VOLUME 2: APPENDIX G

RISK INFORMING THE MATERIALS AND WASTE ARENAS:

A Case Study on Risk Informing Site Decommissioning

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ABSTRACT

A case study on the use of risk information for site decommissioning has been performed by Brookhaven National Laboratory under the sponsorship of the Risk Task Group of the NRC Office of Nuclear Materials Safety and Safeguards. The facility selected for the study is the Trojan Nuclear Power Plant, which has requested unrestricted release of its site from its license under 10 CFR 50. Information related to the specific decommissioning activities and the past and potential use of risk information are presented. Discussion is provided of the characterization of risk, past and potential approaches to its assessment, and the formulation of safety goals in this area. The study follows the approach outlined in the Case Study Plan that was developed by the Risk Task Group. Responses are given for the Draft Questions and the Draft Screening Criteria contained in the Plan.

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TABLE OF CONTENTS

Page No.

TABLE OF CONTENTS (Cont.)

Page No.

TABLE OF CONTENTS (Cont.)

Page No.

TABLES

FIGURE

1. INTRODUCTION

A case study exploring the use of risk information for site decommissioning has been performed by Brookhaven National Laboratory under the sponsorship of the Risk Task Group of the NRC Office of Nuclear Materials Safety and Safeguards. The facility selected for the study is the Trojan Nuclear Power Plant (TNPP) that is currently undergoing facility decommissioning under the provisions of 10 CFR 50.82. TNPP is aiming to terminate its Part 50 license and achieve unrestricted release of the site in compliance with the requirements of the license termination rule, 10 CFR 20 Subpart E. This draft report presents information related to the provisions of the license termination rule, a review of specific decommissioning activities at TNPP and the past and potential use of risk information. Discussion is provided of the characterization of risk, past and potential methods for its assessment, and the formulation of safety goals in this area. The study follows the approach outlined in the Case Study Plan [1] that was developed by the Risk Task Group. Responses are given for the Draft Questions and the Draft Screening Criteria contained in the Plan.

A framework for risk-informed regulation in the Office of Nuclear Material Safety and Safeguards has been issued as SECY 99-100 [2]. It was subsequently noted that identification of candidate regulatory applications that are amenable to expanded use of risk assessment information is a key step in implementing the framework. Draft screening criteria were developed by the staff to identify the candidate regulatory applications. It was determined that a series of case studies will be performed that spans a wide range of materials use and waste activities. The Case Study Plan delineates the strategy for accomplishing this objective.

Site decommissioning is one of eight case study areas that have been identified in [1] for evaluation. In this study, the Trojan Nuclear Power Plant decommissioning activity has been selected as the focal point for evaluation. The objective of this case study is to test the draft screening criteria and to derive safety goals, implicit or explicit, or elements of safety goals in Commission decisions related to the site decommissioning activity. Reference [1] also contains a series of draft questions that define the approach to be taken in this case study evaluation.

The Trojan decommissioning case study is focused on a very specific aspect of the overall decommissioning activity. This focus is the License Termination Plan proposed by the licensee, Portland General Electric Company, the NRC License Termination Rule, and the review of the licensee's Plan by the NRC. The objective of the licensee is to obtain an unrestricted release of the site as per the requirements of 10 CFR 20, Subpart E, from its license. The case study has been defined to not include the potential risks posed by the following activities: 1) the spent fuel in the pool, 2) the plans for removing the spent fuel to an independent spent fuel storage installation, and 3) the removal of the reactor vessel, steam generators, and other equipment. These are also activities related to the decommissioning process. Some of them are addressed elsewhere in NRC programs. For example, spent fuel pool risks have been addressed recently in studies by the NRC [3] and by the industry. Similarly, the removal of the vessel and other equipment has also been addressed separately in analyses carried out by NRC staff [4]. The transfer of the fuel from the pool to the independent spent fuel storage facility is currently planned for the 2003-2004 time frame. Thus this case study is not a comprehensive view of the risks associated with TNPP decommissioning. Rather, it extracts a specific aspect of the overall decommissioning activity, namely the license termination process, for analysis. From a risk perspective, it represents a partial contribution to risk while spent fuel is still on the site.

As part of the case study, a stakeholder meeting was held on May 11, 2001 to solicit stakeholder input. The transcript of this meeting can be found on the ADAMS Reference System [5] on the website maintained by NRC. The authors of this case study also toured the Trojan plant and met with representatives of Portland General Electric, NRC, and the State of Oregon.

In Section 2 of this report, we present an overview of site decommissioning with a focus on the regulatory aspects, in particular, the criteria provided by the License Termination Rule, that are broadly applicable to other nuclear facilities also, besides nuclear power reactors. Section 3 reviews the specific Trojan case including the current status and planned activities. Section 4 provides background to risk-informed concepts pertinent to decommissioning and Section 5 discusses risk-related methods for this case study. Section 6 introduces and reviews, in a preliminary fashion, Draft Safety Goals that can potentially be useful in risk-informing site decommissioning. In Section 7, we provide a preliminary review and response to the Draft Questions of the Case Study Plan. The Draft Questions call for responses to the Draft Screening Criteria and our preliminary responses are provided in Section 8. Finally, Section 9 contains the summary and conclusions of this work.

2. SITE DECOMMISSIONING - GENERAL

We present below an overview of the decommissioning process with a focus on the relevant areas of site decommissioning appropriate to nuclear power reactors. We recognize that site decommissioning has a broader focus beyond nuclear power reactors. Facilities and sites that are licensed under 10 CFR 30 (byproduct material facilities), 10 CFR 40 (source materials), 10 CFR 70 (special nuclear materials), and 10 CFR 72 (spent fuel) are also subject to a regulatory process for decommissioning that includes compliance with the License Termination Rule [6]. The focus of this overview is first the License Termination Rule and the regulatory framework for decommissioning. This is followed by a review of the decommissioning activities at the Trojan nuclear power plant and the License Termination Plan prepared by the licensee, Portland General Electric Company. Risk informing the decommissioning process, in particular the achievement of compliance with the criteria in the License Termination Rule, is discussed and a brief summary of methods for risk assessment pertinent to decommissioning is provided. One important feature of the familiarization process was to have discussions with staff in the relevant offices of the Nuclear Regulatory Commission, in particular NMSS staff who have responsibility for review of the methods chosen by the licensee to comply with the License Termination Rule and NRR staff who are responsible for regulatory oversight of the Trojan site. Another important familiarization activity was the visit to the Trojan site on April 3 and 4, 2001. This visit included a tour of the plant, discussions with the staff of the Trojan plant and Portland General Electric, and a discussion with a representative of the Oregon Office of Energy.

2.1 Regulatory Framework

Decommissioning is defined as permanently removing a nuclear facility, such as a commercial power reactor, from service and reducing radioactive material on the site to levels that would permit termination of the license issued by the NRC and release of the site for either unrestricted or restricted use. A final rule amending earlier regulations on decommissioning procedures was published by the NRC in 1996 [7]. This rule clarified the regulations for decommissioning nuclear power facilities by amending specific regulations of 10 CFR 2 [8], 10 CFR 50 [9], and 10 CFR 51 [10]. It was designed to establish a level of NRC oversight commensurate with the level of safety concerns expected during decommissioning activities and it also increased opportunities for the public to be informed about a licensee's decommissioning plans and activities. Basically, the new rule allowed power reactor licensees to use the process under 10 CFR 50.59 [11] to make changes to their facilities while decommissioning. Also, the radiological criteria for license termination were changed from the older, deterministic radioactive contamination- based requirements to more risk and performance-based dose requirements by amending 10 CFR 20 [12].

The regulatory framework for decommissioning of a nuclear power reactor site is provided in 10 CFR 50.82. Similar regulations for the decommissioning of byproduct material facilities are given in 10 CFR 30.36, for source material facilities and sites in 10 CFR 40.42, for special nuclear material facilities in 10 CFR 70.38, and for spent fuel storage installations in 10 CFR 72.106. 10 CFR 20 Subpart E provides the regulatory basis for determining the extent to which the land and the structures housing the licensed nuclear facility must be remediated, i.e., cleaned up of the remaining radioactive contamination, before the license under Part 50, or under Parts 30, 40, 70, and 72, is terminated and the site can be considered decommissioned. A summary of the Part 20 criteria for license termination is given below in Table 1.

Table 1: Radiological Criteria for License Termination

	Unrestricted Release	Restricted Release	
Dose Criterion	25 mrem/year TEDE peak dose to average member of critical group	25 mrem/yr. TEDE peak dose to average member of critical group while controls are in place*	100 mrem/year or 500 mrem/year TEDE peak dose to average member of critical group upon failure of controls
Time frame (years)	1000	1000	1000
Other require ments	ALARA	ALARA, financial assurance, and public participation	ALARA, financial assurance, and public participation

***Note: Alternate criteria for TEDE peak dose up to 100 mrem/year with specific Commission approval**

10 CFR 20.1402 provides the criteria for unrestricted release of the site. A site is acceptable for unrestricted release if the residual radioactivity above background that is present results in a total effective dose equivalent (TEDE) to an average member of a critical group that does not exceed 25 mrem (0.25 mSv) per year, including that from groundwater sources of drinking water, and the residual radioactivity has been reduced to levels that are as low as reasonably achievable (ALARA). Determination of ALARA must take into account any detriments, e.g., deaths from transportation accidents that are expected to result from decontamination and waste disposal.

For reactor licensees,10 CFR 20.1403 provides criteria for restricted release. A site is acceptable for license termination under restricted conditions if the following conditions are met:

6. A licensee can demonstrate that the residual radioactivity associated with restricted conditions is ALARA, and that further reductions to comply with the levels in 20.1402 would result in net public or environmental harm;
7. A licensee has provided legally enforceable institutional controls such that the TEDE from residual radioactivity above background to the average member of the critical will not be above 25 mrem/year;
8. The licensee has provided sufficient funds to enable an independent third party to provide necessary control and maintenance of the site;
9. A licensee has submitted a license termination plan (LTP) or decommissioning plan specifying a restricted release and documenting how public comments have been obtained and incorporated in the plan;
10. Residual radioactivity at the site has been reduced such that if there were no institutional controls, the TEDE to the average member of the critical group is ALARA and would not exceed either 100 mrem/year, or, under certain conditions, would not exceed 500 mrem/year.

11. If the 500 mrem per year value is used, the licensee: (a) must demonstrate that further reductions in residual radioactivity needed to comply with 100 mrem per year are either not technically achievable or would cause net public or environmental harm or be prohibitively costly, (b) makes provision for durable institutional controls, and (c) provides funds to enable an independent third party to carry out periodic rechecks of the site every 5 years to assure that controls remain in place.

Part 20.1404 of Subpart E also provides alternate criteria for license termination. To meet the alternate criteria, the licensee has to demonstrate that:

1. based on analysis of all possible sources of exposure, it is unlikely that the dose from all man-made sources combined, other than medical, would be more than the 100 mrem/year TEDE limit for individual members of the public from all licensed activities stated in Subpart D of 10 CFR 20,
2. restrictions have been employed at the site to minimize exposures,
3. doses are ALARA taking into consideration detriments such as traffic accidents expected to result from decontamination and waste disposal activities,
4. a decommissioning or license termination plan has been submitted specifying that the licensee proposes to decommission using alternate criteria and documenting how the advice of individuals and institutions in the community who may be affected has been sought and addressed.

Specific licenses, including expired licenses, will be terminated by written notice to the licensee when the Commission determines that (1) The decommissioning has been performed in accordance with the approved final decommissioning plan and the order authorizing decommissioning; and (2)(i) A radiation survey has been performed which demonstrates that the premises are suitable for release in accordance with the criteria for decommissioning in 10 CFR part 20, subpart E; or (ii) Other information submitted by the licensee is sufficient to demonstrate that the premises are suitable for release in accordance with the criteria for decommissioning in 10 CFR part 20, subpart E.

2.2 Decommissioning of Nuclear Power Reactors

Three methods for decommissioning of nuclear power reactors were discussed and evaluated by the NRC in a Final Generic Environmental Impact Statement published in August 1988 [13]:

- DECON involves removing or decontaminating structures, equipment and land at the facility such that the facility license can be terminated relatively soon after operations are terminated,
- SAFSTOR, in which the facility is placed in a stable condition, spent fuel is removed from the reactor vessel, radioactive fluids are drained and processed, and radioactive decay is allowed to occur for an extended period thus reducing the amount of contaminated material that must ultimately be disposed of.
- ENTOMB, in which the radioactive structures and systems are encased in a structurally stable and long-lived substance such as concrete and the entombed structure is maintained and surveyed until radioactive decay allows license termination.

Most of the reactor sites currently undergoing decommissioning, such as Trojan, Maine Yankee, Connecticut Yankee, and Saxton, are using a combination of the first two methods that has been recognized as acceptable by the NRC staff. The third method has been used in the past, e.g., at Hallam, and has been discussed recently in SECY-99-187, SECY-00-0129 and in the Staff Requirements Memorandum related to SECY-00-0129, and in a letter of May 24, 2001 from ACNW to the Commission.

The amended rule is based on the recognition that power reactors that are permanently shutdown and have no fuel remaining in the reactor vessel represent a reduced risk to public health and safety.

The time frame for decommissioning of power reactors is generally divided into three phases: (1) initial activities following the change over from an Operating License to a Possession Only License, (2) major decommissioning and storage activities, and (3) license termination activities. Within two years of permanent shutdown of operations, the licensee is required to submit a post-shutdown decommissioning activities report (PSDAR). The PSDAR is prepared in accordance with Draft Regulatory Guide DG-1067 [14] and contains a description of the planned decommissioning activities, the significant milestones in the process, an estimate of costs, and an evaluation of associated environmental impacts. Ninety days after receipt of the PSDAR by the NRC, the licensee can start to carry out major decommissioning activities without specific NRC approval using the 50.59 process.

