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TASK 1—INSTRUMENTATION IN VHTRS FOR PROCESS HEAT APPLICATIONS

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1. INTRODUCTION

The objective of this project is to support NRC in identifying and evaluating regulatory issues concerning the instrumentation systems proposed for use in the Department of Energy's (DOE) Next Generation Nuclear Plant (NGNP). The NGNP is to provide commercial industries with high-temperature process heat for industrial processes including hydrogen production, as well as electrical power. The high temperature gas-cooled reactor (HTGR) can provide process heat at temperatures from 700 to 950°C. Note that for the upper range of these operating temperatures, the HTGR is sometimes referred to as the Very High Temperature Reactor (VHTR). For the purposes of this project, since DOE's NGNP program language refers to a VHTR, the gas-cooled reactor described herein will sometimes be referred to as the VHTR even though DOE's current plans focus on the lower end of that temperature range for ultimate deployment of NGNP.

VHTR and process heat application instrumentation will be required to operate in high-temperature environments. This work will provide NRC staff (a) insights and knowledge about instrumentation used in other high-temperature reactors, either those operating or that have operated, and what is currently considered state-of-the-art; (b) information to assist in developing criteria to qualify instrument hardware and embedded software; and (c) input for developing the bases for potentially new regulatory guidance to assist in the review of NGNP license applications and providing support in NRC reviews and inspections.

The main function of Task 1 is to provide information about the instrumentation likely to be used in the NGNP primary system. This includes surveys and critiques of instrumentation designs used in prior and current HTGRs, as well as looking forward to potential applications of new technologies. Information about instrumentation for the process heat supply and delivery systems is covered in Task 2.

Work has been closely aligned with the companion project on Advanced Reactor Controls (N6177), since many of the systems obviously interact closely, and functional requirements involved are often common to the two.

The overall effort was impacted by the fact that the NGNP design and functional requirements have not yet been established. In the latter part of this effort, it has appeared that the initial NGNP design will likely have a steam generator in the primary loop, with the steam produced used both to power a conventional steam-turbine generator and to provide high-temperature steam to a relatively nearby industrial process that requires heat in the 600–700ºC range. In addition to the direct steam generator option, designs requiring intermediate heat exchangers (IHX) between the reactor primary system and the "processes" are also investigated.

2. OUTLINE AND SUMMARY OF SUBTASKS

2.1 Document Collection

This subtask included literature searches and the creation of a library with collections and organization of reports and presentations stored in an internal (ORNL electronic) library. Those reports intended for NRC use were forwarded to a SharePoint system accessible to NRC program participants. An organization (taxonomy) scheme, with the goal of making it easier for the user to track down information for specific missions, is still being developed. Collection topics and important findings gained from the literature surveyed are summarized, with emphasis on regulatory and licensing implications.

2.2 Instrumentation Categorized by Subsystem and Function

Some system components will require specific instrumentation for which the attributes will depend on function and operational conditions (such as normal range operation and postulated accident conditions). Example subsystems are:

1. reactor core,
2. reactor pressure vessel (RPV),
3. shutdown cooling system (SCS),
4. reactor cavity cooling system (RCCS),
5. steam generator (SG),*
6. intermediate heat exchanger (IHX),
7. gas turbine,*
8. heat transport loop (to a process heat plant),* and
9. containment/confinement systems.

Some core instrumentation would be markedly different for pebble bed and prismatic designs. Since there is little information currently available about potential process heat plant design and the I&C needs for such a plant would be entirely dependent on features of that selection, work was concentrated mostly in areas common to the potential plant designs. Requirements for postulated accident conditions will be covered in Task 2.

Instruments will have varied requirements depending on many things such as sensor installation location, environment, accuracy and dependability needs, and safety classification. The requirements may also depend on plant design features (e.g., primary system dust measurement and moisture detection needs will differ with reactor type). There have been significant advances in many of the relevant technologies in the past decades, including better “smart sensors,” and extended capabilities may be needed to operate in the high temperature, pressure, and radiation environments. Descriptions of measurement functional requirements and the attributes of existing instrumentation are summarized for the following parameters:

a. temperature,
b. flow,
c. pressure,
d. neutron flux,
e. gamma,
f. fission product release (circulating and plated-out activity),

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*Covered mostly in Task 2, where appropriate.

g. dust concentration,
h. moisture (in primary system),
i. pebble burnup (PBRs),
j. confinement atmosphere chemistry,
k. position (control rods), and
l. vibration

In this report, the nuclear and process instrumentation systems are mostly described as they were applied to specific reactor designs, giving the reader both operational and experiential information.

2.3 Impacts of Instrumentation Errors or Failures on Plant Safety

Because of the many inherent and passive safety features of the modular HTGR, reactor behavior tends to be “self correcting” for many transients and accidents, so instrumentation and control failures are likely to be less damaging than for “non-inherently safe” plants. For example, safety demonstration tests at HTR-10 and AVR showed how the reactors easily survived a loss of flow without scram test and, for HTR-10, an add-on test had a simultaneous control rod withdrawal. In all cases, the reactors shut themselves down. However, one can postulate a number of cases where erroneous measurements could lead to operator actions resulting in fuel failure.

2.4 Summary of Major Instrumentation Issues

Issues covered in this section are those which are of particular licensing interest and which are likely to receive special attention. For these items, considerations should include:

a. perceived gaps between instrumentation needs and the existing capabilities and availabilities,
b. regulatory and licensing implications of the gaps,
c. R&D and special experiment needs, and
d. how variations in the ultimate design features could affect instrumentation requirements.

2.5 Measurements Needed in TRISO Fuel Manufacture

A case was made in a proposal (white paper) to include in this project an assessment of TRISO fuel measurement (and control) technology as it applies to fuel manufacture. This is based on the fact that the fission product barriers in the TRISO particles represent the first line of defense-in-depth arguments for plant safety and licensing. While NRC management decided not to include such an assessment as part of this project, a case for NRC conducting this study (Sect. 7) is presented.

3. DOCUMENT COLLECTION

The document collection subtask involved all Instrumentation (N6668) project participants (as well as the participants in the Advanced Reactor Controls Project—N6177), as a part of their usual literature search activities. The work also included concerted efforts to build up a specific reference library that would focus on instrumentation technology state of the art, as well as on I&C-related experiences with operating HTGRs. There was also some useful overlap with another NRC-RES program (Knowledge Management for HTGRs—N6217) that covered broader areas of HTGR technologies related to safety and licensing. Further contributions to the document collection subtask are expected as more information is found in the course of future work on the project. More thought also needs to be given to document organization (taxonomy) to aid users of the collections in their specific quests.

User goals are expected to include the following elements:

a. historical uses of instrumentation in gas reactor and related processes;
b. state-of-the-art instrument technology, including accuracy and reliability;
c. I&C-related plant design descriptions;
d. parameters of NGNP operation and postulated accidents that need to be monitored;
e. assessment of the phenomena for which measurements are crucial;
f. assessment of the potential consequences of measurement errors (e.g., drift), misinterpretations, and instrument failures; impacts on licensing; and
g. sensor interfaces with control and safety systems.

To help accomplish these goals, documents are arranged in the following categories or groupings:

a. instrumentation in gas reactor designs—includes overviews of current designs; instrumentation in NGNP/VHTR, MHTGR [1980s 350-MW(t) steam cycle]; HTR-10 and HTR-PM (China); HTTR (Japan); GT-MHR and PBMR; Fort St. Vrain; and AVR (Germany);
b. O&M issues (historical);
c. new instrumentation—digital design, reliability, ISA-IEC-IAEA standards, software QA and validation, and wireless; research activities;
d. PIRT studies (what's important, where are the gaps?);
e. control rooms—HMI, multiple units;
f. experimental facilities—potential for testing;
g. accident analyses—determinations of extreme conditions; and
h. process heat plant and heat transport loop instrumentation.

4. SURVEY ON INSTRUMENTATION USE AND EXPERIENCE IN PAST HIGH-TEMPERATURE GAS REACTORS

4.1 Introduction

Information on high-temperature gas reactor (HTGR) instrumentation is presented. Reactor instrumentation is typically divided into four major categories: (1) in-core instrumentation, (2) process instrumentation, (3) analytical instrumentation, and (4) other instrumentation.

4.1.1 In-core instrumentation

In-core nuclear information provides data on the status of key nuclear parameters, such as neutron flux. Out-of-core sensors in gas-cooled reactors are generally the same as those used in water reactors except for their installation. High temperatures involved in these reactors create challenges in the design of in-core neutron detectors.

Neutron sensors used for most gas-cooled reactors are (1) low-level pulse proportional counters (boron or fission counters) for the source or start-up range, (2) compensated DC ionization chambers for the intermediate or log power (log N) range, and (3) uncompensated DC ionization chambers for the linear power range. These technologies are mature, and relevant design information can be found from manufacturers' technical specifications.

Fuel element failure detection in HTGRs depends on the fuel element design. The graphite coatings prevent the release of long-lived gaseous and metallic fission products into the coolant. The short-lived

fission products are not released unless the graphite structure fails and exposes the fuel. Since the primary coolant is helium, no corrosive effects are involved, and operation with a relatively small number of failed fuel elements (TRISO particles) presents no immediate hazard.

Temperature measurements could be used in prismatic core fuel assemblies. Nuclear reactors are generally provided with multiple temperature sensors across the core to generate a two- or three-dimensional temperature profile of the core. This is especially challenging in high-temperature reactors because of the much harsher operating conditions involved in these systems. As the operating temperature of the system increases, the response of thermocouples starts to exhibit drift as well as much higher temperature uncertainty primarily due to increased thermal noise.

In addition to the challenges due to the operating environment, pebble-bed reactors bring additional difficulties for measurement. In a prismatic high-temperature gas reactor, the fuel columns could provide space to accommodate the requisite sensors. In a pebble-bed reactor, however, this access is not available since the fuel elements (pebbles) are not stationary due to the on-line refueling.

4.1.2 Process instrumentation

This report presents information on process instrumentation that was used in earlier gas-cooled reactors. Information on instrumentation used in the Peach Bottom and the Fort St. Vrain reactors was mostly compiled and summarized from Refs. 1 and 2.

The process instrumentation is considered in two major categories: (1) primary coolant instrumentation and (2) secondary coolant instrumentation.

Primary coolant instrumentation is generally manufactured to higher quality specifications to be operational in a much harsher operating environment—such as higher temperatures, much higher neutron and gamma fields, etc.—as well as to meet the regulatory requirements.

4.1.3 Analytical instrumentation

Gas analysis, other than moisture determination, in the primary system is necessary to measure both the radioactivity and concentrations of contaminant gases in the main coolant, the helium dryer, and the purification system.

4.1.4 Other instrumentation

Other reactor instrumentation includes sensors and associated equipment provided for gathering additional plant data. These include reactor vessel instrumentation, reactor vessel cavity instrumentation, etc.

4.2 Arbeitsgemeinschaft Versuchsreaktor (AVR)

4.2.1 Temperature

The AVR was used to test specially designed neutron flux detectors that can function up to temperatures of approximately 800ºC. These chambers were tested at different locations in a temperature range up to 500ºC and over the entire power range from startup to full power operation (Ref. 3). The chambers functioned satisfactorily as per specifications and were made available for high-temperature applications in addition to the self-powered neutron detectors (SPND) developed at the Nuclear Research Centre Jülich of the Federal State of North Rhine-Westphalia (KFA).

There is no conventional technique for measurement of active core temperatures in pebble bed reactors, in contrast to other reactors, because the pebble movement would destroy any standard detection equipment.

An average coolant gas core outlet temperature is generally specified as the reference parameter for the hot-coolant temperature of the AVR, which is calculated from the core power, coolant gas inlet temperature, and helium mass flow. No direct measurement of the gas temperature was done in the AVR core. Hot-gas temperature profiles outside of the active core were measured occasionally, but core temperatures were based on calculations only.

An unorthodox experiment was developed to determine the radial distribution of the maximum hot-gas temperature in the core. The experiment "HTA-8—Maximum Gas Temperatures in the Core" aimed to create a radial temperature profile of the core, for which 200 unfueled spheres containing meltable wires were inserted into the core (Ref. 4). Each monitor sphere contained 20 melt wires with different melting points from 655 to 1280°C. The radial position of the pebble in the core was derived from its exit time. For this purpose, labeled monitor spheres were selectively loaded via the fuelling machine. X-rays made the wires visible (melted or not), giving the maximum temperature during the core passage. Post-irradiation examination indicated that approximately 20% of all spheres had all wires melted and had, therefore, seen surface temperatures in excess of 1280ºC. Unfortunately, 1280ºC was chosen as the highest melting point. In contrast, for an average passage of pebbles, a maximum surface temperature of 1150ºC had been predicted by calculations during the 950ºC operation. Figure 1 shows the frequency distribution of temperatures determined by the melt-wire examination at the end of the experiment at full power and an average hot-gas temperature (Fig. 2) of 950ºC. The spheres that experienced hot-gas temperatures in excess of 1280ºC passed through the core zone between the graphite noses where a locally elevated power density prevails.

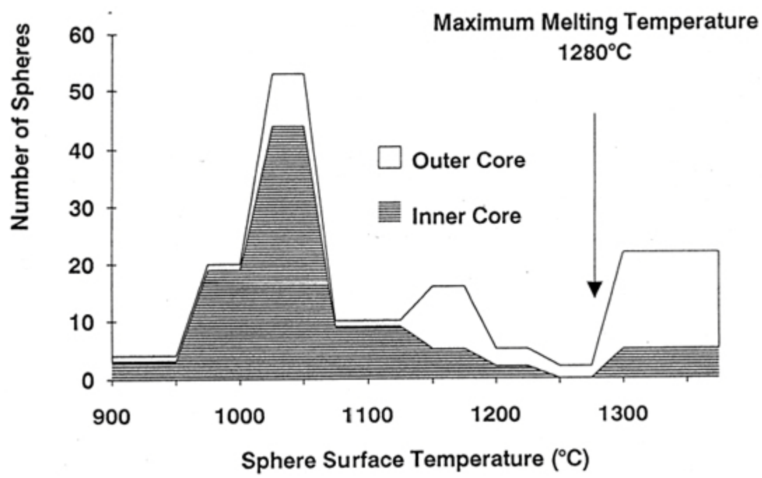


Fig. 1. Frequency distribution of maximum temperatures detected by temperature monitor spheres passing through the AVR core.

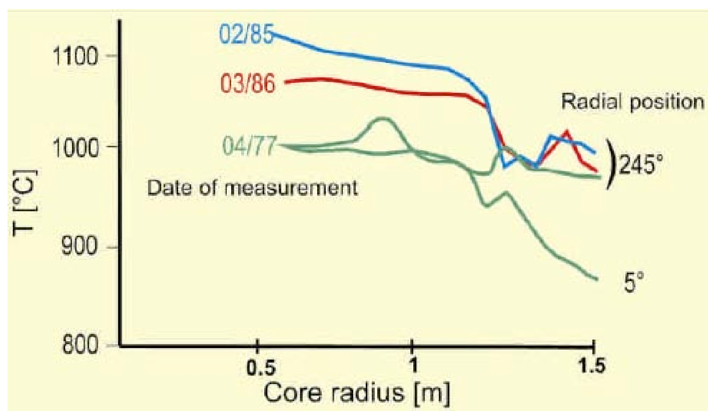


Fig. 2. Measured AVR gas temperature profiles in the hot gas chamber for identical average hot-gas temperatures (950ºC).

For average AVR hot-gas temperatures of 950ºC maximum, core temperatures were originally calculated to be between 1070 and 1123ºC. In the inner core maximum, temperatures are about 75 K lower. Depending on the fission power of the fuel element, maximum coated particle temperatures are up to 120ºC higher than core temperatures.

The AVR was also used for testing advanced techniques for measuring hot-gas stream temperatures in the region near the core. One example is the combined thermocouple/noise thermometer developed by KFA

and Interatom for reliable and accurate high-temperature measurements of long-term stability near the core region. The principle of the noise thermometer is based on the temperature dependence of the thermal noise of selected resistance material. The noise thermometry can be used either as a standalone technique or in combination with thermocouples. A combined system was deployed in the top reflector of the AVR and was successfully tested. Due to demonstrated stability and accuracy, the system was proposed for in-situ calibration of thermocouples. [Refs. 5 & 6]

4.2.2 Dust measurements

In PBRs especially, graphite dust generated from abrasion in passing through the core is of significant concern in its transport and deposition of radioactive fission and activation products. To measure and evaluate dust phenomena, AVR developed the Vampyr test facilities (Fig. 3 and Ref. 7).

Routine testing of coolant activity showed noble gas radioactivity dominating (by orders of magnitude) other major elements; however, they are readily removed by the gas purification system. Noble gas circulating activity, which can be measured continuously, was shown to be an excellent indicator of the major (exposure concern) isotope levels (^{131}I, ^{137}Cs, ^{90}Sr,…), whose detection typically requires accumulation filters with batch measurements following week-long collection periods.

The Vampyr facilities enabled measurements of dust grain-size distributions, long-term contamination profiles as functions of various incidents (water ingress), and bad fuel batches. These required generally quite complex testing procedures [Ref. 8] that would likely be models for dust characterization systems for modern PBR designs. Details of the testing that showed the very large increase in circulating dust activity with blower speed changes are also described (see also Sect. 6.1). A planned test of dust releases in a (simulated) sudden depressurization accident was not executed, perhaps because of the uncertain consequences.

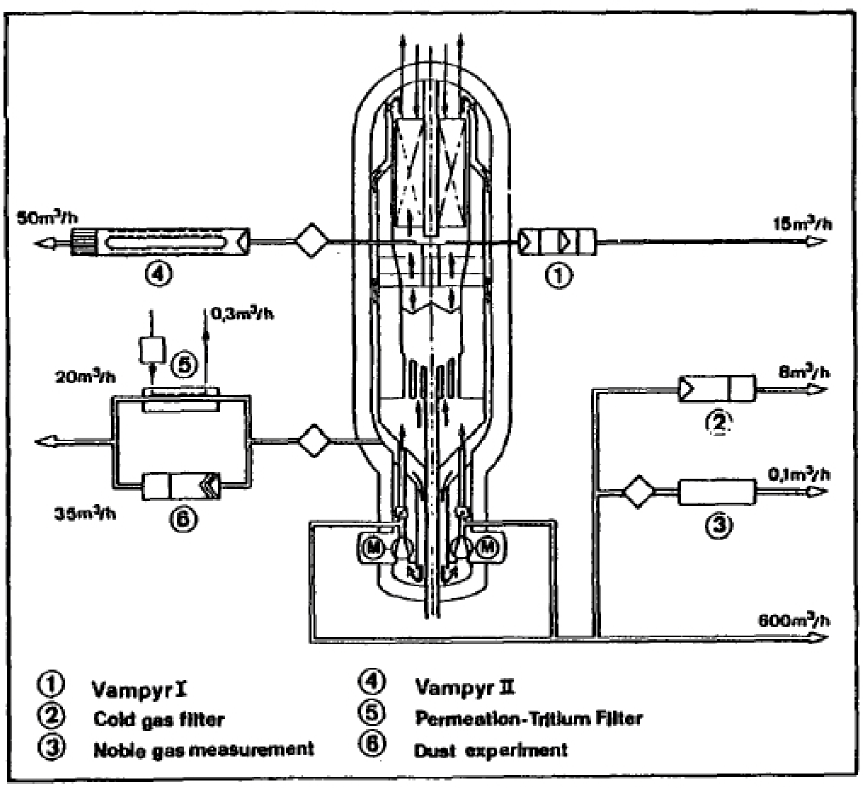


Fig. 3. Test facilities of the AVR primary system.

4.3 Peach Bottom

As a precaution, nearly all instrument pressure lines from the main coolant system and other large helium volumes in Peach Bottom were provided with excess-flow check valves as shown in Fig. 4. These valves were specifically designed for HTGR conditions and operated to shut off automatically any instrument pressure or flow line should the pressure drop across the valve exceeded 96.5 kPa (~0.95 atm), as might happen in a downstream line rupture. The valve is a self-actuated spring-loaded type in which the spring holds the poppet off the seat until high flow rate through the valve closes it. Seat leakage at closure is less than 30 cm^3/min. These excess-flow check valves were installed as close as possible to the helium pressure source.[2]

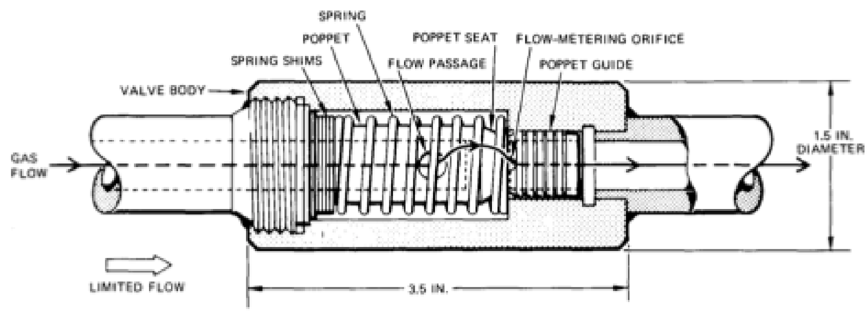


Fig. 4. Excess-flow check valve used in Peach Bottom reactor instrument pressure lines.

Peach Bottom in-core instrumentation includes nuclear instrumentation, failed fuel instrumentation, and temperature sensors.

4.3.1 Nuclear instrumentation

Peach Bottom was equipped with nine channels of permanently installed nuclear instrumentation. These included three source-range channels—^{10}B-lined proportional counters, two intermediate-range channels, and four power-range channels. Additional temporary in-core instrumentation was used for the first core loading. Installation of nuclear instrumentation is shown in Fig. 5.

As typically used in most reactors, approximately two decades of overlap were provided between adjacent ranges. Comparators between redundant channels were used to alarm excessive deviation. The system used solid-state electronics for the count-rate logarithmic and rate circuits, the linear power channels, and all channel bi-stables, comparators, and power supplies. Online testing of each channel is possible at any time during operation.

The neutron sensors were shielded with lead around the bottom to reduce the gamma field. The ^{10}B-lined chambers also employed motor-driven cadmium shields to limit neutron exposure during power operation. For low-level start-up readings, one source channel had triple the sensitivity of the other two.

The intermediate-range instruments covered approximately eight decades down from 500% power. Flux level was indicated, recorded, and compared between the channels. The rate of change in the channels was indicated, compared, and used for scram.

The four power-range channels covered 0 to 150% full power with linear indication. The measurements from the four channels were also used for inter-channel comparison and a variety of special functions. An automatic interlock at 10% power provided protection to inhibit the rate-of-change scram. High-power alarm was set to 120% power, power setback was at 130%, and high-power scram was set to 140%. In addition, all four channels were made available to the automatic flux controller with the possibility of

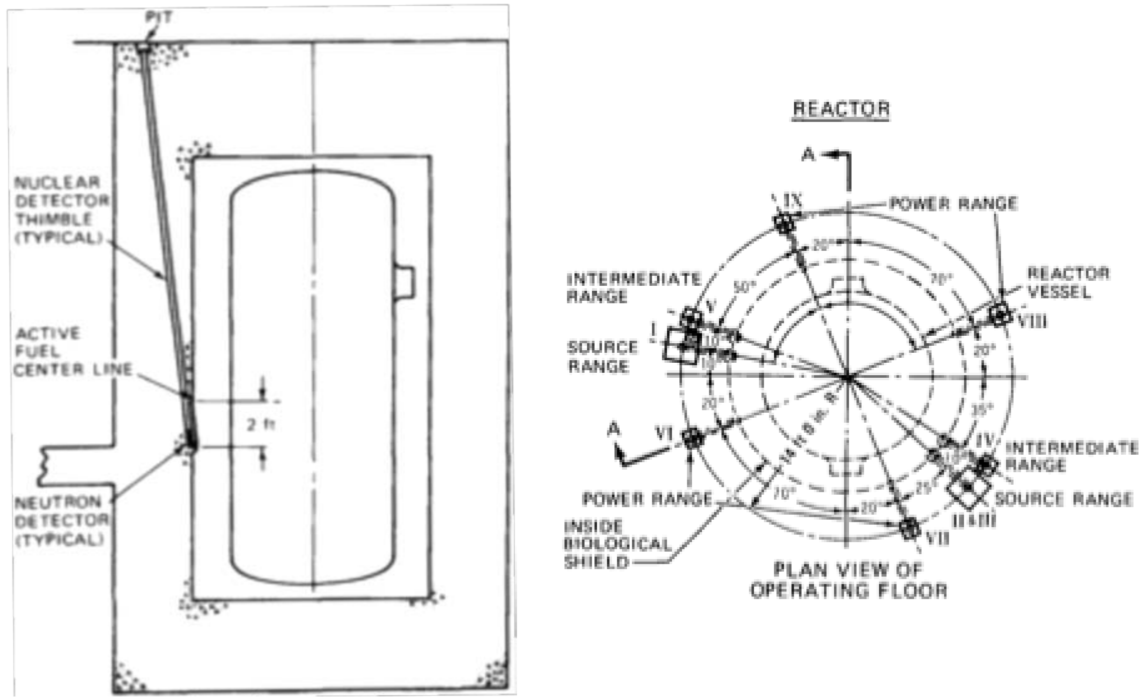


Fig. 5. Peach Bottom nuclear detector installation.

using any single channel, the average of four, or the highest (auctioneered) value of the four. The averaging circuit was arranged so that, on excessive deviation of one channel from another, both channels were automatically removed from the average to prevent hazardous control actions.

Because all neutron sensor wells were water filled, each sensor was hermetically sealed in a waterproof housing. These housings were made of aluminum with welded and O-ring-sealed flanged connections. Each housing was tested for leak tightness by helium mass spectrometry.

4.3.2 Failed fuel instrumentation

In the Peach Bottom reactor, a small purge stream (approximately 1/2 kg/h per element) was drawn from the primary coolant over the compacts inside the fuel element tubes to sweep fission products out to an external trapping system. Should a tube crack, this purge flow becomes ineffective and the primary coolant activity rises. A cracked fuel element tube raises the primary coolant activity by about 3 to 4 Ci—compared to the full-power activity of about 0.5 Ci—which can be monitored on gas flow ionization chambers sampling the bulk primary coolant.

4.3.3 Temperature

For the Peach Bottom reactor, the hot junction of the chromel–alumel thermocouples was formed by brazing No. 28 AWG wires. The hot junction was then insulated with high-purity, reactor-grade magnesia, sheathed in 300-series stainless steel and swaged to 1/16-in. diameter. Ninety-seven thermocouples of this type were used to measure reflector, vessel metal, reactor coolant, and fuel element temperatures below 538°C.

For thermocouples exposed to higher fuel temperatures (up to 1310°C), 59 tungsten-rhenium thermocouples were installed in 36 of the 804 fuel elements. The thermocouples were originally formed by brazing the hot junction, but many open circuited during installation due to tungsten embrittlement. A sheath of high-purity seamless molybdenum tubing clad with seamless high-purity niobium tubing was

used for insulating the hot junction. These were essentially the identical thermocouples used in the first HTGR in the UK, the Dragon reactor.

As an added in-core temperature measurement system, acoustic thermometers were installed in eight selected fuel element spines. These were installed to supplement in-core thermocouple measurements during start-up and to provide at least minimal hot-spot temperature indication following anticipated failure of the adjacent thermocouples near the end of life (approximately 3 years). A cross sectional drawing of the transducer is shown in Fig. 6 along with a graphite fuel element (Ref. 5).

4.3.4 Process instrumentation—temperature

Peach Bottom process instrumentation includes primary coolant sensors and secondary coolant sensors for temperature, pressure, and flow measurements. Chromel–alumel thermocouples were used to measure the primary coolant temperature.

4.3.5 Analytical instrumentation—moisture and gas analyzers

The moisture detector used in the Peach Bottom reactor was of the electrolytic hygrometer type. These detectors are used to monitor the outlet helium stream from each steam generator. A two-out-of-three coincidence circuit is used in the protection system for automatic isolation of a loop if high moisture content is detected. A simplified drawing of a single monitor and a cell element is shown in Fig. 7.

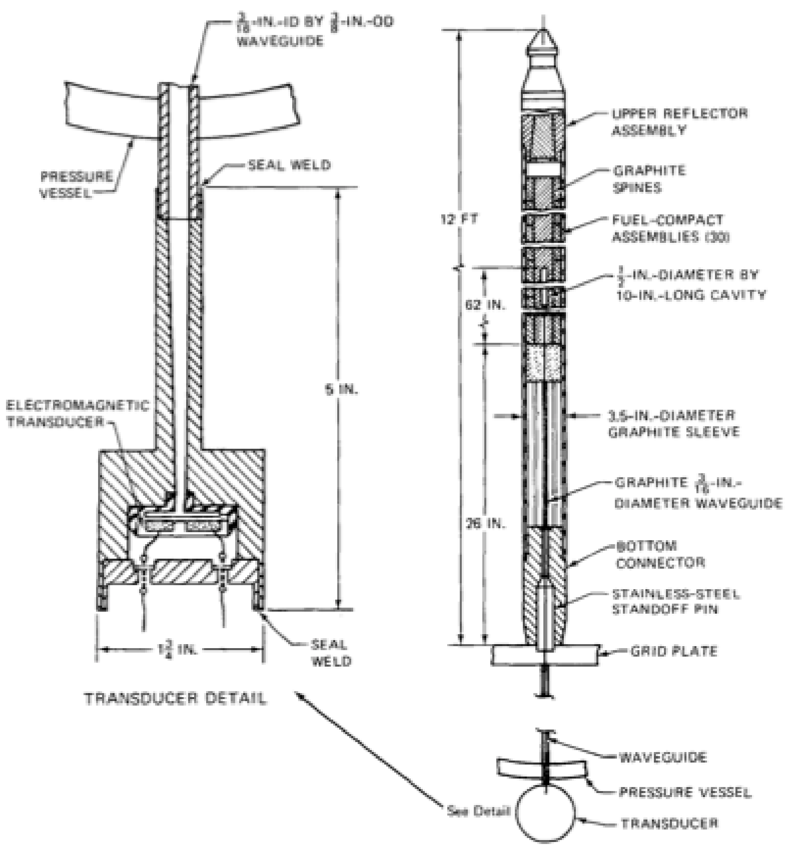


Fig. 6. Cross section of the stainless steel transducer holder, waveguide, and transducer used in the acoustic thermometer at Peach Bottom.[1]

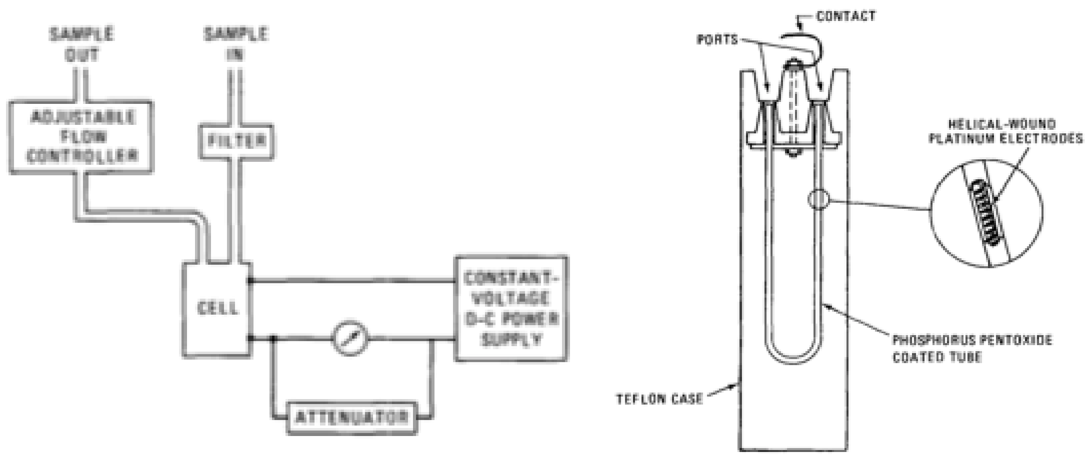


Fig. 7. (*a*) Schematic diagram of the moisture sensor monitor; (*b*) simplified drawing of the electrolytic cell.

Accuracy with the moisture monitoring system was within ±5% full scale in the 0 to 1000 parts per million by volume or 10^{-1} volume % range with a resolution of 1 part per million by volume and a repeatability of ±2%. After a prolonged exposure to very dry gas (<1 part per million by volume), these detectors lose their response and must be reconditioned periodically.

Further monitoring was obtained by an additional hygrometer that sequentially sampled helium purge flow from both steam generator tube sheet baffles. The joints between tubes and tube sheet are the more likely sources of leakage. Detection at this point facilitated an orderly manual shutdown. Four other moisture detectors monitor the purification system at points downstream of the water-cooled delay beds, purified helium, and helium transfer compressors and at the dehydrator beds. Finally, a secondary measurement was made to detect small water leaks in the 0 to 100 parts per million by volume range by using redundant infrared monitors to detect the presence of carbon monoxide (formed in the reaction of water with core graphite) in the main loops.

Peach Bottom used an automatic process gas chromatograph for continuous monitoring of the helium gas for CO_2, CO, O_2, N_2, H_2, and CH_4, measured in the few parts per million range. Samples are taken from the main coolant loop and the exits of the liquid-nitrogen-cooled trap, oxidizer bed, and dehydrator beds. In addition, a non-automatic laboratory chromatograph was made available in the secondary containment air room for analyzing samples from various points in the system.

4.3.6 Reactor vessel instrumentation

Peach Bottom reactor vessel was highly instrumented to monitor vessel temperature and stress. Peach Bottom is typical of a system with a steel pressure vessel. The Peach Bottom pressure vessel internal structure employed eight strain gauge elements, 34 thermocouples that monitor Charpy test samples—to evaluate possible radiation damage to the steel—and 13 thermocouples on the vessel steel. The exterior of the vessel was monitored by 10 strain gauges and 12 thermocouples. These elements sent signals to control room indicators or recorders. They were used during initial startup to prove the design of the vessel; they were occasionally monitored during operation.

4.4 Fort St. Vrain

The Fort St. Vrain reactor used a prestressed concrete reactor vessel (PCRV), which significantly changed the installation of all reactor instrumentation. The PCRVs can be built in a variety of shapes, and the steam generators are usually placed within the vessel. The water content in the steam generators shields the normal neutron sensor location so that the sensors must be mounted within the pressure circuit (i.e.,

either in the core reflector or the inner steam generator shield wall. The steel tendons and reinforcement steel in PCRVs imposed additional difficulties than in a normal concrete biological shield wall.

4.4.1 In-core instrumentation

The installation of nuclear instrumentation in Fort St. Vrain is shown in Fig. 8.

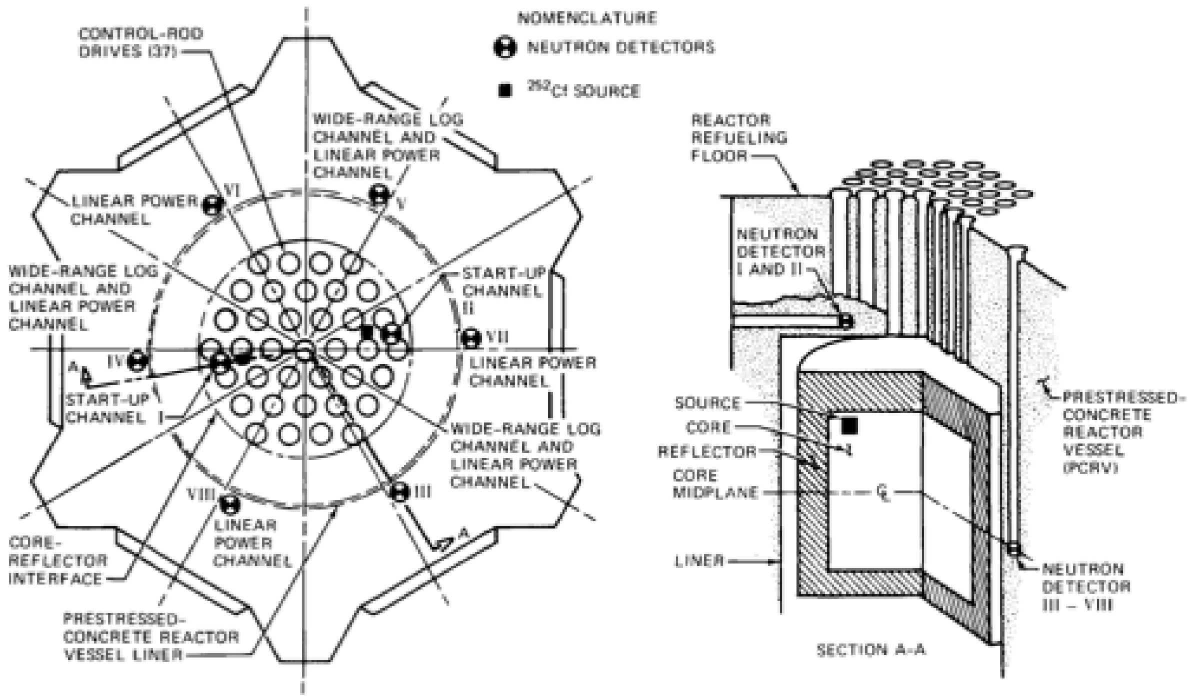


Fig. 8. Installation of nuclear instrumentation in Fort St. Vrain reactor.

4.4.2 Nuclear instrumentation

Fort St. Vrain introduced several advances in nuclear instrumentation design. In addition to two start-up channels, the Fort St. Vrain employed three wide-range log-and-linear power-level channels and three separate linear power-level channels (Refs. 9–11). Each of the wide-range channels derived a wide-range logarithmic power signal and an independent linear power signal from the same fission chamber. The log power readings covered 10 decades and consisted of a conventional log count-rate circuit reading up to 3×10^5 counts/s ($\sim 2 \times 10^5$ neutrons/cm^2/s) and a log mean square (mean-square voltage) circuit to cover from 2×10^5 to 2×10^{10} neutrons/cm^2/s. The linear power signal is a separate channel using the upper two decades of fission-chamber current for linear power measurement.

The six linear power-level output and the three wide-range log power rate outputs were used as inputs for the protection system. Six separate fission chambers were located in the wide-range log and linear power-range sensor wells and were connected to an averaging circuit to produce the flux controller input signal.

The start-up channels used high-sensitivity ^{10}B-lined chambers. These channels were designed to trip with low count rates to prevent control rod withdrawal without source indication. A high-count-rate scram was also provided for use in the core loading and physics tests.

4.4.3 Failed fuel instrumentation

In Fort St. Vrain, gaseous activity of the primary coolant was monitored with a flow-through ionization chamber to detect failed fuel elements. The failed fuel element was located by physical inspection during refueling operations. Generally, the older and more suspect fuel was carefully inspected. The inspection was extended to newer fuel regions until a failed fuel block was located.

4.4.4 Process instrumentation

Fort St. Vrain included 18 instrument penetrations located at two different elevations in the PCRV. These penetrations contain the connecting piping, tubing, isolation valving, and transducers associated with primary coolant monitoring. A list of penetrations is given In Table 1. The measurements made included primary coolant pressure, core differential pressure, fuel-region outlet temperatures, circulator inlet temperatures, reactor and loop moisture, temperatures of various PCRV internals, and differential pressures across various portions of the circulators. The circulator inlet pressure drop was calibrated to give a measurement of helium flow.

Table 1. Fort St. Vrain PCRV instrument penetrations

Name	Purpose	Quantity
Moisture and process instrument penetration	Reactor and coolant-loop moisture monitors; reactor pressure and core ΔP measurements	6
Outlet coolant thermocouple penetration	Each assembly redundantly monitors outlet temperature from several fuel regions	7
Thermocouple penetration	24 non-replaceable thermocouples	1
Circulator instrument penetration	ΔP transducers for helium circulators	2
Spare instrument penetrations	Possible for future use	2

4.4.5 Temperature

Thermocouples in Fort St. Vrain to measure various core, coolant, and steam generator temperatures within the cavities of the prestressed concrete reactor vessel (PCRV) are of Geminol-P and Geminol-N type. These thermocouples supported service temperatures up to 1093°C. Others within the reactor vessel to measure the circulator and core support structure temperatures were of chromel–constantan type, which were for temperatures up to 538°C. A list of reactor and primary loop thermocouples used in Fort St. Vrain is given in Table 2.

Table 2. Fort St. Vrain reactor and primary circuit thermocouples[1]

Service	Quantity	Type	Remarks
Primary coolant	37	Geminol-P and N	N/A
Reactor internals	24	Geminol-P and N	30-year life; 4×10^{18} n/cm^2
Coolant outlet gas	148	Geminol-P and N	Service to 1093°C
Steam generators	220	Geminol-P and N	Service to 1093°C
Steam sub-header	216	Chromel–Constantan	N/A
Core support floor	35	Chromel–Constantan	Water-cooled concrete structure
Helium circulator bearings	60	Chromel–Constantan	Helium, water, and steam environment

All of the Fort St. Vrain thermocouples used magnesia insulation and Inconel seamless sheath. Lead wire sizes varied from No. 18 to No. 26 AWG depending on the service life and intended use.

4.4.6 Analytical instrumentation

The Fort St. Vrain analytical instrumentation included fluid chemistry and moisture measurement systems and primary coolant gas analysis systems.

4.4.7 Moisture

The Fort St. Vrain moisture detector system used the measurement of dew point of a continuously flowing gas sample that passed through it. The dew-point detector head is shown in cross section in Fig. 9. The primary sensor is a rhodium-plated mirror onto which the sample stream of 20 cm^3/s is directed after passing through a sintered metal inlet filter and fixed orifice. The sample stream is in the temperature range of 66 to 121ºC. A beam of light is also directed at the mirror, and the light reflected and scattered from the mirror is detected by photocells. Fiber-optic light pipes are used as part of the moisture detector to allow the light source and photocells to be placed in an area that is accessible during reactor operation. The mirror temperature is measured with a thermocouple and is controlled by a combination of electric heating and gaseous nitrogen cooling flow. The reflected-light signal is used for moisture detection, and the scattered-light signal measures mirror contamination.

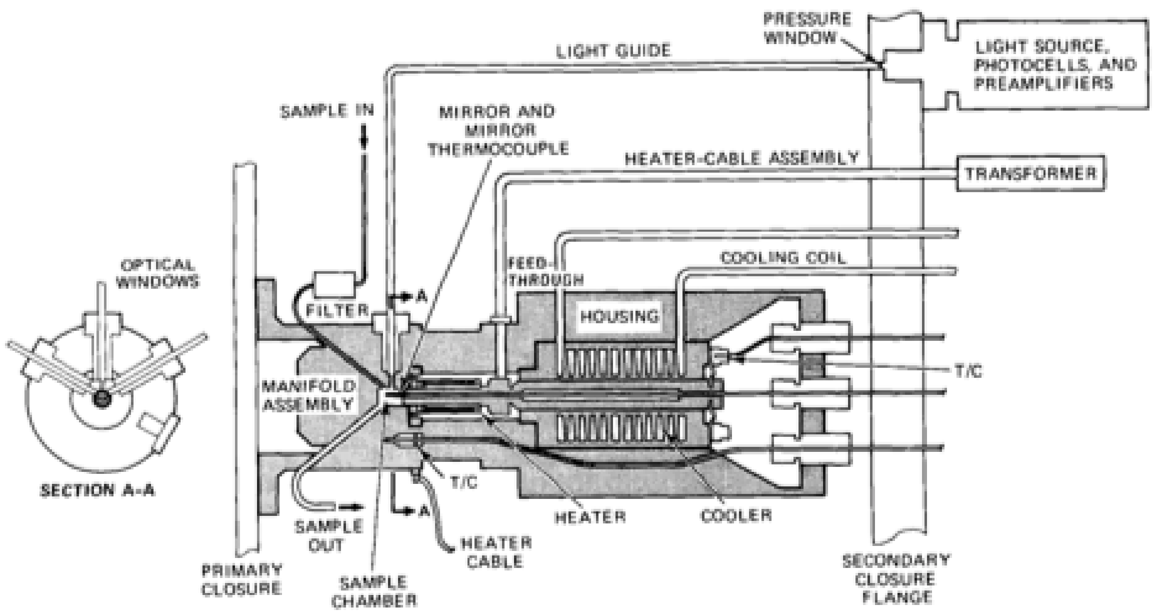


Fig. 9. Simplified cross section of Fort St. Vrain moisture detector.

The system was designed to operate in two modes: (1) indicating mode and (2) trip mode. In the indicating mode of operation, the temperature of the primary sensor surface is varied to the point where condensation is detected. In the trip mode of operation, as shown in Fig. 10, the primary sensor surface is kept fixed at the desired trip temperature and causes a trip when condensation is detected. The trip mode has a shorter time constant since it is not affected by the time delay inherent in changing the sensor temperature to reach a trip condition.

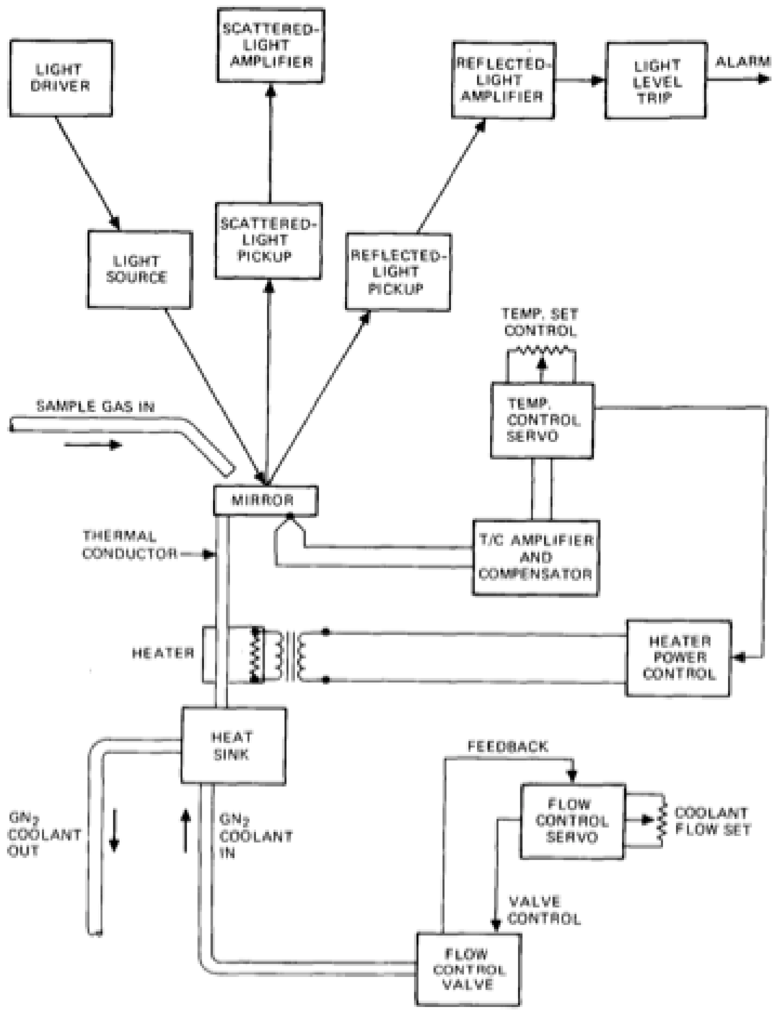


Fig. 10. Fort St. Vrain moisture detection system connected to the trip system.

In the trip mode of operation, dew points are detected from –2.8 to 53ºC, corresponding to moisture concentrations of 100 to 3000 parts per million by volume at approximately 5 MPa. Mirror temperature can be controlled to better than ±0.6ºC. The monitor operating in this mode trips in less than 1 s for a signal 25% above the trip setpoint.

4.4.8 Gas analysis

At Fort St. Vrain, continuous monitoring was required only for CO, for which an infrared analyzer was used. Total gaseous activity of the main coolant was measured, as was the ^{85}Kr activity at the outlet of the helium-purification system, to check for noble gas breakthrough. Gas concentration measurements Table 3 were made for CO, CO_2, N_2, H_2O, and H_2. These measurements were required for regular check on the performance of the purification system and on the graphite reaction rates due to steam ingress. They also provided indication of any air ingress during refueling operations at atmospheric pressure.

Table 3. Gas analysis at Fort St. Vrain

Component	Instrumentation	Sensitivity
Primary coolant radioactivity	NaI(Tl) scintillation crystal	5×10^{-7} to 5×10^{-3} $\mu Ci/cm^3$ for ^{85}Kr
Purification system radioactivity	Plastic scintillator	5×10^{-6} to 5×10^{-1} $\mu Ci/cm^3$ for ^{85}Kr
CO	Infrared analyzer	50 ppm
CO_2	Gas chromatograph	20 ppm
N_2	Gas chromatograph	15 ppm
H_2O	Optical dew point	20 ppm
H_2	Oxidizer bed and dew point	20 ppm

4.4.9 Other Instrumentation

Fort St. Vrain used a prestressed concrete reactor vessel (PCRV). Being the first nuclear installation with a PCRV, the vessel was highly instrumented to support data collection for this new technology and to provide additional monitoring capability.

Reactor vessel instrumentation

A list of sensors used is given in Table 4. The liner strain gauges and the load cells used to measure PCRV tendon strain—typical loading ~4.5 MN—are standard bonded resistance wire type. The Carlson gauges are also resistance-sensitive devices used for various concrete strain measurement applications in the construction industry. The vibrating-wire strain gauges have more than an order of magnitude better resolution—0.1 microstrain, twice the range—3000 microstrains, are about half the size of the Carlson gauge and are ruggedly designed for long life in concrete.

Table 4. Pre-stressed concrete reactor vessel sensing elements at Fort St. Vrain

Element	Quantity
Liner strain gauges	111
Carlson strain gauges	25
Vibrating-wire strain gauges	110
Thermocouples	235
Load cells	27
Moisture detectors	25

4.5 High-Temperature Engineering Test Reactor (HTTR)

The HTTR reactor core is designed to generate 30-MW thermal power and consists of an array of hexagonal graphite fuel assemblies, control rods, and graphite reflectors. The fuel element of the HTTR is ‘pin-in-block type,’ which is made up of fuel rods and a hexagonal graphite block.

The HTTR employs approximately 4000 sensors. The signals from these sensors are centralized by the plant computer. A schematic diagram of the HTTR control system structure is shown in Fig. 11.

Table 5 gives a listing of safety signals for the HTTR Reactor Protection System (RPS) and Engineered Safety Features Actuation System (ESFAS).

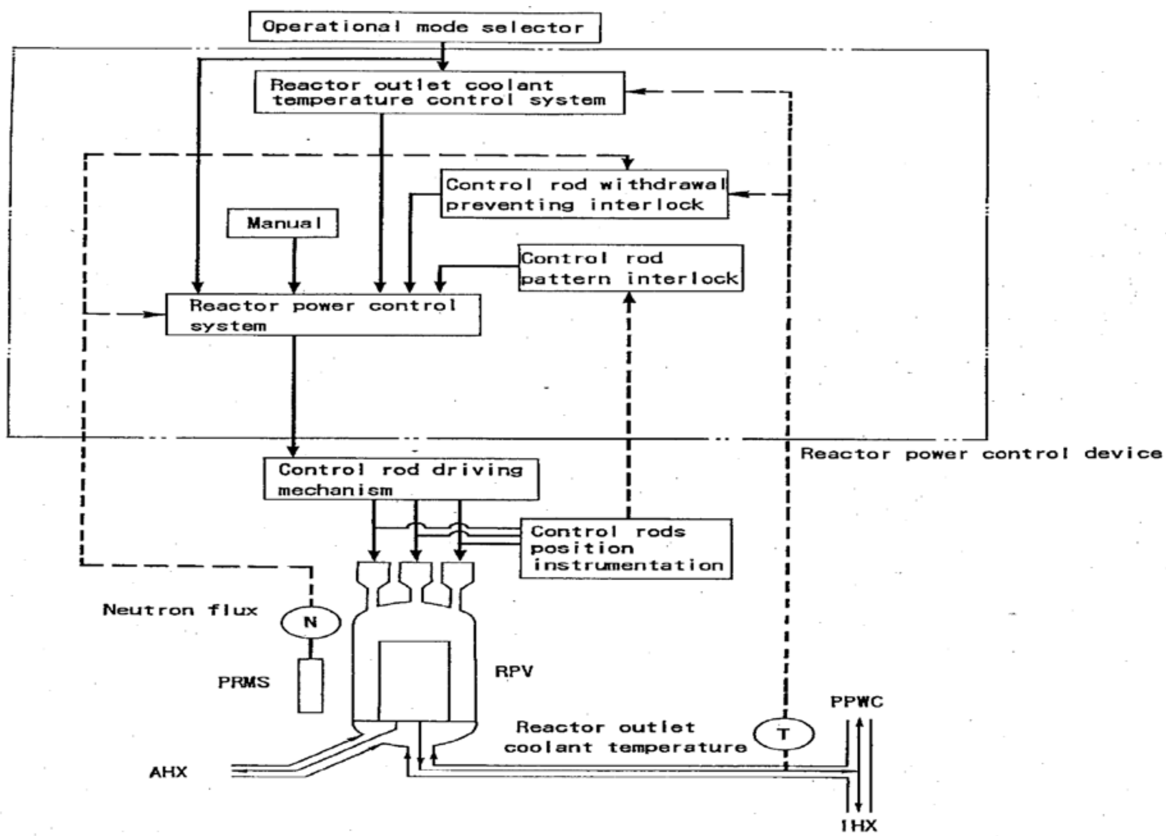


Fig. 11. Schematic diagram of the HTTR reactor control system.

Table 5. Safety signals for the HTTR Reactor Protection System (RPS) and Engineered Safety Features Actuation System (ESFAS)

Reactor scram signals in reactor protection system	
WRMS	High
PRMS	High
IHX primary coolant flow rate	Low
PPWC He flow rate	Low
Primary coolant radioactivity	High
IHX outlet primary coolant temperature	High
Reactor outlet temperature	High
Core differential pressure	Low
PPWC pressurized water flow rate	Low
Primary/pressurized water differential pressure	High
Primary/pressurized water differential pressure	Low
Primary/secondary He differential pressure	Large
Secondary He flow rate	Low
Seismic acceleration	Large
Manual	–

Engineered safety features	Signals	
CV isolation	CV pressure	High
	CV radioactivity	High
	Primary/pressurized water differential pressure	Low
	Primary purification flow rate	High
	SA radioactivity	High
	Manual	–
ACS startup	Reactor scram	–
	Manual	–
Auxiliary water isolation	Primary/auxiliary water differential pressure	Low
	Manual	–

4.5.1 In-core instrumentation

HTTR in-core instrumentation includes neutron detectors, fuel failure detection system, and control rod position instrumentation.

4.5.2 Nuclear instrumentation

Two types of neutron detectors were used in the HTTR: (1) a fission counter that is prepared for the wide-range monitoring system (WRMS) and (2) an uncompensated ionization chamber that is prepared for the power range monitoring system (PRMS), which can detect a low neutron flux level outside of the reactor pressure vessel.

The WRMS operates under a high-temperature environment placed at the top of the permanent reflector. The WRMS is required to be available as a post-accident monitor under accident conditions such as rupture of the primary concentric hot gas duct. For the WRMS, high-temperature fission counter chambers were developed by JAERI. Accelerated irradiation tests were conducted at up to 600ºC at the

JMTR. The WRMS is designed for operation from 10^{-8} to 30% rated power. Three fission chambers are installed in the permanent reflector blocks through the standpipe as shown in Fig. 12. The temperature around the wide-range detector becomes about 600°C, and the neutron flux level around the power range detector becomes about 10^7 n/cm^2/s during the rated power operation of 30 MW.

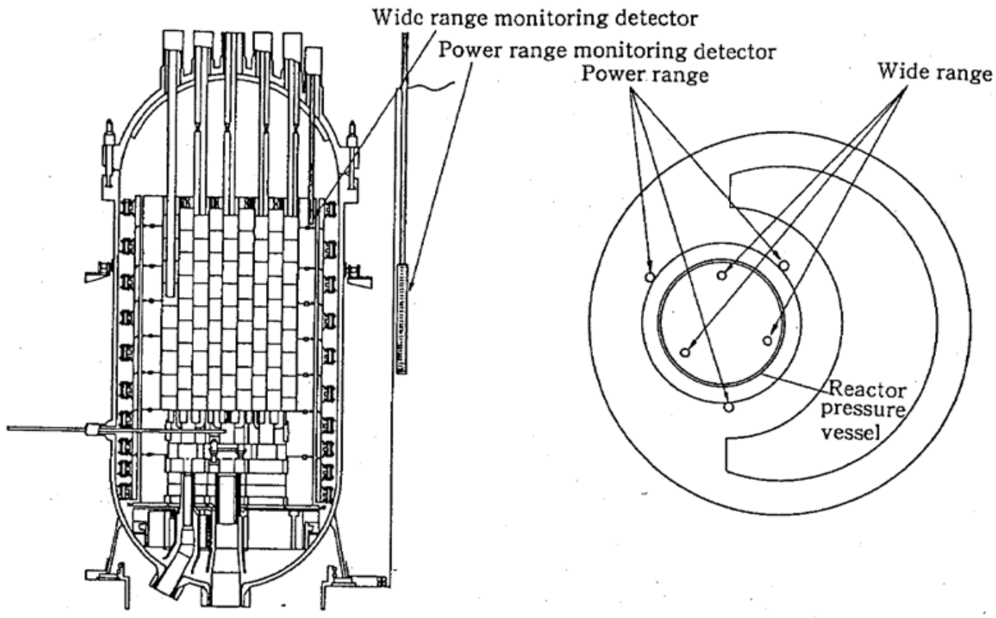


Fig. 12. Detector arrangements for wide-range and power range monitoring systems in HTTR.

The neutron detectors for the WRMS are required to be able to detect neutron flux at 400 and 600ºC for normal operation and during a design-basis accident, respectively. The accelerated irradiation test at 600ºC; long-term, in-core operation test at 600ºC for 1000 days; and overheating test at 800ºC for about 500 h to simulate an accident condition were all carried out. The monitoring system was demonstrated to withstand the test conditions and perform as per specifications.

The signal from WRMS is also used as scram initiator by the reactor protection system (RPS) in the event of a high value.

The PRMS employs high-sensitive gamma-uncompensated ionization chambers that operate in the power range from 0.1 to 120%. The PRMS also drives the control signal for the power control system. The signals from each channel of the PRMS are transferred to three controllers using microprocessors. In the event of a deviation between the process value and the set value, a pair of control rods is inserted or withdrawn at the speed from 1 to 10 mm/s in proportion with the deviation. The relative position of 13 pairs of control rods, except for 3 pairs of control rods used only for scram, are controlled within 20 mm of one another by the control rod pattern interlock to prevent any abnormal power distribution.

The PRMS is located outside the reactor pressure vessel, also shown in Fig. 2, due to increased temperature and neutron flux level within the pressure vessel during full-power operation. The neutron detectors for the PRMS are required to have high sensitivity because of their location outside the RPV where the neutron flux is about 10^7 neutrons/cm^2/s, which is lower than that of LWRs by almost 2 orders of magnitude. In order to improve sensitivity, a ^{3}He counter was developed and used for the PRMS. The sensitivity of the detector was measured to be 4.7×10^{-12} A/nV. Post-irradiation tests showed no noticeable deviation from output linearity.

The PRMS output is also used by the RPS to scram the reactor.

4.5.3 Fuel failure detection system

The fuel failure detection system (Figs. 13 and 14) monitors the primary flow for indication of failure of coated particles. The primary method is detection of short-life fission products, such as ^{88}Kr and ^{138}Xe.

The fuel failure detection system consists of two precipitators, a preamplifier, a compressor, and other supportive components. Helium gas from the primary heat transport system from the seven regions in the hot plenum is transferred to the precipitator chambers. Fission products in the helium gas are collected by the precipitating wire. Collected fission products are transferred to the front of the scintillation detector by wire. The scintillation detector counts beta rays emitted by short-lived gaseous radionuclides.

4.5.4 Temperature

Type-N thermocouples were used to monitor the condition of fuel elements.

4.5.5 Process instrumentation

HTTR process instrumentation utilizes standard temperature, pressure, and flow sensors.

4.5.6 Temperature

Thermocouples used for in-core temperature measurements are required to have long lifetime and high reliability. Performance tests were conducted to monitor the stability of thermo-electromotive force at 1200ºC for about 20,000 h. The test results are shown in Fig. 15. The N-type thermocouple was chosen because the deviation of thermo-electromotive force was found to be small compared to other thermocouples in a high-temperature environment (Ref. 12). The coating materials for the N-type thermocouple were developed to avoid carburization of sheath material by carbide deposits.

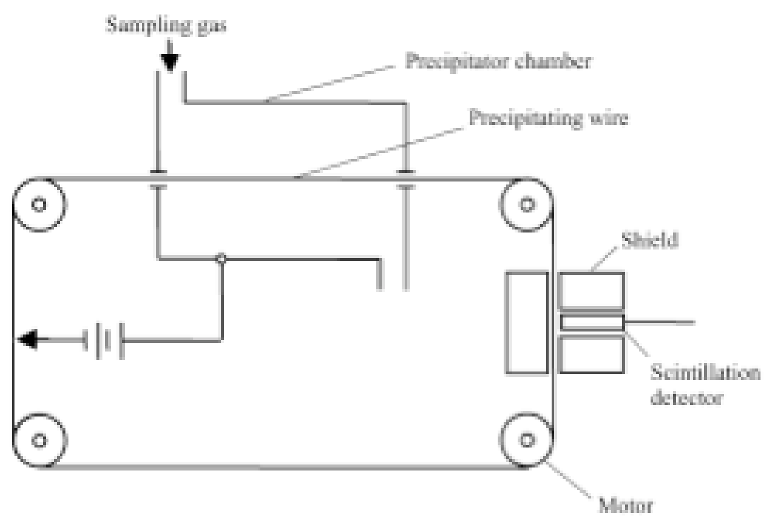


Fig. 13. Schematic drawing of the precipitator for fuel failure detection system.

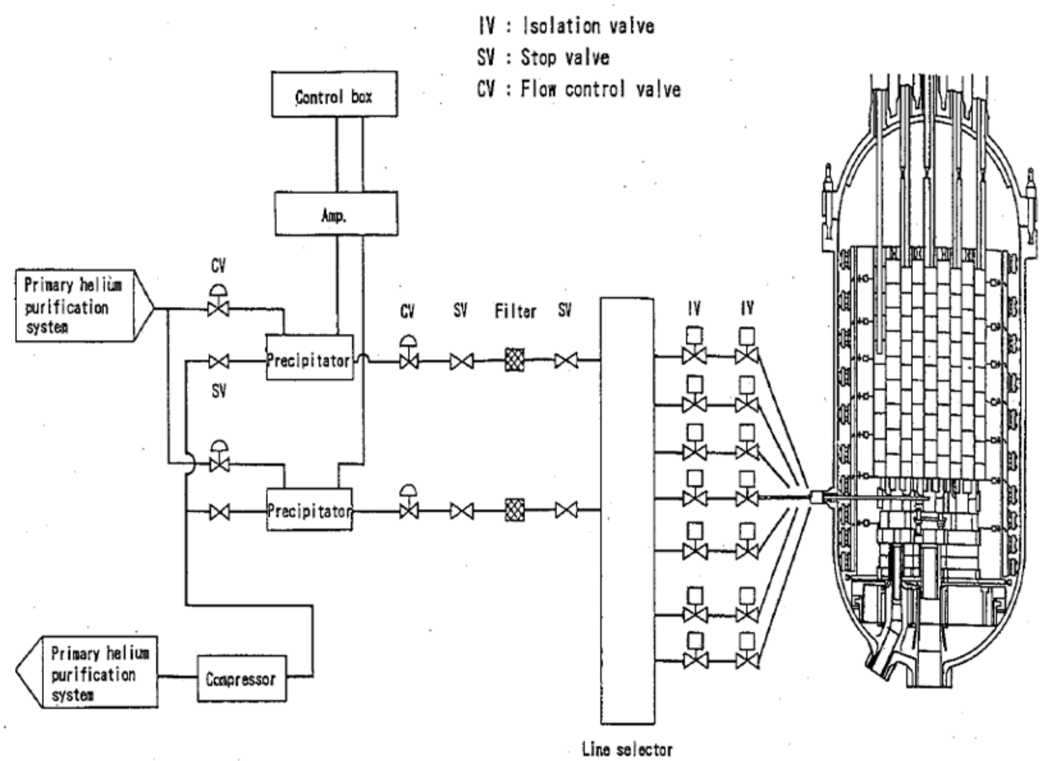


Fig. 14. HTTR fuel failure detection system.

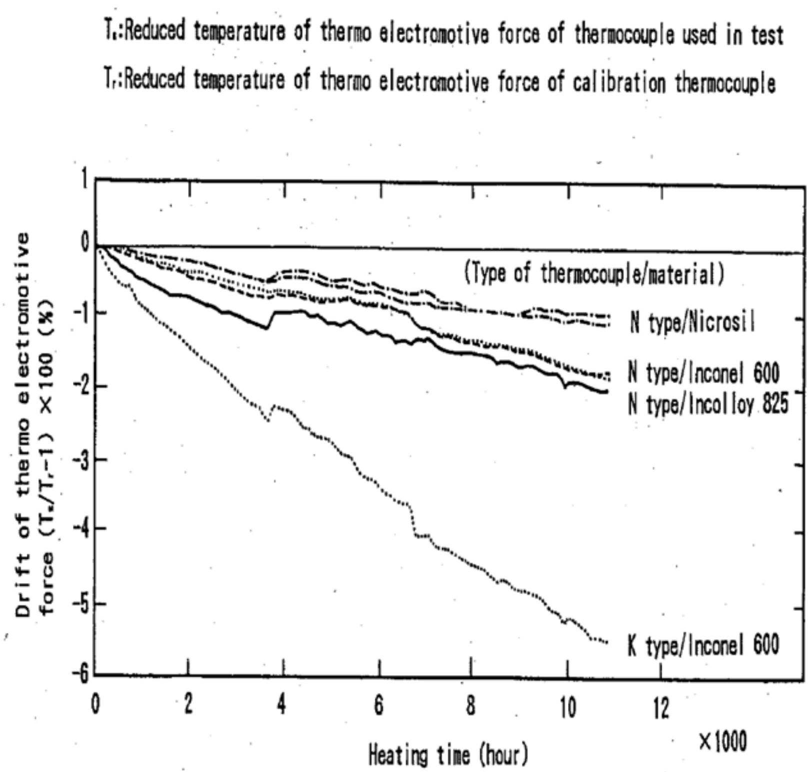


Fig. 15. High-temperature testing of thermocouples used in HTTR.

