LICENSING EXPERIENCE FOR DECOMMISSIONING FORT ST. VRAIN

Vincent F. Likar and Dennis M. Popp
Westinghouse Electric Corporation

ABSTRACT

The Fort St. Vrain (FSV) Nuclear Generating Station is located about 35 miles north of Denver, Co. The *utility* decided in 1989 to prematurely shut down the reactor, and this paper will discuss the experience gained in achieving an NRC license for decommissioning the reactor.

REACTOR DESIGN

The Fort St. Vrain reactor is shown in Fig. 1. The reactor is a High Temperature Gas-Cooled Reactor (HTGR) rated at 330 MWe. The Prestressed Concrete Reactor Vessel (PCRV) is a structure over 105 feet high, with interior dimensions of approximately 75 feet in height and 31 feet in diameter. The top and bottom heads are about 15 feet in thickness, and the side walls are approximately 9 feet thick. The core is located in the upper part of the vessel and consists of over 5000 individual graphite blocks, including about 1500 blocks containing nuclear fuel. The fueled blocks have been removed by the utility and stored in an on-site air cooled storage vault and replaced in the core with unirradiated defueling blocks. The core sits on the Core Support Floor, which is about 29 feet in diameter, 5 feet thick and weights about 270 tons. The lower part of the vessel contains the steam generators and the helium circulators. In operation, the helium circulators forced helium around the Core Support Floor and up the exterior annulus of the Core Barrel. The helium then passed down through the core and through the steam generators, and then back through the helium circulators.

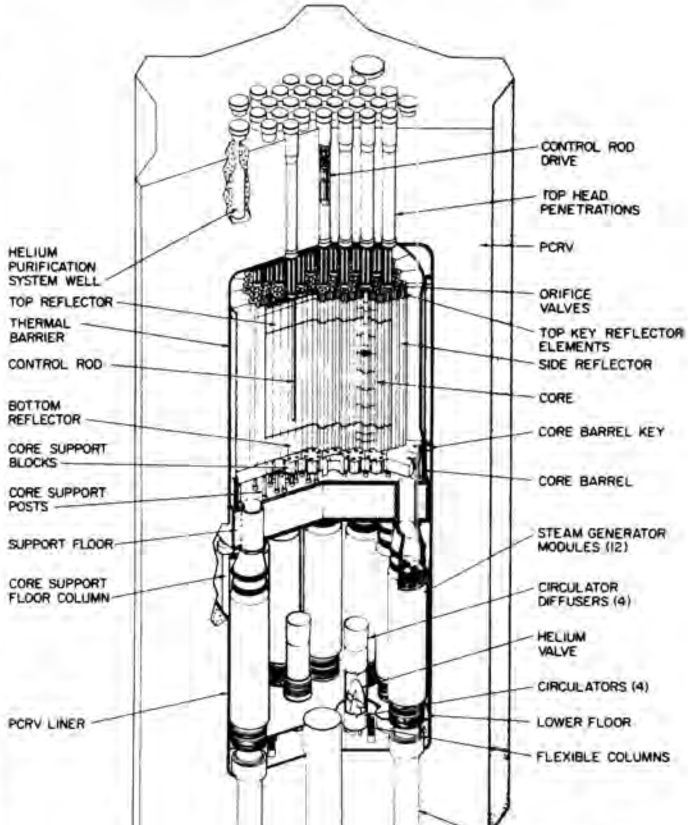

Fig. 1. Fort St. Vrain Reactor.

CONSTRUCTION AND OPERATING HISTORY

The Fort St. Vrain construction permit was issued in September, 1968. Construction was completed and the operating license issued in December, 1973. Initial criticality was achieved on January 31, 1974. The plant then went through over 5 years of start-up testing, low power operation, and required plant modifications, until the plant went into commercial operation in July, 1979. The plant operated sporadically from 1979 through 1988, with a historical capacity factor of less than 15 percent. In September, 1986, the utility signed a settlement agreement with the Public Utility Commission and the Office of Consumer Counsel of the State of Colorado over several areas of litigation and long term issues. As a result of that agreement, the nuclear assets were written down and further operation of the plant was to be based on 4.8 cents per kilowatt-hour of electricity produced. Based on these conditions, the plant operated at a $25 to $35 million dollar loss per year. As a result, the utility decided in December, 1988 to permanently cease FSV operation by June 30, 1990. However, on August 18, 1989, the plant was shutdown because of control rod drive and steam generator problems. On August 29, 1989 the utility decided not to restart the reactor.

