January 3, 2012

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
	)	
Florida Power & Light Company	)	Docket Nos. 52-040-COL
	)	52-041-COL
(Turkey Point Units 6 and 7)	)	
	)	ASLBP No. 10-903-02-COL
(Combined License)	)	

FLORIDA POWER & LIGHT COMPANY'S MOTION FOR SUMMARY DISPOSITION OF CASE CONTENTION 7

I. INTRODUCTION

Pursuant to 10 C.F.R. § 2.1205, Florida Power & Light Company ("FPL") moves this Atomic Safety and Licensing Board (the "Board") for summary disposition of CASE Contention 7 on the grounds that no genuine issue of material fact exists and FPL's legal position is correct with respect to that Contention. CASE Contention 7 argues that it is reasonable to assume that FPL will be forced to rely upon on-site storage of low-level radioactive waste ("LLW" or "LLRW") due to the closure of the Barnwell LLW disposal facility to out-of-compact waste. CASE Contention 7 invokes the legal question of whether the means by which FPL intends to manage Class B and C LLW, as set forth in FPL's Combined Construction Permit and Operating License Application ("COLA") for the proposed Turkey Point Units 6 and 7 ("Units 6 and 7") is sufficient to satisfy the requirements set forth in 10 C.F.R. § 52.79(a)(3). FPL recently revised its Final Safety Analysis Report ("FSAR") to amend its LLW management plan ("Revised LLW Management Plan"). FPL moves this Board to grant summary disposition of CASE Contention 7 because FPL's Revised LLW Management Plan provides as a matter of law information sufficient to enable the Commission to reach a final conclusion regarding whether FPL's means for controlling and limiting radioactive effluents and radiation exposures will be within the limits set forth in 10 C.F.R. Part 20. FPL's Revised LLW Management Plan[1] satisfies 10 C.F.R. § 52.79(a)(3) as a matter of law. No dispute on a material fact exists. Therefore, summary disposition should be granted.

II. Procedural and Factual Background

In June, 2009, FPL submitted an application (the "Application") for a combined license ("COL") for two AP1000 pressurized water nuclear reactors to be located adjacent to the existing Turkey Point power plants, Units 1 through 5, at the Turkey Point site near Homestead, Florida. The proposed nuclear reactors would be known as Turkey Point Units 6 and 7 (the "Units 6 and 7"). The COLA incorporates by reference the information in the AP1000 Design Control Document ("DCD") codified by regulation (10 C.F.R. Part 52, App. D, § III.A) up through the recently approved amendment to Revision 19.[2] On September 4, 2009, the NRC staff ("Staff") accepted the Application for docketing. *See* 74 Fed. Reg. 51,621 (Oct. 7, 2009).

FPL's COLA, as originally filed, stated that FPL had signed a letter of intent with Studsvik, Inc., a licensed LLW treatment facility in Erwin, Tennessee to enter into negotiations for a contract under which Studsvik would accept and process, as well as take responsibility for storage and disposal of LLW produced by Turkey Point Units 6 and 7.[3] COLA Rev. 0, FSAR at 11.4-2. Citizens Allied for Safe Energy, Inc. ("CASE") filed a

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[1] To be clear, there is not a standalone document or plan entitled "Revised LLW Management Plan." That name is simply shorthand for the relevant revisions to FPL's FSAR. FPL's FSAR demonstrates compliance with Part 20 dose limitations through those revisions together with several other provisions, as discussed below. *See* Declaration of Paul Jacobs (Attachment 3) at ¶ 12.

[2] Final Rule, AP1000 Design Certification Amendment, 76 Fed. Reg. 82,079 (Dec. 30, 2011).

[3] FPL is currently shipping Class B and C LLW to Studvik for its operating nuclear plants. Declaration of Paul Jacobs at ¶ 12.

Petition to Intervene and Request for a Hearing on August 17, 2010 raising eight contentions. CASE filed a revised petition ("Revised Petition") on August 20, 2010. CASE Contention 7 argues that, with the closure of the Barnwell facility in South Carolina to out-of-compact LLW, it is reasonably foreseeable that FPL will not have off-site storage or disposal capacity for LLW generated by Units 6 and 7. Revised Petition at 43-44. CASE argues that the Studsvik plan, which relies upon ultimate disposal at the Waste Control Specialists ("WCS") facility in Texas is not sufficient because WCS is not yet licensed and, in any event, would not be allowed to import out-of-compact waste.[4] *Id.* at 42-43.

On February 28, 2011, this Board issued an order admitting, *inter alia*, CASE Contention 7. *Florida Power & Light Co.* (Turkey Point Units 6 and 7), LBP-11-06, 71 NRC __ (2011). As admitted, CASE Contention 7 reads as follows:

> FPL's COLA fails to provide information sufficient to enable the NRC to reach a final conclusion on safety matters regarding the means for controlling and limiting radioactive material and effluents and radiation exposures within the limits set forth in Part 20 and ALARA in the event FPL needs to manage Class B and Class C LLRW for an extended period.

LBP-11-06, slip op. at 112.

On Dec. 16, 2011, FPL submitted to the NRC Revision 3 of its COL Application ("COLA Rev. 3"). *See* Letter from M. Nazar to NRC Document Control Desk, "Submittal of the Annual Update of the COL Application - Revision 3" (Dec. 16, 2011). (ADAMS Accession No. ML________). Among the revisions included in COLA Rev. 3 is a

[4] There is now reason to question the underlying predicate of this contention as WCS is currently licensed as a LLW disposal site by the State of Texas. *See* Radioactive Material License No. R04100 (available at: http://www.tceq.texas.gov/permitting/radmat/licensing/wcs_license_app.html). Further, WCS is authorized by the Texas Low-Level Radioactive Waste Disposal Compact to accept and dispose of out-of-compact LLW. *See* 36 Tex. Reg. 571 (Feb. 4, 2011) (amending 31 Tex. Admin. Code. § 675.23). Finally, earlier this year, Texas Governor Rick Perry signed into law Senate Bill No. 1504, authorizing the disposal of out-of-compact LLW. SB 1504 (June 17, 2011) (Available at:
http://www.capitol.state.tx.us/BillLookup/History.aspx?LegSess=82R&Bill=SB1504).

revision to Section 11.4 of the FSAR to include the Revised LLW Management Plan (Attachment 1). The Revised LLW Management Plan maintains its reliance on the initial plan to contract with Studsvik for LLW management, storage, and disposal, but adds a contingency plan in case off-site disposal capacity is unavailable. The revised FSAR now states that "[i]f additional storage capacity for Class B and C waste were required, further temporary storage would be designed, constructed, and operated in accordance with the design guidance provided in NUREG-0800, Standard Review Plan 11.4, Appendix 11.4-A." COLA Rev. 3, FSAR at 11.4-3.

III. NRC Regulations Governing Summary Disposition

Summary disposition is appropriate where no genuine dispute exists regarding any fact material to a contention and the moving party is entitled to a decision as a matter of law.[5] 10 C.F.R. § 2.710(d)(2). The movant is required to attach to the motion a separate concise statement of material facts not in dispute. 10 C.F.R. § 2.710(a). FPL's statement of material facts not in dispute is provided herein as Attachment 2.

IV. The Board Should Grant Summary Disposition of CASE Contention 7

A. Resolution of CASE Contention 7 Requires a Legal Determination and There Are No Material Factual Issues In Dispute

As an initial matter, the Board should find that CASE Contention 7 poses a legal, rather than a factual, question as to whether FPL's LLW Management Plan satisfies 10 C.F.R. § 52.79(a)(3). In LBP-11-06, the Board identified one potential issue of fact involving CASE Contention 7 – whether FPL's letter of intent with Studsvik adequately establishes where LLW will be disposed of while maintaining compliance with Part 20. LBP-11-06, slip op. at 111-12. But in light of FPL's recent FSAR amendment, the facts

[5] In a Subpart L proceeding, the Board is required to apply the summary disposition standard set forth in Subpart G (Section 2.710(d)(2)) of the Commission's regulations. 10 C.F.R. § 2.1205(c).

alleged by CASE in Contention 7 regarding the viability of FPL's Studsvik plan, even if assumed to be true, are no longer dispositive of the matter in dispute. If FPL's initial LLW storage capacity is inadequate, its Revised LLW Management Plan identifies the means through which FPL would increase that capacity.

The question before the Board is whether FPL's FSAR, which relies upon both offsite shipment to Studsvik as well as an onsite storage contingency plan, along with several other provisions, complies with 10 C.F.R. § 52.79(a)(3). Resolution of this contention requires a legal, rather than factual, determination.[6] Therefore, there are no material facts in dispute, and the Board can grant summary disposition as a matter of law.

B. FPL's COLA Contains Information Sufficient To Satisfy The "Means" Requirement In 10 C.F.R. § 52.79(a)(3)

As pled and admitted CASE Contention 7 alleges that FPL's Application fails to meet the requirements of 10 C.F.R. § 52.79(a)(3). CASE Revised Petition at 41, 43; LBP-11-06 at 108-112. That regulation requires:

> The final safety analysis report shall include the following information, at a level of information sufficient to enable the Commission to reach a final conclusion on all safety matters that must be resolved by the Commission before issuance of a combined license:
>
> * * *
>
> (3) The kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in part 20 of this chapter.

10 C.F.R. §§ 52.79(a), (a)(3).

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[6] Attachment 3 to this Motion is a Declaration of FPL New Nuclear Project Engineering Supervisor Paul Jacobs, which addresses a number of factual issues, including the length of time available until additional storage may be needed for Class B and C LLW and the amount of time needed to construct a LLW storage facility. Nevertheless, FPL maintains that the Board need not consider factual issues to resolve CASE Contention 7. This declaration was provided in order to address reasonably anticipated counterarguments, in keeping with the expectation set forth in 10 C.F.R. § 2.323(c).

The Declaration of Diane D'Arrigo, upon which CASE Contention 7 relies, argues that FPL's COLA must include specific design information regarding potential on-site LLW storage. D'Arrigo Decl. at 10, ¶ 28. According to D'Arrigo:

> The applicant must provide greater detail about the amount of waste, its condition, the processes it will undergo, how it will be stored and where, considering the likelihood that extended onsite waste management will be necessary. Will storage be in buildings, and if so what will the structures be? If outside, exposed to the elements, how will safety and security be assured? Where will the storage area or building(s) be located? Will they be within the "protected" area? What treatment options will be carried out onsite and where? Simply referring to generic guidance documents does not substitute for responsible planning for virtually inevitable waste management needs at this specific site.

Id. While CASE might like this to be the case, the question of law before the Board is whether 10 C.F.R. § 52.79(a)(3) *requires* such design detail – or instead, whether a contingency plan that relies in part upon a well-established regulatory mechanism to incorporate a proven model for expanding onsite Class B and C LLW storage capacity is sufficient.

Neither the rule nor its regulatory history provide significant assistance in interpreting the phrase "means for controlling and limiting radioactive effluents and radiation exposures." Nevertheless, the Commission recently provided some interpretive guidance:

> We agree that the plain language of section *52.79(a)(3) does not explicitly require a description of LLRW storage for a specified duration. On its face, therefore, section 52.79(a)(3) sets no quantity or time restrictions relative to onsite storage of such waste*. Rather, it requires that a COL application contain information of first, the "kinds and quantities of materials expected to be produced" during plant operation, and second, the "means for controlling and limiting radioactive effluents and radiation exposures" to comply with Part 20 limits. In short, the rule pertains to how the COL applicant *intends, through its design, operational organization, and procedures, to comply with relevant substantive radiation protection requirements in 10 C.F.R. Part 20.* This includes, but is not limited to, low-level radioactive waste handling and storage.

Southern Nuclear Generating Company (Vogtle Electric Generating Plant, Units 3 and 4), CLI-09-16, 70 NRC 33, 37 (2009) (footnotes omitted and emphases added).[7] As a result, "the required information [to meet 52.79(a)(3)] is tied to the COL applicant's particular plans for compliance through design, operational organization, and procedures." *Id.* The Commission later quoted this sentence in the *Levy County* proceeding and concluded that an Applicant "must address, in its COL application, how it *intends to handle* an accumulation of LLRW." *Progress Energy Florida, Inc.* (Levy County Nuclear Power Plant, Units 1 and 2), CLI-10-02, 71 NRC __ (slip op. at 24-25) (2010) (emphasis added).

The *Vogtle* Licensing Board was first to interpret this Commission language. *See Southern Nuclear Generating Company* (Vogtle Electric Generating Plant, Units 3 and 4), LBP-10-08, 71 NRC __ (slip op. at 13-14) (May 19, 2010). That Board held that nothing in 10 C.F.R. § 52.79(a)(3) or the two Commission decisions interpreting that rule indicate that it requires detailed design, location, and health impacts information. *Id.* at 13. The *Vogtle* Board compared the requirements of section 52.79(a)(3) to those of 52.79(a)(4), and found that, unlike 52.79(a)(4), "section 52.79(a)(3) does not list 'principal design criteria,' 'design bases,' or '[i]nformation relative to materials of construction, arrangement, and dimensions' as items that must be discussed in the FSAR." *Id.* (Comparing 10 C.F.R. § 52.79(a)(3) with *id.* § 52.79(a)(4)(i)-(iii)).

The *Vogtle* Board also noted the Commission's language from CLI-09-16, which it reiterated in *Levy County* (CLI-10-02), regarding design, organization, and procedures, "seems merely to have been stating that the information required under section 52.79(a)(3)

[7] This Board has already ruled that the first requirement of 10 C.F.R. § 52.79(a)(3) is not a part of CASE Contention 7 because FPL referenced the AP1000 DCD, which describes the "the kinds and quantities of radioactive materials expected to be produced in the operation" of Units 6 and 7. LBP-11-06, slip op. at 110, n. 112.

is tied to the applicant's 'particular plans for compliance through,' but not necessarily the details of, 'design, operational organization, and procedures' associated with any contingent long-term LLRW facility." *Id.* (citing CLI-09-16, 70 NRC at __ (slip op. at 6)). Rejecting the *Vogtle* intervenor's claims, that Board found no requirement to include detailed design information regarding contingent LLW storage facilities. *Id.* at 14.

FPL's FSAR identifies the means by which exposures to radiation from LLW will be maintained within the dose limits of Part 20 of the NRC's regulations because it states "how it intends to handle an accumulation of LLRW." *See Levy County*, CLI-10-02, (slip op. at 24-25). As explained in FPL's FSAR, FPL intends to handle accumulated LLW by shipping it offsite and potentially employing waste minimization strategies. COLA Rev. 3, FSAR at 11.4-1. If additional onsite storage capacity for Class B and Class C LLW is required because sufficient off-site storage or disposal capacity is unavailable, FPL will expand the capacity of its licensed storage facilities, consistent with existing NRC guidance. *Id.*; *see also id.* at 11.4-3. This additional onsite storage would be designed and built utilizing the design guidance provided in NUREG-0800, Standard Review Plan Chapter 11 Radioactive Waste Management Appendix 11.4-A, Design Guidance for Temporary Storage of Low-Level Radioactive Waste. (Attachment 4).[8] *Id.*

FPL will be able to utilize the NRC's existing regulatory framework, described in NRC Regulatory Issue Summary 2008-32, "Interim Low-Level Radioactive Waste Storage at Reactor Sites," to conduct written safety analyses under 10 C.F.R. § 50.59. RIS 2008-32

[8] FPL's plan to ship LLW offsite and rely on onsite LLW storage only as a contingency plan is the precise course of action called for in NRC's guidance documents: "While it may be prudent and/or necessary to establish additional onsite storage capability, waste should not be placed in contingency storage if it can be disposed at a licensed disposal site." NUREG-0800, Appendix 11.4-A at 11.4-25. *See also* RIS 2008-32 at 2 (outlining NRC's policy preference of avoiding onsite activities not directly related to the licensed operation of a reactor).

at 2 (Attachment 5). These written safety analyses allow a licensee to "make changes in the facility as described in the final safety analysis report," such as expanding the capacity of the LLW storage facility already described in the FSAR, without a license amendment if certain conditions are satisfied. 10 C.F.R. § 50.59(c)(1). If the conditions of 10 C.F.R. § 50.59 are not met, FPL would still be able to add on-site storage capacity by seeking an amendment to its COL. RIS 2008-32 at 3.

A split Board in the *Levy County* COL proceeding denied Progress Energy Florida's ("PEF") original motion for summary disposition based upon its similar LLW management plan. *Progress Energy Florida, Inc.* (Levy County Nuclear Power Plant, Units 1 and 2), LBP-10-20, 72 NRC __ (Nov. 28, 2010) ("Levy County I"). Reading the Commission's "design, operational organization, and procedures" language, the *Levy County I* Board concluded that PEF's plan (to rely upon § 50.59 analysis or a license amendment to build onsite storage, if necessary) offered only procedures. *Id.* at 34-35. The *Levy County I* Board ruled that a plan cannot consist merely of procedures or a promise to develop additional onsite storage. *Id.* at 35. What the *Levy County I* majority failed to recognize was that PEF's means for limiting radiation exposures within the limits set forth in part 20 included more than the specific contingency plan for adding on-site LLW storage capacity and that the NRC's review of an applicant's compliance with § 52.79(a)(3) must address the full scope of the FSAR.

Here, FPL's plan includes both design and operational organization, in addition to its reliance on licensing procedures such as § 50.59 in the recent FSAR revisions (the Revised LLW Management Plan). The plan includes specific commitments regarding the kinds and quantities of waste (DCD § 11.4.2.1 at 11.4-3 to 11.4-6), the design of the storage containers (DCD § 11.4.1.3 at 11.4-3), and the methods for processing and packaging the LLW (DCD § 11.4.2.3.3 at 11.-10 to 11.4-11).[9] The FSAR also relies on FPL's operational organization, most notably the Radiation Protection Plan, provided in FSAR Chapter 12, and the Radiation Protection Plan Program Description, in FSAR Appendix 12AA, which will ensure that any contingency planning decisions involving the handling of accumulated LLW are made with full awareness of the implications for Part 20 compliance. Finally, if the LLW cannot be sent offsite for storage or disposal, FPL commits to construct additional storage capacity utilizing the guidance in NUREG-0800, Appendix 11.4-A. This NRC document provides guidance on how to design temporary storage facilities for many different types of LLW. Contrary to the *Levy County* Board's ruling, the PEF plan was not, and FPL's plan is not, merely procedural – only a single contingent portion is.

In any event, as the *Levy County I* dissent explained, a "means" either consists of or includes a "method" or "strategy" for achieving an end. LBP-10-20, slip op. at D3 (citing *inter alia*, American Heritage Dictionary of the English Language (4th ed. 2010) ("A method, a course of action, or an instrument by which an act can be accomplished or an end achieved")). Therefore, the "means" for ensuring compliance with Part 20 can include reliance upon a regulatory procedure, such as 10 C.F.R. § 50.59, and the guidance offered in NUREG-0800, Appendix 11.4-A, as does FPL's Revised LLW Management Plan.

The *Levy County I* majority was also concerned with whether PEF's LLW contingency plan contained enforceable commitments.[10] *See* LBP-10-20, slip op, at 25-29

[9] A copy of DCD Section 11.4 is provided as Attachment 6.

[10] In fact, the *Levy County* Board ultimately granted PEF's motion for summary disposition based upon a more detailed plan that it found to contain "enforceable commitments." *Progress Energy Florida, Inc.* (Levy

and 25 n.28. But the *Levy County I* Board's enforceability concerns were unfounded. That Board seemed to assume that in a situation where the applicant's then-existing LLW storage facilities prove insufficient, the applicant may simply take no action and allow LLW to accumulate uncontrolled to such a degree as to violate the NRC's Part 20 dose standards, so long as there is not a specific FSAR commitment regarding the design of onsite LLW storage.[11] But with good reason, the NRC "does not presume that a licensee will violate agency regulations wherever the opportunity arises.'" LBP-10-20, Dissent at 8-9 (citing *Private Fuel Storage, L.L.C.* (Independent Spent Fuel Storage Installation), CLI-01-9, 53 NRC 232, 235 (2001)).

An applicant for an NRC license "is not obliged to meet an absolute standard but to provide 'reasonable assurance' that public health, safety and environmental concerns were protected, and to demonstrate that assurance 'by a preponderance of the evidence.'" *AmerGen Energy Co., LLC* (Oyster Creek Nuclear Generating Station), CLI-09-7, 69 NRC 235, 262 n.142 (2009) (citing *Commonwealth Edison Co.* (Zion Station, Units 1 and 2), ALAB-616, 12 NRC 419, 421 (1980)). The finding of the *Vogtle* Board and of the *Levy County I* dissent that the "means" requirement of 10 C.F.R. § 52.79(a)(3) does not necessitate the inclusion of detailed design information for a merely contingent LLW storage facility is fully consistent with the NRC's reasonable assurance standard. *Vogtle*, LBP-10-20 (slip op. at 15); *Levy County I,* LBP-11-20 (slip op. at D-4).

County Nuclear Power Plant, Units 1 and 2), LBP-11-31, 73 NRC __ (slip op. at 4, 7, 9) (Nov. 4, 2011) ("Levy County II").

[11] Moreover, the *Levy County* Board's reasoning further implies that the NRC would have no available enforcement mechanism to prevent such an unrealistic scenario. The NRC would absolutely have a legal basis to take enforcement action if onsite storage proves inadequate for Part 20 compliance and yet FPL did not follow its commitment to either ship the LLW to a licensed offsite storage facility or construct an onsite facility while adhering to the principles in NUREG-0800, Ch. 11.4, App. 11.4-A. *See* COLA Rev. 3, FSAR at 11.4-3.

Accordingly, FPL has identified the "means," by which FPL will control or limit radioactive effluents and radiation exposures in the absence of access to an offsite disposal facility to store Class B and Class C LLW. FPL's Revised LLW Management Plan is consistent with NRC regulations and guidance. Together with the other significant portions of the DCD and FSAR, the Revised LLW Management Plan provides the Commission with adequate assurance that radiation exposures from LLW stored onsite at Turkey Point Units 6 and 7 will at all times be within the limits set forth in 10 C.F.R. Part 20.

V. Conclusion

For the reasons stated above, the Board should grant FPL's Motion for Summary Disposition of CASE Contention 7.

VI. Certification

In accordance with 10 C.F.R. §2.323(b), counsel for FPL has made a sincere effort to contact the other parties in this proceeding to resolve the issue raised in this motion but has not been successful. In particular, CASE has indicated that it will oppose the Motion.[12]

Respectfully submitted,

/Signed electronically by Steven Hamrick/

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Counsel for FLORIDA POWER & LIGHT COMPANY

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12 Both CASE and the NRC Staff have agreed not to oppose the motion on the grounds of timeliness if it is filed on January 3, 2012, in order to accommodate holiday schedules.

ATTACHMENT 1

11.4 SOLID WASTE MANAGEMENT

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

11.4.2 SYSTEM DESCRIPTION

Add the following information after DCD Subsection 11.4.2.4.2:

11.4.2.4.3 Contingency Plans for Temporary Storage of Low-Level Radioactive Waste (LLW)

PTN SUP 11.4-2 In the event that offsite shipping of radwaste is not available when Units 6 & 7 become operational, temporary storage capability is available on site for greater than two years at the expected rate of radwaste generation and greater than one year at the maximum rate of radwaste generation, as described in DCD Subsection 11.4.2.4.2 paragraph ten. Implementation of waste minimization strategies could extend the duration of temporary radwaste storage capability.

If additional onsite radwaste storage capability were required, then onsite facilities would be designed, constructed, and operated in accordance with the design guidance provided in NUREG-0800, Standard Review Plan Chapter 11 Radioactive Waste Management Appendix 11.4-A, Design Guidance for Temporary Storage of Low-Level Radioactive Waste.

11.4.5 QUALITY ASSURANCE

Add the following information to the end of DCD Subsection 11.4.5:

STD SUP 11.4-1 Since the impact of radwaste systems on safety is limited, the extent of control required by Appendix B to 10 CFR Part 50 is similarly limited. Thus, a supplemental quality assurance program applicable to design, construction, installation and testing provisions of the solid radwaste system is established by procedures that complies with the guidance presented in Regulatory Guide 1.143.

11.4.6 COMBINED LICENSE INFORMATION FOR SOLID WASTE MANAGEMENT SYSTEM PROCESS CONTROL PROGRAM

Add the following information to the end of DCD Subsection 11.4.6.

