August 15, 2011

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Re: Request for Additional Information regarding Alan J. Blotcky Reactor Facility Decommissioning Plan, License R-57, Docket # 50-131

The Reactor Safeguards Committee for the Alan J. Blotcky Reactor Facility submits the enclosed document. "*VA Final Responses to 13 May 2008 NRC RAIs*"

The Responses to the Request for Additional information were prepared by AECOM. AECOM is the technical support service contracted by the VA Nebraska Western Iowa Healthcare System to assist with the decommissioning of the Alan J. Blotcky Reactor Facility.

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VA Final Responses to 13 May 2008 NRC RAIs

Introduction

This document represents an update to that submitted by the Department of Veterans Affairs (VA) in November 2010 in response to NRC's 13 May 2008 Request for Additional Information (RAI). The information provided in the Nov 2010 Response was based upon additional document research available at that time and results of the October 2010 on-site meeting with NRC and VA.

Since that time, VA worked with NRC to develop, and then implement, Additional Characterization field sampling efforts to further resolve the outstanding concerns. This final Response to RAIs incorporates these new data as well as additional information gathered since November 2010. The responses are also noted as Updated or Unchanged vs. Nov 2010 submittal.

Background

Based on the NRC's RAI and the following significant factors that impact the information provided in the Decommissioning Plan, (DP) VA will provide a major revision of the Decommissioning Plan. In the time since the original DP was submitted some of the radiological conditions have changed (e.g., a half-life of Co-60 has transpired), and waste disposal options have changed (Barnwell, SC has closed; a disposal facility in Texas has opened). This will necessitate a reevaluation of desirable LLRW disposal options. Furthermore, the NRC has finalized NUREG 1757, *Consolidated Decommissioning Guidance,* since the DP was originally submitted in 2004. The revised DP will be structured in accordance with the guidance.

In response to the NRC's comments and to attempt to expedite the decommissioning process, the revised DP will proved greater details of the proposed Final Status Survey (FSS) and will include a request for changes to the reactor facility's technical specifications (TS). The VA has also performed additional site characterization in response to several of the RAIs. The purpose of the characterization was to better understand the isotopes of concern and to examine other areas or systems of the not fully characterized. Specifics are provided below.

NRC Comments and Responses

1(a). Provide criteria for soil and volumetrically contaminated materials to be left in place

Basis:

Section 5.1.2 of the DP indicates that DCGLs are not needed for soils. However, Section 8.1.1 of the DP indicates that after removal of the reactor internals, core bores will be made into the surrounding material (i.e., epoxy/gunite, steel, concrete, and surrounding soil). This section states that some or all of the pit structure may be left intact or that further remediation may be required, potentially including removal of the pit structure and concrete and surrounding soil. However, the DP does not include specific criteria that would be used to determine if such remediation is needed. Criteria for soils are also needed to determine if minimum detection levels were sufficient for the characterization of soils around and under the reactor pit.

Path Forward:

The VA should provide criteria that will be used to determine if soil and other potentially volumetrically contaminated materials around the reactor pit can remain in place.

VA Response (Updated):

VA will use the soil screening levels as provided in NUREG-5512, Volume 3 (Table 6-91; $P_{crit} = 0.10$) and NURGE-1757, Volume 1, Revision 2, Appendix B (Table B.2) to demonstrate compliance with 10 CFR 20, Subpart E decommissioning criteria for soil and concrete remaining at the site. The sum-of-fractions rule will apply to isotopes detectable above background levels (including hard-to-detect isotopes).

After removing activated concrete to a level that will considered as low as reasonably achievable (ALARA), VA will apply the release criteria referenced above to concrete that will remain in place. The determination of ALARA will be based on safety considerations associated with deploying personnel into the reactor tank/pit. Once the structural limits of the concrete have been reached (in the opinion of a professional structural or civil engineer) and radioactivity is below the screening criteria, remediation will be considered complete. While it is currently expected that the surrounding soils are not activated/contaminated, (as evidenced by the site characterization), soil samples will be collected (either horizontally through the concrete or vertically from the basement floor) to demonstrate compliance with the above criteria. The sampling approach will be defined in the Final Status Survey Plan (FSSP).

1(b). Describe criteria for drains, ventilation ducts, and any other miscellaneous components to be left in place

Basis:

Section 8, of the DP states that decommissioning activities will include removal of some ventilation ducts and drains and perhaps other miscellaneous components. The DP is not clear on what criteria will be used to determine if such components may remain in place. It may be that the VA intends to apply the building surface DCGLs to these components, but the NRC staff has not seen that stated in the DP.

Path Forward:

The VA should clearly describe what criteria are to be applied for leaving components in place other than building surfaces. If the VA intends to use the DCGLs developed for building surfaces, the VA should justify the appropriateness of the DCGLs for these other components.

