STATUS OF
NRC RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATION ACTIVITIES*

Charles A. Willis & Frank J. Congel
U. S. Nuclear Regulatory Commission

Abstract

The current NRC position on radiological effluent technical specifications (RETS) was established in 1978 (NUREG-0472 and -0473). Progress in implementing current RETS requirements was interrupted by the accident at Three Mile Island. Since then the RETS requirements have been implemented at the operating license stage. Efforts to implement current RETS requirements for operating reactors are now being resumed using a new approach. Operating reactors are being asked to meet the intent, not the letter, of the model RETS requirements. This flexible approach increases the time required for review so contractor support was found necessary. The current schedule calls for action on all operating reactors within a year.

Scope of the Radiological Effluent Technical Specifications (RETS)

The RETS are the technical specifications that deal with radioactive waste management systems and with environmental monitoring. The RETS are commonly but mistakenly called "Appendix I" Tech Specs. This misnomer has been a source of considerable misunderstanding despite repeated clarifications.

The model RETS did grow out of the 1970 requirements for keeping releases "as low as reasonably achievable" (ALARA) which were later quantified in "Appendix I"[2]. These regulations did not directly impose limits on releases but instead required Tech Specs that would limit releases. This approach was intended to provide the flexibility necessary to ensure that the constraints would indeed be reasonable (as well as low). Thus the RETS are closely associated with "Appendix I" but as other waste management system problems arose other appropriate provisions were added to the RETS.

The model RETS include a number of requirements not directly related to keeping releases ALARA. Specific "non-Appendix I" issues include the following.

1. Control of explosive gas mixtures
2. Curie content of waste decay tanks
3. Activity in BWR main condenser offgas
4. Curie content of certain outdoor liquid radwaste tanks
5. Level-measuring devices in liquid radwaste tanks

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6. Monitoring potentially contaminated liquid effluents
7. Automatic flow termination in some liquid effluent lines
8. Participation in an interlaboratory comparison program for radiochemical analyses
9. Process control program for solid radwaste management
10. Monitoring and control of in-process radwastes
11. Venting/purging Mark II containment drywell.

While it can be argued that the non-Appendix I matters should have been handled some other way, they are in fact integral parts of the RETS.

Regulations on Keeping Releases ALARA

On December 3, 1970, two new sections (10 CFR 50.34a and 50.36a) were added to the reactor licensing regulations[1]. In essence, these new requirements are to keep releases ALARA. Section 50.34a requires designing plants to keep releases ALARA and Section 50.36a requires Tech Specs for keeping releases ALARA during operations.

Section 50.36a explicitly requires Tech Specs in the following four areas.

1. Releases shall comply with the Section 20.106 limits.
2. Procedures shall be established and followed for operating radwaste treatment systems.
3. Radwaste systems shall be maintained and used.
4. Semiannual release reports shall be prepared and submitted to the NRC.

Section 50.36a also requires that the licensee be guided by the following considerations in developing operating procedures.

1. Releases, on the average, should be a small fraction of the limits of Section 20.106[3].
2. Best efforts shall be exerted to keep releases ALARA.
3. Appendix I provides numerical guidance on limiting conditions for operation for effluents.

In addition to the NRC regulations, the EPA has established a regulation (40 CFR 190) requiring that offsite doses from uranium fuel cycle facilities (including reactors) be ALARA. The EPA requirements are somewhat different from the NRC requirements. Nevertheless, Frank Congel (NUREG-0543) has shown that meeting the Appendix I "design objectives" provides reasonable assurance of compliance with 40 CFR 190[13].

Appendix I

Appendix I[2] provides numerical guides for ALARA design objectives and for limiting conditions for operation. Four design objectives are specified in Section II.

A. Liquid effluents from each reactor each year shall not expose any individual to more than 3 mrem to the total body or 10 mrem to any critical organ.

B. Gaseous effluents from each reactor each year (a) shall not produce, at any occupiable off-site location, air doses greater than 10 mrad gamma and 20 mrad beta or alternately, (b) shall not expose any individual to more than 5 mrem to the total body or 15 mrem to the skin.

C. Airborne iodine and particulate effluents from each reactor each year shall not expose any individual to more than 15 mrem to any critical organ.

D. Doses shall be further reduced as much as practical up to the expenditure of $1000 per person-rem saved.

Appendix I also specifies Limiting Conditions for Operation (LCO).*

A. If releases during any calendar quarter cause doses exceeding half an annual design objective, the licensee shall investigate, correct and report to the NRC in 30 days.

B. Surveillance programs shall be established to monitor releases, monitor the environment and identify changes in land use.

The annex to Appendix I, usually refered to as "RM 50-2", provides an alternative to the Appendix I cost/benefit design objective. The RM 50-2 criteria were proposed by the staff and used prior to the promulgation of Appendix I. The One advantage of the RM 50-2 criteria is that they allow the cost-benefit criterion to be replaced by annual release criteria (5 Ci per reactor in liquid and 1 Ci per reactor of airborne iodine-131), (a) if the dose criteria are changed from "per reactor" to "per site", and (b) if liquid dose criterion is changed to 5 mrem/yr to any critical organ or the total body. This option is most likely to be of value to licensees who had Tech Specs based on these criteria before the promulgation of Appendix I.

*Clearly the Appendix I LCOs play a role different from that of the Tech Specs on essential safety systems.

Early Progress in Implementation

Appendix I required licensees to file, by June 3, 1976, information needed for evaluation of means for keeping releases ALARA and proposed ALARA Tech Specs. However, in February 1976, the NRC Staff issued guidance recommending that proposals to modify Tech Specs be deferred until the NRC completed the model RETS. A draft version of the model RETS was transmitted to licensees in May 1976 but this draft was subsequently recalled.

On May 16, 1978, the NRC's Regulatory Requirements Review ("rachet") Committee approved the model RETS. Copies were sent to licensees in July 1978, along with a request to submit proposed site specific RETS on a staggered schedule over a 6 month period. Licensees responded with requests for clarifications and extensions.

The Atomic Industrial Forum (AIF) formed a Task Force to comment on the model RETS. NRC Staff members first met with the AIF Task Force on June 27, 1978. The model RETS were subsequently revised to reflect comments from the AIF and others. The principal changes were the removal from the model RETS of much material concerning dose calculations and the addition of the "off-site dose calculation manual" (ODCM) to contain this material.

The revised model RETS were sent to licensees on November 15 and 16, 1978. This transmittal also included guidance on preparation of the RETS and the ODCM (NUREG-0133)[6], as well as a new schedule for responses, again staggered over a 6 month period.

Four regional seminars on the RETS were conducted by the staff during the last week of November and the first week of December 1978. The purpose of these seminars was to clarify matters for the licensees but the seminars also convinced the staff that there was a need for further guidance on ODCM preparation, a need for guidance on the process control program (PCP), and a need for further revision of the model RETS.

Revision 2 of the model RETS and the additional guidance on the ODCM (Appendix B) and the PCP (Appendix C) were completed in February 1979, and were given to individual utilities in individual meetings.

Submittals on the RETS were received from all but one licensee by December 1979. Many of the submittals were reviewed by the staff and individual discussions were held with several licensees. However, the process was halted in March 1979, by the accident at Three Mile Island (TMI). Since then the model RETS have been implemented at those plants undergoing operating license review but today none of the reactors that were operating before January 1979, (except TMI-1) have Tech Specs that are consistent with the model RETS.

Reasons for Delay in Implementation

Possibly the single most important delay was the time required for the NRC staff to establish its position and provide guidance. Rev. 2 of the model RETS was not published until nearly 4 years after the adoption of Appendix I. Even today, after extensive review and formal approvals, staff members disagree about RETS implementation.

The TMI accident was a second important factor in delaying implementation. The people who had been working with the RETS were diverted to other tasks after the accident. Further, reactions to the accident created such a backlog of work that little staff time has been available for RETS work. Little progress was made with ORs (apart from internal discussions) until mid-1981.

Other important contributors to delay included the following.

1. Lack of explanation of deviations in licensee submittals
2. Inflexible attitude of the staff in initial reviews
3. Licensee resistance to new requirements
4. Staff manpower shortage
5. Licensee manpower shortage
6. Low priority; failure to implement the RETS generally is not seen as a threat to public health and safety
7. Widespread feeling that some model RETS requirements are excessive for ORs
8. Lack of a common understanding of the meaning and purpose of RETS.