This COL Item is addressed below.

STD COL 11.4-1 A Process Control Program (PCP) is developed and implemented in accordance with the recommendations and guidance of NEI 07-10A (Reference 201). The PCP describes the administrative and operational controls used for the solidification of liquid or wet solid waste and the dewatering of wet solid waste. Its purpose is to provide the necessary controls such that the final disposal waste product meets applicable federal regulations (10 CFR Parts 20, 50, 61, 71, and 49 CFR Part 173), state regulations, and disposal site waste form requirements for burial at a low level waste (LLW) disposal site that is licensed in accordance with 10 CFR Part 61.

Waste processing (solidification or dewatering) equipment and services may be provided by the plant or by third-party vendors. Each process used meets the applicable requirements of the PCP.

No additional onsite radwaste storage is required beyond that described in the DCD.

Table 13.4-201 provides milestones for PCP implementation.

PTN SUP 11.4-1 Low-level radioactive waste is packaged to meet transportation and disposal site acceptance requirements. Packaging of waste for offsite shipment complies with applicable DOT (49 CFR Parts 173 and 178) and NRC regulations (10 CFR Part 71) for transportation of radioactive material. The packaged waste is stored on site on an interim basis before being shipped offsite to a licensed processing, storage, or disposal facility. Onsite storage for more than a year at the maximum rate of generation is provided in the waste accumulation room of the radwaste building. Radioactive waste is shipped offsite by truck.

Consistent with current commercial agreements, a third-party contractor processes, stores, owns, and ultimately disposes of low-level waste generated as a result of operations. Activities associated with the transportation, processing, and ultimate disposal of low-level waste comply with applicable laws and regulations in order to ensure the public's health and safety. In particular, the third-party contractor conducts its operations consistent with NRC regulations (e.g., 10 CFR Part 20).

Under 10 CFR 20.2001, reactor licensees may transfer low-level radioactive waste material to another licensee that is specifically licensed to accept and treat waste prior to disposal. Studsvik, Inc., has a licensed low-level radioactive waste treatment facility in Erwin, Tennessee. FPL has signed a letter of intent with Studsvik to enter into negotiations for a contract for the performance of work by Studsvik to include the shipment, processing, storage, and disposal of low-level radioactive waste produced by Units 6 & 7 (Reference 205). Under the proposed contract, Studsvik would treat the Class B and C waste at its Erwin, Tennessee facility and thereafter take responsibility for storage and final disposal.

All packaged and stored radwaste is shipped to offsite disposal/storage facilities and temporary storage of radwaste is only provided until routine offsite shipping can be performed. Accordingly, there is no expected need for permanent onsite storage facilities at Units 6 & 7.

If additional storage capacity for Class B and C waste were required, further temporary storage would be designed, constructed, and operated in accordance with the design guidance provided in NUREG-0800, Standard Review Plan 11.4, Appendix 11.4-A. The change to the facility to provide additional onsite storage would be evaluated by performing written safety analyses in accordance with 10 CFR 50.59. If the acceptability of the proposed additional storage could not be demonstrated by 10 CFR 50.59 analyses, a license amendment would be sought to approve the proposed storage.

11.4.6.1 Procedures

STD SUP 11.4-1 Operating procedures specify the processes to be followed to ship waste that complies with the waste acceptance criteria (WAC) of the disposal site, 10 CFR 61.55 and 61.56, and the requirements of third party waste processors.

Each waste stream process is controlled by procedures that specify the process for packaging, shipment, material properties, destination (for disposal or further processing), testing to verify compliance, the process to address non-conforming materials, and required documentation.

Where materials are to be disposed of as non-radioactive waste (as described in DCD Subsection 11.4.2.3.3), final measurements of each package are performed to verify there has not been an accumulation of licensed material resulting from a buildup of multiple, non-detectable quantities. These measurements are obtained using sensitive scintillation detectors, or instruments of equal sensitivity, in a low-background area.

Procedures document maintenance activities, spill abatement, upset condition recovery, and training.

Procedures document the periodic review and revision, as necessary, of the PCP based on changes to the disposal site, WAC regulations, and third party PCPs.

11.4.6.2 Third Party Vendors

Third party equipment suppliers and/or waste processors are required to supply approved PCPs. Third party vendor PCPs describe compliance with Regulatory Guide 1.143, Generic Letter 80-09, and Generic Letter 81-39. Third party vendor PCPs are referenced appropriately in the plant PCP before commencement of waste processing.

11.4.7 REFERENCES

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201. NEI 07-10A, *Generic FSAR Template Guidance for Process Control Program (PCP)*, Revision 0, March 2009 (ML091460627).

202. *Florida Annual Statistical Bulletin 2008*, National Agricultural Statistics Service. Available at http://www.nass.usda.gov/Statistics_by_State/Florida/Publications/Annual_Statistical_Bulletin/fasd08p.htm.

203. *Commercial Red Meat: Production, by State and U.S.*, U.S. Department of Agriculture, National Agricultural Statistics Bulletin, p. 102. Available at http://www.nass.usda.gov/Statistics_by_State/Iowa/Publications/Annual_Statistical_Bulletin/2007/07_102.pdf.

204. *2002 Census of Agriculture*, Florida State and County Data, Volume 1, U.S. Department of Agriculture, June 2004. Available at www.nass.usda.gov/census/census02/volume1/fl/FLVolume104.pdf.

205. FPL 2009. Florida Power & Light Company. Letter of Intent Between Florida Power & Light Company and Studsvik, Inc., May 22, 2009.

ATTACHMENT 2

Statement of Material Facts on Which No Genuine Dispute Exists

FPL submits, in support of its Motion for Summary Disposition of Contention 7, this Statement of Material Facts as to which FPL contends there is no genuine dispute to be heard.

1. On June 30, 2009, FPL submitted a Combined Construction Permit and Operating License Application ("COLA") for two AP1000 units at its Turkey Point site ("Turkey Point Units 6 and 7"). The COLA Part 2 (FSAR), Section 11.4 incorporates by reference and supplements those sections of the AP1000 Design Control Document that are cited in FPL's Motion for Summary Disposition of Contention 7.

2. On August 20, 2010, CASE filed its revised Petition to Intervene and Request for Hearing ("Revised Petition"), which included Contention 7 alleging:

> FPL's application (FSAR Chapter 11, section 4.6) is inadequate because the Safety Analysis Report assumes that the Class B and C so-called "low-level" radioactive waste generated by the proposed Turkey Point Units 6 & 7 will be promptly (e.g. in approximately 2 years per the AP1000 DCD: page 11.4-6) shipped offsite despite lack access for disposal. The FSAR fails to address compliance with Part 20 and Part 50 Appendix I (ALARA) in the event that PEF will need to manage such waste on the Turkey Point Site for a more extended period of time, possibly its entire licensed operating period or longer.

Revised Petition at 41.

3. In its Order of February 28, 2011, the Board narrowed and admitted Contention 7 as follows:

> FPL's COLA fails to provide information sufficient to enable the NRC to reach a final conclusion on safety matters regarding the means for controlling and limiting radioactive material and effluents and radiation exposures within the limits set forth in Part 20 and ALARA in the event FPL needs to manage Class B and Class C LLRW for an extended period.

Florida Power & Light Company (Turkey Point Units 6 and 7), LBP-11-06, 71 NRC __ (2011) (slip op. at 112).

4. On December 16, 2011, FPL submitted Revision 3 to its COLA, which included revisions to Section 11.4 of its FSAR. The revised Section 11.4 provides FPL's plan, if needed, for controlling exposures from storage of an extended accumulation of LLRW.

ATTACHMENT 3

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
	)	
Florida Power & Light Company	)	Docket Nos. 52-040-COL
	)	52-041-COL
(Turkey Point Units 6 and 7)	)	
	)	ASLBP No. 10-903-02-COL
(Combined License)	)	

DECLARATION OF PAUL R. JACOBS IN SUPPORT OF FLORIDA POWER & LIGHT COMPANY'S MOTION FOR SUMMARY DISPOSITION OF CASE CONTENTION 7

PAUL R. JACOBS who, after being duly sworn, states as follows:

1. I am the New Nuclear Project Engineering Supervisor for Florida Power & Light Company (FPL)'s Turkey Point Units 6 & 7 nuclear power plant project.

2. My professional and educational experience is summarized in the curriculum vitae attached as Exhibit 1 to this declaration. I hold a Bachelors of Science Degree in Nuclear Engineering from the State University of New York, Maritime College. I have a Professional Engineers License in Nuclear Engineering from the State of California (NU1334).

3. In my capacity as New Nuclear Project Engineering Supervisor, I am responsible for preparing engineering studies and reviewing engineering documents in support of federal and state applications, including proper planning of low level radioactive waste (LLRW) storage. I have reviewed the Final Safety Analysis Report (FSAR) amendment to FPL's Combined License Application for Turkey Point Units 6 and 7 (COLA) submitted by FPL on December 16, 2011, and provided

advice and input in its preparation. Specifically, I am knowledgeable of, and provided advice and input on, the revised contingency plan for LLRW storage for Turkey Point Units 6 and 7.

4. I am familiar with CASE Contention 7, which was raised in the NRC licensing proceeding for Turkey Point Units 6 & 7. As admitted into the proceeding by the Atomic Safety and Licensing Board, CASE Contention 7 asserts that FPL's Application fails to provide information sufficient to enable the NRC to reach a final conclusion on safety matters regarding the means for controlling and limiting radioactive material and effluents and radiation exposures within the limits set forth in Part 20 and ALARA in the event FPL needs to manage Class B and Class C LLRW for an extended period.

5. My declaration addresses the timeframe by which FPL can implement its contingency plan for providing additional Class B and C LLRW storage on-site beyond two years, if needed.

6. FPL's LLRW contingency plan can be successfully implemented if required. Almost all of the Class B and C LLRW is generated during planned outages. There is adequate information from plant chemistry monitoring to anticipate the amount of Class B and C LLRW that will be generated during an outage and will need to be stored.

7. The Class B and C LLRW is primarily generated from purification media discharges and these occur during planned outages. Westinghouse AP1000 Design Control Document (DCD) § 11.4.2. The media discharges are first held in a catch tank in the Auxiliary Building, then processed into storage containers. While the DCD provides

the average annual generation rate for LLRW, Class B and C LLRW will likely be generated almost entirely in batches at an outage (expected to be at eighteen month intervals).

8. The expected annual volume and isotopic influent activity of the spent resin and filter cartridge wastes described in DCD Tables 11.4-1 and 11.4-2 account for almost all of the potential to generate Class B and C LLRW at Turkey Point Units 6 & 7. The estimated maximum annual activity is described in DCD Table 11.4-3. The AP1000 plant design has sufficient storage capacity to accommodate the maximum generation rate of Class B and C LLRW.

9. When Turkey Point Units 6 & 7 start operation, the first media discharge will be to spent resin storage tanks in the rail car bay of the Auxiliary Building. DCD § 11.4.2. Depending on operating performance, the first media discharge will be months, if not years, after the plant starts operating.

10. In planning for subsequent media discharges, FPL will determine whether there is adequate capacity in the spent resin storage tanks for Class B and C LLRW. DCD § 11.4.2.3.1. The installed capacity may be supplemented by additional temporary mobile systems, if needed. DCD § 11.4.1.3.

11. When sufficient spent resin has accumulated, the spent resin is processed (primarily by dewatering) and placed into storage/shipping containers. DCD § 11.4.2.3.1. There is adequate space in the Auxiliary Building to store the expected generation of spent resin for one year. DCD § 11.4.2.1. Because higher content and lower radioactive concentrations are mixed as they are accumulated, not every storage/shipping container

of resin will be classified as Class B or C LLRW. Those containers that can be shipped for disposal or off-site storage will be shipped.

12. FPL's plan for controlling and limiting radioactive material and effluents and radiation exposures from Class B and C LLRW is found in Section 11.4 of its FSAR, "Solid Waste Management," which incorporates by reference the corresponding section of Revision 19 to the DCD. This includes specific commitments regarding the kinds and quantities of waste (DCD § 11.4.2.1 at 11.4-3 to 11.4-6), the design of storage containers (DCD § 11.4.2.1 at 11.4-4, and how the waste will be processed and packaged (DCD § 11.4.2.3.3 at 11.4-10 to 11.4-11). It also includes FPL's stated plan to transfer Class B and C LLRW to Studsvik for treatment, storage, and ultimate disposal, as FPL is doing currently for its existing operating nuclear plants. FSAR § 11.4.6 at 11.4-2. It also includes FPL's contingency plan in the event additional onsite storage capacity for Class B and C waste is required. In that case, FPL's FSAR states that additional temporary storage "would be designed, constructed, and operated in accordance with the design guidance provided in NUREG-0800, Standard Review Plan 11.4, Appendix 11.4-A." FSAR § 11.4.2.4.3 at 11.4-1.

13. In planning for each media discharge, FPL will evaluate the capacity in the tanks, temporary mobile systems, and storage containers to determine whether additional storage of Class B and C LLRW will be needed. Because the DCD provides for enough storage for at least two media discharges to be stored in tanks and shipping containers in the Auxiliary Building, it will be the third outage involving media discharge before even additional temporary storage could potentially be needed (about four and a half years). Therefore, FPL will have sufficient time after Turkey Point Units 6 & 7 start

operating to complete all activities to construct additional storage, if needed, as called for in its contingency plan – at least two refueling outages (about three years).

14. The LLRW storage facility can be constructed within six months. Therefore, an additional storage facility could be constructed prior to the third outage requiring media discharge, even if work is not started until after completing the second outage (about 36 months from the start of operations).

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

Signed on December 27, 2011

State of Florida

ss.

County of Palm Beach

MARIE S. WICKS
Commission DD 810190
Expires November 15, 2012
Bonded Thru Troy Fain Insurance 800-385-7019

FPL

Paul R. Jacobs

EDUCATION
B.S. in Nuclear Engineering, State University of New York, Maritime College, New York
Graduate Courses toward Masters Degree, Nuclear Engineering, New York University

LICENSES & CERTIFICATIONS
Professional Engineer - State of California
U.S. Coast Guard - Third Assistant Engineer

SUMMARY OF QUALIFICATIONS
Forty five years of experience in the power generation industry as a design engineer, consulting engineer, and independent business owner.

PROFESSIONAL EXPERIENCE

Florida Power & Light Company May 2006-Present
New Nuclear Project - Engineering Supervisor
Performed initial reviews of all proposed reactor types being considered for the new nuclear project contemplated by FPL.
Engineering lead for preparation of licensing documents submitted in support of obtaining federal and state permits including final safety analysis report (FSAR) and environmental report (ER) submitted to the Nuclear Regulatory Commission (NRC) and the Site Certification Application submitted to the State of Florida. Provides engineering studies supporting federal and state applications. Continuing support for responses requests for additional information from the NRC and responses to questions from state and local agencies.

Florida Power & Light Company - Turkey Point Nuclear Plant January 2006-May 2006
Procedure writer
As a member of the Life Cycle Management Team, provided assistance to the Maintenance Department, Instrumentation and Control, in support of design modification to the feedwater control system. Specific activities included review of the plant design modification for replacement of the feedwater control valve (main and bypass) control system with digital positioners, revision of all maintenance procedures to incorporate required changes, and development of new procedure for calibration of the positioners and position sensors using hand held communicator or valve link software.

Developed new procedure for performing dynamic testing of the feedwater control system to optimize feedwater control system parameters.

Completed procedure upgrades for conversion of Copes Vulcan valve actuators from D100 to D1000. Developed new procedure for the inspection and overhaul of Anchor Darling double disc gate valves. Provided general support for the revision of maintenance procedures to incorporate feedback and corrective actions. Reviewed and provided input to plant modification for the Feedwater Pump Recirculation Flow Transmitter Replacement project.

Entergy - Indian Point Nuclear Plant, Units 2 & 3 January 2003-November 2005
I&C Power Uprate Project Engineer
Provided assistance to the Indian Point 2 and 3 Maintenance Instrument and Control Department in the preparation, technical review and revision of Technical Specification Surveillance Procedures, Technical Requirement Manual and Offsite Dose Calculation Manual Surveillance Procedures and Instrument Calibration Procedures. During this assignment, Indian Point 2 and 3 implemented a power uprate program. Assigned as the Instrument and Control representative to interface with Power Uprate Project Management and Design Engineering Department to evaluate the required instrumentation hardware and software changes. Responsible for revising the Surveillance and Calibration Procedure for the required uprate changes.

Performed a review of Surveillance Test procedures as part of the Design Basis Initiatives Project. The project objective was to determine if the I&C surveillance procedures were in compliance with design basis requirements. The procedure review included a comprehensive review of the purpose statements, conduct of the test and test acceptance criteria to assure a well defined and documented basis. The review also included the impact of the newly implemented Improved Technical Specifications on the surveillance test performance.

Energy and Environmental Management Corporation 1985-2000
President

Company provided services to the utility industry and to the energy conservation market. Was directly involved in providing services to the nuclear utility industry and was responsible for several major projects as described below.

Indian Point Nuclear Plant, Unit 2 June 2001-December 2002
Inservice Test Engineer
Responsible for performing Inservice Testing (IST) Program related tasks.

- Review, revision and issuance of ASME pump and valve surveillance tests.
- Issuance of quarterly, cold shutdown and refueling valve tests.
- Review and evaluation of valve and pump tests for compliance with ASME Section XI and Technical Specification requirements. Perform pump and valve analyses and prepare 96 hour evaluations.
- Review of Post Maintenance Tests and establishment of reference values.
- Maintain B&C Leak Monitoring System Running Total.
- Maintain External Recirculation Leakage Running Total.
- Incorporate plant changes into IST Basis Document and prepare IST Program for submittal to the NRC.
- Maintain IST Augmented Program.
- Maintain 10CFR50 Appendix J administrative and test procedures and database.
- Provided outage related engineering support for testing of valves and pumps during cold shutdown and refueling.

Indian Point Nuclear Plant, Unit 2 July 1998-June 2001
IST Assessment Engineer
Led a multi-disciplined team that performed a comprehensive re-evaluation of the existing Inservice Test Program (IST) Program at Indian Point 2. The review encompassed all valves and pumps in ASME and non-ASME support systems that provide safety related functions to establish the basis for including or excluding components from the IST Program. The review resulted in the development of an IST Basis Document and a revision to the IST Program that was approved by the NRC without comment.

The effort also included the development of a computerized IST Program Database to capture design and licensing basis information for each component covered by the review team. The database maintains surveillance test information for pumps and valves and is used for trending and analysis of component performance.

Indian Point Nuclear Plant, Unit 2 July 1997-June 1998
Project Engineer
Performed a review of the Indian Point 2 Snubber Program, including a review of all historical records for the visual and functional testing of all snubbers, development of a management data base, revision of all snubber procedures, and resolution of outstanding Quality Assurance Audit items.

Indian Point Nuclear Plant, Unit 3 January 1997-June 1997
Design Basis Engineer
Part of the design engineering group performing design basis and licensing analysis to determine the safety function and safety classification of mechanical and electrical components.

J.A. Fitzpatrick Nuclear Plant November 1996-December 1996
Design Engineer
Assigned to the J.A. FitzPatrick plant as part of a special design-engineering group assembled to assist station management in the performance outage related and general station activities. Included were preparation of nuclear safety evaluations, preparation of design calculations, and review of engineering documents prepared by discipline engineers.

Indian Point Nuclear Plant, Unit 2 July 1996-November 1996
Safety & Licensing Engineer
Provided assistance to the Indian Point 2 Nuclear Safety and Licensing Department including NRC interface, preparation of 10CFR50.59 analyses, Licensee Event Reports preparation, resolution of outstanding technical and licensing issues, review and revision to department procedures, and response to Quality Assurance audits.

Energy & Environmental Management Corp. January 1996-June 1996
Energy Conservation Business Development

Participated in the development of the energy conservation business for Energy and Environmental Management Corporation. The company signed and implemented a marketing and engineering agreement with a major Northeast utility to provide energy conservation services. The company provided services to private, municipal and governmental clients in New York State and other states in the Northeast.

Indian Point Nuclear Plant, Unit 2 June 1991-December 1995
Project Engineer
Was a member of the team performing the Component Declassification Evaluation Project for Con Edison's Indian Point Unit No. 2 Nuclear Power Station. This project involved the evaluation of over five hundred selected safety related components, parts and commodities to determine if the item could be declassified. This effort involved a detailed review of plant systems, operating and emergency procedure and the review of Commercial Grade Dedication packages.

Supervised Service Water Pump operation and maintenance review and an analysis of the lubrication requirements for several thousand components.

Acted as the Project Coordinator for design engineering projects being performed for NYPA Indian Point 3 associated with Cataract's off-site engineering support contract. Mr. Jacobs was also involved in the preparation of dedication packages to support the 1992 refueling Outage.

Indian Point Nuclear Plant June 1988-June 1990
Led a multi disciplined team responsible for performing an analysis of all safety and safety-related systems to identify the required operation in response to various plant events. The purpose of the analysis was to evaluate the effect of the failure of various mechanical components on the ability of the system to perform its intended function. The program resulted in recommendations to management regarding the inclusion of components in the inservice testing program.

Susquehanna Nuclear Station February 1986-September 1986
EAL Project Engineer
Member of the project team that prepared the Emergency Action Level (EALs) for PP&L's Susquehanna Station, including . a complete rewrite of existing EALs and a detailed review of operating and emergency procedures.

Comanche Peak Nuclear Power Plant November 1985-January 1986
Assistant Chief Engineer
Involved in the Design Adequacy Program for Comanche Peak Nuclear Power Plant. Responsibilities included review of system design against applicable design criteria, including design documentation, NSSS specifications, design drawings, NRC regulations, single failure criteria, pipe rupture, etc.

Impell Corporation 1973-1985
Vice President and Northeast Region Manager; Manager, Systems Engineering and Management Services

Responsible for the overall operation, complete responsibility, and authority for the technical, administrative and financial aspects of the operation. The office had in excess of 150 engineers and clerical staff and generated over 15 million in annual revenues.

Responsible for the management of mechanical engineering, pipe support, structural design and analysis, systems engineering, design review, licensing and quality assurance projects for utility and engineering clients.

Responsible for developing a methodology for analyzing plant system response to various initiating events. The methodology, Safety Sequence Analysis, was used by utility clients to verify system design (mechanical, electrical, I&C, etc.) and to evaluate the ability of the plant to respond to pipe break events, single failure criteria (active and passive) and environmental considerations.

Supervised a staff of more than 70 professionals assigned to the Systems Engineering and ManagementServices Division; responsible for overall coordination of division and project activities including technical review of work, client liaison, division and project budget, and schedule control.

Was the Project Coordinator for a five (5) year effort at the Shoreham Nuclear Power Station during its construction. Managed a group of twenty five (25) engineers and designers involved in the layout and design of mechanical systems, large bore and small bore piping and support design and conduit and conduit support design.

Ebasco Services, Inc. 1968-1973
Principal Mechanical/Nuclear Engineer

Principal Mechanical/Nuclear Engineer for a large architect-engineering firm. As Lead Mechanical Job Engineer and Project Engineer on several large nuclear power plant projects was responsible for the preparation of detailed system designs, design and analysis of manufacturer equipment proposals for the reactor, auxiliary, and steam conversion systems.

As Project Engineer on the Chin-Shan Nuclear Power Station, responsibilities included cross-discipline coordination of licensing and engineering activities. Duties also included responsibility for work assignments and review of all mechanical work on the project.

As Mechanical Nuclear Engineer on Tsuruga Nuclear Power Station, Vermont Yankee Nuclear Power Station, and WPPSS Nuclear Project No. 3, duties included total technical responsibility for the design of safety related engineering activities.

Assignments also included preparation and review of the Preliminary Safety Analysis Report, ER, and State Application for the WPPSS Nuclear Project. During the early WPPSS Project stages, assignments included NSSS evaluations, site studies and conceptual design.

Military Sea Transportation Service 1966-1968
Coast Guard Licensed Third Assistant Engineer
Performed duties of a Third Assistant Engineer on a United States merchant vessel. Responsible for the operation of the main steam boilers and propulsion systems and ship auxiliaries. Performed maintenance activities on ships engineer room equipment.

PROFESSIONAL AFFILIATIONS
American Nuclear Society
American Society of Mechanical Engineers

ATTACHMENT 4

NUREG-0800

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

11.4 SOLID WASTE MANAGEMENT SYSTEM

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of effectiveness of radwaste systems.