VA Response (Unchanged):

The VA plans on applying the same DCGLs developed for interior building surfaces to inaccessible areas such as ventilation ducts and drains. Contaminated systems (accessible or inaccessible) that do not meet the DCGLs will be removed. If removal of a system is unreasonable, the VA may propose specific alternative DCGLs such that the contamination can remain in place or fixed within the system so that they would not present a future exposure source. Application of such alternative DCGLs will require NRC approval as an addendum to the DP.

1(c). Provide justification for radionuclide mix

Basis:

The radionuclide mix assumed by the VA is based on results of a single sample of resin that was analyzed (described in Attachment 2 of the Facility Characterization Report (FCR)). It is unclear to the NRC staff how this single sample would be representative of all different areas of the VA facility that might be contaminated. The VA has not justified this sample as being representative of all areas. In addition, the NRC staff was unsure why the naturally occurring radionuclides thorium-230 (Th-230) and lead-210 (Pb-210) would be present as contaminants. If these nuclides are under the licensee's control and elevated above background, then they should be addressed, but there was no discussion about these in the DP.

Path Forward:

The VA should provide additional information on the expected radionuclide mix. If one sample is thought to be representative, justification should be provided. If not, the VA should describe an alternative for determining the radionuclide mix and when the determination would be made.

VA Response (Updated):

Through additional characterization efforts in May 2011 (AECOM 2011), VA demonstrated that Th-230, Pd-210, Po-210, and plutonium isotopes should not be considered as isotopes of concern during decommissioning. This was demonstrated by the results of demineralize resin and soil sampling anlaysis that specifically looked for these potential contaminants. Also, total and removable surface contamination measurements did not indicate the presence of alpha contamination. The revised DP will remove these isotopes from discussions regarding isotope mix or isotopes of concern.

1(d). Reevaluate Th-230 and Pb-210 DCGLs.

Basis:

As described in Attachment 2 of the FCR, the VA calculated DCGLs for building surfaces for Th-230 and Pb-210 using the DandD screening code. The NRC staff has concerns about two of the DCGL values. First, for Th-230, it appears that the results were interpreted incorrectly. Based on the DandD report for the Th-230 run, the radionuclide used was "Th-230+C" and the nuclide concentrations for the source term were distributed among all progeny. This means that the input concentration is the concentration of all radionuclides in the Th-230 decay chain summed together. This differs from the interpretation by the VA that the concentration is the concentration of Th-230 only. The way that the DandD code distributes activity for decay chains is described in NUREG-1757, Volume 2, Section 4.2.2.3 (April 2001).

For Pb-210, the NRC staff believes that if Pb-210 were present as a contaminant, then most likely Polonium-210 (Po-210) would also be present, because the reactor facility has been shut down and relatively undisturbed long enough that the Po-210, with a half life of 138 days, would have ingrown close to equilibrium with Pb-210. However, the VA did not include Po-210 as part of the radionuclide mix, nor did it calculate a DCGL for Po-210.

Path Forward:

If Th-210 and Pb-210 are retained in the radionuclide mix, the VA should reevaluate the DCGLs for these nuclides, or should provide justification that the proposed DCGLs are appropriate.

VA Response (Updated):

The revised DP will remove these isotopes from discussions regarding isotope mix or isotopes of concern. See response to Comment 1(d) above.

1(e). Clarify what radionuclides are detectable for building surfaces DCGL

Basis:

Attachment 2 of the FCR provides, for building surfaces, the proposed DCGLs for individual radionuclides and the calculated DCGL for gross measurable activity. A note to the table in this attachment attempts to describe which nuclides are detectable. The note states "Detectable refers to nuclides which are detectable when the detector is operated at an alph+beta counting voltage using an efficiency for Tc-99." The table does not describe what instrument is proposed for use in the measurements. It is unclear to the NRC staff what radionuclides would actually be measured during the proposed measurements. The note indicates counting of alpha and beta particles, but indicates an efficiency for a beta-emitter, Tc-99, would be used. In the table, the alpha-emitting nuclides are indicated as not detectable, which seems to conflict with alpha plus beta particles being measured. Also, the table does not appear to account for differences in actual efficiency compared to the efficiency proposed for use (i.e., efficiency for Tc-99).

Path Forward:

The DP should clearly describe modifications to the gross DCGL to account for detectability of certain radionuclides or certain radiation emissions. If the instrumentation intended to be used impacts the detectability, details about the instrumentation should be provided.

