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July 30, 1991
Fort St. Vrain
Unit No. 1
P-91248

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

ATTN: Dr. Seymour H. Weiss, Director
Non-Power Reactor, Decommissioning and
Environmental Project Directorate

Docket No. 50-267

SUBJECT: RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION - DECOMMISSIONING TECHNICAL SPECIFICATIONS AND REPRESENTATIVE COST ESTIMATE

REFERENCES: 1. NRC Letter, Erickson to Crawford, dated June 7, 1991 (G-91121)

2. PSC Letter, Crawford to Weiss, dated June 6, 1991 (P-91198)

Dear Mr. Weiss:

Attached is Public Service Company of Colorado's (PSC's) response to your request for additional information (Reference 1) regarding the Fort St. Vrain (FSV) proposed Decommissioning Technical Specifications and Decommissioning Representative Cost Estimate.

The attached responses discuss several changes to the proposed Decommissioning Technical Specifications. PSC will incorporate these changes and submit a revised set of Decommissioning Technical Specifications as a proposed license amendment by August 30, 1991.

As is discussed in the attachment, PSC notes that the detailed FSV Decommissioning Cost Estimate was submitted in Reference 2. The NRC's concerns regarding the representative cost estimate had previously been provided to PSC and were considered prior to submittal of the detailed cost estimate.

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If you have any questions regarding the attached information, please contact Mr. M. H. Holmes at (303) 480-6960.

Very truly yours,

A. Clegg Crawford

A. Clegg Crawford
Vice President
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ACC/SWC/lmb

Attachment

cc: Regional Administrator, Region IV

Mr. J. B. Baird
Senior Resident Inspector
Fort St. Vrain

Mr. Robert M. Quillin, Director
Radiation Control Division
Colorado Department of Health

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

The following are PSC's responses to the NRC's Request for Additional Information regarding the Proposed Decommissioning Technical Specifications and Decommissioning Representative Cost Estimate, dated June 7, 1991:

NRC Question 1: General Comment

The Technical Specifications (TS) should be expanded to cover dismantlement activities (i.e., cutting) that may be conducted in the PCRV and require the Reactor Building to maintain subatmospheric pressure. The TS should also address how the PCRV fluid level will be maintained as well as sealing the PCRV and dealing with leakage that may occur while the PCRV is flooded.

PSC Response:

This question will be addressed in two parts. The first part deals with the scope and extent of the FSV Decommissioning Technical Specifications (DTS). PSC considers that the proposed DTS submitted on December 21, 1990, include all the dismantlement activities that require the Reactor Building to maintain subatmospheric pressure, consistent with the accident analysis in Section 3.4 of the Proposed Decommissioning Plan (PDP).

The NRC requirements in 10 CFR 50.36 provide that Technical Specification requirements should be derived from the analyses and evaluations in the Safety Analysis Report. Further, ANSI/ANS 58.4 guidance indicates that Technical Specification Limiting Conditions for Operation are provided for items when they are relied upon in the Safety Analysis.

In the PDP accident analysis, Section 3.4.8 analyzes a loss of AC power during the cutting of a large activated graphite reflector block, and concludes that the loss of ventilation results in an acceptably small potential release. For all of the analyzed accidents, the proposed DTS adequately bound activities that may be conducted within the Reactor Building. The proposed DTS therefore ensure that off-site doses to the public are well below 10 CFR 100 guidelines and within a small fraction of the EPA guidelines provided in EPA-520/1-75-001-A, dated January 1990.

The proposed DTS provide requirements for the Reactor Building to be maintained at subatmospheric pressure during all activities involving activated graphite blocks. The activation level of other PCRV materials, including graphite, concrete, and various metallic items, is significantly less, as identified in the PDP activation analysis. In the event of a load drop accident or release of cutting debris involving other PCRV materials, the resultant doses are low enough that subatmospheric conditions are not relied upon.

In addition to the requirements for subatmospheric conditions and Reactor Building confinement integrity, the DTS provide Administrative Controls for a radiation protection program and for a Decommissioning Safety Review Committee. These Administrative Controls will ensure that activities are conducted in accordance with Radiation Work Permit controls, as applicable. Also, activities that could create the greatest potential for airborne contamination, such as certain cutting operations, will utilize engineered controls for radioactive containment.