These contributors are not independent of each other and the relative importance of each has not been established. Nevertheless, it is manifest that these factors must be taken into account if future implementation efforts are to be effective.

Renewed Efforts at Implementation

Plans for renewed efforts to implement the RETS for ORs were guided by a desire to minimize delays and to limit impact on licensees. Of course, failure to meet the objectives of the model RETS could not be accepted. These considerations led to several key decisions.

First, the existing model RETS (NUREG-0472 and -0473, Rev. 2) are to continue to be the basic guidance. Developing new guidance would have taken considerable time and would have needed new clarification.

Second, initial action is to be based on the 1979 submittals from licensees. Requesting new submittals would have taken time and added to licensee workload. Furthermore, new submittals might not have enhanced understanding or moved us closer to resolution of differences.

Third, where practicable, a flexible aproach is to be taken in the review. Licensee are being asked to meet the intent, rather than the letter, of the model RETS requirements. This approach should minimize impacts on licensees. It does, however, seriously complicate the reviewer's job.

Fourth, contractor support is being utilized. This approach delayed the initial reviews but the workload did not permit the staff to handle the OR RETS in a timely manner. In general, plants in the northeast are being reviewed by Franklin Research Center while EG&G Idaho is reviewing the others (Table 1).

Fifth, clarification will be achieved primarily through direct meetings with individual licensees. Coordination with the AIF Task Force will be continued and papers will be presented at technical meetings but direct contacts are given highest priority.

Role of the Contractors

The assignments of the two contracts differ only in the plants they are to review. These assignments, for each plant, are as follows.

1. Review the existing RETS and the RETS submittal and compare them to current requirements as experssed in the model RETS and in NUREG-0133.

2. Obtain the necessary additional information from the FSAR and from discussions with licensees.

3. Identify deficiencies in the existing and proposed RETS in the sense of not meeting the intent of the requirements.

4. Discuss deficiencies with the licensee and, where practicable, work out mutually acceptable RETS.

5. Where differences cannot be resolved, recommend appropriate staff action.

6. Provide a brief written explanation of why the RETS meet (or do not meet) the intent of the guidance.

7. Update the relevant SAR on compliance with Appendix I; the update is to reflect recent release data, changes in radwaste systems and the RETS revisions.

Schedule

The current tentative schedule calls for the completion of all reviews by October 1982. Experience is not yet sufficient to provide a measure of our ability to meet this schedule.

The plant-by-plant schedule is being developed. This is complicated by conflicting considerations. First, it seems desirable to start with plants which seem relatively close to compatibility with the model RETS to provide experience before proceeding to the more difficult plants. On the other hand, those plants furthest from meeting current criteria merit prompt attention. Furthermore, the staff is trying to give priority attention to those older plants in the systematic evaluation program (SEP). Also, all the plants operated by a single licensee should be evaluated at about the same time. Clearly not all these objectives can be met, but they are being considered.

Implications of Flexibility in Review

The model RETS and the supporting documents were developed primarily for plants not yet in operation. The difficulties in meeting new requirements can be much greater for operating reactors. Even the addition of a new flow meter or radiation monitor can be problematic in older plants. Thus, it becomes appropriate to consider alternate approaches on a case by case basis.

Formats other than that of the model RETS are acceptable if appropriate.

Phraseology changes are acceptable if justified. In particular, there is no objection to the addition of clarifying statements which the licensee feels may obviate problems with "over-zealous" inspectors. We are, of course, striving for a common understanding of the model RETS phrasiology.

Alternate ALARA "design objectives" based on RM 50-2 (rather than Appendix I) are acceptable provided that the full set from RM 50-2 are used. This offers the advantage of meeting the cost-benefit criterion by annual curie limits rather than by operating treatment equipment whenever doses from releases in one month are projected to exceed 1/48 the annual dose objective.

In the ODCM and in procedures, design objectives may be expressed in curies released, rather than in dose, if it is shown that the constraints on releases actually will keep offsite doses below intended levels. This approach might alleviate some problems in training operators.

The process control program (PCP) for processing solid wastes need not be submitted to the staff for approval prior to implementation at operating reactors (only). This should reduce problems in coping with changing burial ground requirements, as well as problems in using or changing contractors.

Conversly, there are numerous areas with little room for flexibility. Specifically, the specified environmental monitoring program is the minimum acceptable. Also the lower limits of detection in effluent samples are inflexible; in particular, analysis for P-32 and Fe-55 is required, at least until it can be shown that these nuclides are not important contributors to offsite doses.

Generally, all the requirements of the model RETS must be addressed but some items are more important than others. The list of "esssential elements" (Table 2) indicates which items are considered most important. Other decisions and considerations are discussed briefly in Appendix A.

Detailed guidance has not been developed for the flexible review process primarily because the radwaste systems and the problems with the RETS differ from plant to plant.

Current Status of Operating Reactors

The general objective of the effluent control program evidently is being met. That is, radiation doses to the public from nuclear power plants (NPP) are low. The conservative calculations summarized in Table 3 indicate that in 1977 NPP effluents increased doses less than 0.008% over natural background. Subsequent improvement of the offgas treatment system of 4 BWRs has reduced the annual population dose since then, probably from 700 to less than 300 person-rems. Careful calculation of the doses from liquid effluents from six reactors would further reduce doses, probably to about 150 person-rems.

Available data (Tables 4 and 5) also suggest that the specific dose design objectives of Appendix I also are generally met. Conclusions on this point are tentative because less than half the plants now report calculated off-site doses and because the methods for such calculations (ODCM) have not been approved. One reason for implementing the RETS is to obtain this information in an orderly manner.

While the off-site doses seem acceptably low, one cannot be so sanguine about some other RETS issues. As indicated in Table 6, almost all ORs have Tech Specs that address certain issues (specifically meeting Part 20 limits, monitoring releases and environmental monitoring) but other issues are only occasionally addressed. For example, only 72% of the ORs have a Tech Spec on the land use census, even though it is explicitly called for by Appendix I. Only 58% of the plants have Tech Specs addressing a design objective on doses from liquid effluents, even though this is the heart of Appendix I. Thus, it is manifest that less than half the ORs meet the explicit ALARA requirements for Tech Specs.

Generally, the "non Appendix I" items are less consistently addressed.

At this time it is not clear how many of the Tech Spec shortcomings represent real deficiencies and how many are merely paperwork problems. It is known that in a number of plants some of the deficiencies are real. The solid waste management programs generally need improvement. Some monitoring equipment, particularly hydrogen-oxygen monitors, is missing. Several radiochemical analysis programs exclude important nuclides, particularly P-32 and Fe-55. The initial review process is expected to improve our understanding of the problem.

TABLE 1

OPERATING REACTOR RETS ASSIGNMENTS

FRC			EG&G		
1.	Beaver Valley	(PA)	1.	Arkansas 1	(AK)
2.	Big Rock Pt.**	(MI)	2.	Arkansas 2	(AK)
3.	Brunswick 1-2	(NC	3.	Browns Ferry 1-3	(AL)
4.	Calvert Cliffs 1-2	(MD)	4.	Cooper	(NB)
5.	Cook 1-2	(MI)	5.	Crystal River	(FL)
6.	Davis-Besse	(OH)	6.	Dresden 1-3**	(IL)
7.	Fitzpatrick	(NY)	7.	Duane Arnold	(IO)
8.	Ginna**	(NY)	8.	Farley 1	(AL)
9.	Haddam Neck**	(CN)	9.	Ft. Calhoun*	(NB)
10.	Indian Pt. 2	(NY)	10.	Ft. St. Vrain	(CO)
11.	Indian Pt. 3	(NY)	11.	Hatch 1-2	(GA)
12.	Maine Yankee	(ME)	12.	Kewaunee	(WI)
13.	Millstone 1**	(CN)	13.	La Crosse**	(WI)
14.	Milstone 2	(CN)	14.	Monticello	(MN)
15.	Nine Mile Pt.	(MY)	15.	Point Beach 1-2	(WI)
16.	North Anna 1-2	(VA)	16.	Prairie Island 1-2*	(MN)
17.	Oconee 1-3	(SC)	17.	Quad Cities 1-2	(IL)
18.	Oyster Creek**	(NJ)	18.	Rancho Seco*	(CA)
19.	Palisades**	(MI)	19.	Robinson	(SC)
20.	Peach Bottom	(PA)	20.	San Onofre 1**	(CA)
21.	Pilgrim	(MA)	21.	St. Lucie	(FL)
22.	Salem 1-2	(NJ)	22.	Trojan	(OR)
23.	Surry 1-2	(VA)	23.	Turkey Pt. 3-4*	(FL)
24.	Vermont Yankee	(VT)	24.	Zion*	(IL)
25.	Yankee Rowe**	(MA)			

*Plants participating in the source term measurements program.
**Systematic evaluation program (SEP) plants.