Four N-type thermocouples were arranged at each hot plenum block to monitor the primary coolant temperature as shown in Fig. 16. The maximum coolant temperature around the thermocouples was calculated to be around 1100ºC.

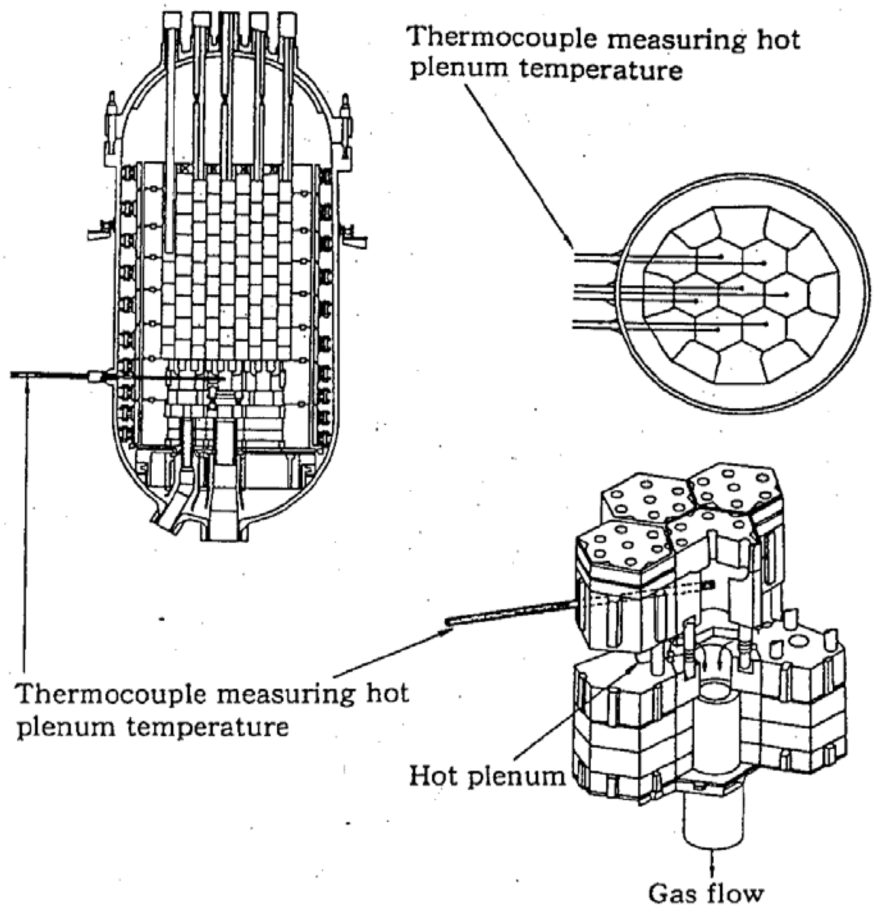


Fig. 16. Thermocouple arrangement for core outlet temperature measurement in HTTR.

4.5.7 Pressure

Primary coolant pressure is monitored by safety-grade Class 1E pressure transmitters.

4.5.8 Flow

The core differential pressure instrumentation measures the change in primary coolant flow through the reactor core between the inlet and the outlet. The signal from this instrumentation is transmitted to the central control room and is used by the safety protection system.

4.5.9 Control rods position instrumentation

The control rods position instrumentation monitors the position of 16 pairs of control rods. The position is measured by the encoder sensor in the control rod drive mechanism, and the signal from this instrumentation is used for the reactor control system and the safety protection system.

4.6 High Temperature Engineering Reactor—10 MW (HTR-10)

The thermal hydraulic instrumentation system of the HTR-10 conforms to the requirements of various international standards such as IEC 60231 (Ref. 12) and the supplementary publication IEC 60231E (Ref. 13). All the safety-related instrumentation used in the HTR-10 conforms to the Class 1E requirements and were fabricated and tested according to the criteria as per IEEE 323 (Ref. 14) and IEEE 344 (Ref. 15). The in-core temperature measurement system and thermocouple penetration assembly conform to the requirements of ASTM E 235 (Ref. 16) and IEC 60772 (Ref. 17), respectively (Ref. 18).

The thermal hydraulic instrumentation system in HTR-10 provides thermal parameters to monitor and control the operation of the reactor. It also provides safety-related thermal parameters to the protection system to initiate protective actions. The system is designed to transmit safety-related parameters prior to, during, and after accident conditions to plant operators for monitoring and control of the plant (Ref. 18). The safety classification of functions and components of the instrumentation and control system is given in Ref. 19.

4.6.1 Temperature

HTR-10 uses Type K (NiCr-NiAl) sheathed thermocouples throughout the high-radiation environment. The outer diameter is 3.17 mm, the shell material is stainless steel 316L, and the isolation material is MgO. The thermocouple measurement error is rated ±1.5°C for temperatures below 375°C, and ±0.4% of the measured temperature for temperatures between 375 and 800°C. Two main types of Class 1E thermocouples used in HTR-10 are shown in Fig. 17.

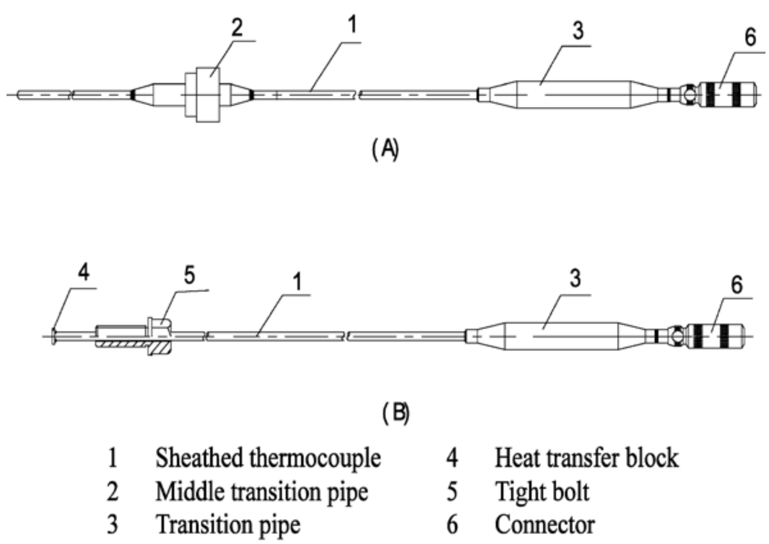


Fig. 17. Two types of Class 1E Type K thermocouples used in HTR-10 (Ref. 18).

In-core component temperature measurement was provided in HTR-10 to monitor the temperature distribution of the core graphite and metal components. Forty temperature measurement locations are made available for this purpose. Thermocouples penetration assemblies were used to prevent helium leak. Each assembly consisted of a penetration body, ten sheathed thermocouples, a middle transition pipe, and a protective tube as shown in Fig. 18. The penetration body is welded to the tube on the reactor pressure vessel. The penetration body, sheathed thermocouples, and whole thermocouple penetration assembly passed the water pressure test at 5.5 MPa for 15 min. The helium leak rate test was also performed and

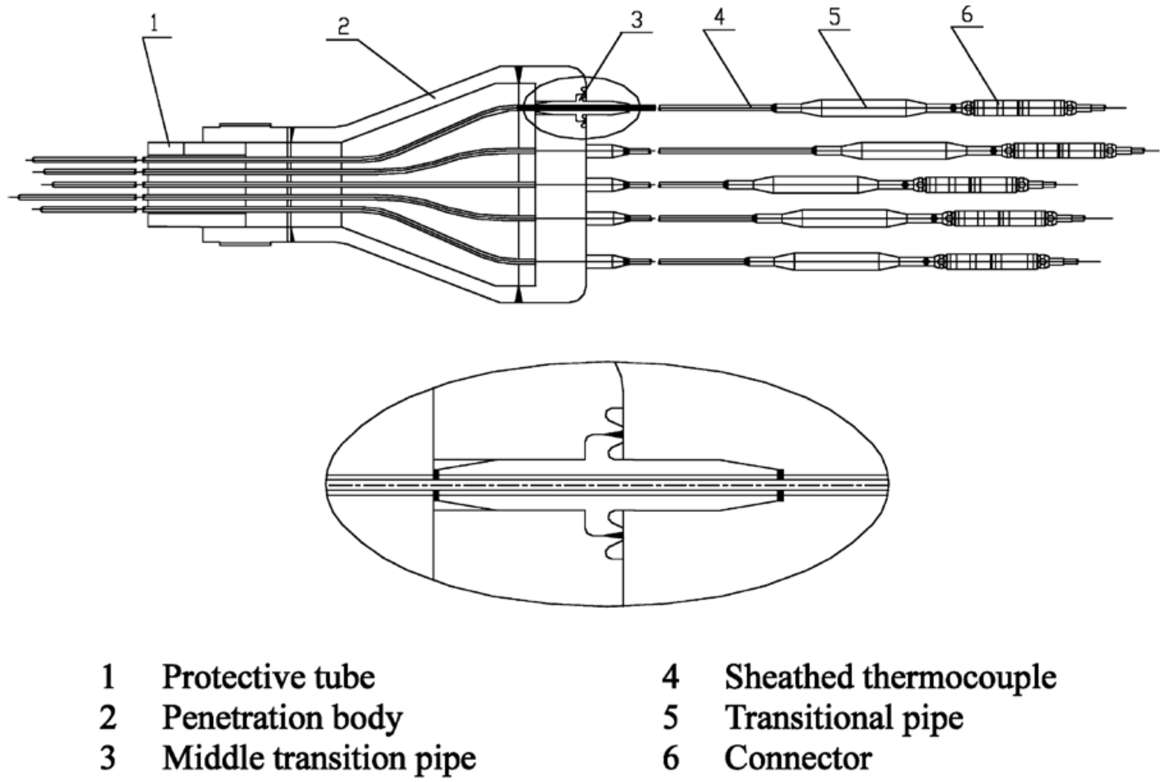


Fig. 18. Thermocouple penetration assembly for pressure vessel to provide in-core temperature measurements.

found satisfactory (Ref. 18). Pressure vessel surface temperature is also monitored to assure that the vessel has not been subjected to temperatures beyond the specification limits. The surface temperature at normal operating conditions is to be below 250°C. During off-normal operating conditions such as anticipated transient, the surface temperature can rise to 350°C for limited periods of time. Fifty-seven thermocouples were installed on the surface, four of which were Class 1E thermocouples. The heads of the surface thermocouples are bolted on the pressure vessel to ensure heat conduction.

4.6.2 Pressure

The primary coolant pressure was measured at the outlet section of the helium blower. Three tubes are connected with three Class 1E pressure transmitters in the confinement (Ref. 18).

Flow

The flow rate in the HTR-10 was not measured directly by ordinary flow meters, primarily due to space limitations. Instead, the flow rate was calculated using the pressure head of the helium blower, helium pressure, and the helium temperature by the protection system. The pressure head of the helium blower was measured by three Class 1E pressure transmitters between the inlet and outlet sections (Ref. 18).

Moisture

The primary coolant humidity was monitored for potential steam generator tube rupture incident. The gas was extracted from the outlet of the helium blower. The hot gas was cooled before it entered the humidity sensor. A valve is provided to protect the sensor from hot gas. The transmitters process the pulse-modulated signals from humidity sensors and transmit the data to the protection system.

The humidity measurement instrumentation consisted of the Hygrotec™ MMY 170 transmitter and DY 75 probe by GE General Eastern (Fig. 19, Ref. 20). The probe combines a planar capacitive gold/aluminum oxide moisture sensor and an RTD temperature sensor in a single package. The temperature signal is used for temperature compensation of the moisture reading as well as for the measurement of the process temperature.

The MMY 170 operates with a 24 V DC power supply and can measure dew point temperatures from –100°C to +20°C with ±2°C accuracy. The DY 75 sensor supports humidity measurements up to 50% at dew points above 0°C. The temperature coefficient is less than 0.2°C/°C over the entire operating range of –20°C to +40°C.

Fig. 19. DY 75 probe that combines a gold/aluminum oxide moisture sensor and an RTD temperature sensor for temperature compensation (Ref. 20).

The safety-related thermal parameters of the secondary loop are steam pressure and feedwater mass flowrate. These measurements are obtained through conventional process measurement devices (Refs. 19 and 21).

Other Instruments

4.6.3 Gamma thermometers

While gamma thermometers have existed in some form since the 1950s (Refs. 22 and 23) and, indeed, the NRC approved their use for local power measurement in PWRs in 1982, gamma thermometers are only now beginning to emerge into widespread use in commercial nuclear power plants. Of particular note, gamma thermometers are currently being proposed for LPRM calibration in the ESBWR. Gamma thermometers, however, remain an emerging technology in that they have not yet achieved widespread, long-term deployment within U.S. commercial nuclear power plants.

Gamma thermometers function based upon the heating of the sensor assembly by gamma rays and the subsequent controlled differential cooling of the sensor body (Fig. 20). The temperature differential developed along the cooling path is proportional to the rate of heating by the incident gamma rays, which is in turn proportional to the local power generation rate during power range operation. An electrical heating element is included within the gamma thermometer to provide an alternate heating source for calibration.

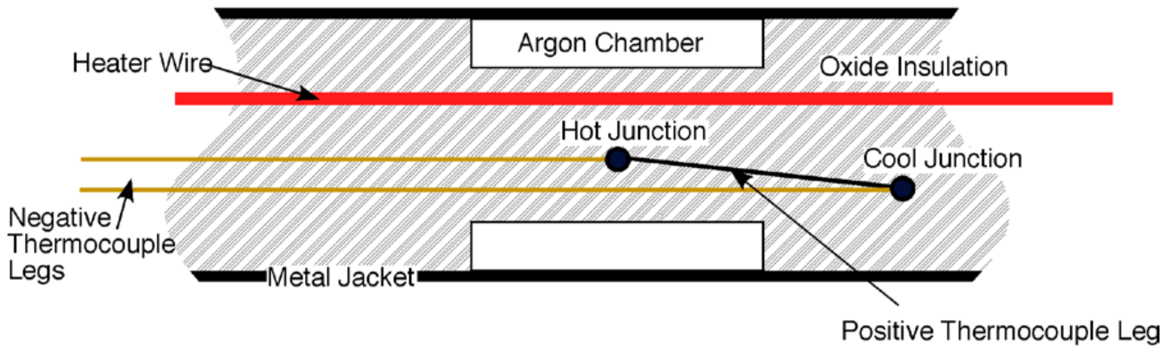


Fig. 20. Gamma thermometer schematic.

4.6.4 Pebble burn-up and damage measurement systems

The on-line refueling feature of pebble bed reactors comes with the requirement that the burn-up of each fuel sphere be measured after each pass through the reactor core. The burn-up value is used to determine whether the pebble has reached its full target burn-up or whether it should be recirculated for another pass through the core. When a pebble reaches its target burn-up, it is routed to a spent fuel tank. The detection system also has the (easier) job of determining if the pebble is “all graphite,” since in the early stages of operation, PBRs typically have a population of moderator pebbles to reduce the average power density of the otherwise fresh-fuel core.

Each discharged sphere is also checked for physical damage. A candidate PBMR system for damage detection automatically diverts (rejects) the sphere if it doesn’t arrive at its destination within a preset time period.

The burn-up measurement and analysis process has relatively little time to decide the fate of each pebble, since (at full power) typically a pebble would need to be discharged every 20–30 seconds. There are several candidate measurement methods. A proposed PBMR burn-up measurement system (Ref. 24) is based on the principle of measuring the concentration of the fission product ^{137}Cs, which has some unique properties that make it an ideal nuclide to measure burn-up. Its burn-up measurement accuracy requirement is to be within 5%.

The PBMR system is a gamma spectroscopic instrumentation system with key subsystems that includes:

- photon collimator to ensure that the photons reaching the detector represent an accurate sample of the photons emitted from the entire fuel sphere,
- high-purity germanium coaxial detector system,
- electromechanical detector cooling system, and
- digital signal processor and burn-up analysis computer.

A PBMR prototype system was developed based on preliminary data obtained in Germany using fuel spheres similar to PBMR fuel spheres. Since no data on actual fission product gamma spectra from high burn-up fuel existed within the PBMR project and no irradiated PBMR fuel spheres were expected for “a few years,” PBMR used irradiated SAFARI reactor fuel elements for validation. The test results confirmed the feasibility of the principle. Further validation tests were planned to demonstrate that the specifications could be met using fuel spheres that emit the gamma spectra representative of that expected in the PBMR and in a geometrical set-up similar to PBMRs.

A potential complication in the use of the ^{137}Cs gamma spectrum is the interference of the intense 658 keV line from ^{197}Nb with the signature 662 keV line from ^{137}Cs. Using Monte Carlo simulations

(Refs. 25 and 26), detailed gamma-ray spectra of pebble bed reactor fuel at various levels of burn-up were calculated. A fuel depletion calculation was performed using the ORIGEN-2.1 code, which yielded the gamma-ray source term as the input to an MCNP-4C simulation that accounted for detection and geometric factors. A means of avoiding the ^{197}Nb interference problem was proposed by using the 1333 keV line of ^{60}Co (introduced as a dopant) and selected ^{137}Cs lines, which are free from spectral interference, thus enhancing the possibility of their utilization as relative burn-up indicators.

An alternative method using neutron emission rate has also been proposed (Ref. 27). The investigation assessed the feasibility of using passive neutron counting techniques to analyze fuel pebbles in real time that provide the speed, accuracy, and range required for the burn-up determination. Numerical simulations of the correlation between passive neutron emission rate of an irradiated pebble and its burn-up level, the detectability of passive neutron emission from an irradiated fuel pebble, and the ability of the detectors to discriminate the gamma interference. The overall conclusion was that there is an acceptable correlation between burn-up and passive neutron emission rate of an irradiated pebble, at least at high-burnup levels, and passive neutron counting could be used to provide the on-line, go/no-go decisions.

5. IMPACTS OF NGNP INSTRUMENTATION ERRORS OR FAILURES ON PLANT SAFETY FOR BOTH NORMAL OPERATION AND POSTULATED ACCIDENT CONDITIONS

5.1 Introduction

Because of the many inherent and passive safety features of the modular HTGR, reactor behavior tends to be “self correcting” for many transients and accidents, so instrumentation and control errors and failures would tend to be less of a safety problem than for “non-inherently safe” plants. For example, safety demonstration tests at HTR-10 and AVR showed how the reactors easily survived loss-of-flow without-scram tests and, for HTR-10, an add-on test even had a simultaneous control rod withdrawal. In all those cases, the reactors shut down automatically due to inherent feedback mechanisms. Studies have concluded that the safety of modular HTGRs is primarily determined by initiating events of very low probability (e.g., structural failures due to rare external events). However, one can postulate a number of cases where erroneous measurements or interpretations could lead to conditions or operator actions resulting in some core damage.

The potential for problems that impact plant safety depends on plant design and, in particular, the proximity of the reactor primary system to components such as a steam generator, intermediate heat exchanger (IHX), or direct cycle gas turbine. For the DOE MHTGR design [circa 1980, 350 MW(t) steam cycle plant], the dominant accident sequence initiator was steam ingress (from a steam generator tube leak or break) (Ref. 28). The core design (pebble bed or prismatic) could also have a bearing on postulated accident sequences, as could the IHX secondary heat transport fluid. The reference design for modular HTGRs is considered (here) to have a confinement, rather than a “sealed” containment building surrounding the primary system.

This report includes descriptions of cases of interest for both normal and accident conditions for HTGRs of potential relevance for DOE’s Next Generation Nuclear Plant (NGNP). A comprehensive study of safety concerns for a full range of reactor conditions and accidents may be found in an IAEA TECDOC reporting on a consultancy that developed safety requirements for modular HTGRs (Ref. 29). At this time, the design features of the NGNP have not been established, and the core may be either a prismatic block design such as the gas turbine modular helium reactor (GT-MHR) or a pebble bed design (such as the PBMR). Versions of these two alternatives are shown in Fig. 21.

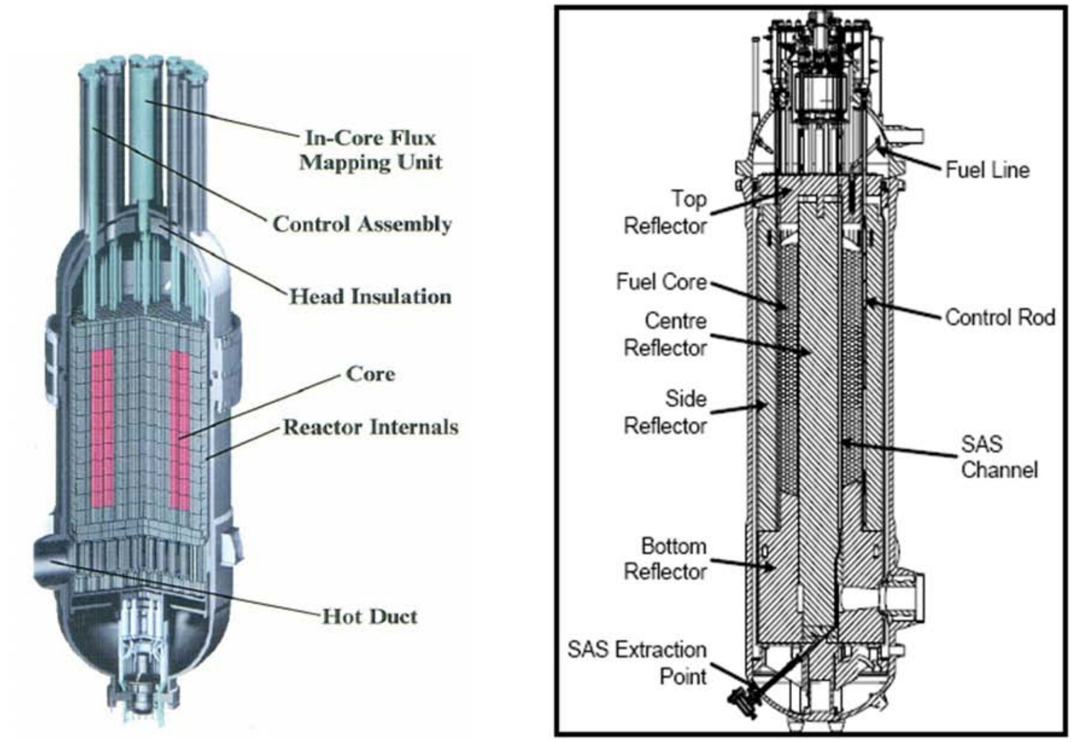


Fig. 21. NGNP reactor core options—prismatic (L) and pebble bed (R).

5.2 Normal Operations

The reactor's three top-level safety considerations are the effective control of heat generation (reactivity), core heat removal, and confinement of radioactive material. This is a general maxim for both normal and accident conditions.

Radioactive material (fission products) first need to be confined within the TRISO fuel particles (the first line of defense), where the barriers in these particles are typically maintained as long as their operating limits (primarily temperature and time-at-temperature constraints) are not exceeded. However, the reactor might be operated with the heat generation and core heat removal well under control ("equal with opposite signs"), but with some of the fuel running too hot or exposed to excessive moisture or other chemical attack, thus releasing radioactive material.

Observations for instrumentation issues under normal operations in this section are summarized in Table 6 following the discussions.

Fuel and coolant temperatures

The core (fuel) operating temperatures normally span a very wide range (e.g., ~400–1200ºC) depending on the design and the fuel's location within the core. However, if fuel in a hotter region is subjected to a combination of higher than expected coolant temperature or power peaking or lower (localized) coolant flow rate, fuel temperature limits may be exceeded. Power peaking can also vary widely, depending on location in the core and burnup. Local coolant flows can vary as functions of bypass paths, localized packing fractions (pebble-bed), and temperature.

Fuel temperatures are not easily measured in prismatic cores and cannot (currently) be measured in pebble-bed cores. Thus, operation within prescribed limits typically relies on design and calculations, as

well as adding margins for error and uncertainties. For new core designs especially, it is advisable to provide as many measurements as practical (e.g., coolant outlet and outlet plenum temperature distributions, and temperatures within the reflectors) to aid in the estimations of actual fuel temperatures. “After-the-fact” measurements can be made using melt-wires (e.g., wires embedded in dummy graphite pebbles), and, in fact, some surprisingly high core temperatures were found this way in the AVR (Germany) (Ref. 30). The final report on AVR operation (Ref. 31) concludes that the melt-wire results were not predicted well because “only 2-D calculations were used, which underestimated the influence of the graphite ‘noses’ in the core on the shutdown rods.” There remains some “heated” controversy about the ultimate safety of PBRs due to the apparent inability to predict maximum fuel temperatures (Refs. 32 and 33).

Measurements of total nuclear power are typically based on a heat balance calculation since neutron level measurements tend to drift long term (i.e., $P = WCp\Delta T$, where P = power, W = mass flow rate of the primary helium, Cp = helium specific heat, and ΔT = mean coolant temperature rise through the core). An accurate reading of mean coolant outlet temperature may be difficult to obtain because of wide local (spatial) variations in core outlet temperatures and imperfect mixing in the outlet plenum. Temperature variations at a downstream measurement point may be fluctuating as well as biased. Backup heat balances (e.g., in the secondary system) are often used to help calibrate primary power calculations.

Primary flow rate and temperature sensor calibrations are, therefore, crucial. At the high coolant outlet temperatures typical of NGNP and VHTR designs (noting also that some local temperatures could be at least 100ºC higher than the mean), thermocouple drift might be a significant problem. (A separate section of the final letter report will cover characteristics of various thermocouple types, thermocouple output drift, and other errors.)

To give an idea of the effect nuclear power measurement errors could have on peak fuel temperature estimates, Figs. 2 and 3 show calculations by the ORNL GRSAC code (Ref. 34) of coolant outlet temperatures that one could expect from a 600 MW(t) prismatic core GT-MHR with a nominal mean outlet temperature of 850ºC and with the reactor at 100% and 110% power. The calculated peak fuel temperatures for these two cases are 1066 and 1124ºC, respectively. Peak fuel temperature increases in the 50 to 60ºC range are typical for 10% increases in power level (or 10% decreases in flow rates).

The outlet temperature distributions shown are for an annular core. A pebble bed NGNP alternative proposed recently has a cylindrical core (and lower rated power). Cylindrical-core PBRs tend to have higher radial power peaking, which would result in larger differences in individual coolant outlet temperatures than those shown in Figs. 22 and 23, thus making it more difficult to obtain reliable and accurate mean temperatures.

Hence, thermal power estimates can be affected by both sensor (thermocouple and flowmeter) errors and difficulties calculating true mean (outlet) temperatures given the wide variations in individual outlet plenum coolant temperatures. Individual measurements of coolant outlet temperatures could be used to help estimate regional fuel temperatures. As a result of all these factors, maximum fuel temperature uncertainties would need to be accounted for in setting requirements for operation with additional margin.

There is another point of interest in the two figures—the large differences between adjacent (local) coolant outlet temperatures. The core support structures would also experience these gradients, which would clearly need to be considered in thermal stress estimates, both for steady state conditions and normal operation transients.

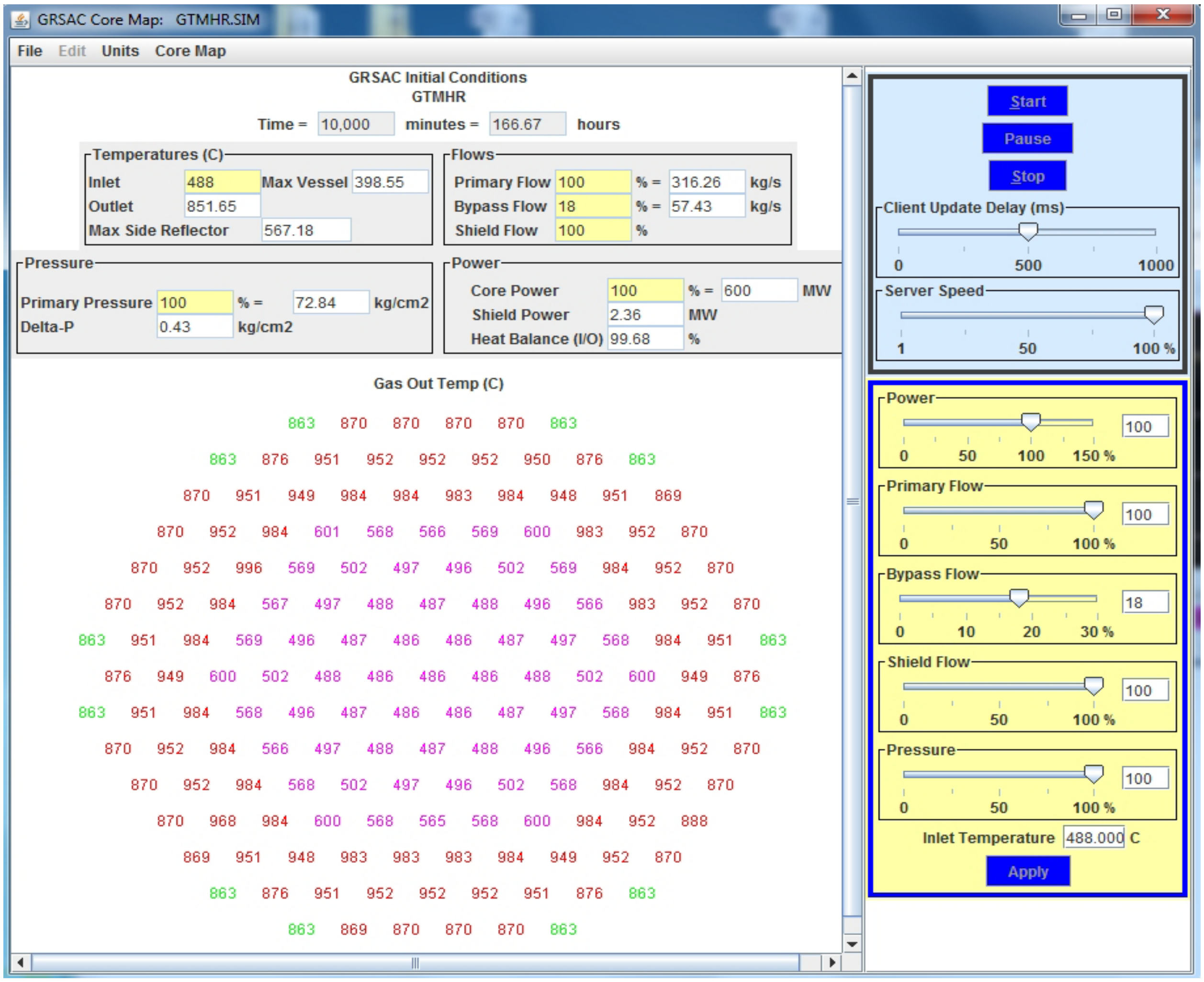


Fig. 22. Map of local coolant outlet temperatures calculated (by the GRSAC code) for a 600 MW(t) GT-MHR operating at 100% power. The lower values of outlet temperatures shown in the center (purple) are from the central reflector, while numbers in the outer three rings are outlets from the active annular core regions. Outer (side) reflector coolant temperatures are calculated but not shown.

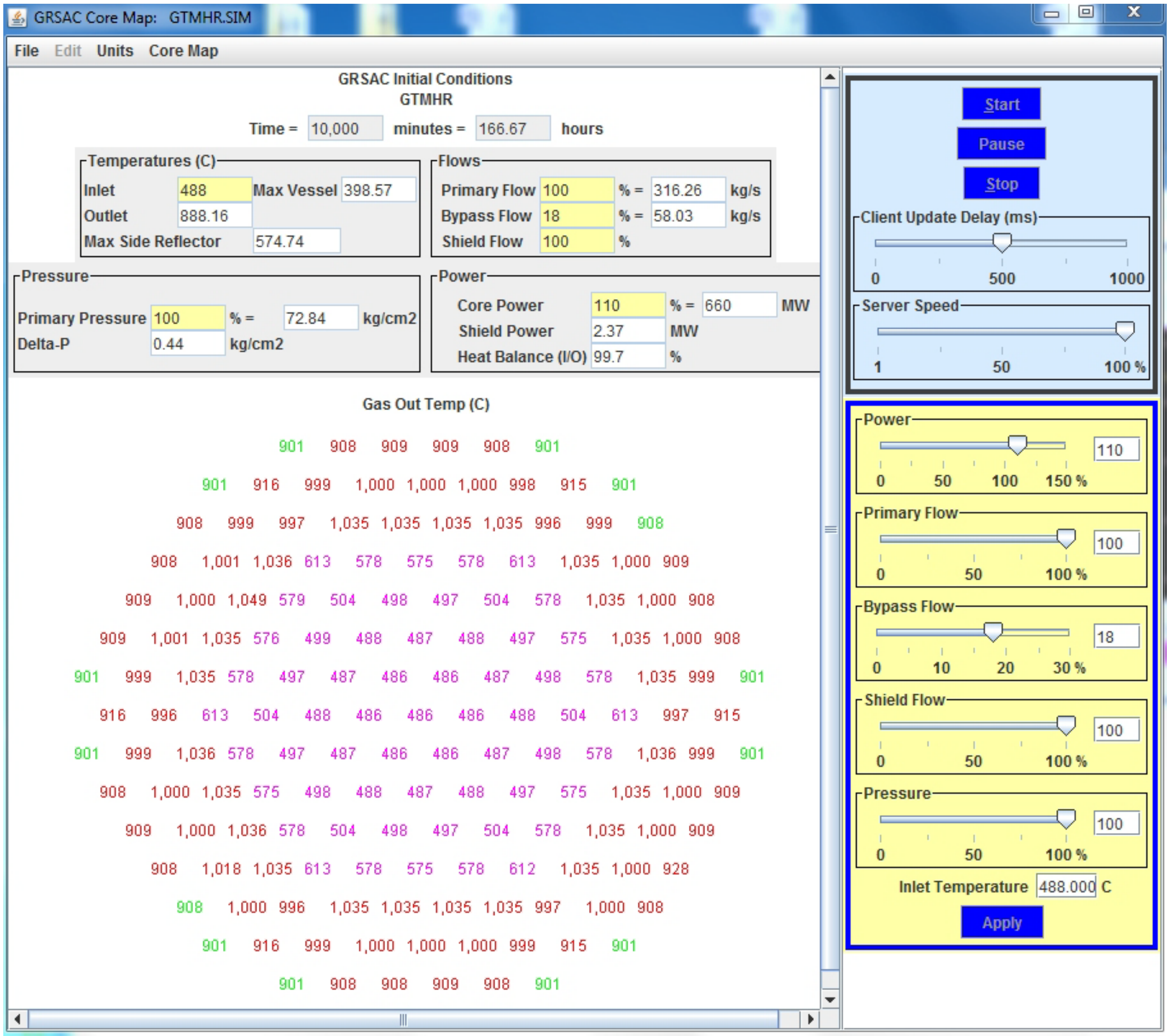


Fig. 23. Coolant outlet temperatures calculated as in Fig. 22 but with the reactor operating at 110% power.

Primary flow rate

In addition to the direct effect that the primary system flow measurement has on the total reactor power calculation (noted above), peak fuel temperatures are also sensitive to the amount of flow bypassing the coolant channels (prismatic) or the fueled section flow path (pebble) in normal operation. Bypass flows are those cooling the reflector regions, in control rod holes, through gaps between fuel blocks (prism) and reflector blocks, and those flows bypassing the core entirely. It is very difficult to estimate, and impossible to measure, the bypass flows since pathway gaps depend on differential expansions, graphite block deformations from temperature gradients and irradiation, lateral pressure differences, and perhaps other factors as well. As an example, the calculated peak fuel temperature in the example shown above (at 100% power) increased by 20ºC when the assumed bypass flow was increased from 18 to 24%. As noted before, such peak temperature uncertainties would need to be accounted for in the operating margin. Figure 24 shows representative changes in the cumulative width of gaps in a prismatic core design as it varies with exposure (and temperature).

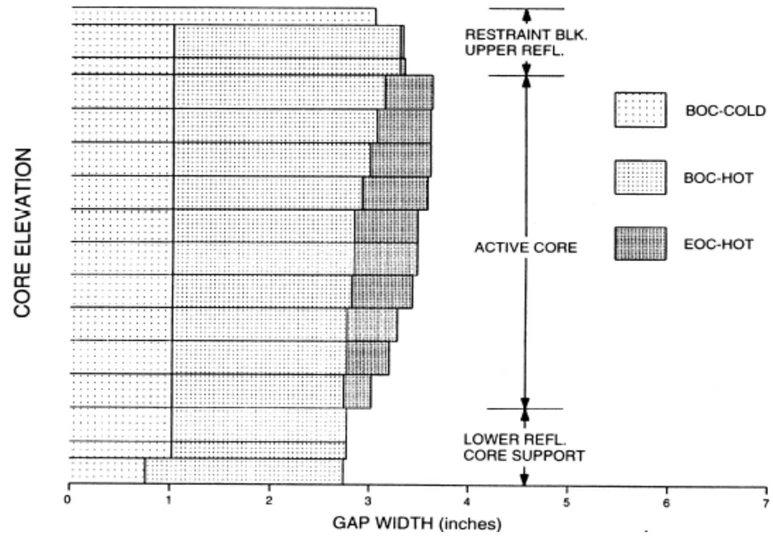


Fig. 24. Prismatic core average gap width at beginning and end of cycle (BOC and EOC).

Besides the effects of bypass flows, the flow nonuniformities in the fueled regions need to be considered. These are due to the changes in flow resistances due to temperature effects: gas viscosity (and, hence, flow resistance) increases with temperature, which results in reducing the cooling in the hotter regions—a positive feedback effect.

Neutron and gamma power level detectors

Neutron power level detectors in HTGRs have typically been located outside the vessel (due to temperature limitations) and used to detect changes in overall power level. Since their output signals tend to drift, they would usually be recalibrated continuously using heat balance measurements. For the very tall cores typical of NGNP designs, a string of neutron detectors spaced axially could be used to detect (and correct) for axial xenon oscillations. Gamma thermometry, as noted previously, might be used to measure localized power levels at very high (nearly in-core) temperatures.

Primary system pressure

Primary pressure measurement is essential for helping to keep track of the helium coolant inventory. A rapid (unplanned) decrease in pressure would indicate a primary system leak or break, and a rapid increase could be signaling a steam ingress event. For direct Brayton cycle designs, helium pressure is routinely varied with power levels (via inventory changes) typically between 20 and 100% power. This is done to maintain high-turbine efficiency by keeping gas velocities nearly constant. In this case, interpretations of pressure variations as possible signs of leaks or steam ingress would need to account for normal pressure variation control.

A failure of the pressure signal (and its interpretation) to detect a depressurization (and scram the reactor) would probably not be a critical safety problem since a core heatup that could result from the event (depending on control actions) would shut down the reactor due to the large negative reactivity-temperature feedback. On the other hand, a failure to detect a steam ingress event (and isolate the water side of the offending steam generator) could be a significant safety problem, although the moisture detection system (see below) should be a backup and able to effect the isolation.

Primary system moisture

Most NGNP designs will have at least one water-cooled heat exchanger as part of the primary system. In shutdown cooling systems (SCS) in prismatic core designs (see Fig. 25); the water pressure in the heat exchanger is typically less than primary helium pressure (except during shutdown). Thus, tube leaks and ruptures would not be likely to introduce much steam/water into the primary. Likewise, this would be the case for the water in precoolers and intercoolers in direct-cycle gas turbine designs. For designs with steam generators in the primary system, however, pressures on the coolant side are typically much greater than helium's. Therefore, steam generator tube leaks (and ruptures) present a much more significant source of accidental moisture ingress, as discussed later in the accident section.

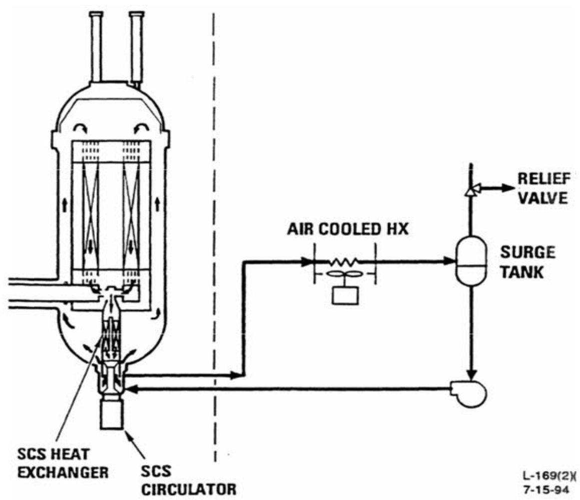


Fig. 25. Example shutdown cooling system (SCS) in a prismatic core modular HTGR.

Some low levels of moisture are likely to be present in the core normally. There is a large amount of moisture initially in the graphite structures that will eventually be removed during operation. Also, a small moisture concentration is desirable to enable an oxygen potential to maintain an oxide film on metallic surfaces. Large, rapid increases in moisture levels would be detected and used to isolate the steam/water side of the steam generator (probably with a corresponding primary pressure signal increase as a backup). Undetected failures of moisture monitoring systems may lead to misinterpretations of otherwise unexplained increases in primary pressure or increases in circulating activity measurements.

Primary system activity

Radioactivity levels in the primary system are monitored continuously. Low levels of releases are expected from releases of tramp uranium embedded in TRISO fuel outer coatings and from modest leakages from defective particles. Any significant upswings in activity could be due to failures in a "bad batch" of fuel or to localized hot spots causing fuel failures and would be a cause for concern that might prompt a reactor shutdown. Some increase in activity could also be due to moisture ingress that reacts with graphite structures, releasing previously absorbed fission products.

Circulating dust in the primary system

A high level of radioactive circulating dust in the primary system is a safety concern, more likely in pebble bed core designs where dust is generated by pebbles rubbing as they pass through the core during the continuous refueling process. Dust releases (in a depressurization) were categorized as one of the major accident sequences for the German 200 MW Modul design. Unexpectedly large quantities of dust were also found in the AVR after operations were completed. Determining the amount, activity, and

physical characteristics of dust that would be discharged in a depressurization is a difficult task. Measurements (during normal operation) that could reasonably characterize the potential source term are yet to be established and are currently the subject of pebble bed reactor safety research efforts.

Vibration (earthquake detectors included)

Quake detectors may be needed in high activity sites. Aside from major quakes that could damage core support and other critical structures, quakes could cause compacting of pebble bed fuel, introducing positive reactivity, and possibly a power surge. Studies have shown such surges to be inconsequential, however, due to the strong negative temperature feedback coefficient for the fuel. Vibrations from other events could cause releases from otherwise stable pockets of dust or edifices with significant fission product plateout. Core overcooling events (during normal operation) that cause positive reactivity insertions (due to the negative reactivity-temperature feedback) have, likewise, been shown (by calculation) to cause benign responses.

Rod control and operator actions

The large negative reactivity-temperature feedback and the wide margin between fuel operating and damage temperatures both help to reduce the consequences of inadvertent rod withdrawals or faulty rod position indications. Reliable position sensors would be essential to prevent inadvertent power peaking. Scenarios would need to be evaluated on a case-by-case basis.

RCCS monitoring

The reactor cavity cooling system (RCCS), the modular HTGR's "ultimate heat sink," is usually classified as a safety-grade system. Typically, it is operating during normal operation as well as (assumed) during all postulated accident sequences. Its operation, at least during accidents, is "passive," not relying on any active systems or operator action to perform any critical functions (at least within a day or so of the accident). RCCS requirements typically include providing adequate cooling to the reactor pressure vessel (RPV) and reactor cavity internals during normal operation and ultimate cooling of the RPV and core during long-term loss of (primary system) cooling accident scenarios. The fact that its proper functioning can (probably) be verified continuously during normal operation implies that it would be able to perform its intended functions during an accident, at least barring any catastrophic external events. Heat balance measurements on the RCCS, from coolant flow and temperature rise, would also be used in the overall reactor heat balance calculations.

Reactor cavity (atmosphere) monitoring

Cavity atmosphere monitoring during normal operation would probably be mainly for helium leak detection (but possibly for other reasons?). Pockets of helium concentration in the higher elevations of the confinement can be hazardous to maintenance personnel.

Pebble bed fuel burnup monitoring

There are a variety of techniques that can be used to determine the extent of burnup in pebbles discharged from the core during operation. The monitor will categorize the pebble's degree of burnup to determine if it can be recycled for another pass through the core or sent to spent fuel storage. In the reactor startup phase, where a large fraction of the core's pebbles are pure graphite, the detector would signal that the dummy fuel ball should go to the graphite bin. In the detector considered for the PBMR, the spectroscopic evaluation system is an integral part of fuel handling and storage system operations. It uses a germanium detector along with spectral analysis signal processing to determine the cesium-137 gamma peak. With that information, it classifies spheres as graphite, used fuel, or spent fuel. It is comprised of three major components—a high-purity germanium (HPGe) detector assembly with positioning table and cryogenic refrigeration system, a digital signal processor (DSP)-based burn-up analysis system, and a photon

collimator. The analyses performed are first a graphite/fuel discrimination and then (if not graphite) a discrimination between low burn-up and high burn-up fuel including the categorization of higher burn-up fuels as either used fuel to be recycled or spent fuel. Given the recycle rates projected at full power operation, the detector would have only about 10 seconds to make a decision and disposition for each pebble.

Fuel manufacturing monitoring

Fuel particle (TRISO) coating integrity provides the first line of defense against release of fission products into the primary system. Consequently adequate instrumentation and control of the manufacturing process (particularly the fluidized bed coating and fuel compacting steps) is essential.

Summary

Results from the qualitative assessment of the safety implications for key measurements during normal operation as discussed in this section are summarized in Table 6. The impact ratings are the relative and "approximate" opinions of the author.

Table 6. Summary of normal operation measurement aberration impacts

Parameter measurement	Comment	Backup measurement?	Safety consideration Impact (H, M, L)*
Fuel temperature	Calculated	None	Fuel failure mechanisms (H)
Coolant outlet temperature[d]	High-temperature drift, fluctuations, bias	Other heat balance calculations	In heat balance power calculation. Operating margin affected (M)
Primary flow rate	Primary cooling	Other heat balance calculations	In heat balance power calculation. Operating margin affected (M)
Neutron,[s,d] power level	Fast response; ex-vessel signal strength	Heat balance	Reactivity transients (M)
Primary pressure	Detect leakage, steam ingress; big variations normal for direct Brayton cycle	Moisture (for steam ingress)	Steam ingress (H) Depressurization (M)
Primary moisture[a,f]	Detect steam ingress; maintain O_2 potential	Pressure (for steam ingress)	Reactivity, graphite corrosion, fuel failure (if exposed) (H)
Primary activity	Liftoff potential	Filters	Primary source term (M)
Circulating dust—primary[a]	Escape potential	Filters	A primary source term—for pebble bed (H)
Pebble burnup on-line monitoring	Decide to recycle or store each discharged pebble	Reactivity balance	Minor effects (L)
Vibration (quake)	Pebble bed compacting	Power level	Minor effects (L)
RCCS performance monitoring[a]	Safety grade, ultimate heat sink	Vessel temperatures	Slow response (L)
Reactor cavity atmosphere	Helium leak	Pressure loss	Minor effects (L)

*Impact: H = high, M = medium, L = low).
Note: Superscript keys for sensor concerns: d = decalibrate/drift potential; f = failure concerns; a = availability (development?) concerns; s = signal strength concerns.

5.3 Postulated Accident Conditions

In this section, the consequences of instrumentation errors or failures during accident progressions are described. Although for modular HTGRs, where the safety case is typically made that operator actions are

not required, it is extremely unlikely that during the hours and days following any of the typically slow-moving events there would be little or no operator action taken. Hence, in some of the sequences, one should consider possible operator responses, which in some cases could make the situation worse, particularly if instrumentation errors or failures provided them with misleading information.

More detailed descriptions of the postulated accident sequences described here are given in other sources including an ORNL /NRC letter report (Ref. 35) and in Section 6 of an IAEA Safety Reports Series report (Ref. 36). These cover the important accident sequences for the reactor but not necessarily those that would involve a yet-to-be-determined heat process connected to an NGNP reactor design.

While the Normal Operation section was organized by "measurement type," this section looks at the major categories of initiating events and sequences of interest, noting the roles of pertinent instrumentation with postulated errors and failure modes plus the possibilities of misinterpretations and subsequent errors in operator response.

Pressurized loss of forced circulation (P-LOFC) accidents

Long-term P-LOFCs may result from a variety of initiating events or event sequences. In fact, in typical HTGR designs, along with a scram signal, the primary helium flow is intentionally stopped by the reactor protection system to avoid a rapid overcooling of the core. Subsequently, the shutdown cooling system (SCS) or equivalent reactor pressure vessel (RPV) cooling system is started up to remove the afterheat. In a P-LOFC, it is assumed that the SCS (or equivalent) fails to provide forced cooling. A station blackout can also result in a long-term LOFC.

In a P-LOFC, the reactor core will heat up due to the decay heat but, because the primary system is still under pressure, natural circulation of helium within the core will help equalize the core temperatures with the maximum temperatures appearing near the top of the core. The maximum fuel temperatures typically remain well below prescribed limits. Some metallic structures, such as the core barrel and RPV, could heat up enough to approach limiting values. In this and other long-term LOFC scenarios, proper operation of the RCCS is essential, in particular, to mitigate adverse heatup effects on the RPV and reactor cavity. In-core and RPV temperature instrumentation, where available, plus primary pressure and RCCS performance calculations during LOFCs, are all essential in monitoring the plant safety status. If primary pressure is maintained, there would be minimal concern for any fission product releases to the confinement building and beyond.

Depressurized loss of forced circulation (D-LOFC) accidents

The major consequence of long-term D-LOFC accidents is the core heat-up and a potential radioactivity release (prompt and delayed) into the confinement and eventually to the environment. Compared to P-LOFC conditions, ambient pressure helium natural circulation in the core is ineffective in averaging out core temperature distributions. Maximum fuel temperatures typically reach peak values in a few days, near the middle or beltline of the core, and then begin a long, slow decrease. In general, the reactor is designed so that the maximum fuel temperature does not exceed an allowable level such that no significant fuel failure is expected. The heat-up of metallic structures also needs to be evaluated. Reactor and RPV temperatures and RCCS performance measurements would be the same as those required for the P-LOFC but with an expectation of significantly higher temperatures.

During the initial depressurization process, some of the circulating activity and other fission products "lifted off" from the primary system components may be released along with the outgoing helium. Depending on the break size, the exiting helium would also take with it some of the radioactive graphite dust residing in the primary system. A measurement of the rate of pressure decrease would be useful in estimating the extent of the releases. The effects of vibration and shocks related to a break also need to be considered in estimating the dust and other releases. Instrumentation characterizing dust contamination and other activity discharged to the cavity and beyond would need periodic monitoring.

Piping breaks and the depressurizations that follow cause pressure distribution transients within the primary system that could be very dynamic for large breaks. Pressure redistributions (especially shock waves) need to be evaluated to ensure structural stability of the system components and reactor internals. It may be necessary to analyze pipe whipping effects on surrounding structures or equipment and to check the potential impact of out-blowing hot helium on safety-related equipment. Pressure and temperature transients at different places inside the reactor confinement need to be evaluated to check the functionality of relevant equipment and structures. Closed-circuit TV monitoring of affected areas would be essential for evaluating damage and devising repair strategies.

The potential for air ingress following a D-LOFC is discussed in a later section.

Anticipated transients without scram (ATWS) accidents

Normally the initiating event for an ATWS event sequence is an anticipated occurrence that calls for a scram (also triggering a loss-of-forced helium flow) followed by a failure to shut down the reactor. There are normally two safety-grade reactivity shutdown systems in a modular HTGR in addition to the control rods. The first shutdown system (rods) actuates automatically on a scram signal, while the second, a backup system typically employing neutron-absorbing balls, is a reserve shutdown system that is actuated manually.

With a loss-of-forced helium flow, for either a P-LOFC or D-LOFC, the average fuel temperature will increase quickly, and the effect of the negative temperature reactivity coefficient will quickly make the reactor subcritical, reducing the fission power to zero. This loss of power causes the maximum fuel temperature to decrease, temporarily, even as the average fuel temperature increases.

Due to the buildup of xenon poisoning, the reactor would typically remain subcritical, at least until the xenon decays (1–2 days). Once the xenon concentration drops sufficiently (and assuming no scram has occurred), the reactor becomes critical again. Then, following recriticality, the increasing reactor power may oscillate and gradually achieve a stable core temperature and low power level that depend on the total residual reactivity and the net core heat removal rate.

Instrumentation requirements for these sequences are those which allow the operators to evaluate the progression of the accident and develop strategies for accident management. Most important would be monitoring of neutron flux indications of recriticality and any progress made to turn it around. Without some primary flow, outlet temperature measurements are not likely to give useful information about actual core (fuel) temperatures, and, likewise, primary activity sensors may not detect fuel failures resulting from over-temperature. The good news in this sequence is that there is a very long time to effect a “scram” or equivalent.

An interesting ATWS variation is one in which, after recriticality, an operator succeeds in restarting the SCS (with still no scram). This added cooling reduces the core nuclear average temperature and, thus, increases reactivity and the power level. However, in the hotter (higher power peaking factor) channels or regions, the convection cooling flows would be lower than average (a higher gas temperature leads to increased viscosity, which leads to a higher friction factor that, in turn, leads to a reduced localized coolant flow). This effect is called “selective under-cooling.” In typical accident simulations where a SCS restart at reduced capacity occurs after recriticality, increases in peak fuel temperatures are predicted over the period of extra “emergency” cooling, significantly adding to, rather than mitigating, fuel failure rates.

Reactivity accident sequences also include pebble bed core compaction during an earthquake (noted in the “during normal operation” section and dismissed as relatively inconsequential), plus reactivity increases due to inadvertent rod withdrawal and reactivity increases due to major steam/water ingress events, discussed in a later section.

An inadvertent rod withdrawal during a LOFC-ATWS accident has been shown to cause only minor and temporary increases in core temperatures, again due to (1) the strong negative reactivity-temperature feedback coefficients and (2) assuming a scram is implemented before the recriticality from xenon decay.

If an “undetected” withdrawal occurs during full power and flow conditions, the core temperature would increase enough to compensate for the added rod reactivity, resulting in a higher power level. This could be detected in several ways, including rapid increase in flux signals and coolant outlet temperatures and probably not result in any fuel damage.

Steam/water ingress accidents

While most postulated modular HTGR accidents are slow moving and relatively benign, large steam/water ingress events are not. Some effects from a steam generator tube leak or break that quickly result in a steam/water ingress into the core could be (1) an increase in primary pressure; (2) an endothermic chemical reaction with the core and support graphite that could release some radioactivity; (3) an increase in (neutron) moderation that could cause an increase in reactivity (and power, if a scram does not occur); and (4) a decrease in control rod worth. In the case of heat exchanger tube leaks from an SCS or Brayton cycle cooler, the likely steam ingress would be minimal due to their lower water-side pressures.

The extent of ingress effects is, of course, very dependent on core design and operational features and the total quantity and flow rate of steam entering the core. The reactivity increase depends on the degree of under-moderation of the core. In any case, the ability of moisture monitors to detect in-leakages (early) and prompt a rapid isolation of the water source could be essential to avoiding serious damage. If the pressure increase were large enough to pop the pressure relief valve, there would be a depressurization event added to the mix and a dispersal of the primary system activity (plus other plated-out fission products that lift off and desorb from the primary graphite and metallic components). If the relief valve fails to re-seat following the depressurization, there would be an open path to continue the releases into the confinement building (and beyond).

For these events, there would likely be scram signals from several sensors— high-fission power, high pressure, and high moisture (and possibly high primary activity and short period). These measurements and trips would all be safety grade, redundant, and likely sufficient to bring the reactor to a safe state. Periodic measurements of building confinement atmosphere activity and chemistry would be essential for accident characterization and for accident management planning. Such accident progressions would be very slow, and using (external) lab sampling would probably be sufficient.

The Fort St. Vrain reactor had 12 steam generators, so the moisture detectors were designed to act quickly to determine which one of the 12 to isolate. In the NGNP designs (ones with a single steam generator in the primary loop), high-moisture detection requirements would be much simpler.

Air ingress accidents

Air ingress into the primary system is a safety concern because of the damage it could cause by oxidizing graphite structures within the vessel and by potential oxidation damage to the fuel (TRISO particles). Although these accidents are typically categorized as very low probability events, beyond design-basis, there is still considerable interest in them due to false perceptions of “graphite burning.” At the operating and accident temperatures in the core following a D-LOFC, however, significant oxidation is possible. The extent of the air ingress flow rates and the oxygen content of the available air depend on a wide variety of reactor and reactor cavity design features, initiating event factors, and details of the subsequent accident progression scenarios. The high core flow resistances and resistances in other parts of potential air ingress flow paths tend to limit convection ingress flow rates to relatively low values.

The major variations in accident scenarios include the sizes and locations of the breaks, which affect initiation time delays and magnitude of the potential air ingress flow rates from natural convection and the oxygen content (vs. time) of the "air" at the intake point.

In the D-LOFC sequence, vessel or other primary system breaches would likely result in blowdown of the helium into the reactor and/or power conversion unit (PCU) cavity, displacing cavity air. The resulting atmosphere for the duration of the potential ingress event depends on confinement system releases from the initial discharge to the atmosphere, outside air ingress after the confinement re-seals, and subsequent leakage to the outside (which would be filtered upon exiting).

Typical scoping calculations of air ingress accidents have indicated that although the oxidation rates for reactor grade graphite are quite low, the oxygen in the entering gas is typically completely consumed, generating heat and forming CO_2 well before the gas exits the core. In fact, at least for the first several days of any significant ingress, analyses typically show it to be mostly consumed in the lower support blocks and lower reflector with relatively little oxygen reaching the active core (Ref. 37). Also, any heat released from oxidation in the active core typically affects only the lower regions and does not add to the peak fuel temperatures that would be reached in non-air-ingress D-LOFC accidents. CO_2 entering the mid-to-upper part of the core is likely to encounter higher temperatures that can cause an endothermic (Boudouard) reaction with core graphite and produce CO. Sustained large-scale oxidation of the core graphite could cause structural damage and expose fuel to potential chemical attack. Structural damage from oxidation in the lower-temperature (chemical) range is substantially greater for a given oxidation weight loss than for the higher temperature (mass transfer limited) range, where most loss from oxidation is near the surface. Periodic analyses of confinement gas chemistry and activity would be especially crucial during the course of such accidents to help assess the damage occurring in the reactor and to help determine appropriate remedial action strategies. Due to the very slow progression of these types of accidents, the analyses could probably be done via periodic sampling and lab test results.

If the oxygen in the confinement volumes could be limited to approximately that initially present in the confinement space (and less, considering air initially displaced by the helium), scoping calculations have shown that the total damage to the core would probably not be significant. Analyses of other scenarios have also shown that if the onset of significant oxidation occurs late in the D-LOFC accident sequence (e.g., several days), the lower core regions may have cooled down to temperatures below the range of significant oxidation rates, in which case oxidation would be more likely to occur further up the core and into the fueled regions. Some Japanese studies have shown that TRISO SiC fuel still retains fission products even when the outer pyrolytic carbon layer is oxidized in those temperature ranges (below ~1300–1400ºC) that are predicted for the lower core regions (Ref. 38).

Another potentially significant accident sequence occurs when a SCS is operated following depressurization, in which case forced circulation flow rates (possibly containing air) could be much greater than in the natural circulation flow cases. Primary coolant activity and temperature instrumentation, along with heat balance anomalies, should provide proper guidance to the operators for such situations. There are a number of cases in reactor (and other process) accident histories where the operator response to an accident has proven to be counterintuitive, such as the Chernobyl accident and the Windscale reactor fire in 1957.

Because of the likelihood of very limited access to critical areas during air ingress accidents, provisions should be made for ad hoc measurements of gas chemistry within the confinement areas and possibly for injection of inert gas to reduce oxidation rates. Meaningful sampling of the confinement atmosphere would be challenging because of the likely wide variation of gas compositions within the space (e.g., helium would rise to the top areas). Confinement gas analyses should be designed to determine the extent of oxidation damage of the core and support structures, the amount of radioactivity potentially available for (filtered?) release to the environs, and compositions that may result in explosive mixtures.

Summary

Some of the accidents described are quite complex. Due to the reactor core geometry, it would be especially difficult to make direct measurements to assess the conditions and progressions of beyond-design basis accidents. It appears that a useful tool to help with such assessments and accident mitigation planning would be an "on-line simulator" tied in to the reactor's I&C to help operators evaluate the scenarios and predict outcomes for candidate mitigation strategies.

Results of this section are summarized in Table 7. As before, the impact ratings are the relative and "approximate" opinions of the author.

Table 7. Summary of accident scenario measurement aberration impacts

Parameter measurement	Comment	Backup measurement	Safety consideration impact (H, M, L)*
Fuel temperature	Calculated	None	Fuel failure mechanism (H)
Coolant outlet temperature[d]	High-temperature drift, fluctuations, bias	Heat balance, overpower detection	In heat balance power calculation (L)
Primary flow rate	Scram on low flow (adaptations for direct Brayton cycle)	Pressure	Trip signal (L)
Neutron[s,d], power level	Fast response; ex-vessel signal strength	Heat balance	Reactivity transients (L)
Primary pressure	Detect D-LOFC, steam ingress	Moisture (for steam ingress)	Steam ingress (M) Depressurization (L)
Primary moisture[a,f]	Detect steam ingress for major leaks	Pressure (for steam ingress)	Reactivity, graphite corrosion, fuel failure (M)
Primary activity	Liftoff potential D-LOFC	Filters	Primary source term (H)
Circulating dust—primary[a]	Escape potential—for D-LOFC events	Filters	A primary source term—for pebble bed (H)
Vibration (quake)	Pebble bed compacting	Power level	Minor effects (L)
RCCS performance monitoring[a]	Safety grade, needed for all LOFC events	Vessel temperatures	Heat balance, slow response (M)
Reactor cavity atmosphere chemistry and activity	Air and steam ingress effects; accident mitigation planning	None	Major needs for very unlikely scenarios (H)

*Impact: H = high, M = medium, L = low.
Superscript keys for sensor concerns: d = decalibrate/drift potential; f = failure concerns; a = availability (development?) concerns; s = signal strength concerns.

6. SUMMARY OF MAJOR INSTRUMENTATION ISSUES

6.1 The Dust Dilemma (mainly for PBRs)

The major problem—from a safety and licensing standpoint—is mainly the uncertainty in how much contaminated dust would be ejected into the confinement (and subsequently into the environs) in postulated rapid depressurization accidents. The reference confinement building designs typically allow for a rapid discharge of the primary helium (from a large break) to exit the building unfiltered. Any subsequent or small-leak (low-flow) discharges would pass through filters before exiting the building. The safety case here postulates that releases of both the (primary system) circulating dust-borne activity

and any additional contaminated dust "stirred up" by the event (due to velocity changes or vibrations) would result in a minimal offsite dose. In addition to dust-borne contamination there may also be releases of other forms of circulating activity and from lift-off of plated-out fission products.

From an instrumentation standpoint, there would need to be on-line means for (semi-) continuous detection of circulating dust concentrations in the primary system. Measurements done in AVR (see Sect. 4.2.2) successfully showed dust concentrations and variations in circulating dust with blower speed changes (Fig. 26, Ref. 4).

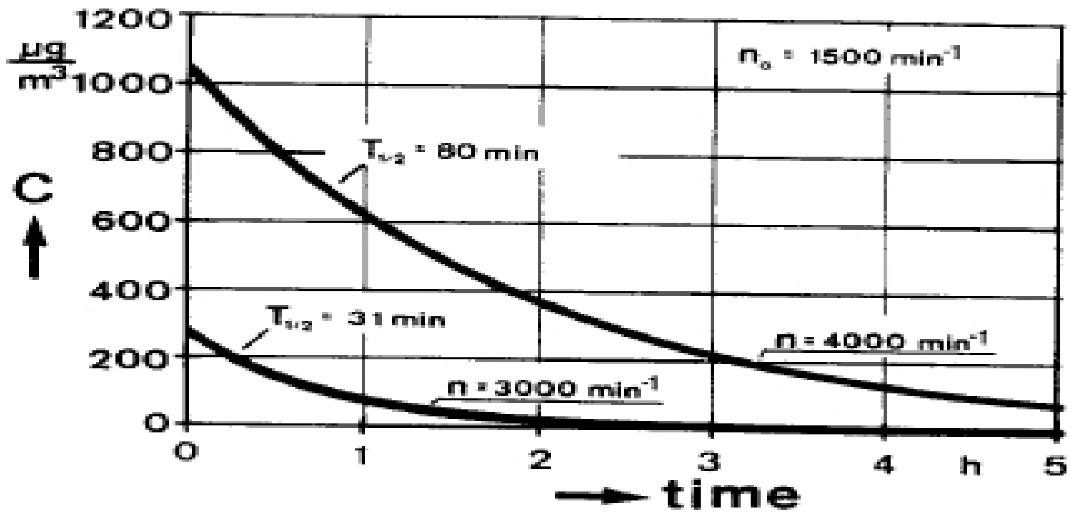


Fig. 10. Curves of dust concentration in the primary gas after a flow increase caused by a blower transient.

