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Letter Report

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

Contributors: S. M. Cetiner, D. E. Holcomb, and S. J. Ball
Oak Ridge National Laboratory

JCN N6668, Advanced Reactor Instrumentation
NRC Technical Monitor: Yaguang Yang

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Prepared by
OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37831-6165
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CONTENTS

LIST OF FIGURES

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LIST OF TABLES

Table **Page**

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.

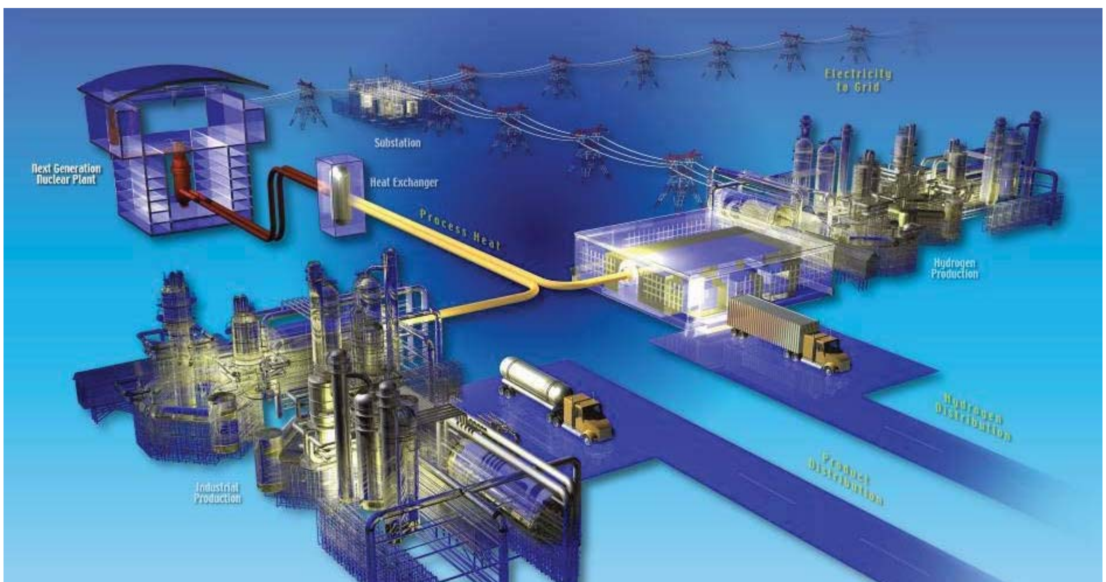


Fig. 1. Schematic drawing of the Next Generation Nuclear Plant with various industrial services.
[Adapted from Ref. 3]

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.

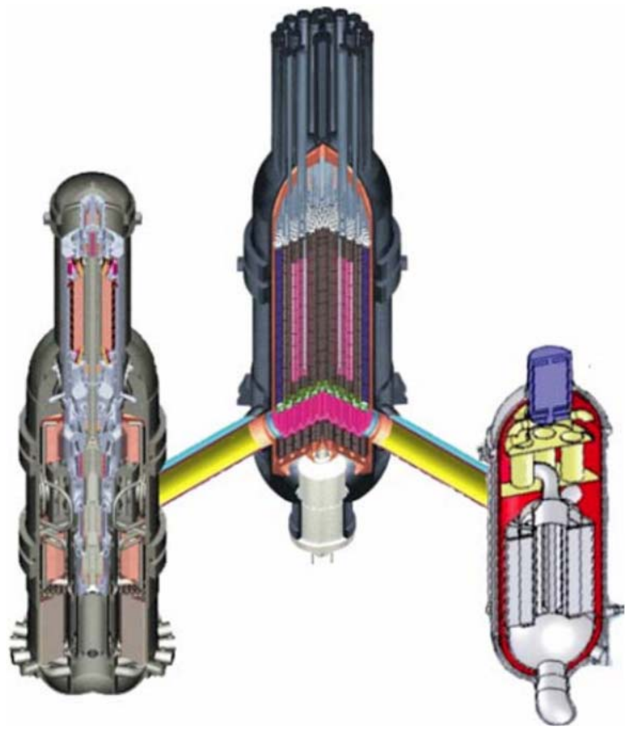


Fig. 2. General Atomics earlier proposed NGNP configuration.

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.