TABLE 2

ESSENTIAL ELEMENTS OF THE RETS
Technical Specifications must be included which address each of the following requirements

1. All significant releases shall be monitored.*
2. Off-site concentrations shall not exceed the 10 CFR 20 Table 2 values.*
3. Off-site doses shall be ALARA.*
4. Equipment shall be maintained and used to keep doses ALARA.*
5. Radwaste tank inventories shall be limited so failure would not cause off-site doses exceeding the 10 CFR 20 limits.
6. Waste gas composition shall be controlled to prevent explosive mixtures.
7. Wastes shall be processed to burial ground criteria under a documented program subject to QA verification.
8. An environmental monitoring program, including a land use census, shall be implemented.*
9. The radwaste management program shall be subject to regular audit and review.
10. Procedures for control of effluents shall be followed.*
11. Periodic and special reports on environmental monitoring and on releases shall be submitted.*
12. Off-site dose calculations shall be performed using documented methods that are consistent with NRC methodology.

*These requirements are expressly stipulated by regulation.

TABLE 3

1977* TOTAL BODY POPULATION DOSES FROM NUCLEAR POWER PLANT EFFLUENTS
(FROM NUREG/CR-1498)

	Site	OL date	Population doses, person-rems Gaseous	Liquid	Total
1.	Millstone (1GE[a], 2CE)	70,75	220 (200)	- (-)	200 (200)
2.	Dresden (GE)[b]	60,70,71	180 (170)	- (17)	180 (187)
	Subtotals		400 (370)	- (17)	400 (387)
3.	Pilgrim (GE)[c]	72	52 (7)	- (-)	52 (7)
4.	Oyster Creek (GE)[d]	69	41 (110)	- (-)	41 (110)
5.	Oconee (B&W)[e]	73,74,74	1 (1)	37 (11)	38 (12)
6.	Hatch (GE)[e]	75	- (-)	35 (-)	35 (-)
7.	Cook (W)[e]	75,78	- (1)	23 (39)	23 (40)
8.	Zion (W)[e]	73,73	9 (7)	13 (16)	22 (23)
9.	Davis Besse (B&W)[e]	77	- (-)	14 (-)	14 (-)
10.	Indian Point (W)	62.74,76	12 (8)	1 (1)	13 (9)
11.	Peach Bottom (GE)[e]	74,74	5 (5)	6.4 (10)	11.4 (15)
	Subtotals		120 (139)	129.4 (77)	249.4 (216)
12.	LaCrosse (AC	69	1.6 (0.4)	7.8 (5.5)	9.4 (5.9)
13.	Brunswick (GE)	75,77	6.3 (1.8)	0.1 (0.5)	6.4 (2.3)
14.	Quad Cities (GE)	72,72	1.3 (1.3)	2.7 (6.0)	4.0 (7.3)
15.	Surry (W)	72,73	1.3 (0.3)	2.4 (1.1)	3.7 (1.4)
16.	Browns Ferry (GE)	74,75,77	2.7 (1.3)	0.5 (0.9)	3.2 (2.2)
17.	Nine Mile Point (GE)	69	0.1 (0.1)	3.0 (-)	3.1 (0.1)
18.	Big Rock Point (GE)	62	0.3 (0.4)	2.3 (2.2)	2.6 (2.6)
19.	Haddam Neck (W)	68	2.2 (4.4)	0.2 (1.2)	2.4 (5.6)
20.	TMI (B&W)	74	1.7 (1.6)	0.3 (0.6)	2.0 (2.2)
21.	Kewaunee (W)	74	- (-)	1.9 (0.6)	1.9 (0.6)
22.	Calvert Cliffs (CE)	75,77	0.7 (0.7)	1.2 (1.9)	1.9 (2.6)
23.	Arkansas (B&W)	74	0.1 (0.1)	1.5 (2.1)	1.6 (2.2)
	Subtotals		18.3(12.4)	23.9(22.6)	42.2(35.0)
24.- 45. (47)	All Others		1.7 (8.6)	6.7(13.4)	8.4 (22)
	TOTALS		540 (530)	160 (130)	700 (660)
	Mean, person-rems per site		12.3(11.3)	3.6 (2.8)	15.9 (14)

Exposed Population, with 50 miles
- Total 92 million
- Average 2 million
- Maximum 16 million (Indian Point)
- Minimum 0.1 million (Humboldt Bay)

Average Dose 7.6 micro-rems (about 0.0076% of natural background)
2 Sites: 400/700 = 57% (59%)
11 Sites: 650/700 = 93% (91%)

*Preliminary 1978 values are given in parentheses based on private communication from David A. Baker (PNL)

[a]Offgas system augmented in 1978; 1979/80 releases were 2.7% of 1977/78 releases.
[b]Dresden-1 is shutdown; 1979/80 releases were 3.1% of 1977/78 releases.
[c]Augmentation now effecive; 1979 releases were 3.5% of 1977 releases.
[d]Offgas system augmentation no functioning.
[e]These calculated doses from liquid effluents appear unrealistically high.

TABLE 4

CURIES OF RADIOACTIVITY RELEASED IN LIQUID EFFLUENTS*

Plant**	1977	1978	1979	1980
Oconee 1, 2, 3	4.5	6.5	0.92	1.5
Hatch	0.075	0.032	0.036	0.068
Cook 1, 2	1.6(0.048)***	1.5(0.7)	2.6(0.27)	1.4(0.004)
Zion 1, 2	0.95(0.003)	1.2	0.84(0.0076)	0.47(0.0012)
Davis Besse	0.026	0.090(0.082)	0.043(0.082)	0.21
Peach Bottom 2, 3	2.2	5.1	19.	1.9
Dresden 1, 2, 3	0.92(0.018)	0.81	0.26(0.13)	0.48
(Dresden 1	0.60	0.33	0	0

*Data from licensee effluent reports.

**Listed in order of descending 1977 population doses.

***Numbers in parentheses are maximum individual total body doses from liquid effluents as reported by the licensees; the "worst" case shown here is Cook in 1978 where the dose of 0.7 mrem is a factor of 4 below the 3 mrem design objective.

TABLE 5

KILO CURIES OF RADIOACTIVITY RELEASED IN GASEOUS EFFLUENTS*

Plant	1977	1978	1979	1980
Millstone 1	620	570	20	12 (0.24)**
Millstone 2	2.4	0.36 (0.14)	0.24	1.3 (0.1)
Dresden 1, 2, 3	830 (7.4)	1330	31 (0.18)	36 (0.49)
(Dresden 1)	520	850	0	0
Pilgrim	410 (5.3)	33 (1.9)	14 (0.73)	1.4
Oyster Creek	180	1000	1000	31
Oconee 1, 2, 3	12	43	48	19
Cook 1, 2	3.9 (0.07)	49 (1.4)	11 (0.04)	3.8
Zion 1, 2	32 (1.0)	50 (.44)	24 (0.17)	5.8 (0.007)
Davis Besse	1.3	2.0	1.7 (0.04)	1.7

*Data from licensee effluent reports.

**Numbers in parentheses are maximum individual total body doses from gaseous effluents as reported by the licensees. The "worst" case shown here is Dresden in 1977 where the dose was 7.4 millirem. This would be a factor of 2 below the design objective of 5 mrem per reactor if releases from the 3 units were equal. Releases from Unit 1 may have produced some 4.6 mrem.