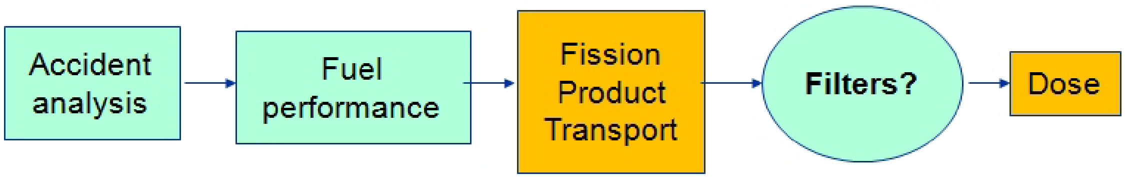
Fig. 26. Fig. 10 from Ref. 4.

Dust experiments are being designed and conducted for the NGNP project and for other foreign projects. Detector designs for eventual use in the reactor could best be tested for accuracy and reliability if used in the experiments. If the NGNP design included a direct Brayton cycle (gas turbine), variations in dust circulation might be detectable with the changes in helium mass flow rates (via inventory changes) that would normally accompany power level changes.

A proposed (partial) solution for fission product release capture during postulated rapid depressurization accidents (with a confinement building) has been suggested by R. N. Morris (ORNL). This involves the use of large, relatively inexpensive filters that would capture any rapid initial discharges from a depressurization accident (see Figs.27 and 28 for clarification). This approach takes advantage of recent major improvements in filter technology and would reduce the uncertainties in EAB dose estimates. The system would be applicable to both prismatic and pebble bed designs. The South African PBMR test facilities HTF (Helium Test Facility) and/or HTTF (Heat Transfer Test Facility) may be good candidates for conducting dust capture experiments. Experiments are amenable to small-scale (modular) testing. Instrumentation requirements for the tests include, at a minimum, rapid response temperature, flow, and particulate concentrations at the inlet and outlet. Other measurements of fission product absorption could be made after the transient tests to determine absorption profiles within the filters. Such tests could be part of an instrument development program for a prototype system.

An answer to the "Early Release" problem for Modular HTGRs Without Leak-Tight Containment Building, Vented for Large Breaks

Design focus is typically on steps 1 & 2 to prevent FP release:



- FP releases quite low even for most serious accidents; however
- Large error bars (especially step #3–FPT) = major licensing concern
  - Including "dust releases" for pebble bed core designs (large breaks)
- FPT experiments: complex, costly, typically under-funded, long time frame, inconclusive

- ***Bob Morris (ORNL) suggestion***
  - ***Consider filtering (step #4), a well-understood technology, as a possible cost-effective alternative to additional FPT experiments***
    - *Review state-of-the-art for high-flow filter systems*
    - *Re-examine trade-offs in light of R&D costs and long schedules*

Fig. 27. Clarification of large filter proposed solution (part 1).

A tight building with high-flow passive filters may make the FPT task more tractable

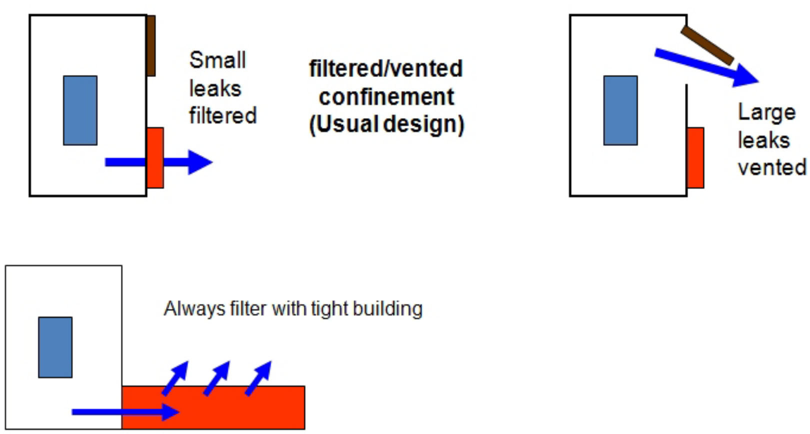


Fig. 28. Clarification of large filter proposed solution (part 2).

6.2 The Maximum Pebble Temperature Dilemma (Due to the Melt-Wire Mystery from AVR)

Unexpectedly high temperatures of AVR fuel (detected via melt-wire pebbles) have cast doubt upon PBR designers' ability to predict maximum pebble fuel temperatures during operation (see Moormann and Koster articles—Refs. 32 and 33). In addition to refined calculations, PBMR instrumentation plans should clearly include provisions for melt-wire pebbles.

6.3 RCCS Heat Balance (To Calculate Heat Loss)

In addition to the more conventional primary system measurements needed to calculate gross thermal nuclear power generation (flow and temperature rise), the power dissipated by the "passive" RCCS is relatively more important in modular HTGRs than in other reactors because it is a greater-than-usual percentage of the total power generated (by design). Usual means of getting power estimates from secondary side measurements would not account for RCCS power. The degree of difficulty in RCCS power measurements depends greatly on the specifics of the design. Designs that rely on natural circulation flows and relatively large spatial coolant flow cross-sections would make it difficult to measure accurate mean flow rates and temperatures.

6.4 Other Important (but More Conventional) Instrumentation Issues

These are addressed in Sects. 4 and 5 and include:

a. thermocouple drift, failure, measurement of non-mean values, and fluctuations;
b. helium mass flow (to get heat balance);
c. neutron power (from detectors outside the vessel); and
d. primary system moisture.

As noted previously, the instrumentation involved in TRISO fuel manufacture, crucial to the safety case's "First line of defense," is discussed in Sect. 7.

7. TRISO FUEL MANUFACTURING I&C NEEDS

[Extracted from NUREG/CR-6844, Vol. 1, *TRISO-Coated Particle Fuel PIRTs for Fission Product Transport Due to Manufacturing, Operations, and Accidents* (main report), July 2004, by R. N. Morris (ORNL) et al.]

The purpose of this letter report section is to establish the case that instrumentation (and control) of the TRISO fuel manufacturing process deserves licensing scrutiny at a level comparable to that given I&C systems involved in reactor operation, although not covered as part of N6668.

Table 8 (Table 2-16 from NUREG/CR-6844, Vol. 1) summarizes the major TRISO fuel manufacturing process issues, as identified by the PIRT panel members, and Fig. 29 (Fig. 2-17 from NUREG/CR-6844, Vol. 1) shows the steps involved in a continuous coating process.

Table 8. Manufacturing process phenomena identified by the PIRT panel
(Table 2-16 taken from NUREG/CR-6844, Vol. 1)

Manufacturing factor	Rationale
Layer coating process specifications: Gases (levitation gas and coating gas)	The gases used in the coater directly influence the quality of the layer and the operation of the coater.
Layer coating process specifications: Ratio of gases	The gas mixtures affect the layer properties and production rate.
Layer coating process specifications: Temperature	The properties of the coating layer are dependent on the coater temperature.
Layer coating process specifications: Coating rate	The microstructure of the coating layer is influenced by the coating rate.
Layer coating process specifications: Pressure	Pressure affects reaction rates. (The coaters are generally operated at atmospheric pressure.)
Layer coating process specifications: Coater size	Coater size affects the distribution of layer properties.
Layer coating process	Continuous versus interrupted coating may affect coating layer interface properties.
Process control	Controlling the process is important. Coating product measurements may not be sufficient to guarantee good irradiation performance.
Product control	Coatings must meet designer specifications.

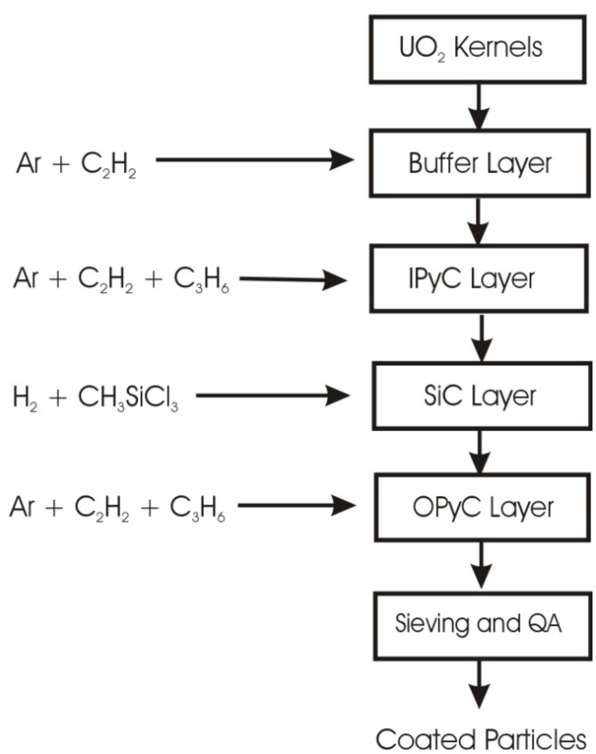


Fig. 29. Flow diagram of the continuous coating process. (Figure 2-17 taken from NUREG/CR-6844, Vol. 1)

Coated Particles

The coating layers are deposited on the kernel in a fluidized bed by the thermal cracking of the appropriate gas in a fluidizing gas such as argon. Hydrocarbon gases such as acetylene and propylene are used for the carbon layers. MTS is used for the SiC layer, and it is reduced by hydrogen. Temperatures are in the range of 1200 to 1500ºC and the flow rates of the gases are adjusted to achieve the desired deposition rate.

Layer properties are controlled by temperature, coating rate, coating gas composition, bed loading, and particle size. In general, each layer has its own optimal combination of parameters that are determined experimentally for a particular coater. A flow diagram of the process is shown in Fig. 29 (Fig. 21-7 from NUREG/CR-6844, Vol. 1). Note that the process may be continuous or interrupted. In the continuous process, the particles remain in the coater and the composition of the gases and furnace temperature is changed so the coatings can be put on one after another. In the interrupted process, the coater is unloaded after each coating and the particles can be checked and sorted for gross defects such as out-of-roundness. Sampling can be used for destructive investigation. The bad particles (or a bad batch) are discarded before the next layer is applied.

At the present time, the continuous coating method has been demonstrated to give acceptable results, but this conclusion is still tentative. The current trend is toward continuous coating and the highest quality fuel (reference material) has been produced by this method.

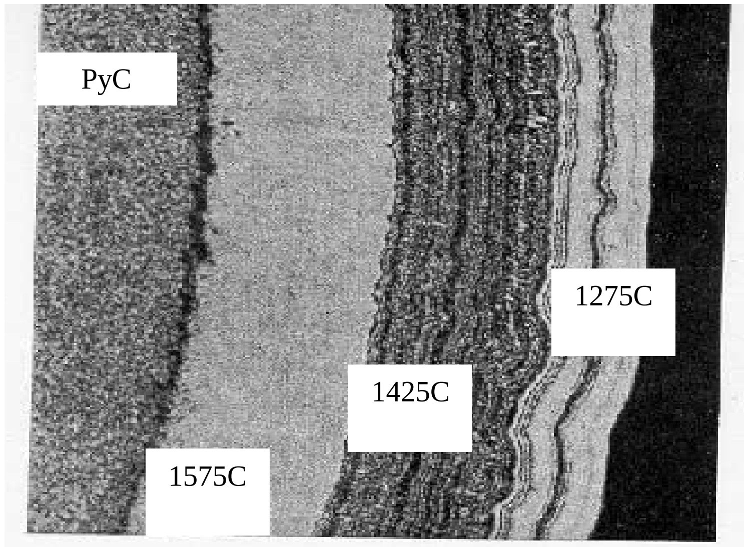
An item of interest from the PIRT review is that the specification of layer product properties is not sufficient to ensure satisfactory irradiation performance. At the present state of the art, modest changes in the operation of the coater (such as design issues, coater size, and exactly where the bed temperature is measured) can lead to coating property changes that could result in substandard irradiation performance. These changes either cannot be observed by the present QC methods, or the changes in material properties are not currently measured.

This is addressed by having both layer fabrication process and layer product specification. Thus, both process knowledge and product measurements are required to determine if the fuel has been properly fabricated. This issue appears to be particularly important for pyrocarbon. The Bacon Anisotropy Factor (BAF) measurement technique is also important. The SiC layer is very important for the control of fission product transport, and it is sensitive to the details of the fluidized bed coater operation.

Figure 30 (Fig. 2-18 from NUREG/CR-6844, Vol. 1) shows how the nature of deposited SiC can change with temperature. The fabricator would like to control free silicon, grain size, and grain orientation. SiC has shown good irradiation properties, but like pyrocarbon, a clear one-to-one correlation between measured properties and irradiation behavior is not available at present.

Table 9 (Table 2-19 from NUREG/CR-6844, Vol.1) shows the coating layer product properties measured during fabrication and the measurement methods that are typically employed. The reader is cautioned that measurements alone do not provide a complete picture of the fabrication parameters and must be used in conjunction with process knowledge.

In summary, precise and reliable measurements and control processes during the coating processes in the fluidized bed coaters are crucial to the “success” of the fuel in terms of its performance (retention of fission products) during normal reactor operation and postulated accident conditions.



Etched SiC
ORNL/TM-5152

Fig. 30. SiC structure as a function of coater temperature.
(Figure 2-18 taken from NUREG/CR-6844, Vol. 1)

Table 9. Coating layer product factors and typical QC methods
(Table 2-19 taken from NUREG/CR-6844, Vol. 1)

Layer attribute	QC method
Buffer Layer	
Thickness	Radiography, metallography
Density	Mercury pycnometry and carbon content analysis (LECO)
Missing or thin layer (a failure mechanism)	Radiography
IPyC Layer	
Thickness	Radiography, metallography
Density	Liquid gradient column
Anisotropy	BAF (other methods under study)
Microstructure	Coating rate and process conditions (temperature, coating gases, time) [Process Knowledge]
Permeability (the heavy metal dispersion will signal a missing layer)	Heavy metal dispersion into layers (Radiography, chemical analysis)
SiC Layer	
Thickness	Radiography, metallography
Density	Liquid gradient column

Table 9. (continued)

Layer attribute	QC method
SiC Layer (continued)	
Microstructure	Coating rate and process conditions (temperature, coating gases, time) [Process Knowledge], metallography
Spatial defects or missing layer	Burn/leach
Strength	Crush tests, brittle ring tests
OPyC Layer	
Thickness	Radiography, metallography
Density	Coating weight and pycnometry
Anisotropy	BAF
Microstructure	Coating rate and process conditions (temperature, coating gases, time) [Process Knowledge]
Missing or defective layer	Optical microscopy
Surface connected porosity	Mercury porosimetry

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APPENDIX A

Table A-1 provides a summary of the information provided in the report (Section 4).

Table A.1. Instrumentation used in high-temperature gas reactors

			AVR—GERMANY (1967–1988)	PEACH BOTTOM—USA (1967–1974)	FORT ST. VRAIN—USA (1979–1989)	HTTR—JAPAN (1999–)	HTR-10—CHINA (2003–)
IN-CORE INSTRUMENTATION	Nuclear Instrumentation	**Source-range monitoring system**	N/S	Three ^{10}B-lined proportional counters with motor-driven cadmium shields to limit neutron exposure during power operation One source-range channel had triple the sensitivity of the other two. The other two channels had alarm comparators, and log count-rate and rate-of-change indication.	Two high-sensitivity ^{10}B-lined counters Channels had low count-rate trips to prevent control rod withdrawal without source indication. A high-count-rate scram was provided for the core loading and physics test.	N/S	N/A
		Wide-range monitoring system	N/S	Two compensated ionization chambers covering almost eight decades down from 500% power	Three fission chambers as wide-range log-and-linear power-level channels Channels derived a wide-range logarithmic power signal and an independent linear power signal from the same fission chamber.	High-temperature fission counter chambers	N/A
		Power-range monitoring system	N/S	Four uncompensated ionization chambers covering from 0% to 150% full power with linear indication	Three linear power-level channels	High-sensitivity gamma-uncompensated ionization chambers	N/A
	Fuel Failure Detection System		N/S	Monitoring the purge stream activity	Flow-through ionization chamber	Two precipitators, a compressor, and a scintillation detector and associated electronics The primary method of detection is monitoring short-lived fission products, such as Kr-88 and Xe-138.	N/A
	Temperature		- melt-wire experiment		24 Geminol-P and Geminol-N	{TBD}	{TBD}

Table A-1. (continued)

			AVR—GERMANY (1967–1988)	PEACH BOTTOM—USA (1967–1974)	FORT ST. VRAIN—USA (1979–1989)	HTTR—JAPAN (1999–)	HTR-10—CHINA (2003–)
PROCESS INSTRUMENTATION	Primary coolant	**Temperature**	- no direct measurement of hot gas temperature - hot gas stream temperatures near the core were measured by thermocouple/noise thermometer as part of instrumentation development/testing	- 97 chromel-alumel thermocouples (formed by brazing No. 28 AWG wires insulated with reactor-grade high-purity magnesia, sheathed in 300-series stainless steel) were used to measure reflector, reactor coolant and fuel element temperatures below 538 ºC - For temperatures above 538 ºC up to 1310 ºC, 59 tungsten-rhenium thermocouples (formed by brazing the hot junction, sheathed in high-purity seamless molybdenum and insulated with seamless high-purity niobium tubing) were installed in 36 of the 804 fuel elements.	- 37 Geminol-P and Geminol-N for primary coolant - 148 Geminol-P and Geminol-N for coolant outlet	Type-N thermocouples	Type-K thermocouples
		Pressure	{N/S}	{N/S}	{N/S}	{N/S}	Safety-related Class 1E commercial pressure transmitters (no further specifications found)
		Flow	{N/S}	{N/S}	{N/S}	{N/S}	Safety-related Class 1E commercial Δp pressure transmitters (no further specifications found)
	Secondary coolant	**Temperature**	{N/S}	{N/S}	- 220 Geminol-P and Geminol-N for steam generators	{N/S}	Safety-related Class 1E commercial thermocouples (no further specifications found)
		Pressure	{N/S}	{N/S}	{N/S}	{N/S}	Safety-related Class 1E commercial pressure transmitters (no further specifications found)
		Flow	{N/S}	{N/S}	{N/S}	{N/S}	Safety-related Class 1E commercial Δp pressure transmitters (no further specifications found)
CAL INSTRUM ENTATIO	**Moisture**		{N/S}	Electrolytic cell	- Rhodium-plated mirror for measurement of dew point	{N/S}	{N/S}

Table A-1. (continued)

			AVR—GERMANY (1967–1988)	PEACH BOTTOM—USA (1967–1974)	FORT ST. VRAIN—USA (1979–1989)	HTTR—JAPAN (1999–)	HTR-10—CHINA (2003–)
	Gas analysis		{N/S}	Automatic gas chromatograph	- Infrared analyzer for CO - Gas chromatographer for CO_2 - GC chromatographer for N_2	{N/S}	{N/S}
	Dust		{N/S}	{N/S}	{N/S}	{N/S}	{N/S}
	Other impurities		{N/S}	{N/S}	{N/S}	{N/S}	{N/S}
OTHER INSTRUMENTATION	Reactor vessel and vessel cavity	Temperature	{N/S}	34 thermocouples (internal structure)	235 thermocouples	{N/S}	{N/S}
OTHER INSTRUMENTATION	Reactor vessel and vessel cavity	Strain	{N/S}	- 8 strain gauges (internal) - 10 strain gauges (external)	Bonded resistance wire type strain gauges	{N/S}	{N/S}
OTHER INSTRUMENTATION	Reactor vessel and vessel cavity	Moisture	{N/S}	{TBD}	{TBD}	{N/S}	{N/S}
OTHER INSTRUMENTATION	Control rod drive position		{N/S}	{N/S}	{N/S}	- position of 16 pairs of control rods is monitored by encoder sensors. - signal is used for the reactor control system and the safety protection system.	{N/S}

N/S—Not specified.
TBD—To be determined.

DRAFT

LTR/NRC/RES/2011-002

Letter Report

Task 2. Impact of Operating Conditions on Instrumentation During Normal Operation and Postulated Accidents

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ACRONYMS

AOO	anticipated operational occurrence
ATWS	anticipated transient without scram
BDBE	beyond design basis event
BOP	balance of plant
BTP	branch technical positions
CFR	code of federal regulations
CFD	computational fluid dynamics
DCD	design control document
D-LOFC	depressurized loss-of-forced circulation
DOE	(U.S.) Department of Energy
EMC	electromagnetic compatibility
EMI	electromagnetic interference
EPA	Environmental Protection Agency
ESF	engineered safety function
FHR	fluoride salt cooled high temperature reactor
GA	General Atomics
GDC	general design criteria
GRSAC	graphite reactor severe accident code
GT-MHR	gas turbine modular helium reactor
HPT	high-pressure turbine
HTGR	high-temperature gas-cooled reactor
HTR	high temperature reactor
HTSE	high temperature steam electrolysis
IAEA	International Atomic Energy Agency
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronic Engineers
IHX	intermediate heat exchanger
INL	Idaho National Laboratory
IPT	intermediate pressure turbine
ISA	International Society of Automation (formerly Instrumentation Society of America)
LOFC	loss-of-forced circulation
LPT	low pressure turbine
LSHT	liquid salt heat transfer
LWR	light-water reactor
MEMS	micro-electro-mechanical systems

MHTGR	modular high-temperature gas-cooled reactor
NERI	Nuclear Energy Research Institute
NGNP	next generation nuclear plant
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
OSHA	Occupational Safety and Health Administration
PBMR	Pebble Bed Modular Reactor (South Africa)
PHX	process heat exchanger
PIRT	phenomena identification and ranking table
P-LOFC	pressurized loss-of-forced circulation
PCU	power conversion unit
PRT	platinum resistance thermometer
R&D	research and development
RCCS	reactor cavity cooling system
RCS	reactor coolant system
RFI	radio-frequency interference
RG	Reg. (regulatory) Guide
RPV	reactor pressure vessel
RTD	resistance temperature detector
SAR	safety analysis report
SC-MHR	steam cycle modular helium reactor
SCS	shutdown cooling system
SG	steam generator
SHX	secondary heat exchanger
SRM	staff requirements memoranda
SMR	steam methane reformer
SRP	standard review plan
SSC	structures, systems, and components
T/F	thermal fluid
TMI	Three Mile Island
TRISO	tri-layer isotropic (particle fuel)
V&V	verification and validation
VHTR	very high temperature gas-cooled reactor
VS	vessel system
WSP	Westinghouse sulfur process

1. INTRODUCTION

This letter report identifies the adverse conditions imposed on instrumentation in very high temperature gas-cooled reactors (VHTRs) during normal operation as well as off-normal operations and postulated accident conditions, and focuses as well on the adverse operating conditions associated with coupling the VHTR to a hydrogen production plant or other high-temperature process heat system. It supplements the information provided in the Task 1 Letter Report for this project,[1] looking at special needs for sensors and diagnostics in adverse conditions, for process heat plant alternatives with various heat transport loop designs and for perceived "gaps" between needs and existing capabilities.

The two major high temperature gas-cooled reactor (HTGR) candidate designs for NGNP—the pebble-bed and prismatic core designs—are instrumented differently primarily because of the limitations imposed by their core configurations. For instance, in-core instrumentation in pebble-bed reactors is a major challenge because of the continuously moving fuel elements (due to on-line refueling). On the other hand, earlier prismatic HTGRs have employed some limited in-core instrumentation, as noted in Sect. 6.1. However, as design gas pressures and outlet temperatures for HTGRs increase due to needs of prospective process heat customers and the enhanced efficiency of gas turbines, sensors are subjected to increasingly harsher conditions.

HTGRs can provide process heat at temperatures from 700 to ~950°C. Note that for the upper range of these operating temperatures, the HTGR is sometimes referred to as the Very High Temperature Reactor (VHTR). For the purposes of this project, however, since the U.S. Department of Energy (DOE) Next Generation Nuclear Plant (NGNP) program language refers to the VHTR, the gas-cooled reactor described herein will sometimes be referred to as the VHTR even though DOE's current plans for NGNP deployment focus on the lower end of that temperature range during normal operation.

However, the instrument designer must also take into account the higher temperatures and other potential environmental conditions from the most severe conditions resulting from transients and postulated accidents, and demonstrate that the instrumentation systems will function reliably as needed for diagnostics and recovery actions.

2. NGNP SYSTEM DESCRIPTION

The NGNP project was initiated at Idaho National Laboratory (INL) by DOE and pursuant to the Energy Policy Act of 2005 (Public Law 109-58). The mission of the NGNP project is to broaden the environmental and economic benefits of nuclear energy technology by demonstration through deployment in industrial applications its use for market sectors not currently served by light-water reactors (LWRs).[2]

NGNP relies upon the design and operation experience from earlier gas-cooled reactor technology. A number of prototype and demonstration HTGRs have been operated over the past 60 years that focused on the generation of electrical power. Two such reactors, Fort St. Vrain and Peach Bottom, were licensed and operated commercially in the United States. Internationally, both pebble bed and prismatic block reactors have been licensed and operated in the United Kingdom, Germany, Japan, and China.

The proposed operating ranges for NGNP could provide process heat to be used for diverse chemical process applications and hydrogen production as well as for electricity production. Key characteristics of the HTGR concept are the use of helium as a coolant, graphite as the moderator of neutrons, and ceramic-coated particles as fuel. Helium is chemically inert and will not react under any condition. The graphite core slows down (moderates) the neutrons and provides high-temperature strength and structural stability.

The ceramic-coated fuel particles are extremely robust and retain the radioactive byproducts of the fission reaction. Two major core design concepts—a prismatic block reactor and a pebble bed reactor—are currently under consideration for the NGNP.

The prismatic block reactor core configuration consists of hexagonal graphite blocks stacked to fit in a cylindrical steel pressure vessel. Cylindrical passages are located within each block for the helium coolant and for graphite compacts that contain the coated particle fuel. Additional graphite blocks surround the core to shape and reflect the neutron flux. The reactor is refueled with blocks containing new fuel approximately every 18 months. The pebble bed design uses fuel particles that are formed into pebbles, approximately the size of a racquetball, with graphite reflectors surrounding the pebble core to provide structural support and reflect neutrons back into the core. The pebbles continuously circulate through the core and are re-circulated six to ten times over the course of 3 years before being permanently discharged from the reactor. Fresh fuel pebbles are added to replace those discharged.

The NGNP project issued a Request for Proposals for the deployment of preconceptual designs of HTGR plants for the production of electricity and hydrogen at the INL site. Major applications from three teams headed by vendors of HTGR design included the Pebble-Bed Technology Team principally lead by Westinghouse Electric Company and Pebble Bed Modular Reactor Pty. Ltd. (PBMR) from South Africa; AREVA NP, Inc.; and General Atomics (GA). These design teams were international in nature and each consisted of multiple team members providing specific capabilities relevant to HTGR development, nuclear power applications, and hydrogen production. A total of 26 companies managed by the three teams' leaders participated in the design work.[2] A schematic representation of the NGNP with the potential industrial sectors is shown in Fig. 1.

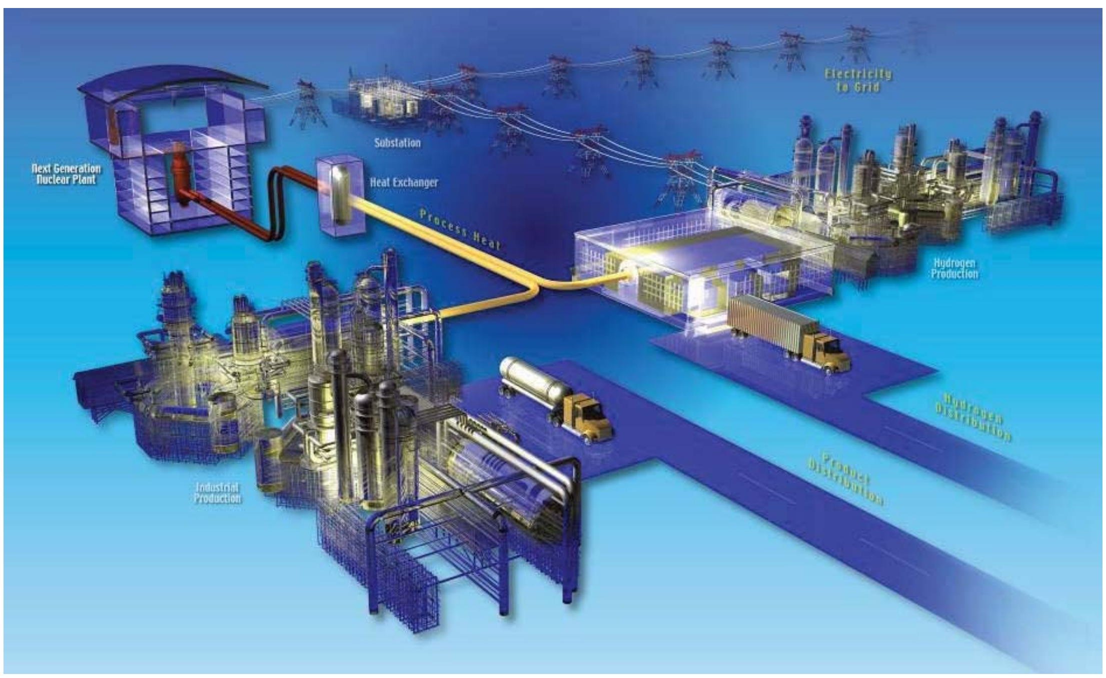


Fig. 1. Schematic drawing of the Next Generation Nuclear Plant with various industrial services.
[Adapted from Ref. 3]

2.1 General Atomics Design

An earlier GA NGNP design[4] is a prismatic block core design, as shown in Fig. 2. This GA design is derived from the gas-turbine modular helium reactor (GT-MHR) plant, which includes a direct Brayton

cycle gas turbine in a vertical configuration to produce electricity. Subsequently, GA added a second parallel loop, as shown in Fig. 3, supplying a compact intermediate heat exchanger that in turn supplies heat to a prototype sulfur-iodine hydrogen production facility.

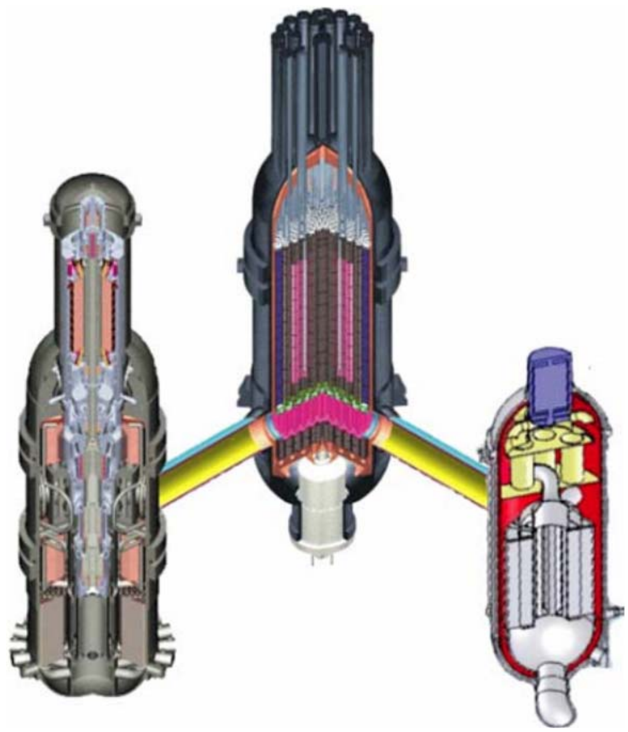


Fig. 2. General Atomics earlier proposed NGNP configuration.

Following the preconceptual design report submission, GA revised the plant operating conditions and configurations as a result of the interactions with the potential end users. While GA original plant concept used gas turbines, the revised concept[5] proposes a plant configuration that has a steam generator in the primary loop supplying both the process steam and a conventional Rankine steam turbine cycle, as shown in Fig. 3. GA has considered two reactor power levels, 350 and 600 MW(t), with reactor outlet temperatures in the 750 to 800ºC range. The 350 MW(t) design is based on the Modular High Temperature Gas Reactor (MHTGR) design that DOE/GA developed in the late 1980s.[6] Having the choice of two different power levels is expected to offer flexibility in applying the technology in multiple-module configurations to satisfy variations in demand and availability requirements.

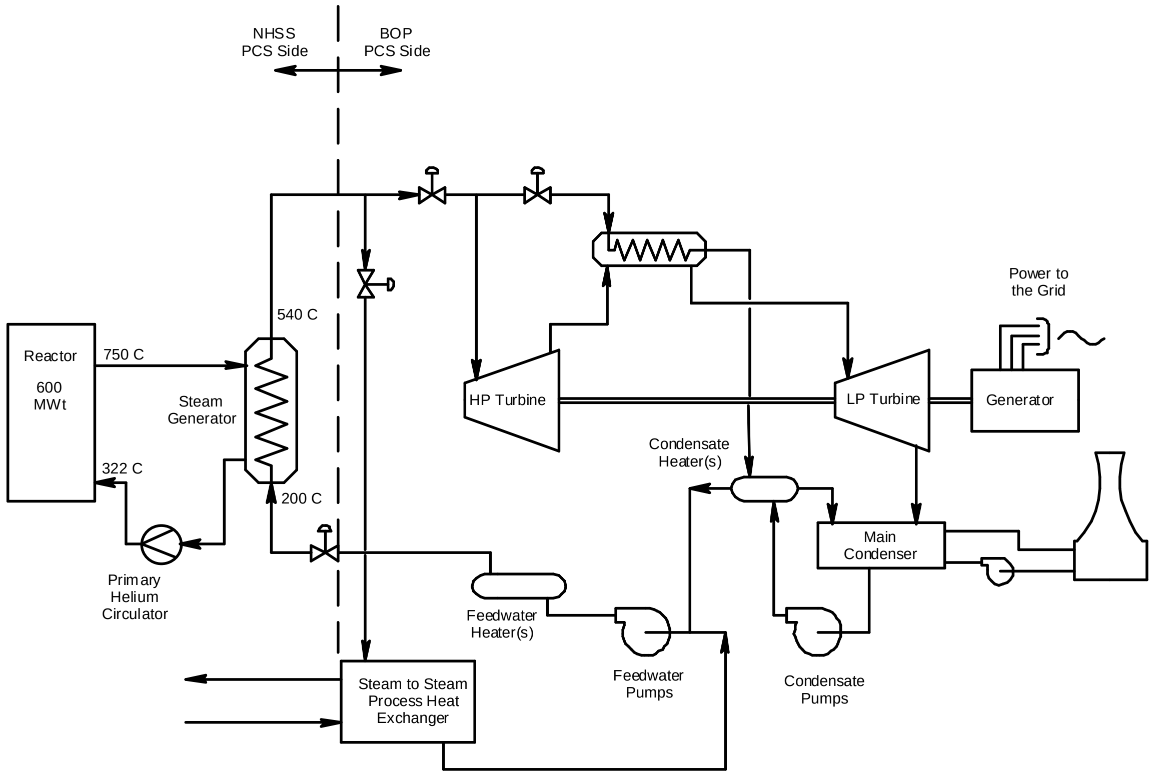


Fig. 3. Schematic of the NGNP configuration considered by GA.
[Adapted from Ref. 5]

2.2 AREVA Design

The AREVA design for NGNP also uses a prismatic block reactor. The original concept proposed by AREVA (Fig. 4) was derived from AREVA's Antares plant, with 550 to 600 MW(t) power levels and 900 to 950ºC reactor outlet helium temperatures.[4]

The Antares plant was designed specifically for electricity production, and included an indirect Brayton cycle gas turbine in a horizontal configuration. For the NGNP proposal, AREVA kept the majority of the Antares design characteristics but included a parallel loop supplying high-temperature gas for the process facility. The first AREVA NGNP design did not include design specifications for the hydrogen production facility, only brief evaluations of the high-temperature steam electrolysis process.

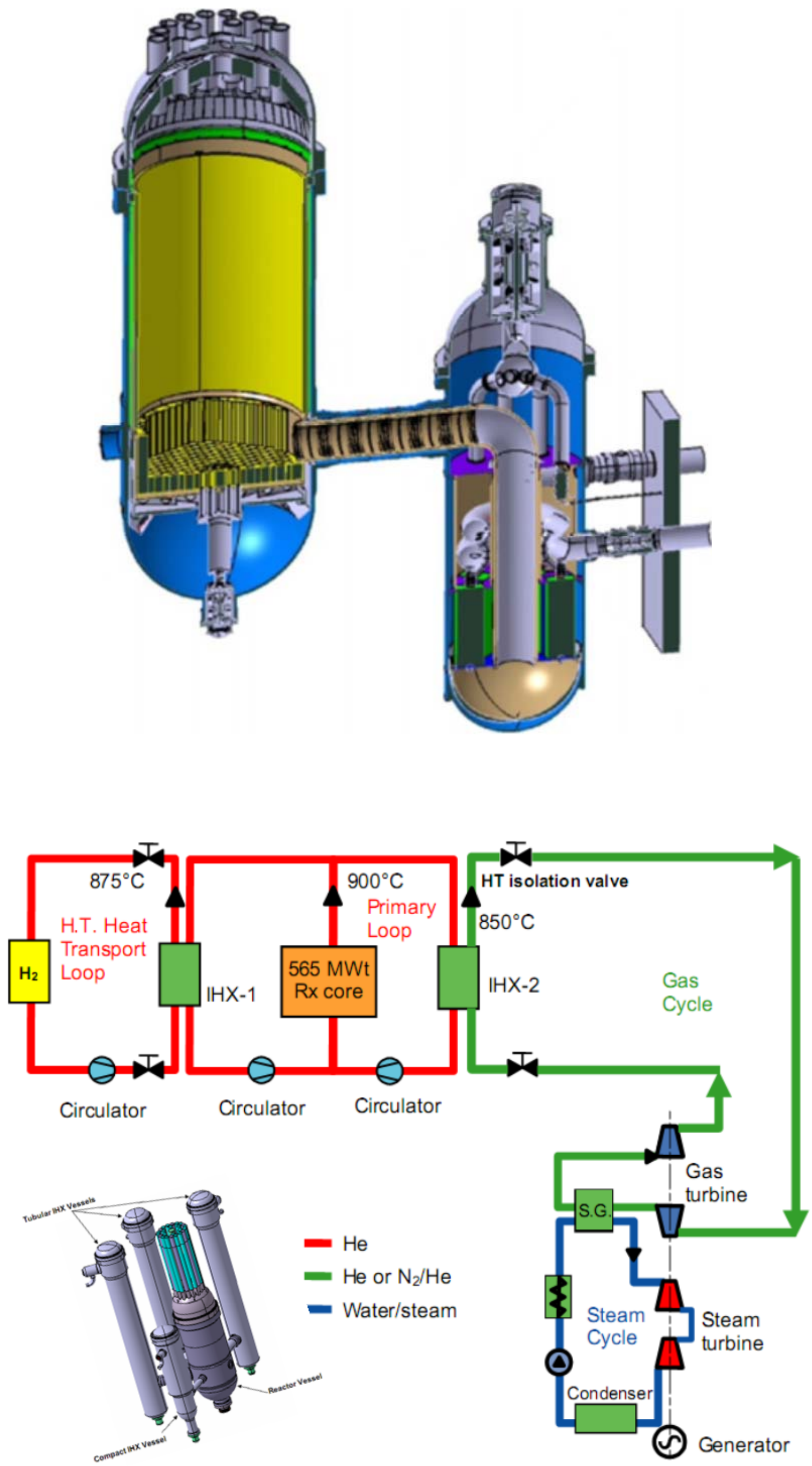


Fig. 4. Original AREVA NGNP design.

As with GA, AREVA also revised their preconceptual design configuration in 2008 and 2009, primarily because of market demand and interactions with potential end users. AREVA also revised the reactor

power level, and produced design concepts with 350 and 600 MW(t) power levels with reactor outlet temperatures between 750 and 800ºC. This concept revision also eliminated the gas turbine and included a steam generator in the primary loop with high-pressure and low-pressure re-boilers taking the heat from the high-pressure and low-pressure turbine midstages, as shown in Fig. 5.

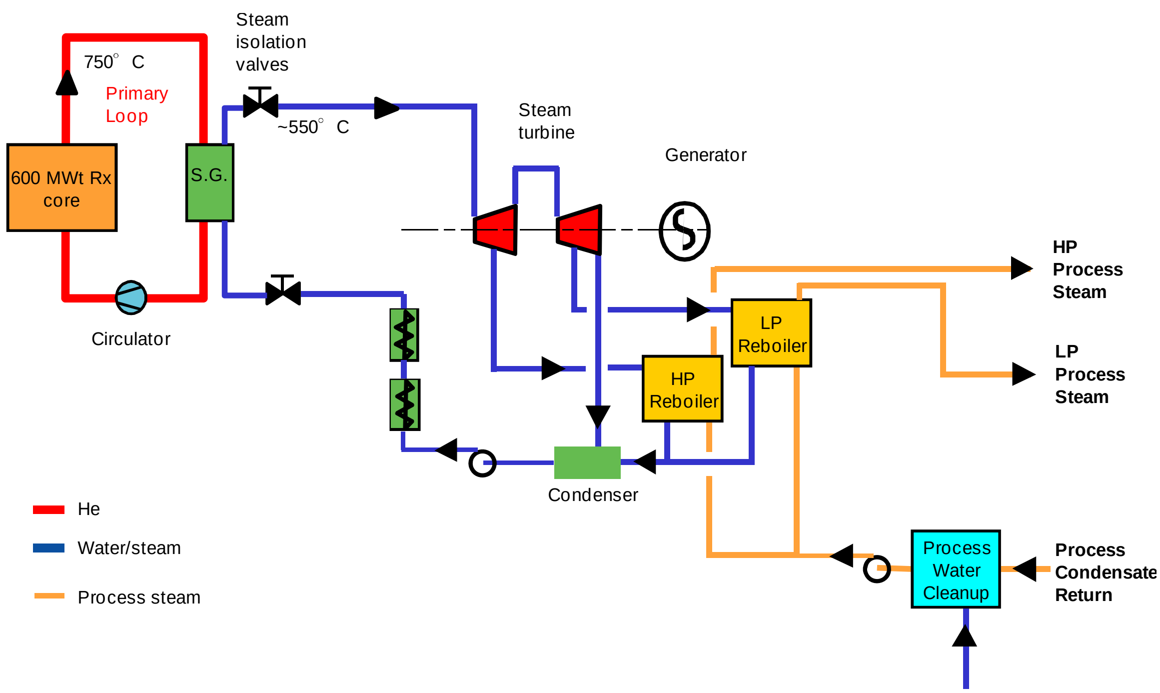


Fig. 5. Recent NGNP flowsheet proposed by both AREVA and GA.
[Adapted from Ref. 2]

2.3 Westinghouse Design

In a preconceptual report, the Westinghouse/PBMR (Pty) Ltd. team proposed an annular core pebble bed reactor design, as shown in Fig. 6, based on the South African PBMR (Pty) Ltd. Demonstration Power Plant design. This design featured a reactor outlet temperature of 900 to 950ºC and a direct cycle gas turbine.[4]

In this design, pebbles are in the annulus formed by the side reflector and the center reflector, all contained within a core barrel. The pebble bed reactor is refueled online. Over a period of about 6 months, each pebble travels down the core and exits from the bottom of the vessel. Each pebble is then examined in the plant fuel handling system to determine if it has reached its burnup limit or is damaged.

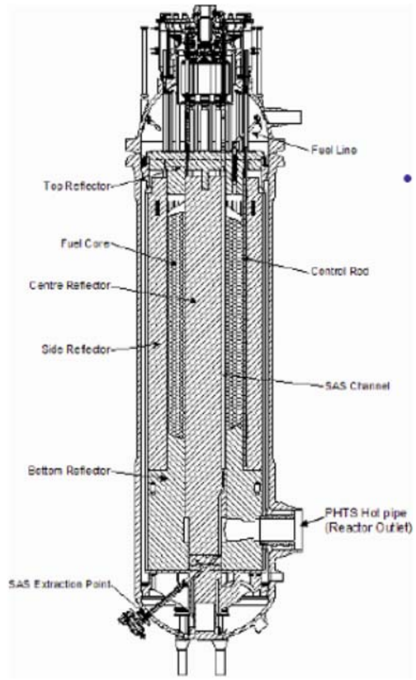


Fig. 6. Original Westinghouse/PBMR (Pty) Ltd. team design.
[Adapted from Ref. 4]

The primary loop included two intermediate heat exchangers in series supplying heat to a secondary helium loop, which ultimately transferred heat to a steam generator and the hydrogen process, as shown in Fig. 7. The steam generator drove a steam turbine for production of electricity. The hydrogen production facility used hybrid-sulfur process.

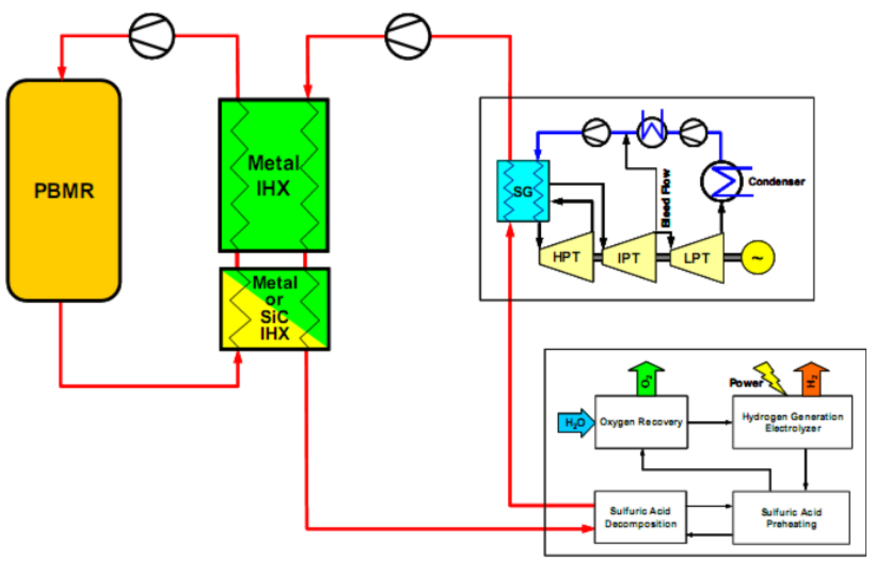
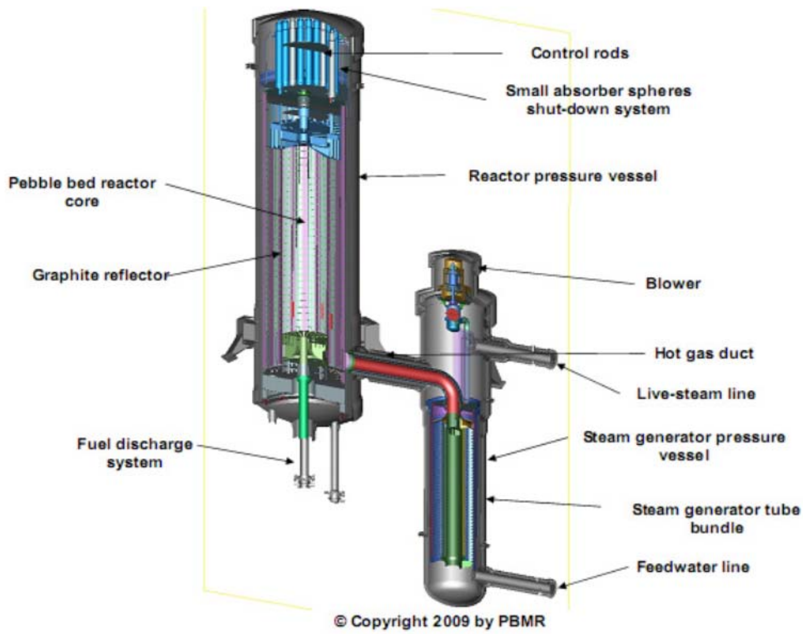


Fig. 7. Flowsheet for the original Westinghouse/PBMR (Pty) Ltd. team design.

Flowsheets and configurations for later NGNP designs are shown in Figs. 8 and 9. Note that the later pebble bed designs had cylindrical rather than annular cores.



Courtesy of Pebble Bed Modular Reactor (Pty) Ltd.

Fig. 8. Later Westinghouse/PBMR conceptual configuration.
[Adapted from Ref. 4]

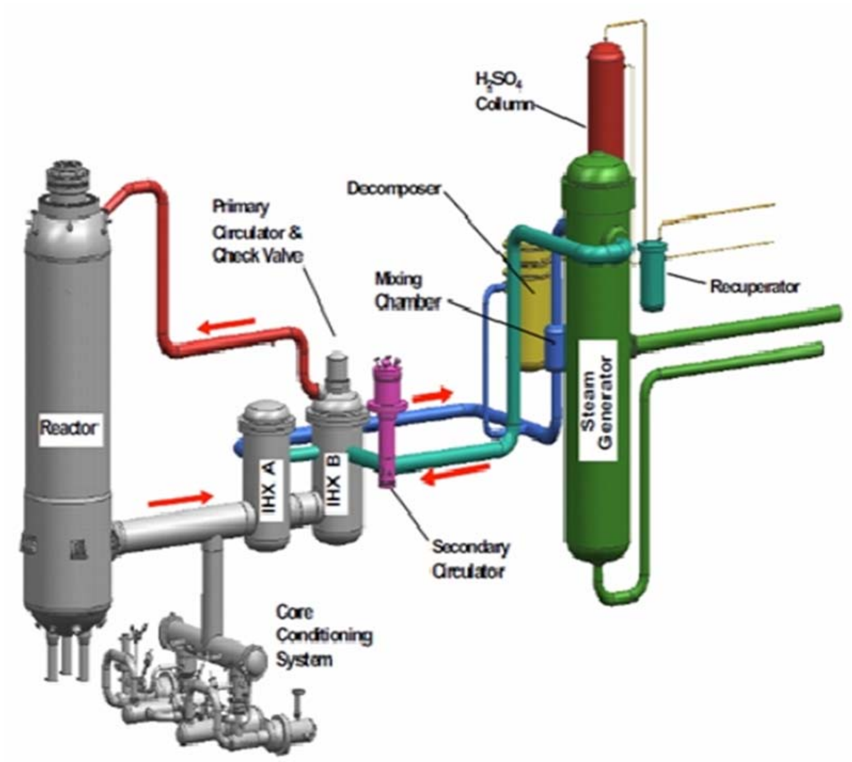


Fig. 9. Westinghouse/PBMR team design with an indirect configuration for electricity and hydrogen production. [Adapted from Ref. 2]

3. IDENTIFICATION OF ADVERSE CONDITIONS DURING NORMAL AND OFF-NORMAL OPERATIONS

What constitutes "normal" operation will depend on the NGNP application (i.e., what end use process heat plant is selected). There are a wide variety of possibilities for NGNP, as shown in an INL survey[7] and an MPR report[8] characterizing the process heat temperature ranges for the various applications. These are shown in Table 1 and Fig. 10. LWRs are typically not used for high-temperature process heat systems since their coolant outlet temperatures are in the 300–350ºC range. To use LWRs as a heat source for (flash) desalination plants, it would be necessary to "cut off" the back end of the low-pressure turbine (with an attendant loss of efficiency) to obtain steam hot enough to drive typical flash evaporators.

Table 1. Prioritization of potential industrial applications of the HTGR technology

Industry	Assessment	Priority
Petroleum Refining	Multiple refining processes have very high energy demands and suitable process temperatures.	High
Coal and Natural Gas Derivatives	In situ bitumen extraction has a high energy demand, suitable process temperature, and high growth expectations.	High
Petrochemicals	Multiple petrochemical production processes have very high energy demands and suitable process temperatures.	High
Industrial Gases (Hydrogen)	Steam methane reforming and advanced hydrogen production methods have high energy demands and suitable process temperatures.	High
Fertilizers (Ammonia, Nitrates)	Ammonia production has high energy demand and suitable process temperatures.	High
Metals	Direct-reduced iron (DRI) production has high energy demands, suitable process temperatures, and strong global growth.	High
Polymer Products (Plastics, Fibers)	Certain polymers have large energy demands, suitable process temperatures, and strong global growth.	High
Cement	The current cement process temperatures are too high, but production is possible at suitable temperatures with technology development.	Low
Pharmaceuticals	The process energy needs of the pharmaceutical industry on a per plant basis are relatively low.	Low
Paper	The typical energy requirements for a mill is low and byproducts, having little value otherwise, are burned to provide half of the steam and electricity needs of paper products.	Low
Glass	Glass production process temperatures are too high.	Low

Table 1 does not list a flash desalination option. However, desalination would be an especially attractive option for direct (or indirect) cycle gas turbine power plants, since the normal coolant discharge temperature from the recuperator (to the precooler) is nearly ideal for driving a brine heater, with little or no degradation in electric power. A follow-up report by Sandia also provides useful information on the high-temperature process heat market potentials in the United States.[9] Elaboration on the characteristic of various process heat plant alternatives is found in Sect. 4.

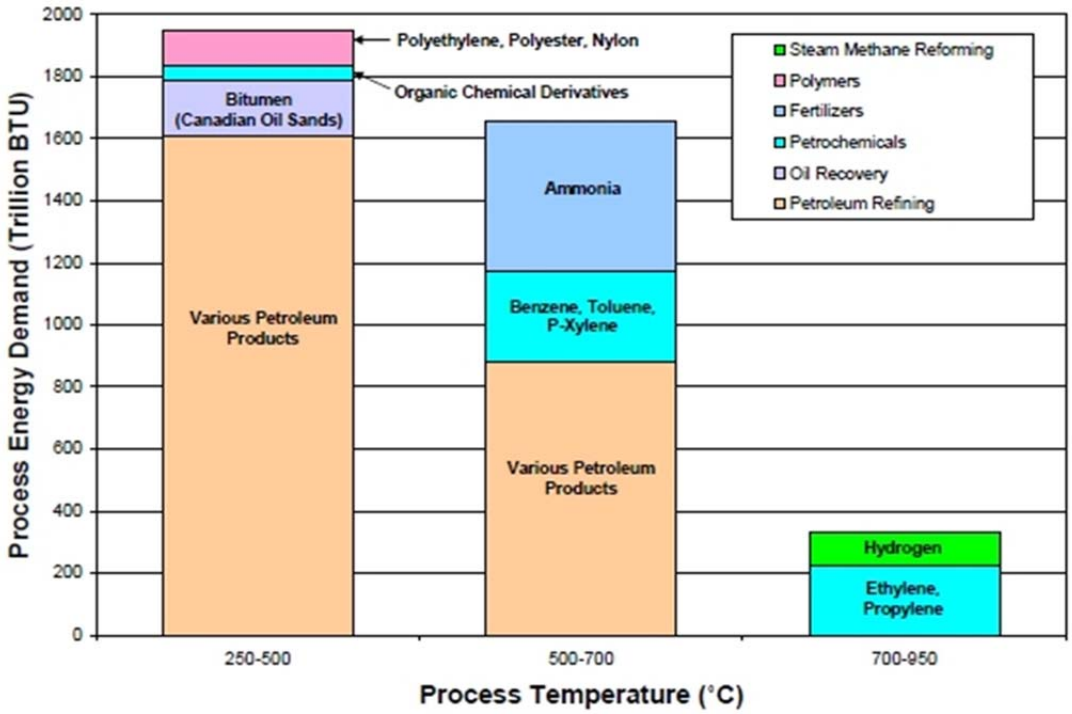


Fig. 10. MPR survey of industrial needs (in the United States) for high-temperature process heat.

A similar study of process heat applications as a function of driving temperatures showed similar results to the MPR study (Fig. 11) [Ref. 7].

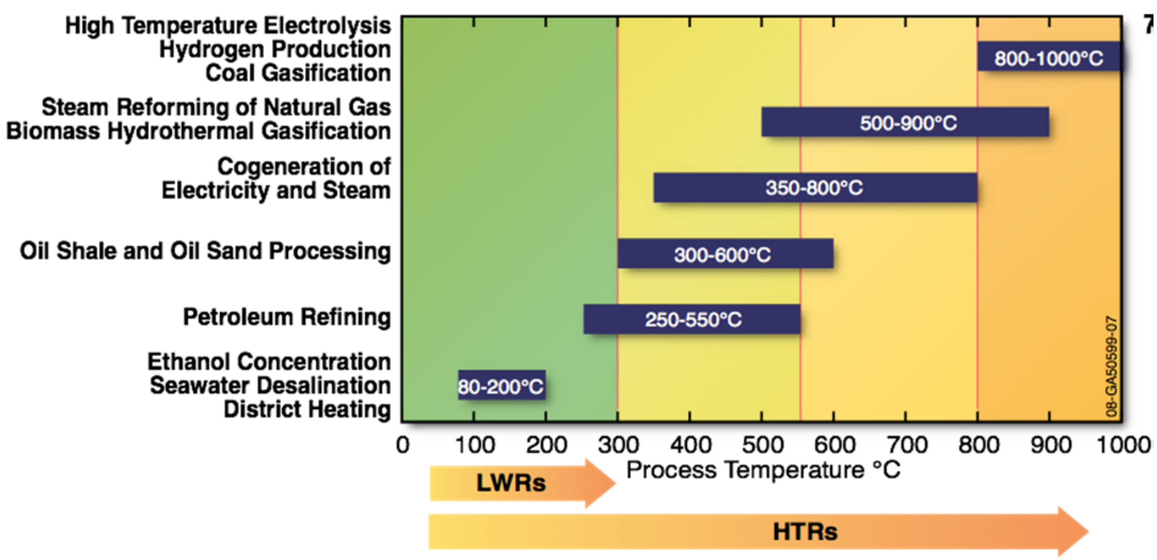


Fig. 11. Summary of temperature requirements for potential end users of HTGRs.

3.1 Normal Operation

The purpose of this section is to provide supplementary information about instrumentation environments and design requirements in addition to that presented in Sect. 5.2 of the Task 1 Letter Report.[1]

Power operation for reactors is normally defined as operation from 5% core power to nominal full core thermal power, which corresponds to “Mode 1” for an LWR. Low-power operation is normally defined as operation with the core at between 1 and 5% power. Operation in these two modes is controlled within a normal operating envelope of coolant pressures, temperatures, flow rates, and core power distributions.

Hot standby is typically at a no-load set of pressure, temperatures, and flow rates where the reactor is normally taken critical (roughly corresponding to “Mode 3” for LWRs).

Shutdown operation encompasses all normal operation from ambient pressure and temperatures to hot standby, with the core kept at least 1% (reactivity) subcritical. For HTGRs, “cold” shutdown temperatures are usually much higher than for LWRs—with temperatures typically determined by the need to have only minimal oxidation of the graphite core structures in air (~250–300ºC). With the very large negative temperature coefficient of reactivity typical of modular HTGRs, much less shutdown reactivity would be needed at these higher temperatures.

Depending on the design, circulators in HTGRs can be used to supply enough heat to the reactor coolant system (RCS) to bring the plant from ambient pressure and temperature (as in LWR’s “Mode 5”) to a stable hot standby (Mode 3). Like in pressurized-water reactors (PWRs), the energy delivered by the motor to the circulator shaft eventually appears as heat to the RCS.

One notable feature of modular HTGRs, as compared to LWRs, is that the average coolant temperature rise across the core at full power is typically about a factor of 10 greater, or ~400ºC. Furthermore, due to the variations in core radial power peaking and core coolant flow spatial distributions, core outlet temperatures can vary significantly from the mean (mixed) temperature. This can result in some outlet temperatures in the order of 100–150ºC higher (and lower) than the mean for annular core designs. For cylindrical cores (PBRs), these differences could be even higher due to the higher central peaking factors. This results in several significant considerations:

1. temperature measurements at the core outlet (support structures and/or coolant) must be capable of (stable, long-term) measurements at temperatures well above the rated mean;
2. temperature gradients in the support structures can be quite high;
3. large differences in the exit coolant flow stream temperatures can cause problems in ensuring adequate mixing to attain a valid “mixed mean” temperature for power calibrations; and
4. temperature fluctuations resulting from the mixing process can cause high-frequency thermal stress fatigue in support structures and cover plates in the outlet plenum, as well as “noise” in temperature signals.

Operating primary coolant pressures in the newer modular HTGR designs are also significantly higher than in earlier HTGRs. Especially in PBRs, where the core pressure drops are high, increasing the operating pressure reduces the pressure drop (proportionally), thus reducing the pumping power required for circulation.

3.2 Anticipated Operational Occurrences (AOO)

There are a number of transient conditions that are expected as relatively minor perturbations (or concerns) from the normal operating mode. These example transients are typically classified as AOOs:

1. turbine trip,
2. loss of load (full or partial),
3. gas cycle valve failures (failed closed or open),
4. control rod or rod bank withdrawal,
5. inadvertent control rod movement,
6. accidental reactor shutdown,

7. unexpected sudden increase or decrease of primary heat removal rate, and
8. control rod drop.

Anticipated transients or operational occurrences are upsets that are externally imposed on the plant such as loss of electrical load or internally imposed from events such as single initiating failures of equipment. The plant either continues to operate within the abilities of the control systems, or one or more parameters exceed the normal operating capabilities and result in actuation of one or more reactor protection system trips that shut down the reactor and may, in the process, actuate safety systems. The function of the safety grade (and other) equipment is to prevent a transient from progressing from an AOO condition to the next level, a design basis event (DBE). DBEs are typically classified as incidents that are not expected to occur within the lifetime of the plant. By definition, the operating conditions during AOOs are kept within normal bounds, and so instrumentation environments are typically not challenged any more than normal.

3.3 Postulated Accident Conditions, Including Design Basis Events (DBEs) and Beyond Design Basis Events (BDBEs)

A discussion of plant conditions expected during postulated accident conditions was included in Sect. 5.3 of the Task 1 Letter Report.[1] This section provides supplementary information about additional modeling and data, currently seen as potential "gaps" in our understanding and capabilities, that could contribute to better estimates of the accident conditions seen by plant instrumentation.[10]

3.3.1 Pressurized loss-of-forced circulation (P-LOFC)

Events involving P-LOFC are assumed to occur during power operation, where the primary helium flow stops and the primary system remains pressurized. The analysis for the P-LOFC accident requires 2-D or 3-D modeling of the core thermal-fluid (T/F) behavior to calculate heat transfer by conduction, natural convection, and radiation; and to evaluate the temperatures of the fuel and metallic components in the core, vessel, and cavity regions. A detailed modeling of the heat transfer, from the fuel through the core barrel and other core internals and vessels to the reactor cavity cooling system, is essential. Temperature measurements in these areas would be especially useful for code validation. Improved models for a failed or degraded reactor cavity cooling system (RCCS) may be required, depending greatly on the design features of the RCCS. Heat transfer models of the RCCS heat removal processes for conductivity, convection, and radiation (emissivity) coefficients, as well as heat capacity, depend on a number of factors such as temperature, reactor operation history, and special material properties, all of which must be taken into account in the modeling. An uncertainty evaluation may be necessary.

3.3.2 Depressurized loss-of-forced circulation (D-LOFC)

In D-LOFC events, starting from power operation, the primary helium inventory is lost to the point that the primary system is depressurized to atmospheric pressure. For the analysis of the depressurization phase, detailed T/F computer codes and models are necessary to evaluate the pressure and temperature transients at different places inside the reactor and reactor confinement building.

For the evaluation of source terms, specialized computer codes and models are necessary to estimate the fission product transport and release mechanisms. For evaluation of the prompt release, the needs include an estimate of the circulating activity, along with an estimate or model for the release of graphite dust with entrained radioactivity. For the analysis of delayed releases, modeling is necessary for the calculation of fuel failure plus the fission product retention capability of the primary system, reactor building, and filtering. All of these features will impact the environment for the instrumentation in the areas.

3.3.3 Anticipated transient without scram (ATWS)

Normally the initiating event for an ATWS event sequence is an AOO followed by a failure to shut down the reactor. In analyses of an ATWS event, all control and safety rod positions are assumed fixed, and no rods drop in response to scram signals. Other protective actions, such as core heat removal via the RCCS, are assumed successful; however, there may be situations where other assumptions result in adverse consequences. For example, the termination of active cooling is a protective action for accident conditions, where failure of such action in an ATWS can represent a serious hazard, and these eventualities should also be considered. The reactor power, primary pressure, and the maximum fuel temperature should be carefully evaluated for the short-term responses. The temperature histories of key components, such as the core barrel, also need to be measured and assessed against acceptance criteria. In a conservative analysis, uncertainties in measurement and modeling should be taken into account; either conservative value or uncertainty analyses should be performed.

3.3.4 Steam/water ingress

Water/steam ingress into a HTGR core can result from steam generator heat transfer tube leaks or breaks in steam cycle designs, where the pressure of the secondary water/steam is much higher than that of the primary helium. Water ingress events can involve complex interactions of neutronics, thermofluids, chemical reactions, and radioactivity releases. Detailed computer codes and models would be needed to calculate the rate and amount of water/steam ingress, the reactivity effects, and any resulting power transients, pressure and temperature transients, production of oxidization gases, and the added radioactivity source terms. Measurements of any or all of these parameters would be very useful in providing mitigating actions and post-accident analyses. Uncertainty analyses are likely to be necessary as part of understanding the potential range of accident parameters.

3.3.5 Air ingress

Air ingress into the primary system is a safety concern because of the damage it could cause by oxidizing graphite structures and components within the vessel and by potential oxidation damage to the fuel (TRISO [tri-layer isotropic] particles). Since the oxidation rates for graphite are very sensitive to temperature, dynamic models need rather fine structure nodalization of the core lower support structure and reflector regions in addition to the fueled areas. Careful analysis is recommended for determining any potentially significant loss of mass and strength in critical areas. At least 2-D, and preferably 3-D core T/F models, with oxidation modeling, would be advisable. The model should account for the differences in rate equations for the various types of graphite used in the structure and the fueled regions. Temperature measurements in the core support areas would be needed to evaluate damage rates.

Oxidation rate data, particularly for lower- and medium-range temperatures, can also be dependent on test specimen size and the rate of oxidant supply. Graphite oxidation rate models should account for the differences in the lower-temperature “chemical” range and the mass-transfer-limited rates in the higher temperature ranges.[11] The differences in prism block designs should also be accounted for, since there are variations in the protective graphite structures around the fuel compacts.

Modeling and data needs for the D-LOFC and P-LOFC would apply here also, since the air ingress accident is an add-on to modeling of these cases. For scenarios involving operation of a shutdown cooling system (SCS), models and design data for that system would be required as well, including details of potential access to air at the intake.

