an acceptable alternative to nuclide depletion, cites NUREG/CR-6683, "A Critical Review of the Practice of Equating the Reactivity of Spent Fuel to Fresh Fuel in Burnup Credit Criticality Safety Analyses for PWR Spent Fuel Pool

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			Storage," and a license amendment for Monticello as precedent. The method involves modeling the depletion of fuel assemblies to a point and then determining a fresh fuel assembly with the same k_{eff} as the depleted fuel assembly. The fuller title for the method is reactivity "enrichment" fresh fuel equivalent (REFFE).The original REFFE methods were shown to be non-conservative. NUREG/CR 6683 evaluates the REFFE method for pressurized water reactor fuel. NUREG/CR-6683 indicates the method can be used successfully, within constraints and limitations. The cited precedence is a boiling water reactor. These examples indicate the method could be used successfully in future applications. However, the NRC is declining to add the method to DG-1373 for the following reasons: • The REFFE method is not included in NEI 12-16 Rev. 4, which means it is out of scope of DG-1363 which endorses, with clarifications and exceptions, NEI 12-16 Rev. 4. • Other potentially acceptable methods have not been included in DG-1373. • Regulatory Guides are not requirements and do not prevent applicants from requeesting to use alternate methods.
Gary Peters	Reference 8	Reference 8 of DG-1373 lists a publication date of September 2012 for NUREG/CR-6683. The correct date, as listed in the ML archive (ADAMS Accession No. ML003751298), is September 2000.	The NRC agrees with the comment as an editorial error and DG-1373 Reference 8 was changed from September 2012 to September 2000.

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7. Gary Peters		NOT APPLICABLE: This is a duplicate submission from Commenter 6	Submission number 7 is a duplicate of Submission number 6, see the resolution of the comments for Submission number 6 accordingly.