The utility filed a Preliminary Decommissioning Plan with the NRC in June 1989 based on SAFSTOR. However, in June of 1990, because of a desire to utilize the non-nuclear portions of the plant for continued power production, the utility went out for fixed price bids for decommissioning the reactor and repowering the plant with natural gas. This request also asked for entrepreneurial bids to decommission and repower the plant as an Independent Power Producer (IPP).

The utility decided to pursue the described early dismantlement option based on the following major considerations:

- There would be continuing long term costs for the SAFSTOR Option (Maintenance, Security, Testing, etc.)
- Risks for early Dismantlement were more quantifiable (Fixed Price Bids, Regulatory Environment, Public Attitude, Waste Disposal)

Although the utility decided to pursue the DECON option, it retained the option to go to SAFSTOR.

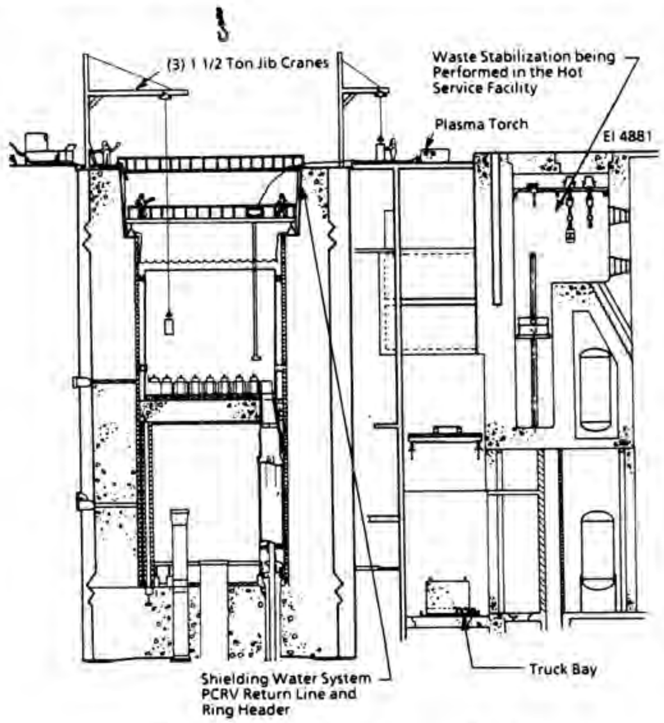

Fig. 2. Decommissioning method.

DECOMMISSIONING CONTRACT

A Consortium of Westinghouse-MK Ferguson-Black and Veach was selected for a fixed price contract to decommission and repower the plant. In July, 1990 Westinghouse and MK-Ferguson were directed to initiate decommissioning planning efforts under an existing Services Agreement between Westinghouse and the utility. Efforts on plant repowering were not initiated at this time.

The "Westinghouse Team" (Westinghouse/MK Ferguson) decommissioning method is to cut open the top of the PCRV to create a clear opening to the interior of the vessel. That method is shown pictorially in Fig. 2. The interior of the vessel will be flooded with water to act as shielding for removal of components. A work platform will be installed on the ledge at the top of the vessel opening, and components will be removed through openings in the work platform, using shielded casks as necessary for the removal and transfer. After upper components are removed, the Core Support floor will be raised out of the PCRV and positioned above the vessel where it will be cut into smaller pieces for shipment. The steam generators and helium circulators and other components in the lower portion of the vessel will then be removed. Other components and portions of the reactor will be decontaminated or removed as necessary. The Fort St. Vrain reactor design is such that the Balance of Plant components and systems have very little and very low levels of contamination. Those systems and individuals components will be dismantled or decontaminated as necessary. About 95 percent of the effort and waste volume is expected to come from the PCRV and its internal components.