The availability of oxygen in the ingressed gas is probably most crucial to the final outcome for core and other structural damage if the oxidation is not stopped by other means. It has been found to be difficult to limit air leakage into a large confinement volume for situations like those following a D-LOFC, in which

initial primary system inventory release has been vented.[12] Data from representative experiments may be needed to check models of these effects to enable determination of a validated range of expected leakage rates and, thus, the oxygen potentially available to the core.

Previous GRSAC predictions[13] of core graphite oxidation used pessimistic assumptions about the availability of "fresh air" ingested into the vessel and core (i.e., it was unlimited). More realistic cases would account for the reactor pressure vessels being in a cavity or confinement building (vault) where fresh air in-leakage is limited, and gaseous products of the accident would collect and become components of the gas for subsequent core ingress.[14] Because of the wide range of potential scenario details, bounding calculations are necessary.

4. PROCESS HEAT PLANT ALTERNATIVES

4.1 Hydrogen Production

The production of hydrogen from water is being developed by a number of countries (including the United States) using electrolysis and/or thermochemical processes. In 2008, it was determined that up to $24 M could be saved by focusing limited funding on a primary technology with a backup rather than continuing to advance the three most promising technologies simultaneously. In 2009, INL led an effort to systematically evaluate and select the best technology for deployment with NGNP. This paper describes that trade study.

Figure 12 depicts the technology options for nuclear hydrogen production, and Table 2 lists a summary overview of nuclear hydrogen production techniques, with details of the processes given in subsequent subsections.

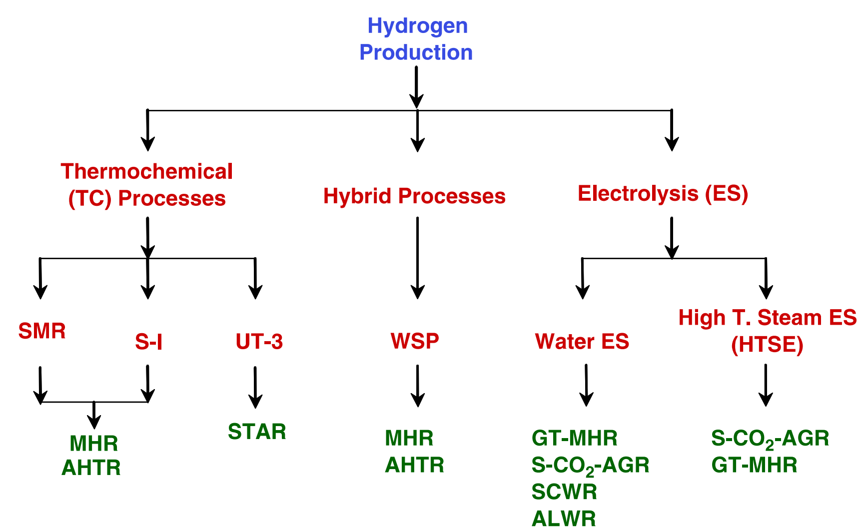


Fig. 12. Technology options for nuclear hydrogen production.

Table 2. Overview of nuclear hydrogen production techniques
[From Ref. 15]

Feature	Electrochemical		Thermochemical	
	Water electrolysis	**High-temperature steam electrolysis**	**Steam-methane reforming**	**Thermochemical water splitting**
Required temperature (ºC)	<100 (~1 atm)	>500 (~1 atm)	>700	• >800 for S-I • >800 for WSP • >700 for UT-3 • >600 for Cu-Cl
Efficiency of the process (%)	85–90	90–95 (T > 800ºC)	>60	>40
Advantage	+ Proven technology	+ High efficiency + Can be coupled to reactors operating at intermediate temperatures + Eliminates CO_2 emission	+ Proven technology + Reduces CO_2 emission	+ Eliminates CO_2 emission
Disadvantage	– Low energy efficiency	– Requires development of durable, large-scale electrolysis units	– CO_2 emissions – Dependent on methane prices	– Aggressive chemistry – Requires very high-temperature reactors – Requires development at large scale

4.2 Steam Methane Reforming

Steam methane reforming (SMR) is the most common commercial technology for hydrogen production. The SMR process requires high temperature, which is mostly generated by burning natural gas.

The SMR process is as follows:

Reforming (endothermic, 750–800ºC):

$$CH_4 + H_2O \longrightarrow CO + 3H_2$$

Shift (exothermic, 350ºC):

$$CO + H_2O \longrightarrow CO_2 + H_2$$

VHTR's can provide the necessary heat at high temperatures. This approach will reduce the CO_2 emissions to the atmosphere in large quantities. However, due to the nature of the chemical reforming and shifting processes, there is still a need for natural gas feedstock, and consequently CO_2 would still be emitted.

4.3 Sulfur Iodine Water Splitting Cycle

The details of the thermo-chemistry of the process are outside the scope of this document. However, a brief account of the process is summarized below.[16] The sulfur iodine (SI) cycle consists of three sets of reactions expressed by the following equations:

$$I_2 + SO_2 + 2H_2O \rightarrow 2HI + H_2SO_4 \text{ (Bunsen reaction)}, \tag{4.1}$$

$$H_2SO_4 \rightarrow H_2O + SO_2 + \frac{1}{2}O_2 \text{ (Sulfuric acid decomposition)}, \tag{4.2}$$

$$2HI \rightarrow H_2 + I_2 \text{ (Hydrogen iodide decomposition)}. \tag{4.3}$$

The reaction given by Eq. (4.1)—Bunsen reaction—proceeds in the liquid phase. This reaction produces two kinds of acid: sulfuric acid (H_2SO_4) and hydriodic acid (HI dissolved in water) from sulfur dioxide (SO_2), iodine (I_2), and water (H_2O). The mixed acid separates into two types of acid of its own accord (liquid–liquid separation). After separation of acids, they are purified, concentrated, and decomposed in the other two reactions. The reaction given by Eq. (4.2) is the sulfuric acid decomposition reaction that produces oxygen, sulfur dioxide, and water. The third reaction in Eq. (4.3) is the HI decomposition reaction that produces hydrogen and iodine. With the exception of hydrogen and oxygen, the other products in Eqs. (4.2) and (4.3) can be reused in the Bunsen reaction step as the reactant material. The endothermic H_2SO_4 decomposition reaction can be operated at about 800–1000ºC. The decomposition of hydriodic acid involves an endothermic reaction around 400–500ºC. The Bunsen reaction occurs exothermically at temperatures of about 100ºC. Heat source of two endothermic acid decomposition reactions in the SI cycle can be provided by the nuclear heat. A flowsheet schematic of the process is given in Fig. 13.

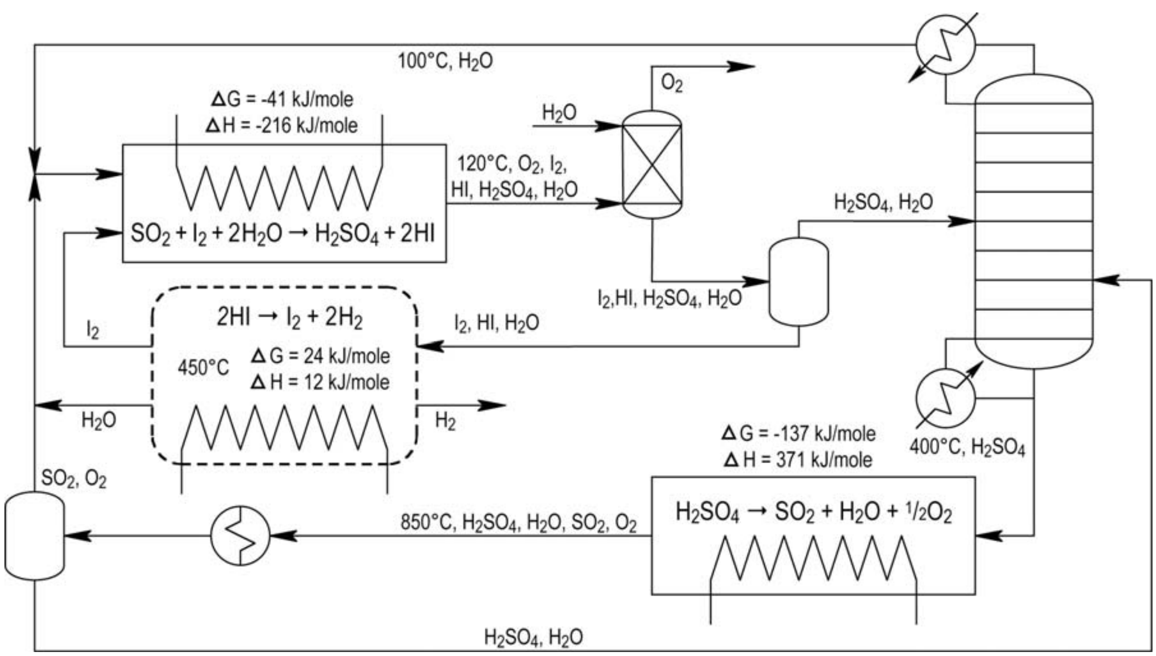


Fig. 13. Simplified schematic for sulfur iodine water splitting flowsheet for hydrogen production.

The net reaction of the SI cycle is the water splitting into hydrogen and oxygen:

$$H_2O \rightarrow H_2 + \frac{1}{2}O_2 \text{ (net reaction)} . \tag{4.4}$$

In the SI cycle, all process fluids are recycled, and no greenhouse gases are emitted. Also, the SI cycle has been fully flow-sheeted and operated at the bench scale in the United States and Japan. This cycle has the highest efficiency (~52%) of any thermochemical water splitting process that has been fully flow-sheeted. Since the hydrogen is produced at high pressure, it eliminates the necessity of compressing the hydrogen for pipeline transmission or other downstream processing. One of the most challenging issues regarding the SI cycle is the material issue, which comes from the high process temperature (800–1000ºC) and the corrosive reactants such as the sulfuric acid and hydrogen iodide.

4.4 High-Temperature Steam Electrolysis

High-temperature steam electrolysis (HTSE) is the electrolysis of steam at high temperatures. The total energy required, ΔH, which is composed of the required thermal energy, Q, and the Gibbs free energy (electrical energy demand), ΔG, is shown in Fig. 14 [Ref. 15]. The total energy increases slightly with temperature. The electricity demand, ΔG, decreases with increasing temperature leading to increased direct heat requirement. The decrease in electrical energy demand drives the thermal-to-hydrogen energy conversion efficiency to higher values, which is one of the primary advantages of HTSE. The higher temperature also favors electrode activity and helps lower the cathodic and anodic over-voltages. Therefore, it is possible to increase the electric current density at higher temperatures and consequently lower polarization losses, which yields an increase in process efficiency. Thus, the HTSE is advantageous from both thermodynamic and kinetic standpoints.

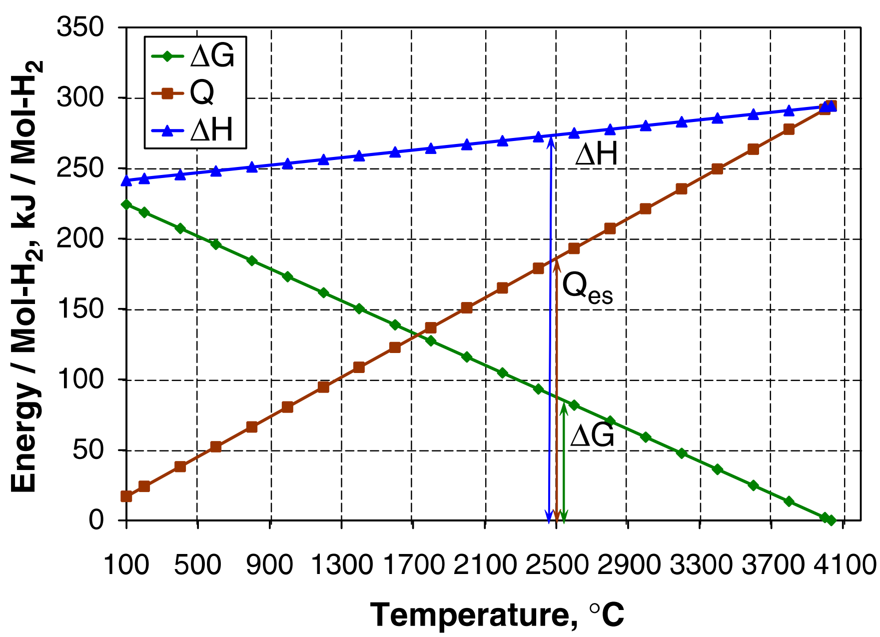


Fig. 14. Energy required for steam electrolysis.

The materials of the HTSE cell can be made of ceramics, which avoid corrosion problems. High-temperature steam electrolysis process using ceramic electrolysis cells is representative of the new advanced technologies. The reaction scheme in the HTSE process is the reverse of that in a solid oxide fuel cell, which is being developed vigorously for application in the power industry.[17] Water vapor molecules are dissociated at the porous cathode, producing an enriched H_2O/H_2 mixture, while the oxygen ions are transported through the nonporous, ion-conducting solid electrolyte to the porous anode where they recombine. Thus, the product gases, hydrogen and oxygen, are automatically separated by the solid electrolyte membrane. A representative electrolysis cell is illustrated in Fig. 15.

The HTSE is an environmentally friendly process in that only the gases H_2O, O_2, and H_2 are circulated in the electrolysis plant; no other chemicals are used that could raise safety concerns or lead to environmental problems.

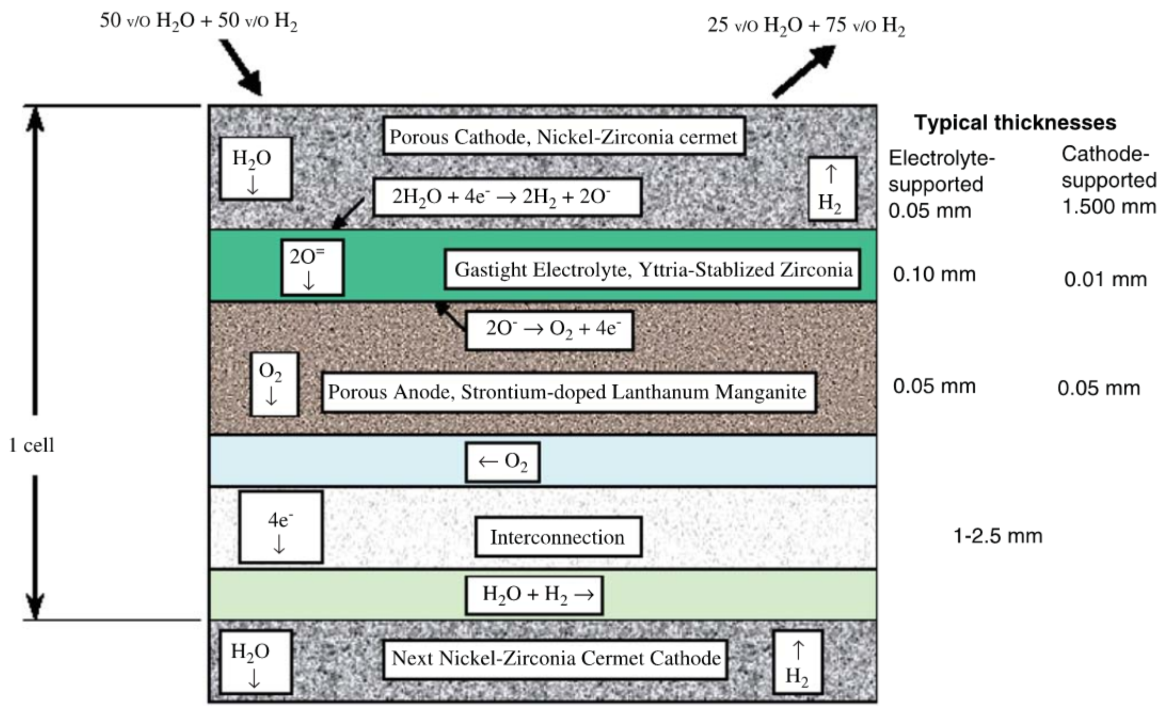


Fig. 15. Schematic for representative solid oxide electrolysis cell.

A possible plant configuration where an HTSE unit is employed is shown in Fig. 16.

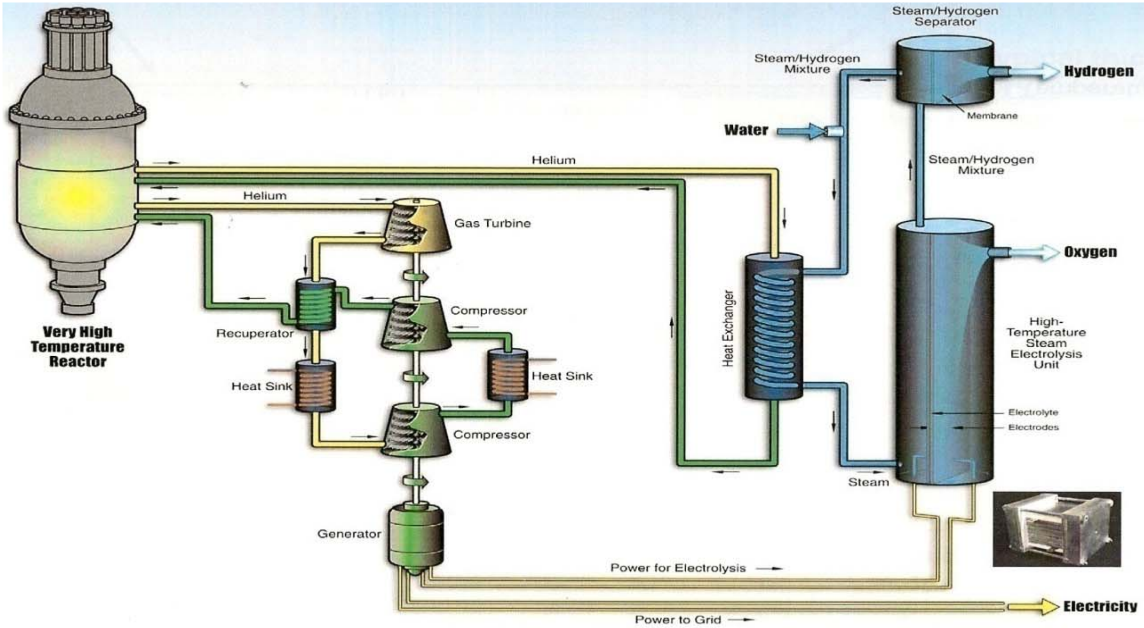


Fig. 16. Schematic of the high-temperature steam electrolysis (HTSE) plant.

4.5 Hybrid Cycles

A thermochemical hybrid process is a combined cycle with thermochemical and electrolysis reactions of water splitting. The hybrid process offers the possibility to run at low temperatures using electricity. One example of cycle-to-hybrid cycles is the *sulfuric acid hybrid cycle* or the *Westinghouse Sulfur Process* (WSP) (Fig. 17).

The WSP[18] has two reactions. From the two reactions electrolysis produces sulfuric acid and hydrogen from water and sulfur dioxide at low temperature. The thermodynamic properties of the chemical species are well known. The two reactions can be written as:

I. Thermochemical (800ºC):

$$\{H_2SO_4\}_{(aq)} \longrightarrow H_2O + \{SO_2\}_{(gas)} + \frac{1}{2}\{O_2\}_{(gas)}$$

II. Electrolysis (80ºC):

$$2\{H_2O\} + \{SO_2\}_{(gas)} \longrightarrow \{H_2SO_4\}_{(aq)} + H_2 - \{\text{electrolytic}\}$$

The first reaction, sulfuric acid decomposition, is the same reaction in the SI cycle.

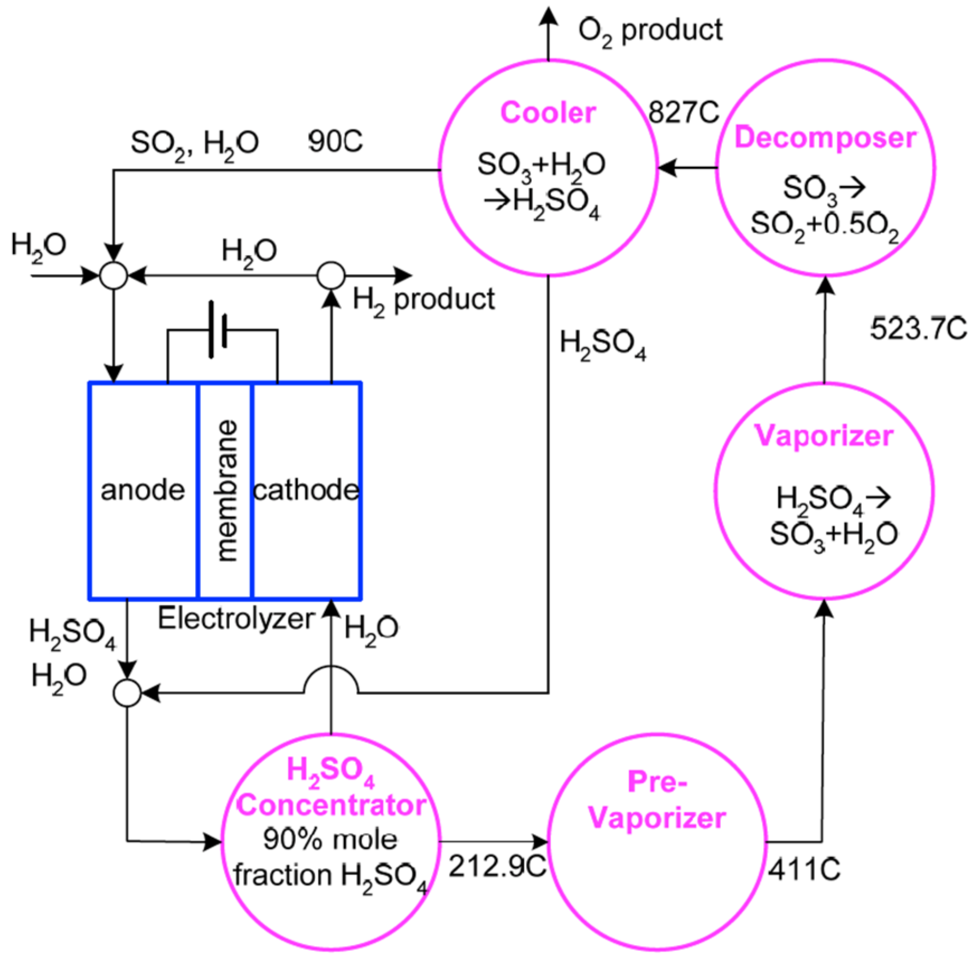


Fig. 17. Schematic diagram for the simplified hybrid sulfur model.

The hybrid sulfur cycle was the highest-ranked cycle from the preliminary screening process in previous Nuclear Energy Research Initiative (NERI) project.[19]

5. HEAT TRANSPORT SYSTEM INSTRUMENTATION

5.1 Introduction

The major NGNP design goal is to produce high-temperature process heat for nearby chemical plants as well as electricity. Because high-temperature heat can only be transported limited distances, the two plants will of necessity be relatively close to each other, but with enough separation distance to minimize potential damage emanating from one to affect the other. Typical separation distances are in the order of 500 m. In these deployment scenarios the reactor could be located at the site of large chemical or petrochemical processing plants. The industrial process heat plants present potential physical hazards to the reactor heat transfer components due to the required process coupling. Additionally, the physical security requirements for nuclear power reactors are significantly greater than for other industrial plants. Both the potential physical hazard to the reactor and the desired security boundary around the reactor provide incentives for maximizing the physical separation between the reactor and the heat consuming plant.

For the NGNP, it is possible that only a small fraction of the heat produced will be used for producing hydrogen or other chemicals, with most of the heat used to produce electricity. In contrast, for a commercial HTGR, all of the heat might be used for chemical production. Since the local chemical inventory determines the potential hazard to the nuclear plant from a chemical plant, the NGNP chemical plant may present much less of a hazard than a commercial system. The NGNP as a demonstration reactor may be connected to multiple generations of hydrogen production or other chemical production systems. Consequently, the safety analysis needs to envelope the safety implications of all of the different technologies.

The primary safety tenets for accidental releases at nuclear and chemical plants are almost entirely opposite. A basic design criterion at a nuclear plant is to contain radioactive material under all conditions. At chemical and petrochemical plants, in contrast, unplanned releases are often vented to the atmosphere (or flared if combustible) to disperse the chemicals to below harmful concentrations. The site layout and major structural elements of chemical and nuclear plants reflect these divergent philosophies. Nuclear plants are contained within strong structures, while chemical plants are frequently built outdoors to prevent trapping and hold-up of toxic or combustible materials.

The regulatory structures governing nuclear and chemical plants are very different. Nuclear plants are bound by Nuclear Regulatory Commission (NRC) rules, whereas chemical plants are governed by a combination of Environmental Protection Agency (EPA), Occupational Safety and Health Administration (OSHA), and state rules. Additionally, becoming an owner of a nuclear power plant would represent a significant increase in responsibility for a chemical company. Consequently, an HTGR, even though proximately located with and interconnected to a chemical plant, may have different ownership.

Given the differences between the nuclear and chemical plants, the NGNP and its hydrogen plant are most logically viewed as separate entities (i.e., the chemical process plant would be most effectively treated not as an extension of the nuclear plant but as an external facility that can impact reactor operation). In this scenario, accidents such as chemical releases would be treated as external events to the reactor.

5.2 Process Heat Plant Interface and Heat Transfer Loop Description

The intermediate heat transfer loop transfers energy from hot primary helium to the nearby chemical plant. The required separation distance such that an accident or incident at one plant does not adversely impact the other plant has not yet been established. The type of chemical plant and the heat transfer medium interconnecting the two plants have also not been selected. Both high-pressure vapor phase loops (steam and helium) as well as low-pressure liquid phase loops (fluoride salts, chloride salts, sodium, lead, and lead-bismuth) loops are candidate systems for heat transfer.

The efficiency, technological difficulty, and expense of transporting heat vary strongly with the heat transfer medium selected. Helium and steam heat transfer loops have substantial industrial pedigrees and may be the initial technology selected for the NGNP because of the lower development requirements. Vapor phase systems, however, contain much less energy per unit volume. Consequently, vapor phase systems require much higher flow velocities, larger pipes, and higher system pressures to transfer the same amount of heat as a liquid system. The combined high-flow velocity and high pressure increases the pressure drop per unit length of pipe providing strong incentive to minimize the loop length. The expense of the much thicker pipe walls of high-temperature alloys required for high-pressure vapor phase loops also argues strongly in favor of minimizing the length of the loop.

Low-pressure liquid loops do not have large pressure drops over reasonable heat transfer loop distances (<1 km) at the flow velocities typically employed. The thinner pipe walls and smaller pipe diameters of low-pressure liquids also decreases the expense of longer loops. However, high-temperature liquid-phase heat-transport systems are not yet commercial items and some development risk accrues with any of the

liquid phase coolants selected. Also, all of the liquid phase heat transfer materials are solids at ambient temperature. Thus, a loop preheating system would be required. Further, especially at higher temperatures, all of the liquids can be corrosive necessitating unconventional alloys and attention to fluid chemistry control. Overall, a fluoride salt heat transfer loop appears to be the most technologically appealing loop over the longer term with the proviso that the supporting technologies for liquid-fluoride salt-based heat transfer are less well proven than those for either helium or steam.

A number of trade studies on options for connecting a heat transport system to an NGNP for the purpose of producing hydrogen (or potentially for another high-temperature process heat application) have been performed.[20] In the reference, seven configuration options were identified, and a series of performance analyses was conducted. The selected configurations included both direct and indirect cycles for the production of electricity. All the options included an intermediate heat exchanger (IHX) to separate the operations and the safety functions of the nuclear and hydrogen plants. For the heat transport system that transfers heat from the reactor to a hydrogen production facility, both helium and liquid salts were considered as the working fluid.

Based on high-level engineering analysis, out of seven configurations, four options were eliminated and three were down-selected for further consideration. One of the viable options used a direct electrical cycle—the primary helium is sent directly to the turbines for electricity generation—and a parallel IHX as shown in Fig. 18. The process heat exchanger (PHX), which delivers heat to the hydrogen production facility, is directly connected to the IHX. This configuration offers the smallest mass and the highest thermal efficiency. In this configuration, the heat is transferred from the primary helium to the liquid salt through the IHX interface. Of the down-selected three configurations, two employ direct electricity generation cycles, and one uses indirect electricity generation cycle. In this report, the direct cycle with the highest thermodynamic efficiency (Fig. 18) and the indirect cycle (Fig. 19) were used as examples as available options that are being considered as well as to provide a contrasting picture from the electricity generation point of view.

Another option similar to the above used an indirect electrical cycle. The PHX was connected to a secondary heat exchanger (SHX), which is then connected to the IHX, as shown in Fig. 19. This configuration provided the better separation between the nuclear island and the process facility. Because of additional components, this option turned out with the largest mass and the lowest thermodynamic efficiency—within the down-selected options—as expected. However, considerations such as operation and license acquisition may favor this option notwithstanding the increased cost and engineering complexity.

The advantage of the configuration shown in Fig. 19 is the possibility of using another working fluid for the power cycle. Recent design studies using supercritical CO_2 (S-CO_2) demonstrate that these systems can deliver electricity with a high-thermal efficiency. If the liquid salt heat transport loop is connected to a heat source in this configuration, the heat is transferred from the secondary fluid to the liquid salt via an SHX.

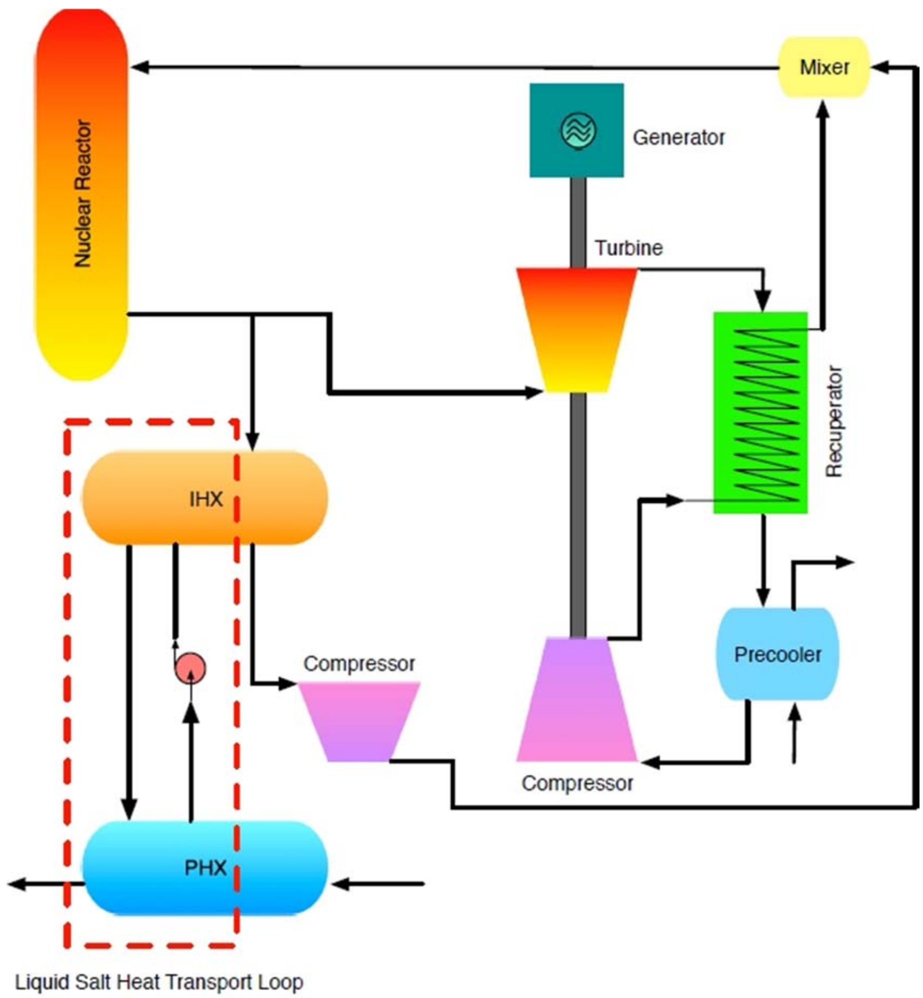


Fig. 18. Possible configuration option for a liquid-salt, heat-transport loop—direct electrical cycle and a parallel IHX.

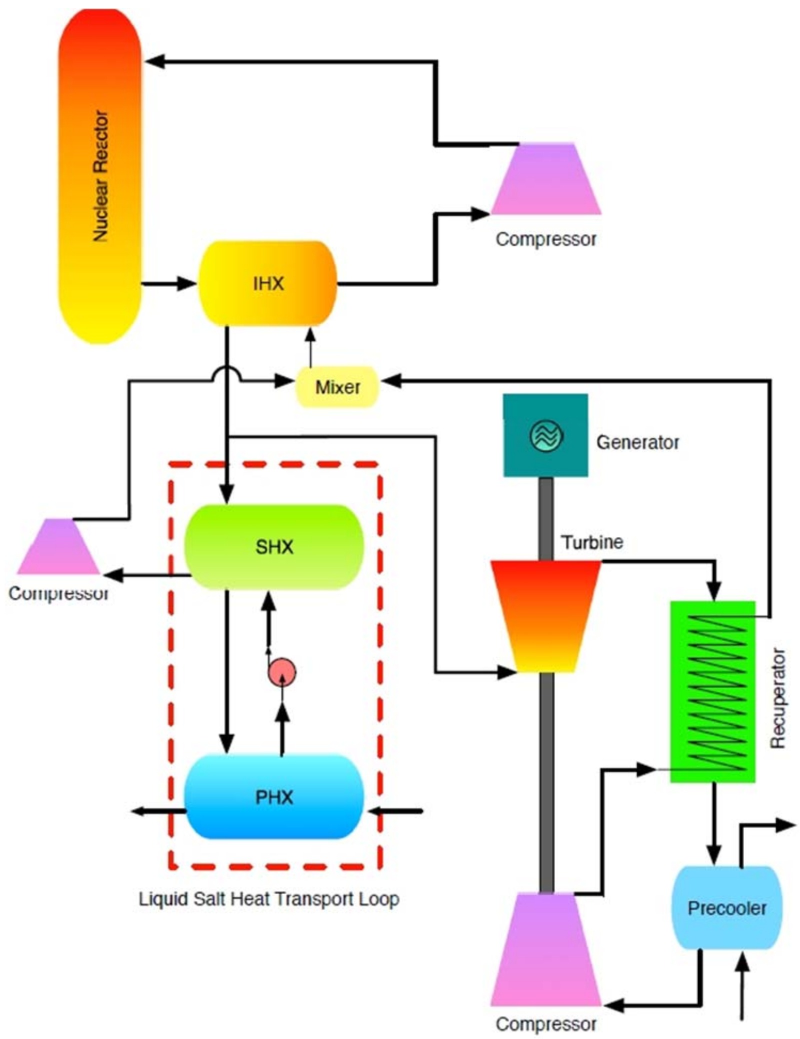


Fig. 19. Possible configuration option for a liquid-salt, heat-transfer loop—indirect electrical cycle and a parallel SHX.

5.3 Parametric Analysis of the Heat Transport System

The following calculations provide a highly simplified analysis of the design and comparative operational aspects of the several heat transfer fluid options for a high-temperature loop. The calculations are not of sufficient detail or fidelity design purposes and are intended only to illustrate the overall implications of selecting a particular heat transfer fluid.

Physical parameters used in the calculations are listed in Table 3. The heat rating of the source was taken to be $\dot{Q} = 125$ MW(t). The temperature rise across the heat source for each fluid is listed in Table 4. The separation between the heat source and the heat sink (e.g., hydrogen production facility), is taken to be 500 m, which gives a total pipe length of L = 1000 m. All fluid calculations were performed for an average fluid temperature of 700ºC, except for water—calculated at 290ºC—and steam—calculated at 300ºC. FLiNaK and sodium were considered at near atmospheric pressure; while water, steam, and helium properties and sodium were considered at near atmospheric pressure; while water, steam, and helium properties were taken at 7.5 MPa. A block diagram of a representative heat transport loop is shown in Fig. 20.

Table 3. Thermophysical parameters for fluids included in the analysis

Fluid	ρ (kg/m^3)	c_p (kJ/kg K)	ρc_p (kJ/m^{3}K)	$\mu \times 10^4$ (Pa s)	k (W/m K)
FLiNaK	2019.9	2.01	4060	29	0.60
Sodium	790	1.27	1000	1.9	62.0
Water*	732.3	5.49	4018	0.9	0.56
Steam†	37.4	4.73	176.9	0.2	0.063
Helium#	3.7	5.26	19.34	0.5	0.29

* 7.5 MPa, 290ºC
† 7.5 MPa, 300ºC
\# 7.5 MPa, 700ºC

Table 4. Selected temperature rise values for fluids included in the analysis

Fluid	ΔT (ºC)
FLiNaK	50
Sodium	100
Water	100
Steam	200
Helium	400

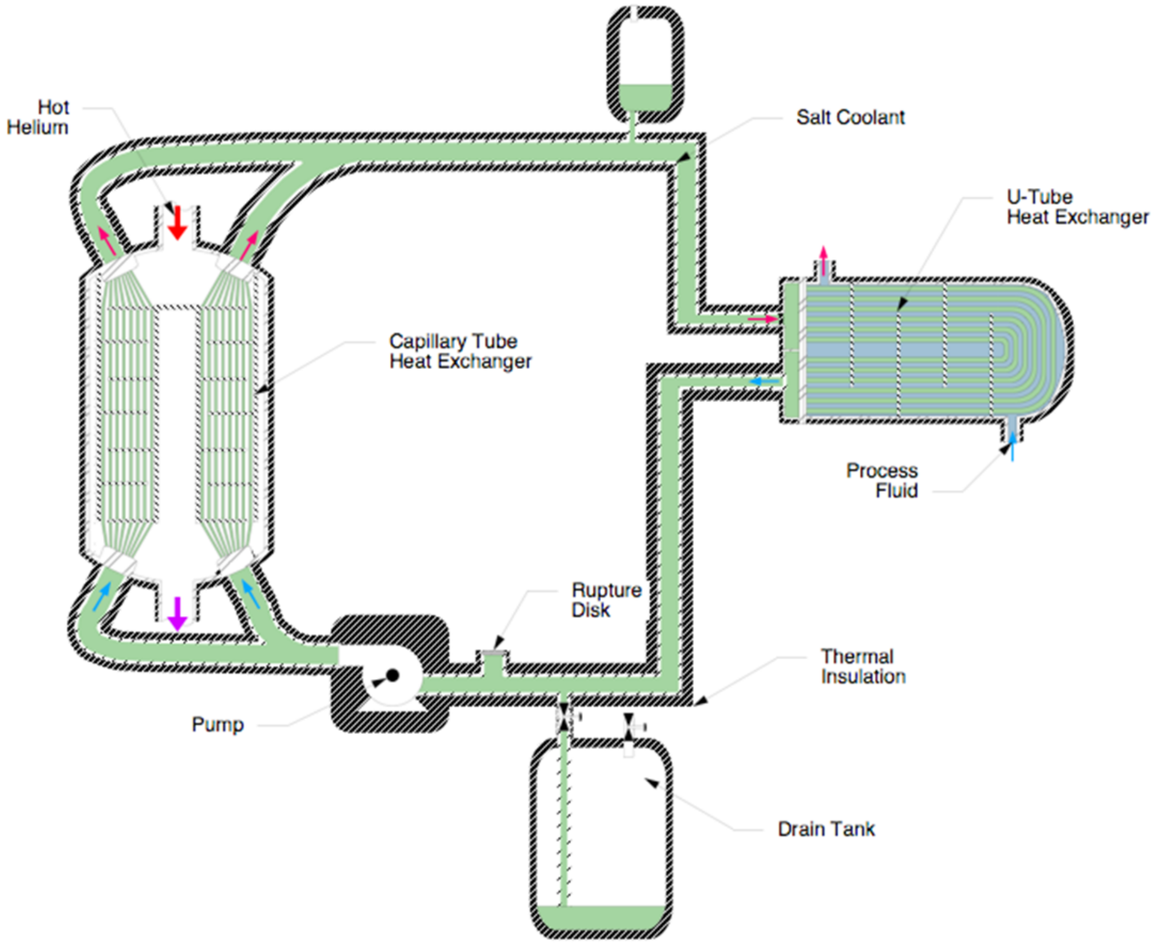


Fig. 20. Principal elements of an NGNP relevant heat transfer loop.

5.3.1 Flow requirements

The volumetric flow rate Q in m^3/s is calculated from the total energy balance using

$$Q = \frac{\dot{Q}}{\left(\rho c_p\right) \Delta T} \tag{1}$$

where $\dot{Q}$ is the thermal rating of the heat source in W, $\left(\rho c_p\right)$ is the volumetric heat capacity in J/m^3K, and ΔT is temperature rise across the heat source in ºC. The mass flow rate $\dot{m}$ in kg/s is calculated using

$$\dot{m} = \rho\, Q\,. \tag{2}$$

The required pipe diameter *D* that satisfies the flow rate and bulk fluid velocity requirements can be calculated by

$$D = 2\sqrt{\frac{Q}{\pi\, V}} \tag{3}$$

where *V* is the bulk fluid velocity in m/s.

For an estimation of the pipe diameter, a parametric analysis has been performed as a function of bulk fluid velocity. The fluid velocity is varied to find a channel dimension that yields a reasonable pumping power. Table 5 provides the results of the analysis—also plotted in Fig. 21. Values listed in bold indicate those combinations of fluid velocity and pipe diameter considered most reasonable.

Table 5. Required pipe diameter with respect to bulk fluid velocity

Bulk velocity (m/s)	FLiNaK Φ (m)	Sodium Φ (m)	Water Φ (m)	Steam Φ (m)	Helium Φ (m)
0.1	2.80	3.99	1.99	6.71	14.34
1.0	**0.89**	**1.26**	**0.63**	2.12	4.54
2.0	**0.63**	**0.89**	**0.45**	1.50	3.21
5.0	**0.40**	**0.56**	**0.28**	**0.95**	2.03
10.0	0.28	0.40	0.20	**0.67**	**1.43**
20.0	0.20	0.28	0.14	**0.47**	**1.01**
50.0	0.13	0.18	0.09	0.30	**0.64**
100.0	0.09	0.13	0.06	0.21	0.45

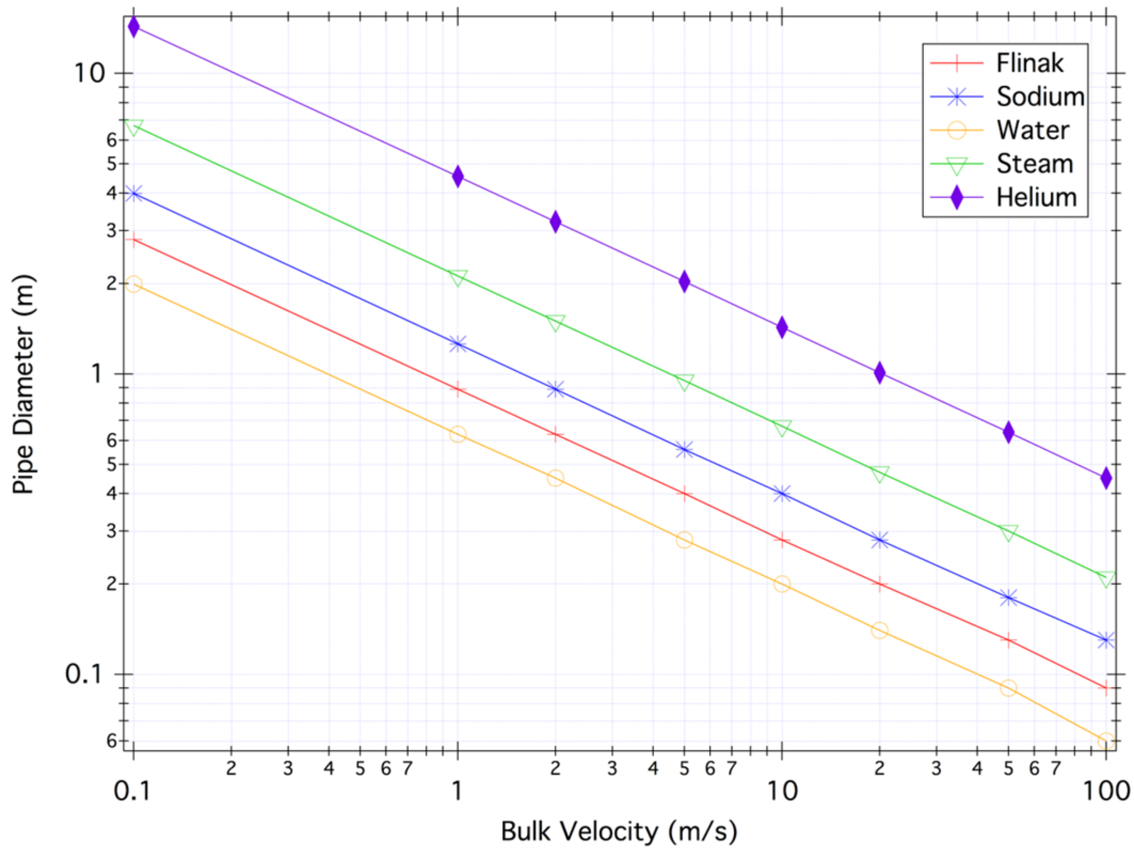


Fig. 21. Variation of pipe diameter as a function of bulk fluid velocity.

Note that the piping wall thickness—a primary piping cost differentiator—is not included in this estimate. The amount of metal volume necessary for the heat transport system piping can be calculated by

$$\mathbf{V}_{\text{pipe}} = \frac{\pi}{4}\left[(D+2w)^2 - D^2\right]L \tag{4}$$

where $\mathbf{V}_{\text{pipe}}$ is the metal volume of the piping in m^3, D is the pipe inner diameter in m, w is the wall thickness of the pipe, and L is the total length of the pipe. With some algebraic operations, Eq. (4) can be reduced to

$$\mathbf{V}_{\text{pipe}} = \pi\, L\, w\, (D+w) \tag{5}$$

For sufficiently large pipe diameters (i.e., $w << D$), it is possible to state

$$\mathbf{V}_{\text{pipe}} \propto w \tag{6}$$

The high-pressure water, steam, or helium systems will require much thicker piping walls than do the low-pressure sodium and FLiNaK. As shown in Eq. (6), the piping mass will increase in proportion with the wall thickness resulting in higher capital expenses, with all other considerations being similar.

5.3.2 Pressure loss

The two main components of pressure drop along the flow loop are frictional and form pressure drops. The form losses for the loop estimate consist of eight 90-degree pipe bends between the heat source and the heat sink. The friction pressure drop is calculated by

$$\Delta p_{fric} = f \left(\frac{L}{D}\right) \frac{\rho V^2}{2} , \tag{7}$$

where f is the friction factor, L is the channel length, D is the pipe diameter. The form pressure drops are irrecoverable energy losses due to sudden change in geometry of the channel or direction of the fluid. They are calculated using:

$$\Delta p_{form} = K \frac{G^2}{\rho} , \tag{8}$$

where K is the form factor, G is the mass flux in kg/m^{2}s and ρ is the fluid density. The form factor for 90-degree turns was taken $K = 0.9$. The total pressure drop is the sum of frictional and form pressure drops. Figure 22 shows the variance of the loop pressure drop with the fluid velocity for each of the evaluated heat transport media.

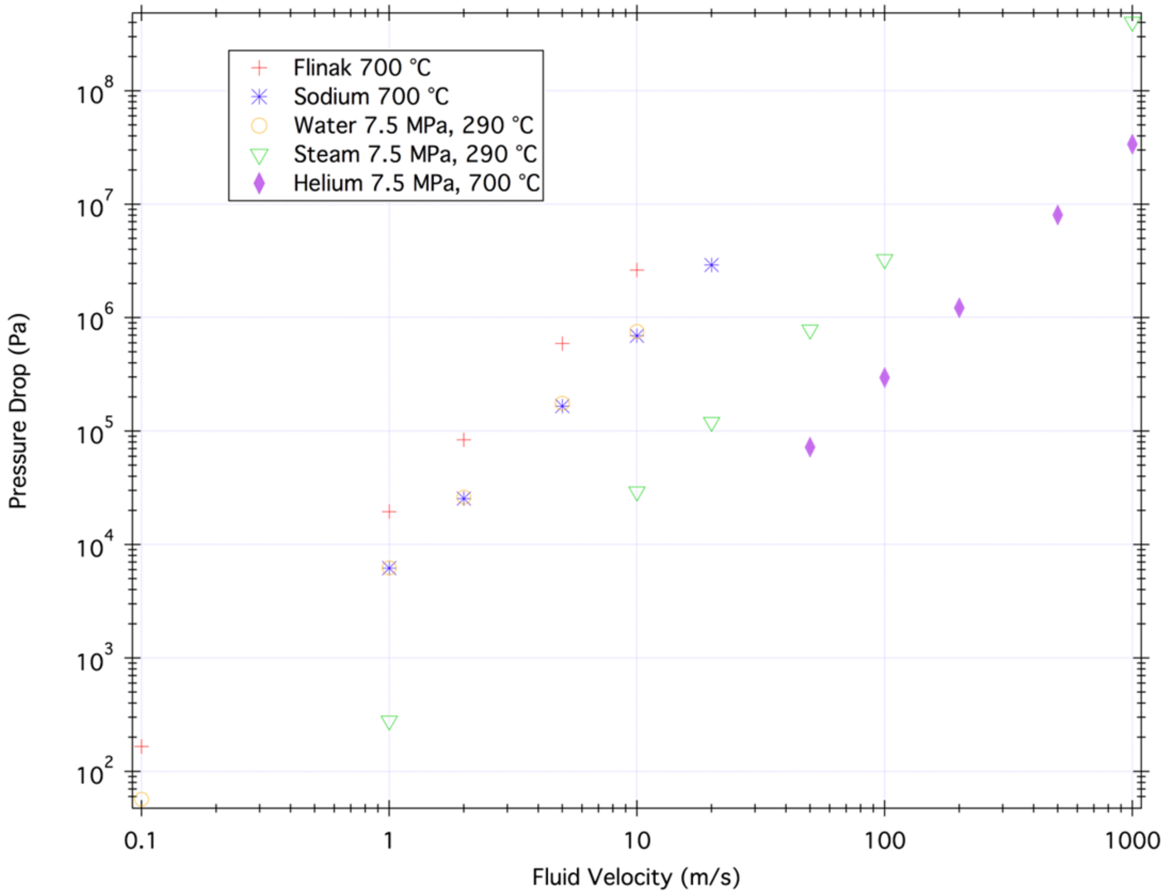


Fig. 22. Variation of total pressure drop with respect to bulk fluid velocity.

5.3.3 Pumping power

The required hydrodynamic pumping power is calculated using the following equation:

$$P_{\text{pump}} = Q\,\Delta P_{\text{total}} \;, \tag{9}$$

where P_{pump} is the pumping power in kW, Q is the volumetric flow rate in m^3/s, and ΔP_{total} is the total pressure drop in Pa.

Table 6 lists the key quantities calculated with respect to the parameterized bulk fluid velocity for a number of fluids that can be considered as the heat transport medium. Pump power calculations do not include head losses due to elevation differences between heat source and the heat sink.

The pressure drop (markers), pumping power (solid lines) and resulting fluid velocity (dashed line with markers) for each candidate fluids to transfer the required amount of heat as a function of pipe diameter is shown in Fig. 23.

Table 6. Calculated thermal fluid quantities for selected fluids at various bulk velocities

V **(m/s)**	***D*** **(m)**	Δp_{fric} **(kPa)**	Δp_{form} **(kPa)**	$\Delta p_{\mathrm{total}}$ **(kPa)**	P_{pump} **(kW)**
FLiNaK					
1.00	0.89	3.22	14.5	17.8	10.9
2.00	0.63	16.7	58.2	74.9	46.1
5.00	0.40	147	364	511	315
10.0	0.28	764	1,450	2,220	1,370
Sodium					
1.00	1.26	0.52	5.69	6.20	7.75
2.00	0.89	2.67	22.8	25.4	31.8
5.00	0.56	23.5	142	166	207
10.0	0.40	122	569	691	864
Water					
1.00	0.63	0.97	5.27	6.24	1.94
2.00	0.45	5.01	21.1	26.1	8.12
5.00	0.28	44.2	132	176	54.8
10.0	0.20	229	527	756	235
Steam					
10.0	0.67	3.69	26.9	30.6	108
20.0	0.47	19.1	108	127	448
50.0	0.30	169	673	842	2,970
Helium					
10.0	1.43	0.31	2.65	2.96	47.8
20.0	1.01	1.60	10.6	12.2	197
50.0	0.64	14.1	66.2	80.3	1,300

These calculations were not performed based on optimal parameters.

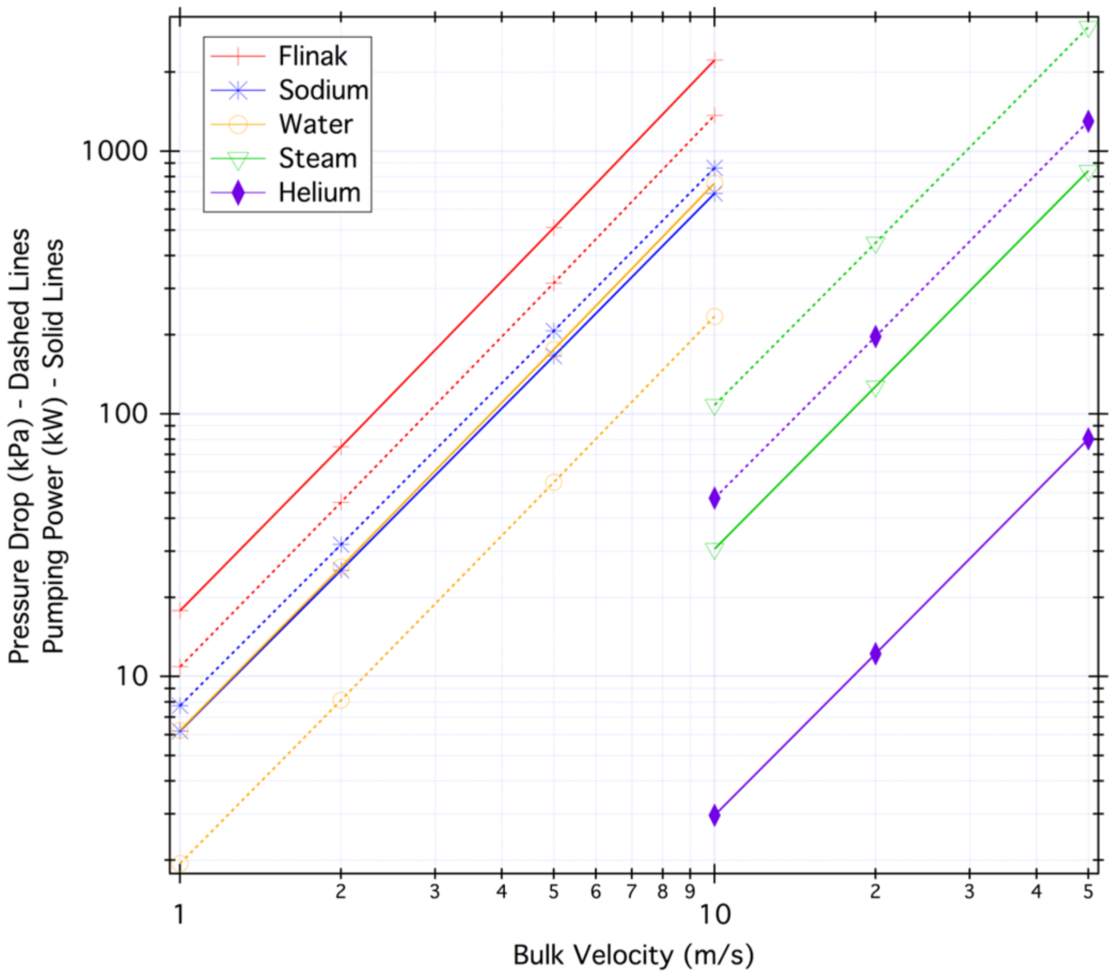


Fig. 23. Required pumping power for selected fluids as a function of fluid velocity.

5.4 Safety Issues

A recent phenomena identification and ranking tables (PIRT) report[21] provides descriptions of safety issues relevant to interconnecting HTGRs with chemical plants. The discussion of accident scenarios included here is based upon those introduced in Ref. 21 with added emphasis on measurement and communication requirements.

Several different technologies are candidates for hydrogen generation, some of which use hot, high-pressure caustic fluids. Only reforming hydrogen from natural gas requires large-volumes of high chemical energy fluid. If oxygen is produced along with the hydrogen (i.e., from high-temperature electrolysis) and stored on site, the oxygen also represents a potential explosion hazard. However, if both natural gas and oxygen were to disperse in the atmosphere, the explosion hazard would be local to the chemical plant. Hydrogen itself is also combustible but disperses readily, rising into the atmosphere. Hydrogen is difficult to bring into a combustible mixture with oxygen outside of a confined space, thus the hydrogen presents negligible hazard outside of the chemical plant.

Some of the potential chemical processes at an HTGR coupled chemical plant involve heavy gases that can form ground-hugging plumes upon release. Oxygen released from storage can also form a hazardous plume before dispersing. The heavy gases can be both toxic and caustic and present a hazard to personnel at the adjacent nuclear plant.

HTGRs have a high-pressure helium primary loop operating at high temperature. The lowest cost, most energy efficient heat transport loops from the HTGR primary to a nearby chemical plant employ low-pressure liquids as the heat transfer medium. Minimizing heat exchange surface thickness (tube wall thickness in a shell and tube heat exchanger) is a key heat transfer efficiency goal. In addition to temperature and pressure stressors, the high temperature heat transfer surface will have different chemical environments and, consequently, corrosion potentials on opposing sides. Thus, heat transfer surface rupture represents a potential primary system failure mode.

If the NGNP elects to employ a high-pressure helium or steam intermediate heat transfer medium, the pressure drop across the heat exchanger will be less as compared to a low-pressure liquid under normal mode operation enabling thinner walls. If, however, either the primary or secondary systems decreases in pressure, the heat exchanger walls could then be subjected to a large pressure differential potentially rupturing the thinner walls.

Rupturing the primary boundary is a loss-of-cooling accident, a means for radioactivity to leave controlled space, as well as a potential means for nonintended fluids to enter the reactor core. Steam is a particular hazard as it can cause a positive reactivity input and will react with the core graphite at high-temperatures. None of the potential low-pressure heat transfer fluids—fluoride or chloride salts, sodium, lead, or lead-bismuth—would provide significant reactivity additions to the core or are normally at high pressures that would enable transport into the core.

If the secondary heat exchanger at the chemical plant was to fail, the intermediate heat transfer loop could undergo rapid unplanned emptying (blowdown). Blowing down the heat transfer loop would remove the heat sink from the reactor, providing a stressor to primary components normally at lower temperatures.

The heat transfer fluids themselves can represent physical and chemical hazards. Rupture of the intermediate loop, if the fluid is at high pressure, represents the largest physical hazard. Hot sodium reacts violently with water. Rupturing a large, pumped, high-temperature sodium loop thus represents a significant thermal and chemical hazard to the immediate surroundings. Hot fluids have significant stored thermal energies. A pumped or pressure driven spray of heat transfer fluid thus represents a local personnel hazard.

Liquid heat transfer lines require connection to an expansion volume to avoid pressure spikes. If the interconnection between the expansion volume and the loop were to become clogged, large pressure transients could occur in the normally low-pressure loop. Pressure relief mechanisms (e.g., rupture disks) are employed to mitigate the pressure transient hazard.

High-pressure heat transfer loops have significant stored potential energy. Rupture of any high-pressure variant of the intermediate heat transfer loop can potentially result in a missile impacting reactor structures. The pipe chase between the chemical plant and nuclear plant represents a potential intrusion route through the nuclear plant security boundary. Thus, a dual-purpose missile shield and intrusion prevention boundary is a necessary part of the loop design.

5.5 Measurement and Information Issues

Under normal operating conditions the primary information transferred from the chemical plant to the nuclear plant operators is the heat demand. While HTGRs are naturally load following, receiving a load request signal enables a more rapid nuclear plant response that minimizes temperature temporal variances and, thereby, decreases stress on the plant equipment. An erroneous load demand signal will result in increased stress on the plant equipment as the reactor thermal feedback adjusts to the actual load.

Loss of heat sink (e.g., from blowdown of the intermediate heat transfer loop) is functionally similar to an erroneous load request. The combination of temperature and flow in the intermediate heat transfer loop

provides confirmation of proper heat transfer. The temperature and flow signals would not be required to be safety grade as a loss of heat sink is not an initiator of radiation release in a passively safe plant.

Under chemical plant severe accident conditions, a ground-hugging chemical plume can approach the nuclear plant. Air intake and upstream chemical environmental monitoring signals are thus necessary to assure that the nuclear control room is isolated from any airborne chemical contamination and to provide warning to plant personnel of the hazardous conditions. As control room environmental isolation is an active response, confirmation of the isolation (typically by differential pressure monitoring across the one-way air valve) is also required.

Pressure mismatch across the heat exchanger would be a safety-grade measurement for designs in which the heat exchanger heat transfer surface thickness has been decreased. For example, in high-pressure helium to high-pressure steam heat exchanger, loss of steam will increase the differential pressure across the heat exchange surface. If the heat exchange surface has not been designed to take full primary system pressure, the differential pressure transient may lead to primary pressure boundary rupture.

An expansion tank needs to be provided for in an incompressible fluid-based heat transfer loop. If the interconnection between the tank and the loop becomes blocked, the loop becomes vulnerable to pressure spikes and consequent mechanical failures. Liquid level measurement within the expansion tank of sufficient precision to observe normal liquid level shifts and, thereby, avoid stuck indicator errors is required to confirm flow.

As all of the candidate liquid heat transfer media have freezing points well above ambient, an expansion tank vulnerability would be the freezing up of the tank and/or the interconnecting piping. Tank and interconnection line temperature measurements are thus also necessary. A tank and loop heating system will also be necessary prior to loop filling or for longer-term reactor shutdown conditions if undrained. The expansion tank is also likely to include the gas vent employed during loop filling and the backpressure gas source to empty the loop for maintenance. As the gas connection port is a potential vent under accident conditions, both flow and radiation measurements will be necessary on the gas line interconnection.

Corrosion occurs more rapidly at higher temperatures. Structural measurements of the heat transfer surface between the primary and secondary fluids need to be performed sufficiently regularly so as to assure that the material strength has not been compromised. In particular, if the pressure boundary includes an anticorrosion cladding, the integrity and bonding of the cladding needs to be assured. The heat transfer media become much more corrosive under improperly maintained chemical conditions. For example, the fluorine potential (and corrosivity) of fluoride salt loops increases several fold if an electronegative impurity such as oxygen from environmental water is permitted to contaminate the system. Thus, high-quality intermediate loop chemical condition monitoring is required.

A leak of hydrogenous material into the core is the most hazardous accident identified for an HTGR. Large-scale water leakage into the core is only a significant concern for those designs that include a primary loop steam generator or a steam/water-based heat transfer loop. In addition to periodic corrosion surveillance measurements, safety-grade water ingress measurements need to be provided for in the primary loop. Additionally acoustic monitoring of the IHX would be a useful diverse measurement to confirm the leak existence and diagnose its size.

The piping chase between the nuclear and chemical plant represents an intrusion path into the nuclear plant. The pipe chase needs to be in the plant security plan and include intrusion monitoring. A physical barrier within the pipe chase would be a useful adjunct to the intrusion monitoring.

5.6 Steam Instrumentation

A water-steam heat transport loop would have many of the same features as a PWR secondary side or a fossil-fuel boiler—hence, most of the required steam instrumentation would not be distinctive, and much of the relevant information could be obtained from standard overviews of steam cycle instrumentation and control such as that available in a comprehensive book by Lindsey.[22]

Some of the recent NGNP candidate designs, however, are focusing on steam generator designs such as those developed by General Atomics in the 1980s and shortly thereafter. Of those, the direct cycle (steam generator in the primary loop) versions are also being favored for a number of reasons, but mainly because they would avoid the problems associated with development, cost, and operation of an IHX.[23] For electrical production, these Rankine direct cycle designs have the potential for significant improvements in efficiency (~20%) over LWRs due to the higher steam temperatures (Fig. 24), so their development is well-warranted for both process heat and electrical production applications.