TABLE 6

EXTENT TO WHICH ORs ADDRESS LCOs OF THE MODEL RETS*

	Limiting Conditions for Operation	Yes**	No	Pct. Yes
3.3.3.9	Liquid Effluent Monitor	67	5	93
3.3.3.10	Gaseous Effluent Monitor	47	8	89
3.11.1.1	Liquid Concentration Limit (Part 20)	71	72	99
3.11.1.2	Liquid Dose Design Objective	42	30	58
3.11.1.3	Liquid Waste Treatment	63	9	88
3.11.1.4	Liquid Tank Curie Limit	48	24	67
3.11.2.1	Gas Dose Rate Limit (Part 20)	72	72	100
3.11.2.2	Noble Gas Dose Design Objective	48	24	67
3.11.2.3	Iodine & Particle Design Objective	43	29	60
3.11.2.4	Gas Waste Treatment	40	32	56
3.11.2.5	Gas Explosive Mixture	11	61	15
3.11.2.6	Gas Tank Curie Limit	43	29	60
	Venting/Purging BWR Containment	13	11	54
3.11.3	Solid Radwaste, PCP	10	62	14
3.11.4	Total Dose (40 CFR 190)	2	70	3
3.12.1	Environmental Monitoring	70	2	97
3.12.2	Land Use Census	52	20	72
3.12.3	Interlab. Comparison	1	71	1
		760	488	61
	Liquid Curie Design Objective (RM 50-2)	45	27	63
	Gaseous Curie Design Objective (RM 50-2)	28	44	39

*Based on an unpublished review by J. J. Hayes (NRC).
**The existing Tech Specs have LCOs that roughly correspond to the LCOs of the model RETS; compliance with the model RETS is not implied.

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12. E. F. Conti, et al, "Radiological Environmental Monitoring by NRC Licensees for Routine Operation of Nuclear Facilities," NUREG-0475 (October 1978).

13. F. J. Congel, "Methods for Demonstrating LWR Compliance with the EPA Uranium Fuel Cycle Standard (40 CFR 190)," NUREG-0543 (January 1980).

14. "Standard Review Plan for the Review of Safety An alysis Reports for Nuclear Power Plants," NUREG-0800 (Formerly NUREG-75/087), (1981).

C. Regulatory Guides

15. "Reporting Operating Information," Regulatory Guide 1.16, Rev. 4 (August 1975).

16. "Measuring, Evaluating and Reporting Radioactivity in Solid Wastes and Releases of Radiaoctive Materials in Liquid and Gaseous Effluents From Light-Water-Cooled Nuclear Power Plants," Regulatory Guide 1.21 (June 1974).

17. "Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I," Regualtory Guide 1.109, Rev. 1 (October 1977).

18. "Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors," Regulatory Guide 1.110 (March 1976).

19. "Methods of Estimating Atmospheric Transport and Dispersion of Gaseous Effluents n Routine Releases from Light-Water-Cooled Reactors," Regulatory Guide 1.111, Rev. 1 (July 1977).

20. "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors," Regulatory Guide 1.112 (March 1976).

21. "Estimating Acquatic Dispersion of Effluents from Accidental and Routine Releases for the Purpose of Implementing Appendix I," Regulatory Guide 1.113, Rev. 1 (April 1977).

D. Papers

22. Collins, John T., "NRC Model Radiological Effluent Technical Specifications," Atomic Industrial Forum Conference (April 1979).

23. Willis, Charles A., and Hayes, John J., Jr., "Radiological Effluent Technical Specifications: Where We've Been, Where We Are, and Where We're Going," Health PHysics Society Annual Meeting (June 1981).

Appendix A

OBJECTIONS TO THE MODEL RETS RAISED BY THE AIF AND OTHERS*

General

1. RETS are too voluminous.

   The model RETS constitute about 10% of the Standard Tech Specs (STS). This may not be excessive considering the public sensitivity to the issues. Also the staff is considering dividing the Tech Specs to emphasize those issues of immediate importance to safety. With this system the principal problem with volume will be eliminated.

2. Clarification is needed.

   Considerable supporting documentation has been provided (see bibliography) as well as papers at technical society meetings, regular discussions with the AIF working group, and regional seminars. This clarification effort is expected to continue but in the immediate future the staff plans to concentrate on meetings with individual licensees. This affords the advantages of dealing with the issues that are causing problems and of obtaining maximum feedback.

3. RETS should rely on existing equipment (no backfit).

   To the extent practicable, this is the staff's intent. In some instances, however, it is not clear how a licensee can show that the intent of a model RETS requirement is met in the absence of key instrumentation or other equipment.

4. The RETS go beyond the regulations.

   While the RETS contain many "non-Appendix I" requirements, a regulatory basis is given for each LCO. Even the much-maligned PCP has a regulatory basis in 50.34a, 50.36a, and GDC-60. The "surveillance" and "action" requirements generally are not addressed in the regulations, either for radwaste systems or for other parts of the plants, but they are important parts of the program.

5. The TMI "Action item" requirements are not included.

   True; the "lessons learned" Tech Specs will be handled separately.

6. STS format should not be required.

   It is not required.

*Comments are partly based on personnal communications from J. S. Boegli (NRC).

7. Tech Specs based on dose necessitate operator training.

   It is acceptable to express the Tech Specs in curies if the ODCM shows that these constraints are equivalent to, or more conservative than, the standard dose criteria. This would not eliminate the need for calculating and reporting doses (Reg. Guide 1.21).

Definitions

1. Some standard definitions are inappropriate for older plants.

   The problems with "channel check", "calibration", etc. are recognized and the staff is prepared to accept other appropriate definitions on a case by case basis.

2. Many terms used in the model RETS need definitions.

   Most of the important definitions have been provided though they are not all in one place. Specifically, for a definition of "restricted area" see Part 20, for "exclusion area" see Part 100, for "continuous composite sample" see STS 1.6, etc. Where a licensee sees potential problems in ambiguity, or other problems with definitions, the staff may accept reasonable definitions proposed by an individual licensee.

   One clarification here seems in order. The expression "on-site general public" is limited to those situations where a portion of the site is or may be occupied by members of the public for extended periods for recreational, occupational or other purposes. This does not include the visitor center or non-employees who occasionally come on site to service vending machines, etc.

Instrumentation

1. Tech Specs should cover only the final release points.

   This contention seems based on the common misconception that the RETS are limited to ensuring compliance with Appendix I. In fact, the RETS are also intended to meet other requirements including General Design Criterion (GDC) 61 which requires the control of radioactivity and of radioactive wastes and GDC 63 which requires monitoring of waste storage. The need for process instrumentation is inescapable. Deviations from the model RETS are acceptable where appropriately justified by the licensee.

2. Inoperable hydrogen and oxygen monitors should not require shutdown.

Appropriate alternatives to the model RETS "actions" can be accepted. In this case the "action" must provide incentive for timely repair of the monitors and for compliance with GDC-3 (Fire Protection).

3. PWR turbine building sump often is difficult to monitor and generally should not be isolated.

   The problems are recognized and plant-specific alternative means of monitoring and controlling these effluents can be accepted.

4. Pump curves should be adequate for flow determinations.

   The staff agrees where conditions are right.

5. Flow rate is irrelevant for some off-line radiation monitors.

   The RETS requirement is for alarm on loss of flow, not for flow rate measurement.

Effluents

1. P-32 analysis should not be required.

   This analysis is needed because P-32 can be an important contributor to dose due to its high bioconcentration factor. This may be changed by NRC and AIF studies now underway but the requirement stands until change is technically justified.

2. No crosscheck is available for P-32 or Fe-55.

   The statement is true but it does not affect the need for the analysis of effluents for these nuclides.

3. Actions on total dose should be on a calendar year basis.

   This LCO is based on the EPA regulation (40 CFR 190) so it follows the EPA time basis. This Tech Spec should cause few practical problems because the staff has concluded that compliance with Appendix I provides reasonable assurance of compliance with 40 CFR 190 (see NUREG-0543).

4. Why use 1/4 the dose objectives as a trigger for equipment use?

   The staff has agreed to accept this trigger level as satisfying the cost-benefit criterion of Appendix I. Two alternatives are available to licensees. One option is to use the Design Objectives of RM 50-2 which include annual curie release criteria. The other option is to provide a cost-benefit analysis justifying some other level for triggering the use of the equipment. The "trigger levels" may vary but the regulations explicitly require a Tech Spec on the maintenance and use of effluent treatment equipment.

5. Why require liquid level monitors on outdoor tanks?

   The primary purpose is to detect leaks that might otherwise go unnoticed.

