Initial Start-Up and Testing of the Fort St. Vrain HTGR – Lessons Learned which May Be Useful for the HTR-PM

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Abstract - Although the activities presented in this paper occurred 40 years ago, there are many observations and lessons associated with Fort St. Vrain (FSV) which may be beneficial in support of the start-up, testing and licensing of the HTR-PM.

This report includes a review of the FSV NPP design including an overview of the requirements and testing program utilized to bring the plant from initial start-up to full power. A sampling of the test results as well as a comparison of the plant design characteristics to actual values achieved at 100% power along with selected overall experiences gained through operation of this plant is also included.

1. INTRODUCTION

Initial construction of the Fort St. Vrain plant began in September 1968 with the issuance of the construction permit by the U.S. Atomic Energy Commission (U.S. AEC, predecessor to the Nuclear Regulatory Commission, U.S. NRC). This plant, located on the electrical system of the utility, Public Service Company of Colorado (PSC), featured the largest HTGR ever constructed.

Fuel loading of this reactor began in December 1973 with initial criticality achieved on 23 January 1974. The reactor was comprised of hexagonal (H-451) carbon blocks and featured a highly enriched uranium235/thorium232 fuel cycle with a conversion ratio of ~ 0.60.

A very comprehensive testing program was then initiated starting with low power physics testing to 2% power and continuing on to Rise-to-Power tests to 100% power. As this plant featured a first-of-a-kind nuclear reactor, the testing program was the principle means for assuring plant nuclear safety on License-by-Test basis with hold points at power levels of 2%, 5%, 30% and 70%.

Initial electrical power generation occurred in December 1976. Full power was subsequently achieved in November 1981. A review of the major design values with those achieved at 100% power is presented later in this report.

2. GENERAL DESIGN OVERVIEW

The plant utilized helium as the primary coolant and had a graphite moderator and reflector core. The entire primary coolant system including the active core, steam generators and helium circulators (shown to the left on Figure 1.) were contained within a 32 m. prestressed
concrete reactor vessel (PCRV). Helium at 4.8Mpa was discharged from 4 helium circulators and passed down through the core, picking up heat from the fission process. The heat was then distributed equally through 12 steam generator modules. With the exception of the unique once-through steam generators, the secondary side of the plant was much like that of a conventional fossil-fueled unit with 16.7Mpa main steam pressure, 538°C main steam/538°C reheat steam temperature.

Main steam leaves the steam generator modules and enters the high pressure turbine throttle. Upon leaving the high pressure turbine, the steam then passes through a single stage turbine which provides the driving force for the four helium circulators. This steam is then reheated to 538°C in the upper (reheat) section of the 12 steam generator modules. The hot reheat steam is then directed back to the intermediate pressure turbine and then on to low-pressure turbine and finally to the condenser.

Table 1: Plant Design Characteristics

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal Power</td>
<td>842 MWt</td>
</tr>
<tr>
<td>Net Electric Power</td>
<td>330 Mwe</td>
</tr>
<tr>
<td>Power Density</td>
<td>6.3 MW/m3</td>
</tr>
<tr>
<td>Core Outlet Temperature</td>
<td>775°C</td>
</tr>
<tr>
<td>Helium Pressure</td>
<td>4.8 Mpa</td>
</tr>
<tr>
<td>Main/Reheat Steam Temperature</td>
<td>538°C/538°C</td>
</tr>
<tr>
<td>Reactor Type</td>
<td>Hexagonal Block</td>
</tr>
<tr>
<td>Fuel</td>
<td>TRISO HEU - Thorium Carbide</td>
</tr>
<tr>
<td>Primary System Vessel</td>
<td>PCRV</td>
</tr>
<tr>
<td>Total Lifetime Electric Generation</td>
<td>~ 5,500 GWh</td>
</tr>
</tbody>
</table>

The plant was shutdown in August 1989, and has subsequently been decommissioned.

3. INITIAL PLANT LICENSING

FSV, because of its uniqueness as a HTGR rather than a LWR, was initially licensed under the Code of Federal Regulations section 10CFR50.21 with a Class 104 license. This class of license was for medical therapy and research facilities. As such, no Standard Review Plan or Regulatory Guides initially existed for HTGRs, so definitive acceptance criteria did not exist for NRC staff in determining if a system or program was acceptable at FSV.