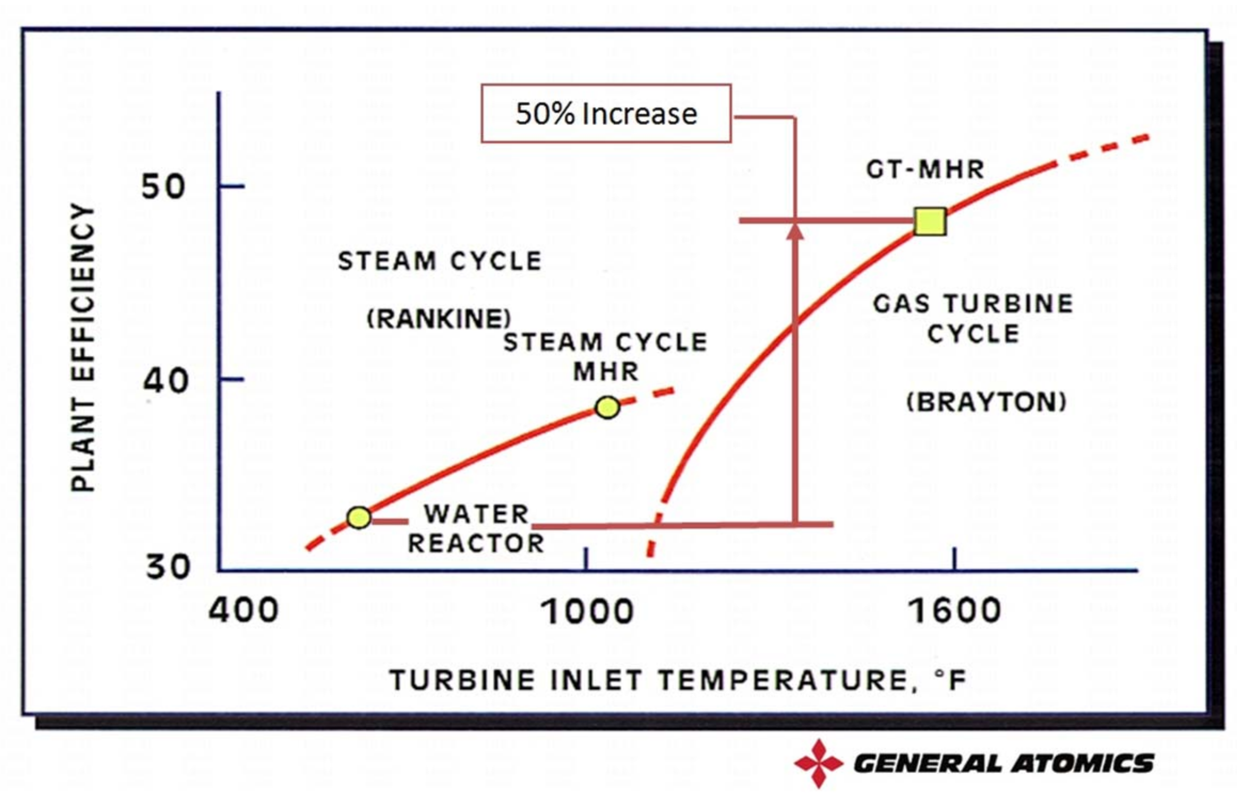


Fig. 24. Electrical production efficiency as a function of turbine inlet temperature.

There are several variations of the higher-temperature steam generator designs, but a typical feature is the high-temperature section needed to produce the final steam outlet temperature, which requires special high-temperature materials. An example design is shown in Fig. 25.

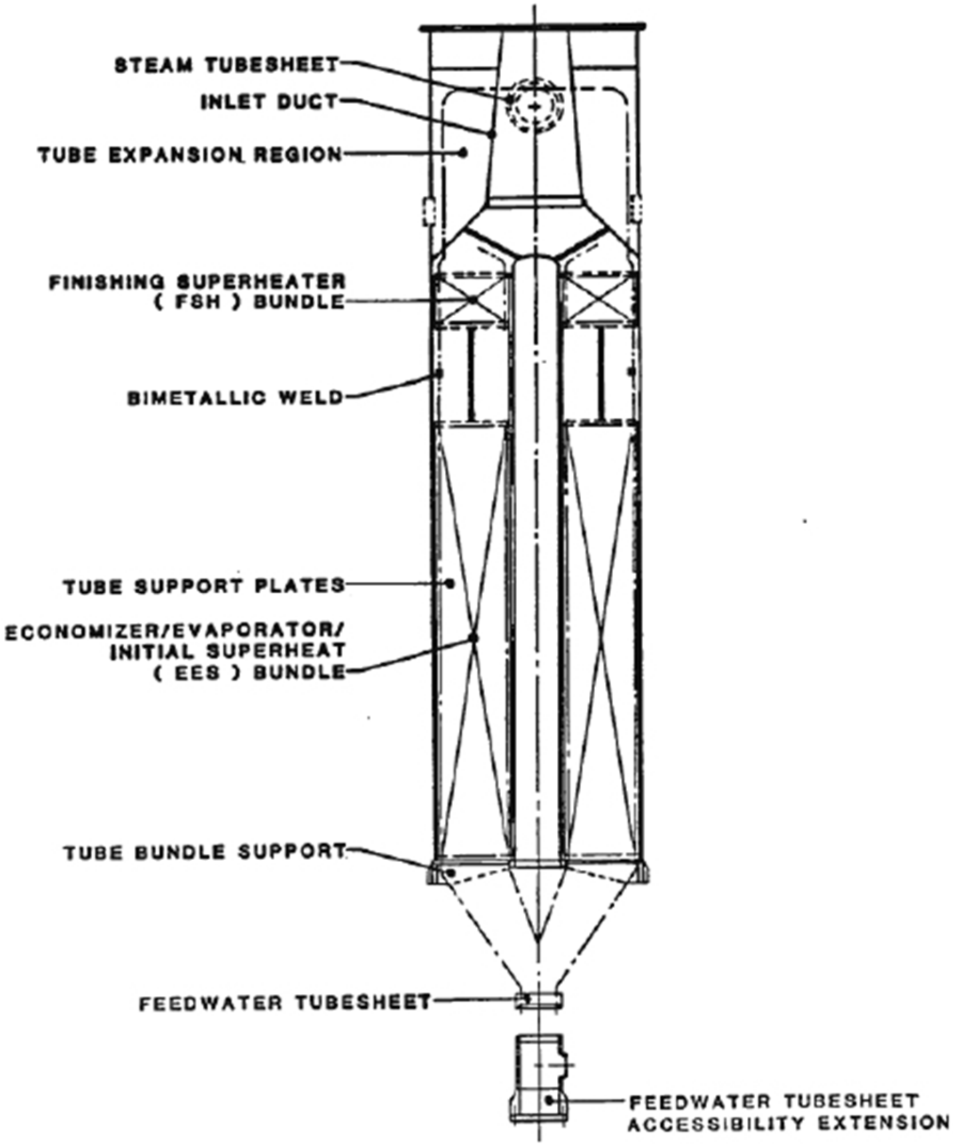


Fig. 25. Typical once-through steam generator employing a finishing superheater section.

The steam generator is a vertically oriented, up-flow boiling, cross-counter flow, once-through shell-and-tube heat exchanger that utilizes multiple tube, helically wound tube bundles. The design shown employs two sets of bundles, where the lower bundle contains economizer, evaporator, and initial superheater sections using 2-1/4 Cr–1 Mo material for the tubing. The upper bundle that contains a finishing superheater section uses the higher temperature Inconel 617 material. A bimetallic weld located between the two bundles is required to join the two dissimilar tube materials.

Locating the steam generator in the primary circuit raises a number of safety concerns that impact the I&C requirements. The major concern is for water/steam ingress into the primary coolant system due to the fact that the secondary (water) side operating pressure is much greater than the helium coolant pressure. Steam contacting the hot core has a number of adverse effects, including corrosion of the graphite as a result of CO, CO_2, and CH_4 production. A detailed analysis of these issues[24] was generated

for the reference steam-cycle modular helium reactor (SC-MHR) plant. Regarding moisture ingress into the primary coolant system from steam generator leakage, the GA analysis predicted that, while a concern, as long as the prescribed corrective actions are taken, it is not expected to result in unacceptable average or localized oxidation of either the bulk core moderator graphite or the graphite core support components, and leakage is not expected to result in radionuclide releases in excess of regulatory limits.

The alternative (i.e., placement of the steam generator in a secondary loop connected to the nuclear heat source through an IHX), would eliminate most issues associated with moisture ingress into the core; however, there are also safety-related and other issues associated with including an IHX in the primary circuit. These include the following.

a. The probability of a major pressure difference developing between the primary and secondary sections of an IHX. Either the IHX would have to be designed as a Class I primary pressure boundary component, or the secondary system must contain Class I isolation valves near the IHX, or the secondary system must be designed as the primary pressure boundary.
b. Loss of secondary helium flow without tripping the primary helium flow would result in rapid IHX heatup with possible damage to the IHX internals.
c. There is uncertainty that an IHX can be designed as a Class 1 component having a reasonable lifetime, taking into account the creep fatigue damage caused by occasional high-pressure differentials at the high-operating temperatures.
d. It is not certain that suitable isolation valves could be developed. No suitable designs of large-size, high-temperature helium leak-tight valves are currently available.

A reactor protection system (including a related investment protection system) would have a number of features related to steam production.

a. Detect and provide corrective action if the moisture level in the primary circuit indicated steam inleakage. In the case of multiple steam generators in the loop, the moisture sensing system must be able to determine which steam generator is leaking and initiate steam generator isolation and dump on the appropriate module.
b. Detect and provide corrective action if changes in the reactor building (including changes in temperature, pressure, and radiation levels) indicate the presence of primary coolant or steam at levels that could potentially expose the general public to low-level radiation effects.
c. Detect and provide corrective action if conditions of pressure, temperature, or flow indicate an interruption of normal cooling functions or steam leakages.
d. Detect and provide corrective action if conditions of pressure and temperature, within and around the vessel system (VS) primary coolant boundary, indicate a level of operation that exceeds the normal VS design levels.
e. Detect and provide corrective action if conditions of environment or service to the reactor system indicate potential interruption of processes necessary to protect the reactor (e.g., non-1E electric systems) and are not suited for a particular environmental event.

Figure 26 [from Ref. 23] shows typical system protection logic for the case where the steam generator is in the primary system. Of particular note are the steam generator isolation and dump and reactor building isolation functions.

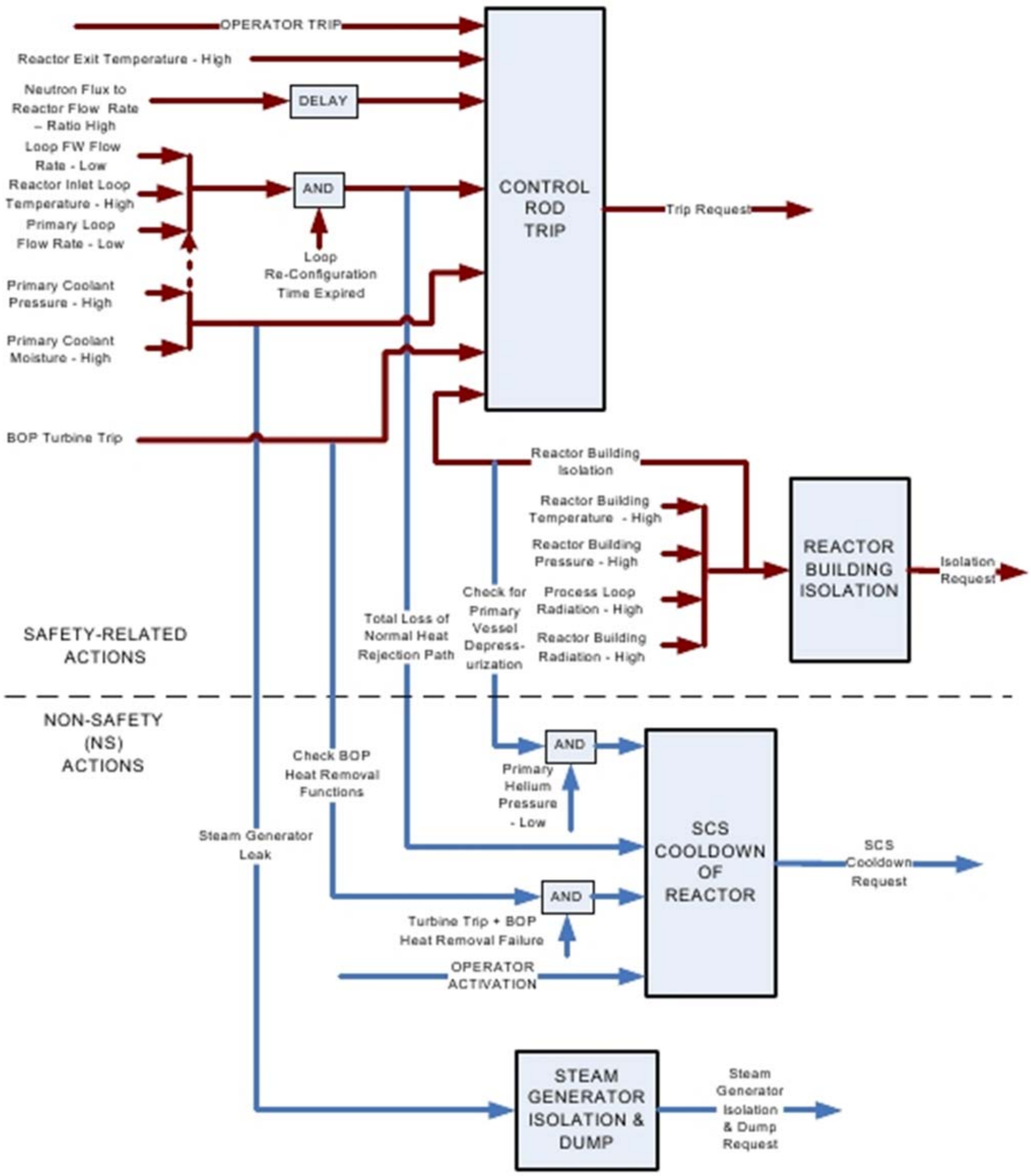


Fig. 26. Integration of steam related protection systems—steam generator in primary loop.

Figure 27 shows the protection logic for the case where the steam generator is in the secondary loop, isolating the primary from potential steam ingress accidents directly impacting the core. As in the previous figure, the red lines indicate safety-related actions, while the blue represent nonsafety functions.

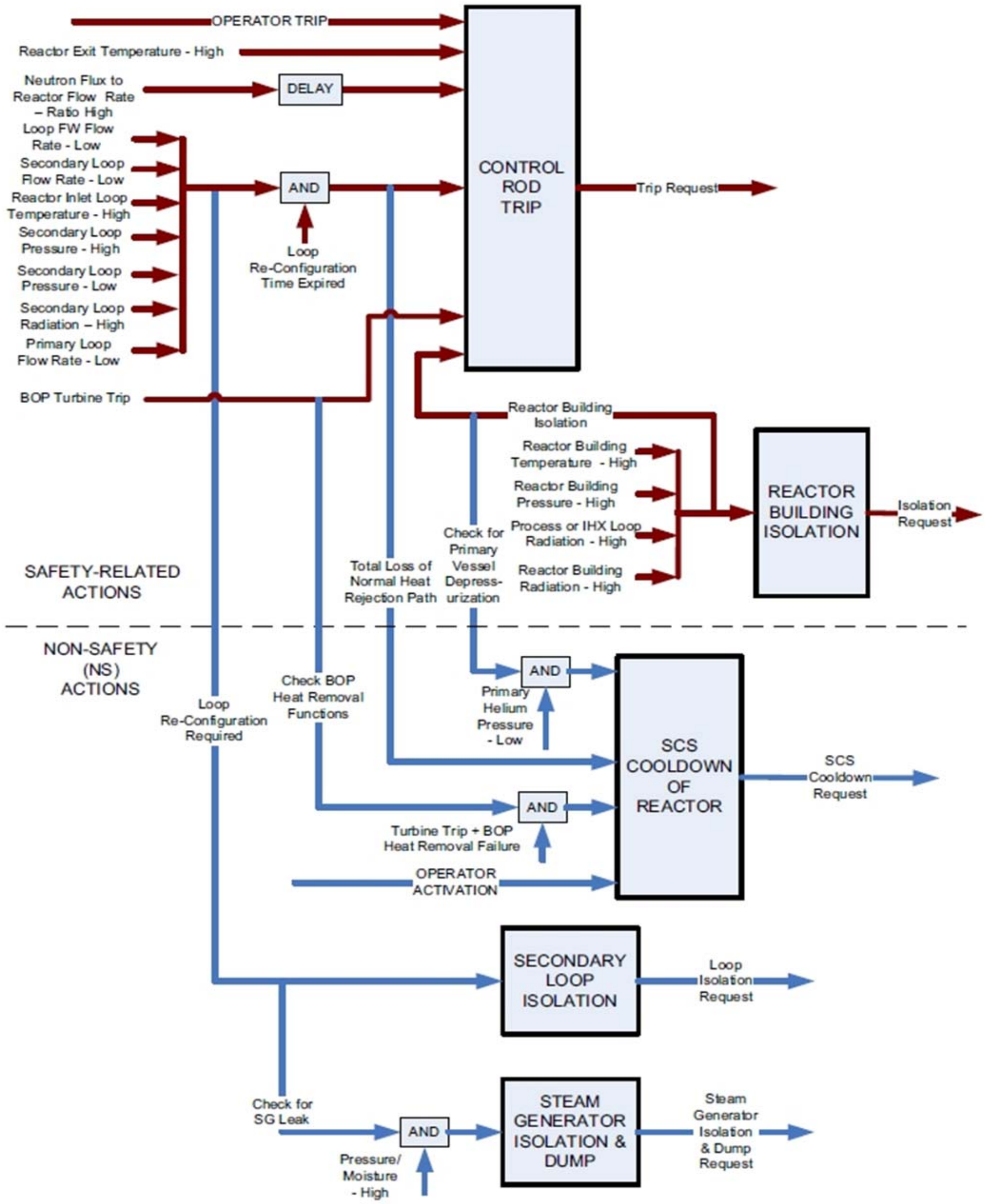


Fig. 27. Steam-related protection system logic for steam generator in secondary loop.

A related design issue for NGNP is tritium control.[24] Tritium is produced in HTGRs by various nuclear reactions. Given its high mobility, especially at high temperatures, some tritium will permeate through the IHX, steam generator, and hydrogen (or other) process vessels contaminating the product (hydrogen) and process steam. Tritium contamination contributes to public and occupational radiation exposures; consequently, stringent limits on tritium contamination in the product of the process heat system are anticipated. Design options are available to control tritium in an HTGR, but they can be very expensive,

so an optimal combination of mitigating features must be implemented in the design. It would be easier to control tritium transport to NGNP end products if the steam generator were located in a secondary loop (rather than a primary loop) because this configuration would allow for inclusion of a second helium purification system in the secondary loop to remove tritium; however, tritium control will be manageable regardless of whether the steam generator is located within a primary or secondary loop.

5.7 Liquid Salt Loop

5.7.1 Liquid salt system description

Functionally, all heat transfer loops consist of a heat source, a heat sink, and a heat-transfer mechanism. A typical liquid salt heat transfer (LSHT) loop, consisting of a single-phase, incompressible liquid coolant, is required to have an expansion volume to prevent pressure spikes. A drain tank is also necessary to enable initial filling and to allow for servicing. Since the fluoride salts have melt points well above ambient loop, preheating is also required. Further, chemistry control is required since fluoride salts only maintain their relatively inert nature when the free fluorine potential is minimized. Further, since a primary purpose for an LSHT loop is to physically separate two energetic processes (possibly at high pressure), sufficient physical loop length is required to prevent severe accidents (blast wave, fire, caustic chemicals) from propagating between the heat source and heat sink processes, and pressure relief mechanisms are required to prevent the liquid salt itself from propagating the accident. Additionally, nuclear security requirements will necessitate either applying nuclear power level security to the heat sink process plant or providing sufficient distance between the reactor and the process plant to incorporate a security boundary.

At the high temperatures of the NGNP, no standard heat transport system is yet available. The desirable physical properties of liquid fluoride salts make them the leading candidate fluid as an improvement over steam to couple the reactor energy to an industrial process heat system. The leading candidate heat transfer salt for NGNP application is a mixture of lithium fluoride, sodium fluoride, and potassium fluoride (46.5-11.5-42 mol %) referred to as FLiNaK. A primary advantage of liquid fluoride salts is their high boiling points (>1400°C for relevant salts) and the consequent low system pressure at operating temperatures. Liquid fluoride salts are composed of the most electronegative element and highly electropositive elements resulting in highly chemically stable compounds that have low reactivity with the environment. Fluoride salts have viscosities a few times that of room temperature water at NGNP operating temperatures and a comparable heat capacity per unit volume to room temperature water resulting in small volumetric pumping requirements and low pressure drop during flow. Relevant FLiNaK heat transfer properties are provided in [Ref. 25].

Table 7. FLiNaK heat transfer properties

Melting point (°C)	454°C
Density (g/cm^3) at 700°C	2.02
Viscosity (mPa-s) at 700°C	2.9
Heat capacity (J/(K-g)) at 700°C	1.884
Thermal conductivity (W/(m-K)) at 700°C	0.92
Volumetric expansion (1/K) at 700°C	3.61×10^{-4}

An overview of LSHT technology and issues is available in an ORNL overview report.[26]

5.7.2 Liquid salt loop instrumentation

LSHT loop operations require measurement of a broad set of process variables including temperature, flow, and level. Coolant chemistry measurements (as a corrosion indicator) and component health monitoring are also important for longer-term operation.

5.7.2.1 Temperature

Temperature measurement is indicative of both process conditions as well as a primary component of the energy transfer measurements necessary for efficient power plant operation. Thermocouples are the most common transducer for process temperature measurement. However, base metal thermocouples lack the long-term accuracy necessary for the heat balance measurements necessary for efficient process operation. Precious metal thermocouples are a possible higher accuracy temperature measurement alternative. Alternatively, two optical instruments potentially have the stability and accuracy required to characterize the heat transfer with low enough uncertainty to maximize efficiency. Both fiber optic coupled pyrometry and Fizeau cavity-type thermometers are candidate technologies for high-accuracy temperature measurement. Ultrasonic wireline thermometry is also a strong candidate technology for low-uncertainty temperature measurement at elevated temperatures.

Although the field of ultrasonic temperature measurement has many embodiments, wireline, pulse-echo ultrasonic sensor is especially applicable to aggressive environment temperature measurement due to its rugged nature. Ultrasonic wireline thermometry has been demonstrated as early as the 1960s in nuclear applications within as severe an environment as molten corium.[27] A review of the technology stressing nuclear power applications was published in 1972 [Ref. 28]. More recently Lynnworth provided a detailed overview of ultrasonic probe temperature sensors.[29] Ultrasonic wireline thermometry is based upon the change in the velocity of sound within a wire with temperature. The speed of sound in a wire varies with its elastic modulus and density as described in Eq. (10):

$$v(T) = \sqrt{\frac{Y(T)}{\rho(T)}} \qquad (10)$$

where *Y(T)* represents Young's modulus and *ρ(T)* represents density of the waveguide, both as a function of local temperature. Although both parameters are temperature dependent, the temperature effect on elastic modulus dominates by about an order-of-magnitude over that of density, which causes sound velocity to decrease with increasing temperature.

Ultrasonic wireline temperature measurement begins by launching an extensional acoustic wave down a waveguide. The return time of reflections of the launched wave pulse are then recorded. The wireline contains a series of notches. The time difference between reflections from each notch is indicative of the temperature between the notches (see Fig. 28).

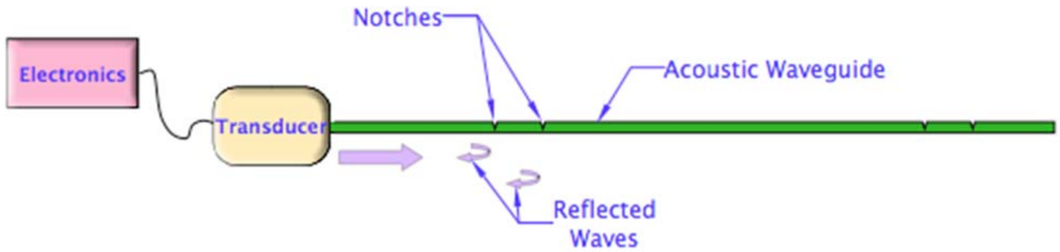


Fig. 28. An ultrasonic thermometry system including a notched waveguide.

Type N (nicrosil-nisil) thermocouples were developed in the 1970s and 1980s as a lower drift alternative to other base metal (particularly Type K) thermocouples.[30,31] Having achieved designation as a standard thermocouple type by the Instrument Society of America in 1983, Type-N thermocouples have been in widespread use for more than 25 years. The Nicrosil and Nisil alloys composing Type N thermocouples were developed after the instability mechanisms of other base-metal thermocouples were understood, specifically to overcome these instabilities. Nicrosil and Nisil alloy compositions feature increased component solute concentrations (chromium and silicon) in the nickel base to transition from internal to surface modes of oxidation and include solutes (silicon and magnesium), which preferentially oxidize to form oxygen diffusion barriers.[32]

5.7.2.2 Flow

Liquid salt flow measurement will most likely either be performed using external, ultrasonic flowmeters or Venturi-type flowmeters that use differential pressure gauges as their active element. Ultrasonic flowmeters are currently gaining wide acceptance in LWRs as a primary coolant flowmeter due to their low uncertainty and high stability. The high temperature of HTGRs requires the use of mechanical stand-offs to limit the ultrasonic transducer temperature exposure. The electronics for water and salt ultrasonic flowmeters would be essentially identical. The differential pressure gauges required for Venturi-base flow measurement either require diaphragm deflection measurement tolerant of NGNP temperatures or impulse line interconnection between a high-temperature and a low-temperature diaphragm, which would be instrumented with conventional low-temperature diaphragm deflection technology. The impulse line fluid would be a lower melting point fluid such as a lead-bismuth eutectic (44.5%Pb-55.5%Bi) with a 123°C melting point or a lower melting point salt. Both optically and capacitively based diaphragm deflection measurements are strong candidates for direct, high-temperature implementation. GP:50 is a commercial supplier with a specialized molten salt melt compatible diaphragm deflection-type pressure gauge that employs NaK (78% potassium, 22% sodium) impulse line isolation of the high-temperature diaphragm and offers diamond-like carbon diaphragm coating for good chemical compatibility with the salt.

5.7.2.3 Level

Several technologies are available for salt level measurement. Bubbler-type level measurements based upon the pressure required for minimal flow in a vertical tube are commercially available technology. Also, radar-type level measurements based upon reflection off the top surface of the salt are commercially available. The radar gun and electronics would be located in a standpipe above the fluid well outside of the high-temperature and high-radiation zones. Mechanical float-type level measurements can also be readily adapted to the salt loop by attaching a mechanical extension to a float on the surface of the salt. The mechanical extension would be configured such that it would extend into a nonmetallic standpipe above the vessel enabling the position of the end of the mechanical extension to be determined magnetically. Heated lance-type level measurements (Fig. 29) within a salt-compatible sheath would also provide discrete position level measurement.[33]

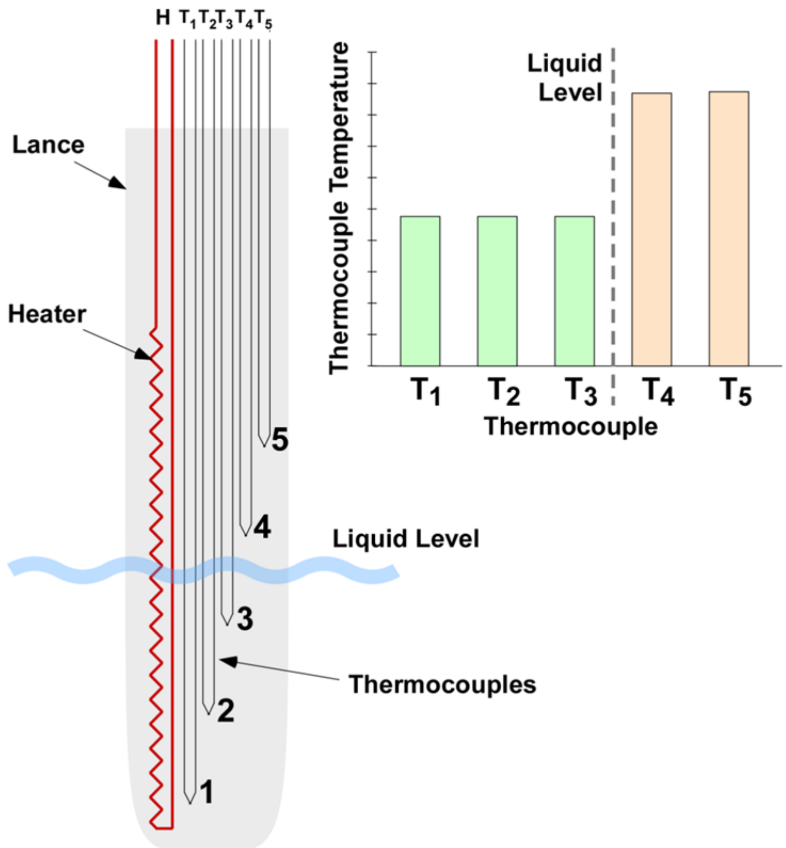


Fig. 29. Illustration of functioning of heated lance type level measurement system.

5.7.2.4 Salt chemistry

Maintaining the relatively low corrosivity of fluoride salts is critically dependent on controlling its reduction-oxidation state. The instrumentation required to characterize the detailed chemical state of fluoride salt exists as laboratory-type instrumentation and is not readily available in an industrial format. Electrochemical measurements are the standard technique for monitoring the redox condition of salt components. Optical absorption spectroscopy is also a potentially useful methodology for identifying trace chemical constituents and their valence state. However, the hot salt is itself a broad-spectrum infrared emitter that makes isolating particular absorption lines challenging. Optical access to the salt is most readily provided through a standpipe above the salt containing an inert gas bubble; a noble metal mirror within the salt would provide the optical return path.

5.7.2.5 Maturity evaluation

Little of the instrumentation is commercially available, and its longer-term reliability and drift performance have not been established. In general, the specialized high-temperature tolerant, high-reliability transducers and the supporting electronics are only available as designs from the literature. A sufficient market has not existed for commercial vendors to establish and maintain sources of supply for the specialized instrumentation.

Fiber-optic couple pyrometry and Fizeau cavity-type thermometers are available commercially. The remaining issues with the technology are in the longer-term performance of the transducers under plant conditions and the stability and reliability of both the opto-electronics and their control logic. Ultrasonic wireline thermometry has been repeatedly demonstrated in harsh, nuclear environments. However, a high reliability, commercial implementation does not currently exist.

Clamp-on, ultrasonic flow meters are a commercial technology in widespread use. High-temperature standoffs to implement the flow meter on very hot piping have been demonstrated in the past and can be ordered as a custom engineered component. However, as a custom component, the long-term reliability under engineering service conditions has not yet been established.

Type-N thermocouples are now widely commercially available at similar cost to other base metal thermocouples and with similar values of thermoelectric voltage output. As commercial nuclear power plants attempt to reduce the required instrumentation margins in their technical specifications, adoption of Type-N thermocouples as a general replacement for other (specifically Type-K) thermocouples should be anticipated.

Venturi flow meters require accurate differential pressure measurement. Pressure measurement is often implemented as a diaphragm deflection measurement. High-temperature tolerant carbon-based ceramic pressure gauges are just entering the commercial market.[34] Precious and refractory metal (or precious metal coated) diaphragms are also compatible with fluoride salts. The pressure-sensitive diaphragm can be implemented at the distal end of a small diameter hollow tube. Optical fibers are commercially available with temperature ratings higher than fluoride salt-cooled high-temperature reactor (FHR) primary coolant temperatures. Interferometric methods are commercially available to measure diaphragm deflection at the distal end of an optical fiber. The central issue for the optical fiber coupled technique is to establish the long-term system reliability under actual service conditions. As an alternative to directly measuring the high-temperature diaphragm deflection, an incompressible fluid (such as NaK) can be employed to transfer the pressure from the high-temperature bellows, along an impulse line, to a lower temperature diaphragm whose deflection can be measured using well-proven technology.

Mechanical float, heated lance, bubbler, and radar-based level measurement technologies are all established commercial technologies. While custom implementations for a specific LSHT variant would be useful, liquid salt level measurement technology is mature.

5.8 Helium Instrumentation

A helium heat transfer loop requires measurement of the major process variables (temperature, pressure, and flow). Additionally, mechanical position monitoring for a magnetic bearing circulator impeller and gas composition analysis will also be necessary.

5.8.1 Temperature

The maximum temperature for a helium flow loop is similar to that for a liquid salt; thus, the same types of thermocouples would be employed for temperature measurement but with potentially different thermowell materials for chemical and pressure compatibility with the different flowing media.

5.8.2 Pressure

Helium heat transfer loops are high-pressure systems necessitating thick piping walls. Also, since gaseous helium has a very small specific heat, pressure is the primary energy storage mechanism high-temperature helium. Pressure measurement thus becomes an important diagnostic for the heat transport loop performance. The helium pressure may be as high as 9 MPa at 750°C.

Diaphragm deflection is the most likely form of pressure measurement for a helium loop. The helium pressure can either be measured at the process temperature using an instrument with a built-in isolation impulse line such as the GP:50 system mentioned in the liquid salt section or an advanced ceramic (silicon carbo-nitride) type pressure gauge including an internally trapped reference pressure such as that developed by Sporian Microsystems would be possible. Alternatively, the helium pressure can be

measured at lower temperature by employing a helium impulse line to transfer the pressure to a diaphragm deflection system located in a cooler environment.

5.8.3 Flow

Direct flow measurement of high-temperature, high-pressure helium is technically challenging. Moreover the helium piping may be annularly configured to minimize the pressure boundary area complicating access to the centrally located flow. One method for determining the helium flow rate is to infer the flow based upon the impeller speed, knowledge of the pump characteristics, and the measured system pressure and temperature. However, accurate determination of the pump characteristic curves, which is necessary for initial sensor calibration, is not practical due to the system size and temperature; consequently flow measurement accuracy is limited.

Several high-temperature, high-pressure, helium-compatible flow meters are progressing through development and demonstration. Silicon carbide cantilever-type anemometry has the potential to endure the erosive (carbon dust), high-temperature environment.[35] However, silicon carbide micro-electro-mechanical systems (MEMS) remain at early phase commercialization.

Optical or microwave tracking of the suspended graphite dust can also be employed to infer flow rate. In this case a fundamental mode waveguide would be employed to channel microwaves from a horn located in a low-temperature environment into the primary piping. The radar waves would be reflected by the suspended graphite particles back up the waveguide. A frequency shift would be imposed upon the reflected microwave based upon the velocity of the dust. The primary limitation to this type of flow meter is that it only samples a relatively small fraction of the pipe cross section to estimate the entire flow.

Clamp-on, ultrasonic transit time type flow meters are also a possibility for high-pressure helium flow measurement within nickel alloy piping. The primary limitation to ultrasonic clamp-on flow metering of a gas within a metal pipe is the acoustic mismatch between the metal pipe wall and the gas. However, higher sensitivity ultrasonic clamp-on flow meters have been developed over the past decade and are now commonly used (especially for natural gas) as a gas within steel piping measurement.[36]

5.9 Impeller Location

With a high-temperature, high-pressure loop, developing the shaft seals necessary for an impeller shaft to penetrate the pressure boundary while having close enough matching to accommodate the different material coefficient of temperature expansion mismatch is technologically challenging. Also, contact-type bearings are known maintenance challenges for high-temperature rotating devices. Active magnetic bearings are currently under development to avoid the shaft penetration and contact bearing issues for helium impellers at gas reactors.[37]

A key measurement requirement of these active magnetic bearing-type canned rotor turbo machines is for high-speed multiaxis impeller position measurement. Active magnetic bearing suspension is based upon changing the drive current to electromagnets based upon rotor displacement measurements. Shaft horizontal and vertical position is independently measured at each radial bearing-motor set. Rotor position measurement can be performed by monitoring the change in the resonant frequency of a driven coil located near the rotor due to the shift in position of the magnetic rotor material or with a Hall-effect-type sensor.

Depending on the specific design requirements, pump shafts can rotate rapidly (thousands of revolutions per minute). The combination of turbulent fluid motion and rapid impeller rotation typically results in vibration frequencies up to roughly 10 kHz. Further, even minor imperfections in the impeller balance or in the rotor position sensor targeting can result in the control system itself enhancing the inherent

oscillations. Additionally, the bearing control response frequency needs to exceed the maximum credible vibration frequency to damp high-frequency impeller oscillations.

6. SUMMARY OF MAJOR INSTRUMENTATION ISSUES

6.1 Instrumentation R&D and Special Development Needs

For normal operation, the primary (essential) measurement is reactor thermal power and, unfortunately, there are usually no simple direct means of making that measurement in modular HTGRs. There are two requirements or components for the power measurement: steady state and transient. For the steady state, the usual means is to derive the power level from heat balance information, while for transients (rapid response measurements as required for protection systems), neutron detectors are used. While neutron detectors have sufficiently fast response, their output signals will drift with time and are also affected by control rod motion. The typical means for correcting the neutron flux signal is to continually reset the gain coefficient(s) with a long-term (very slow response) heat balance signal.

The primary heat balance computed power signal, P, uses mass flow rate w, helium specific heat, C_p, and reactor inlet and outlet temperature difference, ΔT, i.e.,

$$P = w \ C_p \ \Delta T$$

Helium specific heat is essentially constant over the full operating (and accident) ranges of interest. Flow in most HTGR configurations is not easy to measure unless there is an (unlikely) long run of straight pipe incorporating a venturi meter. In the Fort St. Vrain reactor, helium flow was approximated by a calculation using the measured circulator speed, along with helium temperature and pressure at the circulator and factoring in known circulator characteristics. Similar means are likely to be used in NGNP. Depending on the balance of plant (BOP) configuration, secondary side heat balances can also be used to estimate power.

Helium temperatures are measured by thermocouples capable of withstanding high temperatures and radiation with minimal drift. As noted previously,[1] Type-N thermocouples (Nickel-Chromium-Silicon/Nickel-Silicon) are rated for temperatures up to ~1200ºC and are likely to be the best candidates.[38] In general, they would be located in (tough, durable, nonvibrating) thermal wells which would purposely slow their response to filter out temperature fluctuation noise. Steps need to be taken to ensure the temperature measurements give an accurate “mixed mean” signal. Depending on the mixing and measurement distances from heating or cooling components, there can be significant biases and gradients within a pipe or vessel, and those gradients can also vary with changing conditions.

Reactor ΔT measurements for nuclear power using reactor vessel coolant inlet and outlet temperatures would also include the power lost to the RCCS (and other losses within the reactor cavity), while a secondary side power measurement would not, of course. If the reactor “inlet” signal were derived from core inlet (upper plenum) measurements, the RCCS power would need to be factored in as well to obtain total nuclear power. Circulator power would also need to be considered in any heat balance calculation.

Regarding temperature measurement limits, the coolant temperature distributions at the core exit are expected to vary widely due to spatial variations in power and flow rate and also due to bypass flows that may be almost entirely unheated. Unless special means are used in pebble bed cores (without a central

reflector) to flatten the radial power peaking, the spatial variations in core outlet temperatures (for both the coolant and support structures) would be expected to be especially large.

RTDs can generally be used to ~650°C, where the IEC 751 specification for Class A industrial platinum RTDs stops. The issue with using RTDs at that temperature is the stability and qualification, and apparently use of nuclear power qualified RTDs for service temperatures above PWR conditions is not authorized. Commercial drift rates specs are typically ~0.05°C/year. Nuclear plant RTDs are individually calibrated to tighter tolerances than IEC 751 Class A IPRTDs.

For the highest accuracy, at 450°C it would be preferable to qualify and calibrate an RTD. The next best choice would be a precious metal thermocouple (and as these are already available, perhaps they are the best solution). In the hot leg a precious metal thermocouple would be necessary. Heat balance calculations are of high value. For a lower value measurement, a base metal thermocouple such as a Type-N should suffice. The issue with base metal thermocouples above a few hundred degrees Celsius is that their internal alloying and surface chemistry begins to interact with the impurities in the surrounding insulation and metal sheath material, resulting in drift. While the drift rates for Type N thermocouples are lower than those of other base metal options, precious metal thermocouples are much more stable.

Another parameter of great interest is core (fuel) temperature, since fuel operating temperatures are important inputs to fuel performance models. There are no (known) direct means of measuring pebble fuel temperatures in situ, especially since with on-line refueling, the bed of pebbles is continually in (very slow) motion. Melt-wire measurements can be made in dummy (graphite-only) pebbles to determine after-the-fact maximum pebble temperatures, although the path and power/flow history of the pebbles would be unknown. Some in-core measurements were made in Fort St. Vrain (prismatic fuel blocks), where an “instrument package” was temporarily substituted for a control rod for special testing.

Continuous measurements of RCCS power (performance) are also important, since assumption of the safety-grade RCCS capability to perform well in loss-of-cooling accidents is crucial to the safety case. Such a measurement can be difficult, depending somewhat on the design, since the RCCS cooling panels are spread out widely around the reactor cavity, and the spatial flow and temperature distributions in the panels would be expected to vary widely as well. RCCS performance can also be verified (on line) to some extent by monitoring external reactor vessel temperature distributions. If the RCCS has an operating (forced flow) mode instead of being “entirely passive,” means of validating RCCS performance in an accident mode (with natural circulation cooling) would need to be done periodically during normal operation. These tests would require special testing, measurement, and analysis procedures.

Other measurements needing further development and validation for conditions peculiar to NGNP would be continuous monitoring of primary system moisture and circulating activity (radioactivity and dust), and chemical conditions (helium, CO_2 and CO, moisture) in the reactor cavity and elsewhere in the reactor building. In the case of primary moisture measurements, if there is more than one steam generator in the loop, the detection would need to determine which steam generator module had the leak to initiate the proper isolation and dump process.

6.2 Regulatory and Licensing Implications

The acceptance criteria for design, systems and components in nuclear reactors are set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities,” and 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants.”[39,40] Guidance is provided in NUREG-0800, the *Standard Review Plant* (SRP).[41] The primary section of the SRP that covers the instrumentation is Chapter 7, “Instrumentation and Controls.”

The acceptance criteria and guidelines for Instrumentation and Control (I&C) systems are divided into ten categories:

1. Code of Federal Regulations
2. General Design Criteria (GDC) of 10 CFR Part 50 Appendix A,
3. Commission Papers (SECY) and Staff Requirements Memoranda (SRM),
4. Regulatory Guidelines (RGs),
5. Branch Technical Positions (BTPs),
6. NUREG-Series Publications,
7. Institute of Electrical and Electronics Engineers (IEEE) Standards,
8. International Society of Automation (ISA) Standards,
9. International Electrotechnical Commission (IEC) Standards, and
10. International Atomic Energy Agency (IAEA) Publications.

Listed in Table 8 are those regulatory documents, codes, standards, and regulatory commitments that are applicable to instrumentation not required for safety.

NOTE: References to special notes listed in Table 8 will be provided in the next draft of the report. Citations are given in this draft to indicate the special status of the item.

Table 8. List of regulatory documents related to reactor instrumentation for high-temperature reactors

Criteria	Title or Subject	App. to VHTR[a]
1. 10 CFR Parts 50 and 52 (see Section 6.2.1)		
• §50.55a(a)(1)	Quality Standards for Systems Important to Safety	Y
• §50.55a(h)(2)	Protection Systems (IEEE Std 603-1991 or IEEE Std 279-1971)	Y
• §50.55a(h)(3)	Safety Systems	Y
• §50.34(f)(2)(v) [I.D.3]	Bypass and Inoperable Status Indication	Y
• §50.34(f)(2)(xi) [II.D.3]	Direct Indication of Relief and Safety Valve Position	Y
• §50.34(f)(2)(xvii) [II.F.1]	Accident Monitoring Instrumentation	Y
• §50.34(f)(2)(xviii) [II.F.2]	Instrumentation for the Detection of Inadequate Core Cooling	Y
• §50.34(f)(2)(xiv) [II.E.4.2]	Containment Isolation Systems	Y
• §50.34(f)(2)(xix) [II.F.3]	Instruments for Monitoring Plant Conditions Following Core Damage	Y
• §50.49	Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants	Y
• §50.62	Requirements for Reduction of Risk from Anticipated Transients without Scram	Y

Table 8. List of regulatory documents related to reactor instrumentation for high-temperature reactors (continued)

Criteria	Title or Subject	App. to VHTR[a]
2. 10 CFR Part 50, Appendix A General Design Criteria (GDC) (see Section 6.2.2)		
	I. Overall Requirements	
• Criterion 1	Quality Standards and Records	Y
• Criterion 2	Design Bases for Protection	Y
• Criterion 3	Fire Protection	Y
• Criterion 4	Environmental and Dynamic Effects Design Bases	Y
• Criterion 5	Sharing of Structures, Systems, and Components	Y
	II. Protection by Multiple Fission Product Barriers	
• Criterion 10	Reactor Design	Y
• Criterion 11	Reactor Inherent Protection	Y
• Criterion 12	Suppression of Reactor Power Oscillations	P
• Criterion 13	Instrumentation and Control	Y
• Criterion 16	Containment Design	P
• Criterion 17	Electrical Power Systems	Y
	III. Protection and Reactivity Control Systems	
• Criterion 20	Protection System Functions	Y
• Criterion 21	Protection Systems Reliability and Testability	Y
• Criterion 22	Protection System Independence	Y
• Criterion 23	Protection System Failure Modes	Y
• Criterion 24	Separation of Protection and Control Systems	Y
• Criterion 25	Protection System Requirements for Reactivity Control Malfunctions	Y
• Criterion 26	Reactivity Control System Redundancy and Capability	Y
• Criterion 27	Combined Reactivity Control Systems Capability	Y
• Criterion 28	Reactivity Limits	Y
• Criterion 29	Protection Against Anticipated Operational Occurrences	Y
	IV. Fluid Systems	
• Criterion 30	Quality of Reactor Coolant Pressure Boundary	Y
• Criterion 34	Residual Heat Removal	Y
• Criterion 35	Emergency Core Cooling	Y
• Criterion 38	Containment Heat Removal	P

Table 8. List of regulatory documents related to reactor instrumentation for high-temperature reactors (continued)

Criteria	Title or Subject	App. to VHTR[a]
3. Staff Requirements Memoranda		
• SRM to SECY 93-087 II.Q	Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems	Y
4. Regulatory Guides		
• Regulatory Guide 1.22	Periodic Testing of Protection System Actuation Functions (*also addressed in BTP 7-8*)	Y
• Regulatory Guide 1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety System	Y
• Regulatory Guide 1.53	Application of the Single-Failure Criterion to Safety Systems (*endorses IEEE Std 379-2000*)	Y
• Regulatory Guide 1.62	Manual Initiation of Protection Actions	Y
• Regulatory Guide 1.75	Criteria for Independence of Electrical Safety Systems (*endorses IEEE Std 384-1992*)	Y
• Regulatory Guide 1.97	Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident, and Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants (*endorses IEEE Std 497-2002 and BTP 7-10*)	P
• Regulatory Guide 1.105	Setpoints for Safety-Related Instrumentation (*endorses ISA Std S67.04-1994 Part I and BTP 7-12*)	P
• Regulatory Guide 1.118	Periodic Testing of Electric Power and Protection Systems (*endorses IEEE Std 338-1987*)	Y
• Regulatory Guide 1.151	Instrument Sensing Lines (*endorses ANSI/ISA-67.02.01-1999*)	N
• Regulatory Guide 1.180	Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems	Y
5. Branch Technical Positions (BTP)		
• BTP 7-8	Guidance on Application of Regulatory Guide 1.22	Y
• BTP 7-9	Guidance on Requirements for Reactor Protection System Anticipatory Trips	Y
• BTP 7-10	Guidance on Application of Regulatory Guide 1.97	P
• BTP 7-11	Guidance on Application and Qualification of Isolation Devices	Y
• BTP 7-12	Guidance on Establishing and Maintaining Instrument Setpoints	P
• BTP 7-13	Guidance on Cross-Calibration of Protection System Resistance Temperature Detectors	P
• BTP 7-17	Guidance on Self-Test and Surveillance Test Provisions	Y
6. NUREG Publications		
• NUREG-0737	Clarification of TMI Action Plan Requirements	P
• NUREG-1338	Preapplication Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor (MHTGR)	Y

Table 8. List of regulatory documents related to reactor instrumentation for high-temperature reactors (continued)

Criteria	Title or Subject	App. to VHTR[a]
7. The Institute of Electrical and Electronics Engineers Standards		
• IEEE Std 279-1971 or IEEE Std 603-1991	IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations	Y
• IEEE Std 323-1974 and IEEE Std 323-1983	IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations	Y
• IEEE Std 338-1987	IEEE Standard Criteria for Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems	Y
• IEEE Std 379-2000	IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems	Y
• IEEE Std 384-1992	IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits	Y
• IEEE Std 497-2002	IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations	Y
• IEEE Std 7-4.3.2-2003	IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations	
8. The International Society of Automation (ISA) Standards		
• ANSI/ISA–67.02.01-1999	Nuclear Safety-Related Instrument Sensing Line Piping and Tubing Standards for Use in Nuclear Power Plants	Y
• ISA Std S67.04-1994	Setpoints for Nuclear Safety-Related Instrumentation	Y
9. International Electrotechnical Commission (IEC) Standards		
• IEC 60880:2006	Nuclear Power Plants—Instrumentation and Control Systems Important to Safety—Software Aspects for Computer-Based Systems Performing Category A Functions	P
• IEC 61000:1992 Parts 1 through 4	Electromagnetic Compatibility (EMC)	Y
• IEC 61508:1998	Functional Safety of Electrical/Electronic/Programmable Electronic Safety-Related Systems	P
• IEC 61513:2001	Nuclear Power Plants—Instrumentation and Control for Systems Important to Safety—General Requirements for Systems	P
• IEC 61784-1–3:2010	Industrial Communication Networks	P
10. International Atomic Energy Agency (IAEA) Publications		
• IAEA-TECDOC-1366	Considerations in the Development of Safety Requirements for Innovative Reactors: Application to Modular High Temperature Gas Cooled Reactors	N/A

Y–applies to HTGR/VHTR; N–does not apply; P–applies with special provisions.

10 CFR 50, Appendix A stipulates the general design criteria (GDC) for nuclear power plants. The GDC establish minimum requirements for the principal design criteria, providing guidance to ensure that structures, systems, and components (SSCs) provide reasonable assurance that the facility can be operated

without undue risk to the health and safety of the public. The GDC that are applicable to HTGR instrumentation are also found in Table 8.

6.2.1 10 CFR Parts 50 and 52

Part 50 is the part of *Code of Federal Regulations, Title 10—Energy* that sets the acceptance criteria and general requirements for *Domestic Licensing of Production and Utilization Facilities*. §50 has direct implications for the I&C systems in nuclear power generating plants, hence for high-temperature gas-cooled reactors. This section briefly discusses the sections that will potentially have particular impact on requirements for HTGR/VHTR I&C system design and qualification.

§50.55a—Codes and Standards

§50.55a forms the foundation of quality requirements for all SSCs in a nuclear power plant and applies to every SSC. Because of its generic form, it is supported and augmented by other rules, regulations, and guidance documents. The three items under §50.55a have specific application for nuclear plant I&C systems, and have been extracted from NUREG-0800, *Standard Review Plan*, Chapter 7, "Instrumentation and Controls," Table 7-1, "Regulatory Requirements and Standard Review Plan Acceptance Criteria for Instrumentation and Control Systems Important to Safety."

> *(a)(1) Structures, systems, and components must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.*
>
> *(h)(2) Protection systems. For nuclear power plants with construction permits issued after January 1, 1971, but before May 13, 1999, protection systems must meet the requirements stated in either IEEE Std. 279, "Criteria for Protection Systems for Nuclear Power Generating Stations," or in IEEE Std. 603−1991, "Criteria for Safety Systems for Nuclear Power Generating Stations," and the correction sheet dated January 30, 1995. For nuclear power plants with construction permits issued before January 1, 1971, protection systems must be consistent with their licensing basis or may meet the requirements of IEEE Std. 603–1991 and the correction sheet dated January 30, 1995.*
>
> *(h)(3) Safety systems. Applications filed on or after May 13, 1999, for construction permits and operating licenses under this part, and for design approvals, design certifications, and combined licenses under part 52 of this chapter, must meet the requirements for safety systems in IEEE Std. 603–1991 and the correction sheet dated January 30, 1995.*

All the SSCs in HTGR/VHTR designs must meet these requirements.

§50.34—Contents of construction permit and operating license applications; technical information

> *(f) Additional TMI-related requirements.*

§50.34(f) has special provisions for nuclear power plant instrumentation and control systems, based primarily on the unfortunate experience gained during the Three Mile Island (TMI) accident.

> *(f)(2)(v) Provide for automatic indication of the bypassed and operable status of safety systems.*
>
> *(f)(2)(xi) Provide direct indication of relief and safety valve position (open or closed) in the control room.*

The requirements in §50.34(f)(2)(v) and §50.34(f)(2)(xi) also apply to HTGR/VHTR designs.

(f)(2)(xiv) *Provide containment isolation systems that: (II.E.4.2)*

(A) Ensure all non-essential systems are isolated automatically by the containment isolation system,

(B) For each non-essential penetration (except instrument lines) have two isolation barriers in series,

(C) Do not result in reopening of the containment isolation valves on resetting of the isolation signal,

(D) Utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operation,

(E) Include automatic closing on a high radiation signal for all systems that provide a path to the environs.

The requirements in §50.34(f)(2)(xiv) specifically address light-water-cooled reactor containment systems. Whether the HTGR designs will include containment or confinement is still being debated. Furthermore, containment or confinement designs for HTGR/VHTRs have different set of design bases to perform properly during a design-basis accident. For instance, certain containment designs propose rapid discharge systems that will activate during the early stage of a depressurization accident, where the radioactivity levels are presumed to be low enough to prevent significant public exposure and employee exposure that is below the regulatory limits, to relieve the excess pressure within the containment. These kinds of design variations certainly conflict with the requirements set forth in §50.34(f)(2)(xiv) and must be considered within the context of a gas reactor design requirements.

(f)(2)(xvii) *Provide instrumentation to measure, record and readout in the control room: (A) containment pressure, (B) containment water level, (C) containment hydrogen concentration, (D) containment radiation intensity (high level), and (E) noble gas effluents at all potential, accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples. (II.F.1)*

(f)(2)(xix) *Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage. (II.F.3)*

§50.49—Environmental qualification of electric equipment important to safety for nuclear power plants.

(b) *Electric equipment important to safety covered by this section is:*

(1) Safety-related electric equipment,[*]

(2) Nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions specified in subparagraphs (b)(1)(i)(A) through (C) of this section by the safety-related equipment,

(3) Certain post-accident monitoring equipment.[†]

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[*]Safety-related electric equipment is referred to as "Class 1E" equipment in IEEE 323–1974.

[†]Specific guidance concerning the types of variables to be monitored is provided in Revision 2 of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident."

§50.49 requires that safety-related equipment will conform to the requirements set forth in IEEE Std 323-1974, “IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations.”[42] This standard describes the basic requirements for qualifying Class 1E equipment and interfaces that are to be used in nuclear power plants. The qualification requirements demonstrate and document the ability of equipment to perform safety function(s) under applicable service conditions including design basis events, reducing the risk of common-cause equipment failure.

§50.62—Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants.

> (b) Definition. *For purposes of this section, Anticipated Transient Without Scram (ATWS) means an anticipated operational occurrence as defined in appendix A of this part followed by the failure of the reactor trip portion of the protection system specified in General Design Criterion 20 of appendix A of this part.*

Though §50.62 is specifically written for LWRs, the issues addressed in the code also apply to high-temperature gas-cooled reactors in general. The ATWS calculations are usually included in Chapter 15, “Accident Analysis,” Section 8, “Anticipated Transients Without Scram” of the Design Control Document (DCD), Tier 2, submitted by the licensee.

Part 52 is the part of *Code of Federal Regulation Title 10—Energy* that governs the issuance of early site permits, standard design certifications, combined licenses, standard design approvals, and manufacturing licenses for nuclear power facilities licensed under Section 103 of the Atomic Energy Act of 1954, as amended (68 Stat. 919), and Title II of the Energy Reorganization Act of 1974 (88 Stat. 1242). Part 52 cites inspections, tests, analyses, and acceptance criteria (ITAAC), which might have certain impositions on the gas reactor I&C system design and testing.

6.2.2 10 CFR Part 50, Appendix A, General Design Criteria (GDC)

Under the provisions of §50.34, an application for a construction permit must include the principal design criteria for a proposed facility. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

The GDC establish the minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The GDC are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units.

The development of the GDC is not yet complete. Some of the specific design requirements for structures, systems, and components important to safety have not as yet been suitably defined. A particular consideration that has implications for the I&C systems is described as follows:

> *Consideration of the possibility of systematic, nonrandom, concurrent failures of redundant elements in the design of protection systems and reactivity control systems (See Criteria 22, 24, 26, and 29.)*

It is expected that these criteria will be augmented and changed from time to time as important new requirements for these and other features are developed. Augmentation to and transformation of the existing regulatory structure is under consideration. The NRC staff is currently working on a *Technology-Neutral Framework* under a new regulatory structure for new plant licensing. These changes might have ramifications for the design, manufacturing, inspection, and testing of the I&C systems, in particular instrumentation part of the I&C systems, for the HTGRs/VHTRs.

A detailed discussion on implications of each criterion on the instrumentation of HTGRs/VHTRs is beyond the scope of this document. We, therefore, focus on the criteria that provide guidance for or have direct impact on the instrumentation systems in these reactors.

I. Overall Requirements

These requirements apply to any safety system, including the protection system. Provisions in these criteria apply to HTGR/VHTR instrumentation systems that are important to safety. The classification of the instruments according to importance to safety is expected to be design-specific, provided that the proposed design meets the code, rules, and regulations of the Commission.

> Criterion 5—Sharing of structures, systems, and components. *Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.*

Criterion 5 prohibits sharing of safety-related SSCs in nuclear power plants. The applicability of this criterion to modular HTGR/VHTRs designs might be questionable. Characteristic time constants during a design basis accident in a gas reactor are much larger than that of light-water-cooled reactors. Furthermore, the fuel—by design—provides containment functions, which adds an additional protection barrier. Therefore, the requirements in Criterion 5 can be relaxed for SSCs in gas reactors, including the I&C systems, or the conditional provision in the requirement "*... unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions*" can be fulfilled more easily.

II. Protection by Multiple Fission Product Barriers

Instrumentation and control systems in a nuclear power plant are considered as an additional barrier for containing the fission products within the prescribed geometry. One GDC under this section of Appendix A provides generic requirements for I&C systems in nuclear power plants.

> Criterion 13—Instrumentation and control. *Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.*

Criterion 13 requires that the state of the fission process be known at all times. Monitoring of the fission process is usually done by neutron detectors placed outside of the reactor pressure vessel. Because the criterion specifically requires that the monitoring equipment be functional over the entire range of operations, additional means for measuring the status of the fission process should be provided. During a severe accident event, instruments that are in close proximity to the reactor core can become dysfunctional or may provide measurements that are no longer reliable. Wide area monitors placed in the containment or confinement can provide an indirect way of measuring the state of the fission process.

III. Protection and Reactivity Control Systems

This section provides general design criteria for reactor protection systems. Protection systems are deliberately designed as simple systems and usually use point-to-point hard connections to instrument sensing lines. Generally, initiation of a protection system is triggered by the output multiple sensors that are run through a voting logic to essentially prevent unnecessary trip of the plant.

> Criterion 20—Protection system functions. *The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operation occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.*

Criterion 20 specifically cites fuel design limits, which are much higher than the conventional light-water-cooled reactors. Much of the design limits for HTGR/VHTR fuel is temperature dependent—though there are certain design limits for fuel exposed to air or steam at very high temperatures, since such an event deteriorates the quality of the fuel and substantially reduces its capability to contain the fission products.

Direct measurement of temperature in the core, particularly of fuel assemblies in prismatic-type design and pebbles in pebble-bed-type design, seems challenging with the existing, commercially available technologies. ORNL survey of previous gas-cooled reactors indicated that some of these reactors did employ means for fuel temperature measurements (see Sect. 5.1). However, qualification of these sensors as part of a safety system is a significant challenge. Direct temperature measurement in pebble-bed-type designs still remains to be elusive.

> Criterion 21—Protection system reliability and testability. *The protection system shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.*

> Criterion 22—Protection system independence. *The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.*

> Criterion 23—Protection system failure modes. *The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.*

> Criterion 24—Separation of protection and control systems. *The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.*

Criteria 21 through 24 form the basis of common instrumentation and control system design practices and will also apply to HTGR/VHTR I&C designs.

> Criterion 25—Protection system requirements for reactivity control malfunctions. *The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.*

Current gas reactor designs—prismatic or pebble bed—use TRISO fuel, whose design limits are significantly different than those of LWR ceramic fuels. HTGR fuel is more resilient to high-temperature operation; therefore, operation at elevated temperatures can be allowed for a period of time during a transient without any loss of fuel integrity. Major design concerns are migration of certain volatile fission products through the SiC layer. However, the overarching requirement of Criterion 25 still applies.

> Criterion 26—Reactivity control system redundancy and capability. *Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.*

There may be slight departures in design options for certain gas reactor types in conformity to the requirements of Criterion 26, particularly in the pebble-bed designs. However, the rationale behind Criterion 26—provision of redundancy and capacity for reactivity control system—should be met by alternative design options.

> Criterion 27—Combined reactivity control systems capability. *The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.*

Criterion 27 includes the usual LWR-specific lexicon for reactivity control. As indicated earlier for other criteria, the rationale behind Criterion 27 is expected to be fulfilled. Lack of a detailed engineering design precludes the assessment on the applicability of Criterion 27 for HTGR/VHTRs. However, certain design concepts include reactivity control systems that employ a similar approach to the poison addition in a LWR. For instance, PBMR Reserve Shutdown System (RSS) consists of eight units that can insert Small Absorber Spheres into the eight borings of the central reflector, as described in the latest Technical Description of the PBMR Demonstration Power Plant.[43] The design can be quite different, but the functionally is such that they both introduce additional negative reactivity to augment the reactivity margin of the shutdown control rod—with the probability that should any one of the rods stick, sufficient negative reactivity would exist to compensate for reactivity swings. One positive aspect of pebble-beds is that they run on very little excess reactivity because of the online refueling; hence, their shutdown margin is much smaller than those without continuous refueling.

> Criterion 28—Reactivity limits. *The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents*

> *shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.*

Much of the requirements in Criterion 28 relates to the reactivity control system design and has indirect implications for the instrumentation system. However, the Criterion strenuously emphasizes that the reactivity system will perform its function "*... with appropriate limits on the potential amount and rate of reactivity increase ....*" From the design standpoint, these requirements imply that the control system must be furnished with appropriate instrumentation to assure that it performs as per its design specifications.

> Criterion 29—Protection against anticipated operational occurrences*. The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.*

Criterion 29 supports Criterion 28 and Criteria 21 through 25, in that both the reactor protection system and the reactivity control system must be designed, manufactured, installed, and tested to quality standards commensurate with the level of safety functions they perform—a requirement also set forth in §50.55a(a)(1). Reactor protection systems for any nuclear reactor are classified as part of the safety system, therefore, are required to meet the Nuclear Quality Assurance (NQA) requirements as per §50.34 and Appendix B to Part 50. Further guidance can be found in Regulator Guide (RG) 1.28 [Ref. 44], which endorses ASME NQA-1-2008 with the ASME NQA-1a-2009 addenda. Moreover, electrical components—including the sensing lines and actuators—that are part of safety system, or part of a system that interact with a safety system, must be designed, manufactured, installed, and tested to Class 1E quality criteria, as per IEEE Std 323-2003, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations,"[45] which is endorsed—with reservations as explained in *Section C. Regulatory Position*—by Regulatory Guide 1.209, "Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants."[46]

IV. Fluid Systems

Criteria 30 through 46 under Section IV establish minimum requirements for fluid systems in water-cooled nuclear power plants. These criteria impose no direct requirements for instrumentation systems. However, certain criteria have design implications for instrumentation and control systems. Below, a brief discussion is presented on certain Criteria that have design implications for instrumentation systems.

> Criterion 30—Quality of reactor coolant pressure boundary. *Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.*

In the United States, reactor primary system components are designed per specifications of *ASME Boiler and Pressure Vessel Code* (BPVC), Section III, "Rules for Construction of Nuclear Facility Components", Division 1, Subsection NH, "Class 1 Components in Elevated Temperature Service."[47] These requirements also apply to HTGR/VHTRs. There are a number of alloys qualified for extended service life as reactor vessel material under Subsection NH. Those include Type 304 and Type 316 Stainless Steel, Alloy 800H, 2-1/4 Cr–1 Mo, and 9 Cr–1 Mo–V. Currently only Alloy 800H is qualified for high-pressure service at temperatures up to 730ºC. There are draft code cases for several other alloys, including Alloy 617, to be included under Section III, Subsection NH.

The part that is relevant to an instrument designer is the last sentence. Criterion 30 requires that means be provided for detecting reactor coolant leakage and, if possible, for identifying the location of the leak.

> Criterion 34—Residual heat removal. *A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.*
>
> *Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.*

One of the requirements in Criterion 34 is the provision of "*suitable leak detection and isolation*" capabilities for the residual heat removal systems. These systems have drastically different design concepts for removing post-shutdown heat from the reactor core; therefore, they have quite different specifications. However, functional requirements are—to a great extent—similar: assurance of fuel integrity under any anticipated transients and design basis events.

Suitable leak detection refers to sensing capabilities—instrumentation system and isolation of the leak requires actuator capabilities—control system.

> Criterion 35—Emergency core cooling. *A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.*
>
> *Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.*

Existing HTGR/VHTR designs do not employ a safety system similar to an Emergency Core Cooling System (ECCS) in a water-cooled reactor. However, means are provided to remove the excess heat that cannot be removed by conventional cooling mechanisms (i.e., the secondary heat transport system). Most HTGR/VHTR designs include a passive safety system intended to remove core decay heat and sensible heat during a design basis accident. These systems are also recognized as acceptable means of heat removal by the Draft ANSI/ANS-53.1, "Nuclear Safety Criteria and Safety Design Process for Modular Helium-Cooled Reactor Plants."[48]

Similar discussions apply as indicated in Criteria 30 and 34.

> Criterion 38—Containment heat removal. *A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.*
>
> *Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.*

No final HTGR/VHTR containment design exists for the plants considered as candidate designs. Discussions on incorporating a confinement and eliminating the containment system still continue. Even with a containment system, design concepts exist that differ from the conventional leak-tight containments that are used with the existing water-cooled reactors. For instance, safety calculations indicate the viability of pressure relief valves in a containment or confinement environment that activate during the very preliminary phase of a depressurization accident to reduce excessive forces on the structure. At the early stages of a quick depressurization, the radioactivity levels are anticipated to be too low to cause any concern for plant employees and the general public. These relief systems close once the containment internal pressure reduces below a threshold. These measures are intended to improve the effectiveness and reliability of the containment systems.