Secondary - Organizations responsible for the review of (1) radwaste system design and performance, and (2) solid waste materials.

I. <u>AREAS OF REVIEW</u>

The solid waste management system (SWMS) manages radioactive wastes, as liquid, wet, and dry solid wastes, produced during normal operation and anticipated operational occurrences. Review of the SWMS includes design features that are necessary for collecting, handling, processing, and storing wastes. This encompasses the design, design objectives, design criteria, treatment methods, and expected releases, including the description of the SWMS, mobile equipment connected to permanently installed systems, piping and instrumentation diagrams (P&IDs), process and effluent radiation monitoring and control instrumentation, and process flow diagrams showing the operational methods and factors that influence waste treatment. The review includes an evaluation of any additional equipment that may be necessary to process liquid, dry, and wet wastes and route them to the point of discharge from the SWMS or to prepare them for shipment to authorized offsite disposal sites or licensed radioactive waste processors.

The specific areas of review are as follows:

1. Design objectives in terms of expected and design volumes of liquid and wet wastes to be handled and processed (e.g., sludge, resins, filters, process concentrates, and

charcoal) and dry solid wastes and materials (e.g., high-efficiency particulate air (HEPA) filters, contaminated tools, equipment, plastics, glass, metals, rags, paper, and clothing), including expected radionuclide distributions and concentrations, chemicals, and mixed wastes (characterized by the presence of hazardous chemicals and radioactive materials). Expected waste volumes and radioactivity inventories to be shipped for disposal, shipped to waste processors for treatment and disposal, and returned to the radwaste system for further treatment or reuse.

2. Description of the SWMS; P&IDs; process and effluent radiation monitoring and control instrumentation; and process flow diagrams showing the methods of operation, including equipment design capacities, interconnections between plant subsystems (e.g., ventilation, service water, equipment drains) and mobile processing equipment, alternate processing methods, principal parameters assumed in the SWMS design and operation, and the use of such information for the development of the process control program (PCP).

3. Special design features and operational procedures to prevent, control, and collect releases of radioactive materials resulting from overflows from tanks containing liquids, sludge, spent resins, and the like, and measures to prevent the dropping of containers from cranes and forklifts. Corrosion-resistant properties of all system piping and valves associated with transfer lines to storage tanks and discharge piping buried in soils and concrete, including features designed for the early detection of leaks and spills (e.g., leak detection sumps and wells). Provisions and effectiveness of physical and monitoring precautions taken to minimize spills and leaks (e.g., retention basins, curbing, level gauges and alarms, catch containment, and self-sealing quick-disconnects) and measures to prevent interconnections with nonradioactive systems. Provisions for processing radioactive materials associated with the decontamination of leaks and spills and remediation of uncontrolled and unmonitored releases.

4. Description of the methods used for dewatering or stabilize (e.g., removal of free-standing water, encapsulation, solidification, etc.) wet wastes, types of stabilization media or agents, expected waste volume increase factors, and implementation of a PCP to ensure a solid matrix and proper waste form characteristics and/or complete dewatering of wet wastes.

5. Types and characteristics of filtration systems, ion-exchange resins, and adsorbent media to treat liquid and wet wastes, including expected removal efficiencies and decontamination factors.

6. Description of the methods used for volume reduction of dry solid wastes, including sorting methods, technologies (e.g., shredders, crushers, and compactors), system components and their design parameters, and expected waste volume reduction factors.

7. For plants using offgas treatment systems relying on charcoal beds, description of the process for regenerating spent charcoals for reuse and the facilities for storing spent charcoals before shipment for disposal or regeneration via third parties. Radiological and physical properties of spent charcoals. Provisions to manage and ship spent charcoals for disposal and estimates of the projected annual or periodic amounts of spent charcoals that will be disposed of as radioactive waste.

8. Fraction, if any, of all liquid, wet, and dry solid waste processing projected to be contracted out to waste brokers or specialized facilities. Disposition methods of wastes generated from such processing and whether processed wastes will be returned to the plant for later disposal or shipped directly by the processor to an authorized low-level radioactive waste disposal facility on behalf of the applicant.

9. Description of waste container types and sizes; filling and handling methods; spill and leak prevention features; procedures for monitoring for removable radioactive contamination and external radiation; and provisions for decontamination, packaging, and storage of containers.

10. Provisions for onsite waste storage before shipping, including expected design volumes; expected radionuclide concentrations and radioactivity inventories; layout of the packaging, storage, and shipping areas; use of cranes, forklifts, monorails, and similar equipment; storage capacity; fire protection; building ventilation; shielding provisions; expected onsite storage durations; and the design bases for these estimates.

11. Design considerations for the use of shielding around waste processing equipment expected to exhibit elevated levels of external radiation, placement of such equipment in shielded cubicles, and the use of temporary or permanent shielding mounted on or in the immediate vicinity of mobile equipment.

12. Quality group classifications of piping and equipment and the bases governing the classification chosen in accordance with Regulatory Guide 1.143 for wastes produced during normal operation and anticipated operational occurrences. Design, expected temperatures and pressures, and construction materials of permanently installed systems and mobile processing equipment.

13. Design provisions incorporated in equipment and facility to facilitate operation and maintenance in accordance with Regulatory Guide 1.143 and as referenced in topical reports, as well as previous experience with similar equipment and methods referenced in the safety analysis report (SAR) or other supporting documents, as they relate to wastes produced during normal operation and anticipated operational occurrences.

14. Design features to reduce volumes of liquid, wet, and dry wastes handled by the SWMS; reduce radioactivity levels in wastes; minimize, to the extent practicable, contamination of the facility and environment; facilitate eventual decommissioning; and minimize, to the extent practicable, the generation of radioactive waste.

15. For multiunit stations, descriptions and design features of equipment and components (as permanently installed systems or in combination with mobile processing equipment) normally shared between interconnected processing and treatment subsystems.

16. Definition of the boundary of the SWMS, beginning at the interface from plant systems provided for the collection of process streams and radioactive wastes to the point of controlled discharges to the environment, as defined in the PCP and/or Offsite Dose Calculation Manual (ODCM), at the point of recycling to primary or secondary water system storage tanks, or to within plant facilities used for the storage of radioactive wastes and mixed wastes in accordance with Regulatory Guide 1.143 for wastes produced during normal operation and anticipated operational occurrences.

17. Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). For design certification (DC) and combined license (COL) reviews, the staff reviews the applicant's proposed ITAAC associated with the structures, systems, and components (SSCs) related to this SRP section in accordance with SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria." The staff recognizes that the review of ITAAC cannot be completed until after the rest of this portion of the application has been reviewed against acceptance criteria contained in this SRP section. Furthermore, the staff reviews the ITAAC to ensure that all SSCs in this area of review are identified and addressed as appropriate in accordance with SRP Section 14.3.

18. COL Action Items and Certification Requirements and Restrictions. For a DC application, the review will also address COL action items and requirements and restrictions (e.g., interface requirements and site parameters).

    For a COL application referencing a DC, a COL applicant must address COL action items (referred to as COL license information in certain DCs) included in the referenced DC. Additionally, a COL applicant must address requirements and restrictions (e.g., interface requirements and site parameters) included in the referenced DC.

19. Operational Program Description and Implementation. For a COL application, the staff reviews the Process Control Program (PCP) aspect of the Process and Effluent Monitoring and Sampling Program description and the proposed implementation milestones. The staff also reviews final safety analysis report (FSAR) Table 13.x to ensure that the PCP aspect of the Process and Effluent Monitoring and Sampling Program and associated milestones are included.

Review Interfaces

Other SRP sections interface with this section as follows:

1. Review of the SWMS and waste storage facilities given the use or presence of flammable materials is performed under SRP Section 9.5.1.

2. Review of the acceptability of the design analyses, procedures, and criteria used to establish the ability of Seismic Category I structures housing the system and supporting systems to withstand the effects of natural phenomena, such as the safe-shutdown earthquake, the probable maximum flood, and tornadoes and tornado missiles, is performed under SRP Sections 3.3.1, 3.3.2, 3.4.2, 3.5.3, 3.7.1 through 3.7.4, 3.8.4, and 3.8.5.

3. Review of the acceptability of the seismic and quality group classifications for structures and system components is performed under SRP Sections 3.2.1 and 3.2.2.

4. Review of technical specifications (TS) is performed under SRP Section 16.0.

5. Review of quality assurance is performed under SRP Chapter 17.

6. Review of a consequence of a liquid or wet waste tank failure with the potential of releasing radioactive materials to outdoor areas and a potable water supply is conducted under SRP Sections 11.2 and Branch Technical Position (BTP) 11-6.

7. If not included in the review of SRP Sections 11.2 and 11.3, an evaluation of the design features of building exhaust and ventilation systems servicing areas where liquid, wet, and solid wastes are processed and stored (e.g., use of HEPA and charcoal filters) is conducted under SRP Section 9.4 and, for instrumentation used to monitor and control radioactive effluent releases, under SRP Section 11.5.

8. Review of the SWMS design provisions incorporated to control, sample, and monitor radioactive materials in liquid, wet, and solid waste process and effluent streams is performed under SRP Section 11.5.

9. Review of design features of the SWMS process and post-accident sampling subsystems is conducted under SRP Sections 9.3.2 and 11.5.

10. Review of design features for the protection of potable and sanitary water systems is conducted under SRP Section 9.2.4.

11. Review of the Standard Radiological Effluent Controls (SREC) and ODCM, as they relate to elements of the PCP, is conducted under SRP Section 11.5.

12. If not included in the review of SRP Sections 11.2 and 11.3, an evaluation of source terms and dose calculations is conducted to assess the performance of the SWMS against the NRC's requirements set forth in 10 CFR 20.1302 and 10 CFR 20.1301(e), Table 2 of Appendix B to 10 CFR Part 20, and the dose objectives of Appendix I to 10 CFR Part 50, based on information in SRP Sections 11.1 and 11.4.

13. Review of the "as low as reasonably achievable" (ALARA) provisions in system design and operation to assure compliance with the occupational dose limits of 10 CFR 20.1201 and 10 CFR 20.1202 and Table 1 of Appendix B to 10 CFR Part 20 is conducted under SRP Chapter 12.

14. For COL reviews of operational programs, the review of the applicant's implementation plan is performed under SRP Section 13.4, "Operational Programs."

The specific acceptance criteria and review procedures are contained in the referenced SRP sections.

II. ACCEPTANCE CRITERIA

Requirements

Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations:

1. 10 CFR 20.1302 and 10 CFR 20.1301(e), as they relate to radioactive materials released in gaseous and liquid effluents to unrestricted areas. These criteria apply to releases resulting from SWMS operation during normal plant operations and anticipated operational occurrences.

2. 10 CFR 20.1406, as it relates to the design and operational procedures (for applications other than license renewals, after August 20, 1997) for minimizing contamination, facilitating eventual decommissioning, and minimizing the generation of radioactive waste.

3. 10 CFR 50.34a, as it relates to the provision of sufficient information to demonstrate that design objectives for equipment necessary to control releases of radioactive effluents to the unrestricted areas are kept as low as reasonably achievable.

4. 10 CFR Part 50, Appendix I, Sections II.A, II.B, II.C, and II.D, as they relate to the numerical guides for dose design objectives and limiting conditions for operation to meet the ALARA criterion.

5. 40 CFR Part 190 (the U.S. Environmental Protection Agency (EPA), generally applicable environmental radiation standards, as implemented under 10 CFR 20.1301(e)), as it relates to limits on total annual doses from all sources of radioactivity and radiation from the site (with single or multiple units).

6. Appendix A to 10 CFR Part 50, General Design Criterion (GDC) 60, as it relates to the design of the SWMS to control the release of radioactive materials in liquid effluents from the SWMS and to handle solid wastes produced during normal plant operation, including anticipated operational occurrences.

7. GDC 61, as it relates to the ability of systems that may contain radioactivity to assure adequate safety under normal and postulated accident conditions.

8. GDC 63, as it relates to the ability of the SWMS to detect conditions that may result in excessive radiation levels and to initiate appropriate safety actions.

9. 10 CFR 61.55 and 10 CFR 61.56, as they relate to classifying, processing, and disposing of dry solid and wet wastes at approved low-level radioactive waste disposal sites.

10. 10 CFR 20.2006 and Appendix G to 10 CFR Part 20, as they relate to the requirements for transferring and manifesting radioactive materials shipments to authorized facilities (e.g., disposal sites, waste processors).

11. 10 CFR 20.2007, as it relates to compliance with other applicable Federal, State, and local regulations governing any other toxic or hazardous properties of radioactive wastes, such as mixed wastes characterized by the presence of hazardous chemicals and radioactive materials, that may be disposed under 10 CFR Part 20.

12. 10 CFR 20.2108, as it relates to the maintenance of waste disposal records until the NRC terminates the pertinent license requirements.

13. 10 CFR Part 71 and 49 CFR Parts 171–180, as they relate to the use of approved containers and packaging methods for the shipment of radioactive materials.

14. 49 CFR 173.443, as it relates to methods and procedures used to monitor for the presence of removable contamination on shipping containers, and 49 CFR 173.441, as it relates to methods and procedures used to monitor external radiation levels for shipping containers and vehicles.

15. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design

certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;

16. 10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.

SRP Acceptance Criteria

Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are as follows for the review described in this SRP section. The SRP is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with the NRC regulations.

1. The SWMS design parameters are based on expected radionuclide distributions and concentrations consistent with reactor operating experience for similar designs, as evaluated under SRP Section 11.1.

2. Processing equipment is sized to handle the design SWMS inputs, that is, the types of liquid, wet, and solid wastes; radionuclide distributions and concentrations; radionuclide removal efficiencies and decontamination factors; waste volume reduction and increase factors; waste volumes; and waste generation rates.

3. All liquid and wet wastes will be stabilized in accordance with a PCP before offsite shipment, or provisions will be made to verify the absence of free liquid in each container and procedures to reprocess containers in which free liquid is detected in accordance with the requirements of Branch Technical Position (BTP) 11-3.

4. Other forms of wet wastes will be stabilized or dewatered (subject to the licensed disposal facility's waste acceptance criteria) in accordance with a PCP, or provisions will be made to verify the absence of free liquid in each container and procedures to reprocess containers in which excess water is detected in accordance with the requirements of BTP 11-3.

5. SWMS design objectives, design criteria, treatment methods, expected effluent releases, process and effluent radiation monitoring and control instrumentation, and methods for establishing process and effluent instrumentation control set points, as they relate to the PCP and ODCM under this SRP Section and SRP Section 11.5.

6. Waste containers, shipping casks, and methods of packaging wastes meet all applicable Federal regulations (e.g., 10 CFR Part 71, addressing the packaging and transportation of radioactive materials; 10 CFR 20.2006 and Appendix G to 10 CFR Part 20, addressing the transfer and manifesting of radioactive waste shipments; and 49 CFR Parts 171–180, addressing U.S. Department of Transportation (DOT) regulations for the shipment of radioactive materials); and 10 CFR Part 61 or

corresponding State regulations addressing applicable waste acceptance criteria of the disposal facility or waste processors.

7. Onsite waste storage facilities provide sufficient storage capacity to allow time for shorter lived radionuclides to decay before shipping in accordance with the requirements of BTP 11-3. The SAR should give the bases for determining the duration of the storage.

8. SWMS components and piping systems, as well as structures housing SWMS components, are designed in accordance with the provisions of Regulatory Guide 1.143, as it relates to the seismic design and quality group classification of components, and BTP 11-3 for wastes produced during normal operation and anticipated operational occurrences.

9. The SWMS contains provisions to reduce leakage and facilitate operations and maintenance in accordance with the provisions of Regulatory Guide 1.143 and BTP 11-3, as they relate to wastes produced during normal operation and anticipated operational occurrences.

10. For long-term onsite storage (e.g., for several years, but within the operational life of the plant), the storage facility should be designed to the guidelines of Appendix 11.4-A to this SRP section, including updated guidance from SECY 93-323 and SECY 94-198.

11. Liquid, wet, and dry solid wastes will be processed and disposed of in accordance with 10 CFR 61.55 and 10 CFR 61.56 requirements for waste classification and characteristics and with the waste acceptance criteria of the chosen licensed radioactive waste disposal site. The PCP should present the process and methods used to meet these 10 CFR Part 61 requirements.

12. Mixed wastes (characterized by the presence of hazardous chemicals and radioactive materials) will be processed and disposed in accordance with 10 CFR 20.2007, as it relates to compliance with other applicable Federal, State, and local regulations governing any other toxic or hazardous properties of radioactive wastes.

13. All effluent releases (gaseous and liquid) associated with the operation (normal and anticipated operational occurrences) of the SWMS will comply with 10 CFR Part 20 and Regulatory Guide 1.143, as they relate to the definition of the boundary of the SWMS beginning at the interface from plant systems, including multiunit stations, to the points of controlled liquid and gaseous effluent discharges to the environment or designated onsite storage locations, as defined in the PCP and ODCM.

14. <u>Operational Programs</u>. For COL reviews, the description of the operational program and proposed implementation milestone for the PCP aspect of the Process and Effluent Monitoring and Sampling Program are reviewed in accordance with 10 CFR 20.1301 and 20.13.2, 10 CFR 50.34a, 10 CFR 50.36a, and 10 CFR 50, Appendix I, section II and IV. Its implementation is required by a license condition.

<u>Technical Rationale</u>

The technical rationale for application of these acceptance criteria to the areas of review addressed by this SRP section is discussed in the following paragraphs:

1. 10 CFR 20.1302 requires that surveys of radiation levels in unrestricted areas be performed to demonstrate system compliance with the 10 CFR 20.1301 dose limits to individual members of the public. 10 CFR 20.1302 identifies two approaches, either of which can demonstrate compliance with the 10 CFR 20.1301 dose limits. One of these approaches requires the following:

   A. Demonstrate that the annual average concentrations of radioactive material released in gaseous and liquid effluents at the boundary of the unrestricted area do not exceed the specified in Table 2 of Appendix B to 10 CFR Part 20; and

   B. Demonstrate that the annual and hourly doses from external sources to an individual continuously present in an unrestricted area will not exceed 0.5 millisievert (mSv) (0.05 rem) and 0.02 mSv (0.002 rem), respectively.

   Meeting the above requirements provides assurance that the 10 CFR 20.1301dose limits to individual members of the public will not be exceeded. The review in this SRP section will include an evaluation of whether the above-identified dose requirements are met. Meeting the requirements on gaseous and liquid effluent concentration limits in unrestricted areas from all plant sources of radioactivity (including that associated with the operation of the SWMS) is identified as an acceptance criterion in SRP Sections 11.2 and 11.3 and will be evaluated in those SRP sections as well.

2. Meeting the requirements of 10 CFR 50.34a, as it relates to adequate design information on the SWMS, provides a level of assurance that the SWMS will have the necessary equipment and design features to control radioactive effluent releases to the environment resulting from its operation, in accordance with the requirements of 10 CFR 20.1302, Appendix I to 10 CFR Part 50, and GDC 60 and 61.

   The review should evaluate the types and characteristics of filtration systems, ion-exchange resins, and adsorbent and stabilization media proposed to treat liquid and wet wastes. This includes removal efficiencies, decontamination factors, waste volume increase factors for stabilized wastes, and volume decrease factors for compacted wastes, taking into account the expected physical, chemical, and radiological properties of process waste and effluent streams. The review should determine whether performance meets or exceeds that noted in NRC guidance, standard DCs, industry standards, or topical reports. The NRC guidance includes NUREG-0016 or NUREG-0017 and Regulatory Guide 1.112, as they relate to the use of acceptable methods for calculating radionuclide concentrations in process streams and annual effluent releases, and Regulatory Guide 1.110, as it relates to performing cost-benefit analysis in reducing cumulative population doses by using available technology.

3. GDC 60, requires that the nuclear power unit design include provisions to handle radioactive wastes produced during normal reactor operation, including anticipated operational occurrences.

   GDC 60 specifies that the SWMS must provide for a holdup capacity sufficient to retain radioactive wastes, particularly where unfavorable site environmental conditions may impose unusual operational limitations on the release of effluents. Waste processing holdup times and long-term storage capacity also provide decay time for shorter-lived radionuclides before they are processed further or released to the environment. The holdup times are used in the source term calculations, employing the methods described in NUREG-0016 or NUREG-0017 and Regulatory Guide 1.112.

Meeting the requirement of GDC 60 provides assurance that releases of radioactive materials in liquid and gaseous effluents to unrestricted areas during normal plant operation and during anticipated operational occurrences of the SWMS will not result in offsite radiation doses exceeding the dose objectives specified in Appendix I to 10 CFR Part 50 or concentrations of radioactive materials in liquid effluents in any unrestricted area exceeding the limits specified in Table 2, Column 2, of Appendix B to 10 CFR Part 20. Meeting the requirement of GDC 60 provides a level of assurance that the resulting wastes produced from the SWMS will meet the requirements of 10 CFR 61.55 and 10 CFR 61.56 for waste classification and characteristics and DOT shipping regulations under 49 CFR Parts 171–180.

4. GDC 61 requires that systems that may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions.

   Compliance with GDC 61 requires that the SWMS and other systems (as permanently installed systems or in combination with mobile systems) that may contain radioactivity shall be designed to ensure adequate safety under normal and postulated accident conditions. This criterion specifies that the design of such facilities' shall enable inspection and testing of components important to safety and with suitable shielding for radiation protection.

   SRP Section 11.4 and Regulatory Guide 1.143 describe staff positions related to the design of the SWMS, including provisions for equipment to be used to prevent and contain spillage while pumping, filling, pouring, and overfilling waste containers or system tanks and features to contain the contents of resin storage tanks in the event of subsystem failures. Regulatory Guide 1.143 furnishes design guidance acceptable to the NRC's staff on seismic and quality group classification and quality assurance provisions for the SWMS subsystems, structures, and components, as they relate to wastes produced during normal operation and anticipated operational occurrences.

   Meeting the requirement of GDC 61 provides assurance that releases of radioactive materials during normal operation and anticipated operational occurrences, including adverse conditions on system components, will not result in radiation doses that exceed the 10 CFR Part 20 limits. In addition, meeting this requirement will help ensure that the SWMS will continue to perform its safety function(s) under postulated accident conditions.

5. GDC 63 requires that radioactive waste systems be able to detect conditions that may result in excessive radiation levels in waste storage locations and to initiate appropriate safety actions.

   Meeting the requirements of GDC 63 will provide a level of assurance that the SWMS will be equipped with monitoring and detection capabilities to facilitate the initiation of timely corrective actions. It will also ensure that effluent concentrations in unrestricted areas arising from SWMS operation do not exceed the limits for effluents specified in Table 2 of Appendix B to 10 CFR Part 20 and that radiation exposures to occupational workers do not exceed the occupational dose limits of 10 CFR 20.1201 and 10 CFR 20.1202 and Table 1 of Appendix B to 10 CFR Part 20. The review on occupational exposures is conducted under SRP Section 12.0.

6. 10 CFR Part 61 establishes, for land disposal of radioactive waste, the procedures, criteria, and terms and conditions for the disposal of radioactive wastes containing byproduct, source, and other special nuclear material. State and local regulations also apply to the licensing of land disposal facilities.

   The SWMS processes liquid, wet, and dry solid wastes for shipment to a licensed disposal facility. For the SWMS, 10 CFR 61.55 and 10 CFR 61.56 require the inclusion of provisions in the system design and PCP that describe the dewatering and stabilization processes and the classification, processing, and disposition of solid wastes. The SWMS and PCP should also address the criteria that the different waste classes should satisfy and the various characteristics that the processed liquid wet wastes should satisfy. Item 7 of this SRP subsection outlines the technical and procedural elements that the PCP should address and identifies related NRC guidance.

   Meeting the requirements of 10 CFR 61.55 and 10 CFR 61.56 provides a level of assurance that radioactive wastes processed by the SWMS have been properly classified such that controls and resulting waste forms are effective and that the processed waste, when stabilized as required, will not structurally degrade and will be compatible with the disposal site's waste acceptance criteria and the 10 CFR Part 61 requirements. The maximum radionuclide concentrations allowable for land disposal are defined by 10 CFR 61.55 for Class A, B, and C wastes.

7. In the context of 10 CFR Part 61, radioactive wastes shipped to disposal facilities must comply with the requirements addressing waste classifications and characteristics and the shipping regulations under 10 CFR Part 71 and 49 CFR Parts 171–180.