VA Response (Unchanged):

The DP states, "Detectable refers to nuclides which are detectable when the detector is operated at an alpha+beta counting voltage using an efficiency for Tc-99" however the Tc-99 efficiency is for the observed beta radiation only. In fact, the alpha/beta reading instruments (such as the Ludlum 2360) utilize separate efficiencies for alpha particles. The alpha efficiency is normally based on Th-230, with the same voltage used as the Tc-99. The alpha efficiency, which is correspondingly lower than the beta efficiency, can be used to quantify alpha activity separately from the beta activity. The revised DP will specifically address what types of instruments will be used and their ability to measure the isotopes of concern. The revised DP will also indicate what isotopes are to be used for establishing detector efficiencies.

It should be noted VA no longer concurs with derivation of the DCGL based on the sample results presented in the 2003 Characterization Report (Duratek 2003)). VA has demonstrated that the Pu-241, Th-230, and Pb-210 are not present or are within

background levels (as referenced in the Eberline Services Case Narrative in Attachment 1 of the Characterization Report) and should not be considered in the derivation of the DCGL.

1(f). Clarify the criteria for offsite release of materials

Basis:

The DP describes the As-Low-As-is-Reasonably-Achievable (ALARA) criteria for surfaces, but it is unclear exactly how the criteria are intended to be applied. Section 1.8 of the DP states limits for loose (removable) and fixed contamination (1000 and 5000 dpm/100 cm2, respectively) governing the "free release of materials." The same section further states that "Plant systems and reactor components constitute an exception, and will receive decommissioning to meet the calculated DCGLs."

It is unclear to the NRC staff what the VA intends here. First, the limits for loose and fixed contamination are described as for free release of materials, which the NRC staff interprets as for materials that will be released from the site prior to license termination. But the DCGLs apply to building surfaces that will remain on site at license termination. No clear statement is made in this section about what is being done to assure that future doses are ALARA.

In Section 5 of the DP, after discussing the DCGLs, the VA states that "Conservatively, decommissioning is being planned to decontaminate and remove radioactive concentrations at the facility to values in the not to exceed a range [sic] of 5,000 dpm/100 cm2." In this statement, there is no exclusion of the plant systems and reactor components, which seems different than the statement in Section 1.8. In addition, it appears to the NRC staff that this goal of 5000 dpm/100 cm2 is for materials remaining on site, not materials being released from the site.

In Section 7.1 of the DP, the VA restates the limits governing the free release of materials that were given in Section 1.8.

In Section 14.1 of the DP, the VA states that the "DCGLs, in combination with the existing AJBRF [Alan J. Blotcky Reactor Facility] material release limits, will be utilized as the regulatory basis for release of the site for unrestricted use." The DP does not describe the "existing AJBRF material release limits," but it appears to the NRC staff that this statement mixes limits to be applied to building surfaces (and potentially materials) *remaining on site at license termination* with limits to be applied to the *off site release of materials prior to license termination.* The NRC staff distinguishes between these two cases, because the regulatory bases (i.e., in the NRC regulations) for the two are different.

Among these cited sections of the DPs, these statements appear inconsistent and unclear.

In addition, the DP does not address the planned releases of bulk material or aggregated waste (bags, drums, etc.), that may be volumetrically contaminated.

Path Forward:

The VA should clarify the limits it proposes using, being careful to specify how the limits will be used (e.g., are limits for building surfaces or materials to be left on site at license termination or for release of materials off site prior to license termination). The VA should clarify statements that may relate to ALARA goals or limits, if such are proposed. The VA should describe the existing release limits that will be used for the decommissioning work. If the VA intends to release volumetrically contaminated materials from the site (i.e., other than for disposal at a licensed facility), the DP should describe conditions of release and should provide criteria proposed (refer to NUREG-1757, Volume 1, Section 15.11 and to Information Notice 85-92). The VA should be consistent throughout the DP in describing the different limits.

VA Response (Updated):

VA wishes to clarify the issue of free release of material as compared to final building status. By implementing Regulatory Guide 1.86 for releasable items from within the facility, VA will utilize 5000 dpm/100 cm^2 fixed and 1000 dpm/100 cm^2 removable criteria for beta/gamma contamination. However, VA has released most of the loose items and equipment to a more conservative limits of 1000 dpm/100 cm^2 fixed and 200 dpm/100 cm^2 beta/gamma and 100 dpm/100 cm^2 fixed and 20 dpm/100 cm^2 alpha. When executing the FSS the VA shall employee DCGLs based on NRC screening levels in NUREG 1757, Volume 1, Appendix B.

The demonstration of the DGCLs conforming to the principles of ALARA shall occur using appropriate NRC guidance in Appendix N of NUREG 1757, Vol 2.

It should be noted, however, the current proposed DCGL of 27,000 dpm/100 cm^2 is not longer valid based on the results of additional site characterization. New DCGLs will be presented in the revised DP.