As given in the example above, containment system designs should be expected to be significantly different. Containment spray systems, as generally employed in water-cooled reactors to restrict the internal temperature and reduce the pressure by condensation of the steam, would not be effective—and, in fact, detrimental—in a gas reactor with a massive graphite inventory. Therefore, design and performance specifications for HTGR/VHTRs will be quite different than the existing designs. However, functional requirements (i.e., the rationale behind such systems) should be similar.

Hence, it is possible to say that Criterion 38 applies to HTGR/VHTRs with special provisions.

Other criteria under *Section IV*, "*Fluid Systems*" have similar reservations, in that system design and performance specifications might radically differ, but similar functional requirements should apply.

V Reactor Containment

Criteria 50 through 57 under Section V establish the minimum design requirements for containment systems in water-cooled nuclear power plants. Gas-cooled reactors will most likely have different bases for a containment structure and its functional requirements due to substantial differences in the source term—primarily operating temperature and liquid vs gas coolant. Therefore, criteria established under this section of Appendix A are expected to have a minimal applicability. On the other hand, instrumentation requirements might be similar though the structure might differ.

> Criterion 50—Containment design basis. *The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by §50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.*

Please refer to discussion under Criterion 38, "Containment heat removal."

> Criterion 53—Provisions for containment testing and inspection. *The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leak-tightness of penetrations which have resilient seals and expansion bellows.*

Depending on the selection of a containment or confinement structure, the requirements in Criterion 53 apply to HTGR/VHTRs.

> Criterion 54—Piping systems penetrating containment. *Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.*

Criterion 54 applies to HTGR/VHTRs.

VI. Fuel and Reactivity Control

> Criterion 63—Monitoring fuel and waste storage. *Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.*

> Criterion 64—Monitoring radioactivity releases. *Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.*

Criteria 63 and 64 apply to HTGR/VHTRs.

6.2.3 Staff Requirements Memoranda

The only Staff Requirements Memoranda (SRM) that are known to have implications for instrumentation and control system design is SRM to SECY 93-087, Item II.Q, which indicates the regulatory position on "Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems."

Common-mode failures have been a prolific topic of discussion for digital control systems. The complexity of such systems preclude a systematic assessment of defense-in-depth and diversity.

The aforementioned SRM requires that the license applicant provide a detailed assessment of diversity and defense-in-depth—now commonly called D3. It stipulates that the vendor or the applicant analyze each postulated common-mode failure for each postulated event in the Accident Analysis section of the Safety Analysis Report (SAR) using best estimate methods.

New reactor designs, including the HTGR/VHTRs, are expected to use digital instrumentation and control systems for both safety-related and nonsafety-related systems. Some new water-reactor designs use digital systems with an independent, completely analog diverse actuation system that can perform all the safety and protection functions in the event of loss of control in the digital system.

All of these concerns are relevant for HTGR/VHTR designs; therefore, these requirements should apply.

6.2.4 Regulatory Guides

A list of Regulatory Guides was given in Table 8. A brief account on each guide will be discussed below.

6.2.4.1 Regulatory Guide 1.22—Periodic Testing of Protection System Actuation Functions (Current Revision 0, February 1972)

Regulatory Guide (RG) 1.22 [Ref. 49] describes a method that the staff of the NRC considers acceptable to meet the regulatory requirements and supporting guidelines regarding the bypassed and inoperable status indication. These include but not limited to GDC 21, "Protection System," and GDC 22,

“Protection System Independence,” as set forth in Appendix A, “General Design Criteria for Nuclear Power Plants,” to Title 10, of the *Code of Federal Regulations*, Part 50, “Domestic Licensing of Production and Utilization Facilities” (10 CFR Part 50).[39]

Guidance provided in RG 1.22 is acceptable and can be used for HTGR/VHTRs.

An acceptable definition of the protection system is given by IEEE Std 279-1971, “Criteria for Protection Systems for Nuclear Power Generating Stations.”[50]:

> *A “protection system” encompasses all electric and mechanical devices and circuitry (from sensors to actuation device input terminals) involved in generating those signals associated with the protective function. These signals include those that actuate reactor trip and that, in the event of a serious reactor accident, actuate engineered safety features (ESFs), such as containment isolation, core spray, safety injection, pressure reduction, and air cleaning. “Protective function” is defined as the sensing of one or more variables associated with a particular generating station condition, signal processing, and the initiation and completion of the protective action at values of the variables established in the design bases.*

The NRC recognizes that “protection systems” are a subset of “safety systems,” which are covered by IEEE Std 603-1991, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” (including the correction sheet dated January 30, 1995).[51] Safety system is a broad-based and all-encompassing term, embracing the protection system in addition to other electrical systems.

HTGR/VHTR protection system designs will most likely differ from the conventional water-cooled reactor designs. The difference is expected to be twofold: HTGR/VHTR protection system trigger signals will probably not depend on direct measurement of core temperature differential (ΔT) and core absolute and differential pressures (P and ΔP). Most protection system designs for light-water reactors use a logical operation of ΔT and ΔP signals with additional redundancy checks performed by the protection logic. These signals are generated by safety-related sensors, which are periodically tested. Protection system actuators are also safety-related components that control the shutdown rods. These actuators are required to be periodically tested by the code.

A reliable core ΔT measurement might be problematic for the HTGRs and particularly for VHTRs because of the extreme temperatures. Detailed computational fluid dynamics (CFD) calculations indicate that local temperatures at the outlet plenum might vary much more widely due to streaking in the gas stream exiting the core.[52] Local gas temperatures are expected to reach much higher values than the core exit average temperature.

Few existing demonstration and research gas reactors, such as the HTR-10 and HTTR, use Type-N thermocouples for direct measurement of core exit temperature. Type-N thermocouples are made of Nicrosil-Nisil alloys. Nicrosil is a nickel alloy containing 14.4% chromium, 1.4% silicon and 0.1% magnesium; nisil is an alloy of nickel and silicon. Type-N thermocouples are known to be suitable for use at high temperatures exceeding 1200ºC—with a sensitivity of about 30 μV/ºC at 900ºC. For continuous operation, Type-N thermocouples are approved up to 1100ºC. However, uncertainty band at high temperatures becomes exceedingly wide defeating the measurement of temperature. Class 1E Type-N thermocouple has been designed, manufactured, tested, and qualified for a few known applications.[53]

6.2.4.2 Regulatory Guide 1.47—Bypass and Inoperable Status Indication for Nuclear Power Plant Safety Systems (Current Revision 1, February 2010)

Regulatory Guide 1.47 [Ref. 54] describes a method that the staff of the NRC considers acceptable to meet the regulatory requirements and supporting guidelines regarding the bypassed and inoperable status indication. These include but not limited to GDC 1, “Quality Standards and Records”; GDC 13, “Instrumentation and Control”; GDC 19, “Control Room”; GDC 21, “Protection System”; GDC 22,

"Protection System Independence"; and GDC 24, "Separation of Protection and Control Systems," as set forth in Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10, of the *Code of Federal Regulations*, Part 50, "Domestic Licensing of Production and Utilization Facilities" (10 CFR Part 50).

In 10 CFR 50.55 a(h), the NRC requires compliance with IEEE Std 603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," (including the correction sheet dated January 30, 1995) [Ref. 51]. IEEE Std 603-1991 lists requirements with regard to a bypassed and inoperable status indication for safety systems. In addition, Criterion IVX, "Inspection, Test, and Operating Status," as given in Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR 50, requires that measures be established for indicating the operating status of structures, systems, and components (SSCs) of the nuclear power plant, such as by tagging valves and switches, to prevent inadvertent operation. The provisions of 10 CFR 50.34(f)(2)(v) also require an automatic indication of the bypassed and operable status of safety systems.

Digital computer-based I&C systems make extensive use of self-testing. Unlike the analog counterparts, digital computer-based I&C systems exhibit unconventional failure modes. Self-testing and watchdog timers should reduce the time to detect and identify failures. Computer self-testing is most effective at detecting random hardware failures.

A bypass and inoperable status indication system should include the capability of ensuring its operable status during normal plant operation to the extent that the indicating and annunciating function can be verified. Moreover, such a system should include measures against erroneous bypass indications.

Guidance provided in RG 1.47 applies to HTGR/VHTRs.

HTGR/VHTRs that are being considered for possible deployment in the next decade will probably use digital computer-based I&C systems. For such systems as part of a safety system or systems, additional requirements might be imposed, including but not limited to the single-failure criterion of IEEE Std 603-1991, Section 5.1.

6.2.4.3 Regulatory Guide 1.53—Application of the Single-Failure Criterion to Safety Systems (Current Revision 2, November 2003)

Regulatory Guide 1.53 [Ref. 55] endorses IEEE Std 379-2000, "Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems,"[56] which provides methods acceptable to the NRC staff for satisfying the NRC's regulations with respect to the application of the single-failure criterion to the electric power, instrumentation, and control portions of nuclear power plant safety systems.

Single failure means an occurrence that results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. IEEE The Authoritative Dictionary of IEEE Standards Terms[57] describes *single point of failure* as, with respect to a system, a failure that would result in the inability of that system to perform its intended function.

HTGR/VHTR safety system design philosophy should adopt the traditional approach; hence, the guidance provided in RG 1.53 applies.

6.2.4.4 Regulatory Guide 1.62—Manual Initiation of Protection System Actions (Current Revision 1, June 2010)

Regulatory Guide 1.62 [Ref. 58] describes a method that the NRC staff considers acceptable for use in complying with the NRC's regulations with respect to the means for manual initiation of protective actions provided (1) by otherwise automatically initiated safety systems or (2) as a method diverse from automatic initiation. This framework consists of a number of regulations and supporting guidelines

applicable to manual initiation of protective actions including, but not limited to, GDC 1, "Quality Standards and Records"; GDC 13, "Instrumentation and Control"; GDC 21, "Protection System Reliability and Testability"; and GDC 22, "Protection System Independence," as set forth in Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10, of the *Code of Federal Regulations*, Part 50, "Domestic Licensing of Production and Utilization Facilities," (10 CFR Part 50) [Ref. 39].

Regulatory Guide 1.62 endorses IEEE Std 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," and IEEE Std 603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations" and the correction sheet, dated January 30, 1995, as acceptable methods to meet NRC's regulatory requirements.

Regulatory position on the diverse means for manual initiation is discussed in Point 4 of Branch Technical Position (BTP) 7-19, "Guidance for Evaluation of Diversity and Defense-In-Depth in Digital Computer-Based Instrumentation and Control Systems,"[59] of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," issued March 2007 [Ref. 41].

Although the reactor and heat transport system designs of HTGR/VHTRs dramatically differ from that of water-cooled reactors, the safety system design philosophy for instrumentation and control systems should adopt the traditional approach; hence, the guidance provided in RG 1.62 applies.

6.2.4.5 Regulatory Guide 1.75—Criteria for Independence of Electrical Safety Systems (Current Revision 3, February 2005)

Regulatory Guide 1.75 [Ref. 60] describes a method acceptable to the NRC staff for complying with the NRC's regulations with respect to the physical independence requirements of the circuits and electric equipment that comprise or are associated with safety systems.

Regulatory Guide 1.75 endorses IEEE Std 384-1992, "Standard Criteria for Independence for Class 1E Equipment and Circuits," which provides a method that the NRC staff considers acceptable for satisfying the agency's regulatory requirements with few additional requirements.[61]

Although the reactor and heat transport system designs of HTGR/VHTRs dramatically differ from that of water-cooled reactors, the safety system design philosophy for instrumentation and control systems should adopt the traditional approach; hence, the guidance provided in RG 1.75 applies.

6.2.4.6 Regulatory Guide 1.97—Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants (Current Revision 4, June 2006)

Regulatory Guide 1.97 [Ref. 62] describes a method that the NRC staff considers acceptable for use in complying with the agency's regulations with respect to satisfying criteria for accident monitoring instrumentation in nuclear power plants. The method described in RG 1.97 specifically addresses GDC 13, "Instrumentation and Control"; GDC 19, "Control Room"; and GDC 64, "Monitoring Radioactivity Releases," as set forth in Appendix A to Title 10, Part 50 (10 CFR 50).

Moreover, Subsection (2)(xix) of 10 CFR 50.34(f), "Additional TMI-Related Requirements," requires operating reactor licensees to provide adequate instrumentation for use in monitoring plant conditions following an accident that includes core damage.

Regulatory Guide 1.97, Rev. 4 endorses—with certain clarifying regulatory positions—the IEEE Std 497-2002, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations."[63] Regulatory Guide 1.97 is intended for licensees for new nuclear power plants; therefore, it also applies to HTGR/VHTRs.

The aftermath of the accident at Three Mile Island, Unit 2 (TMI-2), in 1979 indicated that a more rigorous regulatory approach be adopted for accident monitoring systems. The initiatives and steps taken following that tragic event resulted in three major sources of related requirements:

1. ANSI/ANS-4.5-1980, "Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors,"
2. IEEE Std 497-1981, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations," and
3. Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident."

Revision 3 of RG 1.97 quickly became the *de facto* standard for accident monitoring. However, technological advances made since the release of the revision requires an update and rewrite of the guidance to incorporate the new technology and address potential vulnerabilities caused primarily by the adoption of modern digital technology.

The contribution of RG 1.97, Rev. 4, is the adoption of performance-based criteria for use in selecting variables instead of prescribing the instrument variables to be monitored and standardization of the criteria based on the accident management functions of the given type of variable rather than providing design and qualification criteria. These efforts resulted in the development of IEEE Std 497-2002 by the IEEE Power Engineering Society, Nuclear Power Engineering Committee, Subcommittee 6, Working Group 6.1, "Post-Accident Monitoring." IEEE Std 497-2002 is endorsed by RG 1.97 subject to a number of regulatory positions.

Key distinctive characteristics of HTGR/VHTRs from light-water-cooled reactors are that (1) the fuel is contained within graphite-coated microspheres (called TRISO), which are in turn dispersed and confined in a larger graphite matrix—either prismatic blocks or pebbles and (2) the core contains a large amount of graphite mass providing a substantial thermal inertia. The TRISO fuel structure provides the functionality of a miniature pressure vessel—as in a light-water-cooled reactor—to contain the fissionable material and fission products that are generated. Therefore, HTGR/VHTRs offer additional barriers against the release of radioactive species compared to a conventional light-water-cooled reactor. Secondly, the large graphite mass provides substantial heat capacitance effectively damping the potential temperature excursions during a design basis accident or during abnormal operational occurrences.

These distinctive features of HTGR/VHTRs create a significantly different set of design bases; therefore require consideration of different design basis accidents. Furthermore, highly elevated temperatures—compared to a conventional light-water-cooled reactor—require special considerations. HTGR/VHTRs are naturally expected to have different monitoring requirements for process variables during an accident.

IEEE Std 497-2002 establishes flexible, performance-based criteria for the selection, performance, design, qualification, display, and quality assurance of accident monitoring variables. These variables are intended to be the primary sources of information for operators to monitor the accident. IEEE Std 497-2002 also cites several industry codes and standards for human factors criteria.[41,64,65]

The flexible structure of IEEE Std 497-2002 and the method of selecting the process variables for accident monitoring based on performance-based criteria makes this guidance an appropriate tool for a wide variety of reactors. Annex A to IEEE Std 497-2002 states that the required accuracy of an accident monitoring channel is established based on the channel's assigned function. The most appropriate set of process variables can be selected based on an objective function—e.g., keeping the pressure vessel temperature below a threshold—subject to safety analysis calculations.

Annex A.3 to IEEE Std 497-2002 suggests typical accuracy values for variables according to their safety significance as indicated in Annex A.2. These values are typical of light-water-cooled reactors, and their validity for HTGR/VHTRs should be reassessed.

Based on the aforementioned discussion, RG 1.97 can be used as an acceptable guidance for HTGR/VHTRs with special provisions.

6.2.4.7 Regulatory Guide 1.105—Setpoints for Safety-Related Instrumentation (Current Revision 3, December 1999)

Regulatory Guide 1.105 [Ref. 66] endorses Part 1 of ISA-S67.04-1994, "Setpoints for Nuclear Safety-Related Instrumentation,"[67,68] which provides a basis for establishing trip setpoints for nuclear instrumentation for safety systems and addresses known contributing errors in a safety-related communication channel. The guidance in RG 1.105 intends to provide an acceptable method to meet the requirements of GDC 13, "Instrumentation and Control"; and GDC 20, "Protection System Functions," of Appendix A to 10 CFR Part 50 as well as Paragraph (c)(1)(ii)(A) of § 50.36, "Technical Specifications," of 10 CFR Part 50 [Ref. 39].

Trip setpoints are chosen to assure that a trip or safety actuation occurs before the process reaches its predetermined analytical limit. Analytical limits of process variables are established by the safety analysis calculations. Trip setpoint calculations incorporate all the uncertainties involved in the measurement loop, including instrument calibration uncertainties, instrument uncertainties due operational variations, drift, etc. ISA-S67.04-1994 suggests that these uncertainties be combined by an acceptable statistical method, including square-root-sum-of-squares, arithmetic sum, probabilistic modeling, stochastic modeling, or a combination thereof.

Regulatory Guide 1.105 was specifically written as a guidance document for light-water-cooled reactors. Because of the drastic differences in operating conditions between a water-cooled reactor and a gas-cooled reactor, methods described by the endorsed standard ISA-S67.04-1994 must be revised and their applicability be reassessed.

Furthermore, a systematic method must be developed to establish the safety-related process and nuclear variables to be used as trip setpoint parameters. This issue was discussed in Section 6.2.4.6 for RG 1.97.

Based on these discussion points, applicability of RG 1.105 for HTGR/VHTRs must be reevaluated.

6.2.4.8 Regulatory Guide 1.118—Periodic Testing of Electric Power and Protection Systems (Current Revision 3, April 1995)

Regulatory Guide 1.118 [Ref. 69] describes a method acceptable to the NRC staff for complying with the Commission's regulations with respect to the periodic testing of the electric power and protection systems. Regulatory Guide 1.118 addresses the requirements set forth by Paragraph (h), "Protection Systems." of Section 50.55a, "Codes and Standards," of 10 CFR Part 50, which requires protections systems meet the requirements in IEEE Std 279-1991 [Ref. 50]. Regulatory Guide 1.118 also addresses GDC 21, "Protection System Reliability and Testability"; and GDC 18, "Inspection and Testing of Electric Power Systems," of Appendix A to 10 CFR Part 50.

Regulatory Guide 1.118 endorses IEEE Std 338-1987, "Criteria for Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems."[70]

Regulatory Guide 1.118 should apply for HTGR/VHTRs.

6.2.4.9 Regulatory Guide 1.151—Instrument Sensing Lines (Current Revision 1, July 2010)

Regulatory Guide 1.151 [Ref. 71] describes a method that the NRC staff considers acceptable for use in complying with the agency's regulations with respect to the design and installation of safety-related instrument sensing lines in nuclear power plants. To meet these objectives, the sensing lines must serve a safety-related function to prevent the release of reactor coolant as a part of the reactor coolant pressure boundary and to provide adequate connections to the reactor coolant system for measuring process variables (e.g., pressure, level, and flow). The rules and regulations, which RG 1.151 addresses include but not limited to GDC 1, "Quality Standards and Records"; GDC 13, "Instrumentation and Control"; GDC 24, "Separation of Protection and Control Systems"; and GDC 55, "Reactor Coolant Pressure Boundary Penetrating Containment," as set forth in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50.

Regulatory Guide 1.151 endorses ANSI/ISA-67.02.01-1999 [Ref. 72] as a standard document that provides an acceptable method in satisfying the agency's regulatory requirements with respect to designing and installing safety-related instrument sensing lines in nuclear power plants with a few exceptions.

Regulatory Guide 1.151 makes references to a number of operational events, in which evolved gases in instrument sensing lines have affected measured water level. These inaccuracies in level measurement can affect the performance of safety functions in light-water-cooled reactors. Other events include failure of pressure transmitters due to hydrogen permeation into the sensor cell.

These failure modes, as explained, are very specific to light-water-cooled reactors. Measurement offset in gauge and differential pressure lines due to trapped gas is a common problem in liquid-cooled systems. There are a number of mechanisms that retain these gas species including surface roughness and adsorption. Once sufficient energy is generated, these species are liberated from their sites. Since they are most likely nondissolvable and noncondensable gases, they tend to migrate and might get trapped within small chambers, such as the closures within instrument sensing lines.

HTGR/VHTRs contain large amounts of graphite, which has micro pores that can adsorb certain molecules including, but not limited to, water vapor. This situation is further exacerbated in pebble-bed designs due to increased graphite surface area. Once the reactor is started, even before criticality, these species disengage from their sites and might find their way into the instrumentation, potentially affecting the performance. These concerns also apply to sampling lines.

The cited standards, such as ANSI/ISA-67.02.01-1999, seem to specifically address the light-water-cooled reactor types. Therefore, the methods suggested in these documents are not directly applicable to HTGR/VHTRs. The functional objectives of these guides must be adapted for gas-cooled reactors and can be incorporated into another regulatory guide, or the guide may be rewritten to allow a technology-neutral perspective with references specific to reactor types.

Regulatory Guide 1.151 need not be used for the purposes indicated therein.

6.2.4.10 Regulatory Guide 1.180—Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems (Current Revision 1, October 2003)

The NRC's regulations in Part 50, "Domestic Licensing of Production and Utilization Facilities," of Title 10 of the Code of Federal Regulations (10 CFR Part 50) state that structures, systems, and components (SSCs) important to safety in a nuclear power plant are to be designed to accommodate the effects of environmental conditions (i.e., remain functional under all postulated service conditions), and that design control measures such as testing are to be used to check the adequacy of design.

Electromagnetic interference (EMI), radio-frequency interference (RFI), and power surges have been identified as environmental conditions that can affect the performance of safety-related electrical equipment. Confirmatory research findings to support this observation can be found in NUREG/CR-5700, *Aging Assessment of Reactor Instrumentation and Protection System Components*;[73] NUREG/CR-5904, *Functional Issues and Environmental Qualification of Digital Protection Systems of Advanced Light-Water Nuclear Reactors*;[74] NUREG/CR-6406, *Environmental Testing of an Experimental Digital Safety Channel*;[75] and NUREG/CR-6579, *Digital I&C Systems in Nuclear Power Plants: Risk-Screening of Environmental Stressors and a Comparison of Hardware Unavailability With an Existing Analog System*.[76] Therefore, controlling electrical noise and the susceptibility of I&C systems to EMI/RFI and power surges is an important step in meeting the aforementioned requirements.

Regulatory Guide 1.180 [Ref. 77] endorses design, installation and testing practices acceptable to the NRC staff for addressing the effects of EMI/RFI and power surges on safety-related I&C systems in a nuclear power plant environment. Of particular interest is the equipment upgrades or replacements in existing analog I&C systems in operating nuclear power plants and new I&C system designs that make extensive use of digital technology. Digital systems may exhibit greater vulnerability to the EMI/RFI fields that exist in a nuclear power plant environment. Moreover, digital technology rapidly evolves and designers push the system limits on a daily basis, either by increasing clock frequencies, which increases the spectral power of the broadcast component; lower logic-level voltages, which makes the circuit more susceptible to external disturbances such as single-event upsets at the device level; and shrinking feature sizes, which increase the leakage current through the gate and makes the whole design more susceptible to cross-talk between independent elements.

The typical environment in a nuclear power plant includes many sources of electrical noise, for example, hand-held, two-way radios; arc welders; switching of large inductive loads; high fault currents; and high-energy fast transients associated with switching at the generator or transmission voltage levels. The increasing use of advanced analog- and microprocessor-based I&C systems in reactor protection and other safety-related plant systems has introduced concerns with respect to the creation of additional noise sources and the susceptibility of this equipment to the electrical noise already present in the nuclear power plant environment.

Regulatory Guide 1.180 was prepared as a regulatory guidance document to complement the previously accepted method proposed by Electric Power Research Institute (EPRI) Topical Report TR-102323, *Guidelines for Electromagnetic Interference Testing in Nuclear Power Plants*,[78] in a Safety Evaluation Report (SER)[79] as one method of addressing issues of electromagnetic compatibility for safety-related digital I&C systems in nuclear power plants.

These concerns equally apply to HTGR/VHTR I&C systems; RG 1.180 must be used as a guidance document to address the EMI/RFI vulnerability issues.

6.2.5 Branch Technical Positions

The Branch Technical Positions (BTPs) represent guidelines intended to supplement the acceptance criteria established in Commission regulations and the guidelines provided in regulatory guides and applicable industry standards. The BTPs are written to resolve technical problems or questions of interpretation that arise in the detailed reviews of plant designs. A BTP is primarily an instruction to staff reviewers that outlines an acceptable approach to the particular issues and is intended to ensure a uniform treatment of the issue by staff reviewers. The approaches taken in the BTPs, like the recommendation of regulatory guides, are not mandatory, but do provide defined, acceptable, and immediate solutions to some of the technical problems and questions of interpretation that arise in the review process. Therefore, they can be used as guidelines for license applicants.

6.2.5.1 Branch Technical Position 7-8—Guidance for Application of Regulatory Guide 1.22 (Current Revision 5, March 2007)

BTP 7-8 [Ref. 80] clarifies that protection system components that cannot be tested during reactor operation should be identified and a discussion of conformity with the provisions of paragraph D.4 of RG 1.22 be provided.

6.2.5.2 Branch Technical Position 7-9—Guidance on Requirements for Reactor Protection System Anticipatory Trips (Current Revision 5, March 2007)

Several reactor designs have incorporated a number of anticipatory or “back-up” trips for which no credit was taken in the accident analyses. These trip systems included nonsafety-grade equipment to perform the protective functions. The NRC staff concurred that this was not an acceptable practice because of possible degradation of the reactor protection system.

BTP 7-9 [Ref. 81] stipulates that all reactor trips included in the reactor protection system should be designed to meet the requirements of IEEE Std 279-1971 or IEEE Std 603-1991. This position applies to the entire trip function—from the sensor to the final actuated device.

Regulatory position in BTP 7-9 applies to HTGR/VHTR instrumentation system designs.

6.2.5.3 Branch Technical Position 7-10—Guidance on Application of Regulatory Guide 1.97 (Current Revision 5, March 2007)

BTP 7-10 [Ref. 82] provides additional guidelines on accident monitoring instrumentation for reviewing an application.

The acceptance criteria in Section B.3 further clarify the agency’s position on accident monitoring instrumentation for various issues and challenges. BTP 7-10 makes special provisions between RG 1.97, Rev. 4, and earlier revisions of the guide. The license applications for HTGR/VHTRs should follow Rev. 4 of RG 1.97 with special provisions for this reactor type, as indicated earlier in Section 6.2.4.6. It might be possible and practical that a new revision is adopted that renders the guide technology neutral.

BTP 7-10 can be used as a regulatory guide with special provisions.

6.2.5.4 Branch Technical Position 7-11—Guidance on Application and Qualification of Isolation Devices (Current Revision 5, March 2007)

BTP 7-11 [Ref. 83] addresses the electrical qualification and application of isolation devices, and amplifies the requirements in RG 1.75 and the acceptance criteria IEEE Std 603-1991 or IEEE Std 279-1971.

BTP 7-11 can be used as a guidance document for HTGR/VHTRs.

6.2.5.5 Branch Technical Position 7-12—Guidance on Establishing and Maintaining Instrument Setpoints (Current Revision 5, March 2007)

BTP 7-12 [Ref. 84] provides additional guidelines for reviewing the process that a license applicant follows to establish and maintain instrument setpoints.

Establishing and maintaining setpoints for safety-related instrumentation was already discussed in Sect. 6.2.4.7 and will not be further treated here. Just as the RG 1.105, BTP 7-12 may apply to HTGR/VHTRs with provision.

6.2.5.6 Branch Technical Position 7-13—Guidance on Cross-Calibration of Protection Systems Resistance Temperature Detectors (Current Revision 5, March 2007)

BTP 7-13 [Ref. 85] is intended to identify the information and methods acceptable to the staff for using cross-calibration techniques for surveying the performance of resistance temperature detectors (RTDs). These guidelines are based on experience in the detailed reviews of applicant/licensee submittals describing the application of in-situ cross-calibration procedures for reactor coolant RTDs as well as NRC research activities.

RTDs—also called resistance thermometers or resistive thermal devices—are temperature sensors that exploit the predictable change in electrical resistance of some materials with varying temperature. As they are almost invariably made of platinum, they are often called platinum resistance thermometers (PRTs). RTDs have advantages in high accuracy, low drift, wide operating range and suitability for precision applications. However, they are rarely used at temperatures above 660ºC since it becomes increasingly difficult to prevent the platinum from becoming contaminated by impurities from the metal sheath of the thermometer.[86]

Because of the temperatures involved in the primary loop of HTGR/VHTRs, RTDs will not likely be used as temperature sensors, particularly not in the sensing lines to trigger protective functions. This BTP does not apply to HTGR/VHTRs.

6.2.5.7 Branch Technical Position 7-17—Guidance on Self-Test and Surveillance Test Provisions (Current Revision 5, March 2007)

BTP 7-17 [Ref. 87] is intended to provide guidelines for reviewing the design of the self-test and surveillance test provisions. It complements and clarifies the guidance provided in RGs 1.22, 1.47, 1.53, 1.118, and 1.152. All of these guidance documents have been previously discussed, excluding RG 1.152, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants," which endorses IEEE Std 7-4.3.2-2003, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations."[88] IEEE Std 7-4.3.2-2003 is the *de facto* standard document that is mandated by 10 CFR Part 50 to be followed during the design, manufacturing, installation, operation, and maintenance of digital computers used in safety systems.

Based on the discussions in the foregoing sections (i.e., Sects. 6.2.4.1-10), guidance provided in BTP 7-13 should apply to HTGR/VHTRs.

6.2.6 NUREG publications

NUREG publications are prepared by the NRC staff to express regulatory position on a subject matter.

6.2.6.1 NUREG-0737—Clarification of TMI Action Plan Requirements

NUREG-0737 [Ref. 89] introduces additional regulatory positions on instrumentation, control, and human factors engineering aspect of a reactor design. A large number of post-Three Mile Island (TMI) requirements require the installation of a number of additional control room indications. This regulatory document requires that due consideration is given to human-factors engineering in planning for the installation of such new control room equipment.

Though HTGR/VHTR dynamics and event time constants significantly differ than that of light-water-cooled reactors, the guidance provided in NUREG-0737 should be useful in designing and implementing the instrumentation and control system for the plant, as well as planning the workforce for operation.

NUREG-0737 should be considered as a guidance document with provisions.

6.2.6.2 NUREG-1338—Draft Preapplication Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor

NUREG-1338 [Ref. 90] is a draft safety evaluation report (SER), which presents the preliminary results of a preapplication design review for the standard modular high-temperature gas-cooled reactor (MHTGR) (Project 672). The MHTGR conceptual design was submitted by the U.S. Department of Energy (DOE) in accordance with the U.S. Nuclear Regulatory Commission (NRC) "Statement of Policy for the Regulation of Advanced Nuclear Power Plants" (51 FR 24643), which provides for early Commission review and interaction. The standard MHTGR consists of four identical reactor modules, each with a thermal output of 350 MW(t) coupled with two steam turbine-generator sets to produce a total plant electrical output of 540 MW(e). The reactors were helium-cooled and graphite-moderated and used coated particle-type nuclear fuel, similar to the suggested fuel designs for various HTGR/VHTR concepts. The design included passive reactor-shutdown and decay-heat-removal features, also similar features as the current gas-cooled reactor concepts.

NUREG-1338 presents the NRC staff's technical evaluation of those features in the MHTGR design important to safety, including their proposed research and testing needs. In addition, it also presents the criteria proposed by the NRC staff to judge the acceptability of the MHTGR design and, where possible, includes statements on the potential of the MHTGR to meet these criteria.

Technical evaluation of the MHTGR presented in NUREG-1338 represents the staff's perception and position on the safety characteristics of earlier gas-cooled reactors. These assessments should be reflective of the understanding of and expectations from the new advanced reactor designs by the staff.

6.2.7 IEEE standards

All of the IEEE standards have been previously discussed in Sect. 6.2.4 and will not be deliberated here. Compliance with some IEEE standards is mandated by law, as required by 10 CFR Part 50; and some of them are endorsed—either partially or as a whole—by regulatory guides.

6.2.8 ISA standards

Published by The International Society of Automation (formerly known as The Instrument Society of America and later The Instrumentation, Systems, and Automation Society), the ISA standards endorsed by the NRC staff have been previously discussed in Sect. 6.2.4.

6.2.9 IEC standards

NRC has not endorsed any IEC standard publications, except for those that have been published in collaboration with ANSI, such as IEC 61000 series on "Electromagnetic Compatibility (EMC)."[91] However, IEC standards are useful resources to develop a technical basis, particularly for areas that have not yet been adequately addressed by the Commission or the NRC staff.

6.2.10 IAEA publications

IAEA publications—and IAEA's position for that matter—on regulatory requirements and acceptance criteria have no binding status for NRC. However, some IAEA publications include very useful information on design, analysis, operation, maintenance, and licensing experience on various reactor types including the gas reactors.

Of particular interest is IAEA-TECDOC-1366, *Considerations in the Development of Safety Requirements for Innovative Reactors: Application to Modular High-Temperature Gas Cooled Reactors*, which focuses on the Modular High-Temperature Gas-Cooled Reactor (MHTGR).[92] IAEA-TECDOC-

1366 proposes a technical basis and methodology, based on principles of defense-in-depth, for conducting design safety assessments, and in the long term, generating design safety requirements for innovative reactors wherein the MHTGR is used as an example to illustrate this process.

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Letter Report

Task 3. New Instrumentation and Requirements for HTGRs

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ACRONYMS

AGR	advanced gas reactor (DOE program)
AOO	anticipated operational occurrence
ATWS	anticipated transient without scram
AVR	Arbeitsgemeinschaft VersuchsReaktor
BDBA	beyond design basis accident
CFR	code of federal regulations
CFD	computational fluid dynamics
CTE	coefficient of thermal expansion
DBA	design basis accident
D-LOFC	depressurized loss-of-forced circulation
DOCXO	double oven controlled crystal oscillator
DOE	(U.S.) Department of Energy
ESA	electrical signature analysis
FP	fission products
FSV	Fort St. Vrain (reactor)
GA	General Atomics
GE	General Electric
GRSAC	graphite reactor severe accident code
GT-MHR	gas turbine modular helium reactor
HTGR	high-temperature gas-cooled reactor
HTR-10	high temperature reactor – 10 MW (China)
HTR-10GT	HTR-10 adapted for gas turbine PCU
HTTR	high temperature engineering test reactor (Japan)
I&C	instrumentation and controls
IAEA	International Atomic Energy Agency
IC	integrated circuit
IHX	intermediate heat exchanger
INL	Idaho National Laboratory
IR	infrared
JNT	Johnson noise thermometry
LDV	laser Doppler velocimetry
LED	light-emitting diode
LOFC	loss-of-forced circulation
LPT	low pressure turbine
LSHT	liquid salt heat transfer
LWR	light-water reactor
MHTGR	modular high-temperature gas-cooled reactor (DOE 1980s design)
MPFD	micro-pocket fission detector
MSRE	molten salt reactor experiment (ORNL)
NGNP	next generation nuclear plant
NRC	Nuclear Regulatory Commission

OFDR	optical frequency domain reflectometry
ORNL	Oak Ridge National Laboratory
PBMR	Pebble Bed Modular Reactor (South Africa)
PBR	pebble bed reactor
PCU	power conversion unit
P-LOFC	pressurized loss-of-forced circulation
PCU	power conversion unit
PNNL	Pacific Northwest National Laboratory
PRV	pressure relief valve
PSID	preliminary safety information document
R&D	research and development
RCCS	reactor cavity cooling system
RPV	reactor pressure vessel
RTD	resistance temperature detector
SCS	shutdown cooling system
SG	steam generator
SSC	structures, systems, and components
SysML	system modeling language
T/H	thermal hydraulics
TMI	Three Mile Island
TRISO	tri-layer isotropic (particle fuel)
UHS	ultimate heat sink
V&V	verification and validation
VHTR	very high temperature gas-cooled reactor

1. HTGR MEASUREMENT AND INSTRUMENTATION VISION

The prior evaluations of measurement and instrumentation at high temperature gas-cooled reactors (HTGRs) performed in this project have focused on specific technical aspects of the sensor system such as the measurement accuracy or sensor environmental tolerance. The instrumentation system, however, extends from the sensor elements through the local information processing and the communication network to the control or safety decision point. In essence, the instrumentation system functions as the power plant's nervous system over its entire lifetime under the full range of adverse conditions it may experience. The system performance requirements, aging, maintenance (including periodic replacement), installation, access requirements, and post-severe accident survival all need to be considered as part of the design and safety evaluation process. The objective of this report is to provide a holistic vision of measurement and instrumentation at HTGRs employing best of breed for the new and emerging instrumentation technologies to enable seeing long-term operational and safety consequences of design phase instrumentation decisions.

The licensing basis and criteria for HTGRs are still being decided and are beyond the scope of this report. However, HTGRs have fundamental physics differences from light-water reactors (LWRs), and their safety and licensing will need to reflect the differences between the reactor classes. The approach taken in this exploratory report is to describe the potential performance and known vulnerabilities of new and emerging reactor instrumentation to support HTGR operation, accident response, and maintenance and to not make any judgment as to the adequacy of any particular architecture or sensor to perform a safety function. Therefore, nothing in this report should be construed as an endorsement or implied approval of any instrument or architecture. Moreover, as the topic area of new instrumentation and issues associated with its application to HTGRs is vast, this report will only identify new instrumentation deployment issues at an overview level while providing a description of new sensing technology at somewhat greater depth.

1.1 Safety Instrumentation Requirements

This section maps the components and functions of an HTGR's instrumentation system to its performance and safety requirements. The overarching goal of any nuclear safety system is prevention of large, off-site release of radionuclides. The instrumentation system supports this goal by providing accurate, timely plant status information under both normal and accident conditions both to the plant operators and to its automatic safety systems. The instrumentation system is also responsible for transporting control decisions to the plant's actuators as well as keeping a record of both the plant status and the control decisions. The HTGR's instrumentation system goal is to provide the process information necessary to both maximize power production and to preserve the plant investment value. The selection and design of the particular sensors and networking technologies that embody an HTGR's instrumentation system thus arises from and amalgam of the component environmental tolerance requirements and the information requirements to safely and efficiently operate the plant.

Nearly all of the conceptual underpinnings relating to the division of a nuclear power plant's instrumentation into safety-related and nonsafety-related systems transfer directly from LWRs to HTGRs. Safety-related instrumentation is generally used in the protection systems to ensure a timely and appropriate response to an accident or anticipated operational occurrence (AOO). Much as in LWRs, a combination of safety-related and nonsafety-related instrumentation is employed during normal operations. For example, control rod position monitors (safety-related) are employed by the control system (nonsafety-related) in performing a power level transition.

The performance requirements for an HTGR's instrumentation system arise from its functional requirements which in turn derive from the plant's physical characteristics and potential accident scenarios. The instrumentation system's basic safety function can be expressed in how it supports the plant's three fundamental safety functions:

1. confinement of radioactive material,
2. control of the core reactivity, and
3. removal of the heat from the core.

The heat removal function can be fulfilled by various design alternatives, but generally three independent systems are common in HTGRs:

1. nominal heat removal path, i.e., the primary and the secondary heat transport systems, the power conversion unit (PCU), and the ultimate heat sink (UHS);
2. shutdown cooling system (SCS), which is used to remove post-shutdown decay heat if the primary system is not functioning; and
3. reactor cavity cooling system (RCCS), which is typically an air (or water) natural circulation cooling system intended to provide the majority of decay heat removal for loss-of-forced circulation accidents, as well as to keep the reactor cavity below ~100°C to prevent damage to the cavity concrete during normal operation.

The RCCS is generally identified as a passive safety system and is designed to be operational at all times (i.e., during normal as well as abnormal conditions). In some RCCS designs, operation during normal conditions utilizes forced cooling but falls back on natural convection in loss-of-power situations. The heat removed by the RCCS during normal operation should be minimized to reduce parasitic losses, but it needs to be adequate to meet the post-shutdown decay heat needs, primarily to prevent overheating of the reactor pressure vessel.

Initiating events often challenge more than one safety function at a time. For instance, primary system pressure boundary breaks directly challenge confinement of radioactivity. However, depending on the size and location of a break, it can also challenge the core heat removal capability. The instantaneous reactor power is a key safety parameter. The reactor power has both neutronic and thermal aspects. Measurements of the two aspects of the reactor power are employed as instrumentation cross checks.

Another commonly used method to determine the instrumentation system requirements is based on providing the information necessary to respond to the dominant phenomena in the event sequence. Example events include:

1. primary system breaks,
2. loss of primary system heat sink,
3. air ingress events,
4. water ingress events,
5. reactivity transients,
6. depressurized loss-of-forced cooling (D-LOFC),
7. pressurized loss-of-forced cooling (P-LOFC),
8. turbine trip, and
9. station blackout.

The instrumentation system's design needs to include sufficient ruggedness and independence to continue to function under adverse conditions imposed by severe accidents. Much as with LWRs, an HTGR's instrumentation system is arranged into divisions to enable cross checking the performance of multiple sensors and measurement methods, thereby avoiding spurious trips and providing an indication of the health of particular sensors.

This section is structured in terms of postulated accidents and how the instrumentation system is configured to acquire and transfer information in support of accident avoidance and mitigation. The particular accidents and AOOs analyzed are those identified in the earlier Task 1 and Task 2 letter reports.[1,2] The measurement technology, sensitivity, and time response characteristics of the instrumentation necessary to respond to accidents and AOOs are identified in this section. Further, both accidents and plant component or structural failures often have precursor indicators that, if measured, can provide advance warning. The technologies and methodologies for providing advanced warning of future failures as well as the current plant health status are the subject of the diagnostics and prognostics section later in this report.

1.2 Normal Operation

The purposes for safety-related measurements at HTGRs during normal operation are to confirm that the plant parameters remain within design limits and plant systems, structures, and components (SSCs) remain capable of responding correctly to accident sequences. Since HTGRs are thermal power plants, the focus of the parameter measurements is on confirming proper heat balances throughout the plant subsystems. The SSC monitoring function confirms that the major radionuclide release barriers remain in place and that both safety-related equipment is functioning as intended (e.g., that the control rods drive mechanisms position the rods as intended) and accident mitigation equipment remains ready to respond (e.g., power is available to the secondary cooling system). Additional monitoring is performed to assess the amount of radioactive material leaking from the plant (e.g., noble gases from the helium cleanup system) and to map dose and/or contamination levels within the plant. The plant security status is also continuously monitored.

The plant monitoring requirements change during refueling. The fuel handling process needs to be monitored to ensure that fuel is not mechanically damaged during handling and that worker dose is minimized. The decay heat from used HTGR fuel needs to be rejected to the environment. Recently removed fuel may require active cooling to avoid fuel failure. If active cooling is used, the cooling system will require safety-grade power. However, unpressurized, below grade fuel storage vaults would not have a strong mechanism to cause release of the radionuclides from even failed fuel to the environment. Given the large decrease in decay heat with time, the function for any active cooling systems for fuel that has been removed from the reactor for at least several months would be to avoid overheating the surrounding structural concrete (preserving the plant value) rather than avoiding failure of the fuel coating layers.

1.2.1 Background

The modular HTGR's core heat transport design is fundamental to achieving the reactor's passive safety objectives. Those inherent safety features are the high-quality ceramic-coated particle (trilayer isotropic—TRISO) fuel, the single-phase inert coolant (helium), and the reactor's post-shutdown decay heat removal features consistent with TRISO fuel temperature design limits. The core's features of a combination of low-power density, large size and heat capacity, high-thermal conductivity, and large fuel thermal margins result in very slow progressions of postulated core heat up accidents. As a result, the design can allow very long times to provide safety-significant responses to loss of all shutdown cooling mechanisms. Under such conditions, core heat removal is accomplished via heat transfer from the core to the noninsulated reactor pressure vessel via conduction, radiation, and (if helium coolant is present) convection, and from the vessel to the reactor cavity and RCCS by radiation and convection.

A primary design characteristic of modular HTGR cores is the limitation of rated thermal power (and power densities) to small fractions of those typical of LWRs. While this difference may give safety advantages to HTGRs, the in-core instrumentation in HTGRs is much less extensive and informative. In pebble bed reactor (PBR) cores, where the pebble fuel is continuously recirculated, it is virtually impossible to measure fuel temperatures and localized flow rates. Measurements within prismatic

(stationary) HTGR cores are also difficult due to the hostile, very high in-core temperature environments, some beyond the range where sensors can operate reliably. Thus, fuel temperatures must be inferred from external measurements and calculations, or after the fact from dummy balls equipped with melt-wire sensors to infer maximum exposure temperature during a pass through the core.

Fuel temperature calculation uncertainties are due to both the variations in localized fuel power densities and the uncertainties in the localized coolant flow rates and temperatures. In both types of cores, there are typically steep flux gradients at the fuel-reflector interfaces. In pebble bed cores, there are wide variations in individual pebble power densities: burn up variations, since they are recycled (~6 times); and pebble locations (e.g., they can either traverse the core near the hot center or near the cooler side reflectors). (Note that current PBR designs have cylindrical—i.e., no central reflector—cores.) Recently noted concerns about pebble flow uncertainties are due to the reduced pebble friction (in helium) at higher temperatures, which tends to make the recirculating pebbles in the central regions flow faster than those in the cooler periphery, thus exacerbating the central power peaking concerns and increasing the uncertainties of core performance predictions.[3]

Localized core flow uncertainties are significant in both types of cores. That is because of the increase in helium viscosity with temperature, which tends to reduce the coolant flow to the hotter areas of the core (a positive feedback effect). In addition, the variable spacing between blocks (side reflector in the PBR and the central and side reflectors plus the fuel blocks in the prismatic cores), lead to uncertainties in the “bypass flow” (defined as the core helium flow that does not directly cool the fuel elements). Bypass flows (which cannot be measured) are estimated to be between 5 and 15% of the total. Gap variations between blocks are due to thermal gradients and the warping due to irradiation of the graphite.

Thermal gradients in HTGR cores are relatively large. The coolant mean-temperature-rise across the core at full power (~400ºC) is about a factor of 10 higher than that for LWRs. Spatial variations in coolant outlet temperatures are also quite large (up to ~200ºC or more) due to variations in radial coolant flows and power densities (especially for cylindrical cores).

Determining the mixed mean primary coolant temperature for an HTGR is technically challenging due to the inadequately mixed cooling streaming and swirling within the piping. Lack of fully mixed primary flow is both deleterious to the downstream structural alloys as well as to performing the primary heat balance measurements. Fort St. Vrain (FSV) included variable orifices for the core’s 27 radial regions to attempt to equalize the coolant outlet temperatures to minimize the mixing problems—such orifices are not present in the current designs.

Thermocouples installed at FSV, HTR-10, and HTTR in the reflector blocks were useful in estimating core thermal performance.[4–6] The extrapolations from these “ex-core” measurements to estimated fuel temperatures require measurements (and estimates) of nuclear power distributions and coolant flow in addition to a comprehensive core T/H on-line simulation. Temperature distribution measurements in the bottom reflector or core support blocks (such as were installed in FSV) can be used to refine estimates of fuel temperature distributions during normal operation. In addition to the proximity problem with ex-core temperature measurements being far from the fuel, there is typically a relatively large time lag involved with sensors imbedded in the large graphite blocks.

Unlike LWR vessels, the reactor pressure vessels (RPVs) for HTGRs are (mostly) uninsulated in such a way that in long-term LOFC accidents, the afterheat can be removed by the RCCS, and that the pressure vessel steel temperature remains below design limits. The RCCS may not be necessary to prevent overheating of the fuel during accident conditions, as its unavailability would only cause a slight increase in peak fuel temperature. However, it is necessary to prevent long-term overheating of the reactor vessel and possible damage to or failure of reactor cavity structural elements and reactor supports. Hence, the normal operation steady-state heat loss from the HTGR RPV is a larger fraction of the nuclear power than is the case for LWRs, and so the RCCS heat removal measurement is crucial in determining the actual reactor thermal power output.

In several instances, the performance of RCCS designs have been found to be difficult to predict with regard to total heat removal capabilities as well as to estimating local temperature distributions in the reactor cavity.[7,8] In its passive operational mode, the RCCS coolant flow and temperature distributions are expected to be quite varied and difficult to measure.

1.2.2 Measurement of neutronic variables

The heat balance calculation should start with measurement of neutronic power generated in the fuel, along with means of estimating gamma heating elsewhere. The measurement is typically used as a support variable in the feedback control of reactor power. A typical device used for this purpose in LWRs is the fission chamber. Challenges associated with the use of this technology include drift and degraded operation at above normal conditions. Gamma thermometers offer an alternative technology that can be adapted for power measurement purpose at HTGRs.

Total neutronic power is typically derived from measurements by all of the flux monitors along with a model for the core flux distribution, fuel composition, and geometry. Single local power measurements are potentially misleading because they do not capture the spatial variations in neutron flux. Spatial variance in the neutron flux is a larger concern for HTGRs since the very good moderation provided by graphite and the lower heat transfer of helium can result in local temperature spikes leading to fuel thermal challenges.

A typical protection signal is the ratio of reactor fission power to primary helium flow, which is a complementary signal to helium coolant temperature rise across the core. This ratio signal requires a very reliable measurement of helium mass flow. No commercial technology currently exists for direct measurement of this flow. Instead, it is typically calculated by measuring the coolant pressure and temperature and the speed of the circulator, along with the differential pressure across the circulator.

Another typical protection signal is obtained from the turbine/generator system. A trip of either the turbine or the generator—commonly called a loss-of-load event—may require a trip of the reactor, although for some designs the plan is to "ride it out" and control the power and flow to reduce to a house load value. This capability is feasible due to the inherent control mechanisms and slow thermal response of the core.

Depending on the reactor design, a trip of the turbine/generator system might disengage the secondary and primary heat transport systems by closure of the main steam isolation valve and remove heat via the SCS, which is generally proposed as a nonsafety system.

The generated nuclear power should match the amount of heat transferred to the primary heat transport system in a calorimetric calculation, which requires that primary helium temperatures at the reactor inlet and outlet be measured as accurately as possible along with helium flow rate. Issues related to accurate measurement of these variables were discussed in the next section.

1.2.3 Measurement of process variables

Mixed-mean helium coolant temperatures can approach 850ºC at the core outlet. The local helium temperature profiles in the outlet plenum have significant deviations from the average—due primarily to nonuniform heat-up in the core as well as flow stratification effects—that create technological challenges in accurate determination of the mean core outlet temperature. Furthermore, measurement or estimation of maximum local outlet plenum temperatures is necessary to determine potentially deleterious effects of high temperatures (including gradients and fluctuations) on critical components downstream, such as an intermediate heat exchanger, gas turbine, or steam generator, plus any connecting piping where the hot helium could exceed the hot duct temperature rating.

Recent studies have been made of nonuniformities of the coolant temperatures and flows in the lower plenum and those well into the hot leg where coolant temperatures at any measurement plane can vary significantly both spatially and temporally.

Computational fluid dynamics (CFD) analyses can reveal the expected severity of the temperature profile nonuniformity, as shown in Fig. 1, for the lower plenum of the Next Generation Nuclear Plant (NGNP).[9] Streamlines in Fig. 1 represent the family of curves that shows the direction that hypothetical fluid elements would take (exiting to the left), and the color-coding represents a temperature distribution which could span several hundred degrees Celsius.

The wide temperature variation of the primary helium as it exits the lower plenum and traverses the hot leg is shown as a CFD contour plot in Fig. 2. The variation in helium temperature shown as it first enters the hot leg (contour plot on the right) is approximately 70ºC—as read from the color scale (max–min). This variation diminishes, but does not vanish, as the gas flows through the pipe.

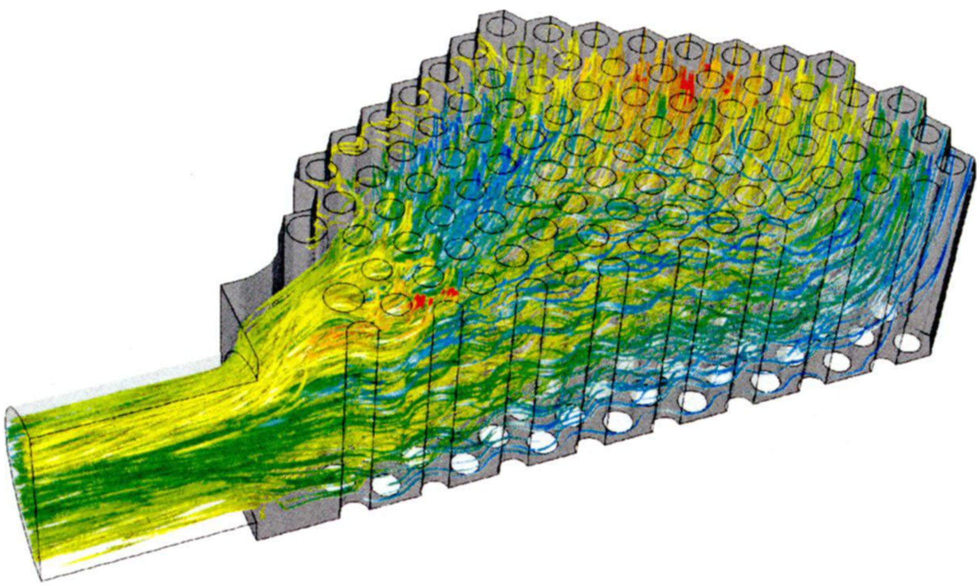

Fig. 1. CFD simulation of flow paths and associated temperature distributions for the lower plenum of an HTGR.[9]

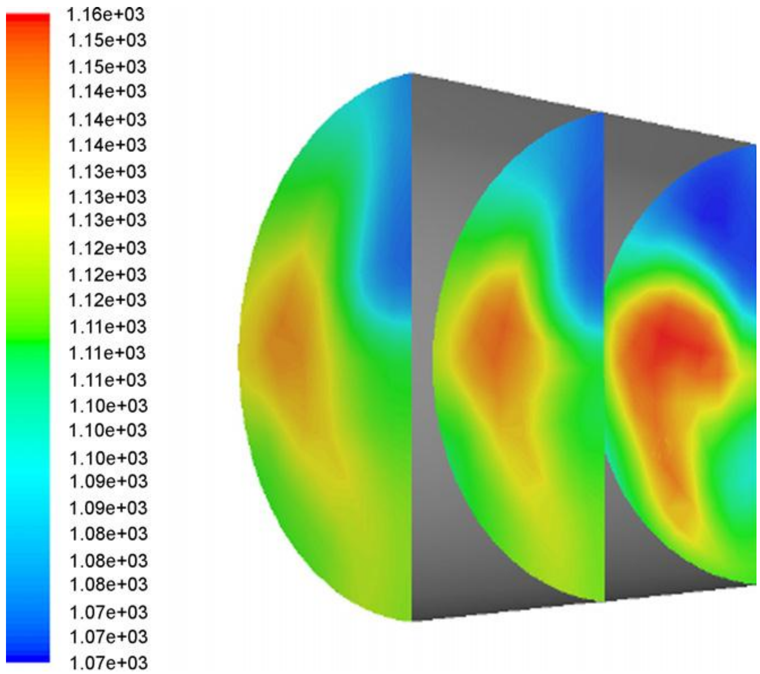

Fig. 2. Temperature contours in the hot leg. The three contour plots represent the entrance (right), the midpoint and exit (left) of the outlet pipe of 1.565-m length.[9]

These projected temperature distribution variations demonstrate the need for multiple temperature measurements in the hot duct and lower plenum. Mean temperature calculations must also account for variations in flow velocities. A discussion of the technology options for temperature measurement, including distributed temperature measurement, is provided in Section 3.1of this report.

In addition to spatial variations, flow may exhibit temporal instabilities, such as swirling and shifting with time, as illustrated in Fig. 3. This may require measurements at multiple locations and employing spatial and time averaging techniques to obtain representative values.

On the secondary side, similar needs arise for heat balance calculations to compare with those of the primary loop. For comprehensive calculations for steam generator designs, heat transfer in the feedwater heaters needs to be taken into account, which includes proper measurement of coolant inlet and outlet enthalpy, along with a measurement of feedwater flow rate.

Typically, turbine manufacturers specify a narrow band of steam temperature at the turbine inlet. This measurement has been used in HTGR designs in the past as a feedback control variable for reactor power. Therefore, measurement of the steam temperature at the outlet of the steam generator should be carried out by a series of redundant safety-grade temperature sensors.

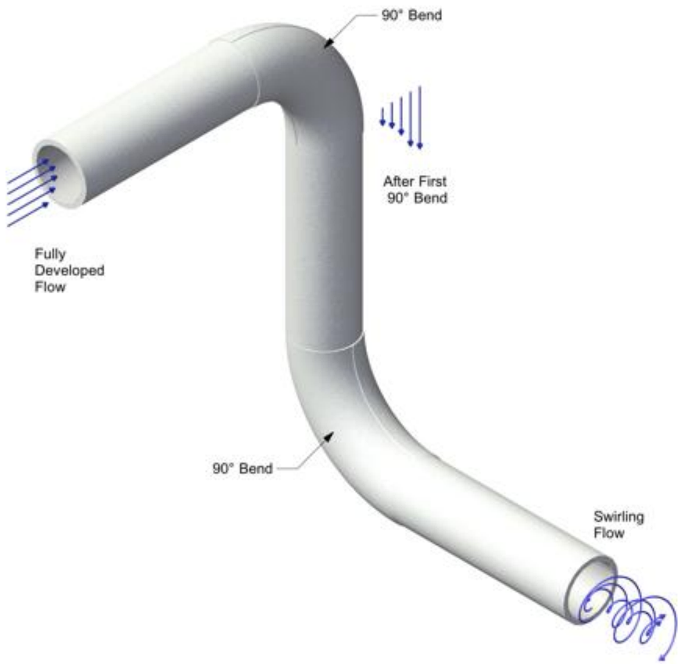

Fig. 3. Illustration of flow and temperature swirl development.

Another issue for temperature measurement in the primary coolant is the mechanical integrity of sensing elements. The primary helium flows at very high velocities—and, consequently, creates high mechanical forces on components in the flow path. The temperature sensors are mounted in thermowells which extend into the primary flow path. The primary flow produces both static and dynamic loads on the thermowell body, including vibrations due to vortex shedding. At operational flow velocities, the frequency of the vortex shedding, f_s, is related to the fluid velocity by the dimensionless Strouhal number, N_S:

$$f_S = \frac{\omega_S}{2\pi} = N_S \frac{V}{D},$$

where D is the tip diameter of the thermowell, V is the velocity of the fluid, and N_S is the Strouhal number (roughly ~0.2 for the high Reynolds number) of primary flow.[10] A thermowell design goal is to ensure that the vortex shedding frequency remains well below the natural frequency of the thermowell probe.

CFD analyses, supported by experimental verification and validation (V&V), are useful in gaining insight on how measurements can be most accurately correlated to determine a mean primary coolant temperature.

Instrumentation specifications for the secondary heat transport system would be different for a Brayton cycle power conversion system. However, these differences are mostly operations-related details; top-level requirements are obtained by fundamental thermodynamic principles. Therefore, reliable, accurate measurement of process variables at the inlet and outlet ports of a cycle is a requirement.

Additional instrumentation is generally used to check equipment health status and to provide redundant data for cross correlation purposes. For example, a sudden variation in a process variable—if credible—would be registered by more than one sensor. Some processes have different time constants; therefore,

certain amounts of delay can be observed between different sensors due to transport times or associated dynamics.

1.2.3.1 Helium leak detection

Helium is one of the most mobile gaseous species because of its very low atomic number. Helium may leak out during normal operations at very low rates, most likely through seals used in various mechanical junctions.

Helium has a much lower density than air. Hence, helium from small leaks from the primary system will collect in the dome of the confinement or compartments. Helium leak detectors installed at high elevations within the confinement volumes should be very effective in confirming small leaks. Sensitivities around 1 ppm should be sufficient for detecting minor leaks. Leak detection systems with 0.1-ppm sensitivity for helium are commercially available. Off-the-shelf systems can be adapted for use in a confinement environment by proper shielding—both thermal and radiation—of the connected electronic equipment.

Helium mass spectroscopy is a technology that can be easily adapted for this application with reasonable accuracy. The same system can also detect hydrogen and tritium presence in a confinement atmosphere.

1.3 Accident Monitoring

Accident and post-accident monitoring provides critical feedback on plant conditions to enable proper corrective and mitigating actions, ultimately to avoid or minimize release of radioactivity.

Accident monitoring focuses primarily on the availability and performance of heat removal systems. For the more severe accidents that involve radioactivity, mapping the movement and escape of contamination would also be a primary focus.

Depending on the accident type, event sequences, and equipment and system conditions, different systems—such as the main secondary heat transport system, SCS, or RCCS—could take the lead role in removing the decay heat from the reactor core.

For breaches of the primary heat transport system, a mapping of the transport of radionuclides within and outside the confinement is needed. Augmented with nuclide identification using spectroscopy, this capability would provide useful and timely information.

HTGRs have distinct design differences vs LWRs that impact the design requirements for accident and post-accident monitoring. Key differences—from the safety instrumentation standpoint—are the primary coolant and the confinement building instead of the containment.

The principal motivation, which also drives the technical basis for determining the critical variables necessary during an anticipated operational occurrence or a postulated accident, is not to dump all the data to operators but rather to provide them with a comprehensive picture of the state of the system to help make informed decisions, particularly under undue stress of accident conditions.

An additional safety task in an accident condition is to initiate a reactor trip upon detection of a credible anomaly in plant operation, mainly to prevent recriticality later in the accident progression sequence. As noted before, there is less emphasis on the scram, since for loss-of-cooling accidents the fission process is effectively terminated due to the strong negative temperature coefficients of reactivity.

The plant monitoring system should provide redundant indications for trip status, such as control rod drive mechanism positions, reserve shutdown system trip status, reactor power, reactor period, etc. In general, there would be a need for information about the accident progression, such as the following:

- coolant chemistry,
- chemical reactions between graphite and air/steam,
- formation of gases and aerosols containing fission products,
- estimates of fuel temperatures, and
- damage to support structures.

This report is intended to provide a high-level walk-through for instrumentation needs from an operator's perspective for each accident as the event progresses. New instrumentation is also proposed that could provide insight about the progression of an accident. For instance, *acoustic emission* monitoring has long been suggested as a feasible method to identify the location of breaches in pressure boundaries or to pinpoint troubling component or components within the containment or reactor cavity. Incorporating new technologies early into the design of the overall instrumentation and control (I&C) system could help with accident mitigation by providing means for timely intervention, especially considering the very long-time scales associated with HTGR accident sequences. In considering the instrumentation performance during accident sequences, it is important to consider the equipment environmental qualification and performance requirements for surviving the conditions that could develop.

The subject of post-accident monitoring is treated extensively in NUREG-0737 (Ref. 11). A detailed "HTGR" explanation of how these requirements were met (by Fort St. Vrain) is in a Public Service Co. of Colorado document available in the NRC Tomoye database.[12]

Experience from major reactor accidents indicates that collection and analysis of credible information is the most critical task in responding to the accident—and reliable instrumentation is the medium for that.

HTGR accident scenarios have been described and analyzed extensively. A summary description of these accidents and event sequences can be found in International Atomic Energy Agency (IAEA) SRS No. 54 (Ref. 13).

1.3.1 Anticipated operational occurrences

Anticipated operational occurrences are those deviations from normal operation that are expected to occur several times during the life of the plant. These may include, for example, loss of power to all circulators, tripping of the turbine or generator, isolation of the main condenser, and loss of off-site power. A long-term loss of power to all circulators is discussed in the P-LOFC section. Since the plant conditions are not significantly altered during AOOs, the instrumentation needs are essentially the same as those noted previously for normal operation.

1.3.1.1 Turbine trip and loss of load

Measurement needs for a steam-cycle design's balance of the plant side are similar to those for the PWRs, except that the temperatures are higher. Because a turbine trip requires dumping the steam into the condensers, reactor may also be tripped and the primary system isolated from the secondary loop. In some designs, reactor power would be sustained but decreased to cover house load. Measurements of interest on the secondary side are the process variables for the steam into the condenser and the status of safety pressure relief valves.

1.3.1.2 AOO reactivity events

"Less severe" AOO reactivity events are those in which a manageable amount of excess reactivity—positive or negative—is accidentally or inadvertently introduced. Typical initiators causing these events are as follows:

1. control rod or rod bank withdrawal;
2. inadvertent control rod movement, including a control rod drop;
3. accidental reactor shutdown;
4. unexpected increase or decrease in primary heat removal rate;
5. compaction of a pebble bed core caused by an earthquake;
6. leak in heat transfer tube(s) in a steam generator or other water-cooled heat exchanger that could result in a minor water or steam ingress into the core; and
7. fuel loading error.

1.3.2 Heat removal accidents—pressurized loss-of-forced circulation (P-LOFC)

This general category of accidents has one major event sequence that should be taken into account for design basis: long-term pressurized loss-of-forced circulation in which the helium circulators stop while the primary system remains pressurized.

A long-term loss of primary coolant flow would be rapidly detected by several of the plant process instrumentation systems. For the case of plant designs with the steam generator in the primary loop, the instrumentation system should provide the operators answers to the following (example) questions [note that this example list would be different for designs with an intermediate heat exchanger (IHX) or a direct-cycle gas turbine]:

1. What caused the failure?
2. Did the plant trip properly upon accident initiation?
3. Was any SCS operation initiated, and what are the prospects for startup?
4. What portion of the primary cooling system is not functioning properly?
5. What is the response of the primary pressure—is it approaching the relief valve limit?
6. Did the steam generator isolation valves (secondary side) close?
7. Has any noticeable plant damage occurred (e.g., causing a loss of primary heat sink, or is there moisture ingress from the steam generator or SCS)?

As the accident progresses, additional information should be provided:

1. estimated core temperatures;
2. estimated reactivity balance (approach to recriticality?);
3. reactor vessel and vessel support temperatures, particularly in the upper regions;
4. RCCS status and heat removal rate;
5. reactor cavity concrete temperatures;
6. primary system pressure;
7. confinement atmosphere temperature and gas composition;
8. electrical power bus status; and
9. evidence of water ingress into the primary system.