PLANT DEFUELING

The removal of the fuel from the Fort St. Vrain reactor was the responsibility of the utility. The fuel is owned by the Department of Energy, which is contractually committed to take final possession and provide storage. The utility intended to begin defueling and shipment of the fuel to INEL in early 1991 to permit the start of decommissioning in January, 1992. However, Governor Andrus of Idaho objected, and a series of legal actions ensued to delay that schedule. The utility had foreseen this possibility and had moved ahead with alternate plans for construction of an Independent Spent Fuel Storage Installation (ISFSI) at the Fort St. Vrain site in February of 1991. It received the license to operate the ISFSI in November, 1991. In December, 1991 the utility elected to defuel the reactor and transfer the fuel to the ISFSI. The defueling of the reactor and movement of all fuel to the ISFSI was completed on June 15, 1992.

DECOMMISSIONING PREPARATIONS

The Team for Decommissioning the reactor, Westinghouse and MK-Ferguson, immediately upon being awarded the contract, initiated efforts with Public Service Co. of Colorado to develop and submit the required documents to obtain a decommissioning license. The first document was the Proposed Decommissioning Plan (PDP), which was written, in compliance to D.G. 1005, Format and Content Guide. The PDP consisted of the following major sections:

- Summary of Plan
- Choice of Decommissioning Alternative and Description of Activities
- Protection of Occupational and Public Health and Safety
- Final Radiation Survey Plan
- Decommissioning Fixed Price Contract and Funding Plan
- Decommissioning Technical and Environmental Specifications
- Decommissioning Quality Assurance Plan
- Decommissioning Access Control Plan
- Westinghouse Team Scope of Work
- Fort St. Vrain Activation Analyses
- Letter of Credit and Standby Trust Agreement

LICENSING DOCUMENT SUBMITTALS

The Proposed Decommissioning Plan (PDP) was submitted to the NRC on November 5, 1990. This was followed in December by submittal of the Technical Specifications for the Decommissioning. Finally, the Supplement to the Applicants' Environmental Report for Decommissioning was submitted in July of 1991. Rather than discuss the chronology and number of questions that have come up in the licensing process since these initial submittals, the following discussion will concentrate on those questions and concerns with the PDP which became major issues with the NRC.

MAJOR NRC QUESTIONS AND CONCERNS

Funding for Decommissioning

Although the utility was pursuing the DECON option for Fort St. Vrain, they also retained the SAFSTOR option. Due to funding uncertainties, they were at first not able or willing to commit to the DECON option to the NRC.

As discussed earlier, the utility initially considered entrepreneurial proposals for decommissioning the plant and repowering it with natural gas as an Independent Power

Producer. This option was later rejected. Several factors influenced that decision. One was the utility's assumption of the assets of Colorado Ute, a utility that went bankrupt. The utility, in acquiring the assets of Colorado Ute, acquired additional generation capacity, and was no longer able to justify repowering the plant in 1995 as was initially projected. It's power needs would only justify a 1998 or later repowering. In addition, the utility went back and reviewed the accounting methodology it had used in 1986 in writing down the Fort St. Vrain plant assets. That review convinced it that it had a legal basis for recovering addition revenue to cover decommissioning. The utility had a continuing discussion with the Colorado Public Utilities Commission, and was successful in reaching an agreement with all interested parties to recover $13.9 million per year for 12 years from ratepayers through its rate base. As part of this agreement, the utility agreed to institute more aggressive conservation programs, and to fund additional programs for those customers that were less able to meet their utility bills.

The formal agreement to this end was signed with the Colorado PUC on January 17, 1992. Thereafter, on January 21, 1992 the utility advised the NRC that they were committed to the DECON option for the Fort St. Vrain plant.