1(g). Clarify applicability of criteria for removable contamination

Basis:

Section 7 of the DP states that loose (removable) contamination will be eliminated where possible and states that a limit of 1000 dpm/cm^2 will apply to removable contamination for the "free release of materials." It is unclear to the NRC staff whether this criteria for removable contamination will be applied to building surfaces and components that will remain at the facility.

Path Forward:

The VA should clarify whether it proposes any criteria for loose or removable contamination for building surfaces, components, or other materials that will remain on site at license termination.

VA Response (Updated):

Please see response to Comment 1(f).

1(h). Justify that conditions for use of screening criteria for surfaces are met

Basis:

As described in the Characterization Report, the screening criteria used by the VA as DCGLs were obtained from a *Federal Register* notice published by the NRC staff, or from the DandD code. The NRC staff provided guidance on criteria for the applicability of these screening values. The guidance states that for use of the building surface screening criteria, the DP should describe how the following conditions would be met: (1) the residual radioactivity should be surficial and non-volumetric; (2) the residual radioactivity on surfaces should be mostly fixed, with no greater than 10% of the total being removable or loose; and (3) the screening values should not be applied to surfaces such as buried structures (e.g., drainage or sewer pipes) without adequate justification. These three criteria are discussed in NUREG-1757, Volume 2, Revision 1, Section H.2.2.

Path Forward:

The VA should justify that the site conditions are appropriate for the use of the screening values.

VA Response (Unchanged):

The revised DP will demonstrate that site conditions are appropriate for the use of the screening value as the DCGL based on the criteria stated above.

1(i). Clarify to what radiation emissions the criteria apply

Basis:

The DP does not describe the type of radioactivity (e.g., alpha, beta, gamma or some combination) to which the DCGLs and the criteria for loose and fixed contamination apply. Attachment 2 of the FCR includes a description of how a "reduced" DCGL was calculated,

including only those radionuclides that are detectable. This information is not presented in the DP when DCGLs are discussed.

Path Forward:

Provide a clarification in the DP regarding to what radiations or radionuclides the DCGLs apply.

VA Response (Unchanged):

The DCGL will be an aggregate of all the radiations encountered. This is to include alpha and beta activity radiation to ensure the most conservative result possible. There will be a DCGL for total activity and a smaller DCGL for measurable activity (based on the type of instruments appropriate for use). Activity calculations will be also performed using the most conservative detector efficiencies.

2. Clarify additional characterization or site historical information needed

Basis:

The FCR, in subsections of Section 8, describe a number of instances where the characterization was incomplete or insufficient, due primarily to difficulty accessing certain areas, but also due to other issues. The DP has not described how these areas will be sufficiently characterized during the remainder of the decommissioning work.

Section 2.3 of the FCR indicates that the pneumatic transfer system was rerouted in the 1960s, from its original terminus in the first floor. The DP and FCR do not discuss characterization and possible remediation for areas of the first floor potentially impacted by contamination from the pneumatic transfer system.

In addition, the DP does not provide information pertaining to the characterization, radiological status, and anticipated disposition of sanitary sewer lines and embedded piping.

Path Forward:

The VA should describe additional site history information or characterization needed and how it will be obtained. The VA should address whether characterization or decontamination or removal of sewer lines and embedded piping is needed.

VA Response (Updated):

VA has performed additional sampling in embedded piping, sewer lines, and the pneumatic transfer system. No contamination above the system detection limits or background levels was identified in these systems (AECOM 2011). The DP will indicate

that the in-wall pneumatic piping, embedded drain line piping, and sanitary sewer likely do not need to be remediated.

3. Provide details on proposed soil sampling around the reactor pit

Basis:

In Section 8.3 of the DP, the VA states "During the decommissioning of the reactor tank, horizontal bores will be made approximately at various depths of the pit in each quadrant to determine if the soil immediately outside the tank is activated or contaminated." No further details about the proposed sampling are provided in the DP.

Path Forward:

The VA should provide additional detail to clarify the commitment for soil sampling around the reactor pit.

VA Response (Updated):

VA believes compliance with decommissioning criteria can be demonstrated by sampling soil around the reactor pit at various depths by penetrating the pit wall horizontally or by sampling vertically from the reactor room floor. VA will propose both methods in the DP and leave the selection of the method up to the decommissioning contractor which will provide the details of the sampling method in their FSSP.

4. Clarify which method will be used to evaluate final status survey results

Basis:

Section 14.1 of the DP discusses general methods that might be used for evaluating the final status survey measurements results. This discussion describes two different methods, both based on recommendations from the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM). The first method is that fixed measurement results would be compared directly with the DCGLs. This is described as an ALARA methodology. However, it is unclear to the NRC staff whether the VA intends to use this method for compliance or intends this as an administrative goal.