   Plant TS require that a PCP be established to provide reasonable assurance of the complete stabilization of process wastes and the absence of free water in process wastes. The PCP and operational procedures should describe, given specific waste-processing technologies and methods, a set of process parameters that are used to process wastes. Among others, the parameters include pH, water content, oil content, presence of hazardous materials, content of chelating agents, and ratio of stabilization agent to chemical additives by types of wastes. The types of wastes may include filter sludge, spent resins, boric acid solutions, process concentrates, and filter media. The PCP should describe the bases in developing waste mixture formulas, sampling, analysis, tests, radionuclide scaling factors, encapsulation and concentration averaging, controls on radiolytic hydrogen gas generation, and methods to demonstrate that the processing of actual or simulated waste samples can be successfully accomplished and ensure compliance with the requirements of 10 CFR 61.55 and 10 CFR 61.56 for waste classification and characteristics; characterizations of waste in shipping manifests in accordance with 10 CFR 20.2006; compliance with 10 CFR 20.2007, as it relates to other applicable Federal, State, and local regulations governing the presence of any other toxic or hazardous materials in waste; conformance with NRC and DOT shipping regulations under 10 CFR Part 71 and 49 CFR Parts 171–180; and compliance with waste acceptance criteria of authorized disposal facilities or waste processors.

   The PCP should identify surveillance requirements consistent with the plant's TS, administrative procedures, operational procedures, operation of the process and effluent radiation monitoring and control instrumentation and procedures for setting instrumentation alarm set points, quality assurance and quality control, radiological controls and monitoring, information to be contained in annual radiological effluent release reports, reporting requirements to the NRC, instructions on the use of the NRC's uniform radioactive shipping waste manifest, and the process for initiating and documenting changes to the PCP and its supporting procedures.

Related guidance may be found in NUREG-1301 for pressurized-water reactors (PWRs) or NUREG-1302 for boiling-water reactors (BWRs), NUREG-0133, and NUREG/BR-0204. Specific guidance on waste form, characterization, and classification is listed in Inspection Procedure 84850; "Issuance Final Branch Technical Position on Concentration Averaging and Encapsulation," dated January 17, 1995; "Final Waste Classification and Waste Form Technical Position Papers," dated May 11, 1983; "Revised Staff Technical Position on Waste Form (SP-91-13)," dated January 30, 1991; and IE Information Notice No. 86-20, dated March 28, 1986, on methodologies used to develop waste-scaling factors. IE Bulletin No. 79-19 and IE Information Notice Nos. 84-72, 85-92, 87-07, and 90-31 present illustrative examples of issues associated with some operational aspects of the PCP.

8. 10 CFR Part 71 establishes requirements for packaging, preparation for shipment, and transportation of licensed material and procedures and standards for packaging and shipping of fissile material or quantities of other licensed materials in excess of Type A quantities, and it defines the applicability of 10 CFR Part 71 to waste generators and common carriers. Regarding allowable external radiation levels and residual surface contamination on external surfaces of shipping containers and packages, 10 CFR Part 71 presents criteria and also refers to DOT shipping regulations under Subpart I (Class 7) of 49 CFR Part 173.

   Meeting the requirements of 10 CFR Part 71 provides a level of assurance that the operation of the SWMS and development of the PCP with regard to packaging, preparation for shipment, qualification of the packaging material, testing of the package, exemptions, quality control and procedures, and transportation of licensed radioactive materials will not result in an undue risk to the public.

9. BTP 11-3 presents guidance on SWMS design guidelines and operation, addressing process parameters, waste stabilization or dewatering, waste form properties, free liquid detection, quality assurance, waste storage, and portable solid waste systems.

   The BTP focuses primarily on wet and liquid wastes for the purpose of ensuring complete stabilization and dewatering. For dry wastes, it emphasizes the use of waste volume reduction technologies for minimizing the amounts of wastes shipped to land disposal facilities. Generic Letter Nos. 80-009, 81-038, and 81-039 provide further guidance.

   Meeting the guidelines of BTP 11-3 provides a level of assurance that the SWMS, as implemented under the PCP, includes the necessary equipment, processes, and procedures to satisfactorily process, monitor, store for decay, and provide storage facilities for radioactive wastes before shipment for offsite disposal or further processing by waste processors.

10. Appendix 11.4-A addresses the long-term storage of wet, stabilized, and dry solid wastes.

    Appendix 11.4-A provides guidance for applicants when considering onsite low-level radioactive waste storage capabilities for periods that may last several years but are significantly less than the life of the plant. The guidance emphasizes safety considerations in the storing, handling, and eventual disposition of radioactive wastes under 10 CFR Part 61 or equivalent State regulations. Generic Letter Nos. 80-009, 81-038, and 81-039, and SECY 94-198 and SECY 93-323 contain further guidance.

Meeting the guidelines of Appendix 11.4-A provides a level of assurance that the SWMS, as implemented under the PCP, will meet the associated requirements of the NRC's regulations (10 CFR Part 20 and 10 CFR Part 71) and DOT shipping regulations (49 CFR Parts 171–180) to ensure that container breaches will not occur during interim storage periods, or minimize the chance of such occurrences, and to preclude or reduce the likelihood of uncontrolled and unmonitored releases of radioactive wastes and materials from processing, handling, transportation, and storage accidents.

11. 10 CFR 20.1406 requires that applicants describe how facility design and procedures for operation will minimize, to the extent practicable, contamination of the facility and the environment; facilitate eventual decommissioning; and minimize, to the extent practicable, the generation of radioactive waste. Regulatory Guide 1.143 presents criteria for SSCs outside containment that contain radioactive wastes produced during normal operation and anticipated operational occurrences..

Specific guidance to meet the 10 CFR 20.1406 requirements is listed below:

A. SWMS processing systems (either as permanently installed systems or in combination with mobile equipment) with a potential for leakage shall provide means to control and contain this leakage to prevent contamination of building floors and interconnected systems (e.g., curbing, floor sloping to local drains, floor-to-floor seals over floor expansion joints, wall-to-floor joint seals, sheathed hoses, drip pans or containment boxes, backflow preventers, siphon breakers, self-sealing quick-disconnects, and operational interlocks). See guidance given in relevant NRC bulletins and circulars (e.g., IE Bulletin Nos. 79-19 and 80-10; IE Circular Nos. 77-10, 77-14, 79-07, 79-09, 79-21, and 81-09; and IE Information Notice Nos. 84-72, 85-92, 87-07, and 90-31).

B. In facilitating decommissioning, designs should minimize, to the extent practicable, embedding contaminated piping in concrete, consistent with maintaining radiation doses ALARA during operations and decommissioning.

C. To minimize waste generation, provisions should be in place to clean contaminated materials (e.g., system components and equipment) and regenerate or reuse resin beds as applicable (e.g., demineralizer resin beds with some remaining ion-exchange capacity when feasible), as opposed to premature disposal.

D. Mobile liquid waste processing systems with interconnections to permanently installed plant SWMS subsystems should include provisions that avoid the contamination of nonradioactive systems, prevent uncontrolled and unmonitored releases of radioactive materials in the environment, and avoid interconnections with potable and sanitary water systems.

E. All temporary and flexible lines (as hoses and connections), system piping embedded in concrete, and effluent discharge lines or piping buried in soils should undergo pressure testing. All system piping and valves associated with transfer lines to storage tanks and discharge piping buried in soils and concrete, including features designed for the early detection of leaks and spills (e.g., leak detection sumps and wells), should have corrosion-resistant properties. See Regulatory Guide 1.143 for wastes produced during normal operation and anticipated operational occurrences.

F. Further guidance is found in Memorandum from Larry W. Camper to David B. Matthews and Elmo E. Collins, dated October 10, 2006, “List of Decommissioning Lessons Learned in Support of the Development of a Standard Review Plan for New Reactor Licensing” (ADAMS Accession No. ML0619201830); and NUREG/CR-3587, “Identification and Evaluation of Facility Techniques for Decommissioning of Light Water Reactors,” and “Liquid Radioactive Release Lessons Learned Task Force, Final Report,” Sections 2.0 and 3.2.2, dated September 1, 2006 (ADAMS Accession No. ML062650312).

12. 10 CFR 20.1301(e) requires that NRC-licensed facilities comply with the EPA generally applicable environmental radiation standards of 40 CFR Part 190 for facilities that are part of the fuel cycle. The EPA annual dose limits are 0.25 mSv (25 millirem (mrem)) to the whole body, 0.75 mSv (75 mrem) to the thyroid, and 0.25 mSv (25 mrem) to any other organ.

Meeting the requirements of 10 CFR 20.1301(e) necessitates the consideration of all potential sources of external radiation and radioactivity, including liquid and gaseous effluents and external radiation exposures from buildings, storage tanks, radioactive waste, storage areas, and N-16 skyshine from BWR turbine buildings. The EPA standards apply to the entire site or facility, which may have either single or multiple units. SRP Sections 11.2 and 11.3 address sources of radioactivity and doses associated with liquid and gaseous effluents, respectively. In turn, SRP Section 11.5 addresses compliance with all sources of effluents. SRP Section 12.3-12.4 addresses sources of radiation and external radiation exposures from buildings housing the SWMS, radioactive waste storage areas, storage tanks, and other site buildings.

III. REVIEW PROCEDURES

The reviewer will select material from the procedures described below, as may be appropriate for a particular case.

These review procedures are based on the identified SRP acceptance criteria. For deviations from these acceptance criteria, the staff should review the applicant’s evaluation of how the proposed alternatives provide an acceptable method of complying with the relevant NRC requirements identified in Subsection II.

In accordance with Regulatory Guide 1.70 or 1.206, the NRC staff will review for completeness the information describing the design features of the SWMS provided in the SAR, the DC application, update of the final SAR, or the COL application, to the extent not addressed in a referenced certified design, including referenced subsections of SRP Sections 11.1, 11.2, 11.3, 11.5, and 12.3-12.4.

1. The P&IDs and the process flow diagrams are reviewed to determine system design, methods of operation, and parameters used in the design (i.e., expected and design flow rates, concentrations of radioactive material, radionuclide distributions, and waste categories). The system design and design criteria, including mobile waste processing

systems, are compared with Regulatory Guide 1.143, BTP 11-3, and available data from operating plants of similar design, as they relate to wastes produced during normal operation and anticipated operational occurrences.

2. The methods to be used for stabilization and/or dewatering are compared with experience gained from previous licensing reviews and with available data from operating plants employing similar methods. The elements of the PCP are reviewed to assure that the proposed stabilization and/or dewatering method is capable of solidifying and/or dewatering the range of constituents expected to be present in wastes. The methods proposed to verify that all wet wastes can be adequately stabilized or dewatered are reviewed, and a determination is made as to their acceptability considering (a) the ability of the technique to detect free, mobile, or uncombined liquids (in the case of encapsulation or solidification) or excess free water (such as in the case of dewatering), (b) the procedures to be employed to solidify or dewater free liquids if detected, (c) the expected final waste form characteristics, and (d) the extent of reliance on mobile processing systems and waste processors. The PCP, including dewatering or stabilization (if performed), is reviewed on a plant-specific basis against the 10 CFR Part 61 requirements and guidance given in BTP 11-3 and Generic Letter Nos. 80-009, 81-038, and 81-039.

3. The description of procedures for the packaging and shipment of solid wastes to an approved offsite disposal facility or waste processor is reviewed, and the reviewer verifies that the applicant makes definite commitments to follow appropriate NRC and DOT regulations, as well as EPA and State regulations addressing the presence of other toxic and hazardous materials. The values given in the SAR for the volumes, radionuclide distributions and concentrations, and radioactive inventories of wastes to be shipped off site are compared with data from operating plants of similar design and information from previous license applications.

4. The solid waste system design capacity is compared with the design basis of expected waste volumes to determine whether the applicant has provided sufficient reserve capacity for greater-than-expected waste volumes, which may occur as a result of anticipated operational occurrences. The inplant storage capacity, for areas designed to accommodate approximately 6 months of waste generation, is compared to the guidelines of BTP 11-3. The comparison will be based on the design criteria as stated in the SAR, the availability of system components to handle surge flows, reliance on mobile processing systems, and whether the storage facilities will provide onsite storage duration periods sufficient to permit the decay of shorter lived radionuclides. For longer term onsite storage (e.g., several years, but within the operational life of the plant), the storage facility is compared to the guidelines of Appendix 11.4-A to this SRP section.

5. The equipment layout, design features, and mode of operation of the solid waste system, as permanently installed systems or in combination with mobile processing equipment, are compared to the guidelines of Regulatory Guide 1.143 and BTP 11-3, as they relate to wastes produced during normal operation and anticipated operational occurrences.

6. Review of the PCP and TS (i.e., administrative controls section proposed by the applicant for process and effluent control) is performed for input to the review of SRP Section 16.0 and this SRP section. The reviewer will determine that the content and scope of the programs identified in the administrative controls section of the TS prepared by the applicant are in agreement with requirements identified as a result of

the NRC staff's review. The review will include the evaluation or development of appropriate limiting conditions for operation or controls and their bases, consistent with the plant design. The programs identified in the administrative controls section of the TS are reviewed according to the requirements of 10 CFR 50.36a.

7. The classification and characterization of wastes are compared to the requirements of 10 CFR 61.55 and 10 CFR 61.56. The requirements address the classification and characteristics of wastes, and they define maximum radionuclide concentrations allowable for land disposal as Class A, B, and C wastes.

8. Meeting the requirements of 10 CFR 50.34a, as it relates to the SWMS, provides assurance that each nuclear power reactor will have necessary design features and equipment to control releases of radioactive liquid and gaseous effluents to the environment in accordance with the requirements of 10 CFR 20.1302 and 20.1301(e); Appendix I to 10 CFR Part 50; and Appendix A to 10 CFR Part 50, GDC 60 and GDC 61. These requirements may be evaluated using the following two approaches:

   A. As part of the review of this SRP section, including a verification of compliance with offsite dose requirements and liquid and gaseous effluent limits associated with the operation of the SWMS; or

   B. With the results of the review incorporated in the evaluation of SRP Sections 11.2 and 11.3, addressing compliance with offsite dose requirements, effluent concentrations limits, and all liquid and gaseous effluents from all sources, including those generated by the operation of the SWMS

9. The SWMS is reviewed to ensure that the design includes provisions to prevent and collect leakage resulting from overflows, leaks, and spillage associated with waste processing, storage, and movement of waste containers; operation of mobile processing equipment; and use of indoor or outdoor storage tanks (including temporary tanks) and is in conformance with 10 CFR 20.1406 requirements and guidelines of Regulatory Guide 1.143 for wastes produced during normal operation and anticipated operational occurrences.

   The review considers information describing design features that will minimize, to the extent practicable, contamination of the facility and environment; facilitate eventual decommissioning; and minimize, to the extent practicable, the generation of extraneous radioactive wastes associated with the operation of the SWMS as a result of operator error and processing equipment failures or malfunctions. In addition, the review may also consider the information contained in the DC application and updates in the SAR or the COL application to the extent not addressed in a referenced certified design. The NRC guidance includes the following:

   A. Memorandum from Larry W. Camper to David B. Matthews and Elmo E. Collins, dated October 10, 2006 (ADAMS Accession No. ML0619201830); and NUREG/CR-3587, as they relate to the design issues that need to be addressed to meet the requirements of 10 CFR 20.1406

   B. "Liquid Radioactive Release Lessons Learned Task Force, Final Report," Sections 2.0 and 3.2.2, September 1, 2006 (ADAMS Accession No. ML062650312)

C. Regulatory Guides 1.11 and 1.143 for wastes produced during normal operation and anticipated operational occurrences

D. SRP Section 9.2.4

E. Relevant NRC bulletins and circulars—for example, IE Bulletin Nos. 79-19 and 80-10; IE Circular Nos. 77-10, 77-14, 79-07, 79-09, 79-21, and 81-09; and IE Information Notice Nos. 84-72, 85-92, 87-07, and 90-31

F. Industry standards, e.g., ANSI/ANS-55.6-1993 (1999), and ANSI/ANS-40.37-1993 (200x updated draft)

10. The PCP and associated plant TS are reviewed to determine whether they identify all regulatory requirements, follow the NRC's guidance, and contain all appropriate operational elements. The regulatory requirements are associated with 10 CFR 61.55 and 10 CFR 61.56 for waste classification and characteristics; 10 CFR 20.2006 for the characterizations of waste in shipping manifests; 10 CFR 20.2007, as it relates to other applicable Federal, State, and local regulations governing the presence of any other toxic or hazardous materials; the NRC and DOT shipping regulations under 10 CFR Part 71 and 49 CFR Parts 171–180; and waste acceptance criteria of authorized disposal facilities or waste processors. The PCP should describe, given specific waste processing technologies and methods, a set of parameters used to process wastes. The PCP should identify surveillance requirements consistent with the plant's TS, administrative procedures, operational procedures, quality assurance and quality control program, radiological controls and monitoring, information to be contained in annual radiological effluent release reports, reporting requirements to the NRC, instructions on the use of the NRC's uniform radioactive shipping waste manifest, and the process for initiating and documenting changes to the PCP and its supporting procedures.

    Related guidance may be found in NUREG-1301 (PWRs) or NUREG-1302 (BWRs), NUREG-0133, NUREG/BR-0204, and Regulatory Guide 1.21. Specific guidance on waste form, characterization, and classification is listed in Inspection Procedure 84850; "Issuance of Final Branch Technical Position on Concentration Averaging and Encapsulation," dated January 17, 1995; "Final Waste Classification and Waste Form Technical Position Papers," dated May 11, 1983; "Revised Staff Technical Position on Waste Form (SP-91-13)," dated January 30, 1991; and IE Information Notice No. 86-20, dated March 28, 1986, on methodologies used to develop waste scaling factors. IE Bulletin No. 79-19 and IE Information Notice Nos. 84-72, 85-92, 87-07, and 90-31 present illustrative examples of issues associated with some operational aspects of the PCP.

11. In determining compliance with the EPA generally applicable environmental radiation standards of 40 CFR Part 190, as implemented under 10 CFR 20.1301(e), the review considers all sources of radiation and radioactivity as potential contributors to total doses to members of the public from the site, whether from single or multiple units. The review focuses on sources of radioactivity and external radiation exposures from waste processing buildings, waste storage buildings, waste storage tanks, and temporary waste storage or staging areas. The source terms and associated doses from liquid and gaseous effluents associated with the operation of the SWMS may be evaluated in this section of the SRP or integrated with the evaluation of SRP Sections 11.2 and 11.3.

In turn, SRP Section 11.5 addresses compliance with all sources of effluents. SRP Section 12.3-12.4 evaluates the doses associated with external radiation from buildings and contained sources of radioactivity.

12. Operational Programs. The reviewer verifies that the PCP aspect of the Process and Effluent Monitoring and Sampling Program is fully described and that implementation milestones have been identified. The reviewer verifies that the program and implementation milestones are included in FSAR Table 13.x.

    Implementation of this program will be inspected in accordance with NRC Inspection Manual Chapter IMC-2504, "Construction Inspection Program - Non-ITAAC Inspections."

    The applicant described the PCP aspect of the Process and Effluent Monitoring and Sampling Program and its implementation which is included in the license condition on operational programs and implementation.

13. For review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the final safety analysis report (FSAR) meets the acceptance criteria. DCs have referred to the FSAR as the design control document (DCD). The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC FSAR.

    For review of a COL application, the scope of the review is dependent on whether the COL applicant references a DC, an early site permit (ESP) or other NRC approvals (e.g., manufacturing license, site suitability report or topical report).

    For review of both DC and COL applications, SRP Section 14.3 should be followed for the review of ITAAC. The review of ITAAC cannot be completed until after the completion of this section.

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the review and calculations (if applicable) support conclusions of the following type to be included in the staff's safety evaluation report. The reviewer also states the bases for those conclusions.

The staff concludes that the design of the SWMS (either as a permanently installed system or in combination with mobile systems), which includes the equipment necessary to process liquid, wet, and dry solid wastes and to control releases of radioactive materials associated with the operation of the SWMS, is acceptable and meets the requirements of 10 CFR 20.1301 and 20.1302, 10 CFR 20.1301(e), 10 CFR 20.1406, 10 CFR 20.2006, 10 CFR 20.2007, and 10 CFR 20.2108; 10 CFR (50.34a) and Appendix I dose objectives; GDC 60, 61, and 63; 10 CFR Part 61, 10 CFR Part 71, and 49 CFR Parts 171–180 for the proper classification, characterization, packaging, shipment, and disposal of radioactive wastes; and applicable NRC BTPs and regulatory guides.

This conclusion is based on the following:

1. The applicant has demonstrated that the SWMS, either as a permanently installed system or in combination with mobile systems, includes the equipment and instrumentation used for the processing, packaging, and storage of radioactive wastes before shipment to an offsite licensed land disposal facility or waste processors. The scope of the review of the SWMS includes line or flow diagrams of the system, P&IDs, process and effluent radiation monitoring and control instrumentation, and descriptive information for the SWMS and for those auxiliary supporting systems that are essential to the operation of the SWMS. The staff has reviewed the applicant's proposed design criteria and design bases for the SWMS, as well as the applicant's analysis of those criteria and bases. The ability of the proposed system to process the types and volumes of wastes, including radionuclides and radioactivity levels, expected during normal operation and anticipated operational occurrences, are in accordance with GDC 60, 61, and 63; provisions for the handling of wastes under the requirements of 10 CFR Part 61 and 10 CFR 71; and applicable DOT regulations under 49 CFR Parts 171–180. The staff found the design features built into the SWMS to control effluent releases to unrestricted areas within the limits of 10 CFR Part 20, arising from system operations, to be acceptable.

   Based on the staff's review, the applicant's proposed PCP, operating procedures, and TS, as they relate to classifying, processing, and disposing of wastes, meet the requirements of 10 CFR Part 61 and 10 CFR 20.2006, 10 CFR 20.2007, and 10 CFR 20.2108. The applicant's proposed methods of assuring complete stabilization, encapsulation, and/or dewatering are acceptable, and the processing, design features, and waste storage also meet the requirements of BTP 11-3 and Appendix 11.4-A to this SRP section (as it relates to plants with temporary onsite storage facilities for low-level radioactive waste). The PCP describes, given the proposed waste processing technologies and methods, a set of parameters that are used to process wastes. The PCP identifies surveillance requirements consistent with the plant's TS, administrative procedures, operational procedures, quality assurance and quality control program, radiological controls and monitoring program, information to be contained in annual radiological effluent release reports, reporting requirements to the NRC, instructions on using the NRC's uniform radioactive shipping waste manifest, and the process for initiating and documenting changes to the PCP and its supporting procedures.

   The basis for acceptance in the staff's review is conformance of the applicant's design, design criteria, design bases, and proposed PCP and TS for the SWMS, including the associated use of mobile processing equipment, to the regulations and regulatory guidance, as referenced above, as well as to branch technical positions and industry standards.

2. The applicant has met the requirements of 10 CFR 20.1406 with respect to providing a description of how facility design and procedures for operation will minimize, to the extent practicable, contamination of the facility and the environment; facilitate eventual decommissioning; and minimize, to the extent practicable, the generation of radioactive waste.

3. The applicant has met the requirements of Appendix A to 10 CFR Part 50, GDC 60, 61, and 63, with respect to controlling releases of radioactive materials to the environment using available technology. The staff has considered the ability of the proposed SWMS and mobile processing equipment to meet the

demands of the plant resulting from anticipated operational occurrences and has concluded that the system capacity and design flexibility are adequate to meet the plant's anticipated needs.

The applicant has fulfilled the requirements of Section II.D of Appendix I to 10 CFR Part 50 with respect to meeting the ALARA criterion. The staff has considered the potential effectiveness of augmenting the proposed SWMS using items of reasonably demonstrated technology and has determined that further waste treatment will not effect reductions in cumulative population doses reasonably expected within an 80-kilometer (50-mile) radius of the reactor at a cost of less than $1000 per man-rem or man-thyroid-rem.

4. The staff has reviewed the applicant's quality assurance provisions for the SWMS, the quality group classifications used for system components, and the seismic design applied to structures housing these systems. The design of the systems and structures housing these systems meet the guidance of Regulatory Guide 1.143 for wastes produced during normal operation and anticipated operational occurrences.