PSC considers that the requirements of the proposed DTS are consistent with the safety analysis provided in the PDP, and that the Administrative Controls provide sufficient assurance of radiation protection measures, such that no expansion of the DTS scope is required.

The second part of the question deals with controls on the PCRV shielding water. Prior to the initial fill of the PCRV, all penetrations which are below the PCRV water line and have had their instrumentation removed will be sealed. Sealing will be accomplished either by welding on cover plates, by cutting and capping (with welded caps), or by installation of blind flanges. All sealing devices will be designed and tested per applicable requirements. It should also be noted that there are two independent PCRV water cleanup and clarification systems, so that repair and maintenance on one train will not affect operation of the other.

During the initial fill of the PCRV, the seals will be monitored for leakage and, if leakage is detected, they will be repaired prior to substantially increasing the level. After the PCRV has been flooded, the PCRV water cleanup and clarification system will be placed into operation. Water level in the PCRV will be monitored on the control panel for this system, located on the refueling deck.

The PCRV water cleanup and clarification system will be designed so that portions of it can be isolated with valves and drained and repaired if necessary. The system will be pressure tested prior to the introduction of contaminated water and it will be checked for leakage during operation.

PSC proposes to add a discussion in the Design Features of the DTS, addressing water leakage prevention provisions, but we do not consider that a Limiting Condition is required. The Loss of PCRV Shielding Water accident scenario postulated in Section 3.4.7 of the PDP assumes that the entire water inventory of the PCRV is released due to a pipe rupture. The dose analysis conservatively assumes that the theoretical maximum amount of tritium is transferred to the PCRV shielding water from the graphite blocks. As such, any leakage that may occur while the PCRV is flooded would be bounded by the accident analysis in Section 3.4.7 of the PDP.

PSC proposes to add the following to the DTS Design Features:

"4.3 <u>PCRV Water Leakage Prevention</u>

The PCRV will be filled with water to provide shielding for workers during initial PCRV internal dismantlement activities. To prevent leakage from the PCRV, all penetrations which are below the PCRV water line and have had their instrumentation removed are sealed. Sealing is accomplished with either welded cover plates, welded caps, or blind flanges.

There are two independent trains in the PCRV water cleanup and clarification system, to allow for maintenance and repair. Each train has sufficient valves and drains to allow isolation as required."

NRC Question 2: Page 3.0-1, Section 3.0.4

Justification for the extension period of 25% should be provided.

PSC Response:

The extension period of 25% allowed by DTS Section 3.0.4 is consistent both with current FSV requirements and with generic NRC guidance for surveillance frequencies.

Section 2.18 of the existing FSV Technical Specifications allows surveillance intervals to be extended up to 25%, as does Specification 4.0.2 of the Westinghouse Standard Technical Specifications, Revision 5. The NRC guidance provided in Generic Letter 89-14, "Line-Item Improvements in Technical Specifications - Removal of the 3.25 Limit on Extending Surveillance Intervals", also allows a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

Based on acceptable past practice and on NRC guidance, PSC considers the allowable 25% extension period appropriate for the FSV DTS.

NRC Question 3: Page 3.1-3, Background

This section states that new outer truck doors may be added. Are the new doors required to maintain Reactor Building integrity? If so, the TS should specify the use of the new doors in controlling releases of radioactivity.

PSC Response:

LC 3.1 requires that Reactor Building confinement integrity be maintained with (among other considerations) either the inner or the outer truck bay closures closed. The word "closures" is intended to apply to either the historical hatches and doors or any future redundant door installation, as described in the Bases section cited by the NRC. In this case, the new outer truck doors would be required for Reactor Building integrity when all inner doors are open.

Since these new outer truck doors are considered "closures", as defined in the Bases, they are included in the requirements of LC 3.1.b.1, for assuring Reactor Building confinement integrity, and no additional DTS requirements are needed.

NRC Question 4: Page 3.1-4, Bases

The TS state that the Reactor Building louvers may be open while activated graphite blocks are being dried or stored. PSC should provide an analysis of the potential release of tritium during the drying process.