As the nuclear power industry in the U.S. was LWR based, regulatory guidance was focused on PWRs and BWRs which, as you are well aware, are considerably different than a HTGR, particularly in the areas of system design and plant safety attributes. Consequently, interaction between PSC and NRC on licensing issues often resulted in negotiation and the need for further considerable (often third-party) analysis in order to obtain resolution. This was particularly evident with new regulations such as environmental qualification of electrical equipment and fire protection.

The end result of this lack of regulatory guidance often resulted in considerable added cost, plant delays, diminished credit for the HTGR’s unique safety features and lack of confidence between the regulator and the licensee.

4. INITIAL STAFFING

This section on plant personnel staffing is being included because both the initial HTR-PM and FSV share a common situation, that of incorporating a new, never built or tested nuclear reactor into their plant designs. Therefore, lessons learned from FSV may be of support to the designers and operating personnel of the HTR-PM.

Training of the initial plant staff began in 1968 and was closely governed by requirements set forth from the US AEC. As this plant included a first-of-a-kind power reactor system that had
never before undergone initial core physics and start-up testing, it was determined that plant operating and engineering personnel be required to secure a Cold Senior License from the US AEC. Although the requirements for licensing of nuclear plant operations personnel have probably changed in the past 40 years, this designation of a cold license involved more stringent requirements than licensing of operating personnel in a standard nth-of-a-kind nuclear power plant.

Initially, Public Service Company selected operations, technical support and management staff from existing fossil-fired plants. These candidates were then sent through a Basics of Nuclear Engineering class conducted by Colorado State University. Final candidates were then selected and sent into the program outlined below which was designed to eventually allow them to receive a Cold Senior Operator license.

The requirements set by the USAEC on the FSV staff for achieving this license included the following:

- Training with subsequent testing of personnel by the USAEC at a nuclear power facility with a reactor design as similar as possible to FSV. We were fortunate to be able to train at Philadelphia Electric’s 110MWt Peach Bottom #1 HTGR plant.

- Personnel successfully completing the program at Peach Bottom were then required to participate in formal training at the facilities of the FSV plant designer.

- Training (both formal and on-the-job) continued for another nearly three years at the FSV site. Approximately midway through this period, plant operating personnel were again tested by the USAEC.

- Approximately four months prior to the time of initial reactor fuelling, the selected plant staff underwent final testing by the USAEC. Successful candidates were then issued Cold Senior Operator Licenses.

The overall testing program required to license the candidates included five comprehensive eight hour exams (three written and two oral) all administered by AEC specialists. Although the personnel licensing program lasted over four years, it was coordinated with scheduled construction of the plant to minimize any delay in initial fuelling. Plant operation in an environment where the reactor core physics as well as the physical plant systems have never been required to perform utilizing nuclear heat, dictated the need for an abundant degree of caution. During the initial rise to power of FSV, plant staff was subjected to a number of situations not previously foreseen. Examples include a number of temporary loss of forced coolant events and reactor operational excursions due to an unanticipated ingress of moisture into the primary coolant and movement of a portion of the reactor core as power approached 70%.

Although FSV incorporated a reactor comprised of hexagonal fuel blocks and the HTR-PM utilizes a core of fuel pebbles, any unpredicted movement within the core can result in serious operational concerns. As an example, during the initial rise-to-power testing, as reactor power approached 70%, individual core region outlet temperatures would tend to fluctuate by as much as 40°C and all six power channels would vary with channel six being the most pronounced showing prompt power changes of up to 8% over cycles of 8 to 20 minutes in duration. After an extensive three year testing and analyses program, the apparent cause of the fluctuations was determined to be thermal induced movements of core components accompanied by periodic changes in bypass flows and crossflows of primary coolant helium as plant power approached 70%. Elimination of the fluctuations was resolved by placement of metal locking devices between the columns of fuel elements at the top of the core.