5. The staff has reviewed the provisions incorporated in the applicant's design to control the release of radioactive materials in wastes resulting from spills, leaks, and inadvertent tank overflows; avoid the contamination of nonradioactive systems; prevent uncontrolled and unmonitored releases of radioactive materials to the environment; and avoid interconnections with potable and sanitary water systems. The staff concludes that the measures proposed by the applicant are consistent with the requirements of GDC 60 and 61 to 10 CFR Part 50, Appendix A, and 10 CFR 20.1406, and the guidance of Regulatory Guide 1.143 for wastes produced during normal operation and anticipated operational occurrences.

6. The applicant described the PCP aspect of the Process and Effluent Monitoring and Sampling Program and its implementation which is included in the license condition on operational programs and implementation.

7. For DC and COL reviews, the findings will also summarize the staff's evaluation of requirements and restrictions (e.g., interface requirements and site parameters) and COL action items relevant to this SRP section.

   In addition, to the extent that the review is not discussed in other SER sections, the findings will summarize the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable.

V. IMPLEMENTATION

The staff will use this SRP section in performing safety evaluations of DC applications and license applications submitted by applicants pursuant to 10 CFR Part 50 or 10 CFR Part 52. Except when the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the staff will use the method described herein to evaluate conformance with Commission regulations.

The provisions of this SRP section apply to reviews of applications submitted six months or more after the date of issuance of this SRP section, unless superseded by a later revision.

VI. REFERENCES

1. 10 CFR Part 20, “Standards for Protection Against Radiation.”

2. 10 CFR 20.1201, “Occupational Dose Limits for Adults.”

3. 10 CFR 20.1202, “Compliance with Requirements for Summation of External and Internal Doses.”

4. 10 CFR 20.1301, “Dose Limits for Individual Members of the Public.”

5. 10 CFR 20.1302, “Compliance with Dose Limits for Individual Members of the Public.”

6. 10 CFR 20.1406, “Minimization of Contamination.”

7. 10 CFR 20.2006, “Transfer for Disposal and Manifests.”

8. 10 CFR 20.2007, “Compliance with Environmental and Health Protection Regulations.”

9. 10 CFR 20.2108, “Records of Waste Disposal.”

10. 10 CFR Part 20, Appendix B, “Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage.”

11. 10 CFR 50.34a, “Design Objectives for Equipment to Control Releases of Radioactive Material in Effluents—Nuclear Power Reactors.”

12. 10 CFR Part 50, Appendix I, “Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion ‘As Low As Is Reasonably Achievable’ for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents.”

13. 10 CFR 50.36a, “Technical Specifications on Effluents from Nuclear Power Reactors.”

14. 10 CFR Part 50, Appendix A, General Design Criterion 60, “Control of Releases of Radioactive Materials to the Environment.”

15. 10 CFR Part 50, Appendix A, General Design Criterion 61, “Fuel Storage and Handling and Radioactivity Control.”

16. 10 CFR Part 50, Appendix A, General Design Criterion 63, “Monitoring Fuel and Waste Storage.”

17. 10 CFR Part 52, “Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants.”

18. 10 CFR Part 61, “Licensing Requirements for Land Disposal of Radioactive Waste.”

19. 10 CFR Part 71, “Packaging and Transportation of Radioactive Material.”

20. 10 CFR Part 100, “Reactor Site Criteria.”

21. 49 CFR Parts 171–180, “Subpart C—Hazardous Materials Regulations.”

22. Regulatory Guide 1.11, “Instrument Lines Penetrating Primary Reactor Containment.”

23. Regulatory Guide 1.21, “Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants.”

24. Regulatory Guide 1.33, “Quality Assurance Program Requirements (Operation).”

25. Regulatory Guide 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants.”

26. Regulatory Guide 1.110, “Cost Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors.”

27. Regulatory Guide 1.112, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors.”

28. Regulatory Guide 1.143, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants.”

29. Regulatory Guide 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition).”

30. Regulatory Guide 8.8, “Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable.”

31. Regulatory Guide 8.10, “Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable.”

32. Branch Technical Position (BTP 11-3), “Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants.”

33. Standard Review Plan, Section 11.4, Appendix 11.4-A, “Design Guidance for Temporary Storage of Low-Level Radioactive Waste.”

34. NRC Generic Letter 80-009, “Low Level Radioactive Waste Disposal.”

35. NRC Generic Letter 81-038, “Storage of Low Level Radioactive Wastes at Power Reactor Sites.”

36. NRC Generic Letter 81-039, “NRC Volume Reduction Policy.”

37. NRC Inspection Procedure 84850, “Radioactive Waste Management—Inspection of Waste Generator Requirements of 10 CFR Part 20 and 10 CFR Part 61,” June 6, 2002.

38. NRC, “Issuance of Final Branch Technical Position on Concentration Averaging and Encapsulation,” January 17, 1995.

39. NRC, "Final Waste Classification and Waste Form Technical Position Papers," May 11, 1983.

40. NRC Inspection Manual Chapter IMC-2504, "Construction Inspection Program - Non-ITAAC Inspections," issued April 25, 2006.

41. NRC, "Revised Staff Technical Position on Waste Form (SP-91-13)," January 30, 1991.

42. NUREG-0016, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWRs) (BWR-GALE Code)."

43. NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWRs) (PWR-GALE Code)."

44. NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants."

45. NUREG-1301, "Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Pressurized Water Reactors."

46. NUREG-1302, "Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Boiling Water Reactors."

47. NUREG/BR-0204, "Instructions for Completing NRC's Uniform Low-Level Radioactive Waste Manifest."

48. IE Circular No. 77-10, "Vacuum Conditions Resulting in Damage to Liquid Process Tanks," July 15, 1977.

49. IE Circular No. 77-14, "Separation of Contaminated Water Systems from Noncontaminated Plant Systems," November 22, 1977.

50. IE Circular No. 79-21, "Prevention of Unplanned Releases of Radioactivity," October 19, 1979.

51. IE Circular No. 81-09, "Containment Effluent Water That Bypasses Radioactivity Monitor," July 10, 1981.

52. IE Information Notice No. 79-07, "Rupture of Radwaste Tanks," March 23, 1979.

53. IE Information Notice No. 79-09, "Spill of Radioactively Contaminated Resin," March 30, 1979.

54. IE Bulletin No. 79-19, "Packaging of Low-Level Radioactive Waste for Transportation and Burial," August 10, 1979.

55. IE Bulletin No. 80-10, "Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to Environment," May 6, 1980.

56. IE Information Notice No. 90-31, "Update on Waste Form and High Integrity Container Topical Report Review Status, Identification of Problems with Cement Solidification and Reporting of Waste Mishaps," May 4, 1990.

57. IE Information Notice No. 84-72, “Clarification of Conditions for Waste Shipments Subject to Hydrogen Gas Generation,” September 10, 1984.

58. IE Information Notice No. 87-07, “Quality Control of Onsite Dewatering/Solidification Operations by Outside Contractors,” February 3, 1987.

59. IE Information Notice No. 85-92, “Surveys of Wastes Before Disposal from Nuclear Reactor Facilities,” December 2, 1985.

60. IE Information Notice No. 86-20, “Low-Level Radioactive Waste Scaling Factors, 10 CFR Part 61,” March 28, 1986.

61. Memorandum from Larry W. Camper to David B. Matthews and Elmo E. Collins, dated October 10, 2006, “List of Decommissioning Lessons Learned in Support of the Development of a Standard Review Plan for New Reactor Licensing” (ADAMS Accession No. ML0619201830).

62. NUREG/CR-3587, “Identification and Evaluation of Facility Techniques for Decommissioning of Light Water Reactors.”

63. Office of Nuclear Reactor Regulation, “Liquid Radioactive Release Lessons Learned Task Force, Final Report,” Sections 2.0 and 3.2.2, September 1, 2006 (ADAMS Accession No. ML062650312).

64. ANSI/ANS-55.6-1993 (1999), "Liquid Radioactive Waste Processing System for Light Water Reactor Plants." Reaffirmed in 1999.

65. ANSI/ANS-40.37-1993 (200x updated draft), "American National Standard For Mobile Low-Level Radioactive Waste Processing Systems." Proposed 2007 draft for public comments.

66. SECY 93-323, “Withdrawal of Proposed Rulemaking to Establish Procedures and Criteria for On-Site Storage of low-Level Radioactive Waste After January 1, 1996,” Nov. 29, 1993. Issued under SRM dated Feb. 1, 1994.

67. NRC Inspection Manual Chapter IMC-2504, “Construction Inspection Program - Non-ITAAC Inspections,” issued April 25, 2006.

APPENDIX 11.4-A
DESIGN GUIDANCE FOR TEMPORARY STORAGE
OF LOW-LEVEL RADIOACTIVE WASTE

I. INTRODUCTION

The objective of this technical position is to provide guidance to licensees considering additional onsite low-level radioactive waste storage capabilities. While it may be prudent and/or necessary to establish additional onsite storage capability, waste should not be placed in contingency storage if it can be disposed at a licensed disposal site. Shipping waste at the earliest practicable time minimizes the need for eventual waste reprocessing caused by potential changes in a disposal facility's requirements, reduces occupational and nonoccupational exposures and potential accident consequences, and, in the event of burial ground closure, maximizes the amount of storage space available for use.

The duration of the intended storage, the type and form of waste, and the amount of radioactive material present will dictate the safeguards and the level of complexity required to assure public health and safety and minimal risk to operating personnel. The longer the intended storage period, the greater the degree of controls that will be required for radiation protection and accident prevention. The duration of the onsite storage safety hazard is predicated on the type of waste being stored, the amount of radionuclides present, and how readily the radionuclides might be transported into the environment. In general, it is preferable to store radioactive material in solid form. Under some circumstances, however, temporary storage in a liquid form may be desirable or required. The specific design and operation of any storage facility will be significantly influenced by the various waste forms; consequently, this document addresses wet waste, stabilized wet waste, and dry low-level radioactive waste.

II. GENERAL INFORMATION

Before implementing any additional onsite storage capacity, licensees should conduct substantial safety review and environmental assessments to assure adequate public health and safety protections and minimal environmental impact. The acceptance criteria and performance objectives of any proposed storage facility or area will need to meet minimal requirements in design, operations, safety considerations, policy considerations, and compliance with other applicable Federal, State, and local regulations governing any other toxic or hazardous properties of radioactive wastes (such as mixed wastes characterized by the presence of hazardous chemicals and radioactive materials). For purposes of this technical position, the major emphasis will be on safety considerations in the storing, handling, and eventual disposition of the radioactive waste. Design and operational acceptability will be based on minimal requirements, which are defined in existing SRPs, regulatory guides, and industry standards for proper management of radioactive waste. Considerations for waste minimization and volume reduction will also need to be part of an overall site waste management plan and the chosen onsite storage alternative. Licensees and applicants should implement additional waste management considerations for ALARA, decontamination, and decommissioning of the temporary storage facility, including disposal, as early as possible, because future requirements for waste forms may make stored wastes unacceptable for final disposition.

Facility design and operation should assure that radiological consequences of design basis events (e.g., fire, tornado, seismic occurrence, and flood) do not exceed a small fraction (10 percent) of 10 CFR Part 100 dose limits (i.e., no more than a few sieverts whole body dose). The added storage capacity should typically consider the anticipated low-level waste volumes generated over the operational life of the plant. Licensees should determine the design storage capacity (volume and radioactive material inventories) from historical and projected waste generation rates for all units, considering both volume minimization/reduction programs and the need for surge capacity due to operations which may generate unusually large amounts of waste. Further guidance is provided in Generic Letter No. 80-09, 81-38, and 81-39, and in SECY 94-198 and SECY 93-323. It should be noted that under SECY 94-198 and SECY 93-323, the provision requiring a Part 30 license for the storage of waste beyond 5 years has been eliminated. However, the balance of the technical information presented in Generic Letter No. 81-38 on the storage of low-level waste remains applicable for the purpose of this guidance.

In considering expanded storage capacity, licensees should consider the design and construction of additional volume reduction facilities (e.g., trash compactors, shredders, incinerators, etc.), as necessary, and then process wastes that may have been stored during their construction. Regional State low-level waste compacts and unaffiliated States may establish new or additional low-level waste disposal sites in the future under 10 CFR Part 61 or equivalent State regulations.

III. GENERALLY APPLICABLE GUIDANCE

1. The quantity of radioactive material allowed and the shielding configurations will be dictated by the dose rate criteria for both the site boundary and unrestricted areas or site. The 40 CFR Part 190 limits will restrict the annual dose from direct radiation and effluent releases from all sources of uranium fuel cycle, and 10 CFR 20.1302 limits the exposure rates in unrestricted areas. Offsite doses from onsite storage must be sufficiently low to account for other uranium fuel cycle sources (e.g., an additional dose of less than or equal to 0.01 mSv (1 mrem) per year is not likely to cause the 40 CFR Part 190 limits, as implemented under 10 CFR 20.1301(e) to be exceeded. Onsite dose limits associated with temporary storage will be controlled per 10 CFR Part 20, including the ALARA principle of 10 CFR 20.1101.

2. Compatibility of the container materials with the waste forms and with environmental conditions external to the containers is necessary to prevent significant container corrosion. Container selection should be based on data that demonstrate minimal corrosion from the anticipated internal and external environment for a period well in excess of the planned storage duration. Container integrity after the period of storage should be sufficient to allow handling during transportation and disposal without container breach.

   Gas generation from organic materials in waste containers can also lead to container breach and potentially flammable/explosive conditions. To minimize the number of potential problems, licensees should evaluate the waste form gas generation rates from radiolysis, biodegradation, or chemical reaction with respect to container breach and the creation of flammable or explosive conditions. Unless storage containers are equipped with special vent designs that allow depressurization and do not permit the migration of radioactive materials, resins highly loaded with radioactive material, such as BWR reactor water cleanup system resins, should not be stored for longer than approximately 1 year.

   Licensees should implement a program providing for at least periodic (quarterly) visual inspections of container integrity (e.g., swelling, corrosion products, leaks, or breach). Inspections can be accomplished by the use of television monitors; by walkthroughs if storage facility layout, shielding, and container storage array permit; or by selecting waste containers that are representative of the types of waste and containers stored in the facility and placing them in a location specifically designed for inspection purposes. All inspection procedures developed should minimize occupational exposure. The use of high-integrity containers (300-year lifetime design) would permit an inspection program of reduced scope.

3. If possible, the preferred location of the additional storage facility is inside the plant's protected area. If adequate space in the protected area is not available, the licensee should place the storage facility on the plant site and establish both a physical security program (fence, locked and alarmed gates and doors, and periodic patrols) and a restricted area for radiation protection purposes. The facility should not be in a location that requires transportation of the waste over public roads unless no other feasible alternatives exist. Licensees must conduct any transportation over public roads in accordance with the NRC and DOT regulations (10 CFR Part 71 and 49 CFR Parts 171–180).

4. Licensees should implement operational safety features to prevent the accidental dropping of containers from cranes and forklifts or the puncturing of containers from forklifts during the movement and transportation of radioactive waste containers. Personnel should receive training in the proper operation of such equipment and instruction on the use of methods to securely hold containers on such equipment (e.g., tie-downs, gates, cages).

5. The facility should include design features, in accordance with 10 CFR 20.1406, that would minimize, to the extent practicable, contamination of the waste facility and environment; facilitate eventual decommissioning; and minimize, to the extent practicable, the generation of extraneous radioactive waste. This requirement applies to storage facilities used to process and store liquid, wet, dry solid, and stabilized wastes.

6. For low-level dry waste and stabilized waste storage, the following criteria apply:

    A. Licensees shall monitor potential release pathways of all radionuclides present in the stabilized waste form as described in Appendix A to 10 CFR Part 50. Surveillance programs shall incorporate adequate methods for detecting failure of container integrity and measuring releases to the environment. For outside storage, licensees shall conduct periodic direct radiation and surface contamination monitoring to ensure that levels are below limits specified in 10 CFR 20.1301 and 10 CFR 20.1302, 10 CFR Part 71, and Subpart I (Class 7) of 49 CFR Part 173. All containers should be decontaminated to these or lower levels before storage.

    B. Licensees should incorporate provisions for collecting liquid drainage, including provisions for sampling all collected liquids. Routing of the collected liquids should be to radwaste systems if contamination is detected or to normal discharge pathways if the water ingress is from external sources and remains uncontaminated by plant-generated radioactivity.

    C. Waste stored in outside areas should be held securely by installed holddown systems. The holddown system should secure all containers during severe environmental conditions, up to and including the design-basis event for the waste storage facility.

D. Licensees should assure container integrity against corrosion from the external environment, including external weather protection where necessary and practical. Storage containers should be raised off storage pads where water accumulation can be expected to cause external corrosion and possible degradation of container integrity.

E. Licensees should establish total radioactive material inventory limits (in becquerels and curies), based on the design of the storage area, dose limits for members of the public, and safety features or measures being provided (e.g., radiation monitoring).

F. Licensees should maintain inventory records by waste types, waste contents, radionuclides and radioactive material, dates of storage, shipment, and other relevant data.

G. The facility design should incorporate provisions for a ventilation exhaust system (for storage areas) and an airborne radioactivity monitoring system (building exhaust vents) where there is a potential for airborne radioactivity to be generated or to accumulate.

IV. WET RADIOACTIVE WASTE STORAGE

1. Wet radioactive waste is defined as any liquid, liquid/solid slurry, or process concentrate. For storage considerations, wet waste is further defined as any waste that contains free liquid in amounts exceeding the requirements for burial as established by the burial ground licensing authority.

2. The design of the facility's supporting structure and tanks should prevent uncontrolled and unmonitored releases of radioactive materials resulting from spillage or accident conditions.

3. The following design objectives and criteria apply to wet radioactive waste storage facilities:

   A. Structures that house liquid radwaste storage tanks should be designed to seismic criteria as defined in SRP Section 11.2 and Regulatory Guide 1.143 for wastes produced during normal operation and anticipated operational occurrences. Foundations and walls shall also be designed and fabricated to contain the liquid inventory that might be released during a container/tank failure.

   B. All tanks or containers should be designed to withstand the corrosive nature of the wet waste being stored. The design shall also consider the duration of storage under which the corrosive conditions exist.

   C. All storage structures should have curbs or elevated thresholds with floor drains and sumps to safely collect wet waste in the event of the failure of all tanks or containers. There should be provisions to remove spilled wet waste to the radwaste treatment systems.

   D. All tanks and containers shall have provisions to monitor liquid levels and to sound an alarm in the event of potential overflow conditions.

E. All potential release pathways of radionuclides (e.g., evolved gases, breach of container) shall be controlled, if feasible, and monitored in accordance with Appendix A to 10 CFR Part 50, GDC 60 and 64. Surveillance programs should incorporate adequate methods for monitoring breach of container integrity or accidental releases.

F. All temporarily stored wet waste will require additional reprocessing before shipment off site; therefore, provisions should be made to integrate the required treatment with the waste processing and stabilization systems. The interface and associated systems should be designed and tested in accordance with the codes and standards described in SRP Section 11.2 and Regulatory Guide 1.143 for wastes produced during normal operation and anticipated operational occurrences.

G. The facility design should include provisions for a ventilation exhaust system (for storage areas) and an airborne radioactivity monitoring system (building exhaust vents) where there is a potential for airborne radioactivity to be generated or to accumulate.

V. <u>STABILIZED RADIOACTIVE WASTE STORAGE</u>

1. Stabilized radwaste for storage purposes is defined as waste that meets stabilized waste criteria for licensed facilities. For purposes of this document, resins or filter sludge dewatered to the above criteria are defined under this waste classification/criteria.

2. Any storage plans should address container protection and any reprocessing requirements for eventual shipment and burial.

3. Casks, tanks, and liners containing stabilized radioactive waste should be designed with good engineering judgment to preclude or reduce the probability of uncontrolled releases of radioactive materials during handling, transportation, or storage. Licensees must evaluate the accident mitigation and control procedures and their ability to protect the facility from design basis events (e.g., fire, flooding, tornadoes) unless otherwise justified.

4. The following design objectives and criteria are applicable to stabilized waste storage containers and facilities:

   A. All stabilized radwaste should be located in restricted areas where effective material control and accountability can be maintained. While structures are not required to meet seismic criteria, licensees should employ good engineering judgment to ensure that radioactive materials are contained safely, such as by the use of curbs and drains to contain spills of dewatered resins or sludge.

   B. If liquids exist in a corrosive form, licensees should implement proven measures to protect the container (i.e., special liners or coatings) and/or neutralize the excess liquids. If deemed appropriate and necessary, highly noncorrosive materials (e.g., stainless steel) should be used. Potential corrosion between the solid waste forms and the container should also be considered. In the case of dewatered resins, highly corrosive acids and bases can be generated, which will significantly reduce the longevity of the container. The PCP should implement steps to assure the above does not occur; provisions should be made to govern container material selection and precoating to ensure that container breach does not occur during temporary storage periods.

C. There should be provisions for additional reprocessing or repackaging in the event of container failure and/or as required by DOT regulations and license disposal facility criteria for final transportation and disposal. Licensees should develop contamination isolation and decontamination capabilities. When significant handling and personnel exposure can be anticipated, licensees should incorporate ALARA methodology in accordance with Regulatory Guides 8.8 and 8.10.

D. Licensees should develop and implement procedures for early detection, prevention, and mitigation of accidents (e.g., fires). Storage areas and facility designs should incorporate good engineering features and capabilities for handling accidents and provide safeguard systems, such as fire detectors and suppression systems (e.g., smoke detectors and sprinklers). If water sprinkler systems are used, floors should be sloped to drain into local floor sumps or curbed to prevent water runoff to uncontrolled areas. Licensees should establish personnel training and administrative procedures to ensure both control of radioactive materials and minimum personnel exposures. Fire suppression devices may not be necessary if combustible materials in the area are minimal.

E. The facility design should incorporate provisions for a ventilation exhaust system (for storage areas) and an airborne radioactivity monitoring system (building exhaust vents) where there is a potential for airborne radioactivity to be generated or to accumulate.

VI. LOW-LEVEL DRY WASTE STORAGE

1. Low-level dry waste is classified as contaminated material (e.g., paper, trash, plastics, glass, metals scraps, air filters, and spent charcoal media) that contains radioactive materials dispersed randomly in relatively small concentrations throughout large volumes of inert material and contains no free water. Generally, this consists of dry materials, such as rags, clothing, paper, and small equipment (i.e., tools and instruments), that cannot be easily decontaminated.

2. Licensees should implement controls to segregate and minimize the generation of low-level dry waste to lessen the impact on waste storage. Licensees should consider the integration of volume reduction hardware to minimize the need for additional waste storage facilities.

3. The following design objectives and criteria are applicable for low-level dry waste storage containers and facilities:

   A. All dry or compacted radwaste should be located in restricted areas where effective material control and accountability can be maintained. While structures are not required to meet seismic criteria, licensees should use good engineering judgment to ensure the radioactive material is contained safely.

B. The waste container design should ensure radioactive material containment during normal and abnormal occurrences. The waste container materials should not support combustion. The packaged material should not cause fires through spontaneous chemical reactions, retained heat, or the like.

C. Containers should generally comply with the criteria of 10 CFR Part 71 and 49 CFR Parts 171–180 to minimize the need for repackaging for shipment.

D. Increased container handling and personnel exposure can be anticipated; consequently, licensees should incorporate all ALARA methodology in accordance with Regulatory Guides 8.8 and 8.10.

E. Facility design should provide for a ventilation exhaust system (for storage areas) and an airborne radioactivity monitoring system (building exhaust vents) where there is a potential for airborne radioactivity to be generated or to accumulate.

VII. REFERENCES

1. 10 CFR Part 20, "Standards for Protection Against Radiation."

2. 10 CFR 20.1301, "Dose Limits for Individual Members of the Public."

3. 10 CFR 20.1302, "Compliance with Dose Limits for Individual Members of the Public."

4. 10 CFR 20.1406, "Minimization of Contamination."

5. 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material."

6. 10 CFR Part 50, Appendix A, General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment."

7. 10 CFR Part 50, Appendix A, General Design Criterion 64, "Monitoring Radioactivity Releases."

8. 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste."

9. 10 CFR Part 71, "Packaging and Transportation of Radioactive Material."

10. 10 CFR Part 100, "Reactor Site Criteria."

11. 49 CFR Parts 171–180, "Subpart C—Hazardous Materials Regulations."

12. Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants."

13. Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable."