PSC Response:

Although the Reactor Building louvers may be open while activated graphite blocks are being dried or stored, the Reactor Building internal pressure will be maintained subatmospheric whenever activated graphite blocks have been removed from the PCRV shielding water and remain inside the Reactor Building, in accordance with LC 3.2. Therefore, all gaseous effluents created as a result of decommissioning operations will pass through the Reactor Building ventilation exhaust system, as was done during normal plant operations. However, the ventilation filters have no provision for removing tritium and consequently, no credit is taken for confinement of tritium. The position of the louvers, therefore, has no effect on the amount of tritium released during the drying process.

PSC has reviewed the amount of tritium that could potentially be released during the drying process. The quantity of PCRV shielding water being evaporated from the surface of the graphite blocks is relatively small, compared to the amounts of tritiated water vapor assumed to be evaporated in the Loss of PCRV Shielding Water accident analyzed in Section 3.4.7 of the PDP. The PDP analysis assumed that tritium would be evaporated from an 848 square meter pool, and concluded that the dose to an individual 100 meters from the Reactor Building would be 34.8 mRem for a two hour period. This is a very small fraction of the 1 Rem whole body dose criteria of the EPA Protective Action Guidelines cited in the PDP. Since the quantities of tritiated water vapor released during drying operations are bounded by the PDP accident analysis, the consequences of drying operations are also bounded by the PDP accident analysis.

NRC Question 5: Page 3.1-1, Action

The completion time allowed to respond to the condition listed should be reevaluated. If the reactor building confinement integrity cannot be maintained, it is recommended that activities be suspended immediately.

PSC Response:

PSC proposes to revise the completion time to suspend activities in the event that Reactor Building confinement integrity is lost, from 12 hours to 1 hour. This completion time is consistent with that proposed in LC 3.2 for the condition where Reactor Building pressure is not subatmospheric.

The 1 hour completion time allows an orderly suspension of activities within a reasonably conservative time frame, so that further problems are not created out of actions taken in a more hurried manner. Also, a 1 hour completion time avoids the ambiguity that is inherent with "immediate" action requirements.

NRC Question 6: Page 3.2-1, Actions

The Required Action and the Completion Time listed in the table for Action A.1 is not consistent with the required Action described on page 3.2-4 for the same activity. This inconsistency must be resolved.

PSC Response:

PSC proposes to revise the A.1 Action discussion in the Bases to be consistent with the required Action table. The second sentence of the A.1 Action discussion on Page 3.2-4 will be revised to read as follows:

> The one hour completion time to <u>suspend activities involving physical handling of ACTIVATED GRAPHITE BLOCKS within the Reactor Building</u> minimizes the time exposure of the Reactor Building to atmospheric or greater conditions and is a conservative time frame (changes underlined).

This revision ensures that the Bases discussion and the Required Action Table are in agreement.

NRC Question 7: Page 3.2-5, Surveillance Requirements, SR 3.2.1

The Reactor Building subatmospheric pressure surveillance should be every 12 hours during critical activities requiring subatmospheric pressure. Action B.1 states that subatmospheric conditions can be maintained for about 12 hours.

PSC Response:

PSC agrees to revise Surveillance Requirement SR 3.2.1 to require a "Once per 12 hours" verification that the Reactor Building pressure is subatmospheric.

NRC Question 8: Page 3.3-2, Table 3.3-2

The required channel calibration frequency should be on a 6-month interval during decommissioning activities.

PSC Response:

PSC agrees to revise Table 3.3-2 to require a 6-month channel calibration frequency for the specified radiation monitors.

NRC Question 9: Page 3.4-2, Surveillance Requirements, SR 3.4.2

The surveillance frequency should be daily while water is being used for shielding.

PSC Response:

PSC considers that a requirement to sample the PCRV shield water daily while it is being used for shielding is an unnecessary burden, since each sample and analysis requires approximately four hours and the PCRV shield water system is expected to be in use for approximately one year, although the tritium concentration is expected to be very low after about 40 days.

PSC considers that daily sampling until the initial tritium level has been substantially reduced, followed by weekly sampling until tritium concentration is decreased below 0.01 microcuries/cc, is acceptable and consistent with NRC regulatory guidance, as follows:

The majority of the tritium within the PCRV is contained within the activated graphite blocks. PSC anticipates that the release of tritium into the shield water will occur within a very short time after the graphite blocks are immersed. During this initial immersion period, daily sampling is warranted to monitor the tritium level and ensure that the maximum limiting concentration is not exceeded.