Estimated Cost for Decommissioning

Another major concern of the NRC involved the level of detail the PDP provided in the estimated cost for decommissioning. The PDP as originally submitted assumed that the estimated cost for the decommissioning was the fixed price of the Westinghouse Team Contract, plus the utility's costs during the decommissioning process, with some contingency for changes. The costs were broken down only into twelve major, line item entries. The utility contended that this approach was acceptable given the fixed price nature of the decommissioning contract. Additional justification, for not providing a significant level of detail, was that the utility had received other fixed price bids that were very close to the Westinghouse bid, thus verifying its accuracy. The utility presented considerable detail on the bid and bid evaluation process. It also pointed out that it had a cost estimate prepared by one contractor as part of the contractor's decommissioning studies, and that when that same contractor bid on the work as a fixed price contract, his bid was more than double his earlier estimate. This was intended to demonstrate that a cost estimate was not nearly as reliable as a fixed price bid. The utility also provided information on the scope of work within each of the twelve major categories of cost, but did not break down the costs beyond that level of detail.

At a meeting held with the NRC to discuss the cost estimate issue, the NRC stated that the above approach did not satisfy their interpretation of the requirements of 10CFR50.82(b) (4). The NRC stated that the fixed price contract was fixed for a given scope of work, but performance of that work did not guarantee that the site would be releasable. The NRC required additional details to determine if the work and funds provided would permit completion of the decommissioning and termination of the FSV license. The NRC stated that it considered it their responsibility to assure that the decommissioning estimate was complete and accurate so that the contractor would not partially complete the contact and run out of money and default on the contract, leaving the utility with insufficient funds to complete the decommissioning and with responsibility possibly reverting to the NRC. The NRC requested that for each individual task an analysis be performed that included details on the method of accomplishment, the estimated duration of the task, crew size required, equipment required, supplies required, volume of waste, radiation levels, and radiation exposure. It also requested transportation and burial costs and required that if any alternate methodology were considered, that the alternate method must also be estimated, and the differential costs included. This resulted in the preparation and submittal of a detailed cost estimate consisting of approximately 500 pages of information. After extensive review, the NRC concluded the level of detail was adequate and accepted the cost estimate submittal.

Radiation Protection Planning (ALARA)

An issue raised by the NRC on the initial PDP submittal was the level of detail provided for the ALARA plan. The NRC requested detailed information on compliance to ALARA principals, controls, procedures and equipment required to protect employees. A response was provided, that revised the ALARA section of the PDP (Section 3.2). The restructured section used Regulatory Guide 8.8, in both form and content, to identify categories requiring discussion. Further conversations and formal RAI's from the NRC indicated that the NRC appeared to be convinced that, with the submittal of our accident analysis, the likelihood of creating an event that would produce significant offsite dose was extremely low. The accident analysis, submitted as part of the PDP, had calculated a maximum whole body dose of 121 m rem at 100 meters from the Reactor Building for the worst postulated event. As a result, a primary focus of the NRC review was the potential on-site exposures.

Occupational Radiation Exposure

The NRC, as part of their review of the PDP, requested in a RAI the "submittal of the final dismantlement methods and supporting safety analysis for NRC review. Alternatively, provide descriptions and safety analysis for potential options that may be used. Include evaluations of, and methods to minimize personal exposure in your safety analysis."

Initial reaction was one of confusion as to the intent of this RAI. After considerable internal discussions, Westinghouse Team concluded that the NRC was requesting a line-item form verification of the occupational radiation exposure estimates that were submitted in the PDP. Telephone conversations with the NRC confirmed this conclusion. The project team had a number of concerns with responding to this RAI. First, the occupational radiation exposure values submitted with the PDP were estimates only. A detailed review of each and every value could potentially lead to the generation of different radiation exposure values. In addition, a mandated change of the numbers could require a reevaluation of the cost study in that the radiation estimates were a key ingredient in our analysis. Of greater concern was the potential for the NRC to have the project commit to the preliminary estimates by making them part of our licensing bases. The project decided to respond to the NRC's concern by presenting a detailed analysis of only those activities that had the greatest occupational exposures. A day long meeting was set up with project and NRC management to present our proposed approach for this RAI. The project team's presentation included specific evaluation details of people, dose source, equipment and locations. The meeting was very successful in that we not only convinced the NRC that justification existed for the occupational radiation exposures but also that the project team was proceeding in developing decommissioning activities in a safe and thorough manner.