10 CFR 50.2 defines major decommissioning activity as "permanent removal of major radioactive components, permanent modifications of the structure of the containment or dismantling components for shipment containing greater than Class C waste". Such components could include " reactor vessel and internals, steam generators, pressurizer, large bore reactor coolant piping, and other large components that are radioactive." However, 10 CFR 50.82(a)(6) requires that the licensee must not perform any decommissioning activity that either (1) forecloses unrestricted release of the site, or (2) causes any unreviewed environmental impact, or (3) results in an inadequacy of funds. If any decommissioning activity could lead to a violation of any conditions, a license amendment request has to be submitted that would result in an opportunity for a public hearing. As part of annual inspection under 50.59, NRC staff will evaluate the licensee's decommissioning procedures to ensure that the above conditions are not violated during changes to the facility.

In order to terminate the Part 50 license, a LTP has to be submitted at least two years before the expected termination of the license. The LTP is prepared in accordance with Regulatory Guide 1.179 "Standard Format and Content of License Termination Plans for Nuclear Power Reactors" [15] and is submitted as a supplement to the Final Safety Analysis Report (FSAR). The LTP is meant to address each of the criteria in 10 CFR 50.82(a)(9) and the radiological criteria in 10 CFR 20 Subpart E. These criteria include: (1) site characterization, (2) identification of remaining site dismantlement activities, (3) plans for site remediation, (4) detailed plans for final radiation surveys to determine release of the site, (5) a description of the end-use of the site for a restricted release, (6) updated site-specific estimate of remaining decommissioning costs, (7) a supplement to the environmental report, under Part 51.33, describing any new information or significant environmental change associated with the

licensee's termination activities, and (8) method for demonstrating compliance with the radiological criteria for license termination, including, for restricted release, documentation on public consultation, institutional controls and financial assurance.

2.3 Achieving Compliance with Dose Criteria

Demonstration of compliance with the dose criteria in the LTR requires a framework for dose assessment and decision making at sites where the licensee has decided to terminate the license. Overall guidance on developing such a framework is provided in NUREG-1549 [16] and further elaborated in the Draft Regulatory Guide DG-4006 [17], now replaced by the Standard Review Plan, NUREG/SR-1727 [18].

NRC guidance on a generic screening dose assessment is based on a screening approach developed in the DandD code, NUREG/CR-5512 [19]. This method is based on the assumption that screening dose assessments are performed with relatively little site-specific information and that the results of these models are conservative, i.e., result in more restrictive dose criteria. Hence licensees using the screening approach would expend less resources than those needed for a more realistic dose estimate and would also have a high assurance that the criteria in Part 20 Subpart E would be met. However, the approach can be modified to incorporate more site-specific information for licensees with more complex situations or where the screening approach would lead to unnecessarily conservative results.

The models, scenarios, and parameters in the DandD code are expected to lead to "reasonably conservative" but not bounding results. The models and generic physical parameters for transport of radionuclides through the environment to a receptor are based on expected variability across the United States and the behavioral and metabolic parameters that translate exposure to dose represent the expected range between individuals in the defined critical group. The approach provides a basis for both screening calculations and for changing model assumptions and parameters in more complex situations. A "reasonable" treatment of uncertainty is incorporated in the model to provide the regulator with a level of confidence that the actions taken are consistent with the regulations.

Guidance and models on site-specific dose assessment are provided in the RESRAD and RESRAD-BUILD codes [20]. These codes were originally developed at Argonne National Laboratory under US DOE sponsorship to evaluate dose to an average individual of a critical group living or working at a site contaminated with residual radioactivity. They have been modified under NRC sponsorship to provide site-specific dose modeling for use with the Standard Review Plan for decommissioning. The RESRAD code applies to doses from contaminated soil and RESRAD-BUILD to contaminated building surfaces. Probabilistic dose modeling is performed within these codes based on parameter distribution functions that incorporate physical, behavioral, and metabolic uncertainty in the system and receptors being modeled and using a stratified Latin Hypercube Sampling (LHS) method.

The decision framework for demonstrating compliance with the dose criteria in Part 20 is illustrated in NUREG-1549 and in more detail in Appendix C of NUREG/SR-1727. The framework consists of the following steps:

Step1: Gather existing data and information from site characterization and the amounts and types of radioactive materials present to define a source term.

Step 2: Define the scenarios and exposure pathways important to the site dose assessment, generic pathways are defined in the DandD code.

Step 3: Conceptualize the system, i.e., identify the migration and exposure pathways for the radionuclides, the potential human activities that can result in exposure, and the members of the critical group who can be potentially exposed in the future. For those licensees using the DandD code this step has been completed by the NRC based on the models, pathways and critical groups described in NUREG/CR-5512.

Step 4: Carry out the dose assessment. For generic screening the DandD code with the models and default parameters is executed with the site-specific source term obtained in Step 1.

Step 5: The results of the dose assessment are compared with the dose criterion of 25 mrem/year. If the calculated dose is less than the criterion, proceed to Step 6. If not, the generic screening approach cannot be used and a site-specific method is needed. Further information on site-specific approaches is contained in NUREG-1549 and in the Standard Review Plan for Decommissioning, NUREG/SR-1727.

Step 6: If the calculated dose meets the 25 mrem per year criterion, then the licensee can proceed to meet ALARA requirements. Methods acceptable to the NRC staff in demonstrating ALARA are contained in NUREG/SR-1727 and DG-4006.

Step 7: Implement the guidance on carrying out the final status survey prior to license termination as provided in NUREG/SR-1727 and in NUREG-1575 [19]**.**

The generic approach uses the concept of a Derived Concentration Guideline Level, DCGL, which is the concentration of residual radioactivity above background that, if distributed uniformly throughout a survey unit, would result in a TEDE of 25 mrem per year to the average member of the critical group. The critical group is conceived of as that group of individuals who would reasonably be expected to have the greatest exposure to residual radioactivity for all future circumstances. Two critical groups and associated exposure scenarios are defined in the DandD code's generic approach:

(a) A building occupant who occupies a previously contaminated building that has now

become a commercial building after license termination. This scenario allows exposure to fixed and removable surface contamination and includes the following pathways: external exposure to radiation from surface sources, inhalation of resuspended surface contamination, and ingestion of surface contamination. A look-up table of screening DCGLs for building surface contamination based on the DandD code is provided in NUREG/CR-5512 and in Appendix C of NUREG/SR-1727.

(b) A resident farmer who lives and farms on contaminated land. This scenario includes the following exposure pathways: external exposure from volumetric soil sources while outdoors and indoors, inhalation from resuspension of contaminated soil both outdoors and indoors including surface soil that is tracked indoors, ingestion of soil including inadvertent ingestion of soil tracked indoors, ingestion of drinking water from a contaminated groundwater source, ingestion of plants grown in contaminated soil, ingestion of plants watered by contaminated groundwater, ingestion of animal products from animals fed onsite and ingesting contaminated plants, groundwater, and soil, and ingestion of fish from contaminated surface water sources. A look-up table of screening DCGLs for soil contamination is provided in NUREG/CR5512 and NUREG/SR1727.

The dose modeling is performed in conjunction with the calculation of area factors and the elevated measurement comparison value in making decisions on remediation and conducting the final status survey to demonstrate compliance with the radiological dose criteria in the license termination rule. The area factor is used to calculate the maximum concentration that can remain in a specific area without necessitating additional clean-up; it is a multiple of the DCGL that is permitted to remain in an area of elevated residual radioactivity. The site is classified into impacted and non-impacted areas based on expected levels of residual radioactivity from licensed operation. Impacted areas are those where there is reasonable potential for residual radioactivity from licensed activities. The impacted areas are further subdivided into 3 classes: Class 1 is that area where, before remediation, the expected residual radioactivity is greater than the DCGL, Class 2 where the expected residual radioactivity is below DCGL, and Class 3 where there is not expected to be any significant residual radioactivity. Non-impacted areas are those where there is no expectation of residual radioactivity from licensed operation and do not have to undergo a final status survey prior to release.

The impacted areas are subdivided into survey units for the purpose of measurement of the radionuclide concentration. Guidance on selecting the size of survey units for building structures and soils is provided in MARSSIM [21]. Further guidance on the design and analysis of the final radiation status survey for buildings and soil is provided in NUREG-1505 [22]. Typical survey unit areas recommended in MARSSIM are: Class 1 areas, 100 m^2 for structure floor area to 2000 m^2 for land, and, for Class 2 areas, 100-1000 m^2 for structures and 2000-20000 m^2 for land. Technical guidance on minimum detectable concentrations for various contaminants and field conditions is contained in NUREG-1507 [23]. Background reference areas are required by the methods in MARSSIM if the residual radioactivity in the impacted area contains a radionuclide that occurs in background or the sample measurements to be made are not radionuclide specific. Background areas are not needed if radionuclide-specific measurements are used to measure concentrations of a radionuclide that is not present in the

background. Guidance on selecting background reference areas for both soils and buildings is provided in NUREG-1727.

All survey units should be evaluated to determine if the average concentration in the unit is below the DCGL. If multiple radionuclides are present at concentrations that are greater than 10 percent of their respective DCGL, then gross activity DCGLs, called $DCGL_{GA}$, are developed and applied for building surfaces and plant systems. The gross activity DCGL enables field measurement of gross activity rather than the estimation of individual radionuclide activity for comparison to the radionuclide-specific DCGL. Gross activity DCGLs, $DCGL_{GA,}$ are defined as the relative fraction of the total activity contributed by each radionuclide within the survey area

$$DCGL_{GA} = \frac{1}{(f_1/DCGL_1) + (f_2/DCGL_2) + ...+ (f_n/DCGL_n)}$$

where f_n = fraction of total activity contributed by the nth radionuclide and $DCGL_n$ = derived concentration guideline level for radionuclide n. The gross activity DCGLs are calculated using the relative nuclide fractions from samples of building surfaces before remediation or plant system materials. When the concentrations of different radionuclides appear unrelated, either the most restrictive DCGL has to be applied to all radionuclides or the unity rule has to be used:

$$(C_1 /DCGL_1) + (C_2 /DCGL_2) + (C_3 /DCGL_3) +.... \leq 1$$

where C_n = concentration of radionuclide n and $DCGL_n$ is the derived concentration guideline level for radionuclide n.

Details of the estimation of gross activity and the application of the unity rule are contained in the MARSSIM report. The methodology requires the licensee to establish data quality objectives for the data obtained in the final status survey. These objectives include:

(1) The selection of an appropriate statistical test discussed in MARSSIM. The Wilcoxon Rank Sum (WRS) test is used when the radionuclides of concern are present in background or gross measurements are made. It requires that background reference areas be identified from which the same sample is collected as from the survey unit, the data from the reference area are adjusted for the DCGL and the two data sets are compared to determine compliance. The minimum number of samples needed in each survey unit for the WRS test is given by equation 5-1 of the MARSSIM report. If the radionuclide of concern is not present in background or is present at a small fraction of the DCGL, the Sign test is employed to determine compliance. The number of samples needed in a survey unit for the Sign test is given by equation 5-2 of MARSSIM.

(2) The input parameters for sample size calculations include the DCGL and the lower bound of the gray region (LBGR), which is the concentration to which the survey unit must be remediated to have an acceptable probability of passing the statistical tests referred to above. The LBGR represents the lower bound of the uncertainty for the

concentration of residual radioactivity in the survey unit; the DCGL is the upper bound. These parameters are used along with the limits on decision errors to calculate the number of samples needed in the above tests.

2.4 ALARA Considerations

The ALARA evaluation is built on the concept of action levels called remediation levels (RL). RL is the level of residual radioactivity defined in terms of the ratio of a concentration of a given radionuclide to the derived concentration guideline level, DCGL. (The DCGL, as defined above, is the average concentration of residual radioactivity at a site that would result in a dose of 25 mrem per year to an average member of the critical group). RL is defined as a level at which the desired benefit of the remediation action, that is, the value of averted future doses to the critical group members, is equal to the impact, i.e., the cost, of the action.

$$RL = CONC / DCGL$$

where,

CONC = average concentration of residual radioactivity in the area being evaluated in units of activity per unit area (Ci/m^2) for building surfaces and activity per unit volume (Ci/m^3) for soils,

DCGL = derived concentration guideline level equivalent to the average concentration of residual radioactivity (in the same units as CONC) that would give a dose of 25 mrem/year to the average member of the critical group.

The benefit of the remediation action, B, in monetary units is given by

$$B = \$2000 \times PV(AD)$$

Where,

PV(AD) = present value of the future averted collective dose through the action measured in person-rem over a period of N years over which the collective dose is calculated,

and, \$2000 = monetary value of a person-rem averted in dollars [24].

PV(AD) is obtained by discounting future averted doses over the N year period and is given by:

$$PV(AD) = PD \times A \times F \times [CONC/DCGL] \times 0.025 \times [1 - e^{-(r+\lambda)N}]/(r+\lambda)$$

Where,

A = area (m^2) over which the averted dose is calculated,

PD = population density for the critical group (people/m^2) in the area,

F = fraction of residual radioactivity removed by the action, i.e., its effectiveness,

0.025 = rem per year dose to average member of critical group from a radioactive concentration equivalent to the DCGL,

r = discount rate (per year) used to discount future averted doses,

λ = radioactive decay constant (per year) for the particular radionuclide of interest, and

N = number of years over which the averted dose is calculated.

If the total cost of the action = \$C, then RL is that value of CONC/DCGL when

\$C = \$B, i.e.,

$$RL = C/ \{(2000) \times (PD) \times (A) \times (.025) \times (F) \times [1 - e^{-(r+\lambda)N}]/(r+\lambda) \}$$

If more than one radionuclide is present, then the total benefit is the sum of the collective averted dose for each radionuclide.

The costs of the action may consist of several components: costs of the action itself, costs for transport and disposal of waste generated, cost of worker accidents while carrying out the action, costs of traffic accidents during waste transport, monetary cost of worker exposure while performing the action and during waste transport, monetary cost of public exposure from excavation, transport, and disposal of waste, and other, miscellaneous costs. A discussion of the various costs and numerical examples comparing costs with benefits in defining remediation levels in the context of ALARA is provided in DG-4006 and, in detail, in NUREG-1727 that has replaced the earlier draft guide DG-4006.