The second method is the use of nonparametric statistical tests.

Section 14.1 appears to state that the VA intends to use both of these approaches. It is unclear to the NRC staff if this is the intent, as details have not been provided.

Path Forward:

The VA should clarify which method is to be used. If the VA intends to use both methods, the VA should describe conditions for use of each. The description should be clear as to which method will be used for regulatory compliance. If a method to be used for ALARA will only be considered an operational or administrative (not for compliance) method, this also should be clearly described in the DP.

VA Response (Updated):

VA will have an administrative goal of decontaminating all surface contamination to levels below the DCGLs. Final status survey measurements will first be compared against the DCGLs. If all measurements in a survey unit are less than the DCGLs, no statistical test is needed. If one or more measurements are above the DCGL, a non-parametric statistical test will be conducted to determine if the survey unit meets the release criteria. Details will be provided in the FSSP.

5. Provide evaluation of hot particles

Basis:

A number of reactor facilities have discovered hot particles in various locations during decommissioning. Origins of hot particles have included particles of failed fuel and particles of activated materials. The DP does not discuss the possibility of hot particles at the VA facility.

Path Forward:

The VA should evaluate the possibility of hot particles at the AJBRF facility. The DP should either justify that significant hot particles do not exist at the facility or demonstrate that characterization and final status surveys will detect such hot particles if they do exist.

VA Response (Updated):

The VA has never had any failed fuel within its facility, and does not expect to encounter hot particles. However, there will be daily, weekly, and monthly surveillance surveys and constant Heath Physics Technician coverage on all ongoing job activities. It should be noted that no hot particles were encountered during either the 2003 or the 2011 Characterization Sampling efforts (2011efforts involved four continuous weeks on-site).

6. Describe how hard-to-detect nuclides (HTDN) will be addressed.

Basis:

The soil sample analyses performed in support of the activation analysis, presented in the FCR (Attachment A of the DP), did not include analysis methods for the detection of hard-to-detect radionuclides (HTDN), alpha-emitters or low-energy beta emitters. Section 8.11 of the FCR (page 62) states that "Many of the radionuclides listed for soil in APPENDIX C are radionuclides that cannot be detected by gamma spectroscopy. Therefore, analysis of a few samples during decommissioning for these radionuclides is suggested." The DP does not describe a commitment to perform such additional analyses.

The DP does not explicitly address the potential for HTDN on structural surfaces and systems, or in soils. Table 4.2, Resin Sample Analysis Results Summary, indicates the presence of tritium (H-3), carbon-12 (C-14), and plutonium-241 (Pu-241) at detectable levels. Section 4.4 (page 45) states that "An environmental analysis of surface soil samples was performed to determine the presence of radiological or hazardous contamination present outside of the containment building. None of the soil samples analyzed by gamma spectroscopy showed any measurable radioactivity." A similar statement appears in Section 4.5, Subsurface Soil Contamination. A review of the derivation of the DCGLs for structural surfaces, as presented in Attachment 2 of the Characterization Report, indicates that HTDN were included in the calculation of the DCGLs. However, this is not explicitly stated in the body of the text of the DP, nor does it address the potential for HTDN in soil.

Path Forward:

The DP should provide clarification about how HTDN have been addressed, for soils and for building surfaces. If additional measurements of HTDN are to be made during decommissioning (including during further characterization, remedial action support surveys, or during final status surveys), the DP should provide commitments to perform such measurements.

VA Response (Unchanged):

To address the HTDNs in soil, VA will employ independent laboratories to determine the concentration of these isotopes.

During the course of remediation and final status survey, 100 cm^2 swipes will be taken. These swipes will be examined in a liquid scintillation counter or like apparatus capable of identifying the low energies of HTDN. Assumptions will be made for removable fractions and removal efficiencies.

7. Provide information about expected doses during decommissioning

Basis:

The DP does not include an individual or collective dose-estimate for workers during the decommissioning, as suggested by NUREG-1537, Part I, Appendix 17.1, Section 3.1.3. This information would be helpful in evaluating whether the radiation protection program is sufficient for the expected doses.

Path Forward:

The DP should provide discussion about the dose levels expected for workers involved in the decommissioning activities.

VA Response (Updated):

The VA will prepare a dose estimate for site workers and the public for anticipated decommissioning activities. However, the VA needs to collect additional information on the activity of some components to provide an accurate estimate. It is not expected that any of the workers involved in decommissioning activities will exceed 100 mrem. The collective dose of the AECOM staff during a month of onsite activities during May 2011 was 1 mrem for approximately 1,000 man-hours on-site.