An additional benefit of this day long meeting was the commitment made by both the NRC and the project team for additional communications. As a result, a weekly conference call was initiated. The weekly conference calls became an efficient and successful technique in providing additional details to the NRC in a timely manner, thus eliminating the need for additional and time consuming written questions and responses.

Additional Detail

The NRC issued its major set of review comments to the PDP in February of 1991 which consisted of 60 individual questions. These questions can best be characterized as request for additional details and explanations, and covered all areas of the PDP. Examples of the type of details requested were:

- Radiological precautions to prevent the spread of contamination during diamond wire cutting must be described in detail
- Describe the specific methods that will be used for cutting the piping of the Balance of Plant Systems (saws, torches, thermal cutting, shears)

In responding to the specific questions, the Westinghouse Team was determined to provide as complete information as possible to the NRC, but was also mindful that if the response was too specific (i.e. provided too much detail) it would become part of the licensing basis, and it would be difficult to deviate or change those details as warranted as work progressed without considerable effort. At the same time we were responding to these RAI's, however, final design details were under development. Therefore, in the responses, we attempted to include possible alternative methods and provide upper tier details in order not to tie things down so tightly that we would have no latitude to make beneficial changes later during the decommissioning process.

Tritium Releases

The Westinghouse Team Decommissioning methodology requires the flooding of the reactor vessel with water to provide shielding for component removal operations. However, the graphite blocks in the reactor contain tritium, some of which would be released into the water. The tritiated water later has to be released. The Westinghouse Team estimated that about 500 curies of tritium would be released into the 325,000 gallons of water that would be put into the reactor vessel. A number of alternatives were evaluated. The planned method of disposal was to bleed tritiated water from the reactor vessel and mix it with cooling tower blowdown water which is then discharged to the river which flows past the plant. This has been the utility's normal discharge path for liquid radioactive effluents from Fort St. Vrain during operation. The NRC had a series of concerns with this planned operation.

The first concern was whether the calculated value of 500 curies of tritium that would be released in the reactor vessel was accurate. This estimate was based on calculations that the "maximum" amount of tritium that could potentially exist in the graphite during the entire plant operational history was 100,000 curies. There were technical reports of British experiments which measured the tritium leach rate and cumulative leach fraction in small samples. These reports were used as a bases for the estimate that 500 curies would leach out of the graphite block. The NRC was initially hesitant to accept this estimate based on lack of known experimental data. After an intensive review of the conservatism used in the estimate and a review of all known existing experimental data, the NRC concluded that the 500 curies was a reasonable estimate. However, the NRC remained concerned with the potential for having higher levels of tritium, and requested that they be advised of what alternative methods were available and what point (value) would cause consideration of alternative methods for discharge.

The Westinghouse Team also had considered, but rejected, three alternate methods for disposal as compared to river discharge. These were evaporation from solar ponds, mechanical evaporators, and solidifying the tritiated water into drums. In evaluating the discharge of tritium into the river, the rate of discharge is governed by 10CFR20 limits at the release point. After mixing with plant cooling tower blowdown flow, it is limited to 3 million picocuries per liter. In addition, Appendix I of 10CFR50 places limits on doses to the public of 3 millirem per year. In analyzing the decommissioning operational schedule, the scheduled time from initial filling of the reactor vessel until all the water had to be discharged was 23 months. Over that period of time, a total of approximately 8000 curies could be discharged. Calculations per 10CFR50 Appendix I indicated that 8000 curies in two calendar years was acceptable. However, if more than 8000 curies were required to be discharged, additional time would be required, causing potential schedular delays, at very high costs. Therefore the NRC was advised that up to 8000 curies of tritium in the water would not cause a problem. It was also estimated that the solidification of the water into drums for disposal would add about 10 million dollars to the cost of the project. The NRC was satisfied that the number of curies that could be handled without perturbing the planned operation (8000 curies) was 15 times more than the expected, and even at that, the cost of alternate disposal was within the expected contingency.