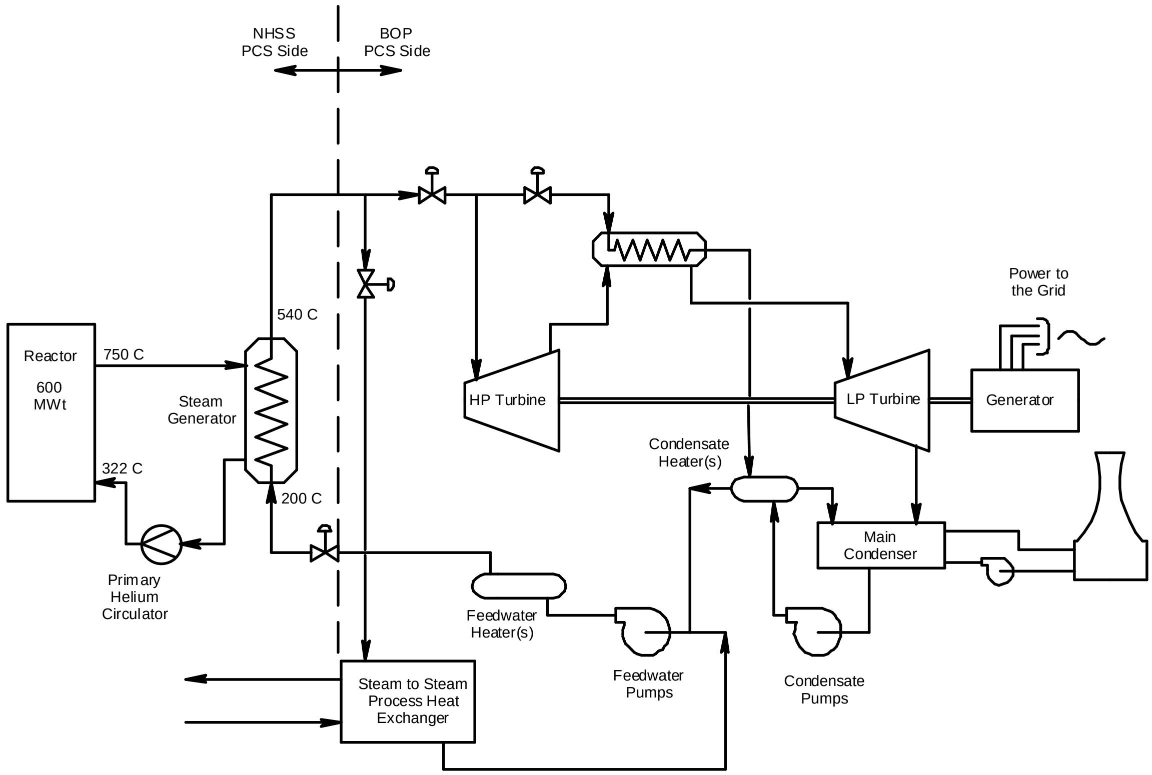


Fig. 3. Schematic of the NGNP configuration considered by GA.
[Adapted from Ref. 5]

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.