Note that for the P-LOFC, with no primary coolant flow, core coolant outlet temperature sensors (in the lower plenum) would not provide any direct information about core temperatures.

There are two major groups of events or event sequences that lead to a P-LOFC: (1) a reactor shutdown with a failure of the SCS (or other shutdown cooling mechanism) to provide forced cooling or (2) a station blackout that results from a long-term loss of off-site power and auxiliary power sources. The first group includes events that result in actuation of the reactor protection system, which typically stops or reduces the primary helium circulation intentionally to avoid overcooling. The second group of events

includes loss of grid load or failures of the primary helium circulating equipment (i.e., helium circulators or turbomachines).

The primary concern with loss of circulation is the core and primary system heat-up transient. The advantage of being at pressure is that the primary helium inventory will enter a natural circulation mode within the core as the core heats up, which will help equalize the core temperatures and with maximum temperatures appearing near the top of the core.

Many of the measurement needs for P-LOFCs are similar to those for normal operations. Because there is an additional mode of effective heat transfer compared to a D-LOFC—namely, the primary helium in natural circulation mode—fuel temperatures will remain well below safety design limits.

Elevated temperatures are of particular concern for metallic structures and equipment. Those include primary system pressure boundary, the reactor vessel, control rod sleeves, and the core barrel. Reliable measurement of the temperature of these structures and components is important, especially for post-accident *time-at-temperature* analyses. These calculations are critical in assessing the potential impact of such temperature increases on structural integrity and code limits.

Depending on the heat removal conditions seen by the entire primary system and actions taken (or not) by the helium inventory control system, a long-term primary pressure increase may occur.

For prolonged LOFC accidents, reactor temperatures eventually rise to levels where RCCS heat removal is crucial. The RCCS is generally proposed as a passive, safety-grade system. Heat rejection by radiation from the outer surface of the reactor vessel is the dominant heat transfer mode, but natural circulation (of air) in the reactor cavity is also important and tends to concentrate heat in the upper regions.

In this heat transfer process, a crucial factor is the thermal emissivity of the reactor vessel and the RCCS opposing surfaces. On-line measurements of emissivity are feasible: (1) emissivity measurement on the outer surface of the reactor vessel, which is indicative of effectiveness of radiating heat off the surface and (2) temperature measurements on the outer surface of the vessel and on the surface of the RCCS used in heat balance calculations for the RCCS. In addition to these, airflow rate in the stacks that connected to the RCCS provides an indication of how effectively the heat is being rejected to the atmosphere—the ultimate heat sink for the RCCS.

For emissivity measurements, radiation pyrometry—most likely an IR pyrometer—can be used. For reliable pyrometry measurements, care must be given to the RCCS atmosphere. Presence of dust or gases absorbs a portion of optical energy, which might result in misreading of surface emission. To overcome this challenge, ratio pyrometers can be used, taking measurements within narrow bands at multiple wavelengths. By taking the ratio of emissions at two temperatures, the effect of variation of surface emissivity with temperature cancels out. This technique also compensates for any intermediary material—such as dust and other gases -- that might be present in the optical path.

An alternative to IR pyrometry is phosphorescent thermography. The technique exploits the temperature dependence of the phospho-fluorescence phenomenon, which provides an emissivity-independent alternative to pyrometry. A special class of materials—known as *thermographic phosphors*—have been widely studied with numerous industrial applications.[14] Applications include measurement of vane and blade temperatures in an operating turbine engine and temperature control of integrated circuit (IC) wafers during plasma etch and chemical vapor deposition processes. Some of these measurement systems are commercially available.

These measurements are also useful during normal operation. For this purpose, another band-filter configuration can be added, which focuses around a wider wavelength that corresponds to the nominal temperature of the vessel surface during normal operation. This way, heat rejected by the RCCS can always be incorporated into the heat balance for a wide range of operating conditions.

1.3.3 Primary system rupture accidents

1.3.3.1 Depressurized loss-of-forced circulation (D-LOFC)

Many of the monitoring needs for a D-LOFC accident would be similar to those for the pressurized LOFC case. For the D-LOFC, there would be an additional concern for air in-leakage to the primary system that could result in long-term oxidation of the core graphite (see Section 1.3.4).

The D-LOFC is a design basis accident with failure of the primary system pressure boundary. Therefore, instrumentation functional requirements are drawn from the needs for monitoring the accident progression as well as the regulatory requirements to quantify the risk to the health and safety of the public.

In the D-LOFC accident, it is assumed in the typical reference case that, in addition to an LOFC, the primary helium inventory is lost to the point that the primary system is depressurized to atmospheric pressure. D-LOFC conditions may arise from primary pressure-containing equipment failures, such as leaks or piping ruptures that are not isolatable. The opening of primary system safety pressure relief valves (without re-closure) may also lead to D-LOFC conditions.

Variations of the D-LOFC include scenarios in which the leak is isolated, and the resulting pressure is greater than atmospheric, thus setting up some natural circulation flows within the core. [With only atmospheric pressure helium, natural circulation within the core has no noticeable effects on core temperatures.] Another variable of much interest is the size of the leak, which determines the rate of depressurization. Rapid depressurization event consequences may include an unfiltered discharge of the primary system inventory into the atmosphere as well as damage to equipment outside the primary system from a hot helium jet or from piping whiplash. Impact of the large momentum transfer and the resulting forces on safety-related equipment will need to be accounted for in the structural evaluation of the postulated accident sequence. As helium leaks into the reactor cavity and the confinement, temperatures in these volumes will initially rise significantly with certain extremely hot spots before it completely mixes.

The major consequence of long-term D-LOFC accidents is the core heat-up and potential radioactivity release (prompt and delayed) into the confinement and, eventually, into the environment. During the course of the accident, the maximum fuel temperature reaches its peak value within a few days, followed by a prolonged decline. Maximum fuel temperatures are generally expected near the midpoint of the core. The major design features of modular HTGRs (maximum power, core diameter, etc.) are limited to ensure that the peak fuel temperature does not exceed the design value (typically 1600ºC).

During the initial depressurization process, much of the circulating activity may be released along with the leaking helium. Depending on the break size and location, graphite dust containing fission products from failed fuel as well as any ^{137}Cs and other isotopes plated out onto colder portions of the loop may also be released. The large prompt helium releases are followed by slower releases into the confinement. In current designs, these discharges would pass through charcoal filters before being released to the environment.

Maximum attainable temperatures in the cavities and the confinement should be estimated using the thermodynamic equilibrium models. Based on the calculations with reasonable engineering margins, requirements for equipment qualification should be written for certain safety-related instrumentation to withstand the anticipated conditions—i.e., pressure, significant forces, and temperature. Maximum and final process parameters will be a function of the primary helium inventory, core mass and temperatures, and the ratio of total primary system volume to total cavity and confinement volume.

Cabling requirements for safety-related instrumentation would be determined based on predicted consequences of postulated events—as per requirements by IEEE Std 323-1974. Class 1E equipment cabling practices should be adopted for any component or system that is identified as the potential safety-related instrumentation during a postulated accident.

1.3.3.2 Nuclear measurements

The most critical parameter to be monitored during a D-LOFC is the radioactivity levels in the reactor cavity, confinement, and the site periphery. Measurement at multiple locations may be useful in identifying the location of the leak. Activity trends might also be a confirmatory indication of approximate size of the breach.

Qualified air monitors used in the operating reactors monitor the activity in the containment atmosphere. However, qualification requirements for HTGR confinement atmosphere will be different from those for LWR containment. Some equipment may need to be designed with larger thermal margins to withstand localized temperature transients following a major depressurization event.

Typical air monitors do not provide spectroscopic information that can be used for isotopic analysis, which would be useful in monitoring accident sequences involving fuel failures.

1.3.3.3 Process measurements

During a D-LOFC, availability of in-core temperature information is of particular importance, primarily because much of the safety-related parameters are linked directly to temperature—such as TRISO fuel integrity. During the course of the event, core temperatures rise, though slowly, well beyond the levels expected during normal operations. Some of the instrumentation used during normal conditions might fail because of the core heat-up. Although direct in-core temperature measurements are likely not available, it is advisable to have some related temperature instrumentation (such as in the reflectors) available to withstand the extreme conditions to provide means of estimating in-core parameters during the transient.

In addition to core temperature instrumentation, means should be provided to monitor the temperature in the cavity—both reactor and the PCS. Because of the excessive forces and subsequent heat released during a depressurization accident, concerns on the availability of the instrumentation apply, as mentioned earlier. The equipment intended for accident and post-accident monitoring in the reactor cavity, confinement, and the site vicinity should be designed to withstand the conditions that might arise within reasonable assumptions during a design basis accident. Instrumentation for safety-related functions is expected to be qualified to stricter requirements to be determined by accident analyses.

Pressure is the key parameter in a D-LOFC for confirming depressurization and potential breach of the pressure boundary. Pressure is most likely measured at multiple locations within the primary system as well as within the reactor cavity and the confinement. Following a breach of the primary pressure boundary, pressure transducers within the cavity and the confinement may also register shock waves. Depending on the break characteristics, sensor locations, and instrumentation design, it may be possible to obtain insight as to the approximate location of the leak.

1.3.3.4 Confinement air filter measurements

During a rapid depressurization, the confinement building will experience a sudden increase in pressure. Depending on the design, excess pressure will most likely open a flapper valve to the environment. Blowout filters (Fig. 4) could also be instrumented with a radiation monitoring system—with or without spectroscopic capabilities. Spectroscopic capabilities would provide additional means for tracking movement of escape of contamination. Rate counters with identical energy windows at the inlet and outlet of the filter can be used to determine its effectiveness.

1.3.3.5 Acoustic Emission Monitoring

A major depressurization process will manifest itself as sudden departure from nominal values of the process variables. Monitoring these variables provides a direct and reliable indication as to the existence of an anomaly. However, determining the location of the leak may not be possible with the instrumentation specifically intended for monitoring the plant variables.

One potentially viable method for (1) confirming a breach and (2) locating the leak is through acoustic transducers. Discharge of high-temperature helium into the cavity with a large pressure differential creates very high-frequency acoustic waves. Furthermore, at the onset of a breach, the initial release should create a strong pressure wave—a shock wave—that should be easily registered by pressure transducers as well as accelerometers mounted on the components. The approximate location of the pressure source might be determined by cross correlation of the readings from multiple sensors and triangulation.

Even if the leak were small, it should create very distinct, high-frequency acoustic waves easily detected by the same method. Noise amplitude analysis would most likely not lend itself to determining the approximate size of the leak.

Leak position-sensing could be accomplished in a number of ways—transducers mounted on the primary piping should help create a baseline for nominal operation. On-pipe transducers are also useful to monitor flow-induced noise, which can provide data for detecting potential loose parts within the heat transport loop. Proximity of the sensors to the leak should impact the amplitude of the signals. Hence, amplitude comparison amongst sensors made available for this purpose should provide an indication as to the approximate location of a leak.

Another method for position sensing is to employ a time-of-flight approach, where the time shift of the waveform is measured by using one transducer as the reference signal source. Other transducers are used with respect to that reference signal to determine the amount of shift in a signature waveform. The approximate location of the wave source—potentially a leak or a failed component with a distinguishable noise characteristic compared to the baseline spectrum—can be calculated using the information from multiple transducers by triangulation.

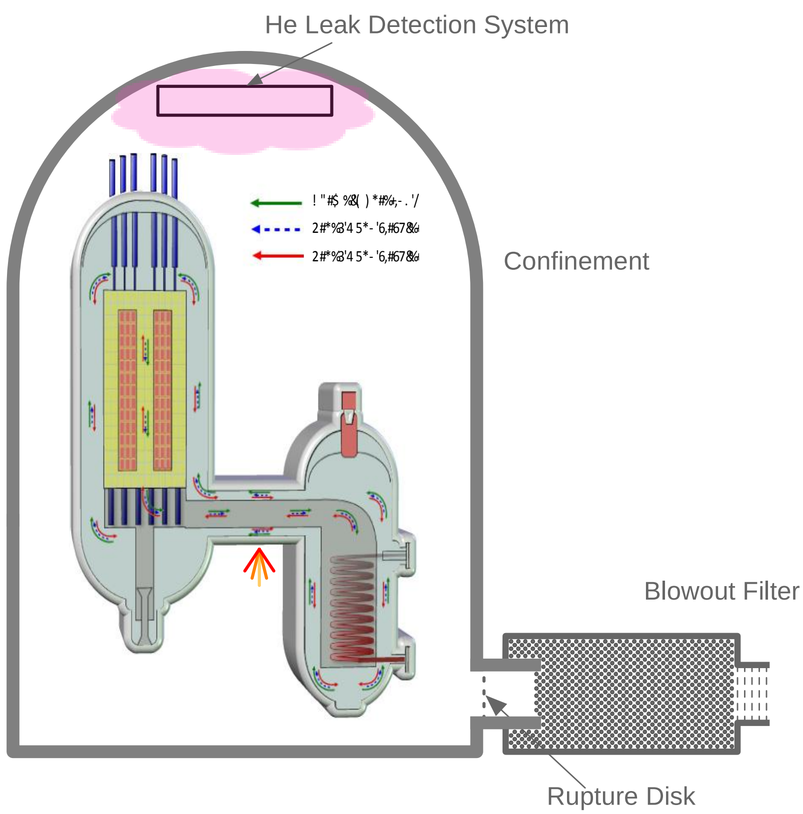

Fig. 4. An illustration of the confinement blowout filter and a helium leak detection system.

1.3.4 Air Ingress

Key factors in air ingress accident scenarios are the net airflow rate into the reactor vessel and core and, ultimately, the availability of fresh air (oxygen) over the course of the accident. The net airflow through the core is strongly dependent on the buoyancy forces due to differential temperatures and the flow resistances in the core and at the break(s). Predictions of the availability of “fresh air” ingested into the vessel and core need to account for the reactor pressure vessel being in a cavity or confinement building (vault), where fresh air in-leakage is limited, and the discharged helium and gaseous products of an accident would collect and mix and become components of the gas for subsequent core ingress.

For a single break or opening in the primary system, calculations and experiments have shown that there may be long delays before a sustained, significant net air inflow into the core is established. The main responsible phenomenon is the persistence of a helium “bubble” in the top area of the vessel, which inhibits recirculation flows within the core. Experiments and analyses by JAERI[15] and others have shown that this bubble may be effectively maintained in the vessel for days. For idealized experimental conditions, these predictions have proven quite accurate. However, for more realistic break situations, actual conditions would be complex and could result in large uncertainties in the time for a significant ingress flow to start. The major uncertainty factors include size and location of the break(s), the core temperature distribution, the temperature and composition of the gas outside the primary system that provide the cold leg driving force for the natural circulation, and other types of perturbations (such as operator actions) to the system likely to occur in the days following a major event such as a D-LOFC. This process involves air diffusion into the helium bubble and, potentially, a density gradient driven stratified flow effect as well for larger break scenarios.[16] Because of the uncertainties, ingress flow start times in these analyses are usually inputs treated parametrically.

For a much less likely case of a double break that allowed outside access to both the top and bottom of the core, this chimney-like configuration would promote a higher net air ingress flow with minimal delay.

Once a net air ingress flow is established, oxidation would typically begin in the lower part of the core or in the bottom support and reflector areas. However, for the low flow rates expected, oxygen could be depleted even before reaching the active core area. In some scenarios, however, oxidation would occur in the lower part of the active core for higher ingress flow rates, and for cases where the lower reflector has cooled sufficiently and no longer oxidizes. Structural damage from low-temperature (chemical regime) oxidation may be considerable in the lower plenum—depending on the delay in the start of air ingress. Hence, this start time is a major determinant of the potential accident consequences.

Variation in the time delay for a net air ingress flow typically has little effect on peak fuel temperatures, which occur higher up the core. Even with no mitigation, the oxidation rate would eventually decrease due to limitations in available oxygen.

While CO_2 is the principal oxidation product with its oxidative heat deposited in the graphite, carbon monoxide can be formed in higher-temperature reactions of the CO_2 with graphite ($CO_2 + C = 2CO$, the Boudouard reaction). Since this is an endothermic reaction, ignoring it is “conservative” when predicting peak core temperatures, but its omission may lead to an underprediction of the amount of graphite consumed and, thus, a greater potential for fuel damage in the hotter part of the core.

Studies have shown that core survival could be in jeopardy if there is very long-term easy access to fresh air after a D-LOFC.[17] Thus, prediction of damage to hot core and support structures that encounter the oxygen requires refinement of design data and further analysis, as well as post-accident monitors designed to determine the extent and nature of damage to the core and support structures. It is important that for long-term air ingress accidents, the ultimate availability of fresh air is considered and limited.

1.3.4.1 Analytical measurements

Much of the basic instrumentation requirements discussed under the D-LOFC also apply to air ingress scenarios. Because depressurization is prerequisite to air ingress, measurement needs are also similar. In addition, analytical methods should provide critical information about the extent and nature of the graphite oxidation. Reaction products (e.g., carbon dioxide and carbon monoxide), as well as potential fission gas releases to the reactor building (confinement) would need to be monitored to help guide accident mitigation efforts. Measurement of airborne concentration of these species should provide insight into the progression of these events. Technology is available for making very precise and accurate measurements at elevated temperatures.

The most likely technology that can be applied here is absorption spectrometry—either optical or infrared (IR). Molecules absorb a narrow band of electromagnetic radiation whose frequency matches the resonant frequency of a vibrating molecular bond. Absorbed frequency bands are characteristic of molecular bonds and provide information about the gaseous species in a measurement volume. The most commonly used analytical equipment for this spectroscopic technique is the Fourier-Transform Infrared (FTIR) spectroscopy.

Other technologies for gaseous species include sensors that exploit the change in electrical conductance of a metal oxide. However, this particular technology has certain drawbacks for reducing gases at high temperature (400–800ºC) including high cross-sensitivity, drift of the sensor response, and poor recovery.

1.3.5 Steam/water ingress

1.3.5.1 Background

The NRC preapplication review of the modular HTGR (MHTGR) in the 1980s as documented in NUREG-1338 (Ref. 18) and the extensive supporting documentation provided by DOE in the *Preliminary Safety Information Document* (*PSID) for the Standard MHTGR*[19] provide thorough documentation of the multiyear regulatory review of a 350-MW(t) plant similar to that currently considered for the NGNP. It is applicable to NGNP in that a dominant risk is from steam/water ingress via steam generator (SG) tube leaks or breaks.

For moisture ingress events, the major risk evaluations would, as with other accidents, focus primarily on dose at the site boundary, radioactive release from the reactor building, and worker dose. Secondary risk evaluations for moisture ingress events would be for loss of structural integrity of the graphite or composite reactor internals, release of primary system contaminants to the reactor building, explosive gas concentrations, and fission product mobilization. A comprehensive safety evaluation of moisture ingress was completed recently and forms the basis for much of this section.[20]

If one adopts the assumption that the only feasible means for fission product escape to the environs in a moisture ingress accident is via operation of the primary system pressure relief valves (PRVs), then the leading critical parameter in the accident sequences is primary system pressure. Simultaneous depressurization events (due to breaks in the primary system)—considered to be independent of water ingress—are usually judged to be beyond the usual beyond-design basis accident (BDBA) probability.

Primary system pressure is affected by the amount of steam (or water) ingress, reaction gas generation, system temperatures (e.g., core heatup), and the timing of various elements of the sequence. For cases where relief valves open on high pressure, cycling of the valves would impact the amount of gas release (and dose) involved.

Typically the cores of modular HTGRs are under-moderated, so that the additional moderation introduced by moisture ingress increases the reactivity and can, thus, cause rapid power increases. In addition, a reduced neutron leakage from the core reduces the worth of control rods as well as reserve shutdown

system effectiveness. Even so, with a very hot core, any moisture ingress would be in the form of low-density steam, which should have only small effects on reactivity.

There is also the need to consider "chronic" degradation of graphite due to low-level concentrations of moisture in the primary system during normal operations. Such long-term degradation may affect the integrity of structural supports and, thus, exacerbate consequences of structural impacts during severe accidents.

1.3.5.2 Ingress event descriptions

Steam-water ingress events are assumed to be primarily caused by steam generator tube leaks or breaks, but also possible would be leaks in other water-cooled heat exchangers in the primary system such as in the SCS. Figure 5 shows the possible sources of water ingress, the dominant ones being from the steam generator since it operates at a pressure higher than the primary helium coolant. The figure implies the dependency of SG event sequences and accident outcomes on the SG leak location, since water (from the feedwater end) or steam (or both) may be injected.

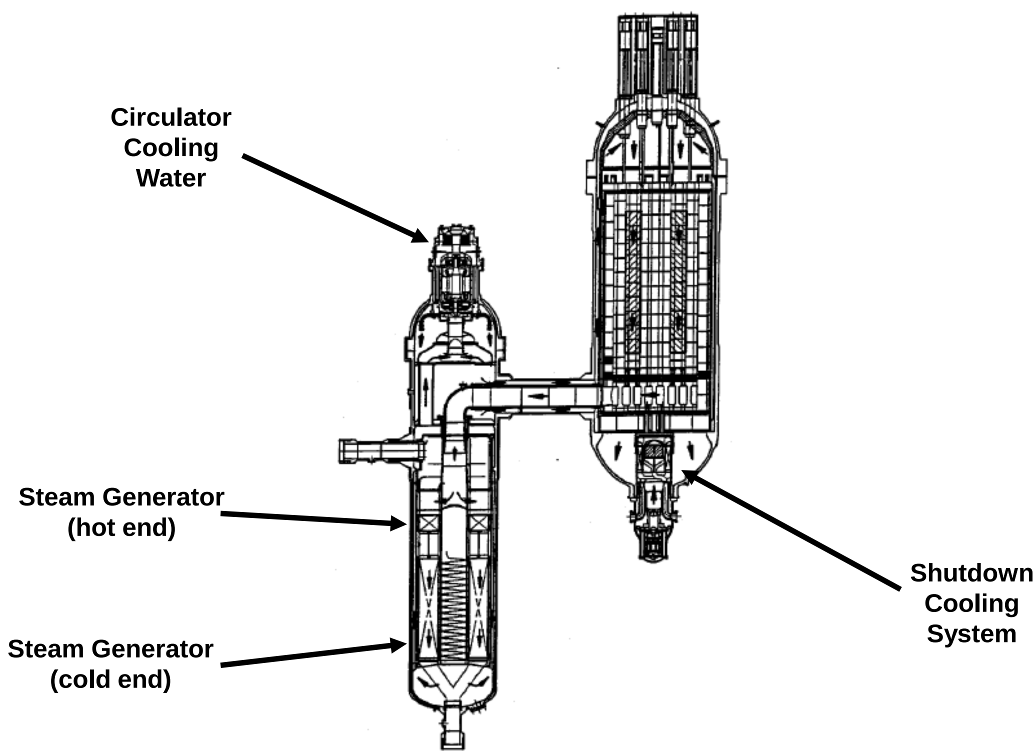

Fig. 5. MHTGR arrangement and water ingress sources. (From Ref. 20)

The steam generator is also equipped with secondary side isolation and dump valves. A typical SG isolation and dump system configuration is illustrated in Fig. 6. Reactor trips are accompanied by closing the feedwater and main steam isolation valves and opening the dump valves to drain the remaining water inventory from the steam generator. The dump valves are reclosed when the secondary pressure has dropped to less than the primary coolant pressure. Draining the excess water from the steam generator prevents additional water from entering the primary via vaporization of remaining water in the tube bundle.

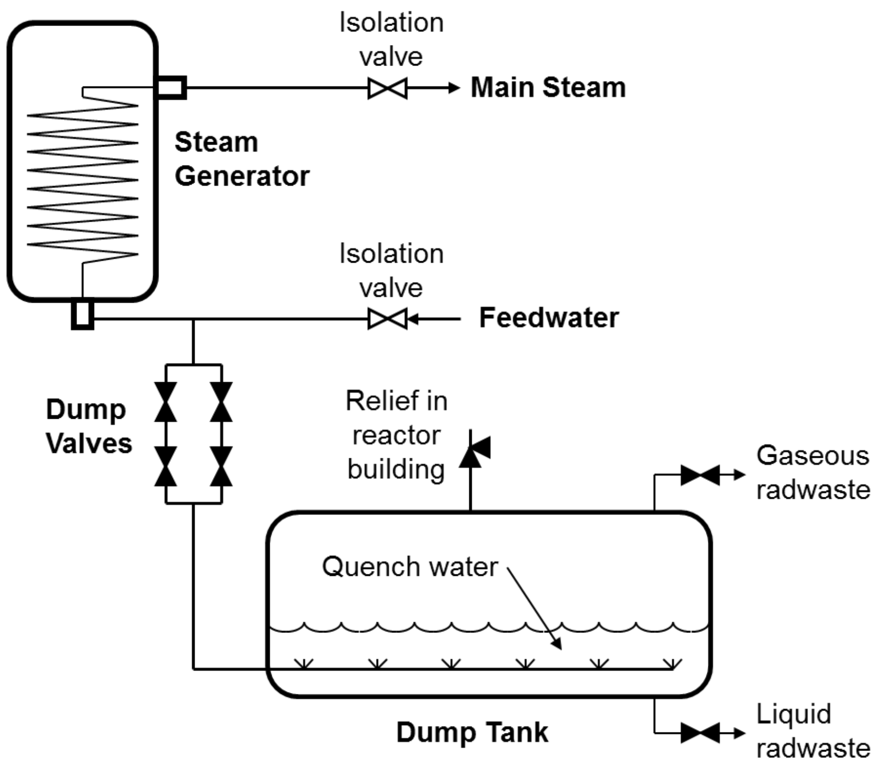

Fig. 6. SG isolation and steam water dump system configuration. (From Ref. 20)

In ingress scenarios where it is assumed that nonsafety-grade moisture monitors (per MHTGR design) operate and successfully activate the SG isolation valves, thus cutting off the water ingress source, the results are a modest amount of total water ingress with very little effect on reactivity, core temperature changes, water gas generation, and graphite oxidation. On the other hand, scenarios that assume failure of a moisture monitor trip action ultimately rely on a high primary system pressure to scram (and potentially for subsequent safety valve actuation), which also activates the SG isolation. In this case the safety valve actuation would maintain the system pressure within design limits but at the cost of releasing fission products (FPs) to the environment. A continuing pressure rise causing repeated openings of the relief valve(s) would continue to maintain a "safe" pressure but would continue to release FPs. In a variation of this scenario in which the high-pressure trip point is not reached, there would be reliance on the operator to manually scram the reactor and actuate the SG isolation and dump valves.

Events involving moisture ingress during shutdown and refueling modes of operation could have more of an impact on reactivity than those during normal operation. In liquid form there could be more moderation occurring than with the high-temperature steam, where the density would be much lower. Moisture monitors also may not be able to detect the ingress and conditions associated with the presence of moisture in the liquid form (e.g., saturation levels do not indicate the status of water concentrations).

In the events and situations noted, there are cases where changes in operating conditions (or model assumptions) could have significant effects on predictions of total water ingress and total FP releases. For long-term accident sequences especially, the possibilities for operator actions need to be factored in (where "intuitive" or prescribed interventions may or may not be beneficial). For such long-term, potentially complex accident scenarios, it is important for the instrumentation to provide both the automated systems and the operators with reliable and accurate information.

The essential information should be collected and analyzed to enable the I&C system and the operators to estimate and track the following:

a. moisture levels in the primary;
b. core and primary system coolant temperatures, flow, and pressure;
c. reactivity effects of steam/water ingress (increased reactivity with under-moderated core);
d. reduction of control/shutdown rod worth;
e. SCRAM (and reserve shutdown system) action;
f. reactor pressure vessel (RPV) pressure relief system activations and response;
g. plant protection and control systems (safety and nonsafety) response;
h. confinement atmosphere conditions (temperature, pressure, composition);
i. confinement release characteristics, including dose;
j. steam generator isolation and dump system operation;
k. graphite oxidation/corrosion products;
l. fission product releases from fuel and graphite; and
m. potential for explosive gas mixtures—in both the RPV and the reactor building.

1.3.5.3 Applicable instrumentation

A steam/water ingress accident requires reliable moisture monitoring in the primary system. Although a major in-leakage could also be detected via a sharp increase in the primary loop pressure, there are other causes of pressure increases so it would not necessarily indicate moisture ingress. Depending on the scale and location of the leak, it may also be detectable by acoustic sensors. Moisture ingress can also cause an increase in the core reactivity due to the added moderation from the hydrogen, although small leaks are not anticipated to produce measureable reactivity changes. No observable reactivity changes were recorded during the Fort St. Vrain reactor operation in spite of the numerous ingress events. Calculations indicate, however, that gross in-leakages could cause reactivity increases that would be readily detectable by power range fission chambers.[21] Additionally, a side effect of water entering the core could be a release of tritium that has been chemisorbed into the core graphite.

Moisture monitoring will likely be performed either with a mechanical resonance or capacitive-type gauge connected to a gas sampling tube (to lower the gas temperature and pressure) or preferably by an on-line, infrared absorbance spectrometer. Descriptions of the sensor options are included in the Section 3 of this report.

In general, the hot graphite of the core is expected to react with all of the trace oxygen, moisture, and carbon dioxide in the coolant to form carbon monoxide and hydrogen. The hydrogen, in turn, could react with cooler graphite to form CH_4. The moisture detection system needs to be sensitive to ~1 ppm of moisture, and periodic (off-line) analysis of primary coolant contamination helps to keep track of the reactions occurring during normal operation.

The recent HTGR moisture ingress evaluation notes that ppm levels of moisture could be a chronic problem at HTGRs, but indicates that further R&D is advised to estimate the effects on core materials and lifetimes.[20] However, with a properly functioning helium cleanup system, even the large amount of moisture that can potentially be present in new fuel or reflector block loadings can be removed relatively rapidly (hours).

Water/steam leaking into a helium environment will have an acoustic signature as the steam flow induces vibration in the leaking crack. The intensity and frequency of the acoustic signature will vary if the crack

enlarges and the steam flow increases over time. The leak's acoustic signal can be detected by attaching a high-temperature strain gauge to the surface of the primary piping near the steam generator and reading out the strain gauge at a few tens of kilohertz.

A rise in the loop tritium level resulting from flushing the chemisorbed tritium from the core graphite would be observable in the helium cleanup system. In the most likely form of the helium cleanup system, a side stream of the primary coolant would be oxidized by flowing it over a copper oxide bed to form tritiated water. The tritiated water would then be adsorbed onto a microporous synthetic zeolite filter—referred to as a molecular sieve dryer. The filter would then be removed from the flow path and heated to drive off the water. The water would then be condensed in a cold trap and analyzed for tritium content most likely using a scintillation cocktail. The tritium collection process inherently takes time; tritium would not be detected until the filter is exchanged and analyzed, likely to be some hours after the water ingress begins. However, tritium detection may be a useful indicator of a small to moderate water ingresses.

1.3.5.4 Summary

Typically in safety studies, the dynamic effects of water ingress into operating and shutdown core are considered of high importance. The 1986 MHTGR PSID included several design basis accident (DBA) and BDBA versions of steam leaks and tube breaks, with and without safety system intervention, including SCRAM, turbine trip, and isolation and dump valve closures. The DBA scenarios resulted in modest power increases from reactivity increases, maximum fuel temperatures well within material limits, and with no fuel failures expected to occur. The consequences of BDBA events (with only safety systems responding) were more significant, with more water entering the primary system leading to more graphite oxidation.

The primary pressure relief valve cycles resulted in releases of radionuclides. However, the PSID analysis concluded that the oxidation and the total dose to the environs would be well within limits. There are variations in the BDBA sequence (from the PSID) that could lead to more oxidation and dose; consequently, sensitivity studies, with variations in both sequence assumptions and models used, are recommended. This will require a systems accident code capable of simulating phenomena associated with moisture ingress used to acquire a better understanding of the potential consequences of moisture ingress events and to optimize the design of mitigation systems in the process.

Fission product releases would result mainly from removal of FP deposits from primary system surfaces and from chemical reactions with the FPs and graphite. Moisture ingress would have no significant effect on in-tact fuel particles, only on defective particles in which the kernels are exposed. Releases to the environment would occur only upon relief valve opening(s). There is a need for more data and improved modeling for decontamination factors for the reactor building.

For the case of long-term, low levels of moisture present during normal operating conditions are a significant concern, as oxidation rates and the physics for diffusion-controlled oxidation are not well understood. Additional research and development would be required to accurately calculate the oxidation rates, the effects on material performance, and the mechanisms controlling the oxidation behavior at high temperatures and low-moisture environments. Long-term structural damage would be a consideration as it may affect "initial conditions" in the evaluation of significant moisture ingress accidents.

1.3.6 Reactivity events and anticipated transients without scram (ATWS)

Anticipated transients without scram (ATWS) events are usually those that are AOOs followed by the failure of the reactor trip portion of the protection system specified in General Design Criteria 20 of Appendix A to CFR Part 50.

A failure to scram involves initiating events (which call for a scram). Typical ATWS initiating events considered for HTGRs are:

1. loss-of-offsite power,
2. loss of heat sink,
3. loss of forced cooling under pressurized or depressurized conditions,
4. withdrawal of one group of control rods,
5. compaction of a pebble bed core caused by a safe shutdown earthquake, and
6. ingress of water or steam into the core (causing a positive reactivity insertion).

High-level requirements for HTGR reactivity control will vary depending on design features such as the balance of the plant characteristics of the power plant (i.e., steam cycle vs gas-turbine-based power conversion system). A discussion of the safety aspects and consequences of reactivity events is found in IAEA SRS No.54 (Ref. 13).

HTGRs are typically furnished with two active reactivity shutdown systems—diverse from the reactivity control system, which functions as a power regulation system. The first shutdown system actuates automatically inserting sufficient negative reactivity into the reactor—typically by inserting control rods into the core or into the reflectors depending on the design. The second shutdown system has a diverse actuation mechanism, which dumps neutron-absorbing balls into the core. The second shutdown system can be actuated either manually or automatically in the event that the first shutdown system fails. Both systems should have sufficiently black reactivity to render the reactor subcritical.

In an ATWS event sequence, when a reactor protection parameter reaches its trip point, a scram signal is initiated and sent to the reactor protection system, which actuates a series of protective actions, such as drop of the control and safety rods, stopping the primary helium circulators, closure of a blower flap, isolation of the secondary system (by closure of the main steam isolation valve in a steam-cycle design).

Furthermore, the excellent reactivity-temperature feedback coupled with very high-thermal inertia of core components—primarily large graphite mass—provide inherent protection against the deleterious of a potential ATWS event and allow for long delay times in case there is need for manual operator actions, such as insertion of the absorbing balls into the side reflector.

High-level instrumentation requirements for ATWS mitigation—similar to PWRs and BWRs—should include a diverse signal route from sensor output to final actuation device, which will de-energize the coils on the reactor shutdown rods. This is a safety-grade system. A diagnostic system that predicts reactivity based on power history, temperatures, control/safety rod positions, and calculated xenon poisoning levels would be useful to monitor any approaches to recriticality.

Note that control rod ejection events may also be associated with a primary system rupture—due to control rod housing failures. However, rod ejection events are omitted from consideration in modular HTGRs by design.

2. MODERN HTGR INSTRUMENTATION ARCHITECTURE

HTGRs have low power density, ceramic, high-temperature cores, a large thermal margin to fuel damage, lack the potential for primary coolant phase change accidents, have strong negative reactivity thermal feedback, don't require auxiliary power in case of long-term station blackout accidents, and have inherently slow accident progression scenarios. As such, HTGRs will require significantly different safety measurement capabilities than would an LWR. The differences will inevitably be reflected in the instrumentation system architecture. While general design principals such as high quality, diversity, and defense-in-depth remain requirements for any reactor, the much larger time and temperature margins inherently afforded by an HTGR to respond to any accident may, in addition to altering the instrumentation, affect the allocation of responsibility for safety response from equipment to operator or even remote expert.

Unlike LWRs, HTGRs have no credible accident scenarios that require a rapid response. All known loss of core coolability accidents evolve slowly. HTGRs have no known mechanisms to rapidly overpressure the primary coolant pressure boundary and even a postulated control rod ejection accident would only lead to local fuel damage before the inherent negative thermal reactivity feedback mitigates the accident. The longer time afforded for communication means that no obvious imperative exists for any signal to have direct point-to-point wiring. Indeed, the timing requirements for an HTGR's digital communications network can be significantly relaxed from those for an LWR. Given the large time (days) for accident response at an HTGR and the resultant ability to manually perform safety system actuation in the event of a communications system failure, separating the digital control network from the safety network may no longer be advisable. The continuous use of the control network provides on-line, albeit limited functionality, end-to-end testing of the communications network, and this may result in a higher reliability automatic system while the manual back-up provides the defense-in-depth ordinarily provided by the separation of the control and safety systems.

In order to preserve the passive decay heat removal capabilities, HTGR prismatic cores are limited to roughly 600 MW(t) in size. Because of their relatively smaller thermal power ratings, HTGR cores are more likely, than would be the case for LWRs, to be grouped together to provide a larger amount of electrical power from a single site. Indeed coupling two 250-MW(t) cores together to feed a single steam turbine in the initial demonstration plant (with the intent to couple together several or even many modules for commercial operation) is the approach being taken by the Chinese HTGR program.[22] Having multiple reactor cores, perhaps constructed at different times and possibly collocated in a single containment (or confinement) feeding a single turbine drastically alters the plant instrumentation architecture. As the recent accident at Fukushima Daiichi have so dramatically shown, co-locating reactors and sharing services between them means that the impact of a serve accident at one reactor needs to be considered in the safety analysis of its neighbors. The newly re-emphasized instrumentation architecture issues (including basic questions such as how far to locate the control room from the containment) are beyond the scope of the current effort. Hence, this report does not address non-HTGR specific instrumentation architecture issues and is instead limited to the technologies embodying a modern (redundant, diverse, separated) HTGR relevant digital communications network.

A proposed I&C system layout for an HTGR with a steam-cycle is shown in Fig. 7. The figure includes process measurement variables and locations (i.e., temperature, pressure, differential pressure, and flow rate to provide a continuously updated portrait of the plant status). The figure does not include the safety-grade RCCS, which must also be considered in the overall heat balance calculation during normal operation.

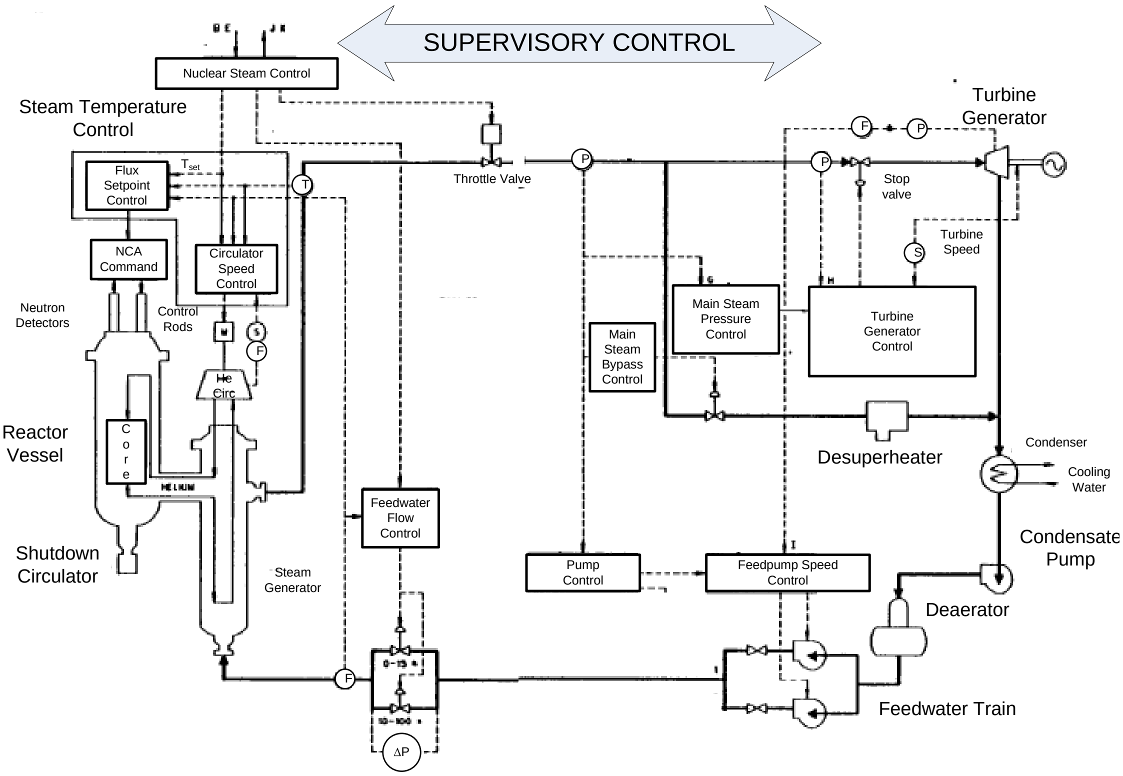

Fig. 7. Typical control system layout for an HTGR steam-electric plant.

(F: flow sensor, P: pressure, ΔP: differential pressure, S: circulator/turbine speed) (Ref. 23)

2.1 Conceptual Digital Network Topology Overview

The basic HTGR instrumentation network topology derives from the purpose of the instrumentation system and, hence, is closely analogous to that of other safety significant industries and prior nuclear power implementations. Evaluation of instrumentation system architecture details that are not specific to HTGRs is beyond the scope of this report. More general modern nuclear power digital networking architectural information is available in NUREG/CR-6991(Ref. 24).

Network information flow begins with transducers that feed information into signal processing electronics, which in-turn are interconnected into a network access point. The network access point is a component of a digital signal network that communicates information from sensors to data loggers and control/decision points as well as transmitting instructions from the decision points to the plant sensor and actuation systems. The basic network layout for any safety significant industrial plant is a redundant collection of high-speed connections. The classical sensing topology is to provide wiring to each sensor as wiring provides a robust, durable means of providing power to the sensor and a high-reliability signal path to the sensor.

The communications network in LWRs is segregated between safety and control systems. Given the much larger time response available for an HTGR, the safety and control functions may be comingled with the human operators providing the diverse backup. Equipment health measurements may also be transmitted over the communications network. Transmitted instructions to the sensor system could be a query for nonstandard reported information such as the amount of drift compensation currently being applied or an instruction to perform a self-calibration. The communication network also includes the interconnection to the system actuators. As the communications network needs to support post-severe

accident sensor communication, the network electronics will need to have appropriate environmental tolerance.

A conceptual illustration of a portion of a potential advanced HTGR network topology demonstrating the architecture features, described in the following sections, is shown as Fig. 8. Two out of a typical four redundant communication networks that would be anticipated in a future HTGR are shown in the figure. A wired-ring topology has been selected as the sensing network backbone. Figure 8 shows the network layout impact of several candidate advanced technology concepts. Much as in LWR instrumentation channels, the sensors themselves may be common to multiple instrumentation networks. Communication path diversity is increased by connecting sensors via both wired and wireless means. Wireless is only shown as in a containment (confinement) technology to minimize the potential for external intrusion. System actuators are shown connected to an IEEE 1901 type power line communication network.[25] Safety actuators are connected to multiple communications networks for redundancy. Safety sensors are shown connected to multiple networks to enable voting. Diagnostic sensors may be connected to one or more networks. Local environment power scavenging is shown supporting both wired and wireless sensor communication networks. Wireless network access points are shown to address more than one sensor.

2.2 Redundancy, Diversity, and Separation

A major portion of the rationale for redundant instrumentation networks is to continue to operate when any single component has failed. Since highly reliable operation is as valuable to an HTGR as any other power plant, their instrumentation redundancy will closely resemble that of an LWR or other large industrial facility. In contrast, it is anticipated that the much larger time allowance for accident response provided by an HTGR may enable the system diversity to be provided by human operators. In essence human thought patterns are diverse from digital instrumentation systems and the many hours to few days available to take safety actions at an HTGR allows human-based action to be highly effective in performing correct safety functions provided the humans have access to correct information about the system.

The basis for the physical separation requirements for signal paths is avoiding vulnerability to common-cause failures such as fires or local flooding. The physical pathway separation requirements within any single HTGR reactor module will closely resemble those for a single unit LWR. However, with the additional depth of integration between multiple smaller HTGR modules (much as with other technology small, modular reactors), logical and physical separation between the signals and signal pathways between different HTGR reactor modules is also necessary.

2.3 Wireless

Laying communications wiring during plant construction is not especially costly or difficult. Further, in most deployment scenarios, a wire (or optical fiber) based communications path provides a defined communications path with higher bandwidth than is readily achievable with a wireless network. Moreover, wireless communications within power plant environments (with their large reflective metal structures and strong electromagnetic emissions) is much more technically restricted than in milder environments. In addition, the cyber intrusion vulnerability is greater for a wireless network. Hence, a fully wired communications network would on initial consideration appear to be technologically preferable. However, wireless communications networks (especially when coupled with environmental power scavenging) provide aging, flexibility, diversity, and severe accident survivability capabilities that cannot be matched by a wired system.

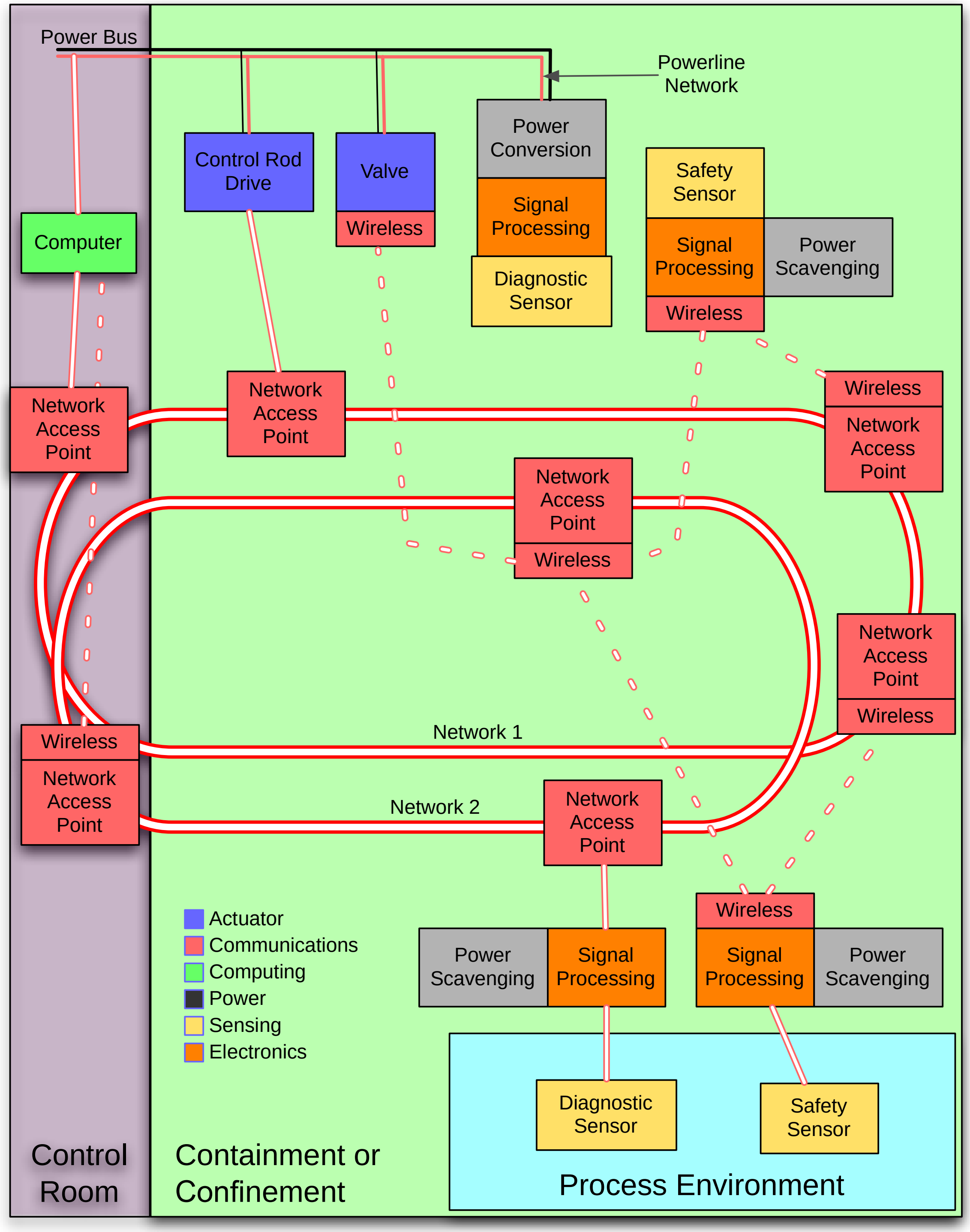

Fig. 8. Modern HTGR instrumentation network topology.

In order to provide protection from fire and/or mechanical damage, communication cables are embedded within protective materials making cable removal and replacement expensive and difficult. The organic insulators used in nearly all electrical communication cables degrade under plant environmental conditions in a few decades. Note that metal clad glass optical fibers do not depend on organic materials

and, hence, do not have the same aging concerns making metal clad glass optical fiber networking a potentially advantageous, albeit initially, more expensive, network physical layer. Embrittled cables can fail suddenly when subjected to severe accident conditions (e.g., being shaken by an earthquake). This failure vulnerability cannot be diagnosed from cable electrical properties measured at the cable ends. Cable replacement technology and cable health diagnostics are, thus, current LWR sustainability issues. Wireless data transmission by avoiding cables altogether offers a potential solution to cable aging issues.

As a rule, the amount of information being gathered and communicated about industrial processes increases with time resulting in continually increasing data networking demands. Further, digital components tend to become obsolete much more rapidly than their analog counterparts. Thus, nuclear power plant instrumentation networks need to be designed for upgrading and replacement (likely more than once) over the plant lifetime. Component and structure monitoring for diagnostics and prognostics is becoming progressively more common. Over time, it is likely that additional measurements, potentially only used intermittently, will be added to the HTGR. Deploying additional wiring at an HTGR is both expensive and can be disruptive of normal operations. Wireless networks in contrast offer the potential to rapidly reconfigure the instrumentation network in response to evolving demands.

Even with protective layers, wires are more vulnerable to fires and mechanical or radiation damage than wireless communication paths. Moreover, wireless networks can be designed for ad hoc reconfiguration routing around damaged segments. Hence, wireless networking provides a unique potential for higher assurance communications under severe accident conditions.

Lacking full control over the communications path, wireless networks are more vulnerable to cyber intrusion or unintended electromagnetic disruption than a wired network. Partially mitigating the vulnerability, many wireless communications technologies are inherently short-range, and the reactor containment (confinement) both provides considerable electromagnetic shielding and is physically distant from uncontrolled space. However, some plant systems are located nearer to the plant boundaries and may involve longer-range wireless communications technologies. A longer-range, spatially distributed networking is much more vulnerable to cyber attack than a short-range wireless instrumentation network deployed only within plant buildings.

Overall, future HTGR network architectures are likely to consist of a mixture of wired and wireless networking. The wired network is likely to consist of high-bandwidth cabling with wireless network access points distributed along its length. The additional accident response time inherently provided by an HTGR is anticipated to eliminate all point-to-point wiring connections to the control room. The wired network may be an all ceramic and metal configuration designed to last the life of the plant or may be designed for periodic replacement. Powered actuators (e.g., control rod drive motors) are likely to route their communications over their associated power lines. Individual sensing elements remain likely to have a wired connection to the nearest environmentally benign area where their measurement electronics will survive, as electronics remain the most environmentally sensitive component in the instrumentation system. Higher-temperature, tolerant electronics are under development to support a number of industries. Some of the high-temperature tolerant electronics technologies are also more tolerant of higher radiation doses. Consequently, the severe accident survival of local instrumentation electronics packages will improve over time, and the first-stage electronics will progressively be located closer and closer to the transducer.

2.4 Instrumentation Power Scavenging

Wireless instrumentation networks lack an obvious means to supply instrumentation power. Instruments, however, only require small amounts of power, and the combination of batteries with power scavenging from the local environment has the potential to enable truly wireless instrumentation. Alternatively, electrical- and optical-wired networks can provide instrumentation power over the same wires used for communications. In particular, IEEE’s recent 1901 broadband power line standard should be considered

for endorsement for use at nuclear power plants. Optical power networks may be especially useful under flooding conditions, as optical fibers cannot be shorted out.

Batteries without automatic recharging would be an on-going maintenance burden at nuclear power plants. In addition, batteries may be in a near-discharged state just prior to a severe accident—when the demand placed upon them would be greatest. Under normal operating conditions, power plants provide an energy-rich environment that can be harvested to provide instrumentation power. Solar cells can be employed to collect ambient lighting. Both mechanical vibration and airflow are also widely available energy sources. The large electromagnetic fields near motors similarly can be exploited for instrument power. Thermopiles can also be employed to harvest the energy inherent in the temperature difference between the process piping and the ambient environment. In short, during normal operations local batteries can be recharged by one of several alternate methods through scavenging power from the ambient environment.

In a post-severe accident environment, however, power scavenging would be technically challenging as the power harvesting devices need to directly couple with a local environment that is no longer well controlled. Given the relatively small power needs of instrumentation, however, dedicated local batteries (recharged under normal operating conditions via power scavenging) could be sized to provide several weeks of data before running down.

2.5 Aging and Obsolescence

The sensors, measurement electronics, and the communication network components all age and either they need to be designed to perform acceptably for the entire plant life or, more likely, to be replaced. At the most fundamental level instrumentation system aging is an expression of the inevitable entropy increase (atomic rearrangement) of the system components and materials over time. The higher temperatures of an HTGR cause more rapid aging of the sensor materials and thus necessitate the use of more durable materials. Sensor aging mechanisms are specific to the particular sensor, and a mechanistic discussion is provided along with the sensor descriptions later in this report. The electronics environment, in contrast, in an HTGR is likely to be either the same as that for an LWR or, near the primary piping, less severe due to the lack of ^{16}N decay gamma rays. The digital signal processing technologies employed in control room type environments in an HTGR are likely to be nearly identical to those used in any other modern industrial plant.

In combination with aging, instrumentation system components become commercially unavailable with time as suppliers change and upgrade their product offerings. HTGR instrumentation systems are more likely than LWRs to rely on standard, industrial safety-grade control electronics because the reactor inherent passive safety relaxes the system performance requirements. Use of standard components instead of the bespoken production common for LWR safety system electronics changes the system obsolescence paradigm.

At an LWR, once production of a particular, qualified digital component or system has ceased, specialized nuclear power specific qualification of a replacement system or component is required. As the replacement component or system is likely to have different characteristics or technologies than its antecedent (e.g., embedded software or different generation digital logic), qualification of the replacement component is generally both time consuming and expensive. Obsolescence avoidance strategies, such as lifetime stockpiling or dedicated remanufacturing of outdated components, do not allow LWRs to take advantage of the continuously improving digital system capabilities (such as built-in health diagnostics or automatic recalibration) that have demonstrably improved the reliability of safety significant control systems over the past few decades. In other words, the safety system qualification process at LWRs limits application of improved, albeit more complex, and difficult to qualify, digital instrumentation technology resulting in lower system reliability than might otherwise be achieved with the enhanced diagnostics available in modern digital instrumentation.

At an HTGR, in contrast, the major issue for instrumentation system upgrade is preservation of the original instrumentation system design objectives/requirements. Use of modern design description languages such as the Systems Modeling Language (SysML) or compliance with the ISO 10303 AP-233 standard on systems engineering data representation in the original plant design specifications is anticipated to alleviate the design intent tracing difficulty. Simply put, the inherent passive safety of HTGRs means that the human operators provide diverse and timely back up to all required safety actions greatly reducing the potential impact of any software or hardware errors and, thereby, reducing the required rigor of the instrumentation system qualification process. Thus, HTGRs are anticipated to be able to make much more extensive use of digital system improvements as they become technologically available resulting in a higher net instrumentation system reliability than is typical at an LWR.

The lifetime of digital systems is becoming progressively shorter and shorter as the speed of technological progression increases. Analog instrumentation was often available for a decade or more while digital instruments often have a model lifetime of just a few years. As most instrumentation components have lifetimes much shorter than the major plant structures, they need to be designed for replacement several times during the plant's operating life. One of the key lessons learned from LWR life extension efforts is that cable replacement is difficult and expensive. While HTGRs are expected to have much less cabling than older LWRs due to their network instrumentation configuration and use of wireless, the network cables will need to be designed for replacement. This means that the mechanical and fire protection layers over the cables need to be removable or the cables routed through conduit. Further, as cable cracking and embrittlement may not be observable from the cable ends and tends to occur locally at points of peak environmental stress, cable test coupons should be deployed along with the cabling to allow periodic destructive testing of cable samples.

The built-in, automatic sensor and electronics recalibration possible in modern digital instrumentation is more important to apply as outage intervals are extended. In modern, high-reliability electronics, measurement standards are becoming more commonly embedded within the electronics to provide higher measurement confidence and to extend the calibration intervals. For example, an embedded band-gap-type voltage reference circuit[26] provides a high confidence voltage level to a measurement circuit that can periodically be automatically switched into the measurement path to recalibrate the amplifier and analog-to-digital conversion circuitry. Similarly, a time (or frequency) standard can be provided either wirelessly or over the network to compensate for any drift in the measurement electronics timing. Temperature compensated crystal oscillators periodically recalibrated with a local GPS-type (double oven controlled crystal oscillator DOCXO) timing signal provide very good local time measurement stability.[27] Indeed this is the timing maintenance configuration used in the commercial cell phone network. A realistic goal for digital measurement instrumentation is to have an external recalibration interval longer than the lifetime of the technology avoiding manual recalibration altogether.

2.6 Deployment and Access Issues

The combination of the higher temperatures and high-pressure characteristic of an HTGR makes deploying primary system instrumentation technically challenging. Of particular note is the relatively large mechanical shift between the core and the vessel due to the differential coefficient of thermal expansion between the graphite moderator and the metal pressure vessel. In essence, the motion of the core would provide large lateral stresses on any sidewall penetrations limiting measurement access to above and below the core. Additionally, access to the hot inner annulus of annular primary piping (hot inside, cooler outside) is restricted by the mechanical shifting of the inner piping as the reactor heats up.

The thick vessel and piping walls make ultrasonic access to the primary loop technically difficult (a potential work around for the ultrasonic access issue is described in the ultrasonic guided wave sensor section of this report). Similarly, a fully qualified optical penetration into primary piping does not currently exist and as the coefficient of thermal expansion of fused silica and sapphire are much less than

that of steel, any optical penetration will most likely be provided at the end of a standpipe extending out to a lower temperature environment.

2.7 Decision Aid Tools

While the high degree of passive safety inherent to HTGRs limits the demands on the operators, human error on the part of an HTGR's operating or maintenance staff remains a potential source for accidents. Operators taking the correct actions are also central to preventing accidents from escalating and to mitigating the consequences of any accidents that do occur. Decision aid tools can support the plant staff by ensuring that relevant information is readily available. For example, an electronic job notepad can monitor its location in the plant, thereby increasing the confidence that a maintenance worker is performing work on the correct piece of equipment. Electronic notepads can also provide much more graphical detail on component maintenance procedures than their paper counterparts. The added support is especially important for infrequently serviced components where the plant staff may have less familiarity with the maintenance procedures.

Emergency operations procedure manuals provide guidance to the plant operators on how to respond in the event of an accident. Emergency operations procedures have traditionally been paper based and as a procedure-based system provide limited insight into the physical processes occurring in the plant. Electronic alternatives to paper manuals have begun to be used in many industries. A key advantage to electronic manuals is that their recommendations can be continuously updated as an accident evolves by tying the recommendation engine into a plant simulation. Further, the plant status information can be displayed in an easier to understand format, such as an augmented reality temperature overlay onto a view of the reactor vessel.

The major purpose of HTGR accident progression simulators is to support operator decision-making by projecting the outcomes of alternate recovery actions. Thus, a premium is placed on having correct physics models for accidents and on correctly interpreting sensor information under degraded conditions. HTGRs in general and especially the pebble bed modular reactors (PBMRs) have few in-core sensors. Employing a simulator coupled with ex-core sensors (inlet and outlet temperatures, flux, rod positions, coolant flow rate, pressure, as well as the gas composition both within vessel and containment, etc.) could provide assurance that any particular recovery action would produce the intended result. Given the slow accident progression scenarios characteristic of HTGRs, the operators (or outside experts) would be able to perform reasonableness checking on the simulation results before there would be a need to take actions.

Techniques for accommodating degraded sensors need to be part of any accident response support simulator. Under accident conditions, sensing elements, the transducer electronics, and the communications hardware, etc., may be operating outside of their qualified bounds. Being able to interpret the received signals and the characteristic changes in the signals if the sensors are stressed enables the best available data to be fed into the accident simulation. The Graphite Reactor Severe Accident Code,[28] GRSAC, is an applicable HTGR accident progression simulation code. However, while the model includes default data to fill in for unavailable measurements, the simulation relies upon external data validation.

The accident progression simulation output to the operators would include the anticipated radionuclide releases, the predicted core temperature distribution, the reactivity balance (predicted flux measurement for the measured control rod positions), and the heat balance. For predictive simulations, a major consideration would be optimizing the displays to minimize the chances of misunderstandings.

3. NEW AND EMERGING HTGR SENSORS

Previous reports produced under the current scope of work have reported on current and prior generation instrumentation that have been or are planned to be deployed at very high temperature gas-cooled reactors (VHTRs). The current effort is focused on describing more advanced sensing and instrumentation technology that, while not yet ready for deployment at nuclear power plants, offers the potential for substantial performance improvements over currently available technologies. Maintaining awareness of the potential capabilities and limitations of emerging instrumentation technologies is important in that future deployment of these systems may change the degree of and confidence in the understanding of the plant conditions and, thus, the safety profile. Additionally, while generally improving on the capabilities of current generation instrumentation, the new instrumentation has different failure modes and vulnerabilities and understanding the performance parameters is important to be able to assess the safety impact of emerging instrumentation technology.

3.1 Temperature

Both accurate primary circuit heat balance and in-core temperature measurement for pebble-bed systems remain challenging instrumentation problems. The drift of current generation thermocouple or resistance temperature detector (RTD) based thermometry at hot, and to a lesser extent cold, leg temperatures would necessitate a larger plant instrumentation margin to avoid unacceptable thermal degradation of the primary circuit materials. Hence, a significant impetus exists to develop and adopt less drift prone, higher-accuracy temperature measurement technology.

3.1.1 Gold-platinum thermocouple

Existing precious metal thermocouples can be used with care outside of high neutron flux regions to provide reasonably high precision temperature measurement (initially roughly ±200 mK above 500°C). Note, however, that the precious metal elements have too high a neutron capture cross sections for use in areas with significant neutron flux. For higher stability and precision temperature measurement, the pure element (99.999%), nonletter designated Au-Pt thermocouple can achieve precision of approximately ±10 mK at temperatures up to 1000°C.)[29] Overall, the Au-Pt thermocouple is markedly superior to the platinum-rhodium alloy thermocouples in terms of stability, homogeneity, and sensitivity (about double a type S) and, thus, appears likely to be the preferred ex-vessel temperature measurement instrument for VHTRs. However, the stability and durability of mechanically rugged, metal-sheathed, mineral-insulated versions of the Au-Pt thermocouple has yet to be demonstrated sufficiently for immediate application to safety important measurements at nuclear power plants. In particular the coefficients of thermal expansion (CTEs) of gold and platinum are sufficiently different that the hot junction mechanical interconnection needs to be flexible to avoid stressing the elements. A thin, gold bridging wire has been shown to provide the necessary strain relief.[30]

The largest concerns for replacing RTDs with Au-Pt thermocouples are the increased electromagnetic noise vulnerability due to the smaller signal and maintaining proper cold junction compensation. Thermocouple signals are typically around 10 mV, whereas industrial RTD signals are at least 100 mV. Thus, both better electromagnetic shielding and higher gain, high stability amplifiers are required for thermocouple readout. No standard compensating wire currently exists for Au-Pt thermocouples and, as the thermocouple wire is expensive, the thermocouple itself will likely only extend a few centimeters past the exterior of the insulation surrounding the primary circuit. The temperature at the cooler, proximal end of the thermocouple will, thus, be required to be independently measured to compensate for its difference from the ice point. As the Au-Pt cold junction temperature is likely to be less than 50°C, readily available thermistors or capsule-type RTDs appear to be capable of measuring the cold junction temperature. Also, while not a significant issue for temperature measurement within large primary piping, the thermal

wicking down the thermocouple leads also needs to be considered in the measurement system design. Both gold and platinum are good thermal conductors, and thermocouple leads have larger diameters than RTD leads—because of the requirement to minimize the impact of impurities. Thus, a longer immersion depth is required for Au-Pt thermocouples as compared to RTDs for accurate measurements.

3.1.2 Johnson noise thermometry (JNT)

The impact of the mechanically and chemically induced changes in RTD electrical resistance with time at high temperature can be avoided by instead basing the measurement on a fundamental property of temperature. Johnson noise is a first principles representation of temperature. Johnson noise results from the vibration of the electronic field surrounding atoms as they thermally vibrate. Since temperature is merely a convenient representation of the mean kinetic energy of an atomic ensemble, measurement of these electronic vibrations yields the absolute temperature.

As a first principles temperature measurement, Johnson noise is insensitive to the material condition of the sensor and, consequently, immune to the contamination and thermomechanical response shifts that plague thermocouples and resistance thermometers. Commercialization of Johnson noise thermometers has the potential to increase the accuracy of both near-core and primary-loop temperatures with the added benefit of reduced calibration requirements.

JNT is best understood as a continuous, first-principles re-calibration methodology for a conventional resistance-based temperature measurement technique. The traditional method of directly measuring temperature from an RTD has unavoidable drift. JNT measurement is applied in parallel to the RTD lead wires of the resistance measurement circuit without altering the traditional resistance measurement circuit. A more detailed overview of the application of Johnson noise to high-temperature reactors was published in 2005 (Ref. 31).