During the course of this review with the NRC, a problem arose at the Savannah River Plant involving tritium discharges that may have been in excess of EPA safe drinking water standards. Although the tritium release plan was in compliance with all current regulations, and the river being discharged into is not used for drinking water, the NRC was concerned with this aspect of discharge as well. The NRC therefore requested compliance with the EPA safe drinking water standard of 20 thousand picocuries per liter in the river. The utility, in accordance with its normal Offsite Dose Calculation Manual and the procedures and methods therein, reported that when discharging within the NRC limits and mixing with the river flow, the EPA standards could be met with minimal impact on operations. As a result, meeting the EPA safe drinking water standards became a commitment.

The NRC then asked if the State of Colorado was fully aware of plans to discharge the tritium into the river. The project replied that the State of Colorado, Radiation Control Division had been supplied with all of the Decommissioning documents, including the Supplement to the Environmental Report which discussed the tritium discharge in detail and the resultant dose. In addition, the Radiation Control Division had been involved in numerous project meetings and had only minor comments. They had not made any comment or response on the Supplement to the Environmental Report. Notwithstanding, the NRC requested that the project confirm that the State of Colorado had possession of the Supplement to the Environmental Report, and that they specifically had no issue with the tritium discharge. In discussing this with the State Radiation Control Division, they decided that the State Water Quality Control Division should also be involved.

Prior to this time the Water Quality Control Division had not been aware of these plans. There was also a question as to whether the State of Colorado had a legal basis for regulating this discharge. The State of Colorado Water Quality Control Division also asked why liquid discharge was being done, and why it was not being evaporated. After reviewing the process in detail with the Water Quality Control Division, they indicated that the river flow rates used to determine the amount of dilution were outdated and that significantly lower flow rates should be used. The utility hired an experienced hydrologist, who reviewed the flow data from the last 10 years and recalculated the "Monthly 30 Day Low Flows". These ranged from 118 to 394 cubic feet per second. However, utilizing these lower river flow rates could significantly impact the planned discharge rate if unexpected high quantities of tritium had to be discharged. However, since there was confidence that 500 curies was a realistic estimate, utilization of these revised flow rates was acceptable. The State of Colorado then advised the NRC that "their concerns had been adequately addressed."

In addition, as the discussions with the NRC had proceeded, the utility was very concerned that due to the criterion issue the NRC might withhold its final Decommissioning Order. The utility had irradiated graphite blocks available from a prior outage, however these blocks had differences from the graphite blocks of concern in terms of tritium content, size, shape, surface-to-volume ratio, and graphite purity. The utility decided to perform a test on two of these blocks and the test results indicated that the tritium was leached from the blocks at a rate of one-fourth the magnitude of the British test during the 30 day test period. This information was also reported to the NRC and helped to obtain approval for river discharge.

PRESENT STATUS

Today all of the NRC concerns have been resolved and the NRC issued a Decommissioning Order and a Safety Evaluation Report (SER) in November 1992. The SER included no surprises or open issues.

In the SER, the NRC indicated that they intended to invoke the standard 10CFR50.59 process for Fort St. Vrain decommissioning, basically only changing the regulation's wording to replace the words "Safety Analysis Report" where it appears in reference to the licensing document with the words "Decommissioning Plan". The Westinghouse Team is confident that this change should allow reasonable capability to make changes to the Decommissioning Plan as may be warranted as the decommissioning proceeds.

The lessons learned from our decommissioning licensing efforts were 1) ALARA is major NRC concern, and sufficient detail will be required, 2) Decommissioning cost details and bases or justification of the cost estimates will be required as part of the license application and 3) frequent communications between the NRC and the decommissioning project team is extremely helpful in responding to NRC questions, heading off licensing delays and achieving the ultimate objective of a Decommission Order.