3. SITE DECOMMISSIONING -TROJAN NUCLEAR POWER PLANT

3.1 Background

The Trojan Nuclear Power Plant (TNPP), a 3411 MWt four-loop Westinghouse pressurized water reactor, operated from 1976 to 1992 and shutdown permanently in 1993. The TNPP License Termination Plan (LTP) [25] has been prepared in accordance with 10 CFR 50.82 and the guidance provided in RG 1.179. It is maintained as a supplement to the TNPP Defueled Safety Analysis Report (DSAR). The objective of the LTP is to demonstrate that the remainder of decommissioning activities will be performed in accordance with 50.82, and will not affect public health and safety and the quality of the environment.

The TNPP Operating License was changed to a Possession Only License on 5/5/93. An Updated Final Safety Analysis Report (UFSAR) for the defueled condition was submitted in October 1993 [26]. The proposed TNPP Decommissioning Plan and Supplement to the Environmental Report was submitted in 1995 and approved by the NRC in 1996. After revision of the License Termination Rule in 1997, TNPP submitted the current LTP report.

TNPP decommissioning is divided into 2 major periods: a Transition Period and a Decontamination and Dismantlement Period. Decommissioning will be followed by site restoration.

3.1.1 Transition Period

The Transition Period began in 1993 and will continue until spent fuel is transferred to the Independent Spent Fuel Storage Installation (ISFSI) planned to be erected on site. Decontamination and Dismantlement will begin once spent fuel is in the ISFSI. Site restoration will begin once the10 CFR 50 license is terminated and will involve the final disposition of structures, systems, and components.

The Transition Period is nearing completion (as of March 1999). The licensee continues to maintain systems and components required to support decommissioning and spent fuel storage in accordance with the Possession Only license. Activities include: assessing functional requirements for systems, structures, and components (SSCs), deactivating SSCs, active decontamination and dismantling of SSCs not needed for spent fuel (SF) storage in the pool, and maintaining safe SF storage.

The concrete pad and chain link fence for the Trojan ISFSI is complete. Fuel transfer is awaiting approval of a new storage cask. Completion of fuel transfer to the ISFSI will permit the removal or decontamination in-place of the SSCs that support the SF pool.

Major activities during the Transition Period also includes the removal of the steam generators, and the reactor vessel and internals (RVI) that was transported from the Trojan site to the US Ecology low level waste (LLW) facility near Richland, WA. RVI removal eliminated approximately 2 million Ci of activity from TNP, or >99% of remaining Ci activity (exclusive of the spent fuel) at the TNP facility.

3.1.2 Decontamination and Dismantlement Period

Planned activities include: removing remaining contaminated systems and components, decontaminating structures, and a final radiation survey to verify rad activity reduction to sufficiently low levels to allow unrestricted release.

Contaminated SSCs will be decontaminated or removed and packaged and shipped to an offsite processing facility, or to a LLW facility. Decontamination of structures may be completed concurrently with removing equipment and systems. Decontamination may be carried out with different methods ranging from water washing to surface material removal. Demolition of some buildings may be needed due to degraded structural integrity after decontamination and/or removal of systems, components, barriers, etc.

The final radiation survey will be performed to demonstrate compliance with criteria of 10 CFR 20.1402 to allow unrestricted release of the site and license termination.

3.1.3 Site Restoration

Non-radiological site restoration activities will occur after termination of the Possession Only license. This is scheduled to begin in 2018 and completed in 2019 (presumably after spent fuel has been removed from ISFSI).

3.2 Site Characterization

Site characterization is provided pursuant to 50.82(a)(9)(ii)(A) and Regulatory Guide 1.179 that requires description of radiological conditions at site. The site characterization incorporates survey results to quantify the extent and nature of contamination.

3.2.1 Methodology

The initial plan was developed in 1993 based on the guidance contained in Regulatory Guide 1.86 [27], NUREG/CR-5849 [28] and NUREG/CR-5512. After the initial plan was developed, revisions to the site release criteria were issued in Part 20 Subpart E and guidance on the methods required to achieve compliance with the release criteria were provided by MARSSIM (NUREG-1575). PGE has undertaken to carry out the final survey in accordance with MARSSIM guidance.

The TNP Site Characterization Plan incorporates the following objectives: determine the initial radiological status of facility, estimate site source term and isotopic mix to support decommissioning cost estimate and decision-making, and determine location and extent of any contamination outside the radiologically controlled area. Four areas were addressed in the process: structures, systems, activation, and environment. QA requirements imposed including training, procedures, records, audits, and surveillances.

3.3 Facility Radiological Status

TNPP operated commercially from May 1976 to November 1992 for 14 fuel cycles and 3300 effective full power days.

Routine gaseous and liquid effluent releases during operation were monitored. Several operational events occurred that could also have an impact on decommissioning: (1) fuel assembly damage in 1981-82 that resulted in damaged fuel pellets being released into the RCS. This caused a high level of transuranics in plant systems and also led to contamination of many surfaces inside containment. (2) As per the information provided in the Trojan LTP, steam generator tube leaks occurred during several operating cycles. Leaking tubes were identified in 1978, 1979, 1981 and 1992. Steam Generator tube leaks released radioactive material to secondary side systems of plant. This contaminated areas outside the radiologically controlled area. However, offsite radiological consequences due to these events were judged to be minimal.

A Phase 1 scoping survey/site characterization was used to characterize radiological status and support objectives mentioned above. Phase II which is ongoing is supporting decontamination and dismantlement activities.

Phase 1 looked at structures, systems, activation and environment, and estimated the total waste volume and curie activity. (*Note that spent nuclear fuel, control rod elements, incore instrumentation hardware, and radioactive fluids, filters, resins contained in piping, equipment, and sumps are not considered to be decommissioning waste and not included in the activity totals. Spent fuel, control rods and incore hardware will be stored in the ISFSI*).

The following data summarizes the Phase 1 scoping and characterization survey results for structures and systems (Trojan LTP, p. 2-6, May 2000):

Structures: 0.031 Ci (includes removable and fixed contamination, but not activation, on structural surfaces in the containment, auxiliary building, turbine building, etc.)

Systems: 1070.5 Ci (not including SGs, pressurizer, or activation)

Activation: 4.2×10^{6} Ci one year after shutdown (mostly in vessel internals; activity in RV,

clad, insulation and concrete is about 3.1 x 10^4 Ci). Five years after shutdown, the calculated activity was estimated at about 2 x10^6 Ci (Trojan LTP, p. 2-6).

With the transfer of the RV and internals to the LLW site in Hanford, WA it is estimated that more than 99% of the activity has been removed from the Trojan site.

3.4 Remaining Site Dismantlement Activities

The spent fuel pool and associated systems for fuel storage have been mechanically and electrically isolated to create a SFP island so as to minimize any adverse impact by the ongoing decommissioning activities. During the Transition Period, the following activities have been completed:

(1) Removal of the 4 steam generators and pressurizer.

(2) Removal of the reactor vessel with internals.

(3) Assessment of functional requirements for plant SSCs to support safe storage of spent fuel and spent fuel pool cooling.

(4) Deactivation and/or removal of SSCs not required for spent fuel storage or to support other decommissioning activities.

(5) Redefinition of the regulatory basis for the defueled plant. The plant technical specifications were revised to reflect its permanently defueled status, the NRC issued a Possession only license, the FSAR was revised and retitled as the DSAR and the licensee submitted a proposed Decommissioning Plan, that was later approved by the NRC.

(6) Licensing and Construction of the ISFSI; a license application was submitted in March1996, NRC issued an Environmental Assessment and a Finding of No Significant Impact (FONSI) in November 1996 and also approved a physical security plan for the ISFSI contingent on PGE receiving a 10 CFR 72 license. Spent fuel will be transferred to the ISFSI and the 10 CFR 50 license terminated concurrent with the issuance of the Part 72 license.

The decontamination and dismantlement activities (D&DA) are designed to reduce radioactivity to acceptable levels to allow unrestricted release of the site. Decontamination of structures and systems may occur concurrently with equipment removal. Contaminated structural material may be sent for processing or directly to a LLW facility. The principles governing D&DA viz., the planning of work activities and the use of shielding and isolation to restrict exposure and transport of radioactivity to the environment, the decontamination methods, and the sequence of removal of materials and structures is described in the LTP.

A detailed list of the status of the major SSCs as of January 1999 is provided in the LTP. The major components already removed are identified along with those remaining to be removed. Remediation considerations for the remaining SSCs are also identified. The D&DA for the remaining SSCs are classified into 3 phases: phase 1 involves SSCs not required for spent fuel storage, phase 2 involves SSCs associated with spent fuel storage, and phase 3 other SSCs.

The LTP provides a system by system description of the decontamination of structures and removal of equipment in each of the above phases. A description of the various radiation monitoring systems, that include process and effluent radiological monitoring systems and the area radiation monitoring system, is also provided in the LTP.

During the Transition Period and D&DA, the actual and projected radiological impacts to workers from major activities is identified in the LTP as follows (Table 2)

Table 2: Worker Doses (actual and projected) from Decommissioning Activities

Activity	**Collective dose (person-rem)**
Steam Generator/Pressurizer removal	54 (actual)
Reactor Vessel/Internals removal	67
Dismantlement	
Nuclear Steam Supply System	51
Spent Fuel Racks	19
Balance of plant systems	165
Structures	46
Miscellaneous	20
Normal plant operations	30
Fuel Transfer to ISFSI	99 (2.9 p-r/cask x 34 casks)

The total collective exposure to workers from decommissioning operations is projected to be 551 person-rem of which 121 person-rem has already been incurred. The projected exposure from decommissioning is based on TNP site information. It assumes that area dose rates are based on radiation surveys that have been adjusted to account for radioactive decay to the estimated start of the decommissioning period. Exposure during the transition period is estimated to be about 4 person-rem per year excluding activities during the decontamination and dismantlement period.

The forms of waste generated during decommissioning include: contaminated water, used disposable protective clothing, expended abrasive and absorbent materials, resins and filters, contamination control materials such as strippable coatings and plastic enclosures, and contaminated equipment used in the decommissioning process. Projected waste volumes are as follows (Table 3)

Table 3: Projected Waste Volume from Trojan Decommissioning

(Trojan LTP, Table 3-4, March 1999)

Class of Waste	Volume (x 1000 cubic feet)
Class A Burial Volume	294
Class B Burial Volume	4
Class C Burial Volume	8

Decommissioning planning at TNP assumes that cost-effective methods for reducing waste volumes are limited. Significantly contaminated or activated materials are planned to be sent to a disposal facility.

3.5 Site Remediation Plan

The site remediation plan prepared as part of the Trojan LTP describes the remediation actions that will be applied to various areas of the site to ensure that future doses are as low as reasonably achievable (ALARA) and comply with the regulations in 10 CFR 20 to allow for unrestricted release. The plan identifies the remediation methods to be adopted and demonstrates that the methodology is adequate to ensure unrestricted release of the site. The remediation actions include: chemical decontamination, wiping, washing, vacuuming, scabbling, spalling, abrasive blasting, and high-pressure washing.

The Trojan LTP uses the following values of PD, r, and N, based on NRC guidance, to evaluate RL (Table 4); see Section 2.4.

Table 4: Acceptable Values for Evaluating Remediation Levels

Variable	Acceptable Value	
	Building	Land
PD (pop. density, persons/m^2)	0.09	0.0004
r (discount rate, %/year)	0.07	0.03
N (years for dose calc.)	70	1000

3.6 Summary of Final Survey Plan

As stated above, TNPP performed an initial Phase I site characterization to evaluate the extent and nature of radiological contamination at the site. The methodology provided in MARSSIM permits use of data developed during site characterization to be used as final survey data provided the data meet the same Data Quality Objectives. The major radionuclides and their isotopic distributions are provided in the LTP as per 10 CFR 61 requirements for the waste stream analysis. These fission and activation products are typical of what would be expected in PWRs.

More extensive survey information is being obtained from Phase II characterization efforts that are ongoing. These will include surface activity measurements on building interiors, in particular cracks, construction joints, etc. that are difficult to reach and interior surfaces of embedded piping, and sampling of open land, structures and environment not conducted during Phase I that are potentially contaminated with transuranics (TRU) and hard-to-detect (HTD) radionuclides. The final site survey will be based on the guidance provided in MARSSIM.

The industrial area (IA) of the site contains the main power plant structures and the radiologically controlled area (RCA) consists of the Containment Building, Aux Building, Fuel Building, and the Aux Building Piping facade. The LTP identifies the impacted and non-impacted (i.e., areas without residual radioactivity from licensed operation) areas of the site. All land and structures within the IA are classified as impacted and divided into Class 1, 2 or 3 areas. Table 5-2 in Appendix 5-2 of the LTP identifies the initial classification of the Trojan Facility and site. Class 1 areas include: the floor, lower wall and other surfaces of the structures and embedded piping within the RCA. Areas outside the IA are generally non-impacted except the oil and waste storm drains that are classified as Class 2 and temporary warehouse structure classified as Class 3.

3.6.1 Derived Concentration Guideline Levels

The dose models contained in the DandD code (NUREG/CR-5512) were used to calculate the

derived concentration guideline levels (DCGLs). Screening DCGLs are based on the building occupancy and resident farmer scenarios of the DandD code. Table 5.1 of the LTP contains the screening DCGLs based on the generic values published earlier by the NRC [29]. For certain specific radionuclides, in particular, Cs-134, Eu-152, Pu-238, Pu-239, Pu-240, Pu-241, Am-241, Cm-243 and Cm-244, building surface contamination values were calculated by DandD version 1.0. For structures where radiation from embedded piping is present, direct exposure due to the shine dose from the embedded pipes was assumed to contribute up to 5 mrem per year. This is equivalent to a uniform residual surface contamination activity level of 100,000 dpm per 100 cm^2 on the internal surface of the embedded pipe. Thus for a survey unit with embedded pipes, the applicable DCGL for the building surface is equivalent to 20 mrem per year or 0.8 of the DCGL for building surface contamination.