8. Provide Method for Approving Changes to the DP

Basis:

Section 9.1.6 of the DP, page 80, cites one of the functions of the Reactor Safeguards

Committee (RSC) as follows: "The committee will review decommissioning procedures, decommissioning activities dealing with radioactive material and radiological controls as well as review and approve changes to the decommissioning plan." This section also lists the following as an RSC function: "Review proposed changes to the decommissioning plan or procedure changes, including changes in monitoring or control equipment, systems or testing to determine if there are safety questions as defined in 10 CFR 50.59." However, the DP does not discuss the conditions that would allow changes to the DP without prior NRC approval and the method that is proposed for the licensee to approve such changes, nor does it describe changes or conditions that would require prior NRC approval.

NUREG-1537, Part I, Appendix 17.1, Section 9, states that "Unless the DP contains a method to make changes without prior NRC approval, NRC will have to approve all changes to the plan, no matter how small or insignificant with regard to safety." The NRC staff has additional guidance on changes to DPs in NUREG-1757, Vol. 1, Chapter 18 (the licensee should note that Volume 1 is

generally applicable to materials licensees, but may provide helpful information for reactor licensees).

Path Forward:

The licensee should incorporate into the DP the change control criteria and method for modifications to the DP. This could be done by listing: (1) what in the DP may be changed by the licensee; and/or (2) what changes must be approved by the NRC. The DP should also describe how changes to the DP will be documented, and how the NRC will be notified of changes.

VA Response (Updated):

Any changes or alterations to the methods described in the DP necessary for execution of the decontamination process will ensure:

- The change does not conflict with requirements specifically stated in the license or impair the licensee's ability to meet all applicable NRC regulations.
- There is no degradation in safety or environmental commitments addressed in the NRC–approved DP.
- There are no significant adverse effects on the quality of the work, the remediation objectives, or health and safety.
- The change is consistent with the conclusions of actions analyzed in the technical specifications revised/established for the decommissioning project.

VA will not change programs and procedures related to dose modeling, final radiological surveys or restricted use/alternative criteria without prior NRC approval. All substantive changes will first be discussed with the NRC and an appropriate method for documenting the changes will be established (i.e., DP addendum, DP revision, letter to file, etc.)

9. Provide a Radiological Accident Analysis

Basis:

The DP does not discuss or analyze potential radiological accidents that could affect the public or occupational health and safety. A Radiological Accident Analysis, as described in NUREG 1537, Part I, Appendix 17.1, Section 3.3, would describe accidents that are directly related to decommissioning with emphasis on how they differ from accidents related to normal maintenance activities. Sufficient detail is needed to clearly define and analyze the consequence of any significant potential accidents, including all potential releases of

radioactivity to or from controlled areas of the facility. This information is needed for NRC staff preparation of the Environmental Assessment.

Path Forward:

The DP should include a Radiological Accident Analysis.

VA Response (Updated):

The revised DP will address credible accident scenarios in a Radiological Accident Analysis. These scenarios will examine potential uncontrolled releases of radioactive materials during decommissioning or during transport that could result in a measurable dose to a member of the public.

10. Provide proposed revision to Technical Specification

Basis:

Since there is no longer nuclear fuel at the facility, most of the operating license technical specifications will not apply after the DP is approved by amendment. However, the DP should describe the applicable decommissioning technical specifications (TS) to include the safety precautions necessary during the decommissioning phase (refer to NUREG-1537, Part I, Appendix 17.1, Section 5).

Section 8.0, page 58, states, "Requests for changes to the technical specifications will be prepared and submitted to address such impacts prior to the start of the actual decommissioning work." Based on this statement, NRC staff concludes that the licensee intends to submit the decommissioning technical specifications to the NRC for review and approval prior to the start of decommissioning work.

Path Forward:

The DP should include, at a minimum, the commitment by the licensee to include the information described in NUREG-1537, Part I, Appendix 17.1, Section 5, in the revisions to the TS. The NRC staff also notes that it may be preferable to include a request for changes to the TS as part of the DP package. The revised TS should include the four sections described in Section 5 of Appendix 17.1: 1) a section imposing limiting decommissioning conditions at the facility that is comparable to the limiting conditions for operation for required equipment and conditions; 2) a section providing for surveillance of the required equipment and conditions for decommissioning; 3) a section describing the residual facility and site to which the DP applies, and; 4) an administrative section that outlines the management structure, provides for review and audit functions, provides for development and use of the necessary procedures, and contains reporting and record-retention requirements.

VA Response (Unchanged):

The DP revision will include a section discussing the TS as suggested above. To expedite the decommissioning process, VA will include a request for revisions to the TS in the revised DP package.