One significant advantage of JNT is that the actual resistive element is not required to follow a particular temperature to resistance curve. This allows consideration of high-stability, high-mechanical strength alloys or even a cermet as the temperature transducer—likely significantly increasing the lifetime and reliability of the sensor element. While the JNT amplifier electronics restrict the allowable range of transducer resistance shift with temperature to roughly a factor of three around the operational resistance of tens or hundreds of ohms, the exact shape of the resistance to temperature correspondence is not otherwise constrained apart from the basic requirement for a monotonic change with temperature.

Conventional industrial RTD elements are composed of high-purity, well-annealed platinum in order to achieve a repeatable resistance to temperature correlation. However, RTDs are not industrially deployed at temperatures over 850°C and are often restricted to temperatures less than 450°C for high accuracy deployments. As the deployment temperature increases, the CTE mismatch between the platinum element and the support structures exerts progressively greater and greater amounts of stress onto the temperature measurement element shifting its calibration. This is compounded by the softening of platinum at temperatures over ~650°C and the overall increase in impurity atom mobility at higher temperatures.

Molybdenum disilicide has been demonstrated as an alternate high-temperature resistor for Johnson noise thermometry up to 1600°C.[32] Hot-pressed silicon nitride—molybdenum disilicide composites (Si_3N_4–30 vol % $MoSi_2$) also have useful electrical resistivity for Johnson noise resistive elements as well exhibiting high mechanical toughness.[33]

Johnson noise thermometry has three significant sources of potential measurement error that may, in general, be avoided or compensated for by proper design and implementation. However, reviews of Johnson noise thermometers being implemented in safety systems will need to assess the design and implementation methodologies selected to overcome the vulnerabilities. While digital implementations of Johnson noise thermometry can compensate for some electromagnetic noise pick-up, Johnson noise is a small signal phenomenon. Consequently, severe electromagnetic pick-up (as would be anticipated from

improperly implemented grounding or shielding) would prevent Johnson noise measurement. Also, Johnson noise thermometry relies on high gain (~10^6) wide-bandwidth signal amplification. If the amplifier gain shifts over time or with electronics temperature, the Johnson noise thermometry would provide an incorrect RTD recalibration. Consequently, the amplifier gain characteristics need to be verified either on-line or via periodic maintenance. Finally, the intervening cabling between the sensor and the first stage signal amplification will cause the higher frequency components of the wideband Johnson noise signal to roll off restricting the allowed upper measurement frequency. Understanding and compensating for any shifts in the cable properties is required for a successful long-term implementation.

Developing a prototype commercial-style JNT is the subject of a U.S. Department of Energy (DOE) small modular reactors project that is currently getting underway.

3.1.3 Vacuum microtriodes

Measuring the pebble temperature on-line in-core is extremely technically challenging. Core temperatures and radiation fluxes are well beyond where semiconductor electronics can function. Vacuum tubes, in contrast, consist entirely of ceramics and metals (which are inherently extremely radiation tolerant) and, thus, have the potential to operate under the severe operating environment of an HTGR core. Vacuum tubes, in fact, have a heritage of core-type radiation environment operation due to their use in first-stage amplifiers for early fission chambers.[34]

The lack of measured pebble-bed core data requires that the pebble flow, neutron flux, and core temperature distribution be conservatively estimated limiting the achievable reactor core power density. Experience with the Arbeitsgemeinschaft VersuchsReaktor (AVR), a pebble-bed experimental high-temperature reactor, indicated that core coolant temperature simulations significantly underestimated the hot spots in the core.[35]

Several attempts have been made over the years to miniaturize vacuum tubes. Ceramic tubes similar in size to a pencil eraser were developed in the 1970s. Spindt tips-based tubes were developed in the 1990s and have a cavity cross-section of ~100 nm^2 (Ref. 36). The fundamental element of a vacuum tube is the triode. In order to provide an on-line, in-core temperature measurement the microtriode needs to measure the temperature (likely converting an analog voltage into a frequency), amplify the frequency signal, modulate the signal onto a higher frequency carrier signal, and wirelessly transmit the frequency-encoded temperature. These measurement process steps will require a power source, amplification circuitry, and signal transmission circuitry. From a circuit design perspective, four fundamental components are required to perform the measurement function: a resistor, a capacitor, a triode (amplifier), and a rectifier. An electronic oscillator (and, for that matter, any electronic component) can be built from these four fundamental elements.

Vacuum microtriodes are technically the furthest out of the emerging technologies. An estimate of their failure modes and vulnerabilities will need to await further development of the technology.

3.1.4 Ultrasonic guided wave

Ultrasonic probe thermometry has been under development since the mid-1960s.[37] Probe-based ultrasonic temperature measurements have been applied to nuclear reactor core temperature measurements since the late 1960s.[38-40] Ultrasonic probe-based thermometry was first compared to Johnson noise thermometry for reactor thermometry in 1974 (Ref. 41). Progressive development of high-temperature materials, high-speed electronics, and signal processing methods has pushed the technology forward. Simply put, probe-based ultrasonic thermometry can be deployed in very high temperatures and hostile environments due to lack of requirement for an electrical insulator and its compatibility with refractory alloys.

The most significant unresolved issue for implementing ultrasonic-guided wave thermometry as a general practice in nuclear reactors is the challenge of transmitting the ultrasonic-guided wave through the

primary pressure boundary. Solid-state measurement electronics are not compatible with HTGR temperatures. The ultrasonic waveguide, however, can extend several meters. Thus, the transduction electronics can be located above the reactor vessel head within a cooled high-pressure helium environment separated from the high-temperature environment by a small diameter standpipe up to a few meters long. The electrical signals would be the only components that would then need to pass through the pressure boundary.

Guided-wave, pulse-echo ultrasonic thermometry operates by measuring the speed of sound in a metal wire or rod, which is a known function of temperature. The speed of sound in a metal is dependent on the elastic modulus and density. Although both parameters are temperature dependent, the temperature effect on elastic modulus dominates by about an order-of-magnitude over that of density, which causes sound velocity to decrease with increasing temperature.

By launching a compressional wave down a cylindrical rod waveguide, the transit time can be measured as the wave travels from the launch location to a point of reflection then returns to the detection location. The temperature can be calculated from the transit time. By creating abrupt impedance shifts (such as notches) at predetermined locations, temperature measurement over multiple zones is made possible. The arrival times of reflections from multiple notches can be measured as shown in Fig. 9. The average temperature between notches can be calculated from timing data.

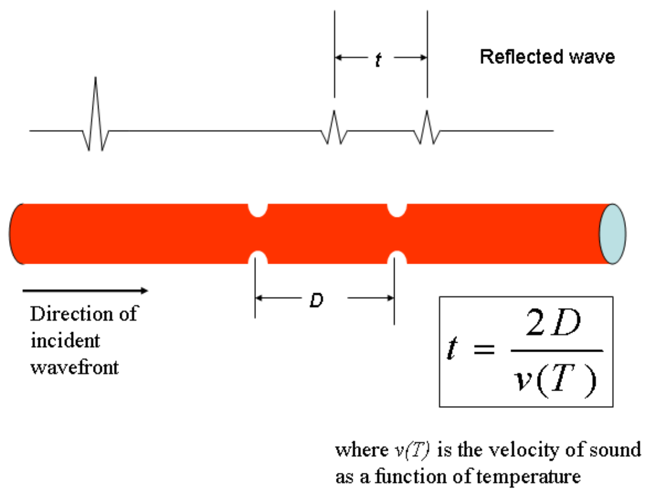

Fig. 9. Ultrasonic waveguide with notches.

Ultrasonic-guided wave thermometry is not immune from drift. The high temperatures and high radiation flux of an HTGR core environment will, over time, alter the mechanical properties of the waveguide and transmute its composition shifting the recorded temperature. Also, any solid deposits onto the waveguide will cause spurious echoes that can confuse the signal analysis. Additionally, relatively complex algorithms are required to reconstruct the temperature profile from the large number of reflections received from a multiply notched waveguide; thus, software quality assurance will require a comparatively larger effort. The DOE Nuclear Energy Enabling Technologies (NEET) program is initiating a project focused on deploying guided-wave ultrasonic thermometry as a fuel qualification thermometry technique.

3.1.5 Distributed fiber-optic Bragg thermometry

Distributed fiber-optic Bragg thermometry is based upon a series of Bragg gratings arranged along the core of a single-mode optical fiber (see Fig. 10). The simplest readout technique for a limited number of gratings along a fiber begins by launching a band (range of wavelengths such as from a light-emitting diode [LED]) of light into the optical fiber. Each grating reflects a specific wavelength within the band. The particular wavelength reflected is determined by the Bragg grating period with each individual grating having a slightly different spacing. Temperature causes the grating period to shift both by thermal expansion as well as change in the refractive index. A shift in the reflected wavelength thereby corresponds to a shift in the temperature of a particular Bragg grating.

The primary advantages of distributed fiber-optic Bragg thermometry are that the sensor is nonconductive allowing for deployments in high-electromagnetic field environments such as pump motors and turbines, and that many sensors can be configured along a single path enabling the acquisition of a distributed temperature map with a single readout system. This would enable applications such as direct observation of the temperature profile across the primary piping instead of relying on single radius sampling. Note that by properly adhering the Bragg gratings to the piping, the distributed thermometer would transform into a distributed set of strain or vibration gauges enabling mechanical diagnosis of piping or components.

Optical Frequency Domain Reflectrometry (OFDR) can also be employed to measure the signal from many (thousands) of individual gratings along a fiber.[42] OFDR is an interferometric technique requiring an adjustable wavelength, coherent light source. Tunable lasers remain somewhat expensive and have more limited lifetime than simple, wideband light sources. Consequently OFDR would only be the preferred read-out technique for large sensor arrays.

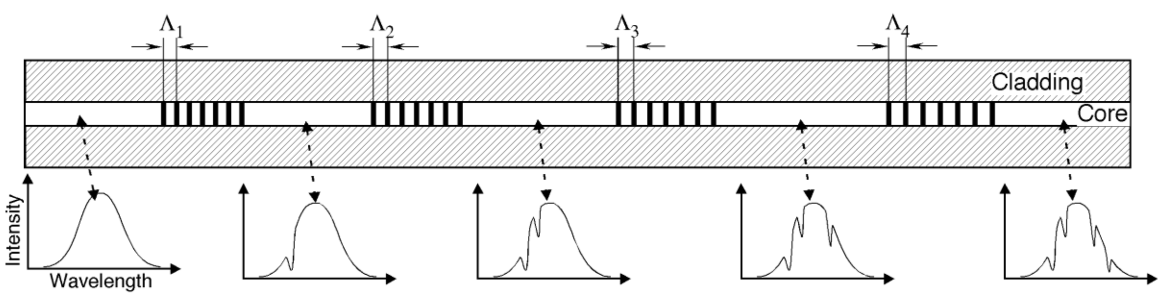

Fig. 10. Transmitted light spectra through a distributed optical fiber Bragg grating.

Distributed fiber-optic Bragg thermometers have been demonstrated to function briefly (few days) in high (core-type) radiation environments and much longer in more moderate radiation environments.[43–45] The optics and electronics for distributed fiber-optic Bragg thermometers can be located hundreds of meters from the sensing elements allowing placement in well-controlled environments at nuclear power plants. For OFDR, in particular, the readout electronics are somewhat more complicated than for competing technologies such as thermocouples requiring more effort in the qualification process. Also, Bragg gratings in standard communication-type optical fibers bleach out upon exposure to combined high temperatures and high-radiation fields. The grating bleaching vulnerability requires employing less common, custom optical fibers expressly designed for higher temperature, higher dose deployments. This contrasts with resistance temperature detectors and thermocouples where devices suitable for nuclear power application are substantially the same as for nonnuclear deployments.

3.2 Flow

Primary helium flow rate is a factor in the measurement of core heat removal. Primary coolant flow is not typically directly measured in HTGRs but is, instead, inferred from either the turbine-compressor or the helium circulator rotational speed depending on whether the reactor is directly or indirectly coupled to the turbine. The high-pressure and temperature primary flow environment effectively eliminates consideration of all commercial flowmeters. However, thermally based flow measurement can potentially be configured to perform under VHTR temperature, pressure, and flow conditions.

3.2.1 Hot wire anemometry

Hot wire anemometry functions by running a current through an electrical resistor, connected as a four-wire element, which is exposed to the flow (see *www.efunda.com*). The voltage across the resistor is measured. Combining the impressed current and the measured voltage yields the element resistance. The resistance is then used to determine the element temperature through a resistance to temperature look-up table. The current through the resistor is varied to maintain a prespecified temperature difference from the flow (whose temperature is measured independently). The amount of current necessary to maintain the temperature difference is a direct measure of the heat transfer from the electrical resistor, which, in-turn, is a measure of the coolant flow rate.

The electrical element used in conventional hot-wire anemometry is directly exposed to the flow environment providing high frequency flow mapping. Primary HTGR flow would rapidly destroy any electrical element that is directly immersed within it. However, sheathed elements, such as a typical RTD, can endure HTGR cold-leg environments, and sheathed, conductive ceramic composite electrical resistors can withstand hot-leg temperatures. Flow measurements made using an RTD as a hot-wire anemometer would have the same time response as would be anticipated when RTD is used as a thermometer (several seconds for response time).

3.2.2 Heated lance—gamma thermometer-type probe

A heated lance-type flowmeter is conceptually similar to a hot-wire anemometer with a nearby electrical resistance heater taking the place of the current supply and a thermocouple being used to measure the temperature.[46] An illustration of the concept is shown in Fig. 11. To minimize the impact of thermocouple drift on the temperature measurement, the thermocouple leads can be configured as in a gamma thermometer with one of the junctions thermally insulated from the flow by an annular gas gap. In this case, the temperature difference between nearby thermally insulated and thermally connected thermocouple junctions is inferred by the voltage measured at the ends of the thermocouple leads. The temperature difference between the two junctions is proportional to the heat deposition and the cooling rate. The cooling rate is directly proportional to the temperature difference between the element and the coolant as well as the coolant flow rate.

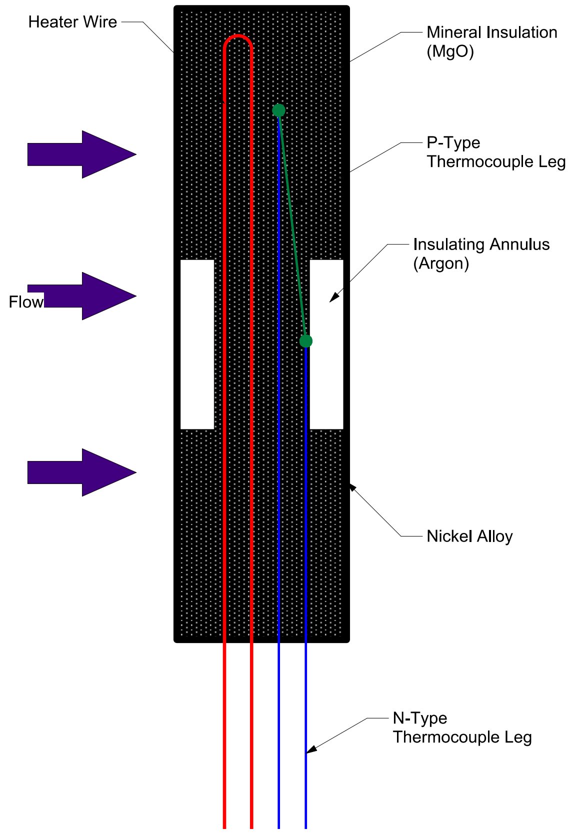

Fig. 11. Heated lance-type flowmeter.

The most significant issue for thermally based measurement of the primary flow in HTGRs is that both techniques are point measurements (although multiple elements can be deployed along a single probe). Flow within the primary ducting of most HTGR designs is not fully developed and may not have a stable profile. Hence, a point measurement may not provide a reliable mass flow rate.

3.2.3 Projection Laser Doppler Velocimetry (LDV)

The motion of graphite dust within the helium coolant can be used to infer the primary coolant mass flow rate. Laser Doppler Velocimetry (LDV) has until recently been difficult to implement in industrial situations due to the precision optical alignment required. Conventional LDV crosses two beams of coherent laser light in the dust containing primary coolant flow creating an interference pattern perpendicular to the flow direction. As particles pass through the constructive interference fringes, they scatter light into a photo-detector. The motion of the dust particle as it crosses the fringe pattern causes a frequency-shift (Doppler) in the scattered light enabling calculation of the particle velocity. Conventional LDV requires alignment of two laser beams and good optical access for imaging the beam intersection point.

Projection LDV is relatively new LDV variant that creates the fringe pattern by placing an optical mask within a single laser beam.[47] Thus, the fringe pattern is projected along the laser beam. As the dust particles cross the fringe pattern, the Doppler shift in their scattered light wavelength indicates their velocity. The detector can be configured coaxially with the laser (as shown in Fig. 12) by employing an oval optical mask oriented 45° to the circular laser beam. The optical mask would be absorptive on the incident laser side and highly reflective on the return side to direct the scattered light into the detector.

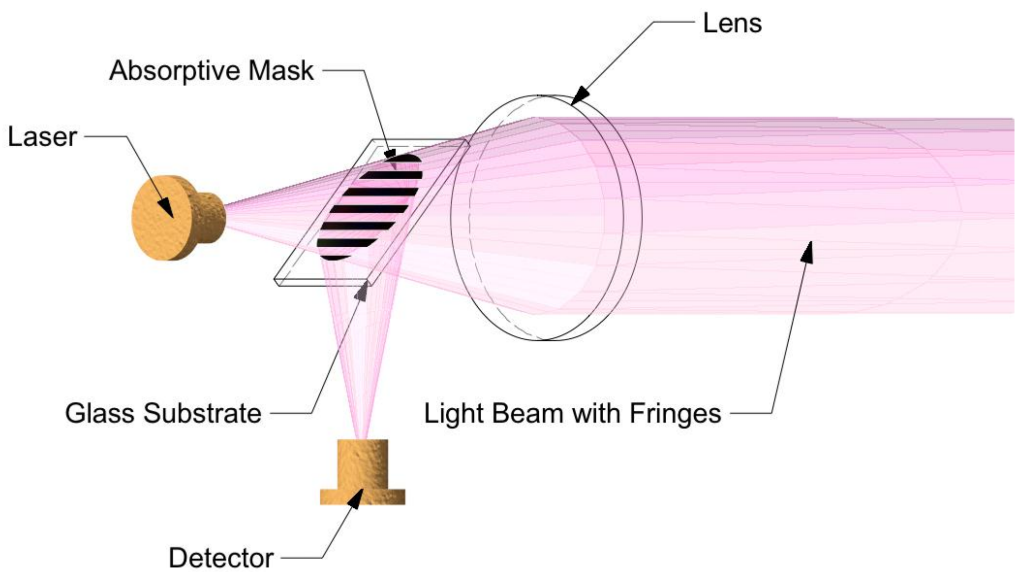

Fig. 12. Projection LDV optical components.

Projection LDV provides a measure of the particle flow across a chord across the flow. Chirping the laser would enable a flow profile to be measured along the laser path. The optical components of the projection LDV system would be located outside of the primary coolant with optical access provided through a high-pressure window located in a standpipe to lower the window temperature. The number of scatters could also be counted to provide a measure of the dust level within the coolant.

3.3 Pressure

Pressure measurements are useful at HTGRs as supporting measurements to confirm proper functioning of the heat transport system. Differential pressure measurement is useful at HTGRs as a corollary measurement to indicate failure of system components or structures. For example, differential pressure measurement across the core could serve as an indication whether flow resistance has changed. Pressure is typically measured by converting the pressure into a mechanical deflection, which is, in-turn, converted to an electrical signal. For solid (or liquid) elements only, a pressure differential across an element will result in an unbalanced force and, hence, a mechanical deflection. Creating a controlled pressure difference at operating temperature between HTGR pressure and a reference pressure is technically challenging due to the high temperature mechanical property shifts of structural alloys. While the primary helium itself can serve as the fill fluid for impulse lines running back to more moderate environments where conventional measurement electronics can survive, access concerns frequently make running impulse lines impractical.

3.3.1 SiCN internally self-referenced composite ceramic

An innovative approach to developing a differential pressure is to employ a closed-pore ceramic sensor body. The sensor body is deformed by the difference between the internal and external pressure and the electrical resistance; the sensor body is dependent on its strain state.[48] A unique characteristic of polymer-derived ceramics is their formability prior to firing allowing creation of complex shapes with designed pore structures.[49] Polymer-derived ceramic bodies based on these principles are already under

development as high-temperature, high-pressure sensor materials focused on power generation.[50] In this case, the sensor body as a whole can serve as a strain gauge with a nonporous sensor body employed as a sensor reference leg. A particular challenge for this type of sensor is wiring the signals back outside the primary circuit. As the ceramic element and electrical wiring pads are directly exposed to the primary fluid, avoiding shorting the electrical leads (due to carbon dust or metallic deposits arising from evaporation impurities within the structural alloys) becomes a significant technical issue.

3.3.2 Liquid metal impulse line

Impulse lines are the most common method to transfer a pressure from a harsh environment to a more benign one where measurement electronics can survive and, indeed, a primary coolant impulse line is typically used at HTGRs wherever mechanical access is available. Liquid-filled impulse lines, however, both provide shorter response times as well as providing an additional isolation barrier to the primary coolant. In order to serve as an impulse line fluid, the enclosed liquid needs to remain in liquid state at both the hot process temperature and the lower measurement temperature. Sodium–potassium mixtures meet the melt point criterion. While the alkali metals can react violently with some materials, the impulse line can have a very small diameter limiting the amount of material present. High-temperature pressure sensors incorporating a NaK impulse line are currently available as special order item.[51]

3.3.3 Polarization rotation

An internally self-referenced pressure sensor can also be realized by the use of coefficient of thermal expansion matched glass and metal sensor assembly. In this case, the glass piece serves as a diaphragm sealing off a cavity in the sensor body (see Fig. 13). This type of mechanical configuration has been demonstrated for turbine engine control.[52] High-temperature application requires matching the CTE of the glass with that of the supporting structure. Aluminosilicate glasses have CTEs that nearly match those of molybdenum alloys enabling development of a sealed-glass diaphragm—refractory metal system.

The pressure-induced stress on the glass causes local realignment of the electric field within the glass; changing the refractive index for light propagation in the direction of net applied stress (birefringence). Placing the measurement glass between a set of crossed polarizers and monitoring the intensity of the transmitted light provides a measure of the polarization rotation and, thereby, the pressure.

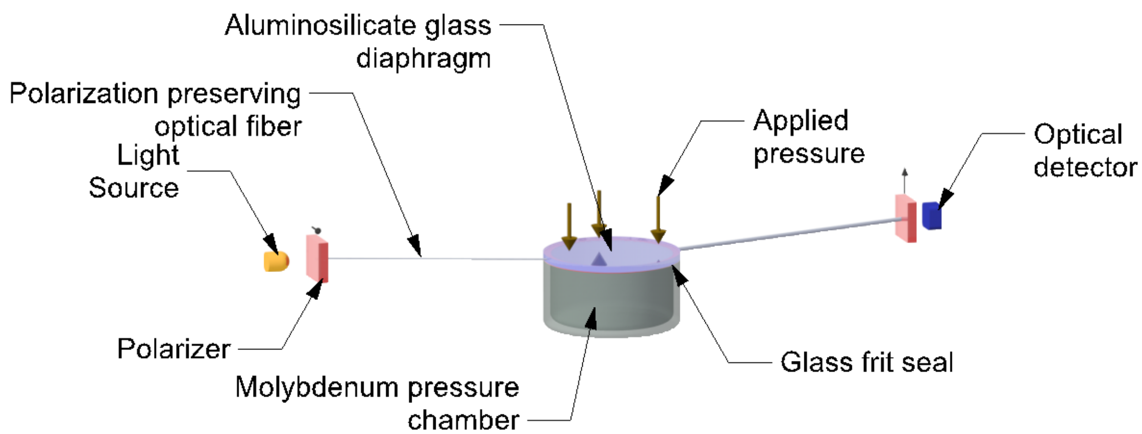

Fig. 13. Polarization rotation pressure sensor.

A polarization rotation pressure sensor has to overcome a number of challenges to become a reliable pressure sensor in HTGR environments. Although metal-jacketed optical fibers can be brazed into standard gland-type pressure seals, polarization preserving optical fiber penetrations into HTGR-type environments are not yet standard items. Also, a multilayered glass diaphragm and chamber seal will be required as many glasses (including the aluminosilicates) are somewhat permeable to helium at high temperature. A helium leak into the pressure chamber of this type of sensor (such as via helium percolation through either the diaphragm or the glass frit) would be difficult to detect in the measurement signal leading to progressively larger measurement error with time.

3.3.4 Extrinsic Fizeau cavity

The position of a diaphragm between the high-temperature helium and exterior pressure can also be read out optically. Essentially, a Fizeau cavity is formed by the distal end of an optical fiber and the diaphragm.[53] The diaphragm deflection changes the cavity tuning and, thus, the wavelength specific reflectivity of the cavity. The large mismatch between the CTE of silica glass and metallic support structures makes maintaining a precise spacing between the diaphragm and the fiber tip with varying temperature challenging. However, proprietary graded joint approach to maintain precise alignment between the silica optical fiber and a metallic housing has now been demonstrated to enable wide dynamic range pressure sensing to above 800°C (Ref. 54). Similar technology has also been employed for high-temperature measurements ex-core HTGR-type radiation environments, but continues to have measurement drift limitations.[55]

3.4 Nuclear

Monitoring a fission reactor's neutron flux is a key safety and performance measurement. Neutron flux measurement instrumentation is important for reactor control because it provides unambiguous indication of spatial and temporal levels and variation of neutron flux, guides the operator with respect to the approach to criticality, and provides a measurement of reactor power.

3.4.1 High-temperature fission chamber

No suitable neutron flux measurement technology is commercially available that functions at temperatures above 550°C. HTGRs have core temperatures well above 550°C. Prior HTGR development programs have demonstrated fission chambers that function up to 800°C (Refs. 56 and 57). However, these detectors are not commercially available. The current design intent for the NGNP is to back fission chambers away from the core and infer core flux conditions from the remote measurements. Developing fission chambers that are directly usable near the core is a significantly superior measurement approach. Neutron flux can peak sharply within the core, and the peaking has both safety and performance implications. It may be difficult to reconstruct local flux peaking from remote sensors due to its fine spatial structure.

The fundamental processes of fission chamber-type neutron chambers are not temperature dependent. The failure of fission chambers at high temperatures is most commonly due to metallic deposits, arising from evaporation of contaminants (or alloy components) from the structural alloy, forming across the electrical insulator between the central anode and the wall shorting out the chamber. In order to overcome the temperature vulnerability of fission chambers, low outgassing structural materials and high-temperature tolerant sealing materials and methods need to be devised.

The fissile material of a fission chamber is consumed in the measurement process. The chamber sensitivity, thus, decreases as the neutron fluence increases. Longer-lived fission chambers combine a fertile with a fissile material, ^{234}U with ^{235}U, to extend the chamber's useful period of performance. While the general burn-out characteristics of the fission chamber are known and tracked, sufficient uncertainty

typically remains in the chamber sensitivity that the devices need to be periodically recalibrated using the heat balance measurements provided by the primary coolant temperature and flow measurement.

A DOE NEET project to fabricate and demonstrate a high-temperature (900°C) tolerant fission chamber has recently been initiated.

3.4.2 Gamma thermometers

The local core power distribution is a primary reactor performance and safety parameter. While traveling in-core probes and self-powered neutron detectors are commonly used as local power monitors in LWRs, both technologies would require significant redesign to function at the higher temperatures of VHTRs. The materials and structures of gamma thermometers are similar to thermocouples and heated lance-type level probes and, as such, are anticipated to be suitable for higher temperature. However, high-temperature reactors present different chemical environments as well as higher temperatures and as such gamma thermometers for high-temperature reactor cores require alternate sheath material selection as well as structural re-engineering.

Gamma thermometers function based upon the heating of the sensor assembly by gamma rays and the subsequent controlled differential cooling of the sensor body.[58] Gamma thermometers (as shown in Fig. 14 below) essentially consist of a series of differential thermocouples embedded within an electrically insulating body. At periodic intervals along the body, annuli of the electrical insulation are removed. A linear electrical heating element is also typically included to enable sensor calibration. The resulting structure is then swaged into a metal jacket. The temperature differential developed between an insulated and an uninsulated thermocouple junction within the body of the sensor is proportional to the rate of local heating by the gamma radiation, which is in turn proportional to the local power generation rate during power range reactor operation. The combination of insulated and uninsulated temperature measurement decouples the sensor response from the coolant flow rate as nearly all of the deposited heat is removed via the coolant flow, yet the two temperature measurement junctions have different thermal flow path lengths to the same flow-cooled surface.

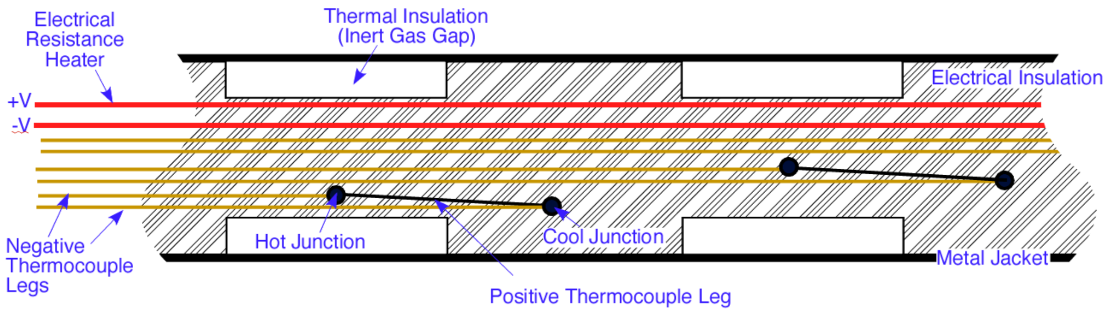

Fig. 14. Gamma thermometer conceptual layout.

Gamma thermometers are anticipated to be reliable sensors with their largest issues being properly compensating for the neutron component of the heating, pinhole leaks in the outer jacket, and susceptibility to electromagnetic interference in the region between the end of the metal jacket and the signal amplifier. Gamma thermometers have not been demonstrated at the higher temperatures of HTGRs, and the additional radiative heat removal across the insulation will need to be compensated for either in the sensor design (low emissivity coating of the annular gap) or in the sensor calibration by supplemental temperature measurement.

3.4.3 Micropocket fission chambers

Micropocket Fission Detectors (MPFDs) are pancake-style, highly miniaturized fission chambers that employ sealed alumina plates as their structural backings and coatings of uranium or thorium as their neutron sensitive element.[59] Typical device dimensions are 0.5–2 mm in thickness and 1–3 mm in diameter and are operated with ~200 V of bias. Multiple chambers can be co-located on the same substrate providing the ability to subtract leakage current (blank), detect thermal neutrons with enriched uranium, and detect fast neutrons with thorium.[60] A temperature measurement can also be incorporated into the detector body enabling controlled leakage current compensation.

While highly promising and composed of materials that are anticipated to be high-temperature tolerant, MPFDs remain to be qualified for the high-temperatures and long performance times required for local flux measurements in VHTRs. The DOE NEET program is currently initiating a project to mature the technology in support of fuel qualification testing.

3.5 Primary System Gasses

Primary coolant chemistry monitoring has most commonly been performed off line with a low-pressure, cooled sample of primary coolant. A wide range of technologies is available for off-line measurement of the trace constituent chemistry and moisture content of helium. The chilled mirror technique employed at Fort St. Vrain can provide accurate moisture measurements but is both labor and maintenance intensive. Future helium composition measurements are most likely to be performed through

1. optical absorption spectroscopy,
2. capacitive shift due to water absorption, or
3. vibration frequency shift of a resonant structure due to water absorption into a hygroscopic coating.

3.5.1 Optical absorption

Water has a well-known optical absorption spectrum in both the infrared and ultraviolet, and trace atmospheric moisture measurements are commonly performed in both wavelength regions. A water or air ingress accident will also result in carbon dioxide and/or carbon monoxide as well as CH_4 in the primary helium coolant. All of these combustion gases have infrared absorption signatures, and the particular wavelength bands selected for monitoring needs to avoid overlap to avoid loss of specificity.

Optical absorption measurement is both rapid and sensitive and can, thus, provide a prompt, unambiguous indication of a major water or air ingress accident or provide a diagnostic indication of small leak initiation. To minimize measurement drift, optical absorption-based measurement is almost always performed in a relative manner, dividing by the simultaneously measured optical absorption at a nearby insensitive wavelength. The measurement sensitivity is largely a function of the optical path length. The effective optical path length can be made large by placing the sample in a Herriott-type optical cavity (see Fig. 15) that multiplies the sample path length through multiple reflections.

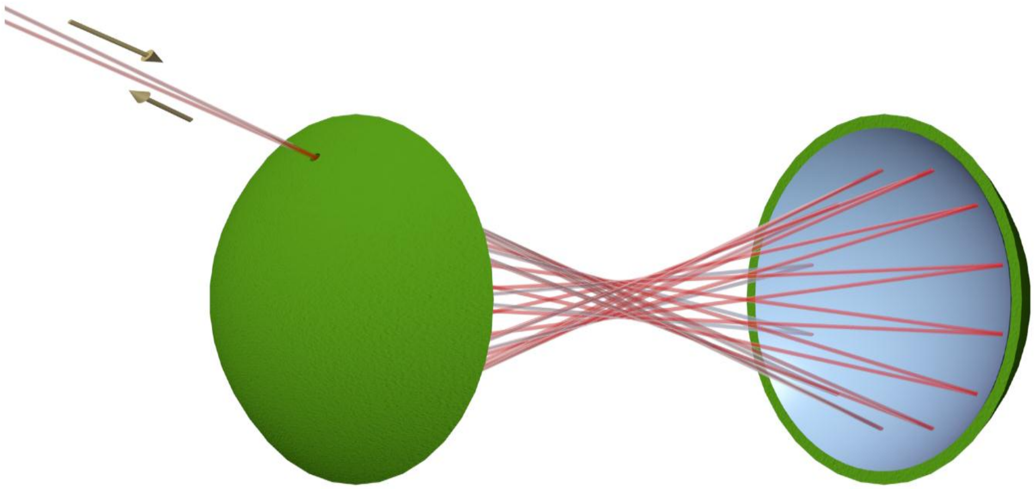

Fig. 15. Herriot-type optical cavity.

Optical absorption spectroscopy can be performed at elevated temperatures and pressures. Performing the optical absorption measurement under primary coolant operating conditions, however, both significantly alters the absorption spectrum and necessitates optical access across the primary coolant pressure boundary. Optical access into an HTGR primary coolant path would either be through a fiber optic port or a small high-pressure window. High-temperature, high-pressure fiber optic ports, that mechanically closely resemble the compression fittings employed for electrical wiring penetrations, are commercially available with either fused silica or sapphire windows. However, as helium is transparent, a small diameter, high-pressure tolerant window can provide optical access across the primary pressure boundary. Sapphire and fused silica windows are commercially available with significantly higher pressure ratings than would be employed in HTGRs. In most optical penetration designs, in order to reduce the temperature on the window's ceramic-to-metal seal, the window is located on a short pipe flange extending out from the high temperature process. While both fused silica and sapphire are transparent in the near infrared and ultraviolet, water's ultraviolet optical absorption does not become large until below available fiber optic cut-off wavelengths (~160 nm). Hence, only the infrared optical absorption spectroscopy appears feasible for an on-line moisture measurement.

Both temperature and pressure alter the water vapor absorption spectrum. By selecting measurement wavelengths that have different temperature and pressure sensitivities, infrared optical absorption spectroscopy can be configured to provide simultaneous, on-line measurement of primary coolant moisture content, pressure, and temperature. Nagali and Hanson[61] have provided an overview of infrared optical absorption-based simultaneous moisture, temperature, and pressure measurement in a high-pressure, combustion-type environment that includes suggested measurement wavelengths (1.30–1.41 μm range). A conceptual layout of a high-pressure, high-temperature tolerant optical absorption-based measurement system is shown as Fig. 16. Note that this type of temperature and pressure measurement system will not work without some included moisture and would, thus, be most useful in postaccident conditions. Adequate system sensitivity would need to be experimentally validated under normal operating conditions. In general, optical absorption-based process condition measurements would be especially useful in postsevere accident environments, since no sensor element would be required to withstand the primary coolant environment, and the measurement electronics can be backed well away from the reactor.

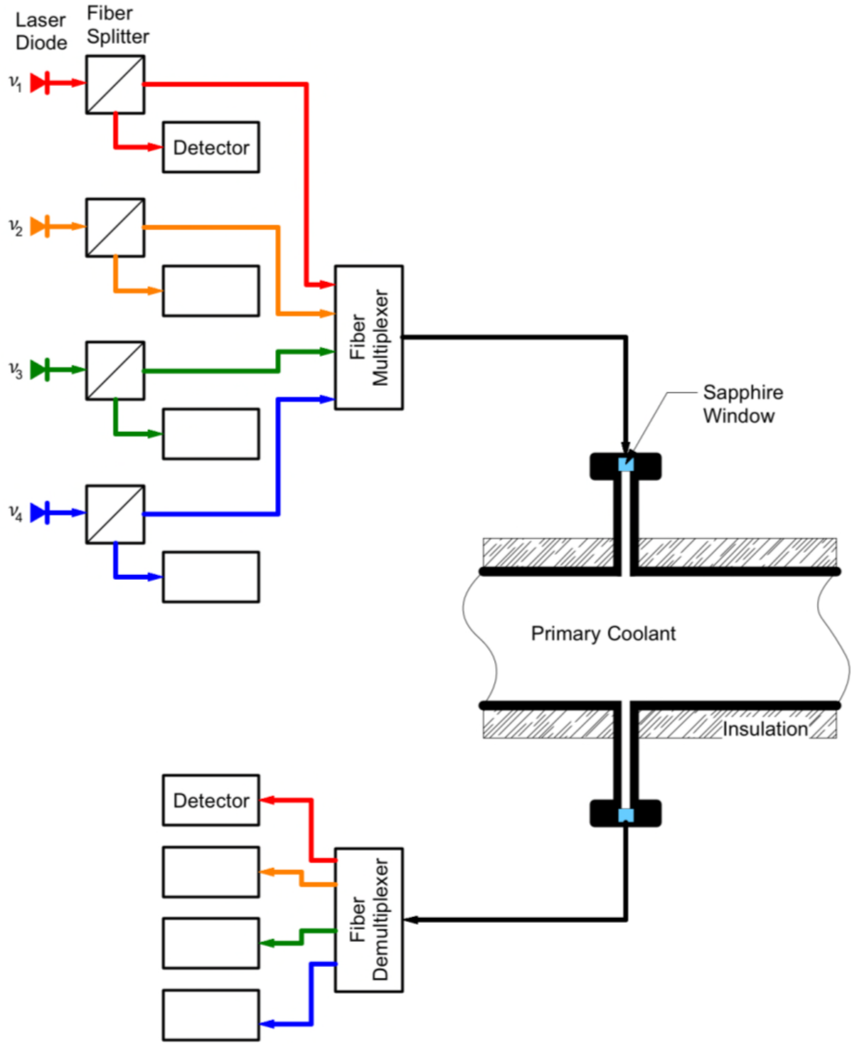

Fig. 16. High-pressure optical absorption spectroscopy component layout.

Optical absorption-based moisture content measurements can also be made on lower-temperature, atmospheric pressure coolant samples. Primary coolant samples would be extracted using a bleed valve attached to a sampling line. At low pressure and temperature, water exhibits several narrow (Angstrom) absorption lines in the infrared. To measure the moisture content of a gas sample, a tunable narrowband laser diode would be tuned across a water absorption line.[62] The optical attenuation at a wavelength just above or below the absorption line would be used to compensate for measurement drift. Moisture is also strongly absorbing at the Lyman–alpha wavelength (121.56 nm), and an ultraviolet optical absorption spectrometer employing fluoride glass windows and a hydrogen discharge lamp is another candidate moisture measurement technology for lower pressure and temperature samples.

3.5.2 Capacitive shift

In a capacitive shift-type moisture detector, water vapor is adsorbed onto the dielectric layer between a capacitor's two plates. The water changes the effective dielectric constant of the layer, which in turn shifts the capacitance value. In the most common implementation format (including the GE General Eastern model deployed at HTR-10), the capacitor is formed by first oxidizing an aluminum plate producing a few micron thick alumina layer. Tubular pores are then formed into the top surface of the alumina layer

(typically by combination of anodic oxidation and chemical dissolution). The pore diameter (few nm) is controlled to preferentially admit water molecules. The pores also significantly increase the surface area available for water adsorption increasing the device sensitivity. A thin (~30 nm) top gold electrode is deposited onto the structure. Moisture diffuses through the thin gold layer and is adsorbed onto the surface of the alumina.

Capacitive shift-type hygrometers are moderately sensitive to pressure and temperature changes necessitating a controlled sample environment and are generally insensitive to combustion gases. Measurement times of a few seconds are possible for high moisture content changes and of roughly a minute for low moisture concentrations. Capacitive-type moisture monitors offer only modest accuracy and repeatability and require periodic recalibration.[63]

3.5.3 Resonant frequency shift

Piezoelectric quartz tuning forks exhibit high-stability resonant oscillation. A moisture sensor can be formed from the tuning fork by coating it with a hygroscopic polymer (e.g., cellulose acetate).[64] Moisture absorption into the polymer increases the tuning fork’s mass thereby shifting its resonant frequency. A matched set of tuning forks is employed to provide drift compensation by maintaining the duplicate turning fork in a dry atmosphere at matched temperature and pressure. While mechanically resonant structures are more common, surface acoustic wave-type humidity sensors have also been built and also function based upon the shift in the acoustic properties of the sensor with water absorption into a surface film. Measurement response time is determined by the adsorption characteristics of the polymer ranging from a few seconds for large shifts to roughly a minute for small shifts. Measurements are sensitive to temperature and pressure necessitating control of the measurement environment. Parts per billion measurement sensitivities are possible with resonant frequency-type measurements.

3.6 Confinement Gases

Measuring the confinement gas composition is primarily of interest following severe accidents and as an indicator of helium leakage. Monitoring combustion by-products both in confinement and within the primary coolant loop provides supports monitoring severe accident progression. Monitoring the oxygen content of the confinement provides an indication of how much additional graphite combustion can occur following a loss of pressure accident. The spatial distribution of the gases needs to be taken account of in the measurement design. The helium will rise to the top of the confinement, and the amount of mixing of the other gases will depend on the specifics of the accident. Likely, multiple measurements of gas composition at multiple elevations will be necessary.

3.6.1 Helium

The resonant frequency of a tuning fork directly shifts with the density of the surrounding gas. As the density of the containment gas directly varies with helium concentration, a gas density measurement provides a surrogate for helium concentration measurement. A matched set of tuning forks would be employed to provide drift compensation. The matched fork would be located in matched pressure and temperature air or nitrogen environment (i.e., within a sealed, helium impermeable bellows) providing compensation for the confinement pressure and temperature shifts. Measurement response time of resonant frequency devices is short and 0.2% system sensitivity has been demonstrated.[65]

3.6.2 Combustion products

Monitoring the carbon dioxide and carbon monoxide in a post-depressurization accident would provide indication of the amount of air ingress as well as a general tracking of the accident progression. Measuring the ratio of CO_2 to CO would give information about where in the core the oxidation is taking

place. Combustion diagnostics at both high and low pressure across a wide range of temperatures can be performed on-line using infrared optical absorption spectroscopy. Both carbon monoxide and carbon dioxide can be observed simultaneously along with moisture. Infrared absorption measurements can be quite mechanically and thermally robust with measurements being made in atmospheres as aggressive as a coal-fired power plant combustion chamber[66] and a full sized natural gas burner.[67] The instrumentation necessary to perform the measurement is much the same as that necessary for the infrared optical absorption-based moisture measurement described earlier. Infrared absorption measurement times are small and typical measurement sensitivities are ~100 ppm. Performing combustion measurements within the primary coolant boundary (or in-core) would require assuring optical access across the primary coolant pressure boundary in a post-severe accident environment. Monitoring of combustion products in confinement can be performed with reasonably available, well-known hardware.

3.6.3 Oxygen

The confinement environment following a severe accident may contain several gas species as well as moisture droplets and carbon particles. Maintaining specificity of the measurement of a mixed, aggressive environment is key to selecting the most advantageous measurement technology. Infrared optical absorption spectroscopy is especially effective at selecting out oxygen from the interfering species. All of the other conventional oxygen analyzer technologies have some cross sensitivity to interfering gases that may be present in confinement in a post-severe accident scenario. A molecular oxygen absorption band near 760 nm has been shown to provide 0.1–1% oxygen measurement under fire-suppression conditions.[68] The measurement configuration for a fiber optic coupled, tunable laser diode, oxygen infrared absorption spectroscopy measurement system closely resembles that for either moisture or combustion product measurement.

3.7 Tritium

Tritium contamination is regularly measured as part of nuclear plant environmental release monitoring. Measurements can readily be made in helium, air, and/or water using well-established techniques. The "NGNP Contamination Control Study"[69] does not describe any anticipated difficulties with measuring tritium, and classical instrumentation is anticipated to be suitable for tritium measurements at next generation HTGRs.

Tritium concentration measurement in helium would most readily be performed by placing the helium sample into an ion chamber or flow through type proportional counter. If additional contamination with other radioactive species is suspected, the helium sample would first need to be passed through a liquid nitrogen cold trap to plate out the radioactive noble gasses. An instrument for the measurement of tritium within air would be similar to one for measuring tritium in helium. Established procedures (filtering, electrostatic precipitation, etc.) need to be followed for compensating for smoke, aerosols, moisture, and/or gamma-ray fields.[70] In a high background area, such as an HTGR containment following a severe accident, a gamma-insensitive flow-through ion chamber would likely be required for an accurate tritium concentration measurement.[71]

Tritium measurement in water would most likely be performed by dissolving the sample into a liquid scintillation cocktail and measuring the resulting scintillation light using a photomultiplier tube. Both environmental and medical tritium contamination monitoring in aqueous solutions is commonly performed using this scintillation cocktail method.[72] Alternatively, the water suspected to contain tritium can be electrically separated into hydrogen and oxygen and a flow through type ion chamber employed to count the tritium in the hydrogen.

3.8 Fuel Qualification

The particular TRISO fuel intended for use at the NGNP is currently undergoing qualification testing through the DOE Advanced Gas Reactor (AGR) program.[73] In-fabrication, in-core, and post-exposure testing are all underway. Measuring the parameters of the fuel qualification environment is technically challenging due the high levels of radiation, high temperatures, and small spaces available. Several of the sensors described in the preceding sections are likely candidates for deployment for fuel qualification. Most fuel property measurements are made by post-irradiation examination using classical laboratory techniques. Similarly, the fuel test conditions are confirmed off-line using melt and flux wires. However, irradiation temperature and neutron flux are typically measured on-line. The micropocket fission chambers described previously may enable improved local neutron flux measurements as compared current self-powered neutron detector techniques with their known temperature sensitivity.[74]

Johnson noise thermometry has been shown to be advantageous for fuel temperature measurements in-core since it is fundamentally immune from radiation-induced shifts in the sensor value.[75] While type-N thermocouples are generally stable and reliable temperature measurements to 1000°C, fuel qualification temperatures often are performed at higher temperatures to simulate accident conditions. A particular challenge for fuel qualification thermocouples is the small diameter wire required. The small wire diameter increases the technical challenge of the wire drawing, increases the surface area to volume ratio making the wires more vulnerable to corrosion, and makes the wires more vulnerable to grain growth across the thin wires. Additionally, the electrical resistivity of insulators decreases by about one order of magnitude per hundred degrees above 1000°C. The 0.1MoPt–5MoPt thermocouple developed by the Dragon reactor project in support of irradiation experiments would also be useful for fuel qualification testing at temperatures up to 1700°C (Ref. 76). Molybdenum/niobium alloy thermocouples have also demonstrated good high-temperature, high-neutron flux performance.[77] However, niobium and its alloys are particularly vulnerable to oxygen-induced embrittlement (at concentrations of as low as 1 ppm).[78] Consequently, radiolytically liberated oxygen from the oxide electrical insulators is a concern for long-term thermocouple survival. Also, pure niobium tends to exhibit grain growth and consequent wire cracking. Nb–1Zr alloy is commonly used to avoid the rapid grain growth. Additionally, molybdenum becomes brittle at ambient temperature after being exposed to high temperatures, which can be a concern for welded thermocouple junctions or cyclic testing. The recent use of lanthanum oxide dispersion strengthened molybdenum has decreased the thermocouple vulnerability to molybdenum loss of ductility.[79]

3.9 Control Rod Drive Instrumentation

Due to the tall HTGR core format, full-height control rods do not fit within the primary vessel when extracted from the core. In order to avoid needing to have the control rods penetrate through a pressure boundary, the rods retract partially into standpipes located above the vessel. To minimize the required standpipe height, the control rod leaders need to be either flexible, telescopic, or both. In FSV the control rods were attached to a cable and drum drive system with the control rod motors located at the top of the standpipes to lower the motor temperature. The rods had an electromagnetic clutch that could be depowered to allow the rods to fall into the core under gravity.[80]

Both control rod position and cable tension need to be monitored to provide evidence that the control rod is at its specified position and that it remains capable of moving when called upon. For a cable and drum system, the simplest form of position sensor is an absolute shaft encoder. Both electromagnetic and optical absolute shaft encoders are widely available. Alternatively, the amount of cable remaining on the drum can be measured using capacitive gauges. The position of the top of the control rod could also be directly measured using coherent laser radar with optical access to the top of the standpipe provided through a high-pressure window. Cable tension is measured to demonstrate that the control rod is free to move. Too low of a tension indicates that the rod is resting upon an obstruction, and too high of a tension

indicates that the movement system is wedged and the rod may not drop when released. Cable tension is measured by monitoring the amount of force necessary to deflect the cable using a pulley and strain gauge apparatus. Similar information can be derived from shaft torque sensors on the drive motor. Several different style torque gauges are commonly used in industry.

4. DIAGNOSTIC AND PROGNOSTIC TECHNOLOGIES FOR HTGRs

Advanced instrumentation technologies can provide information necessary to assess a plant's current "health" and predict the remaining useful life of the plant's SSCs. Diagnostics and prognostics techniques, however, are becoming ubiquitous in industrial systems. Some diagnostic techniques, such as lubricating oil condition and contamination monitoring, have been widespread for several decades now throughout industry. This section provides a descriptive overview of the instrumentation and methodologies for diagnostics and prognostics as they might apply to HTGRs and does not attempt to address the more general issues associated with applying the technologies to nuclear power plants in general or other industrial SSCs.

Active components are particularly amenable to integration with instrumentation in that the sensor data obtained can support control as well as diagnostics and prognostics. Overall, integrating advanced controls and instrumentation deeply into nuclear power plant components offers the potential to profoundly improve their capabilities, performance, and maintainability.

The recent reports by PNNL staff provide a good general foundation on the application of instrumentation to component and structure health and remaining useful life estimation.[81–83] Material prognostics are especially important at HTGRs since at the higher temperatures the structural alloys have a greater tendency to creep under load, and careful monitoring and consequent adjustment of operating conditions can significantly extend the life of expensive, difficult to replace structural components.

Diagnostics and prognostics techniques can also be applied to the instrumentation system itself to increase confidence in the measurement values and to provide an estimate of remaining useful life. For example, an increase in the measured noise on an instrument's signal likely indicates a degradation of the sensing channel rather than a process variable shift. The recent textbooks by H. M. Hashemian provide a good review of use of signal variance as a measure of instrumentation system health.[84,85]

4.1 Helium Circulator

The helium in the primary circuit of an HTGR needs to be kept very clean, and conventional oil lubricated bearings are a potential source of contamination. Also, bearing maintenance within the primary pressure boundary is both expensive and time consuming. The helium circulator for FSV was the source (especially its seals and bearings) of much of the plant reliability problems.[86] Modern HTGRs are consequently likely to adopt active magnetic bearings. The HTR-10GT helium circulator has already made this design choice employing high-temperature, active magnetic bearings controlled using a multiaxis, high-speed feedback scheme.[87]

The development of high-temperature materials technologies over the past few decades was a key enabling technology for instrumentation to more intimately couple into the helium circulator to support active control. Additionally, development of high-speed, fault-tolerant computing and controls technology was required for active management of circulator performance to become possible; this technology was unavailable for the HTGRs of the 1960s and 1970s.

The basis of active magnetic bearing suspension is varying the drive current to electromagnets as a result of rotor displacement measurements. Shaft horizontal and vertical positions are independently measured at each radial bearing-motor set. Rotor position measurement is made by monitoring the change in the resonant frequency of a driven coil located near the rotor due to the shift in position of the magnetic rotor material or with a Hall effect-type sensor.

An HTGR helium circulator shaft will rotate rapidly under normal, at-power conditions. The combination of turbulent fluid motion and rapid impeller rotation results in multi-mode shaft vibrations with frequencies up to roughly 10 kHz. Complicating the measurement interpretation, even minor

imperfections in the impeller balance or in the rotor position sensor targeting can result in rotational instability (i.e., the control system itself can enhance the inherent oscillations).

The relative and absolute motions of the circulator rotor are the primary measured parameters for both the control and diagnostics system. Rotor rotational characteristics are computed from the measured force and position data. Flaws within the circulator structure or operations are impressed onto its rotational characteristics. System models are used to correlate the measured rotational characteristics with a particular flaw. Once a flaw has been identified, the control system is called upon to adapt to degraded operating conditions. A prognostic model for flaw progression can also then be employed to predict the component remaining useful life. Fault tolerant active magnetic bearing control design remains an active development area[88] as does active magnetic bearing-based fault detection in centrifugal pumps.[89]

Fault tolerant control can be combined with system diagnostics to improve the overall circulator reliability. For example, the circulator active magnetic bearing control would typically be powered by three-phase power supplies. For any one radial bearing, each phase could control the rotor position along axes separated by 120°. Since position control only needs to be supplied on the x-y axes, in any one radial bearing control, any one of the power supplies is redundant. Thus, if one of the power supplies were to fail, the other two phases of the magnetic bearing control power supply system could maintain control of the rotor position, provided that they were sized appropriately and that the I&C system could respond rapidly enough to recognize the problem. This would both avoid a rotor crash as well as enabling on-line system repair by hot-swapping a working third phase into the power supply rack.

4.2 Control Rod Drive Mechanism

HTGR control rod drives are substantially different than those for LWRs. The taller core and desire for a less than full core height standpipe above the vessel combined with the lower density and compressibility of the coolant combine to create substantially different design requirements. As the environment at the top of the standpipe, where the control rod drive motor is located, is inert and relatively low temperature; the motor, clutch, drum, and instrumentation can be fabricated from traditional materials. As Electrical Signature Analysis (ESA) can be applied without deploying additional instrumentation outside of a well-controlled environment, it is the most likely technique to be employed to observe the onset of mechanical changes to the control rod drive system.[90] The central premise of ESA is that mechanical changes to an electrically powered system, such as clutch slippage or changes in shaft torque (rod jamming), will be impressed onto the higher frequency components of the motor current and voltage. As such, ESA is a general diagnostic technique (that was originally developed at ORNL under NRC sponsorship[91]) with widespread applicability to any electrical motor operated component.

4.3 Noise Analysis

Noise analysis has been applied to a wide variety of reactor designs, and the results have sometimes been useful in obtaining information about system operation that would otherwise not be available. Perhaps the most common current use deals with monitoring vibrations of rotating equipment, where changes in the noise spectrum characteristics can be attributed to various degradations, such as bearing wear, and can usefully be applied to maintenance or replacement scheduling. Likely candidates for this application in HTGRs would be for the control rod drive motors (noted in Section 3.2) and for circulators (or the gas turbine for the Brayton cycle designs). Today's noise analyzers are fast, cheap, reliable, and abundant.

The signal typically obtained from noise analysis is the product of two elements—the source of the noise and the characteristics of the process or system (and sensor) between the source and the measurement. For example, if bubbles in a reactor core coolant are the noise "source," the ensuing reactivity fluctuations and their effect on the neutron population (and the flux sensor) are the "filter" between the source and the

measurement. The filter characteristics are functions of the void coefficient of reactivity and other thermal-hydraulic processes. In some cases then, it may be difficult to "back out" the desired information.

One useful variation on noise analysis makes use of forced perturbations on the system, which then can allow the analysis to characterize the measured response as system behavior without the uncertainty of the size or nature of the perturbation. Such tests can usually be done without significant "interference" with system operation (which is often a requirement in commercial power plants), since this information can usually be obtained with small perturbations. For example, tests have been run on operating PWRs using small variations in control rod and steam valve positions with time. The resulting information is in the form of the frequency response of the system. Examples are the reactivity-to-power and reactivity-to-core outlet temperature characteristics. These are linearized functions and will typically change with such things as power level, coolant flow, and burnup. By comparisons of the measured frequency responses with the theoretical models, useful information can be obtained to help with model validation. Because of the differences between the response times of various phenomena, which can be seen in frequency response results, parameters can be inferred that would not be evident in the analysis of steady state measurements.

The use of several types of perturbation signals allows for tradeoffs in experimental difficulty and accuracy of the information obtained. The use of pseudo-random binary sequences (PRBS) has the advantage of generating perturbations with many known distinct frequencies, where at these frequencies the signal-to-noise ratio would be high (and thus, hopefully, the results "clean"). A comprehensive study and implementation of these options was performed for the ORNL Molten Salt Reactor Experiment (MSRE).[92]

Historically (in HTGR history, at least) the most unusual case of reactor noise occurred in 1979 at Fort St. Vrain, where large periodic variations were observed in core coolant outlet temperature and flux detector signals (see Figs. 17 and 18).[93] The mystery was solved when it was determined that the cycles were due to fuel blocks tilting (back and forth). The cyclic nature was due to thermal expansion and cooling variations from bypass flows from adjacent blocks, along with varying radial pressure drops. The effect in FSV was exacerbated by differences in core regional flow orifice settings, which resulted in large lateral pressure drops. Current modular HTGR designs do not have regional orifices, so the lateral pressure drops would be small; however, in NGNP designs, the active core is ten blocks high (vs six high for FSV), making it more prone to wobbling. The newer cores have design features (e.g., supports) to inhibit wobbles.

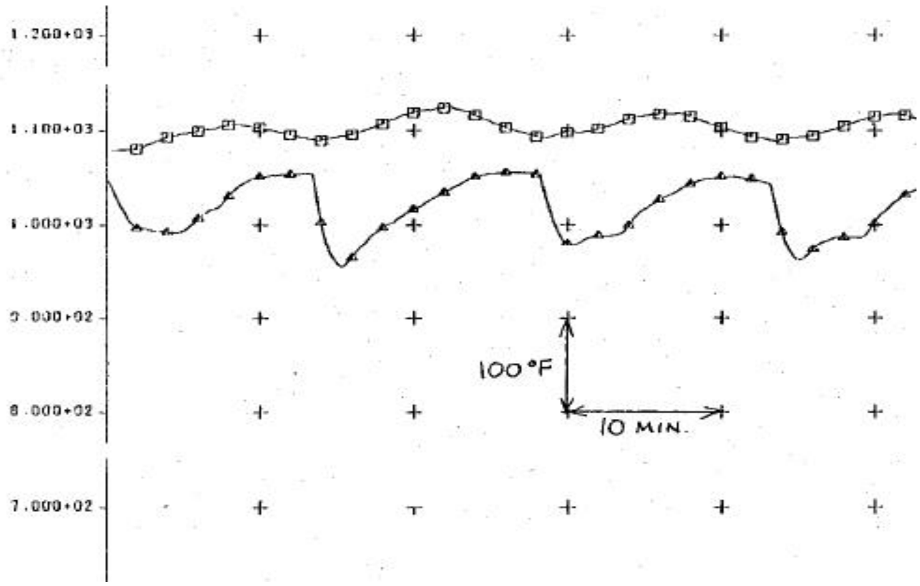

Fig. 17. Core coolant outlet temperature perturbations at Fort St. Vrain.

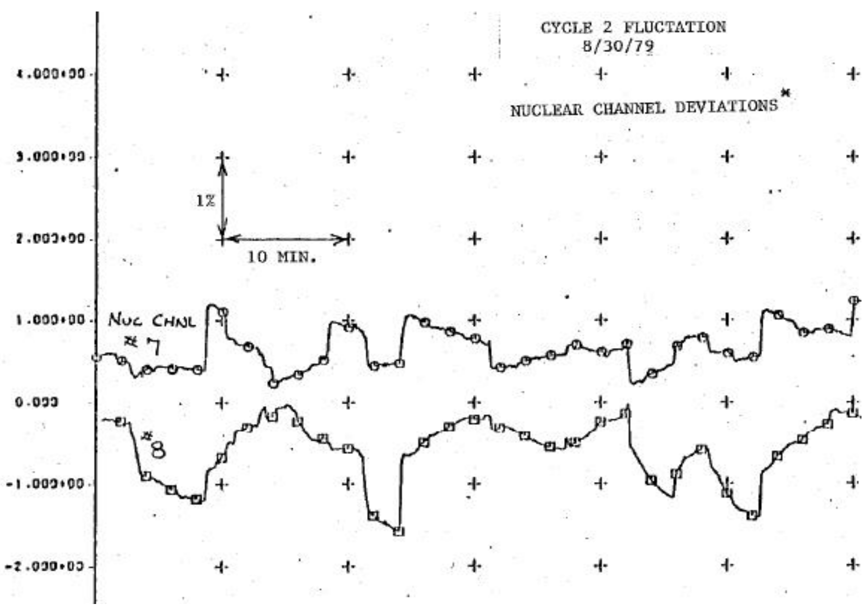

Fig. 18. Flux channel perturbations at Fort St. Vrain.

4.4 Fission Product Escape

Some of the fission products will escape the TRISO fuel. The degree to which FPs leak is a potentially important diagnostic both to ensure the quality of deployed fuel as well as to ensure that the primary coolant remains sufficiently clean such that a depressurization accident does not excessively contaminate

the confinement, filter system, or environment. While some of the FPs released from the fuel will adsorb onto the nearby graphite, the released FPs may also be transported in the graphite dust suspended in the primary coolant or as a vapor mixed in with the primary coolant.

Dust particles can be removed from the primary flow using electrostatic precipitators, much as coal ash is removed from the flue gas emerging from coal combustors is stripped using electrostatic precipitators.[94] The gamma intensity and spectrum of the deposited particulate layer would then serve as an indicator of FP leakage. Iodine-131 and ^{133}Xe will both circulate as gases in the primary coolant. The ^{133}Xe will be trapped in the cold trap of the helium cleanup system. The ^{131}I will tend to plate out on the cold surfaces. However, the boiling point of iodine is only 184.3°C. Consequently, a significant amount of the iodine will evaporate and circulate. Iodine will also be removed by the cold traps in the helium cleanup system. Measuring the radionuclide content of the helium cleanup system cold trap is, thus, a measure of the fuel failure fraction.

Cesium has a higher vapor pressure than most other FPs would preferentially transport in vapor phase. The boiling point of cesium is 671°C. Consequently, the cesium would tend to plate-out either in the colder sections of the steam generator or in the cold leg. Performing gamma detection and spectroscopy on the inlet section of the cold leg would provide an indication of the amount of cesium leakage from the fuel. The cesium may form a friable layer on the surface of the cold leg and, thus, be releasable during a severe depressurization accident. Consequently, measurement of the cesium concentration plated onto the cold leg would be a useful measurement of whether the coolant contains an unacceptable level of contamination.

4.5 Dust Diagnostics

The initial letter report in this series1 summarized the dust measurement issues in HTGRs (primarily PBRs) noting that from a safety standpoint, the concern is mainly about how much contaminated dust would be released to the environment in rapid D-LOFC accidents. For low-pressure containment designs, a rapid discharge of the primary helium would exit the building unfiltered and subsequent discharges would pass through filters before exiting.

Recently a comprehensive assessment of the dust problem took place at the Very High Temperature Reactor Dust Assessment Meeting held in Rockville, MD. The objective was to identify and prioritize the phenomena and issues that characterize the effect of carbonaceous dust on high-temperature reactor safety.[95] The major R&D needs were identified as (1) generating adsorption isotherms for fission products that display an affinity for dust, (2) investigating the formation and properties of carbonaceous crust on the inside of high-temperature reactor coolant pipes, and (3) confirming that the predominant source of dust is the abrasion between fuel spheres and the fuel handling system (for PBRs).

A major effort addressing the dust problem is also taking place in Europe. The study is being done as part of the RAPHAEL project.[96] The T/H code SPECTRA is used to model the dust behavior, and 200 kg was estimated as the primary system inventory in PBMR after 40 years of operation. The graphite dust generated in an HTR/PBMR during normal reactor operation would be deposited inside the primary system and become radioactive due to sorption of fission products. Doses were predicted to be strong functions of particle size distributions, with most deposition occurring in the recuperator (hot side) and with higher doses for smaller particle sizes.

5. SUMMARY AND CONCLUSIONS

This report takes a broad view of the instrumentation systems for a modular HTGR, characterizing them as nerve centers for implementing safe and efficient plant operation. To this end, the I&C requirements, especially noting differences vs LWRs, are elaborated upon with background discussions of plant behavior for both normal and postulated accident conditions.

Emphasis is put on new capabilities for measurements likely to be employed in an NGNP design, noting how these could be used along with on-line simulation analyses to learn more about details of event consequences that would not otherwise be available. Such information would be very useful in guiding any potentially beneficial remedial action.

One section (2) is dedicated to a discussion of a modern instrumentation architecture that takes advantage of new computation and network systems coupled to the HTGR with its unique response and safety characteristics.

Section 3 describes advanced sensors and other instrumentation with potential application to HTGR systems. Section 4 describes potential use of advanced diagnostics that would be beneficial to HTGR operation.

It can be concluded that applications of new and emerging instrumentation and networking systems can be beneficial to the safe operation of modular HTGRs with NGNP design features.

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