5. PLANT TESTING PROGRAM

The plant testing program consisted of four independent phases. The first phase was Cold Check-Out testing and included validation of systems and associated instrumentation to assure that construction was in keeping with design requirements. This phase was conducted as significant portions of individual systems became available upon completion of construction.
The second testing phase was performance of Preliminary Operating Tests on systems and components as they were completed and passed on to the operating staff for testing. This was a formal testing program that did not involve use of nuclear heat to assure that system and component operation was in accordance with design values. This phase of testing culminated with performance of a Hot Flow Test. This test utilized the compressive heat generated by the helium circulators to heat up the primary coolant system to operational temperatures thereby assuring that the system components would operate properly at high temperatures.

The start-up testing program was divided into two groups, the low power physics tests (A) and the rise-to-power test program (B). Individual low power physics tests that were performed are shown in Table 2.

<table>
<thead>
<tr>
<th>Test/Activity</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>Initial core loading</td>
<td>Loading of fuel elements, temporary absorbers &amp; reflector elements while maintaining a subcritical configuration.</td>
</tr>
<tr>
<td>Pulse source measurements &amp; initial criticality</td>
<td>Using a pulse neutron source, the core was repetitively pulsed to determine shutdown margin in various configurations.</td>
</tr>
<tr>
<td>Test of control rod drive &amp; orifice mechanisms</td>
<td>Temporary absorbers were removed and permanent control rod and orifice mechanisms were installed &amp; tested</td>
</tr>
<tr>
<td>Differential control rod worth measurements</td>
<td>Using a reactivity computer, the worth of control rod pairs and groups were calibrated</td>
</tr>
<tr>
<td>Neutron flux distribution measurements</td>
<td>Axial flux distributions were measured for unrodded, rodded and partially rodded regions</td>
</tr>
<tr>
<td>Reactivity coefficient measurements</td>
<td>Core pressure and temperature coefficients of reactivity were measured</td>
</tr>
<tr>
<td>Fuel Handling Machine (FHM)</td>
<td>Using the computerized FHM, one region of the core was unloaded, transferred to the fuel storage facility and then returned to the core</td>
</tr>
<tr>
<td>Helium purification test</td>
<td>Purification test performed to remove impurities from the reactor core</td>
</tr>
<tr>
<td>Helium purification system during initial heating and outgassing of the reactor core was demonstrated</td>
<td>Helium circulator performance Operability of the circulators driven by both water and steam was verified</td>
</tr>
</tbody>
</table>

Table 2: Low Power Physics Tests (A)

The Rise-to-Power (B) testing program consisted of a large number of tests that were individually scheduled throughout eighty separate sequences. A sequence was a particular separable test entity performed during defined conditions such as steady state, a power change or a transient. The individual Rise-to-Power tests are tabulated below.

B1.) Steam Generator Performance Verification:

This test was performed at a large number of steady state power levels where steam generator helium, feedwater and steam temperatures were logged, as were pressure and steam generator module strain gauge readings. One of the 12 steam generator modules had been highly instrumented with strain gauges and thermocouples in order to assess its performance throughout startup. Similar parameters were logged during and after nine transient conditions which included trips and load shedding. The performance of an instrumented steam generator module with simulated plugging of one feedwater tube was evaluated. The steady state and transient performance of the turbine-generator was also tested. When possible, all safety valves were tested for lifting and reseating pressures by momentarily lifting each valve with a simulated pressure to its actuator.

B2.) Analysis of Chemical Impurities in the Primary Coolant:

At selected time intervals, some as short as four hours, during the entire Rise to Power test program, but at lease on a daily basis, analyses were made of the primary coolant and helium purification system impurities. Table 3. is an example of chemical impurities at 67% power and 688°C helium core outlet. The normal flow rate through the helium purification system was 15% of the primary coolant per hour. Outgassing of impurities, particularly oxidants, after a refueling would take ~ 2 weeks with power increases in a step-wise manner to increase primary system temperature.
<table>
<thead>
<tr>
<th>Chemical</th>
<th>Concentration (units)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Hydrogen</td>
<td>5.0 ppmv</td>
</tr>
<tr>
<td>Oxygen/Argon</td>
<td>0.0</td>
</tr>
<tr>
<td>Nitrogen</td>
<td>3.8</td>
</tr>
<tr>
<td>Methane</td>
<td>0.52</td>
</tr>
<tr>
<td>Carbon Monoxide</td>
<td>5.4</td>
</tr>
<tr>
<td>Carbon Dioxide</td>
<td>1.7</td>
</tr>
<tr>
<td>Water</td>
<td>0.5</td>
</tr>
</tbody>
</table>