6. How can curie limits be set for tanks of unknown size?

   Methods are described in NUREG-0133 and SRP 15.7.3. Generally the tank volume does not determine the limit. The staff has accepted 10 curies as a default value in cases where substantial dilution would occur before a spill could reach a potable water supply.

7. Why sample gaseous wastes daily following a power change if there is no activity "spike" in the reactor coolant?

   This provision reflects the uncertainty about the "iodine spiking" phenomenon. Iodine concentrations have been observed reaching maximum values as much as 24 hours after the transient. Other "spikes" have persisted over several days and in some cases the concentrations have varied with time over a period of days. Also, under some circumstances the activity increase may be most readily observed in the waste gas system. Where this requirement poses problems it may be practical to justify other controls.

8. Sampling waste gas tanks should be minimized.

   Sampling is not required. The staff will accept the results of a gross gamma monitor calibrated to curies Xe-133 equivalent.

9. Hydrogen and oxygen monitors are unreliable and hard to calibrate.

   Redundant monitors provide some relief but this continues to be a problem.

10. Some PWR waste gas systems (WGS) are designed for 3% hydrogen or oxygen.

    Generally, BWR WGS can withstand an explosion while PWR WGS cannot. SRP 11.3 sets the limit at 2% for systems not designed to withstand an explosion. The staff, however, has accepted deviations where justified on a case by case basis.

11. Concurrent meteorology dose calcualtions are costly.

    Appropriate and conservative approximate methods are acceptable.

Offsite Dose Calculation Manual (ODCM)

1. A simplified ODCM should be accepted.

   Simplicity is no barrier to acceptance so long as the methods are conservative and clearly explained.

2. The ODCM should contain only methodology.

   Certain parameter values are also needed, as is the flow diagram. In essence, we need enough information to permit a review of the substance of the document. This document is the basis for accepting licensees' calculated offsite dose values so it should contain enough information to justify confidence.

3. Methods for determining setpoints should suffice.

   Specifics are needed for inspectability. Fixed setpoints should be given and justified. For variable setpoints, the methodology should be sufficiently detailed and the parameter values so identified that (1) the reviewer can verify the adequacy of the approach and (2) an inspector can check actual setpoint calculations.

Process Control Program (PCP)

1. The "philosophy" of the PCP should be expounded.

   The need for improved control of waste processing has been repeatedly demonstrated and the NRC has committed to making improvements.

   The PCP approach was selected because it promised both reasonable assurance of adequate protection and the flexibility to accommodate diverse systems. The essence of the PCP is the requirement to identify the important parameters in the process and specify their acceptable ranges (how much oil?; what temperature?; etc.). Other important aspects are committments to testing, QA and compliance with procedures. Thus, the PCP should provide a sound basis for the licensee's control of his own activities, as well as a basis for inspection.

2. The PCP is not needed because the burial grounds and the shipping regulations establish the requirements.

   The purpose of the PCP is preventative not punitive; the principal concern is proper implementation not requirements.

3. The PCP should not be required; plant procedures should be sufficient.

   The PCP addresses issues not normally covered in procedures, viz. the the background bases and justification for the procedures. The PCP actually provides a basis for the review of procedures. Without a PCP there is no assurance that the procedures even address the important parameters.

4. There should be no need for the PCP to address dewatering of resins.

   Unfortunately, the potential for deficiencies in this simple process has been demonstrated so dewatering cannot be ignored. However, the potential for wasted effort in this area is small. If one knows what needs to be done writing this part of the PCP is easy and if one does not know, the learning effort is justified.

5. Why require seismicly qualified pads for portable solidification equipment?

   The staff has not required the application of Regulatory Guide 1.143 to such pads.

<u>40 CFR 190</u>

1. Some plants have a problem with direct radiation from stored solid wastes.

   This is not a RETS problem. Compliance with 40 CFR 190 is required, whether or not it is specified in the Tech Specs.

APPENDIX B

GENERAL CONTENTS OF THE OFFSITE DOSE CALCULATION MANUAL (ODCM*)
(Rev. 1, February 1979)

Section 1 - Set Points

Provide the equations and methodology to be used at the station or unit for each alarm and trip set point on each effluent release point according to the Specifications 3.3.3.8 and 3.3.3.9. The instrumentation for each alarm and trip set point, including radiation monitoring and sampling systems and effluent control features, should be identified by reference to the FSAR (or Final Hazard Summary). This information should be consistent with the recommendations of Section I of Standard Review Plan 11.5, NUREG-75/087, (Revision 1). If the alarm and/or trip set point value is variable, provide the equation to determine the set point value to be used, based on actual release conditions, that will assure that the Specification is met at each release point; and provide the value to be used when releases are not in progress. If dilution or dispersion is used, state the onsite equipment and measurement method used during release, the site related parameters and the set points used to assure that the Specification is met at each release point. The fixed and variable set points should consider the radioactive effluent to have a radionuclide distribution represented by normal and anticipated operational occurrences.

Section 2 - Liquid Effluent Concentration

Provide the equations and methodology to be used at the station or unit for each liquid release point according to the Specification 3.11.1.1. For systems with continuous or batch releases, and for systems designed to monitor and control both continuous and batch releases, provide the assumptions and parameters to be used to compare the output of the monitor with the liquid concentration specified. State the limitations for combined discharges to the same release point. In addition, describe the method and assumptions for obtaining representative samples from each batch and use of previous post-release analyses or composite sample analyses to meet the Specification.

Section 3 - Gaseous Effluent Dose Rate

Provide the equations and methodology to be used at the station or unit for each gaseous release point according to Specification 3.11.2.1. Consider the various pathways, release point elevations, site related parameters and radionuclide contribution to the dose impact limitation. Provide the

*The format for the ODCM is left up to the licensee and may be simplified by tables and grid printout. Each page should be numbered and indicate the facility approval and effective date.

dose factors to be used for the identified radionuclides released. Provide the annual average dispersion values (X/Q and D/Q), the site specific parameters and release point elevations.

Section 4 - Liquid Effluent Dose

Provide the equations and methodology to be used at the station or unit for each liquid release point according to the dose objectives given in Specification 3.11.1.2. The section should describe how the dose contributions are to be calculated for the various pathways and release points, the equations and assumptions to be used, the site specific parameters to be measured and used, the receptor location by direction and distance, and the method of estimating and updating cumulative doses due to liquid releases. The dose factors, pathway transfer factors, pathway usage factors, and dilution factors for the points of pathway origin, etc., should be given, as well as receptor age group, water and food consumption rate and other factors assumed or measured. Provide the method of determining the dilution factor at the discharge during any liquid effluent release and any site specific parameters used in these determinations.

Section 5 - Gaseous Effluent Dose

Provide the equations and methodology to be used at the station or unit for each gaseous release point according to the dose objectives given in Specifications 3.11.2.2 and 3.11.2.3. The section should describe how the dose contributions are to be calculated for the various pathways and release points, the equations and assumptions to be used, the site specific parameters to be measured and used, the receptor location by direction and distance, and the method to be used for estimating and updating cumulative doses due to gaseous releases. The location, direction and distance to the nearest residence, cow, goat, meat animal, garden, etc., should be given, as well as receptor age group, crop yield, grazing time and other factors assumed or measured. Provide the method of determining dispersion values (X/Q and D/Q) for releases and any site specific parameters and release point elevations used in these determinations.

Section 6 - Projected Doses

For liquid and gaseous radwaste treatment systems, provide the method of projecting doses due to effluent releases for the normal and alternate pathways of treatment according to the specifications, describing the components and subsystems to be used.

Section 7 - Operability of Equipment

Provide a flow diagram(s) defining the treatment paths and the components of the radioactive liquid, gaseous and solid waste management systems that are to be maintained and used, pursuant to 10 CFR 50.36a, to meet Technical Specifications 3.11.1.3, 3.11.2.4 and 3.11.3.1. Subcomponents of packaged equipment can be identified by a list. For operating reactors whose construction permit applications were filed prior to January 2, 1971, the flow diagram(s) shall be consistent with the information provided in conformance with Section V.B.1 of Appendix I to 10 CFR Part 50. For OL applications whose construction permits were filed after January 2, 1971, the flow diagram(s) shall be consistent with the information provided in Chapter 11 of the Final Safety Analysis Report (FSAR) or amendments thereto.