11. Provide a Detailed Final Status Survey Plan

Basis:

The DP does not contain a proposed Final Status Survey Plan with adequate detail, as recommended by NUREG-1537, Part I, Appendix 17.1, Section 4. Section 8 of the DP, page 54, refers to Section 14 for the Final Survey Plan "(see Section 14 for the Final Survey Plan)". Section 14 generally references the MARSSIM and NUREG-1505 as "The basis for developing the final survey plan..."

Based on the above and the minimal information provided in Section 14, it appears that the intent of the licensee is to submit a Final Status Survey Plan to the NRC for review and approval at a later date. However, the latter is not explicitly stated in the DP. Clarification is needed as to whether the intent of the licensee to submit a Final Status Survey Plan for approval at a later date, or if Section 14 is the proposed Final Status Survey Plan. The NRC staff prefers the FSS Plan be part of the DP, but separate submittal can be acceptable if the DP commits to an acceptable process. For additional guidance, see NUREG-1757, Vol. 2, Chapter 4 (for guidance on final status surveys) and Section 2.2 (for guidance on flexibility in submittals).

Path Forward:

If Section 14 is to serve as the proposed Final Status Survey Plan, the licensee should provide a complete, detailed plan. The licensee should refer to the applicable guidance documents (NUREG-1537, Part I, Appendix 17.1, Section 4; and NUREG-1757, Volume 2, Chapter 4) to determine the information that should be supplied by the licensee to allow the NRC staff to determine that the FSS design or design process is adequate to demonstrate compliance with the radiological criteria for license termination.

VA Response (Unchanged):

VA intends to submit a FSS separately for the approval of the NRC. It is often best to wait for the DOC to plan decontamination and demolition activities prior to designing the FSS because the physical condition of the site is hard to predict at the time of DP submittal. Furthermore, the DOC will likely be requested to prepare the FSS Plan as part of their project scope. The revised DP will explicitly state that intention.

12. Provide Additional Information on Training Program

Basis:

The Training Program section of the DP, Section 9.4, lacks some of the information recommended by NUREG-1537, Part I, Appendix 17.1, Section 2.5. The items not described include:

- The required frequency for refresher training
- A statement that contractors will receive training on the DP
- A statement that existing training programs have been modified to account for the differences between normal operations and D&D; and
- The DP does not state that training on the principles and techniques of D&D activities will be conducted.

Path Forward:

The licensee should provide additional information in Section 9.4 of the DP, and the licensee's training program modified accordingly:

VA Response (Updated):

VA will describe all the necessary training requirements in the revised DP. It should be noted: 1) refresher training occurs yearly and as such is not expected for the length of D&D operations, 2) the revised DP shall state the training required for execution of the DP, 3) the revised DP will explain how the training programs in D&D operations will differ from normal training on an in use reactor, and 4) D&D operations technique and principles shall be incorporated.

13. Describe responsibilities of Decommissioning Operations Contractor (DOC) Manager

Basis:

The responsibilities of the DOC Manager are not specified in the DP (as recommended by NUREG-1537, Section 2.4). In addition, the DP refers to the "DOC on-site supervisor" in Section 9.1.8. It is assumed that this section should refer to the "DOC Manager".

Path Forward (Updated):

The DP should include the responsibilities of the DOC Manager. In addition, the terminology should be clarified or corrected in the DP.

VA Response:

The description and responsibilities of all parties (VA and DOC) will be updated in the revised DP.

14. Provide description of dismantlement methods and tools

Basis:

Section 8 of the DP discusses the decommissioning activities. However, the DP does not state that standard industry dismantling and decontamination techniques will be used, and does not describe the tools to be used (e.g., wire saws, high pressure/ultra-high pressure water, needle guns, jack hammers, torches/plasma arc torches, hydraulic cutters, and hand tools). Section 2.3.1 of NUREG-1537, Part I, states that "the licensee should describe the methods, techniques, and equipment necessary to segment or otherwise dismantle components and systems."

Path Forward:

The DP should include a description of the methods, techniques and tools that will be used for the dismantlement and decontamination activities. The licensee should commit to informing the NRC in advance if new or unique technologies are to be used.

VA Response (Updated):

VA is not proposing to apply any new or unique technologies in the decommissioning of the TRIGA but will include a list of methods, techniques and tools that may be used for the dismantlement and decontamination activities in the DP revision.

15. Provide Additional Details on Waste Disposal

Basis:

The DP does not specify the classification of the wastes anticipated to be generated during D&D. Section 12.1.1 of the DP states that "Solid radioactive waste generated during the decommissioning of the AJBRF will be primarily comprised of low-level radioactive waste." Section 12.2.2 of the DP states that "Implementation procedures and the Waste Disposition Plan will provide instructions for determining the 10 CFR 61 waste classification of radioactive waste." Table 12.1 of the DP lists the components to be removed, associated waste volumes, and expected waste receiver facility. However, the DP does not specify the classification of anticipated waste volumes.