NRC Staff review of the building surface DCGLs at Trojan is provided in the Staff Evaluation Report issued with the approval to amend the license [30]. The staff considered 47 radionuclides or radionuclide pairs in its review and compared them against the list provided in NUREG/CR-0130 “Decommissioning a Reference PWR” [31]. The licensee performed radiochemical analyses of 15 samples from different locations within the plant. Seventeen of the 47 radionuclides considered were not detected in these samples and thus screened out. Another 8 radionuclides were screened based on decay or age considerations. Of the remaining 22 radionuclides, 8 contributed less than 5 percent to dose and were screened out using the criteria of NUREG-1727 that allows radionuclides contributing less than 10 percent of the dose limit to be screened out. Building surface DCGL values for the remaining 14 radionuclides are displayed in Table 5.

Since the building surfaces in Trojan are radiologically complex, i.e., the contaminated with multiple radionuclides, the LTP proposes that gross activity DCGL values be used for demonstrating compliance. For areas that have not been surveyed or where the nuclide distribution is yet to be determined, the LTP states that the most conservative distribution resulting in the lowest DCGL of those areas will be used.

The resident farmer scenario was used to obtain DCGL values appropriate to the volumetric contamination of soil in the yard areas of the Trojan site. These screening DCGLs were also based on the generic values published earlier by the NRC using the DandD code. Table 2 shows the screening DCGLs for volumetric radioactive contamination. The LTP indicates that surrogate ratio DCGLs will be used to demonstrate compliance in those areas where the radionuclide concentration ratios are reasonably constant. Surrogate ratio DCGLs are based on site characterization data before any remediation activities are conducted. Trojan data indicates that Cs-137 is likely to be the surrogate radionuclide.

The staff review concludes that the DCGL values established for the residual radioactivity on building structures and residual volumetric radioactivity at the Trojan site based on the building occupancy scenario should “provide reasonable assurance” that the dose criterion for unrestricted release will be met.

Table 5: Screening DCGLs at Trojan

Radionuclide	Building Surface DCGL (dpm/100cm^2)	Volumetric soil DCGL (pCi/gm)
H-3	1.2E+08	1.1E+02
C-14	3.7E+06	1.2E+01
Fe-55	4.5E+06	1.0E+04
Co-60	7.1E+03	3.8E+00
Ni-63	1.8E+06	2.1E+03
Sr-90	8.7E+03	1.7E+00
Cs-134	1.3E+04	5.7E+00
Cs-137	2.8E+04	1.1E+01
Eu-152	1.3E+04	8.7E+00
Pu-238	3.0E+01	2.5E+00
Pu-239/240	2.8E+01	2.3E+00
Pu-241	8.8E+02	7.2E+01
Am-241	2.7E+01	2.1E+00
Cm-243/244	3.9E+01	3.2E+00

3.6.2 Final Survey Design, Data Collection, and Evaluation of Statistical Compliance

Since the final survey will be carried out based on the guidance provided in MARSSIM, the design calls for an iterative process that requires site classification and formal planning using DQOs. The factors appropriate to the survey design, data collection, and compliance evaluation have been outlined in section 2.1.2 above. The LTP describes the classification of areas, the survey unit sizes, and a detailed listing of survey units.

The input parameters for the DQOs, including the variables for calculating sample size, the chosen limits on the decision errors, and the selection of the appropriate statistical tests, are listed in the LTP. The input parameters for the sample size calculation are the relevant DCGL, the LBGR and an estimate of the radionuclide variability. For planning of the survey, the LTP chooses the LBGR at 40 percent of the DCGL, estimates variability at 20 percent of the DCGL,

and establishes the default decision error at 0.05. The main concern for the NRC is what is identified in DG-4006 as the Type I or α error where a survey unit is judged to meet the release criterion when it does not. The default value of 0.05 for the Type I error is regarded as acceptable by the NRC staff. A minimum of 30 samples will be collected from each survey unit and the technical rationale for this is provided in the LTP. Staff review of the final survey design indicates that the final status survey meter-detector selection and calibration identified in the LTP is acceptable and the procedures described for assessing minimum detectable concentrations for various radionuclides of concern are acceptable and follow the guidelines of NUREG-1575.

3.7 Compliance with Radiological Criteria in Part 20

Trojan expects to release the site for unrestricted use based on the criterion of Part 20, Subpart E that the dose to the average member of the critical group will be below 25 mrem per year. Information gathered during the site visit indicated that the licensee is likely to meet this criterion without any difficulty. NRC staff review of the LTP and the issuance of the amendment to the facility operating license essentially concurs with this assessment. The site tour which included a walk-down of the radiologically controlled areas of the plant, with the exception of the containment, indicated that a large portion of the disposal and site remediation work is complete. The only significant remaining uncertainty is whether site groundwater contains any residual radioactivity from licensed operations. The licensee has committed to installing wells at the site as part of the groundwater monitoring program described in the LTP. If any radioactive contamination of groundwater from facility operation is detected then the licensee has agreed to submit site-specific volumetric soil DCGLs for Commission review and approval.

4. RISK INFORMING THE DECOMMISSIONING AREA

In this section we discuss how risk-informed notions have been and potentially can be used in the site decommissioning area. Some of the remarks in this section are highly tentative and are subject to revision at a later stage of this case study. First, in subsection 4.1, we perform a retrospective review of how doses, probabilities, and risks have been used in the regulatory structure for site decommissioning. Then, in subsection 4.2, a broader view is taken of the aspects of the current regulations that can be construed to be risk-informed and we tentatively suggest a process for expanding the risk characterization in this area.

We note that the concept of "risk-informed" that we use in this case study is the one advanced in the NRC White Paper on the subject [32]. For the convenience of the reader we excerpt some of the key elements of the NRC definition of "risk-informed" from Reference [32] and present them here.

> *A "risk-informed" approach to regulatory decision-making represents a philosophy whereby risk insights are considered together with other factors to establish requirements that better focus licensee and regulatory attention on design and operational issues commensurate with their importance to public health and safety. Where appropriate, a risk-informed regulatory approach can also be used to reduce unnecessary conservatism in purely deterministic approaches, or can be used to identify areas with insufficient conservatism in deterministic analyses and provide the bases for additional requirements or regulatory actions. "Risk-informed" approaches lie between the "risk-based" and purely deterministic approaches. The details of the regulatory issue under consideration will determine where the risk-informed decision falls within the spectrum.*

4.1 Dose, Probability, and Risk Concepts in the Site Decommissioning Area

The NRC approach to site decommissioning is a mix of deterministic and risk-based regulations. The approach to decommissioning of power reactors exemplified by Regulatory Guide 1.86 "Termination of Operating Licenses for Nuclear Reactors" [27], is a deterministic approach. The criterion for unrestricted release is acceptable surface contamination levels for a variety of radionuclides including isotopes of uranium, thorium, transuranics, Sr-90, and various beta-gamma emitters. Limits of the average, maximum, and removable levels of these radionuclides in amounts of dpm/100 cm^2 are prescribed and licensees have to demonstrate that all contamination at the site was within these limits. Licensees that are decommissioning their facilities and sites under Parts 30, 40, and 70 are required to conduct a radiation survey of the premises where the licensed activities were carried out and submit a report of the results of this survey, unless the licensee demonstrates in some other manner that the premises are suitable for release in accordance with the criteria for decommissioning in 10 CFR part 20, subpart E. The report shall, as appropriate, include: levels of gamma radiation in units of millisieverts (microroentgen) per hour at one meter from surfaces, and levels of radioactivity, including alpha and beta, in units of megabecquerels (disintegrations per minute or microcuries) per 100 square centimeters removable and fixed for surfaces, megabecquerels (microcuries)

per milliliter for water, and becquerels (picocuries) per gram for solids such as soils or concrete.

The approach outlined in Section 2.3 is based on dose criteria. An acceptable dose level of 25 mrem per year TEDE to the average member of the critical group is defined in the regulations as the criterion for unrestricted release of a reactor site that is undergoing decommissioning. Inasmuch as radiation dose translates directly into a health risk, albeit a small additional risk of latent cancer at these low levels of dose, the current regulation can be thought of as incorporating risk concepts in the decision process related to site release. As identified in the Statement of Considerations supporting the License Termination Rule [5], constant exposure to a dose level of 25 mrem/year over a period of 30 years results in an estimated additional lifetime risk of fatal cancer of about 3.8×10^{-4} (based on the BEIR-V [33] risk coefficient of 5×10^{-4} latent cancer fatalities per rem; the exposure period of 30 years is used by the EPA in estimating risk from contaminated sites and assumes that it is unlikely that a person will live or work in the same area for more than 30 years). Thus the dose level that licensees have to meet in demonstrating compliance with the criterion for license termination can be considered as an acceptable risk level established by the Commission for releasing sites that were formerly contaminated. Note that the 25 mrem/yr. level is the residual dose *above* the natural background level, which is approximately an order of magnitude greater.

There are other aspects of the current approach that utilize risk concepts. Demonstration of compliance with the rule via a hierarchical modeling approach that starts with simple screening analyses and then progresses, as needed, to more sophisticated, site-specific models can be considered to be a performance assessment. The NRC staff considers performance assessment to be a probabilistic risk assessment method applied to waste management [34].

The following elements of demonstrating compliance incorporate risk-based concepts:

(a) The simple screening approach used to establish the derived concentration guideline levels, DCGL, for individual radionuclides, while conservative, does include sampling from a range of physical parameters and behavioral parameters for the critical group. The DCGL is chosen at the 90^{th} percentile of the output dose distribution. On the other hand, the deterministic scenario that is established to estimate dose, viz., the choice of the critical group such as the resident farmer growing crops and meeting all his/her food and water needs from the land comprising the site, is designed to be conservative. This is indicated in the Standard Review Plan, NUREG/SR-1727, where it is pointed out that "In this SRP, the term probabilistic refers to a computer code or analysis that uses a random sampling method to select parameter values. The result of the calculation does not include the probability of the scenario occurring".

(b) In demonstrating compliance with ALARA, the remediation level calculation is explicitly based on a benefit-cost approach that uses many of the elements of a regulatory analysis used to analyze safety enhancements at reactors under the Backfit Rule. Under the Backfit Rule, 10 CFR 50.109, the dose reduction in person-rem from a safety enhancement, valued at $2000 per person-rem as per Regulatory Analysis Guidelines (NUREG/BR-0184), is compared with the cost of installing, operating, and maintaining the safety enhancement. The dose reduction is evaluated in risk terms as an expected

value over a distribution.

(c) The approach defined in MARSSIM for determining the acceptability of the measurements of radioactivity in the final status survey to evaluate compliance with the dose limits defined in the regulation is statistically based and incorporates probability-based evaluations of the statistical results. Hence within the boundaries of the concepts to which the statistical tests are applied, namely radiological doses to an average member of the critical group from the small amounts of contamination remaining in soil and on building surfaces, the final status survey procedures include risk concepts.

The Generic Environmental Impact Statement (GEIS) in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities [35], provides an analysis of regulatory alternatives for generic "reference" facilities for establishing radiological criteria for decommissioning structures and lands of licensed facilities. The GEIS includes facilities involved with the nuclear fuel cycle as well as those licensed to use nuclear material for other non-fuel cycle purposes. The GEIS considers both radiological and nonradiological impacts on human health and safety. This includes radiation exposure from occupancy of site buildings and residence on site lands following decommissioning and license termination. It also includes radiation exposure during decommissioning and waste for disposal. It considers nonradiological impacts on humans, including those resulting from workplace accidents and from traffic accidents during transport of decommissioning wastes.

In the GEIS, dose rates are translated into mortality rates, which are compared with mortality from accidents as described above. This is done for a parametric range of residual dose rates from 1 to 100 mrem/yr. and for various activities and facilities. This analysis provides a risk perspective for various alternatives and thus risk-informs the decision process. Cost-benefit analysis is also performed for obtaining various dose levels for release considerations for the reference facilities considered. The report uses the results obtained to support the establishment of radiological criteria for decommissioning as given in the License Termination Rule.

4.2 Towards a Risk-Informed Approach to Site Decommissioning

It is clear from SECY-99-100 [2] and the Risk-Informed Regulation Implementation Plan [36] that NRC intends to examine the extent to which risk-informed approaches can improve the regulatory process with regard to decommissioning. The Case Study Plan [1], for use of risk information in the materials and waste arenas, provides a roadmap for how the examination will be initiated. In this case study, there is a particular interest in examining the process with regard to license termination and site release.

The regulation of decommissioning of nuclear facilities by NRC relies on a process that contains an admixture of prescriptive, performance-based, deterministic, and probabilistic elements. It is prescriptive because the License Termination Rule gives specified radiological limits within a clear set of alternatives for being in compliance with regulations. It is performance-based because the LTR allows the licensee to choose the method to be used for achieving the prescribed dose criteria. It is deterministic because the parameters that are

emphasized are doses to be determined from contamination of structures and soils and the exposure routes and populations are specified in advance. It is probabilistic because the likelihood of attaining certain doses is argued for in a risk-based way by specifying parameter distributions for key variables and using statistical sampling methods like Monte Carlo sampling or Latin Hypercube sampling to sample from the distribution.

The challenge with regard to decommissioning is to consider whether it is possible to introduce a probabilistic risk assessment approach that will facilitate the regulatory process for all parties involved. In Reference [34], an example process for implementing a risk-informed approach is given. It is postulated that a hypothetical licensee for site decommissioning performs a screening analysis that fails to meet the dose criteria of the License Termination Rule. It is then postulated that the licensee performs a realistic, site-specific dose analysis that includes a quantitative uncertainty analysis for key parameters of the dose assessment. The result is then presented as a complementary cumulative distribution function (CCDF); i.e. the probability of exceeding a particular dose is given. The example, while illustrative of the risk assessment technology, stops short of indicating how the regulatory decisionmaker would actually use this information.