14. Regulatory Guide 8.10, “Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable.”

15. Generic Letter 80-009, “Low Level Radioactive Waste Disposal.”

16. Generic Letter 81-038, “Storage of Low Level Radioactive Wastes at Power Reactor Sites.”

17. Generic Letter 81-039, “NRC Volume Reduction Policy.”

18. SECY 93-323, “Withdrawal of Proposed Rulemaking to Establish Procedures and Criteria for On-Site Storage of low-Level Radioactive Waste After January 1, 1996,” Nov. 29, 1993. Issued under SRM dated Feb. 1, 1994.

19. SECY 94-198, “Review of Existing Guidance Concerning the Extended Storage of Low-Level Radioactive Waste, Aug. 1, 1994.”

ATTACHMENT 5

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

December 30, 2008

NRC REGULATORY ISSUE SUMMARY 2008-32
INTERIM LOW LEVEL RADIOACTIVE WASTE STORAGE AT REACTOR SITES

ADDRESSEES

All holders of operating licenses for nuclear power reactors, including those that have permanently ceased operations, and for research and test reactors.

INTENT

The U.S. Nuclear Regulatory Commission (NRC) is issuing this regulatory issue summary (RIS) to clarify the current NRC staff position regarding the long-term, interim storage of low level radioactive waste (LLRW) at facilities licensed under Title 10, Part 50, "Domestic Licensing of Production and Utilization Facilities," of the *Code of Federal Regulations* (10 CFR) and to provide an acknowledgement, with certain conditions, of the proposed NEI/EPRI Guidelines for Operating an Interim On-Site Low-Level Waste Storage Facility, Final Draft, April 2008.

As of July 1, 2008, LLRW generators in 36 States are no longer able to ship Class B and C LLRW to a disposal facility. Therefore, facilities in those States will have to store their Class B and C LLRW for an indeterminate amount of time. Since 1981, NRC has issued a number of generic communications containing information related to interim LLRW storage. This RIS will consolidate relevant information and clarify past positions. This RIS requires no action or written response on the part of the addressees.

BACKGROUND INFORMATION

The Low Level Radioactive Waste Policy Act of 1980 made States responsible for disposing of LLRW generated by commercial entities within their State. The Act also encouraged the States to form regional compacts. To date, there are 10 Compacts and all but 7 States are a member of a compact. The States that are not affiliated with a compact are Maine, Massachusetts, Michigan, New Hampshire, New York, North Carolina, and Rhode Island. The Low-Level Radioactive Waste Policy Amendments Act of 1985 established milestones, penalties and incentives for States or regional compacts to develop their own low-level waste disposal facilities. Currently, there are three operating LLRW disposal facilities in the United States, located in Barnwell, South Carolina (Barnwell), Clive, Utah (Clive) and Richland, Washington (Richland). LLRW is defined in 10 CFR 61.2. Per 10 CFR 61.55, LLRW is classified as Class A, B, or C. Class A waste makes up approximately 99 percent of the LLRW and has the lowest level of radioactivity. Class A waste usually consists of slightly contaminated paper products and clothing, rags, mops, equipment and tools, and filters with low levels of radioactivity. While

Class B and C waste makes up approximately one percent of the LLRW, it has a higher level of radioactivity. Class B and C usually consist of materials such as filters, resins, and irradiated hardware.

The Clive facility only accepts LLRW in waste Class A. All LLRW generators in the United States may ship Class A waste to Clive for disposal, subject to waste acceptance criteria and some compact constraints. The Richland facility only accepts Class A, B and C LLRW from waste generators in the Northwest Compact (WA, OR, ID, MT, UT, WY, AK, and HI) and the Rocky Mountain Compact (NV, CO, and NM). As of July 1, 2008, the Barnwell facility will only accept Class, A, B, and C LLRW generated in States that are members of the Atlantic Compact (SC, NJ, and CT).

SUMMARY OF ISSUE

Since July 1, 2008, LLRW generators in the District of Columbia, the Commonwealth of Puerto Rico, the U.S. Territories, and in the 36 States not part of the Atlantic, Northwest or Rocky Mountain Compacts have no available disposal facility for their Class B and C waste. These LLRW generators will now have to store the LLRW on-site for an indeterminate amount of time.

Previous Information

Since 1981, the NRC has issued a number of generic communications providing information for storing LLRW on licensees' sites. The following is a summary of documents that specifically address interim storage of LLRW on reactor sites.

Generic Letter (GL) 81-38, "Storage of Low-Level Radioactive Wastes at Power Reactor Sites": The NRC issued GL 81-38 in November 1981 as a result of a reduction in the availability of waste disposal in the United States when three disposal sites permanently closed. GL 81-38 informed licensees that if the on-site LLRW storage capacity was to be increased, then the licensee must perform an evaluation under the provisions of 10 CFR 50.59, "Changes, tests, and experiments." If an unreviewed safety question was identified as a result of the evaluation, then the licensee was to apply to the NRC for a license under the provisions of 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material." GL 81-38 stated that the 10 CFR Part 30 license was for the administrative convenience of the Commission and was not intended to be substantively different than an application for amending the 10 CFR Part 50 license. The 10 CFR Part 30 license would be issued for a 5-year term and could be renewed for additional 5-year terms if the need for on-site LLRW continued. GL 81-38 also provided guidance to be used in the design, construction and operation of the LLRW storage facility.

GL 85-14, "Commercial Storage at Power Reactor Sites of Low-Level Radioactive Waste Not Generated by the Utility": The NRC issued GL 85-14 in expectation that no new LLRW disposal facilities would be available for several years. GL 85-14 provided guidance for licensee requests to store LLRW at reactor sites, including storage of LLRW generated elsewhere. GL 85-14 stated that, as a matter of policy, the NRC is opposed to any activity at a nuclear reactor site which is not generally supportive of activities authorized by the operating license or construction permit and which may divert the attention of licensee management from its primary task of safe operation or construction of the power reactor. Accordingly, GL 85-14 determined that interim storage of LLW within the exclusion area of a reactor site, as defined in 10 CFR 100.3(a), was subject to NRC jurisdiction regardless of whether or not the reactor was located in an Agreement State. GL 85-14 reiterated that a Part 30 license is required for LLRW storage and that an amendment to the 10 CFR Part 50 license may also be required. GL 85-14 described the criteria a licensee application for LLRW storage must meet, including a determination by the utility licensee that the proposed LLRW commercial storage activities do not involve a safety or environmental question, and that safe operation of the reactor will not be affected.

Information Notice (IN) 89-13, "Alternative Waste Management Procedures in Case of Denial of Access to Low-Level Waste Disposal Sites": The NRC issued IN 89-13 in February 1989 to address the possibility of restrictions for disposing of LLRW, particularly for licensees in Vermont, New Hampshire and Michigan. IN 89-13 also provided suggestions on ways to minimize possible adverse consequences of interim storage by minimizing the waste generated on-site. Suggested actions included evaluating potential safety problems and technical difficulties arising from long term storage, reviewing ways to minimize waste generation, and reviewing alternative waste management and disposal methods.

SECY-94-198, "Review of Existing Guidance Concerning the Extended Storage of Low-Level Radioactive Waste" (ML071640462): SECY-94-198 consolidated previous staff guidance and clarified that 10 CFR Part 50 licensees no longer have to apply for a 10 CFR Part 30 license to store LLRW because they are already authorized under Part 30, within the limits of their Part 50 operating licenses, to possess and store LLRW on-site.[1] In the event that the storage of LLRW was not within the limits of a given facility operating license, SECY-94-198 stated that the licensee should seek to amend its Part 50 license. For power reactor licensees, SECY-94-198 also eliminated the five-year limit for on-site storage of LLRW generated at the site. SECY-94-198 also clarified that a 10 CFR 50.59 evaluation was not required for LLRW storage in those instances where no changes in the facility or procedures as described in the safety analysis report are involved. The paper also stated that LLRW should be stored safely and that containers for interim long term storage of LLRW should be compatible with the waste type and possible environmental factors to prevent container corrosion. Additionally, the LLRW should be stored in such a manner as to prevent potential gas generation from processes such as radiolysis, biodegradation, or chemical reaction.

On-site Storage Considerations

Since July 1, 2008, licensees in 36 States have had to store their Class B and C waste on-site. The operation of a licensee's on-site LLRW storage facility must comply with the requirements in 10 CFR Part 20, "Standards for Protection Against Radiation," including 10 CFR 20.1801, "Security of Stored Material," which requires that licensed materials stored in controlled or unrestricted areas be secured from unauthorized removal or access. Also, under Part 20 requirements, licensees storing LLRW on reactor sites for an indefinite period of time must

[1] SECY-94-198 noted that "commercial storage of [LLRW] generated by other licensees on the reactor site would still require a separate Part 30 license for the operation of that facility."

ensure that, in connection with such LLRW storage, occupational doses are as low as is reasonably achievable and that doses to individual members of the public are within regulatory limits. In addition, licensees must ensure that the storage of LLRW has been accounted for in their Part 20 radiation protection programs, including meeting the requirements for surveys and monitoring, labeling, and reports and record retention.

When evaluating interim long-term on-site LLRW storage, Part 50 licensees must consider the applicability of the general design criteria listed in Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50, specifically Criteria 61, 63 and 64. Criterion 61, “Fuel Storage and Handling and Radioactivity Control,” specifies that fuel storage and handling, radioactive waste and other systems that may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. Criterion 63, “Monitoring Fuel and Waste Storage,” states that appropriate systems shall be provided in fuel storage, radioactive waste systems, and associated handling areas to (1) detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions. Criterion 64, “Monitoring Radioactivity Releases,” specifies that there must be a method for monitoring the level of radioactivity in effluent release pathways and to the plant environs.

In 2007, the NRC revised NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” in anticipation of receiving new reactor license applications. While NUREG-0800 was revised and updated in anticipation for new license applications, it is also used by staff during license amendment reviews for operating plants. Chapter 11.4, “Solid Waste Management System,” specifies the information that NRC staff has determined should be included in a Construction and Operating License Application. Appendix 11.4-A, “Design Guidance for Temporary Storage of Low-Level Radioactive Waste” provides specific guidance to licensees for increasing on-site LLRW storage capacity.

Proposed EPRI Guidelines

In May 2008, the Nuclear Energy Institute submitted the draft report, “Guidelines for Operating an Interim On-site Low Level Radioactive Waste Storage Facility, Final Draft, April 2008,” prepared by the Electric Power Research Institute. This report, known as the Guidelines Report, includes guidance for licensees on recordkeeping, waste containers and waste forms, monitoring and inspecting, and on combining Class B and C waste into greater than Class C (GTCC) waste for extended on-site storage for LLRW. With the exception of the section on combining B and C class waste into GTCC, the NRC staff finds the guidelines to be consistent with NRC information contained in this RIS and other NRC guidance such as NUREG-0800. The Guidelines Report provides an acceptable method for recordkeeping, determining waste forms and waste containers and monitoring and inspecting the interim long-term storage of LLRW. While NRC has indicated that volume reduction of LLRW is generally appropriate, NRC has not developed a position on combining Class B and C waste together to form GTCC waste.

Summary

With the access to Barnwell now being limited to only licensees in States that are members of the Atlantic Compact, clarification of applicable NRC information was appropriate. This RIS consolidates relevant information on interim long-term storage of LLRW. Of note, Part 50 licensees do not have to obtain a separate Part 30 license for on-site storage of LLRW generated at that site, and therefore, the 5-year limit on storing such LLRW on-site remains not applicable.

BACKFIT DISCUSSION

This RIS reiterates the current staff position that there is no need for power reactor licensees to obtain a Part 30 license for storing LLRW generated at the site for a duration greater than 5 years. Previously, GL 81-38 indicated that a licensee may need a Part 30 license for storage of LLRW when the storage time duration would exceed 5 years. In 1993, the staff proposed rulemaking requiring the need for the Part 30 license for storage of LLRW. This rulemaking effort was withdrawn since Part 30 already allows for LLRW storage at Part 50 licensed facilities with no time limit. In response to the staff's proposal to withdraw this rulemaking, the Commission issued a Staff Requirements Memorandum on February 1, 1994, which directed the staff to establish guidance identifying that a Part 30 License was not required for Part 50 Licensees. These efforts established the current staff position. This RIS requires no action or response. This RIS does not impose a regulatory staff position interpreting Commission rules that is either new or different from a previously applicable staff position and, therefore, it is not a backfit as defined by 10 CFR 50.109. Consequently, the staff did not perform a backfit analysis.

***FEDERAL REGISTER* NOTIFICATION**

A notice of opportunity for public comment on this RIS was not published in the *Federal Register* because the RIS is informational and does not represent a departure from current regulatory requirements. However, a public meeting to discuss the RIS and obtain comments from interested parties was held on September 10, 2008. The meeting summary is available under ADAMS accession number ML082540738.

CONGRESSIONAL REVIEW ACT

This RIS is not a rule as designated by the Congressional Review Act (5 U.S.C. §§ 801-886) and therefore, is not subject to the Act.

Paperwork Reduction Act Statement

This RIS contains information collection requirements that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.) These information collections were approved by the Office of Management and Budget, approval numbers 3150-0011 and 3150-0014.

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting documents displays a currently valid OMB control number.

CONTACT

This RIS requires no specific action or written response. Please direct any questions about this matter to the technical contact listed below.

/RA/

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301-415-8517
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Enclosure:
References

Note: NRC generic communications may be found at the NRC public website at http://www.nrc.gov under Electronic Reading Room/Document Collections.

CONTACT

This RIS requires no specific action or written response. Please direct any questions about this matter to the technical contact listed below.

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REFERENCES

NUREG-0800, Standard Review Plan for the Review of Safety Reports for Nuclear Power Plants, March 2007.

Generic Letter GL-81-38, Storage of Low-level Radioactive Wastes at Power Reactor Sites. November 10, 1981.

Guidelines for Operating an Interim On Site Low Level Radioactive Waste Storage Facility, EPRI, Final Draft April 2008.

Information Notice 89-13. Alternative Waste Management Procedures in Case of Denial of Access to Low-level Waste Disposal Sites. February 8, 1989

Generic Letter GL-85-14, Commercial Storage at Power Reactor Sites of Low level Radioactive Waste not Generated by the Utility. August 1, 1985.

SECY-94-198, Review of Existing Guidance Concerning the Extended Storage of Low-Level Radioactive Waste. August 1, 1994

ATTACHMENT 6

11.4 Solid Waste Management

The solid waste management system (WSS) is designed to collect and accumulate spent ion exchange resins and deep bed filtration media, spent filter cartridges, dry active wastes, and mixed wastes generated as a result of normal plant operation, including anticipated operational occurrences. The system is located in the auxiliary and radwaste buildings. Processing and packaging of wastes are by mobile systems in the auxiliary building rail car bay and in the mobile systems facility part of the radwaste building. The packaged waste is stored in the auxiliary and radwaste buildings until it is shipped offsite to a licensed disposal facility.

The use of mobile systems for the processing functions permits the use of the latest technology and avoids the equipment obsolescence problems experienced with installed radwaste processing equipment. The most appropriate and efficient systems may be used as they become available.

This system does not handle large, radioactive waste materials such as core components or radioactive process wastes from the plant's secondary cycle. However, the volumes and activities of the secondary cycle wastes are provided in this section.

11.4.1 Design Basis

11.4.1.1 Safety Design Basis

The solid waste management system performs no function related to the safe shutdown of the plant. The system's failure does not adversely affect any safety-related system or component; therefore, the system has no nuclear safety design basis.

There are no safety related systems located near heavy lifts associated with the solid waste management system. Therefore, a heavy loads analysis is not required.

11.4.1.2 Power Generation Design Basis

The solid waste management system provides temporary onsite storage for wastes prior to processing and for the packaged wastes. The system has a 60-year design objective and is designed for maximum reliability, minimum maintenance, and minimum radiation exposure to operating and maintenance personnel. The system has sufficient temporary waste accumulation capacity based on maximum waste generation rates so that maintenance, repair, or replacement of the solid waste management system equipment does not impact power generation.

11.4.1.3 Functional Design Basis

The solid waste management system is designed to meet the following objectives:

- Provide for the transfer and retention of spent radioactive ion exchange resins and deep bed filtration media from the various ion exchangers and filters in the liquid waste processing, chemical and volume control, and spent fuel cooling systems

- Provide the means to mix, sample, and transfer spent resins and filtration media to high integrity containers or liners for dewatering or solidification as required

- Provide the means to change out, transport, sample, and accumulate filter cartridges from liquid systems in a manner that minimizes radiation exposure of personnel and spread of contamination

- Provide the means to accumulate spent filters from the plant heating, ventilation, and air-conditioning systems

- Provide the means to segregate solid wastes (trash) by radioactivity level and to temporarily store the wastes

- Provide the means to accumulate radioactive hazardous (mixed) wastes

- Provide the means to segregate clean wastes originating in the radiologically controlled area (RCA)

- Provide the means to store packaged wastes for at least 6 months in the event of delay or disruption of offsite shipping

- Provide the space and support services required for mobile processing systems that will reduce the volume of and package radioactive solid wastes for offsite shipment and disposal according to applicable regulations, including Department of Transportation regulation 49 CFR 173 (Reference 1) and NRC regulation 10 CFR 71 (Reference 2)

- Provide the means to return liquid radwaste to the liquid radwaste system (WLS) for subsequent processing and monitored discharge

The solid waste management system is designed according to NRC Regulatory Guide 1.143 to meet the requirements of General Design Criterion (GDC) 60 as discussed in Sections 1.9 and 3.1. The seismic design classifications of the radwaste building and system components are provided in Section 3.2.

Provisions are made in the auxiliary and radwaste buildings to use mobile radwaste processing systems for processing and packaging each waste stream including concentration and solidification of chemical wastes from the liquid waste management system, spent resin dewatering, spent filter cartridge encapsulation and dry active waste sorting and compaction.

The radioactivities of influents to the solid waste management system are based on estimated radionuclide concentrations and volumes. These estimates are based on operating plant experience, adjusted for the size and design differences of AP1000. The influent source terms are consistent with Section 11.1.

The solid waste management system airborne process effluents are released through the monitored plant vent as described as part of the 10 CFR 50 (Reference 3), Appendix I, analysis presented in subsection 11.3.3.

The solid waste management system collects and stores radioactive wastes within shielding to maintain radiation exposure to plant operation and maintenance personnel as low as is reasonably achievable (ALARA) according to General Design Criteria 60 as discussed in Section 3.1 and Regulatory Guide 8.8. Personnel exposures will be maintained well below the limits of 10 CFR 20 (Reference 4). Design features incorporated to maintain exposures ALARA include remote and semi-remote operations, automatic resin transport line flushing, and shielding of components, piping and containers holding radioactive materials. Access to the solid waste storage areas is controlled, to minimize inadvertent personnel exposure, by suitable barriers such as heavy storage cask covers and locked or key-card-operated doors or gates (see Section 12.1).

The solid waste management system conforms to the design criteria of NRC Branch Technical Position ETSB 11-3. Suitable fire protection systems are provided as described in subsection 9.5.1.

Waste disposal containers are to be selected from available designs that meet the requirements of the DOT and NRC. The solid waste management system does not require source-specific waste containers. Waste containers must meet the regulatory requirements for radioactive waste transportation in 49 CFR 173 and for radioactive waste disposal in 10 CFR 61 (Reference 5) as well as specific disposal facility requirements.

11.4.1.4 Compliance with 10 CFR 20.1406

In accordance with the requirements of 10 CFR 20.1406 (Reference 11), the solid radwaste system is designed to minimize, to the extent practicable, contamination of the facility and the environment, facilitate decommissioning, and minimize, to the extent practicable, the generation of radioactive waste. This is done through appropriate selection of design technology for the system, plus incorporating the ability to update the system to use the best available technology throughout the life of the plant.

11.4.2 System Description

11.4.2.1 General Description

The solid waste management system includes the spent resin system. The flows of wastes through the solid waste management system are shown on Figure 11.4-1. The radioactivity of influents to the system are dependent on reactor coolant activities and the decontamination factors of the processes in the chemical and volume control system, spent fuel cooling system, and the liquid waste processing system.

The parameters used to calculate the estimated activity of the influents to the solid waste management system are listed in Table 11.4-1. The estimated expected isotopic curie content of the primary spent resin and filter cartridge wastes to be processed on an annual basis is listed on Table 11.4-2. Table 11.4-3 provides the same information for the estimated maximum annual activities. The AP1000 has sufficient radwaste storage capacity to accommodate the maximum generation rate.

The radioactivity of the dry active waste is expected to normally range from 0.1 curies per year to 8 curies per year with a maximum of about 16 curies per year. This waste includes spent HVAC filters, compressible trash, non-compressible components, mixed wastes and solidified chemical wastes. These activities are produced by relatively long lived radionuclides (such as Cr-51, Fe-55, Co-58, Co-60, Nb-95, Cs-134 and Cs-137), and therefore, radioactivity decay during processing and storage is minimal. These activities thus apply to the waste as generated and to the waste as shipped.

The estimated expected and maximum annual quantities of waste influents by source and form are listed in Table 11.4-1 with disposal volumes. The annual radwaste influent rates are derived by multiplying the average influent rate (e.g. volume per month, volume per refueling cycle) by one year of time. The annual disposal rate is determined by applying the radwaste packaging efficiency to the annual influent rate. The influent volumes are conservatively based on an 18-month refueling cycle. Annual quantities based on a 24-month refueling cycle are less than those for an 18-month cycle. The estimated expected isotopic curie content of the primary spent resin and filter cartridge wastes to be shipped offsite are presented in Table 11.4-4 based on 90 days of decay before shipment. The same information is presented in Table 11.4-5 for the estimated maximum activities based on 30 days of decay before shipment.

Section 11.1 provides the bases for determination of liquid source terms used to calculate several of the solid waste management system influent source terms. The influent data presented in Tables 11.4-2 and 11.4-3 are conservatively based on Section 11.1 design basis (Technical Specification) values.

All radwaste which is packaged and stored by AP1000 will be shipped for disposal. The AP1000 has no provisions for permanent storage of radwaste. Radwaste is stored ready for shipment. Shipped volumes of radwaste for disposal are estimated in Table 11.4-1 from the estimated expected or maximum influent volumes by making adjustments for volume reduction processing by mobile systems and the expected container filling efficiencies. For drum compaction, the overall volume reduction factor, including packaging efficiency, is 3.6. For box compaction, the overall volume reduction factor is 5.4. These adjustments result in a packaged internal waste volume for each waste source, and the number of containers required to hold this volume is based on the container's internal volume. The disposal volume is based on the number of containers and the external (disposal) volume of the containers.

The expected disposal volumes of wet and dry wastes are approximately 547 and 1417 cubic feet per year, respectively as shown in Table 11.4-1. The wet wastes shipping volumes include 510 cubic feet per year of spent ion exchange resins and deep bed filter activated carbon, 20 cubic feet of volume reduced liquid chemical wastes and 17 cubic feet of mixed liquid wastes. The spent resins and activated carbon are initially stored in the spent resin storage tanks located in the rail car bay of the auxiliary building. When a sufficient quantity has accumulated, the resin is sluiced into two 158 cubic feet high-integrity containers in anticipation of transport for offsite disposal. Liquid chemical wastes are reduced in volume and packaged into three 55-gallon drums per year (about 20 cubic feet) and are stored in the packaged waste storage room of the radwaste building. The mixed liquid wastes fill less than three drums per year (about 17 cubic feet per year) and are stored on containment pallets in the waste accumulation room of the radwaste building until shipped offsite for processing.

The two spent resin storage tanks (275 cubic feet usable, each) and one high integrity container in the spent resin waste container fill station at the west end of the rail car bay of the auxiliary building provide more than a year of spent resin storage at the expected rate, and several months of storage at the maximum generation rate. The expected radwaste generation rate is based upon the following:

- All ion exchange resin beds are disposed and replaced every refueling cycle.
- The WGS activated carbon guard bed is replaced every refueling cycle.
- The WGS delay beds are replaced every ten years.
- All wet filters are replaced every refueling cycle.
- Rates of compactible and non-compactible radwaste, chemical waste, and mixed wastes are estimated using historical operating plant data.

The maximum radwaste generation rate is based upon the following:

- The ion exchange resin beds are disposed based upon operation with 0.25% fuel defects.
- The WGS activated carbon guard bed is replaced twice every refueling cycle.
- The WGS delay beds are replaced every five years.
- All wet filters are replaced based upon operation with 0.25% fuel defects.
- The expected rates of compactible and non-compactible radwaste, chemical waste, and mixed wastes are increased by about 50%.
- Primary to secondary system leakage contaminates the condensate polishing system and blowdown system resins and membranes which are replaced.