Table 3: Chemical Impurities at 67% power

B3.) PCRV Performance

This test was performed at steady state power levels and included evaluation of the effectiveness of both the liner thermal barrier and the liner cooling system. The structural response of the PCRV to temperature and pressure changes was verified at various power plateaus. A daily log was kept of helium flows to penetrations and of make-up for losses. Included above is a typical concentration of primary coolant chemical impurities exhibited at a power level of 67% during the rise to power testing program.

B4.) Primary System Performance

The performance of the four helium circulators and their auxiliaries was evaluated at steady state power levels. The average power density in each of the thirty-seven core regions was measured and compared to the average power density of the entire core.

B5.) Plant Instrumentation Performance

The power range nuclear instrumentation was calibrated using a heat balance over the PCRV and the secondary coolant system. Calibration thermocouples were traversed through the region outlet thermocouple assemblies and their measurements were compared to the four permanently installed thermocouples at each of the thirty-seven region outlets. Feedwater flow instrumentation was calibrated using venturis and by considering extraction steam and feedwater losses.

B6.) Plant Transient Performance

The transient performance of the plant protective system (PPS) was demonstrated after nine planned trips, each selected to duplicate those that could occur during plant operation. Additional data were acquired from a large number of unplanned trips.

B7.) Plant Automatic Control System Performance

The response of eighteen separate control subsystems, which operate independent of and are subservient to the Plant Protective System, were evaluated during steady-state operation and during power shifts.

B8.) Reactivity Coefficient Measurement

The temperature coefficient of reactivity was measured (Fig. 2) during increasing and decreasing power shifts.

Fig. 2: Temperature Defect vs. Fuel Temperature

B9.) Differential Control Rod Worth Measurements

Control rod pairs were calibrated by differential measurements using the reactivity computer and normal plant instrumentation while at selected steady-state power levels. Examples of the differential control rod worths obtained in this test showed that the final measurement for the maximum worth rod pair was 0.036 compared to a preliminary calculated worth of 0.048. The total worth of the first four rod groups was measured as 0.088 compared to a preliminary calculation of 0.089.

B10.) Xenon Buildup and Decay Measurements

The xenon behavior was recorded at steady state power levels, reached after an increase or decrease in power. The xenon reactivity effect was then measured using control rod worths and temperature coefficient data.

B12.) Shielding Surveys
The adequacy of the plant shielding was determined at three steady state power levels and during a rapid regeneration of one of the purification systems.

B13.) Radioactive Analysis of the Primary Coolant

Radioactive gaseous and iodine probe analyses were made at all major steady state power levels.

6. 100% POWER OPERATION

Plant testing continued throughout the remainder of the 1970’s and on 6 November 1981 the plant achieved 100% rated thermal power. Table 4 provides a summary of the some of the major design characteristic versus actual values realized.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Design</th>
<th>Actual</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor Power</td>
<td>842 MWe</td>
<td>842 MWt</td>
</tr>
<tr>
<td>Gross Electric Power</td>
<td>342 MWe</td>
<td>324 Mwe</td>
</tr>
<tr>
<td>Helium Pressure</td>
<td>4.8 MPa</td>
<td>4.8 MPa</td>
</tr>
<tr>
<td>Core Inlet temperature</td>
<td>395°C</td>
<td>387°C</td>
</tr>
<tr>
<td>Core Out Temperature</td>
<td>775°C</td>
<td>767°C</td>
</tr>
<tr>
<td>Helium Flow</td>
<td>442 kg/s</td>
<td>490 kg/s</td>
</tr>
<tr>
<td>Main/Reheat Steam T.</td>
<td>538/538°C</td>
<td>533/524°C</td>
</tr>
<tr>
<td>Feedwater Flow</td>
<td>292 kg/s</td>
<td>296 kg/s</td>
</tr>
</tbody>
</table>

Table 4: Full Power design vs. Actual Results

At 100% power it was necessary to operate the four helium circulators at higher than normal flow rates because of a regenerative heat transfer problem in the steam generator modules. This was attributed in part due to excessive regenerative heat losses to the cold helium from the main steam downcomer tubes where they join their subheaders (the cold helium exits the steam generator at this position), but dominantly due to heat losses by the main steam subheaders where they are in close proximity in the reactor vessel penetration interspaces. Over blowing of the core was required at lower power levels to accommodate the regenerative losses, but this diminished at higher power levels.