Section 8 - Sample Locations

Provide a map of the Radiological Environmental Monitoring Sample Locations indicating the numbered sampling locations given in Table 3.12-1. Further clarification on these numbered sampling locations can be provided by a list, indicating the direction and distance from the center of the building complex of the unit or station, and may include a discriptive name for identification purposes.

APPENDIX C

SOLID WASTE MANAGEMENT SYSTEMS
(Rev. 1, February 1979)

Standard Review Plan 11.4[1] and Branch Technical Position ETSB 11-3[2] require that each applicant for an operating license provide a detail description of a Process Control Program (PCP) to assure that the solid waste system will perform its intended function and that the product produced by this system contains no free water* and is a monolithic solid.

Specification 3.11.3.1 of the model Radiological Effluent Technical Specifications[3] require that the solid radwaste system be maintained and used in accordance with the PCP. NUREG-0133[4] requires that at the time an applicant/licensee submits proposed Radiological Effluent Technical Specifications that he submit the PCP for NRC review.** NUREG-0133 futher requires that the PCP be documented in the plant operating procedures.

To meet this commitment, the staff has prepared a general description of a PCP giving the essential points that should be covered by the applicant/licensee in making this submittal. Due to variations in system design and operation, the applicant/licensee should not interpret this outline to be all inclusive. The PCP is plant specific and must be established on a case-by-case basis since waste characteristics will vary from plant to plant.

PROCESS CONTROL PROGRAM

A "Process Control Program" (PCP) for a solid radwaste system shall be a manual detailing the program of sampling analysis and formulation determination by which solidification of radiaoctive wastes from liquid systems is assured. The PCP shall provide assurance that the system is operated as designed and produces a final product that contains no free water and has completely solidified all waste. If properties of the final product have been determined by the manufacturer, the PCP shall also assure tha the solidified waste products exhibit those physical and chemical properties (leachability, strength, flammability, etc.) that are characteristic of the product as demonstrated by the manufacturer for producing an acceptable solidified waste product. The PCP shall identify interfaces with other plant systems (e.g., liquid and gaseous radwsate systems), identify equipment (interlocks, alarms, monitors, etc.) which are required to be functional before processing can commence, identify administrative controls or equipment features to

*Free water is defined as uncombined water not bound by the solid matrix.
**Current (1981) position is that PCPs for operating reactors need not be submitted for review prior to implementation.

assure that operating procedures will be followed, identify the sampling requirements prior to processing and identify the various processing steps and process parametrs which provide boundary conditions within which the solid radwaste system shall be operated. Depending upon the type of waste (bead resins, powered resins, filter sludge, evaporator concentrates, sodium sulfate solutions, boric acid solutions, etc.) to be solidified and the kind of solidification agent (urea formaldehyde, cement, cement with sodium silicate, asphalt, polyester, etc.) employed, the process parameters shall include but are not limited to the type of waste, requirements for sampling prior to processing, pH, oil content, water content, temperature, ratio of solidification agent to influent waste and the ratio of solidification agent to chemical additive.

NOTE:

For operating reactors which have systems installed that are not capable of solidifying the categories of "wet" waste as defined in SRP 11.4, BTP-ETSB 11-3 or NUREG-0133, the licensee shall define the limitations of his present system and provide a Process Control Program to cover the waste that can be processed by his existing system. The licensee shall identify those wastes which cannot be solidified and indicate the method of packaging currently being employed (dewatered resins, vermiculite, etc.). In addition, the licensee shall provide a schedule for upgrading his solid waste system to provide the capability to process all types of "wet" wastes as defined in these reference documents.*

*For 1981 application to operating reactors, delete this sentence. However, in anticipation of the promulgation of 10 CFR Part 61, "Licensing Requirements for the Land Disposal of Radioactive Wastes," efforts should be initiated by licensees to modify their equipment, procedures and PCP such that they will be in compliance with the new requirements for waste form and packaging when Part 61 goes into effect.

REFERENCES

(1) Standard Review Plan 11.4, Revision 1, Solid Waste Management Systems, NUREG-75/087.

(2) Branch Technical Position - ETSB 11-3, Revision 1, "Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants," NUREG-75/087.

(3) Draft Radiological Effluent Technical Specifications for PWRs and BWRs, NUREGs 0472 and 0473.

(4) Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants, NUREG-0133.

Revision 1
November 1979

Branch Technical Position

Background

Regulatory Guide 4.8, Environmental Technical Specifications for Nuclear Power Plants, issued for comment in December 1975, is being revised based on comments received. The Radiological Assessment Branch issued a Branch Position on the radiological portion of the environmental monitoring program in March, 1978. The position was formulated by an NRC working group which considered comments received after the issuance of the Regulatory Guide 4.8. This is Revision 1 of that Branch Position paper. The changes are marked by a vertical line in the right margin. The most significant change is the increase in direct radiation measurement stations.

10 CFR Parts 20 and 50 require that radiological environmental monitoring programs be established to provide data on measurable levels of radiation and radioactive materials in the site environs. In addition, Appendix I to 10 CFR Part 50 requires that the relationship between quantities of radioactive material released in effluents during normal operation, including anticipated operational occurrences, and resultant radiation doses to individuals from principals pathways of exposure be evaluated. These programs should be conducted to verify the effectiveness of in-plant measures used for controlling the release of radioactive materials. Surveillance should be established to identify changes in the use of unrestricted areas (e.g., for agricultrual purposes) to provide a basis for modifications in the monitoring programs for evaluating doses to individuals from principal pathways of exposure. NRC Regulatory Guide 4.1, Rev. 1, "Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants," provides an acceptable basis for the design of programs to monitor levels of radiation and radioactivity in the station environs.

This position sets forth an example of an acceptable minimum radiological monitoring program. Local site characteristics must be examined to determine if pathways not covered by this guide may significantly contribute to an individual's dose and should be included in the sampling program.

AN ACCEPTABLE RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Program Requirements

Environmental samples shall be collected and analyzed according to Table 1 at locations shown in Figure 1.[1] Analytical techniques used shall be such that the detection capabilities in Table 2 are achieved.

The results of the radiological environmental monitoring are intended to supplement the results of the radiological effluent monitoring by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways. Thus, the specified environmental monitoring program provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures of individuals resulting from the station operation. The initial radiological environmental monitoring program should be conducted for the first three years of commercial operation (or other period corresponding to a maximum burnup in the initial core cycle). Following this period, program changes may be proposed based on operational experience.

The specified detection capabilities are state-of-the-art for routine environmental measurements in industrial laboratories.

Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment and other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, every effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the annual report.

The laboratories of the licensee and licensee's contractors which perform analyses shall participate in the Environmental Protection Agency's (EPA's) Environmental Radioactivity Laboratory Intercomparisons Studies (Crosscheck) Program or equivalent program. This participation shall include all of the determinations (sample medium-radionuclide combination) that are offered by EPA and that also are included in the monitoring program. The results of analysis of these crosscheck samples shall be included in the annual report. The participants in the EPA crosscheck program may provide their EPA program code so that the NRC can review the EPA's participant data directly in lieu of submission in the annual report.

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[1] It may be necessary to require special studies on a case-by-case and site specific basis to establish the relationship between quantities of radioactive material released in effluents, the concentrations in environmental media, and the resultant doses for important pathways.

If the results of a determination in the EPA crosscheck program (or equivalent program) are outside the specified control limits, the laboratory shall investigate the cause of the problem and take steps to correct it. The results of this investigation and corrective action shall be included in the annual report.

The requirement for the participation in the EPA crosscheck program, or similar program, is based on the need for independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid.

A census shall be conducted annually during the growing season to determine the location of the nearest milk animal and nearest garden greater than 50 square meters (500 sq. ft.) producing broad leaf vegetation in each of the 16 meteorological sectors within a distance of 8 km (5 miles).[2] For elevated releases as defined in Regulatory Guide 1.111, Rev. 1., the census shall also identify the locations of all milk animals, and gardens greater than 50 square meters producing broad leaf vegetation out to a distance of 5 km. (3 miles) for each radial sector.