The License Termination Rule does not give criteria for acceptance in probabilistic terms. For unrestricted release of a site, it states that the residual dose must not exceed 25 mrem/yr for the 1000-year period and that ALARA would be used to achieve further reduction of dose. In NUREG/SR-1727, it is stated that the term "probabilistic" refers to a computer code or analysis that uses a random sampling method to select parameter values. The result of the calculation does not include the probability of the scenario occurring (the stylized resident farmer or building occupant scenarios are assumed). Further, criteria for treating uncertainty are discussed in Chapter 8 of this Standard Review Plan. There it is recognized that uncertainty in dose assessment can come from (1) uncertainty in the models, (2) uncertainty in the scenarios, and (3) uncertainty in the parameters. NUREG/SR-1727 notes that the first two sources of uncertainty are large and difficult to quantify. For these reasons, only parameter uncertainty is considered in this document--and thus, provides the only source of a probabilistic expression of the risk due to residual doses.

In order to obtain a more complete characterization of the risk, all major sources of uncertainty should be included. There are methods for working with broad uncertainty ranges to get a fuller perspective for the risk. In the next section a tentative procedure for calculating the risk is outlined. The risk could be expressed in term of doses or it can be translated into expected cancer fatalities.

For example, the likelihood of not exceeding the 25 mrem/yr. dose level could be computed, along with the likelihood of not exceeding a range of dose values. A CCDF for probability vs. dose could be computed for various levels of confidence in the resulting CCDF.

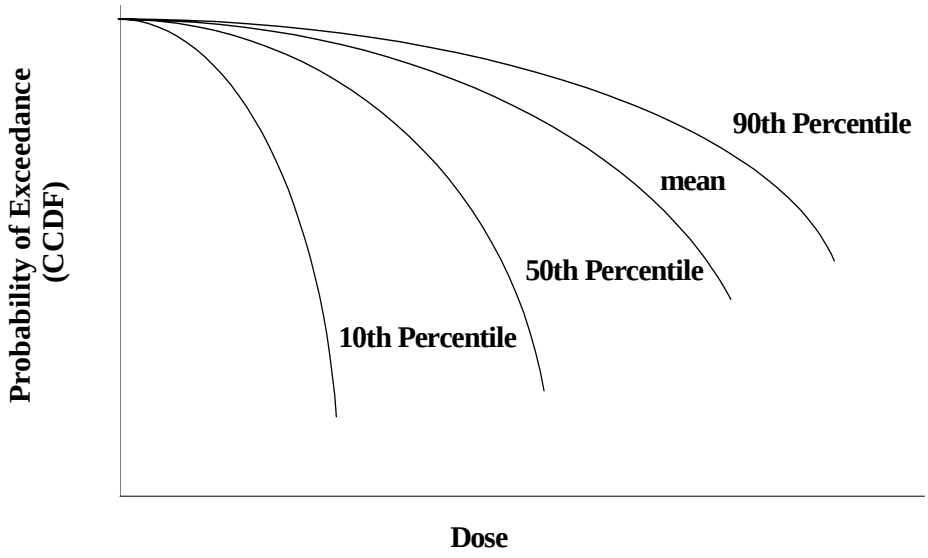

Figure 1: Hypothetical Example of Display of Risk (Dose) Results

Figure 1 is a hypothetical example of how this information would be displayed. The criterion of the LTR would then be put in the context of this enlarged information frame and a more informed decision could then be made. The display of these curves would risk-inform the regulator with regard to the soundness of his/her decision. The particular results may indicate robustness of the conclusion; on the other hand it could indicate that a more careful examination of a decision is warranted.

In this regard, NUREG/SR-1727 recognizes that in the probabilistic approach some fraction of the calculated dose may exceed the regulatory limit. It is further noted that a key Agency issue that needs to be addressed is how to interpret the results of an uncertainty analysis in the context of a prescribed regulatory limit. The NUREG suggests the use of the mean of the distribution as a useful metric. This is an issue that is very familiar in the operating reactor area. The Safety Goals Statement does mention that the mean value of the respective risk metrics (i.e., prompt fatalities or latent cancers) would be compared to the quantitative health objectives (QHOs) values (i.e. 0.1% of the current U.S. risk of prompt fatality or latent cancer from all other causes). However, the QHOs are explicitly stated in terms of frequency while the LTR criterion is not.

Similarly, if restricted site release is the chosen approach, then probabilistic analysis and results, as discussed above could, in principal, be developed to show how the 100 mrem/yr. and 500 mrem/yr criteria levels are met. Scenarios could be developed for how institutional controls could fail and their likelihoods could be quantified. A probabilistic risk expression can then be obtained for assuring that the dose criteria are met. In addition, these probabilistic methods can be used to evaluate different design alternatives for institutional controls that would be required for restricted release. Further, the same probabilistic technology can be

used to advance the ALARA assessment. For these cases, realistic models would be developed for the performance of the engineered and institutional barriers that would be in place for restricted site release.

5. METHODS FOR RISK ASSESSMENT

In order to bring risk insights to a particular assessment or decision, a method is needed to calculate the risk of the condition or activity in question. There are a wide range of activities and conditions for which risk tools have been used to calculate risk. The scope, depth of analysis, end products, and mode of inquiry are essential aspects of a risk assessment that define the particular methodology. For example, in the area of health science (or environmental) risk assessment, the top level approach is to address:

-hazard identification

-dose response assessment

-exposure assessment

-risk characterization.

These four activities are well suited to a situation in which a hazard is continuously present (this could be regarded as a "chronic" risk) and the risk needs to be evaluated. For engineered systems, the favored top level approach to risk assessment addresses the so-called risk triplet:

-what can go wrong

-how likely is it

-what are the consequences.

This approach is naturally suited to "episodic" risks, where something fails or breaks, a wrong action is taken, or a random act of nature (such as a tornado or a seismic event) occurs. This is the approach that has been taken, with much success, in probabilistic risk assessments of nuclear power reactors. The NRC White Paper on Risk-Informed and Performance-Based Regulation, [32], has an evident orientation toward reactor risks that are episodic in origin. The White Paper does note that risk insights are also derived from "predicted dose from decommissioned sites".

The environmental risk assessment methodology and the engineered system methodology do have elements in common, particularly in the dose and exposure areas. Further, in the development of physical and biomedical models in the environmental area, uncertainties in the parameters of the models and the models themselves are expressed probabilistically. This is also sometimes the case in the engineered system area.

Is it more appropriate to use an environmental risk approach or an engineered system risk approach for the dose assessments that are associated with the License Termination Rule for decommissioning? Is the distinction worth making? To understand these questions a bit more, it is instructive to introduce some related concepts. In the waste and materials areas that the NMSS regulates, SECY-99-100 (see Attachment 2 of the SECY) notes that there are methodologies (not quite probabilistic risk assessments) that have been useful. Notably, performance assessment (PA) is the method of choice for evaluation of the risk posed by a high

level waste repository. SECY-99-100 defines PA to be a type of systematic safety assessment that characterizes the magnitude and likelihood of health, safety, and environmental effects of creating and using a nuclear waste facility. Other safety analysis tools, such as systems analysis and integrated safety analysis, are defined in Attachment 2, but they are not central to this discussion. Upon reviewing some applications of PA, PRA, and PA appear to be very similar methodologies. In a recent paper, Eisenberg, et. al. [34] state the connection very succinctly and directly: “performance assessment is a probabilistic risk assessment method applied to waste management”. Further, in a paper on this subject, Garrick [37], notes that: (PRA) is identified with the risk assessment of nuclear power plants, probabilistic performance assessment (PPA), or just PA, is its counterpart in the radioactive waste field. He adds that in the mid-1990s the USNRC began to equate PRA with PA.

In Eisenberg, et. al. [34], and in SECY-99-100, PA has been recommended as the methodology for assessing the risks related to residual contamination associated with decommissioning. In order to obtain some insight on how this might proceed it is worthwhile to briefly review how PA is applied to the repository. In the risk studies described in [34], the probabilistic aspects of the assessment come from two areas: 1) the likelihood of disruption of the repository site, due to natural and man-caused events, at some time after site closure and 2) the uncertainty characterization in the physical models of the transport of radionuclides. The analyses do not provide a probabilistic depiction of habitation of the environs of the repository site at the time of exposure. Further, there is no discussion of the risk during the preclosure phase (a multi year period) of the repository in this reference.

If one now looks to the decommissioning of a nuclear facility in this light, certain parallels can be drawn in connection with the use of PA for decommissioning.

1. The period from cessation of power operations to site release is analogous to the preclosure operations phase of the repository. There is much activity on both sites during these respective phases. Worker risk can be expected to be dominant from (episodic) accidents (both radiological and nonradiological) and from (chronic) exposure to contamination.
2. The period from site release to 1000 years for a decommissioned site is analogous to the postclosure 10000-year period for the repository. The radiological source term for the former is much smaller.
3. The foregoing suggest that for the 1000 year period of the decommissioned site, the likelihood of stochastic events and their consequences be assessed. Fires and floods are two obvious candidates.
4. For both the decommissioned site and the repository the long term dose recipient is treated deterministically. This is a potential area for risk-informing the assessment in both situations.

We outline here a framework for probabilistic risk assessment that could be done in the context of the LTR. We distinguish the two phases of decommissioning that is outlined in item 1 as the pre-release and post-release phases. The post-release phase includes unrestricted and restricted site release conditions. Pre-release is included in the analysis because there may be

some activities that would be carried out during this phase that pose risks for the near term (e.g. to workers) that ought to be weighed, in a risk-informed manner, against the implications of these activities and their alternatives for the longer term post-release risks. In both the pre-release phase and the post-release phase, it is important to tailor the analysis to the risks posed by the activities and situations to be analyzed. In general, relatively more effort in the analysis should be given to the higher risk areas. Smaller risks should be evaluated with bounding and screening approaches.

E. A PRA for the pre-release phase is performed. This is based on the operations that are planned for in the licensee's post-shutdown decommissioning activities report (PSDAR) or decommissioning plan. Risks to the public and to the site workers are assessed. For the workers, both nonradiological and radiological risk scenarios are developed. Routine (chronic) and accident (episodic) risk are computed. Risks are expressed in term of health, environmental, and economic impacts.

F. A PRA for the post-release phase is performed. As discussed above, its content will be similar to the PA that is performed for the repository. Chronic doses and their implied health effects are assessed for the 1000-year period. The likelihood of the resident farmer scenario, the building occupancy scenario, and other potential scenarios as well as the nature of the critical group are evaluated probabilistically. It is recognized that there are great uncertainties in this area and these uncertainties will be expressed in the analysis. Model and parameter uncertainties for the dose assessments will also be reflected in the quantitative assessment of risk. The probabilities for the occurrence of natural events that may disrupt the site during this period will be assessed. The consequences of this event in terms of transport of radionuclides to the environment and their potential health effects will be calculated. Human intrusions to the site will also be evaluated.

For sites that have restricted release, scenarios need to be developed that will challenge the engineered and institutional barriers and controls that have been put in place to meet the requirements for restricted site release. The likelihood of these scenarios and their consequence should be quantified, including assessments of uncertainties. Public, worker, radiological, and nonradiological risks would be included in this analysis.

The long term human and environmental condition of the site is difficult to assess. Current stylized scenarios may be unduly conservative. One attempt to assess the long-term radiological impact on a site is the work sponsored by the International Atomic Energy Agency on the BIOMASS Program [38]. One objective of this program is to develop the concept of "reference biospheres" to assess the long-term safety of repositories for radioactive waste. This would also have implications for the residual radiological impact of a site that has been decommissioned. The basic idea is to develop a subset of example biospheres that can provide a useful point of reference as broadly applicable indicators of potential radiological releases. The program appears to be in an early stage of development and may be worth watching from the point of view of application to site decommissioning. The initial efforts in the program is the development of reference biospheres for radionuclides released to the biosphere via groundwater at an inland site. Alternative pathways, including human intrusion

and atmospheric transport would need to be addressed as well.

Items A and B, as outlined above, would provide an enhanced picture of the relative risks associated with decommissioning. The end uses of this assessment will help to determine the scope and depth of the various elements of the study.

6. PRELIMINARY APPROACH TO DRAFT SAFETY GOALS

6.1 Introduction

In this section, strawman safety goals are presented and discussed. They are based on the insights obtained from this case study.

Safety goals are objectives designed to guide regulatory requirements and to help regulatory decision-making in a manner consistent with NRC's legal mandate of protection of the public, the workers and the environment from licensed facilities, activities, and materials. The goals are statements of the NRC philosophy and approach to safety that provide the public with a clear expression of the objectives underlying NRC regulatory actions. However, safety goals are not requirements. They are, instead, indications of "how safe is safe enough" that should be used to guide licensing, inspection, or enforcement actions.

There are currently two *safety* goals that have been approved by the Commission and both relate to operation of nuclear power plants. These goals are defined by limiting the risk arising from the operation of power plants for two risk measures, early (or prompt) fatalities and latent cancers. The goals are stated in both qualitative and quantitative terms. There are no other formal NRC safety goals that have been explicitly stated in risk language.

In addition to safety goals, the NRC has also adopted *strategic* goals. In the Appendix to its recently published Strategic Plan for the Fiscal Years 2000-2005 (NUREG-1614), the NRC has proposed the following Strategic Goal for Nuclear Waste Safety that includes activities associated with the decommissioning of nuclear reactors:

- Prevent significant adverse impacts from radioactive waste to the current and future public health and safety and the environment, and promote the common defense and security.

As a corollary, the following measures are proposed to assess results in achieving this strategic goal:

- No deaths resulting from acute radiation exposures from radioactive waste,
- No events resulting in significant radiation exposures[1] from radioactive waste,

[1] "Significant radiation exposures" are defined as those that result in unintended permanent functional damage to an organ or physiological system as determined by a physician in accordance with Abnormal Occurrence Criterion I.A.3.

- No releases of radioactive waste causing an adverse impact[2] on the environment.