The dry solid radwaste includes 1383 cubic feet per year of compactible and non-compactible waste packed into about 14 boxes (90 cubic feet each) and ten drums per year. Drums are used for higher activity compactible and non-compactible wastes. Compactible waste includes HVAC exhaust filter, ground sheets, boot covers, hair nets, etc. Non-compactible waste includes about 60 cubic feet per year of dry activated carbon and other solids such as broken tools and wood. Solid mixed wastes will occupy 7.5 cubic feet per year (one drum). The low activity spent filter cartridges may be compacted to fill about 0.40 drums per year (3 ft^3/year) and are stored in the packaged waste storage room. Compaction is performed by mobile equipment or is performed offsite. High activity filter cartridges fill three drums per year (22.5 cubic feet per year) and are stored in portable processing or storage casks in the rail car of the auxiliary building.

The total volume of radwaste to be stored in the radwaste building packaged waste storage room is 1417 cubic feet per year at the expected rate and 2544 cubic feet per year at the maximum rate. The compactible and non-compactible dry wastes, packaged in drums or steel boxes, are stored with the mixed liquid and mixed solid, volume reduced liquid chemical wastes, and the lower activity filter cartridges. The quantities of liquid radwaste stored in the packaged waste storage room of the radwaste building consist of 20 cubic feet of chemical waste and 17 cubic feet of mixed liquid waste. The useful storage volume in the packaged waste storage room is approximately 3900 cubic feet (10 feet deep, 30 feet long, and 13 feet high), which accommodates more than one full offsite waste shipment using a tractor-trailer truck. The packaged waste storage room provides storage for more than two years at the expected rate of generation and more than a year at the maximum rate of generation. One four-drum containment pallet provides more than 8 months of storage capacity for the liquid mixed wastes and the volume reduced liquid chemical wastes at the expected rate of generation and more than 4 months at the maximum rate.

A conservative estimate of solid wet waste includes blowdown material based on continuous operation of the steam generator blowdown purification system, with leakage from the primary to secondary system. The volume of radioactively contaminated material from this source is estimated to be 540 cubic feet per year. Provisions for processing and disposal of radioactive steam generator blowdown resins and membranes are described in subsection 10.4.8. Note that, although included here for conservatism, this volume of contaminated resin will be removed from the plant within the contaminated electrodeionization unit and not stored as wet waste.

The condensate polishing system includes mixed bed ion exchanger vessels for purification of the condensate as described in subsection 10.4.6. Should the resins become radioactive, the resins are transferred from the condensate polishing vessel directly to a temporary processing unit or to the temporary processing unit via the spent resin tank. The processing unit, located outside of the turbine building, dewaters and processes the resins as required for offsite disposal. Radioactive condensate polishing resin will have very low activity. It will be disposed in containers as permitted by DOT regulations. After packaging, the resins may be stored in the radwaste building. Based on a typical condensate polishing system operation of 30 days per refueling cycle with leakage from the primary system to the secondary system, the volume of radioactively contaminated resin is estimated to be 206 cubic feet per year (one 309 cubic foot bed per refueling cycle). Normal disposal of nonradioactive condensate polishing system resins is described in subsection 10.4.6.

The parameters used to calculate the activities of the steam generator blowdown solid waste and condensate polishing resins are given in Table 11.4-1. Based on the above volumes, the disposal volume is estimated to be 939 cubic feet per year. The expected and maximum activities of the resins as generated are given in Tables 11.4-6 and 11.4-7, respectively. The expected and maximum activities of resins as shipped, based on 90 days decay prior to shipment, are given in Tables 11.4-8 and 11.4-9, respectively.

11.4.2.2 Component Description

The seismic design classification and safety classification for the solid waste management system components are listed in Section 3.2. The components listed are located in the seismic Category I Nuclear Island. Table 11.4-10 lists the solid waste management system equipment design parameters. The following subsections provide a functional description of the major system components.

11.4.2.2.1 Spent Resin Tanks

The spent resin tanks provide holdup capacity for spent resin and filter bed media decay before processing. High- and low-activity resins may be mixed to limit the radioactivity concentration in the waste containers to 10 Ci/ft^3 in accordance with the USNRC Technical Position on Waste Form (Reference 6).

Resin mixing capability is provided by mixing eductors in each tank, and resin dewatering, air sparging and complete draining capabilities are also provided. The ultrasonic level sensors and dewatering screens are arranged for remote removal. The vent and overflow connections have screens to prevent the inadvertent discharge of spent resin, and they are routed to the radioactive waste drain system (WRS).

11.4.2.2.2 Resin Mixing Pump

The resin mixing pump provides the motive force to fluidize and mix the resins in the spent resin tanks, to transfer water between spent resin tanks, to discharge excess water from the spent resin tanks to the liquid waste processing system, and to flush the resin transfer lines.

11.4.2.2.3 Resin Fines Filter

The resin fines filter minimizes the spread of high-activity resin fines and dislodged crud particles by filtering the water used for line flushing or discharged from the spent resin tanks to the liquid waste processing system.

11.4.2.2.4 Resin Transfer Pump

The resin transfer pump provides the motive force for recirculation of spent resins via either one of the spent resin tanks for mixing and sampling, for transferring spent resin between tanks, and for blending high- and low-activity resins to meet the specific activity limit for disposal. The resin transfer pump is also used to transfer spent resins to a waste container in the fill station or in its shipping cask located in the auxiliary building rail car bay.

11.4.2.2.5 Resin Sampling Device

The resin sampling device collects a representative sample of the spent resin either during spent resin recirculation or during spent resin waste container filling operations. A portable shielded cask is provided for sample jar transfer.

11.4.2.2.6 Filter Transfer Cask

The filter transfer cask permits remote changing of filter cartridges, dripless transport to the storage area in the auxiliary building, transfer of the filter cartridges into and out of the filter storage, and loading of the filter cartridges into disposal containers.

11.4.2.3 System Operation

11.4.2.3.1 Spent Resin Handling Operations

Demineralized water is used to transfer spent resins from the various ion exchangers to the spent resin tanks. A demineralized water transfer pump provides the pressurized water flow to transfer the spent resins as described in subsection 9.2.4. Before the transfer operation, it is verified that the selected spent resin tank is aligned as a receiver and has the capacity to accept the bed. It is also verified that the resin mixing pump is aligned to discharge excess transfer water through the resin fines filter to the liquid waste processing system.

During the transfer operation the tank level is monitored and the resin mixing pump is operated, if required, to limit tank water level. The operator stops the transfer when the CCTV camera viewing the sight flow glass indicates on a control panel monitor that the sluice water is clear and the transfer line is, therefore, flushed of resins.

After the bed transfer, the tank solids level can be checked by operating the resin mixing pump to lower the water level below the solids level. The solids level can be determined by the ultrasonic surface detector.

Between bed transfer operations the water level in the spent resin tanks is maintained above the solids level. Demineralized water is supplied for water level adjustment as well as a backup water source for flushing resin handling lines after resin recirculation and waste disposal container filling operations.

The solids bed can be agitated and mixed at any time by using compressed air or by operating the resin mixing pump in the resin mixing mode. In the resin mixing mode, water is drawn from the spent resin tank via resin retention screens. The water is returned via tank mixing eductors that generate a resin slurry recirculation within the tank equivalent to about four times the flow rate generated by the resin mixing pump. The solids bed is locally fluidized during this operation.

The resin mixing mode is established to fluidize and mix the solids bed in the spent resin tank before waste disposal container filling. The resin transfer pump is then started in the recirculation mode. A resin slurry is drawn from the spent resin tank and returned to the same tank. A representative resin sample may be obtained during recirculation or container filling modes by operating the sampling device.

The portable system's container fill valve is opened to initiate the filling operation. The resin dewatering pump of the portable dewatering system is started to dewater the resin as it accumulates in the container. The resin dewatering pump discharges the water to the recirculation line. The water flows back to the spent resin tank, thereby preserving the water inventory in the system and retaining any resin fines or dislodged crud within the system.

The resin mixing pump can be stopped at any time during the filling operation. When the solids level nears the top of the container, as detected by level sensors and observed by a television camera, the fill valve is closed and cycled to top off the container. Excessive water or solids level automatically closes the fill valve.

When the filling operation is complete, the line flushing sequence controller is manually initiated to automatically operate the pumps and valves to flush the resin transfer lines back to the spent resin tank. The container fill valve is opened for a short time period to flush the remaining resin to the waste container. The resin mixing pump supplies filtered flush water from the spent resin tank. The portable dewatering system's dewatering pump is operated periodically until no further dewatering flow is detected by the pump discharge pressure indicator and/or audible indications from the pump.

11.4.2.3.2 Spent Filter Processing Operations

A filter transfer cask is used to change the higher-activity filters of the chemical and volume control system and spent fuel cooling system. The filter vessel is drained, and the filter cover is opened remotely. The shield plug of the port over the filter is removed and the transfer cask, without its bottom shield cover, is lifted and positioned on the port directly over the cartridge in the filter vessel.

A grapple inside the transfer cask is remotely lowered and connected to the filter cartridge. The cartridge is lifted into the transfer cask, and the cask is transferred over plastic sheeting to the bottom shield cover. The dose rate of the cartridge is measured with a long probe, and the cask is lowered onto and connected to the bottom shield cover. The transfer cask is then moved to the auxiliary building rail car bay.

If recent applicable sample analysis results are available, the filter cartridge can be loaded directly into a disposal container as described in the following paragraph. If analysis is required, a sample of the filter media is obtained through a port in the transfer cask. The filter cartridge is placed in one of nine high-activity filter storage tubes until sample analysis results are available. The transfer cask bottom cover is disconnected, the transfer cask is lifted by the crane and transferred to a position over one of the temporary storage tubes, and the spent filter cartridge is lowered into the tube. After moving the transfer cask away, the crane is used to install a shield plug onto the storage tube. Any water draining from the filter during storage collects in the storage tube which may be drained to a floor drain for subsequent transfer to the liquid radwaste system.

When sample analysis is complete and packaging requirements are established, the transfer cask is used to retrieve the spent cartridges from storage and deposit them into a waste container via a port in the top of a portable processing and storage cask. Plastic coverings are removed and the container is capped, smear-surveyed, and decontaminated as required, using reach rod tools through a cask port. The dose rate survey is also made through a cask port. Transfer of the filled waste container to the shipping cask, including cask cover handling, is then performed using the rail car bay crane under remote control.

Filters with dose rates less than 15 R/hr on contact may be changed from outside of filter vessel shielding by using reach rod tools. The filter vessel is drained, and the cover is removed. Then the spent filter cartridge is grappled and lifted out and into a filter transfer cask.

At the radwaste building, low and moderate activity filter cartridges are deposited into disposal or storage drums. The drums are stored within portable shield casks in the shielded accumulation room, which is serviced by the mobile systems facility crane. Depending on dose rates and analysis results, stabilization may or may not be required. Cartridges not requiring stabilization are loaded into standard, 55 gallon shipping drums with absorbent and may be compacted using a mobile system. When stabilization is required, the cartridges may be loaded into either high integrity containers or standard drums. If standard drums are used, mobile equipment is used to encapsulate the contents of the drums.

The drum covers are manually installed, and the drums are smear surveyed, decontaminated by wiping, if required, weighed, stacked on pallets, and placed in the packaged waste storage room.

When a truck-load quantity of waste containers accumulates, shipment to a low-level waste disposal facility is initiated by loading pallets of drums and other low-level waste containers into a closed van using the scissor lift or onto a flat-bed trailer using the crane. If the activity level is too high for unshielded shipment, the drums are loaded onto a cask pallet and into a shielded shipping cask using the mobile systems facility crane.

Radioactive filters from ventilation exhaust filtration units are bagged and transported to the radwaste building, where they are temporarily stored. The filters are compacted along with other dry active wastes by a mobile system as described in the following subsection.

11.4.2.3.3 Dry Waste Processing Operations

Dry wastes are segregated by measuring the contact dose rate of the wastes to determine the appropriate processing method. The contact dose rates for initial waste segregation are as follows:

Low activity	<5 mR/hr
Moderate activity	5 mR/hr to 100 mR/hr
High activity	>100 mR/hr

These activity levels may be adjusted by the operator to minimize exposures while maximizing processing efficiency.

Wastes from surface contamination areas in the radiologically controlled area are placed in bags or containers and tagged at the point of origin with information on radiation levels, waste type, and destination. The bags or containers are transported to the radwaste building, where they are placed into low-, moderate-, or high-activity storage, segregated by portable shielding as appropriate.

The high-activity wastes (greater than 100 mR/hr) are normally expected to be compacted in drums using a mobile compactor system in the same manner as lower-activity filter cartridges.

Moderate-activity wastes (5 mR/hr to 100 mR/hr) are expected to be sorted in a mobile system to remove reusable items such as protective clothing articles and tools, hazardous wastes, and larger noncompressible items. The remaining wastes are normally compacted by mobile equipment. The packaged wastes may be loaded directly onto a truck for shipment or may be stored in the packaged waste storage room until a truck load quantity accumulates.

Low-activity, dry active waste (less than 5 mR/hr) generally contains a large amount of nonradioactive material. It is expected that these wastes normally will be processed through a mobile radiation monitoring and sorting system to remove non-radioactive items for reuse or local disposal. A radiation survey allows identification and removal of potentially clean items for the clean waste verification. The remaining radioactive wastes are normally compacted or packaged for disposal as appropriate.

Materials that enter the radiologically controlled area are verified as nonradioactive before being released for reuse or disposal. Tools and equipment belonging to personnel and contractors are surveyed at the radiologically controlled area exit in the annex building. If these items cannot be released or decontaminated, they become plant inventory or dry active waste and are handled as described previously.

Other wastes generated in the radiologically controlled area but outside of surface contamination areas are collected in bags or containers and are delivered to the temporary storage location in the radwaste building. These wastes normally are processed through a mobile radiation monitoring system to verify that they are nonradioactive and suitable for disposal in a local waste landfill.

11.4.2.3.4 Mixed Waste Processing Operations

Mixed wastes from the radiologically controlled area are collected in suitable containers and brought to the radwaste building, where separate containment pallets and accumulation drums are provided for solid and liquid mixed wastes. Mixed wastes are normally sent to an offsite facility having mixed-waste processing and disposal capabilities.

11.4.2.4 Waste Processing and Disposal Alternatives

11.4.2.4.1 Portable and Mobile Radwaste Systems Capabilities

Portable or mobile processing and packaging systems can be located in the auxiliary building rail car bay or the radwaste building mobile systems facility. Chemical wastes are normally processed in the radwaste building by a mobile concentration and/or solidification system when a batch accumulates in the chemical waste tank. Mobile systems are also used to encapsulate high-activity filters, to sort, decontaminate and compact dry active wastes, and to verify nonradioactive wastes.

The spent resin system includes connections in the fill station and rail car bay to allow spent resins to be delivered to a disposal container in either location for dewatering using portable equipment.

Branch Technical Position ETSB 11-3 provides guidance for portable solid waste systems in Section IV. Compliance with the four guidance items is achieved as follows:

IV.I The spent resin tanks are the only tanks that contain a significant volume of wet wastes, and these tanks are permanently installed. Concentrates that may be produced by mobile evaporation systems will be produced and stored by the mobile systems only in small batches prior to being solidified by the mobile systems. As described in subsection 1.2.7, the radwaste building is designed to retain spillage from mobile or portable systems.

IV.2 Permanently installed piping for transport of radioactive wastes to mobile or portable systems is routed close to the mobile or portable systems thereby minimizing the use of flexible interfacing hose. The hydrostatic test requirements of Regulatory Guide 1.143 will be applied to the flexible interfacing hose.

IV.3 Portable or mobile systems will be located in either the rail car bay of the auxiliary building or in the mobile systems facility in the radwaste building. The spent resin waste container fill station or the shipping cask in the auxiliary building collects spillage of spent resin during waste container filling operations. The radwaste and auxiliary buildings contain and drain spillage to the liquid radwaste system via the radioactive waste drain system as described in subsection 1.2.7 and Section 11.2. Portable or mobile systems will, when required, have their own HEPA filtered exhaust ventilation system. HEPA filtered exhaust is required when airborne radioactivity would exceed 10 CFR 20 derived air concentration limits for radiation workers. The mobile systems facility has connections on the exhaust ventilation ducts for connecting exhaust duct from mobile or portable processing systems to the building's exhaust ventilation system.

IV.4 Although the seismic criteria of Regulatory Guide 1.143 are not applicable to structures housing mobile or portable solid radwaste systems, the portable equipment used for spent resin container filling and dewatering and high-activity filter cartridge packaging will be housed within the Seismic Category I auxiliary building. The radwaste building, which provides shelter for mobile or portable radwaste systems, is non-seismic in accordance with Branch Technical Position ETSB 11-3.

11.4.2.4.2 Central Radwaste Processing Facility

As an alternative to the mobile or portable processes for lower-activity wastes, the wastes may be sent to a licensed central radwaste processing facility for processing and disposal. This option requires minimal onsite processing to remove radioactive materials from the waste streams. The wastes are loaded into a cargo container. The mobile systems facility includes a designated laydown area, and the mobile systems facility crane may be used to handle a cargo container.

11.4.2.5 Facilities

11.4.2.5.1 Auxiliary Building

Resin and filtration media transfer lines from the various ion exchangers are routed to the spent resin tanks on elevation 100′ - 0″ in the southwest corner of the auxiliary building. The spent resin system pumps, valves, and piping are located in shielded rooms near the spent resin tanks.

Liquid radwaste system transfer lines to and from the radwaste building are routed to the south wall of the auxiliary building where they penetrate and enter into a shielded pipe pit in the base mat of the radwaste building.

Accessways in the auxiliary building are used to move the filter transfer casks. This includes filter transfer cask handling from the containment, where the chemical and volume control filters are located, to the auxiliary building rail car bay, where the filter cartridges are stored and subsequently packaged using mobile equipment. These accessways are also used to move dry active waste from various collection locations to the radwaste building. Enclosed access is provided between the auxiliary building and the radwaste building on elevation 100′-0″ (grade level).

11.4.2.5.2 Radwaste Building

The radwaste building, described in Section 1.2, houses the mobile systems facility. It also includes the waste accumulation room and the packaged waste storage room. These rooms are serviced by the mobile systems facility crane.

In the mobile systems facility, three truck bays provide for mobile or portable processing systems and for waste disposal container shipping and receiving. A shielded pipe trench to each of the truck bays is used to route liquid radwaste supply and return lines from the connections in the shielded pipe pit at the auxiliary building wall. Separate areas are reserved for empty (new) waste disposal container storage, container laydown, and forklift charging. An area is available near the door to the annex building for protective clothing dropoff and frisking.

The waste accumulation room (pre-processing) is divided as needed, using partitions and portable shielding to adjust the storage areas for different waste categories as needed to complement the radioactivity levels and volumes of generated wastes. The accumulation room has lockable doors to minimize unauthorized entry and inadvertent exposure.

The packaged waste storage room may be separated into high- and low-activity areas, using portable shielding to minimize exposure while providing operational flexibility. A lockable door is provided to minimize unauthorized entry and radiation exposure.

The heating and ventilating system for the radwaste building is described in subsection 9.4.8.

11.4.3 System Safety Evaluation

The solid waste management system has no safety-related function and therefore requires no nuclear safety evaluation.

11.4.4 Tests and Inspections

Preoperational tests are conducted as described in subsection 14.2.9. Tests are performed to demonstrate the capability to transfer ion exchange resins and deep bed filtration media from the ion exchangers and filters to the spent resin tanks or directly to a waste disposal container. Preoperational tests of the solid waste management system components are performed to prepare the system for operation.

After plant operations begin, the operability and functional performance of the solid waste management system is periodically evaluated according to Regulatory Guide 1.143 by monitoring for abnormal or deteriorating performance during routine operations. Instruments and setpoints are also calibrated on a scheduled basis. The preventive maintenance program includes periodic inspection and maintenance of active components.

11.4.5 Quality Assurance

The quality assurance program for design, installation, procurement, and fabrication issues of the solid waste management system is in accordance with the overall quality assurance program described in Chapter 17.

11.4.6 Combined License Information for Solid Waste Management System Process Control Program

The Combined License applicant will develop a process control program in compliance with 10 CFR Sections 61.55 and 61.56 for wet solid wastes and 10 CFR Part 71 and DOT regulations for both wet and dry solid wastes. Process control programs will also be provided by vendors providing mobile or portable processing or storage systems. It will be the plant operator's responsibility to assure that the vendors have appropriate process control programs for the scope of work being contracted at any particular time. The process control program will identify the operating procedures for storing or processing wet solid wastes. The mobile systems process control program will include a discussion of conformance to Regulatory Guide 1.143 (Reference 7), Generic Letter GL-80-009 (Reference 8), and Generic Letter GL-81-039 (Reference 9) and, information of equipment containing wet solid wastes in the nonseismic Radwaste Building. In the event additional onsite storage facilities are a part of Combined License plans, this program will include a discussion of conformance to Generic Letter GL-81-038 (Reference 10).

11.4.7 References

1. "Shippers-General Requirements for Shipments and Packagings," 49 CFR 173.

2. "Packaging and Transportation of Radioactive Material," 10 CFR 71.

3. "Domestic Licensing of Production and Utilization Facilities," 10 CFR 50.

4. "Standards for Protection Against Radiation," 10 CFR 20.

5. "Licensing Requirements for Land Disposal of Radioactive Waste," 10 CFR 61.

6. "USNRC Technical Position on Waste Form," Rev. 1, January 1991.

7. Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants."

8. USNRC Generic Letter GL-80-009, "Low Level Radioactive Waste Disposal," dated January 29, 1980.

9. USNRC Generic Letter GL-81-039, "NRC Volume Reduction Policy (Generic Letter No. 81-39)," dated November 30, 1981.

10. USNRC Generic Letter GL-81-038, "Storage of Low-Level Radioactive Wastes at Power Reactor Sites," dated November 10, 1981.

11. USNRC, "Minimization of Contamination," 10 CFR 20.1406.

Table 11.4-1

ESTIMATED SOLID RADWASTE VOLUMES

Source	Expected Generation (ft^3/yr)	Expected Shipped Solid (ft^3/yr)	Maximum Generation (ft^3/yr)	Maximum Shipped Solid (ft^3/yr)
Wet Wastes				
Primary Resins (includes spent resins and wet activated carbon)	400(2)	510	1700(4)	2160
Chemical	350	20	700	40
Mixed Liquid	15	17	30	34
Condensate Polishing Resin(1)	0	0	206(5)	259
Steam Generator Blowdown(1)(6) Material (Resin and Membrane)	0	0	540(5)	680
Wet Waste Subtotals	**765**	**547**	**3176**	**3173**
Dry Wastes				
Compactible Dry Waste	4750	1010	7260	1550
Non-Compactible Solid Waste	234	373	567	910
Mixed Solid	5	7.5	10	15
Primary Filters (includes high activity and low activity cartridges)	5.2(3)	26	9.4(3)	69
Dry Waste Subtotals	**4994**	**1417**	**7846**	**2544**
TOTAL WET & DRY WASTES	**5759**	**1964**	**11,020**	**5717**

Notes:
1. Radioactive secondary resins and membranes result from primary to secondary systems leakage (e.g., SG tube leak).
2. Estimated activity basis is ANSI 18.1 source terms in reactor coolant.
3. Estimated activity basis is breakdown and transfer of 10% of resin from upstream ion exchangers.
4. Reactor coolant source terms corresponding to 0.25% fuel defects.
5. Estimated activity basis from Table 11.1-5, 11.1-7 and 11.1-8 and a typical 30 day process run time, once per refueling cycle.
6. Estimated volume and activity used for conservatism. Resin and membrane will be removed with the electrodeionization units and not stored as wet waste. See subsection 10.4.8.