If it is learned from this census that the milk animals or gardens are present at a location which yields a calculated thyroid dose greater than those previously sampled, or if the census results in changes in the location used in the radioactive effluent technical specifications for dose calculations, a written report shall be submitted to the Director of Operating Reactors, NRR (with a copy to the Director of the NRC Regional Office) within 30 days identifying the new location (distance and direction). Milk animal or garden locations resulting in higher calculated doses shall be added to the surveillance program as soon as practicable.

The sampling location (excluding the control sample location) having the lowest calculated dose may then be dropped from the surveillance program at the end of the grazing or growing season during which the census was conducted. Any location from which milk can no longer be obtained may be dropped from the surveillance program after notifying the NRC in writing that they are no longer obtainable at that location. The results of the land-use census shall be reported in the annual report.

The census of milk animals and gardens producing broad leaf vegetation is based on the requirement in Appendix I of 10 CFR Part 50 to "Identify changes in the use of unrestricted areas (e.g., for agricultural purposes) to permit modifications in monitoring programs for evaluating doses to individuals from principal pathways of exposure." The consumption of milk from animals grazing on contaminated pasture and of leafy vegetation contaminated by airborne

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[2] Broad leaf vegetation sampling may be performed at the site boundary in a sector with the highest D/Q in lieu of the garden census.

radioiodine is a major potential source of exposure. Samples from milk animals are considered a better indicator of radioiodine in the environment than vegetation. If the census reveals milk animals are not present or are unavailable for sampling, then vegetation must be sampled.

The 50 square meter garden, considering 20% used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and a vegetation yield of 2 kg/m^2, will produce the 26 kg/yr assumed in Regulatory Guide 1.109, Rev 1., for child consumption of leafy vegetation. The option to consider the garden to be broad leaf vegetation at the site boundary in a sector with the highest D/Q should be conservative and that location may be used to calculate doses due to radioactive effluent releases in place of the actual locations which would be determined by the census. This option does not apply to plants with elevated releases as defined in Regulatory Guide 1.111, Rev. 1.

The increase in the number of direct radiation stations is to better characterize the individual exposure (mrem) and population exposure (man-rem) in accordance with Criterion 64 - Monitoring radioactivity releases, of 10 CFR Part 50, Appendix A. The NRC will place a similar amount of stations in the area between the two rings designated in Table 1.

Reporting Requirement

A. Annual Environmental Operating Report, Part B, Radiological.

A report on the radiological environmental surveillance program for the previous calendar year shall be submitted to the Director of the NRC Regional Office (with a copy to the Director, Office of Nuclear Reactor Regulation) as a separate document by May 1 of each year. The period of the first report shall begin with the date of initial criticality. The reports shall include a summary (format of Table 3), interpretations, and an analysis of trends for the results of the radiological environmental surveillance activities for the report period, including a comparison with operational controls, preoperational studies (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the station operation on the environment.

In the event that some results are not available the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; a map of all sampling locations keyed to a table giving distances and directions from one reactor; the results of land use censuses; and the results of licensee participation in a laboratory crosscheck program if not participating in the EPA crosscheck program.

B. Nonroutine Radiological Environmental Operating Reports

"If a confirmed[3] measured radionuclide concentration in an environmental sampling medium averaged over any quarter sampling period exceeds the reporting level given in Table 4, a written report shall be submitted to the Director of the NRC Regional Office (with a copy to the Director, Office of Nuclear Reactor Regulation) within 30 days from the end of the quarter. If it can be demonstrated that the level is not a result of plant effluents (i.e., by comparison with control station or preoperational data) a report need not be submitted, but an explanation shall be given in the annual report. When more than one of the radionuclides in Table 4 are detected in the medium, the reporting level shall have been exceeded if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1$$

If radionuclides other than those in Table 4 are detected and are due from plant effluents, a reporting level is exceeded if the potential annual dose to an individual is equal to or greater than the design objective doses of 10 CFR Part 50, Appendix I. This report shall include an evaluation of any release conditions, environmental factors, or other aspects necessary to explain the anomalous result.

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[3] A confirmatory reanalysis of the original, a duplicate, or a new sample may be desirable, as appropriate. The results of the confirmatory analysis shall be completed at the earliest time consistent with the analysis, but in any case within 30 days.

TABLE 1

OPERATIONAL RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples[a] and Locations	Sampling and Collection Frequency[a]	Type and Frequency and Analysis
AIRBORNE			
Radioiodine and Particulates	Samples from 5 locations: 3 samples from offsite locations (in different sectors) of the highest calculated annual average groundlevel D/Q.	Continuous sampler operation with sample collection weekly or as required by dust loading, whichever is more frequent[e]	Radioiodine Cannister: analyze weekly for I-131
	1 sample from the vicinity of a community having the highest calculated annual average ground-level D/Q.		Particulate Sampler: Gross beta radio-activity following filter change,[b] composite (by location) for gamma isotopic[c] quarterly
	1 sample from a control location 15-30 km (10-20 miles) distant and in the least prevalent wind direction[d]		
DIRECT RADIATION[f]	40 stations with two or more dosi-meters or one instrument for measuring and recording dose rate continuously to be placed as follows: 1) an inner ring of stations in the general area of the site boundary and an outer ring in the 4 to 5 mile range from the site with a station in each sector of each ring (16 sectors x 2 rings = 32 stations). The balance of the stations, 8, should be place in special interest areas such as population centers, nearby residences, schools, and in 2 or 3 areas to serve as control stations.	Monthly or quarterly	Gamma dose monthly or quarterly

TABLE [illegible] (Continued)

Exposure Pathway and/or Sample	Number of Samples[a] and Locations	Sampling and Collection Frequency[a]	Type and Frequency of Analysis
WATERBORNE			
Surface[g]	1 sample upstream 1 sample downstream	Composite sample over one-month period[h,i]	Gamma isotopic analysis monthly. Composite for tritium analyses quarterly
Ground	Samples from 1 or 2 sources only if likely to be affected[j]	Quarterly	Gamma isotopic and tritium analysis quarterly
Drinking	1 sample of each of 1 to 3 of the nearest water supplies could be affected by its discharge 1 sample from a control location	Composite sample over two-week period[i] if I-131 anlysis is performed, monthly composite otherwise	I-131 analysis on each composite when the dose calculated for the consumption of the water is greater than 1 mrem per year.[k] Composite for Gross β and gamma isotopic analyses monthly. Composite for tritium analysis quarterly
Sediment from Shoreline	1 sample from downstream area with existing or potential recreational value	Semiannually	Gamma isotopic analyses semiannually
INGESTION			
Milk	Samples from milking animals in 3 locations within 5 km distant having the highest dose potential. If there are none, then, 1 sample from milking animals in each of 3 areas between 5 to 8 km distant where doses are calculated to be greater than 1 mrem per year[k]	Semimonthly when animals are on pasture, monthly at other times	Gamma isotopic and I-131 analysis semimonthly when animals are on pasture; monthly at other times.

TABLE 1 (Continued)

Exposure Pathway and/or Sample	Number of Samples[a] and Locations	Sampling and Collection Frequency[a]	Type and Frequency of Analysis
Milk (cont'd)	1 sample from milking animals at a control location (15-30 km distant and in the least prevalent wind direction)		
Fish and Invertebrates	1 sample of each commercially and recreationally important species in vicinity of discharge point	Sample in season, or semianually if they are not seasonal	Gamma isotopic analysis on edible portions
	1 sample of same species in areas not influenced by plant discharge		
Food Products	1 sample of each principal class of food products from any area which is irrigated by water in which liquid plant wastes have been discharged	At time of harvest[1]	Gamma isotopic analysis on edible portion.
	3 samples of broad leaf vegetation grown nearest offsite locations of highest calculated annual average ground-level D/Q if milk sampling is not performed	Monthly when available	
	1 sample of each of the similar vegetation grown 15-30 km distant in the least prevalent wind direction if milk sampling is not performed	Monthly when available	

TABLE 1 (Continued)

[a]The number, media, frequency and location of sampling may vary from site to site. It is recognized that, at times, it may not be possible or practical to obtain samples of the media of choice at the most desired location or time. In these instances suitable alternative media and locations may be chosen for the particular pathway in question and submitted for acceptance. Actual locations (distance and direction) from the site shall be provided. Refer to Regulatory Guide 4.1, "Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants."