Allied to the above strategic goal for nuclear waste safety, there is a performance goal: Maintain safety, protection of the environment, and the common defense and security. Four measures are defined to assess progress towards achieving the performance goal. They include: no events resulting in radiation overexposures from radioactive waste exceeding regulatory limits (defined in 10 CFR 20.2203 (a)(2)); no breakdown of physical protection resulting in a vulnerability to sabotage, theft, etc.; no radioactive releases to the environment from operations exceeding regulatory limits (defined in 10 CFR 20.2203 (a)(3)); no instances where waste cannot be handled, transported, stored or disposed of safely now or in the future. These measures represent more conservative thresholds than the ones used for the strategic goal since they utilize specific limits and standards.

In addition to the safety and strategic and performance goals, there are a number of radiation dose limits, contamination limits and related quantities found in the regulations pertaining to site decommissioning that have risk implications. For example, the dose limit of 25 mrem per year to an average member of the critical group for unrestricted release of a decommissioned site can be stated equivalently in terms of a health risk limit to the average member of the critical group. Using the BEIR-V risk factor, this would imply a risk limit of 1.25E-5 latent cancers per year. A comprehensive list of dose limits and target populations established by the NRC, EPA, and other organizations for accomplishing various objectives is provided in Attachment 5 to SECY-99-100. Besides quantitative dose limits, there are other measures such as performance objectives and the traditional statements regarding “adequate protection”, “no undue risk”, and “safety assurance” that are stated in more qualitative terms.

However, the distinction between a goal and a regulatory limit needs to be understood. A goal is an aspiration that may or may not be reached while a limit is a quantity that must be complied with by each applicant or licensee. The various “strawmen” draft safety goals proposed below for site decommissioning reflect this distinction to some extent. They are based on a combination of the qualitative statements and quantitative limits that have been proclaimed, in various contexts, by the Commission.

6.2 Draft Safety Goals

The draft safety goals developed below pertain to two phases of the decommissioning process. The first phase is while decommissioning activities are taking place at the site, i.e., the period between shutdown and license termination. The second phase is the period after license termination when the site is released under the provisions of the license termination rule.

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[2]Releases that potentially cause an adverse impact on the environment are not currently defined. A surrogate quantity is a release that exceeds the limits for reporting abnormal occurrences given by Abnormal Occurrance criterion 1.B.1 that is usually 5000 times the quantities in Table 2 of Part 20, Appendix B.

In developing the draft safety goals, it is useful to keep in mind the approach adopted for the safety goals for nuclear power plant operation outlined in the Commission's Safety Goal Policy Statement of 1986. That approach begins with a broad goal stating that there should be no significant additional risk to individual members of the public from the operation of nuclear power plants, and that societal risks to life and health should be comparable to or less than the risks of generating electricity by other viable technologies and should not be a significant addition to the risks posed by other societal technologies. Then it proceeds to specify the risk measures, early (or prompt) fatalities and latent cancers, and define what "insignificant" means in risk terms. In doing so, the qualitative goals are transformed into quantitative health objectives (QHOs).

The quantitative health objectives that limit the risk of early fatalities and latent cancers from power plant operation are then used to develop subsidiary objectives for the figures of merit, namely core damage frequency and large early release frequency, that relate directly to accident risk at operating plants. Hence the draft safety goals for decommissioning developed below begin with broad overall goals that are aspirational in nature that are then followed by more quantitative objectives that can, in principle, be evaluated and measured or verified.

6.3 Draft Safety Goals for Decommissioning Activities

The draft safety goals that are defined during the reactor site decommissioning activities phase refer to two target populations, the public and the facility workers. A related safety goal addresses the environment. These safety goals are based partly on the NRC strategic goal stated above and partly on the NRC reactor safety goals.

6.3.1 Draft Safety Goal 1: No significant adverse impacts on public health from decommissioning activity

This goal expresses the aspiration that decommissioning activities at the reactor site from the time of permanent shutdown to license termination should pose a negligible risk to the health of the offsite public. Two related goals are proposed that provide a more focused means of addressing the broader goal by defining risk measures.

Draft Safety Goal/Risk Measure 1A: No early (offsite public) fatalities from acute radiological exposure during decommissioning activity

This draft goal/risk measure is very similar to the reactor safety goal that states that the risk of early fatality from reactor operation should be negligible. A quantitative health objective (QHO), similar to the one for reactor operation, could be constructed from this goal, for example, by stating that the risk of early fatality from decommissioning activity should be less than one-tenth of one percent of the risk of early fatality from all other causes. However, a PRA of decommissioning would be needed to assess compliance with such a QHO.

Draft Safety Goal/Risk Measure 1B: No significant radiation exposure of any member of the public during decommissioning activity.

This draft goal/risk measure is based on the risk measure stated above for the NRC strategic goal in the Appendix to NUREG-1614. However, the "significant radiation exposure" as defined above in footnote 1 refers to the abnormal occurrence criterion that specifies it in terms of unintended permanent functional damage to an organ or physiological system. One alternative is to specify the "significant radiation exposure" in terms of the latent cancer fatality risk limit of the reactor safety goal, i.e., define significant radiation exposure as one that results in an additional risk greater than one-tenth of 1 percent of the chance of latent cancer fatality from all other causes. Assessing compliance with such a QHO would also require a risk assessment of decommissioning. Another possibility is to specify "significant radiation exposure" in terms of the dose limit to individual members of the public from licensed activity as stated in 10 CFR 20.1301.

6.3.2 Draft Safety Goal 2: No significant adverse impacts on worker health from decommissioning activity.

Draft Safety Goal/Risk Measure 2A: No early worker fatalities from acute radiological exposure during decommissioning activity

This draft goal/risk measure states that the risk of early fatality to workers from acute radiological exposure should be less than 0.1 percent of worker early fatality from all other causes. If non-radiological accidents are included here, then this particular QHO could be based on limiting worker fatality during decommissioning to 0.1% of the risks of industrial fatalities in general. (However, this may lead to potential jurisdictional issues whether OSHA regulations are controlling worker risk or whether NRC goals would be controlling worker risk.)

Draft Safety Goal/Risk Measure 2B: No significant radiation exposure of workers during decommissioning activity

There are two possible alternatives for this draft goal/risk measure. The first alternative would be to examine the worker fatality rate from routine exposures in other hazardous, e.g., chemical, industries and then to set some limit or fraction, say 1 percent or 10 percent, of that as a goal for defining what is considered to be "significant radiation" exposure for decommissioning workers. In other words, this would be equivalent to limiting the latent cancer fatality risk for the population of decommissioning workers by interpreting no significant radiation exposure of workers to imply no significant additional risk for the workers, however, the comparison would be with workers employed in other hazardous material industries. Data on worker risk in other hazardous industries is potentially available from DOE sources (e.g., the chronic beryllium disease cohort) and from chemical industry sources. The other alternative would be to define "significant" in terms of the occupational exposure limits in Part 20.

It should be noted that a specific safety goal for workers has not been established heretofore by the NRC. The reactor safety goals refer only to the general public. In decommissioning, however, the workers are likely to be the population group most at risk and impacted by both

routine and accidental exposure. SECY 99-100 also indicates that the risk metrics and goals must address the safety of workers.

6.3.3 Draft Safety Goal 3: No significant impacts on the environment

This draft goal/risk measure is difficult to define in quantitative terms. However, one way to approach it is to state that the land or groundwater at the site should not be unduly harmed, e.g., land should not be "scarred", during decommissioning and site remediation. The aesthetic, recreational, social, and economic value of the environment should be respected and preserved. Another possibility is to adopt the criterion stated in the strategic goal expressed in footnote 3 above of releases to environment exceeding the limits of abnormal occurrence 1.B.1.

6.4 Draft Safety Goals for Post License Termination Phase

The draft safety goals for the post-license termination phase deal with a return of the site to pre-existing conditions, maximizing the number of sites released for unrestricted use, and any potential worker health impacts for restricted use sites. In the post-license termination phase, the only affected population for unrestricted release sites is the critical group involved with the site. The license termination rule specifies the conditions for unrestricted release in terms of a dose limit for the average member of the critical group and that limit can be considered to be a draft safety goal/risk measure for unrestricted release. However, for restricted release sites controls have to be employed that may require workers or employees for check and enforcement purposes. Safety goals may be helpful in limiting exposure of these personnel and in providing a criterion to analyze alternative control options.

6.4.1 Draft Safety Goal 1: Return Site to Pre-existing Conditions

A broad aspiration for decommissioning, from a radiological standpoint, could be to return a site to the condition it was in before commencement of the licensed activity, viz. the construction and operation of the nuclear power plant. This goal was included in a preliminary list of six regulatory alternatives for decommissioning that was published for public comment by the NRC in the Federal Register in 1993 (58 FR 33570). The goal would imply that all residual radioactivity at the site that was a result of a licensed activity would be removed and the radiological condition would revert to background.

For background to be used as a goal, it is necessary to consider methods of measuring the very low radiation levels associated with such a criterion and to establish a relationship between the scientific definition of background and its use as a regulatory concept. NUREG-1501 "Background as a Residual Radioactivity Criterion for Decommissioning" provides an in-depth discussion of the issues related to the return-to-background as a criterion for site decommissioning. There are two basic options for applying background as a criterion. One is based on the radiological dose rate produced by background and the other is based on the concentration of naturally occurring radionuclides in the environment. Both approaches,

however, have to account for the temporal and spatial variations of natural radioactivity.

The annual radiological dose (annual effective dose equivalent) from background to a person living in the U.S. varies, depending on location, from about 100 mrem to 1000 mrem with an average of 300 mrem. Roughly two-thirds (200 mrem) of the average is due to inhaled radon and its decay products, approximately 13 percent (40 mrem) is due to inhaled and ingested sources, about 9 percent (28 mrem) is due to external gamma radiation from terrestrial sources, another 9 percent (27 mrem) is due to cosmic rays, and less than 1 percent to cosmogenic radionuclides and fallout from nuclear weapons testing (NCRP, 1987). By comparison, the average dose due to medical and dental X-rays is about 39 mrem/year and nuclear medicine procedures account for another 14 mrem/year. Hence the return to background goal has to be stated in terms of exposure to an amount of radiation due to licensed activity that is indistinguishable from background, in other words, the excess exposure over background is negligible or undetectable.

In addition to spatial variations due to altitude, latitude, radon concentration, etc. the background dose at a particular location is subject to temporal variations as well from short-term weather variability (heavy snowfall can reduce the outdoor external terrestrial gamma dose rate by as much as 90 percent), sunspot activity, industrial emissions, etc. This variability has to be kept in mind in establishing a “return-to-background” goal based on the background dose rate.

Another approach is to set a goal of reducing the concentration of residual radionuclides produced by or due to licensed activities to levels that are “indistinguishable” from their naturally occurring concentrations in the vicinity of the facility. As with the earlier goal, this approach would also have to take into account the spatial and temporal variations at the site in addition to the radionuclides under scrutiny. The implementation of this approach depends on whether the radionuclides occurring as a result of licensed activity are also present in the background. If they are not, it would be more straightforward to demonstrate compliance with such a goal. If they are, however, compliance would be difficult to demonstrate.

Such a draft goal of “return-to-background” is likely to find a high level of acceptance from the general public in the vicinity of the site and from public interest groups and organizations that have expressed reservations about the impact of nuclear power. However, there are some drawbacks associated with such a goal. First, as pointed out above, is the issue of determining and verifying compliance. Detection limits, variations in background that are difficult to distinguish from contamination, and related issues could pose a problem.

A larger problem, however, is that of increasing the overall risk. In order to remove all residual radioactivity from a site, large amounts of soil may have to be removed and transported to another location such as a licensed disposal site for low-level waste. This could lead to a greater number of transportation accidents with the concomitant risk of spreading the radioactive material along the transport route. Extensive soil removal could also leave the facility in an environmentally scarred condition similar to that produced, for example, by strip-mining of coal deposits. The cost of remediation could also increase significantly and exceed

the cost-benefit ratio of $2000 per (avoided) person-rem used by the NRC for ALARA determination and to justify safety enhancements.

6.4.2 Draft Safety Goal 2 - Return Site for Unrestricted Use

This safety goal is already embodied in the license termination rule. The regulatory limit for unrestricted release of 25 mrem per year dose to the average member of the critical group involved with the site and a means for achieving compliance with this limit have been established by the agency. The arguments presented below for maximizing the number of sites for unrestricted release also apply to this goal.

6.4.3 Draft Safety Goal 3 - Maximize number of sites for unrestricted release

Another safety goal that could be adopted would be to maximize the number of sites that could be released without any restrictions. This goal would be beneficial to the public in returning the largest amount of land for future public use. It would be straightforward to estimate and to communicate to the public.

However, to obtain the benefits of this goal, it may be necessary to adopt a more risk-informed approach to the current regulation governing unrestricted release. Under the License Termination Rule, the criterion for releasing a site for unrestricted use is that the dose to an average member of a critical group is less than 25 mrem per year. The concept of a critical group, however, as it is applied in the regulations is based on the notion of a resident farmer, i.e., a particular use of the site for the next 1000 years. While this hypothetical method of applying the rule has the advantage of being conservative with respect to dose estimation, it may not constitute a risk-informed approach. Sites that are unsuitable, in principle, for farming may be forced to comply with unrealistically low DCGL values and thus be forced to either expend additional resources for further decontamination or go for restricted use.

A more risk-informed approach that would improve the chances of achieving the stated goal would be to consider a range of possible future site uses as part of the site decommissioning process. A broad list of possible future site options could be established on a generic basis. These options could include, for example, such uses for the site as a public park, an industrial park, a commercial complex, a residential development, etc. However, the application of the generic list in a specific case would necessarily take into account special features of that site. A critical group could be defined for each option and associated exposure pathways established, as is done for the resident farmer scenario, leading to a set of DCGL values that are pertinent to that option. Each option could have a scenario probability of future occurrence attached to it that would, in essence, vary depending on site characteristics. The resident farmer scenario would, of course, be one possible option with some future probability of occurrence. These future options could then be evaluated and the assessment of the criterion for unrestricted use would incorporate the likelihood of future site use. Consistent application of this approach is likely to increase the number of sites suitable for unrestricted release.