Tritium levels in the PCRV shield water will be reduced by a feed and bleed operation. As shown on the attached Figure (provided in the Proposed Decommissioning Plan as Figure 3.3-1), tritium concentration is expected to peak within 10 days after flooding the PCRV, and to be substantially reduced (to less than 0.1 microcuries/cc) within 40 days after flooding the PCRV. Tritium concentration is expected to continue to decrease thereafter.

NRC regulatory guidance for tritium monitoring programs for occupational exposure is contained in Regulatory Guide 8.32, "Criteria for Establishing a Tritium Bioassay Program." This guidance provides criteria for tritium concentrations above which bioassay programs should be established, and frequencies at which routine bioassay sampling should be conducted.

The most conservative criteria in Regulatory Guide (RG) 8.32 for workers where tritiated water is in contact with the air calls for sampling if the tritium concentration exceeds 10 microcuries/cc, on a once per two weeks frequency. Even if workers can come in contact with tritiated water, the concentration limit is 0.01 microcuries/cc. Below these tritium concentrations, a routine survey program is not required by RG 8.32.

Based on the above, PSC proposes to revise Surveillance Requirements SR 3.4.1 and 3.4.2 to require daily sampling during initial filling of the PCRV with shielding water and until tritium concentration decreases below 0.1 microcuries/cc for three consecutive days. After this period (estimated to be approximately 40 days after flooding), weekly sampling will be required until tritium concentration decreases below 0.01 microcuries/cc. After this point is reached, no further sampling will be required.

This proposed revision to SR 3.4.1 and 3.4.2 is conservative with respect to RG 8.32 in that (1) sampling is required above 0.01 microcuries/cc, where RG 8.32 only requires sampling above 10 microcuries/cc for comparable applications, and (2) sampling is required on a daily or weekly basis, where RG 8.32 only requires once per two weeks or quarterly sampling.

<u>NRC Question 10:</u> Page 3.4-4, Applicability

The LC should be applicable as long as the PCRV has water in it.

<u>PSC Response:</u>

PSC agrees to revise the Applicability of LC 3.4 to "Whenever there is shielding water within the PCRV." This agreement is subject to the position taken in response to Question 9 above regarding sample frequencies.

NRC Question 11: Page 5.0-3, Administrative Controls, 5.3.7

The decommissioning audits should be performed at least once every 6 months for the activities listed in this section. These activities address significant safety areas that are critical during decommissioning and dismantlement.

PSC Response:

PSC considers that performing audits on decommissioning activities every 6 months represents an unreasonable burden on our resources.

PSC proposes to revise Administrative Controls section 5.3.7 to require that audits of decommissioning activities be performed on a one year frequency. This one year audit frequency is consistent with the current FSV Technical Specifications. FSV Administrative Control 7.1.3.c requires that the Nuclear Facility Safety Committee audit conformance of facility operation to the Technical Specifications and various other requirements at least once per year. This is also consistent with the audit requirements of the Westinghouse Standard Technical Specifications, Revision 5, Administrative Control Section 6.5.2 (NUREG-0452).

NRC Comment:

Decommissioning Representative Cost Estimate

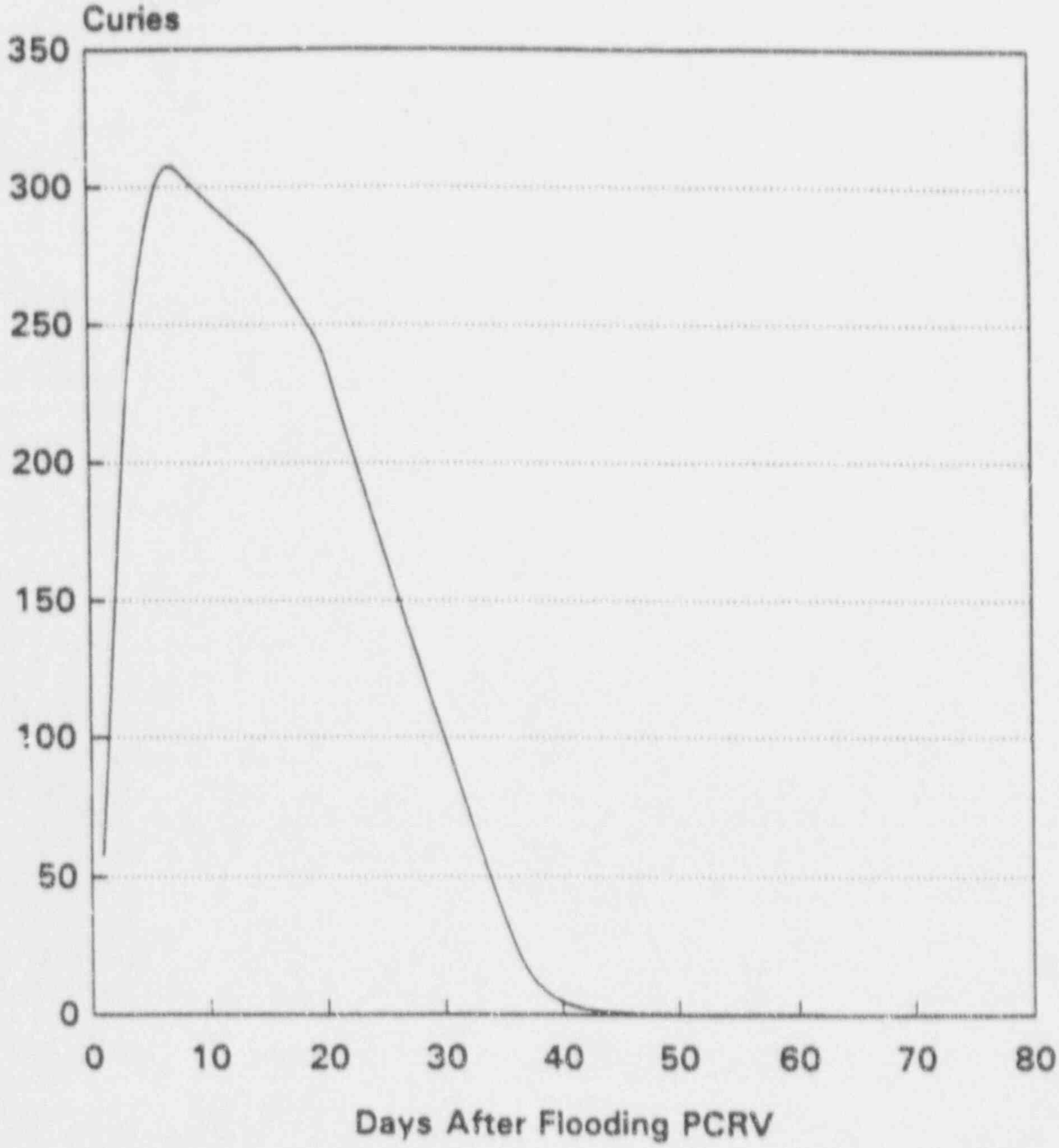
The sample cost estimate (WBS NO. 2.3.4.3) described the task to be performed, identified the estimated duration required to complete the task, estimated crew size necessary to perform the task, equipment, and supply requirements. The example also estimated volume of waste, radiation levels, and radiation exposure resulting from performing the task. The example also indicated that transportation and burial cost would be developed for each WBS although it was not included in the example provided. By providing the cost estimate for each of the identified areas, NRC's concerns should be adequately addressed. However, the example provided a description for an approach for performing the WBS and stated that if an alternative approach is used the contingency allowance would be sufficient to cover any differential cost, etc. This is not an acceptable approach. If an alternative approach is being considered, the estimate must address all the areas discussed above or identify differential cost compared to the initial approach.

PSC Response:

The detailed Fort St. Vrain Decommissioning Cost Estimate was submitted to the NRC in PSC letter, Crawford to Weiss, dated June 6, 1991 (P-91198). The NRC concerns identified above had been relayed to PSC and Westinghouse during the preparation and prior to submittal of the detailed cost estimate.

In preparing the detailed Work Breakdown Structure (WBS) Dictionary descriptions, WBS Element Descriptions and individual WBS Element Cost Estimates, alternative approaches were evaluated for technical and ALARA feasibility. However, for cost estimating purposes, if an alternative approach was considered in the WBS Dictionary and Element Description, only the highest cost option was included in the WBS Element Cost Estimate. Therefore, the total Decommissioning Cost Estimate represents a conservative upper bound on the cost of decommissioning.

Attachment to
P-91248
Page 12



ESTIMATED TRITIUM INVENTORY IN PCRV WATER SYSTEM