[b]Particulate sample filters should be analyzed for gross beta 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air or water is greater than ten times the yearly mean of control samples for any medium, gamma isotopic analysis should be performed on the individual samples.

[c]Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

[d]The purpose of this sample is to obtain background information. If it is not practical to establish control locations in accordance with the distance and wind direction criteria, other sites which provide valid background data may be substituted.

[e]Canisters for the collection of radioiodine in air are subject to channeling. These devices should be carefully checked before operation in the field or several should be mounted in series to prevent loss of iodine.

[f]Regulatory Guide 4.13 provides minimum acceptable performance criteria for thermoluminescence dosimetry (TLD) systems used for environmental monitoring. One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposes of this table, a thermoluminescent dosimeter may be considered to be one phosphor and two or more phosphors in a packet may be considered as two or more dosimeters. Film badges should not be used for measuring direct radiation. The 40 stations is not an absolute number. This number may be reduced according to geographical limitations, e.g., at an ocean site, some sectors will be over water so that the number of dosimeters may be reduced accordingly.

[g]The "upstream sample" should be taken at a distance beyond significant influence of the discharge. The "downstream" sample should be taken in an area beyond but near the mixing zone. "Upstream" samples in an estuary must be taken far enough upstream to beyond the plant influence.

[h]Generally, salt water is not sampled except when the receiving water is utilized for recreational activities.

[i]Composite samples should be collected with equipment (or equivalent) which is capable of collecting an aliquot at time intervals which are very short (e.g., hourly) relative to the compositing period (e.g., monthly).

[j]Groundwater samples should be taken when this source is tapped for drinking or irrigation purposes in areas where the hydraulic gradient or recharge properties are suitable for contamination.

[k]The dose shall be calculated for the maximum organ and age group, using the methodology contained in Regulatory Guide 1.109, Rev. 1., and the actual parameters particular to the site.

[l]If harvest occurs more than once a year, sampling should be performed during each discrete harvest. If harvest occurs continuously, sampling should be monthly. Attention should be paid to including samples of tuborous and root food products.

TABLE 1 (Continued)

Note: In addition to the above guidance for operational monitoring, the following material is supplied for guidance on preoperational programs.

Preoperational Environmental Surveillance Program

A Preoperational Environmental Surveillance Program should be instituted two years prior to the institution of station plant operation.

The purposes of this program are:

1. To measure background levels and their variations along the anticipated critical pathways in the area surrounding the station.

2. To train personnel

3. To evaluate procedures, equipment and techniques

The elements (sampling media and type of analysis) of both preoperational and operational programs should be essentially the same. The duration of the preoperational program, for specific media, presented in the following table should be followed:

Duration of Preoperational Sampling Program for Specific Media

6 months	1 year	2 years
. airborne iodine	. airborne particulates	. direct radiation
. iodine in milk (while animals are in pasture)	. milk (remaining analyses)	. fish and invertebrates
	. surface water	. food products
	. groundwater	. sediment from shoreline
	. drinking water	

TABLE 2

Detection Capabilities for Environmental Sample Analysis[a]

	Lower Limit of Detection (LLD)[b]					
Anaysis	Water (pCi/l)	Airborne Particulate or Gas (pCi/m3)	Fish (pCi/kg,wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)	Sediment (pCi/kg, dry)
gross beta	4	1×10^{-2}				
^{3}H	2000					
^{54}Mn	15		130			
^{59}Fe	30		260			
$^{58,60}Co$	15		130			
^{65}Zn	30		260			
^{95}Zr	30					
^{95}Nb	15					
^{131}I	1[c]	7×10^{-2}		1	60	
^{134}Cs	15	5×10^{-2}	130	15	60	150
^{137}Cs	18	6×10^{-2}	150	18	80	180
^{140}Ba	60			60		
^{140}La	15			15		

Note: This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported.

TABLE 2

NOTES

[a]Acceptable detection capabilities for thermoluminescent dosimeters used for environmental measurements are given in Regulatory Guide 4.13.

[b]Table 2 indicates acceptable detection capabilities for radioactive materials in environmental samples. These detection capabilities are tabulated in terms of the lower limits of detection (LLDs). The LLD is defined, for purposes of this guide, as the smallest concentration of radioactive material in a sample that will yield a net count (above system background) that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

where

LLD is the "a priori" lower limit of detection as defined above (as pCi per unit mass or volume). (Current literature defines the LLD as the detection capability for the instrumentation only, and the MDC, minimum detectable concentration, as the detection capability for a given instrument, procedure, and type of sample.)

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute)

E is the counting efficiency (as counts per disintegration)

V is the sample size (in units of mass or volume)

2.22 is the number of disintegrations per minute per picocurie

Y is the fractional radiochemical yield (when applicable)

λ is the radioactive decay constant for the particular radionuclide

Δt is the elasped time between sample collection (or end of the sample collection period) and time of counting

The value of S_b used in the calculation of the LLD for a particular measurement system should be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicated variance.

In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background should include the typical contributions of other radionuclides normally present in the samples (e.g., potassium-40 in milk samples). Typical values of E, V, Y and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as a posteriori (after the fact) limit for a particular measurement.*

[c]LLD for drinking water samples.

* For a more complete discussion of the LLD, and other detection limits, see the following:
(1) HASL Procedures Manual, HASL-300 (revised annually).
(2) Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry" Anal. Chem. 40, 586-93 (1968).
(3) Hartwell, J. K., "Detection Limits for Radioisotopic Counting Techniques," Atlantic Richfield Hanford Company Report ARH-2537 (June 22, 1972).

TABLE 3

ENVIRONMENTAL RADIOLOGICAL MONITORING PROGRAM ANNUAL SUMMARY

Name of Facility _____ Docket No. _____

Location of Facility _____ Reporting Period _____

(County, State)

Medium or Pathway Sampled (Unit of Measurement)	Type and Total Number of Analyses Performed	Lower Limit of Detection[a] (LLD)	All Indicator Locations Mean (f)[b] Range	Location with Highest Annual Mean: Name, Distance & Direction	Mean (f)[b] Range	Control locations Mean (f)[b] Range	Number of Nonroutine Reported Measurements
Air Particulates (pCi/m^3)	Gross β 416	0.01	0.08(200/312) (0.05-2.0)	Middletown 5 miles 340°	0.10 (5/52) (0.08-2.0)	0.08 (8/104) (0.05-1.40)	1
	γ-Spec. 32						
	^{137}Cs	0.01	0.05 (4/24) (0.03-0.13)	Smithville 2.5 miles 160°	0.08 (2/4) (0.03-2.0)	<LLD	4
	^{131}I	0.07	0.12 (2/24) (0.09-0.18)	Podunk 4.0 miles 270°	0.20 (2/4) (0.10-0.31)	0.02 (2/4)	1
Fish pCi/kg (wet weight)	γ-Spec. 8						
	^{137}Cs	130	<LLD	-	<LLD	90 (1/4)	0
	^{134}Cs	130	<LLD	-	<LLD	<LLD	0
	^{60}Co	130	180 (3/4) (150-225)	River Mile 35	See Column 4	<LLD	0

[a]See Table 2, note b.

[b]Mean and range based upon detectable measurements only. Fraction of detectable measurements at specified locations is indicated in parentheses. (f)

Note: The example data are provided for illustrative purposes only.

TABLE

REPORTING LEVELS FOR NONROUTINE OPERATING REPORTS

Analysis	Reporting Level (RL) Water (pCi/l)	Airborne Particulate or Gases (pCi/m³)	Fish (pCi/Kg,wet)	Milk (pCi/l)	Broad Leaf Vegetation (pCi/Kg, wet)
H-3	2×10^4 [a]				
Mn-54	1×10^3		3×10^4		
Fe-59	4×10^2		1×10^4		
Co-58	1×10^3		3×10^4		
Co-60	3×10^2		1×10^4		
Zn-65	3×10^2		2×10^4		
Zr-Nb-95	4×10^2				
I-131	2	0.9		3	1×10^2
Cs-134	30	10	1×10^3	60	1×10^3
Cs-137	50	20	2×10^3	70	2×10^3
Ba-La-140	2×10^2			3×10^2	

[a] For drinking water samples. This is 40 CFR Part 141 value.

Figure 1

(This figure shall be of a suitable scale to show the distance and direction of each monitoring station. A key shall be provided to indicate what is sampled at each location.)