In addition, based on Table 12.1, it appears that some wastes may be sent to a landfill. However, no details are provided about the landfill or about criteria for what materials could be sent to the landfill.

Path Forward:

The licensee should provide additional information about the classification of wastes. In addition, if the licensee intends to send wastes to a landfill, the licensee should provide information about the intended landfill and criteria the licensee proposes to use to determine acceptability for disposal at the landfill.

> **VA Response (Unchanged):**
>
> The revision to the DP will include the most updated information with regards to waste characterization and disposal options. In the time since this DP was submitted some of the radiological concerns have changed (e.g. a half-life of Co-60 has transpired), and waste disposal facilities have changed (Barnwell, SC has closed and a disposal facility in Texas has opened). This will necessitate a reevaluation of desirable LLRW disposal options.

16. Include consideration of potentially contaminated outdoor areas

Basis:

Section 5.1.1 states "Exterior building surfaces do not contain concentrations of residual radioactivity concentrations. Accordingly, the need to perform dose modeling other than what already has been completed in the process of establishing DCGLs for the site is not a consideration and appears not to be warranted." A similar statement is made about surface soils in Section 5.1.2 of the DP. However, the DP has not justified that all outdoor areas can be considered non-impacted. The NRC staff believes that it may be appropriate to consider at least some part of the exterior of the building (especially the roof near any stack release points) and exterior soils (especially at the ramp that will be built for access into the basement) as potentially impacted. If some areas are potentially impacted, they must be included in final status survey plans.

Path Forward:

If exterior parts of the building and exterior soils cannot be justified as non-impacted, the DP should address these potentially impacted areas. Such areas must be considered in the final status survey plan.

VA Response (Updated):

The exterior parts of the building will be addressed in the FSS, likely classified as Class 3 survey units (according to MARSSIM). Recent soil sampling outside the basement area indicated that the surface and subsurface soils are likely non-impacted (AECOM 2011). Additionally, removable contamination samples in the ventilation system, including sample points at the system exit on the roof, indicate that the ventilation system is not contaminated above release criteria (AECOM 2011) and thus it is acceptable to conclude that roof surfaces are non-impacted. Decommissioning activities could, however, change the designation of some areas.

17. Additional items for clarification

Basis:

The NRC staff noted some items in the DP that are unclear, as follows:

a. Table 4.4 and 4.5 summarize locations and contamination levels for rooms and for systems and equipment. The last column of the Tables contains a sub-heading "(Direct Scans)". The applicability of the heading to the information presented in the column is not clear. The Table should also state the type of radioactivity the values represent (e.g., total beta/gamma).

VA Response (Unchanged):

The tables should note that the measured activity is "total beta/gamma" and will be corrected as necessary. Also, the phrase "Direct Scans" will be deleted from the applicable column heading as it does not apply to the information provided in the columns.

b. The DP contains contradictory statements in Section 7.1, page 52. The last sentence of the second paragraph states "Per the site characterization report results, the structures and components within the AJBRF contain non-detectable levels of contamination." The following paragraph states "Per the site characterization report results, most of the structures and components within the AJBRF contain very low levels of contamination."

VA Response (Unchanged):

The DP should read "Per the site characterization report results, most of the structures contain very low contamination or radiation levels not detectable above background" in both locations.

c. The discussion of the Cost Benefit Analysis in Section 7.2 of the DP requires additional clarification. This section states that "The cost of achieving the lower limits of contamination is

insignificantly greater than decommissioning to the calculated DCGL level of 27,000 dpm/100 cm2." This section does not define the "lower limits of contamination". It appears to the NRC staff that "lower limits of contamination" is meant to represent the 5000 dpm/100 cm2 "release limit" described in Section 7.1, though this is not clear in the document.

VA Response (Unchanged):

The document revisions will reflect the correct verbiage, as VA did wish to express the lower limit as 5000 dpm/100cm^2.

d. There are inconsistencies in the DP regarding references to specific organizational positions. For example, Figure 9.1 references the "DSC". However, this committee is described as the Reactor Safeguards Committee (RSC), in Section 9.1.6. Furthermore, the document describes the RSC as the "Reactor Safeguards Committee" and as the "Reactor Safety Committee".

VA Response (Unchanged):

The description and responsibilities of all parties will be updated in the revised DP.

References

AECOM. 2011. Allen J Blotcky Reactor Facility Additional Characterization Work Plan. April.

Duratek, Inc. 2003. Allen J Blotcky Reactor Facility Omaha Veterans Administration Medical Center, Omaha, Nebrasks, Characterization Survey Report. February.

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