6.4.4 Draft Safety Goal 4 - Worker health impact for restricted use sites

This goal would minimize the impact on worker health for any group of workers that are involved with a site released under restricted conditions. The draft safety goal/risk measure could be stated in terms similar to that of section 6.3.2.

The draft goals stated above are summarized in the Table 6 below.

Table 6: Draft Safety Goals During Decommissioning and After License Termination

Target Population	Decommissioning Activities	Post License Termination
Public	No significant adverse impact on public health • No early fatality • No significant radiation exposure	Residual risk to long-term health of public at or near site should be insignificant
Worker	No significant adverse impact on worker health • No early fatality (including non-radiological accidents) • No significant radiation exposure	Insignificant impact on health of any worker involved with site released under restricted use condition
Environment	• No scarring of land by remediation • Minimize impact on groundwater and land	Maximize number of sites for unrestricted use

7. PRELIMINARY REVIEW OF DRAFT QUESTIONS

7.1 Screening Criteria Analysis/Risk Analysis Questions

1. What risk information is currently available in this area? (Have any specific risk studies been done?)

Site decommissioning of a nuclear power plant site is categorized in SECY-99-100 as a Group 1 activity, i.e., "activities that involve long-term commitment of a site or facility to the presence of nuclear material at a planned, acceptable level". However, decommissioning also involves removal of radioactive material from the site as well as onsite storage of nuclear material in engineered systems. The latter activities are categorized under Group 2 in SECY-99-100 defined as "activities that involve use of engineered casks to isolate nuclear material under a variety of normal and off-normal conditions (e.g. transportation and storage)". The focus of the present case study is on the License Termination Rule (see Final Rule, Federal Register Vol. 62, no. 139, July 21, 1997) which provides the radiological criteria for license termination. Additional and related information is contained in the Final Rule for Decommissioning of Nuclear Power Plants (Federal Register Vol. 61, No. 146, July 29, 1996).

Most of the available risk information is directed at the Group 2 category. Various studies in this regard addressing transportation risk are identified in SECY-99-100, viz. NUREG-0170 and NUREG/CR-4829 and the update to the modal study recently published as NUREG/CR-6672. There is also a specific reference in SECY-99-100 to PA methodology in the NRC staff's approval of the removal and transport of the Trojan reactor pressure vessel from the site to a disposal area in Hanford, WA. This activity is being analyzed in a separate case study.

In the Trojan LTP, the licensee elected to use the generic screening approach in their assessment of the derived concentration guideline levels that would lead to doses satisfying the criteria for unrestricted site release. Hence there is no risk study that has been identified for the Trojan decommissioning process. In the GEIS [35], dose rates are translated into mortality rates, which are compared with mortality from nonradiological accidents. This is done for a parametric range of residual dose rates from 1 to 100 mrem/yr. and for various activities and facilities. This analysis provides a risk perspective for various alternatives and thus risk-informs the decision process.

A preliminary review of the risk literature indicates the following. In the Proceedings of PSAM5 Conference, held in Osaka, Japan in 2000, there is nothing apparent. In the Proceedings of PSAM4 [39] there is a series of papers on risk assessment and decision analysis of environmental remediation alternatives for a hazardous waste site at a DOE facility. These papers illustrate how various stakeholders are involved in an integrated process of risk assessment and decision-making.

There was also a paper at PSAM4 [40] on importance measures for a repository; also see [41]. These papers modified and adopted the methods of importance measures for systems reliability to the special characteristics of the repository and similar passive systems. These authors recognize several distinguishing characteristics of the repository (which are generally held in common with a decommissioned site) relative to an operating reactor that has active systems. These are: continuous vs. discrete behavior; passive vs. active systems; physically dispersed vs. compact; continuous doses vs. abrupt release; consequences evolve over long time periods (comparable to mission time) vs. short time periods. These are important characteristics that impact the style and techniques of the risk assessment.

In the Proceedings of PSA99 there are no papers related to risk assessment for decommissioning. In PSA96 there is a paper by Thompson, et. al. [42] on risk analysis for decommissioning of a uranium recovery facility. This paper provides a good example of how risk concepts can be used to gain perspective on three alternatives for the site in question: in situ reclamation, relocation of contaminants, do nothing. Radiological risks to the public and workers were assessed for the three options. Also, nonradiological risks related to construction and transportation accidents were presented based on actuarial data. Upper and lower bound total risks were then presented and compared for the three alternatives. The study predicted that the in situ option was of clearly lower risk than the relocation option and that the do nothing option sat between them.

Reference [34] outlines a method for assessing decommissioning risk (performance assessment) and provides a hypothetical example of its use in the decision-making process.

Members (approximately 10) of the international risk assessment community were contacted by email and asked to identify any risk analysis studies that have been performed for any aspect of decommissioning. Most responders were not aware of such studies. [We discounted spent fuel risk studies since they are out of scope for this case study and because we already know that such studies exist]. One responder noted that there is a long history of risk studies dating back to the mid-1970's concerned with decommissioning risk. Some of these studies look at radiation exposures during normal operation and then calculate risks to the individual (e.g. worker) or population dose.

2. What is the quality of the study? (Is it of sufficient quality to support decision-making?)

The studies mentioned in response to Question 1 are of sufficient quality to inform some decisions. Much of the radiological risk analysis tends to be generic and approximate estimates are made of doses resulting for expected levels of residual contamination. These doses are then converted to expected fatalities using a standard algorithm (e.g. BEIR V). The suitability of this approach will depend on the end use. For relative comparisons, the uncertainties or inaccuracies may tend to not be significant. However, if an absolute measure (e.g. a criterion) is required or if different sources of risk are being numerically aggregated then some precision will be needed in the risk predictions. If the assumptions of the study and its supporting analysis are clearly stated to the decision-maker, then the decisions could be tempered by this

knowledge.

3. What additional studies would be needed to support decision-making and at what cost?

Absent a full review of the underlying analysis of the studies discussed above, it is difficult to attest to the robustness of the results. Areas for possible improvement of the studies would be completeness of scenarios evaluated and assessment of uncertainties. The role of naturally occurring phenomena over the 1000-year period specified in the LTR does not appear to be addressed. Uncertainties, while clearly wide, should be addressed quantitatively.

Studies of the episodic and chronic risks in Phase 3 decommissioning may be useful in elaborating risk acceptance criteria and prioritizing decision-making related to specified activities and their cost/benefit. A letter by the NRC Advisory Committee on Nuclear Waste (ACNW) dated 11/17/99 suggests a three-region approach to risk acceptance where there is a lower risk bound below which no further action is required, an upper level above which definitive action is needed to control risk, and an intermediate region in which cost-benefit trade-offs can be made. The ACNW letter recommends that the staff investigate these and related concepts for materials regulation.

4. How is/was risk information used and considered by NRC and licensee in this area?

The licensee made a business risk decision by electing not to perform a site-specific dose (and therefore residual risk) analysis. The NRC staff regards the LTR to be (implicitly) risk-informed and therefore a risk-informed framework guides the process for review.

While not in the scope of this case study, the vessel removal activity is regarded to be risk-informed.

5. What is the societal benefit of this regulated activity?

Generally, this activity is designed to protect the health and safety of the public and to ensure that the environmental impact from licensed operation is minimized. In principle, it returns land for unrestricted public use by future generations.

6. What is the public perception/acceptance of risk in this area?

At the NRC stakeholders meeting on February 9, 2001 on the *Use of Risk Information in the Nuclear Materials Regulatory Process,* there was a clear concern for the decommissioning

area. Public comments that were provided in the Final License Termination Rule indicate a strong interest in this area by various stakeholders. During the Trojan site visit, it was noted by the licensee that there was significant public opposition by intervenor groups while the plant was in operation. There was also organized public opposition in the 1994 time frame when the steam generators and pressurizer were being removed and readied for shipment off-site via the Columbia River to the disposal facility in Hanford, WA. However, since the successful and uneventful completion of that activity, there has been comparatively little interest in site decommissioning. For example, there was relatively little interest in the reactor vessel and internals removal project by the same method of shipment up the Columbia River despite the much higher amounts of radioactivity involved. In part, this may be due to the fact that the Trojan decommissioning activity has been very straightforward and is likely to result in very low levels of residual contamination remaining at the site. For other sites, however, this may not be the case.

7. What was the outcome when this application was put through the draft screening criteria? Did this application pass any of the screening criteria? Does the outcome seem reasonable? Why and why not?

As shown in Section 4, this application passed the draft screening criteria.

7.2 Safety Goal Analysis Questions

1. What is the basis for the current regulations in this area (e.g. legislative requirements, international compatibility, historical events, public confidence, undetermined, etc.)?

The Atomic Energy Act requires that NRC have statutory responsibility for protecting health, safety, and the environment related to the possession and use of source, byproduct, and special nuclear material. The NRC includes within this mandate decommissioning of nuclear facilities that it has licensed for operation. The Commission provides guidance to licensees on how to plan for and prepare their sites for decommissioning. In 1988, NRC amended its regulations by defining “General Requirements for Decommissioning of Nuclear Facilities”[43]. These amendments required licensees who had ceased licensed activities to decommission their facilities so that their licenses could be terminated and the property released for unrestricted use. While the 1988 amendments provide guidance and criteria that could be used on a case-by-case basis for many and technologically diverse facilities that NRC regulates, it became evident that codified radiological criteria would provide a more consistent and effective approach to protecting public health and the environment at decommissioned sites.

The arguments put forth in Reference [5] are that codified radiological criteria would provide for:

-more efficient use of licensee and NRC resources;

-consistent application across all types of licensees;

-a predictable basis for decommissioning planning;

-more timeliness in the decommissioning process

-incorporation of knowledge from decommissioning experience and radiation protection standards

With the foregoing as a charter, the Commission develops the current License Termination Rule [5].

Public comments have been provided to NRC at various stages of the development of the License Termination and of the Generic Environmental Impact Statement. These are characterized in those documents.

2. Are there any explicit safety goals or implicit safety goals embedded in the regulations, statements of consideration, or other documents (an example would be the acceptance of a regulatory exemption based in part on a risk analysis and the outcome)?

There are implicit goals that are contained in various documents. For example in the GEIS, it is stated that *regulations require licensees to decommission their facilities so that property can be released for unrestricted use.* This can be regarded as a top-level safety (and environmental) goal. It expresses the notion that there are no residual hazardous conditions remaining on the site that would preclude its future use for any purpose. There is also a discussion in the GEIS specifically of returning a site to preexisting background radiation conditions. The GEIS recognizes that there are some facilities for which this goal could be achieved. However, the GEIS also points out that there are some sites (e.g. with contaminated soils and/or structures) for which its is not reasonable to, in terms of costs vs. impacts, to achieve the preexisting background conditions. For those sites, it is argued, there are competing risks associated with doses that would be received during remediation activities and with nonradiological hazards that would weigh against the benefits to be achieved for dose reductions near the background level.

The License Termination Rule puts forth radiological criteria for unrestricted release and for restricted release. It is interesting to compare the radiological risk that is deemed acceptable by the Safety Goals for nuclear power plant operation with that implied by the unrestricted release criterion in the license termination rule. The safety goals for reactor operation are structured as a three-tier paradigm. The highest tier contains two qualitative statements of the level of safety to be aspired to for nuclear power reactors. The first goal states that no individual should bear a significant additional risk to life and health from nuclear power plant operation. The second goal states that nuclear power plant operation risks should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks. Thus the goals are stated in a relativist fashion.

From these goals, the second tier of the hierarchy is given as two quantitative health objectives. These objectives are expressed in terms of potential health risks to the public through two risk metrics, prompt (or early) fatalities and latent cancer fatalities from reactor accidents. For each risk metric, the goal is expressed by stating that the risk from reactor operation should be a negligible fraction (less than 0.1 percent or one-thousandth) of the corresponding risk faced by an average person in the U.S. For prompt fatality, the comparison is with the risk to an individual of other fatal accidents such as automobile accidents, drowning, electrocution, etc. For latent cancer fatality, the comparison is with the societal risk of getting spontaneous cancer.

In comparison to reactor operation, the level of radiation exposure for a decommissioned reactor allowed by the license termination rule is several orders of magnitude below the threshold needed to cause a prompt fatality. Thus only the latent cancer fatality safety goal is relevant. The current numerical value of the latent cancer fatality safety goal, based on one-tenth of 1% of the 1 in 500 chance for the U.S. population of dying annually from cancer, is 2×10^{-6} latent cancers per year. One can compute an equivalent annual dose, by using the BEIR-V dose-risk factor (5×10^{-4} latent cancers per rem), implied by the reactor quantitative health objective. This translates into an expected value of an exposure limit of 4 mrem per year. This is approximately 6 times lower than the criterion for unrestricted site release of 25 mrem per year.

However, it should be recalled that the safety goals for operation are goals to limit the occurrence of low-probability, high consequence accidents. The dose limits for license termination, on the other hand, apply to a situation where the exposure will occur essentially with unit probability for the average member of the critical group. In other words, the relative stringency of the exposure limit implied by the reactor safety goal is imposed by the need to strictly limit the probability of reactor accidents during operation in the context of the realization that such accidents could have large consequences. The dose criterion for unrestricted release, however, is less than 10 percent of background radiation exposure and is one-fourth of the allowable dose limit to public individuals from licensed operation. The additional lifetime risk imposed by an exposure of 25 mrem per year has to be balanced by the risk of site decontamination and waste disposal activities. It is due to these factors that 25 mrem per year is considered an acceptable criterion by the Commission. As indicated in the Statement of Consideration for the License Termination Rule in FR 62, 39058, the additional risk imposed by the 25 mrem/yr exposure is low. Thus a high level decommissioning safety goal of maximizing the number of unrestricted release sites based on a latent cancer risk corresponding to the 25 mrem per year dose criterion would not necessarily be in conflict with the latent cancer safety goal for reactor operation.