Table 11.4-2 (Sheet 1 of 2)

EXPECTED ANNUAL CURIE CONTENT OF PRIMARY INFLUENTS

Isotope	Primary Resin Total Ci/yr	Primary Filter Total Ci/yr
Br-83	---	---
Br-84	1.98E-01	1.98E-02
Br-85	---	---
I-129	---	---
I-130	---	---
I-131	1.42E+02	1.42E+01
I-132	1.04E+01	1.04E+00
I-133	5.29E+01	5.29E+00
I-134	6.89E+00	6.89E-01
I-135	3.49E+01	3.49E+00
Rb-86	---	---
Rb-88	9.72E-01	9.72E-02
Rb-89	---	---
Cs-134	3.06E+02	3.06E+01
Cs-136	3.16E+00	3.16E-01
Cs-137	4.64E+02	4.64E+01
Cs-138	---	---
Ba-137m	4.44E+02	4.44E+01
Cr-51	3.21E+01	3.21E+00
Mn-54	1.04E+02	1.04E+01
Mn-56	---	---
Fe-55	1.04E+02	1.04E+01
Fe-59	5.00E+00	5.00E-01
Co-58	2.05E+02	2.05E+01
Co-60	9.59E+01	9.59E+00
Zn-65	3.02E+01	3.02E+00
Sr-89	2.67E+00	2.67E-01
Sr-90	1.13E+00	1.13E-01
Sr-91	1.72E-01	1.72E-02
Sr-92	---	---
Ba-140	6.29E+01	6.29E+00
Y-90	---	---
Y-91m	---	---
Y-91	3.74E-06	3.74E-07

Table 11.4-2 (Sheet 2 of 2)

EXPECTED ANNUAL CURIE CONTENT OF PRIMARY INFLUENTS

Isotope	Primary Resin Total Ci/yr	Primary Filter Total Ci/yr
Y-92	---	---
Y-93	---	---
La-140	---	---
Zr-95	2.80E-04	2.80E-05
Nb-95	---	---
Mo-99	---	---
Tc-99m	---	---
Ru-103	5.35E-03	5.35E-04
Ru-106	6.37E-02	6.37E-03
Rh-103m	---	---
Rh-106	---	---
Te-132	---	---
Te-125m	---	---
Te-127m	---	---
Te-127	---	---
Te-129m	1.36E-04	1.36E-05
Te-129	---	---
Te-131m	---	---
Total:	2.11E+03	2.11E+02

Note:
Values shown as "---" Ci/yr are those calculated to be lower than 1.0E-10 Ci/yr, and thus considered to have insignificant contributions to total.

Table 11.4-3 (Sheet 1 of 2)

MAXIMUM ANNUAL CURIE CONTENT OF PRIMARY INFLUENTS

Isotope	Primary Resin Total Ci/yr	Primary Filter Total Ci/yr
Br-83	7.03E+00	7.03E-01
Br-84	3.42E-01	3.42E-02
Br-85	3.74E-03	3.74E-04
I-129	3.44E-03	3.44E-04
I-130	9.00E+00	9.00E-01
I-131	5.45E+03	5.45E+02
I-132	1.97E+02	1.97E+01
I-133	1.66E+03	1.66E+02
I-134	7.31E+00	7.31E-01
I-135	3.81E+02	3.81E+01
Rb-86	2.97E+01	2.97E+00
Rb-88	2.52E+01	2.52E+00
Rb-89	9.83E-01	9.83E-02
Cs-134	9.57E+03	9.57E+02
Cs-136	1.72E+03	1.72E+02
Cs-137	9.14E+03	9.14E+02
Cs-138	1.06E+01	1.06E+00
Ba-137m	8.66E+03	8.66E+02
Cr-51	3.95E+01	3.95E+00
Mn-54	1.18E+02	1.18E+01
Mn-56	4.75E+01	4.75E+00
Fe-55	1.14E+02	1.14E+01
Fe-59	5.84E+00	5.84E-01
Co-58	3.03E+02	3.03E+01
Co-60	2.45E+02	2.45E+01
Zn-65	---	---
Sr-89	4.56E+01	4.56E+00
Sr-90	1.09E+01	1.09E+00
Sr-91	1.16E+00	1.16E-01
Sr-92	9.96E-02	9.96E-03
Ba-140	1.19E+01	1.19E+00
Y-90	1.07E+01	1.07E+00
Y-91m	3.48E-01	3.48E-02
Y-91	5.48E-01	5.48E-02

Table 11.4-3 (Sheet 2 of 2)

MAXIMUM ANNUAL CURIE CONTENT OF PRIMARY INFLUENTS

Isotope	**Primary Resin Total Ci/yr**	**Primary Filter Total Ci/yr**
Y-92	4.19E-02	4.19E-03
Y-93	9.07E-05	9.07E-06
La-140	1.07E+01	1.07E+00
Zr-95	---	---
Nb-95	---	---
Mo-99	---	---
Tc-99m	---	---
Ru-103	---	---
Ru-106	---	---
Rh-103m	---	---
Rh-106	---	---
Te-132	---	---
Te-125m	---	---
Te-127m	---	---
Te-127	---	---
Te-129m	---	---
Te-129	---	---
Te-131m	---	---
Total:	3.78E+04	3.78E+03

Note:
Values shown as "---" Ci/yr are those calculated to be lower than 1.0E-10 Ci/yr, and thus considered to have insignificant contributions to total.

Table 11.4-4 (Sheet 1 of 2)

EXPECTED ANNUAL CURIE CONTENT OF SHIPPED PRIMARY WASTES

Isotope	Primary Resin Total Ci/yr	Primary Filter Total Ci/yr
Br-83	---	---
Br-84	---	---
Br-85	---	---
I-129	---	---
I-130	---	---
I-131	6.04E-02	6.04E-03
I-132	---	---
I-133	---	---
I-134	---	---
I-135	---	---
Rb-86	---	---
Rb-88	---	---
Rb-89	---	---
Cs-134	2.81E+02	2.81E+01
Cs-136	2.61E-02	2.61E-03
Cs-137	4.61E+02	4.61E+01
Cs-138	---	---
Ba-137m	4.61E+02	4.61E+01
Cr-51	3.37E+00	3.37E-01
Mn-54	8.50E+01	8.50E+00
Mn-56	---	---
Fe-55	9.75E+01	9.75E+00
Fe-59	1.23E+00	1.23E-01
Co-58	8.51E+01	8.51E+00
Co-60	9.29E+01	9.29E+00
Zn-65	2.34E+01	2.34E+00
Sr-89	8.05E-01	8.05E-02
Sr-90	1.13E+00	1.13E-01
Sr-91	---	---
Sr-92	---	---
Ba-140	4.80E-01	4.80E-02
Y-90	1.13E+00	1.13E-01
Y-91m	---	---
Y-91	4.03E-04	4.03E-05

Table 11.4-4 (Sheet 2 of 2)

EXPECTED ANNUAL CURIE CONTENT OF SHIPPED PRIMARY WASTES

Isotope	Primary Resin Total Ci/yr	Primary Filter Total Ci/yr
Y-92	---	---
Y-93	---	---
La-140	5.52E-01	5.52E-02
Zr-95	1.09E-04	1.09E-05
Nb-95	1.31E-04	1.31E-05
Mo-99	---	---
Tc-99m	---	---
Ru-103	1.10E-03	1.10E-04
Ru-106	5.38E-02	5.38E-03
Rh-103m	1.11E-03	1.11E-04
Rh-106	5.38E-02	5.38E-03
Te-132	---	---
Te-125m	---	---
Te-127m	---	---
Te-127	---	---
Te-129m	2.10E-05	2.10E-06
Te-129	1.37E-05	1.37E-06
Te-131m	---	---
Total:	1.60E+03	1.60E+02

Note:
Values shown as "---" Ci/yr are those calculated to be lower than 1.0E-10 Ci/yr, and thus considered to have insignificant contributions to total.

Table 11.4-5 (Sheet 1 of 2)

MAXIMUM ANNUAL CURIE CONTENT OF SHIPPED PRIMARY WASTES

Isotope	Primary Resin Total Ci/yr	Primary Filter Total Ci/yr
Br-83	---	---
Br-84	---	---
Br-85	---	---
I-129	3.44E-03	3.44E-04
I-130	---	---
I-131	4.10E+02	4.10E+01
I-132	---	---
I-133	6.27E-08	6.27E-09
I-134	---	---
I-135	---	---
Rb-86	9.76E+00	9.76E-01
Rb-88	---	---
Rb-89	---	---
Cs-134	9.31E+03	9.31E+02
Cs-136	3.47E+02	3.47E+01
Cs-137	9.13E+03	9.13E+02
Cs-138	---	---
Ba-137m	9.13E+03	9.13E+02
Cr-51	1.86E+01	1.86E+00
Mn-54	1.10E+02	1.10E+01
Mn-56	---	---
Fe-55	1.12E+02	1.12E+01
Fe-59	3.66E+00	3.66E-01
Co-58	2.26E+02	2.26E+01
Co-60	2.42E+02	2.42E+01
Zn-65	---	---
Sr-89	3.06E+01	3.06E+00
Sr-90	1.09E+01	1.09E+00
Sr-91	---	---
Sr-92	---	---
Ba-140	2.35E+00	2.35E-01
Y-90	1.09E+01	1.09E+00
Y-91m	---	---
Y-91	3.90E-01	3.90E-02

Table 11.4-5 (Sheet 2 of 2)

MAXIMUM ANNUAL CURIE CONTENTS OF SHIPPED PRIMARY WASTES

Isotope	**Primary Resin Total Ci/yr**	**Primary Filter Total Ci/yr**
Y-92	---	---
Y-93	---	---
La-140	2.70E+00	2.70E-01
Zr-95	---	---
Nb-95	---	---
Mo-99	---	---
Tc-99m	---	---
Ru-103	---	---
Ru-106	---	---
Rh-103m	---	---
Rh-106	---	---
Te-132	---	---
Te-125m	---	---
Te-127m	---	---
Te-127	---	---
Te-129m	---	---
Te-129	---	---
Te-131m	---	---
Total:	2.91E+04	2.91E+03

Note:
Values shown as "---" Ci/yr are those calculated to be lower than 1.0E-10 Ci/yr, and thus considered to have insignificant contributions to total.

Table 11.4-6 (Sheet 1 of 2)

EXPECTED ANNUAL CURIE CONTENT OF SECONDARY WASTE AS GENERATED

Isotope	Secondary Resin Total Ci/yr
Na-24	1.83E-02
Cr-51	4.29E-02
Mn-54	2.95E-02
Fe-55	2.35E-02
Fe-59	4.49E-03
Co-58	7.78E-02
Co-60	1.03E-02
Zn-65	9.56E-03
Br-84	2.22E-05
Rb-88	8.99E-05
Sr-89	2.24E-03
Sr-90	2.37E-04
Sr-91	2.11E-04
Y-90	2.06E-04
Y-91	2.53E-04
Y-91m	1.82E-04
Y-93	9.80E-04
Zr-95	6.53E-03
Nb-95	5.19E-03
Nb-95m	4.74E-03
Mo-99	1.52E-02
Tc-99m	1.41E-02
Ru-103	1.13E-01
Ru-106	1.65E+00
Rh-103m	1.39E-01
Rh-106	2.11E+00
Ag-110	2.12E-02
Ag-110m	2.45E-02
Te-129	2.29E-03
Te-129m	2.79E-03
Te-131	1.14E-03
Te-131m	1.42E-03
Te-132	4.74E-04

Table 11.4-6 (Sheet 2 of 2)

EXPECTED ANNUAL CURIE CONTENT OF SECONDARY WASTE AS GENERATED

Isotope	Secondary Resin Total Ci/yr
I-131	1.70E-01
I-132	7.93E-03
I-133	5.23E-02
I-134	1.18E-03
I-135	2.56E-02
Xe-131m	---
Xe-133	---
Xe-135	---
Cs-134	2.50E-01
Cs-135	4.70E-10
Cs-136	1.48E-02
Cs-137	3.39E-01
Ba-136m	1.39E-02
Ba-137m	3.42E-01
Ba-140	1.17E-01
La-140	1.47E-01
Ce-141	2.13E-03
Ce-143	2.91E-03
Ce-144	7.35E-02
Pr-143	2.04E-03
Pr-144	6.37E-02
Total:	5.96E+00

Note:
Values shown as "---" Ci/yr are those calculated to be lower than 1.0E-10 Ci/yr, and thus considered to have insignificant contributions to total.

Table 11.4-7 (Sheet 1 of 2)

MAXIMUM ANNUAL CURIE CONTENT OF SECONDARY WASTE AS GENERATED

Isotope	Secondary Resin Total Ci/yr
Na-24	4.62E-04
Cr-51	5.17E-01
Mn-54	3.55E-01
Mn-56	2.24E-01
Fe-55	2.78E-01
Fe-59	5.88E-02
Co-58	9.25E-01
Co-60	1.23E-01
Br-83	3.73E-02
Br-84	1.41E-03
Br-85	1.64E-06
Kr-83m	---
Kr-85	---
Kr-85m	---
Rb-88	4.56E-02
Rb-89	1.53E-03
Sr-89	9.10E-01
Sr-90	5.00E-02
Sr-91	2.13E-02
Sr-92	7.25E-04
Y-90	4.60E-02
Y-91	4.34E-02
Y-91m	2.11E-02
Y-92	2.66E-03
Y-93	1.04E-03
Zr-95	7.74E-02
Nb-95	8.25E-02
Nb-95m	5.52E-02
Mo-99	1.52E+01
Tc-99m	1.68E+01
Ru-103	6.28E-02
Ru-103m	3.87E-02
Rh-103m	6.29E-02
Rh-106	5.95E-02

Table 11.4-7 (Sheet 2 of 2)

MAXIMUM ANNUAL CURIE CONTENT OF SECONDARY WASTE AS GENERATED

Isotope	Secondary Resin Total Ci/yr
Ag-110	1.34E-02
Ag-110m	2.24E-01
Te-129	1.19E+00
Te-129m	1.10E+00
Te-131	2.35E+00
Te-131m	2.01E-01
Te-132	6.75E+00
Te-134	1.49E-03
I-130	1.19E-01
I-131	1.37E+02
I-132	6.77E+00
I-133	2.51E+01
I-134	4.99E-02
I-135	3.99E+00
Xe-131m	---
Xe-133	---
Xe-135	---
Cs-134	6.90E+02
Cs-135	6.16E-08
Cs-136	5.15E+02
Cs-137	5.00E+02
Cs-138	3.41E-02
Ba-136m	6.35E+02
Ba-137m	5.14E+02
Ba-140	2.83E-01
La-140	3.31E-01
Ce-141	6.42E-02
Ce-143	4.94E-03
Ce-144	6.33E-02
Pr-143	4.63E-02
Pr-144	6.33E-02
Total:	3.08E+03

Note:
Values shown as "---" Ci/yr are those calculated to be lower than 1.0E-10 Ci/yr, and thus considered to have insignificant contributions to total.

Table 11.4-8 (Sheet 1 of 2)

EXPECTED ANNUAL CURIE CONTENT OF SHIPPED SECONDARY WASTES

Isotope	Secondary Resin Total Ci/yr
Na-24	---
Cr-51	4.55E-03
Mn-54	2.40E-02
Fe-55	2.19E-02
Fe-59	1.14E-03
Co-58	3.25E-02
Co-60	9.95E-03
Zn-65	7.42E-03
Br-84	---
Rb-88	---
Sr-89	6.86E-04
Sr-90	2.36E-04
Sr-91	---
Y-90	2.31E-04
Y-91	6.71E-09
Y-91m	---
Y-93	---
Zr-95	2.52E-03
Nb-95	4.06E-03
Nb-95m	2.32E-03
Mo-99	---
Tc-99m	---
Ru-103	2.34E-02
Ru-106	1.38E+00
Rh-103m	2.87E-02
Rh-106	1.77E+00
Ag-110	1.66E-02
Ag-110m	1.92E-02
Te-129	3.44E-04
Te-129m	4.48E-04
Te-131	---
Te-131m	---

Table 11.4-8 (Sheet 2 of 2)

EXPECTED ANNUAL CURIE CONTENT OF SHIPPED SECONDARY WASTES

Isotope	Secondary Resin Total Ci/yr
Te-132	---
I-131	7.32E-05
I-132	---
I-133	---
I-134	---
I-135	---
Xe-131m	---
Xe-133	---
Xe-135	---
Cs-134	2.31E-01
Cs-135	4.86E-10
Cs-136	1.56E-04
Cs-137	3.36E-01
Ba-136m	1.47E-04
Ba-137m	3.40E-01
Ba-140	8.97E-04
La-140	1.05E-03
Ce-141	3.13E-04
Ce-143	---
Ce-144	5.91E-02
Pr-143	2.38E-05
Pr-144	5.12E-02
Total:	4.38E+00

Note:
Values shown as “---“ Ci/yr are those calculated to be lower than 1.0E-10 Ci/yr, and thus considered to have insignificant contributions to total.

Table 11.4-9 (Sheet 1 of 2)

MAXIMUM ANNUAL CURIE CONTENT OF SHIPPED SECONDARY WASTES

Isotope	Secondary Resin Total Ci/yr
Na-24	---
Cr-51	5.47E-02
Mn-54	2.89E-01
Mn-56	---
Fe-55	2.60E-01
Fe-59	1.50E-02
Co-58	3.87E-01
Co-60	1.19E-01
Br-83	---
Br-84	---
Br-85	---
Kr-83m	---
Kr-85	---
Kr-85m	---
Rb-88	---
Rb-89	---
Sr-89	2.79E-01
Sr-90	4.96E-02
Sr-91	---
Sr-92	---
Y-90	5.12E-02
Y-91	1.12E-06
Y-91m	---
Y-92	---
Y-93	---
Zr-95	2.98E-02
Nb-95	5.19E-02
Nb-95m	2.70E-02
Mo-99	2.72E-09
Tc-99m	3.04E-09
Ru-103	1.30E-02
Ru103m	3.27E-02

Table 11.4-9 (Sheet 2 of 2)

MAXIMUM ANNUAL CURIE CONTENT OF SHIPPED SECONDARY WASTES

Isotope	**Secondary Resin Total Ci/yr**
Rh-103m	1.30E-02
Rh-106	5.03E-02
Ag-110	1.05E-02
Ag-110m	1.76E-01
Te-129	1.92E-01
Te-129m	1.77E-01
Te-131	---
Te-131m	---
Te-132	2.90E-08
Te-134	---
I-130	---
I-131	5.94E-02
I-132	2.36E-08
I-133	---
I-134	---
I-135	---
Xe-131m	---
Xe-133	---
Xe-135	---
Cs-134	6.35E+02
Cs-135	6.36E-08
Cs-136	5.42E+00
Cs-137	4.98E+02
Cs-138	---
Ba-136m	6.69E+00
Ba-137m	5.11E+02
Ba-140	2.18E-03
La-140	2.87E-03
Ce-141	9.41E-03
Ce-143	---
Ce-144	5.08E-02
Pr-143	4.75E-04
Pr-144	5.08E-02
Total:	1.66E+03

Note:
Values shown as "---" Ci/yr are those calculated to be lower than 1.0E-10 Ci/yr, and thus considered to have insignificant contributions to total.

Table 11.4-10 (Sheet 1 of 2)

COMPONENT DATA – SOLID WASTE MANAGEMENT SYSTEM (NOMINAL)

Tanks	
Spent resin tank	
Number	2
Total volume (ft^3)	300
Type	Vertical, conical bottom, dished top
Design pressure (psig)	15
Design temperature (°F)	150
Material	Stainless steel
Pumps	
Resin mixing pump	
Number	1
Type	Pneumatic diaphragm
Design pressure (psig)	125
Design temperature (°F)	150
Design flow rate (gpm)	120
Design head (ft)	160
Air supply pressure (psig)	100
Air consumption (scfm)	130
Material	Stainless steel housing, Buna N diaphragms
Resin transfer pump	
Number	1
Type	Material handling positive displacement
Design pressure (psig)	125
Design temperature (°F)	150
Design flow rate (gpm)	100
Material	Stainless steel housing, Buna N flexible parts

Table 11.4-10 (Sheet 2 of 2)

COMPONENT DATA – SOLID WASTE MANAGEMENT SYSTEM (NOMINAL)

Filters	
Resin fines filter	
Number	1
Type	Filter cartridge for inside to outside flow
Design pressure (psig)	150
Design temperature (°F)	150
Design flowrate (gpm)	120
Filtration rating	10 microns
Material	Stainless steel housing and pleated polypropylene cartridge with stainless steel screen outer jacket
Sampler	
Resin sampling device	
Number	1
Type	Inline sampler, positive displacement sample collection and portable pig for sample jar
Material	Stainless steel and EPDM wetted parts

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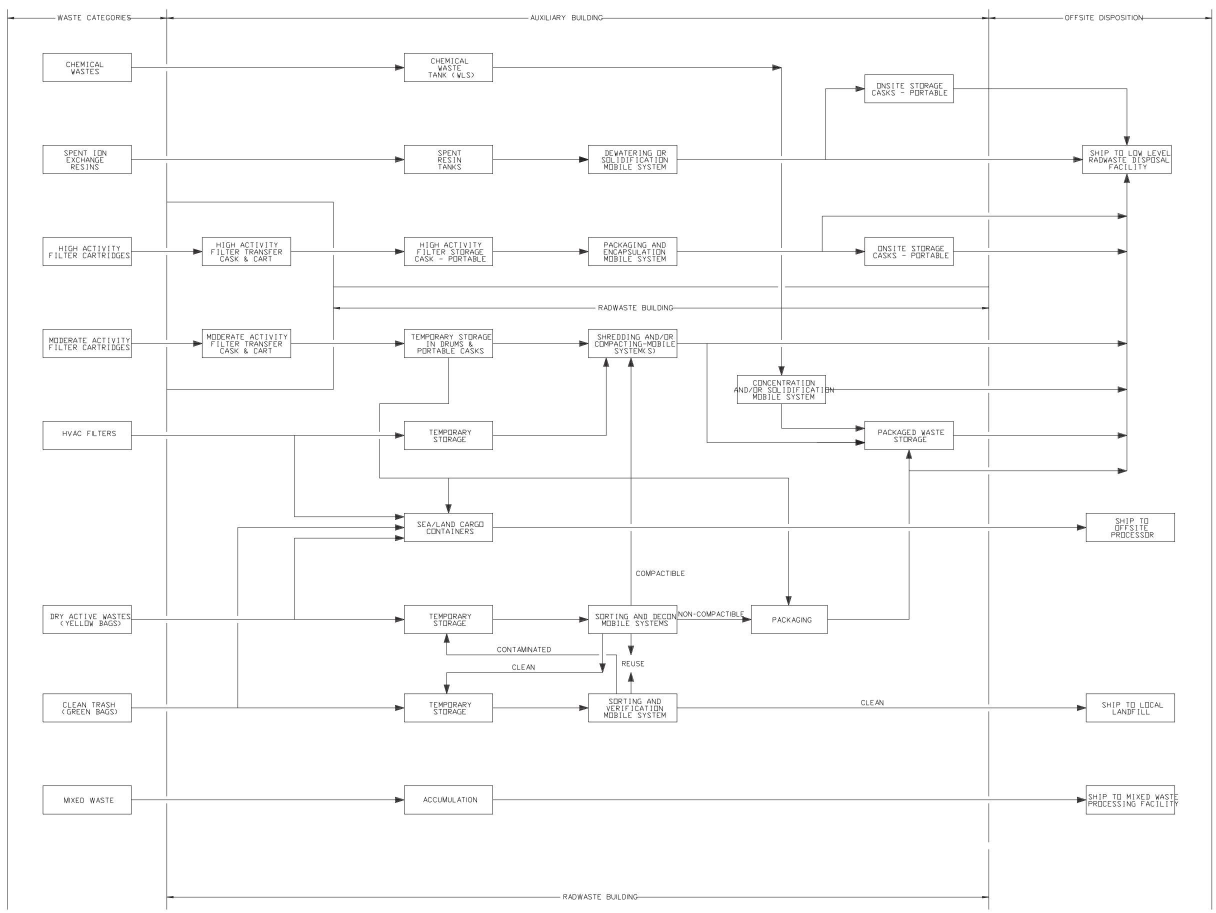

Figure 11.4-1

Waste Processing System Flow Diagram

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
	)	
Florida Power & Light Company	)	Docket Nos. 52-040-COL
	)	52-041-COL
(Turkey Point Units 6 and 7)	)	
	)	ASLBP No. 10-903-02-COL
(Combined License)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing "Florida Power & Light Company's Motion For Summary Disposition of CASE Contention 7" were provided to the Electronic Information Exchange for service to those individuals listed below and others on the service list in this proceeding, this 3rd day of January, 2012.

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