Over blowing of the core and reheat steam attemperation resulted in consistently lower core helium outlet temperatures. These data are suspect because of a known core primary coolant bypass flow that is diverted into the thermocouple sleeves that are positioned at the outlet of each core region. The bypass flow can vary as core thermal-hydraulic conditions vary, but its effect is always to depress the measured core outlet temperature. Figures 3 and 4 provide examples of the design and operational results characteristics achieved in the test program at 100% power.

![Fig. 3: Helium Flow vs. Thermal Power](image)

![Fig. 4: Core Helium Outlet Temp vs. Thermal Power](image)

7. PLANT RADIOLOGICAL CLEANLINESS

The radiological consequences of operating Fort St. Vrain were exceptionally low. The personnel exposure rates were consistently two orders of magnitude below those experienced by the light water power plants in the U.S. The primary reason for this was inherent in the design of the core and primary coolant system including the unique graphite core structure with the TRISO coated fuel particles, the use of helium as the inert primary coolant and the PCRV with its double sealed pressurized (with purified helium) penetrations. For example, the primary coolant circulating activity at 100% power was 439 Ci
versus a design value of 30,900 Ci and an expected value of 2,630 Ci as shown in Figure 5.

A cumulative total of 180 m$^3$ of low-level solid waste was generated at Fort St. Vrain throughout 15 years of operation (through 1988). This included waste generated as the result of initial fuel loading, three refuellings and major modifications and plant maintenance on components within the primary coolant system. Except for tritium (from the circulator bearing water system), noble gas airborne and liquid effluent releases from Fort St. Vrain were consistently an order of magnitude below the average of the U.S. LWR program through 1988.

![Fig. 5: Circulating Activity](image)

### 8. LESSONS LEARNED

Although the Chinese HTR-PM plant incorporates significant system differences from the FSV plant, there are considerations applicable to both plants for the designer and the operator. These are especially important for key systems that incorporate first-of-a-kind equipment.

- **Complexity breeds unavailability.** Incorporate system/components that are ruggedly simple in design and have a history of reliable operation and minimal maintenance.

- **Assure that there is strong plant support for this initial HTR-PM, including technical expertise and associated funding.** Quite likely, the successful startup and operation of this plant will require a level of support considerably higher than a typical nuclear plant.

- **Be very attentive to the design aspects of first-of-a-kind components located in the Class 1, Safety-Related portions of the plant.** For example, a generic metallurgical failure could easily lead to an extensive plant shutdown in order redesign the failed component and then go through the subsequent re-licensing, manufacture, installation and testing process.

- **Where possible, test all key systems to actual plant configuration and design characteristics ahead of actual installation in the plant.** This is especially important for first-of-a-kind equipment.

- **Do not be reluctant to incorporate a generous over-build capability into systems and components.** It is significantly easier to have extra margin in compressors, pumps, motors, etc. than to be required to backfit into larger units.

- **Never attempt to license or construct a new innovative nuclear power plant without first having the proper regulatory guides and criteria in place.**

- **Assure that the Final Safety Analysis Report accurately reflects the actual capability of the plant to conservatively respond to accident conditions.**

### 9. CLOSING

In closing, I would like to say that I am proud and very fortunate to have chosen the HTGR as my profession throughout these past 40+ years. This advanced nuclear power source has repeatedly demonstrated the capability to provide:

- **Exceptionally good radiological cleanliness including personnel...**
exposures consistently an order of magnitude below the LWR industry.

- Outstanding safety features such as handling of temporary loss of forced cooling events without approaching core temperature limits.

- Cycle efficiencies approaching the highest in the industry.

- Generation of high quality steam providing an excellent energy source for a multitude of process heat applications.

### 10. REFERENCES