SECY 99-100 discusses safety goals for nuclear materials use and disposal in the context of risk management. It distinguishes between risks arising from normal operations with low-level exposure of workers and the public from risks arising from accident conditions. For normal operations (for both materials use/disposal and reactor power operations), goals and metrics have been established and adapted from standards-setting organizations such as the International Commission on Radiation Protection and the National Committee on Radiation Protection and Measurement. For accidents and off-normal conditions, SECY 99-100 compares and contrasts the characteristics and situations for power reactor operations and materials use/disposal. Notable is the diverse nature of the latter facilities and operations as well as their low-consequence and high probability feature (in comparison to reactor risks).

The SECY paper suggests that some risk metrics that might be worth considering for materials use/disposal are: 1) overall risk of individual fatality for the population at risk from the particular materials application; 2) frequency of large exposure (e.g. > 25 rem/yr.); 3) maximum possible dose (a dose cap) for a particular material application; 4) the probability of criticality event. With regard to item 1), the SECY paper notes that risk to workers is the principal aspect of risk for most materials applications and that, unlike for reactor safety goals, materials applications *would* address worker protection from accidents.

3. What was the basis for the development of the strategic goals, performance goals, measures and metrics? How are they relevant/applicable to the area being studied and how do they relate/compare with the regulatory requirements? How would they relate to safety goals in this area?

The older NRC approach to decommissioning of power reactors as exemplified by Regulatory Guide 1.86 "Termination of Operating Licenses for Nuclear Reactors", June 1974 [27], was a deterministic approach. The criterion for unrestricted release was acceptable surface contamination levels for a variety of radionuclides including isotopes of uranium, thorium, transuranics, Sr-90, and various beta-gamma emitters. Limits of the average, maximum, and removable levels of these radionuclides in amounts of dpm/100 cm^2 were prescribed and licensees had to demonstrate that all contamination at the site was within these limits. The current approach as outlined above is based on dose criteria. An acceptable dose level of 25 mrem per year TEDE to the average member of the critical group is defined in the regulations as the criterion for unrestricted release of a reactor site that is undergoing decommissioning. Inasmuch as radiation dose translates directly into a health risk, albeit a small additional risk of latent cancer at these low levels of dose, the current regulation can be thought of as incorporating elements of a risk-informed approach in the decision process related to site release.

4. Are there any safety goals, limits, or other criteria implied by decisions or evaluations that have been made that are relevant to this area?

The decision not to do the drop test based on the criteria in Part 71 for the transport of the reactor vessel and internals seems relevant. Additionally, an accident of probability of one in a million was considered “incredible” by the Commission in approving the barge shipment of the Trojan reactor vessel. (See the accompanying case study an the Trojan reactor vessel shipment).

5. If safety goals were to be developed in this area, would tools/data be available for measurement?

It would make sense to develop goals that can be related to measurable or verifiable quantities. For example, a high-level safety goal could be established of maximizing the number of sites for unrestricted release based on the current criterion in the license termination rule of 25 mrem TEDE per year to the average member of the critical group. There are current tools, such as NRC's DandD code, and data such as the derived concentration guideline levels, the final radiation survey, etc. that are in use and could be employed to evaluate compliance with such a high-level safety goal. A larger set of future site uses in addition to the building occupancy and resident farmer scenarios employed in the current codes for dose evaluation could improve the options for more sites qualifying for unrestricted release.

6. Who are/were the populations at risk?

This is tied to the critical group concept for the long-term risks. Industrial hazards that the workers are exposed to while the site is being prepared for release could be important. Long-term (1000 year) risk and short-term (on time scale of decommissioning activities) risk are relevant discussion points. Also, the Trojan vessel transport activity illustrates how an alternate approach can lead to much less radiological exposure and smaller degree of non-radiological hazards along with a significant overall cost savings.

7. What are/were, and what could be/have been, the various consequences to the populations at risk?

For workers it would be industrial-related injuries (possibly death) and cancers induced by radiation -- large doses are not expected, thus radiation sickness would not be a consequence. For public groups it could be cancer resulting from exposures not otherwise expected.

8. What parameters should be considered for the safety goals (e.g. workers vs. public, individual vs. societal, accidents vs. normal operations, acute vs. latent fatality or serious injury, environmental and property damage)?

The metrics noted in this question are possibilities for our future consideration. For some of this we would have to step beyond just the NRC regulatory space, e.g. OSHA.

9. On the basis of the answers to the questions above, would it be feasible to develop safety goals in this regulatory area?

Yes, it is feasible and some examples of draft safety goals that could be used to guide regulatory oversight of this area are provided in section 6.3 of this report.

10. What methods, data results, safety goals, or regulatory requirements would be necessary to make it possible to risk-inform similar cases?

Generally, the same approach that has been outlined here can be used for other decommissioning candidates.

7.3 Questions Upon Developing Draft Safety Goals

Responses to these questions are based on the draft safety goals that are presented in Section 6 of this report.

1. Are the current regulations sufficient in that they reflect the objectives of the draft goals? Would major changes be required?

The License Termination Rule provides adequate protection to the public. The residual dose limit is sufficient to reflect the draft goal given in Section 6.3.2. Draft goal 6.4.3, maximize number of sites for unrestricted release, may force some sites that would otherwise propose restricted release to strive for additional remediation. As discussed elsewhere in this report, a risk-informed analysis could help provide a more realistic assessment of sites that may be candidates for unrestricted release.

Draft goals related to the workers, Sections 6.3.2 and 6.4.4, do not currently have counterparts in the NRC regulations. The exposures covered under 10 CFR 20 are regulatory limits for determining compliance, not goals. Other-agency jurisdiction for non-radiological accident conditions would need to be examined for consistency with these draft goals.

The environmental goals presented in Sections 6.3.3 and 6.4.3 are new and tentative. As such, they have no direct touchstones with existing regulations. It is worth noting that potential safety goals related to land contamination have been discussed for reactor accidents for some time now. In the Staff Requirements Memorandum (June 27, 2000) on SECY-00-0077, the Commission disapproved the NRC staff's recommendation that the Safety Goal Policy Statement include "there be no adverse impact on the environment".

Protection of the public during decommissioning activities would likely be assured under the regulations for reactor operations. However, the latter may be too burdensome for some the activities that take place during decommissioning.

2. Would the regulations need to be tightened?

As discussed above, the License Termination Rule provides adequate protection within its scope. However, if the suggestions for worker protection and environmental protection were to be adopted, then it is possible that an associated regulatory framework would need to be developed.

3. Are the regulations overly conservative and/or too prescriptive with respect to the goals?

No, the License Termination Rule is consistent with the corresponding goal. Furthermore, since

it is dose-based and does not set concentration limits, it is not prescriptive.

4. If these were the safety goals, what decisions would be made?

This may become more clear as additional applications are made for decommissioning. A key issues could become whether a site would be released under restricted or unrestricted conditions. The draft goals may also provide general guidance for those sites that opt for alternative criteria under the License Termination Rule.

5. Would these goals be acceptable to the public?

This was an important topic for the Stakeholders Meeting to be held on October 25, 2001. At the meeting, there was general agreement by the stakeholders that NRC should proceed with development of safety goals in the nuclear materials use and waste area.

8. PRELIMINARY RESPONSES TO THE DRAFT SCREENING CRITERIA QUESTIONS

In Section 4 of the Case Studies Plan [1], draft screening criteria are provided. The intent of these criteria is to identify candidate regulatory applications that are amenable to expanded use of risk assessment information. These questions are presented here along with our preliminary responses for the case study.

8.1 Would a risk-informed regulatory approach help to resolve a question with respect to maintaining or improving the activity's safety?

Yes. It has been demonstrated in the GEIS and the License Termination Rule that a risk perspective is helpful in decision-making for decommissioning. For example, the weighing of competing risks related to allowing residual dose level vs. soil remediation with radiological and nonradiological hazards is helpful to the decision process.

8.2 Could a risk-informed regulatory approach improve the efficiency or effectiveness of the NRC regulatory process?

Yes. It could help focus the regulatory process on areas of highest safety concern. Resources could be applied where they would be most effective in performing a review of the License Termination Plan. In the case of the Trojan application, the review by the NRC was straightforward. The staff verified that the licensee performed the appropriate analysis to support the goal of unrestricted release. However, Trojan is a site with low levels of residual contamination from licensed operation and the same may not be the case at other sites.

8.3 Could a risk-informed approach reduce unnecessary regulatory burden for the applicant or licensee?

Yes. The licensee or applicant may want to examine the criteria associated with restricted and unrestricted release and evaluate the risks associated with alternatives on a site-specific basis to determine the most effective approach, consistent with protecting health, safety, and the environment. In the case of Trojan, the licensee made a business risk decision by opting to use the screening approach in their License Termination Plan because they were reasonably within the criteria for unrestricted site release.

8.4 Would a risk-informed approach help to effectively communicate a regulatory decision or situation?

Yes. It is helpful to support a decision or relate information on a situation by providing as broad a perspective as possible. Understanding the risks and their trade-offs can only enhance the perspective. It is also important to communicate benefits as well as the risks of a given

situation or decision.

8.5 Does information (data) and analytical models exist that are of sufficient quality or could they be reasonably developed to support risk-informing a regulatory activity?

Yes. One of the strengths of the risk-informed approach is that allows the analyst to quantify what he/she knows about what he/she does not know. Central to the risk-informed approach is the concept of uncertainty. Expressing and understanding these uncertainties allows the analyst and decision-maker to assess the risks for the activity in question. There are models for working with uncertainties that have been useful in other context and they should also be useful for assessing the risk related to decommissioning. Physical models for dose assessment already exist and they have the capability for expressing parameter uncertainties (referred to in this literature as probabilities). As [34] indicates, performance assessment, or PA, is the overall methodology of choice for decommissioning risk assessment.

8.6 Can startup and implementation of a risk-informed approach be realized at a reasonable cost to the NRC, applicant or licensee, and/or public, and provide a net benefit?

Yes. It is not necessary to totally revamp the overall process at once. An evolutionary process is more appropriate and likely to be warranted.

8.7 Do other factors exist (e.g., legislative, judicial, adverse stakeholder reaction) which would preclude changing the regulatory approach in an area, and therefore, limit the utility of implementing a risk-informed approach?

There are no factors apparent to us at this time.

9. SUMMARY AND CONCLUSIONS

This Technical Report presents the results of this case study on the site decommissioning. The report also provides responses to the Draft Questions that are presented in the Case Study Plan. These responses are subject to revisions as more information is obtained in documents and from interactions with interested parties. Responses to the Draft Screening Criteria are given in this report. The site decommissioning area passed the screening evaluation and thus can be regarded as a candidate for risk-informed regulation.

The review of the Trojan Nuclear Plant License Termination Plan and the NRC's review of this material against the License Termination Rule revealed that the regulatory evaluation commenced in prescriptive and deterministic manner. However review of the foundations for the rule indicated that it was developed in a risk-informed manner and that it allowed for a performance-based option for complying with the rule.

Options for complying with the Rule include deciding on whether to apply for unrestricted or restricted release of a site. It appears that there is opportunity to develop risk-informed studies to support particular cases for meeting the intent of the regulations. In the case of Trojan, this was not warranted because the licensee was able to use prescriptive screening criteria to demonstrate compliance. This was the most cost-effective way for the licensee to proceed.

Our responses to the Draft questions revealed that there is a reasonable amount of risk-related literature for site decommissioning. However, there is very little that has been done for site decommissioning, per se. It is our impression that further studies can be performed, without any apparent technical obstacles, if the industry, the NRC, or the stakeholders have an interest in such studies. Some technical arguments can be made (e.g. Trojan) without the benefit of a risk study. Stakeholders will weigh the costs of performing risk studies against the perceived benefits to be derived from these studies.

There are no explicit safety goals for the site decommissioning area, but implicit goals can be discerned for the existing regulatory fabric. The most overarching goal (below not doing undue harm to the public and workers) is that "property can be released for unrestricted use". Metrics are provided for meeting the criterion of unrestricted use in terms of an acceptable residual dose above that implied by the background radiation level. Protection of the workers while preparations are made for long-term site release needs additional attention in this case study. Documentation by the NRC revealed some strawman possibilities for safety goals, which were also given consideration.

Draft safety goals have been developed in this report for site decommissioning. They cover public and worker health and safety and environmental protection. They have been developed for decommissioning activities as well as long-term license termination.

With regard to the objectives of the Case Study Plan, we have concluded the following.

1. This case study does provide useful information for the development of a final version of the screening criteria. The draft screening crtieria have been useful for this study and apart from some change of emphasis in the wording, the essential thrust of the draft screening criteria will remain.

2. As suggested in this study, risk information would be useful in developing more realistic long-term scenarios which could make a difference with regard to restricted vs. unrestricted release of a site.

3. This study considered the feasibility of safety goals and has, in fact, developed draft goals.

4. This study has begun the process of identifying suitable methods for developing a risk-informed approach. It show how performance assessment could be used and extended to develop the appropriate analytical tools.

With regard to the broad objectives of the NRC, we have given consideration to how risk-informing the site decommissioning area might be a worthwhile enterprise. With regard to improving safety, it has been demonstrated in the GEIS and the License Termination Rule that a risk perspective is helpful in decision-making for decommissioning. For example, the weighing of competing risks related to allowing residual dose level vs. soil remediation with radiological and nonradiological hazards is helpful to the decision process. With regard to regulatory effectiveness and efficiency, a risk-informed regulatory approach could help focus the regulatory process on areas of highest safety concern. Resources could be applied where they would be most effective in performing a review of the License Termination Plan. With regard to reducing unnecessary burden, the licensee or applicant may want to examine the criteria associated with restricted and unrestricted release and evaluate the risks associated with alternatives on a site-specific basis to determine the most effective approach, consistent with protecting health, safety, and the environment. Finally, with regard to effectively communicating a regulatory decision or situation, it is helpful to support a decision or relate information on a situation by providing as broad a perspective as possible. Understanding the risks and their trade-offs can only enhance the perspective. It is also important to communicate benefits as well as the risks of a given situation or decision.

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