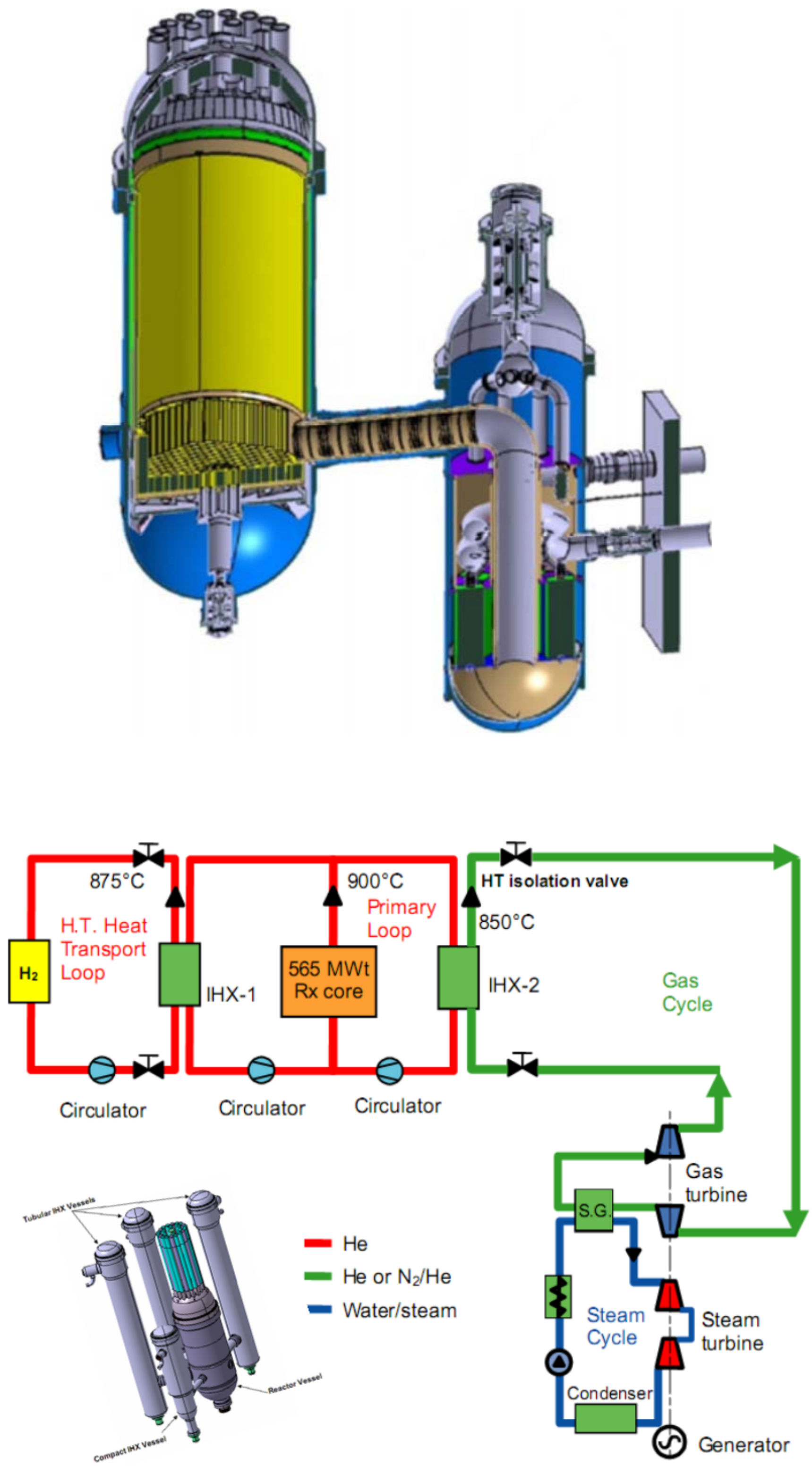


Fig. 4. Original AREVA NGNP design.

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.

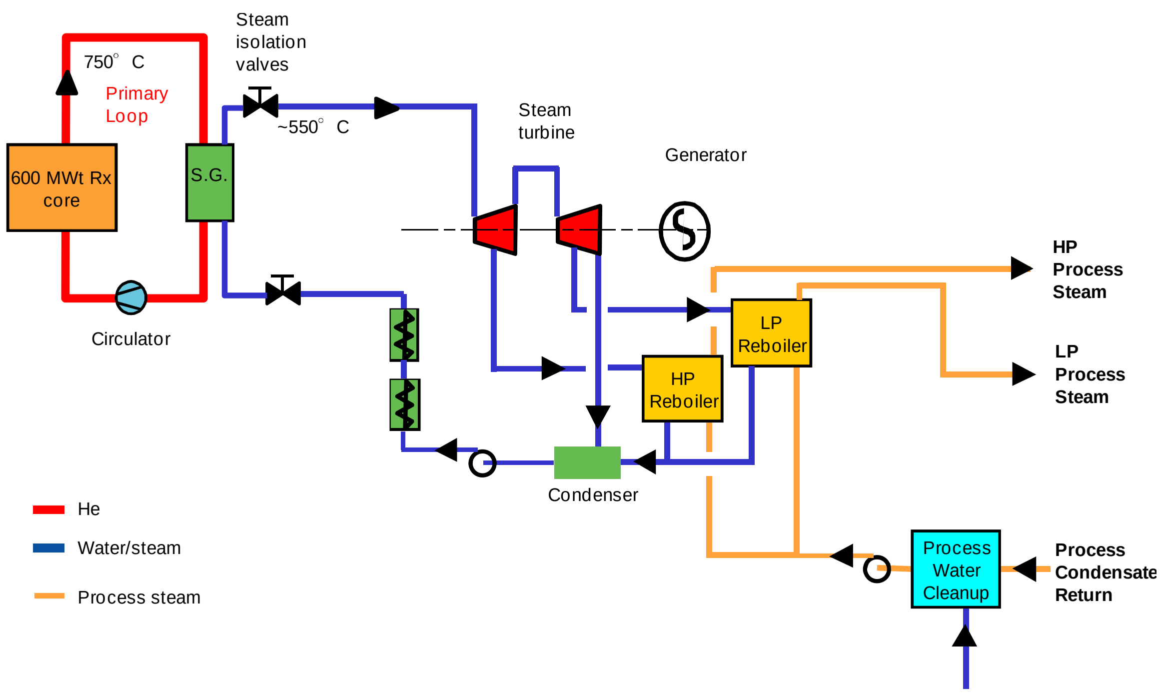


Fig. 5. Recent NGNP flowsheet proposed by both AREVA and GA.
[Adapted from Ref. 2]

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.

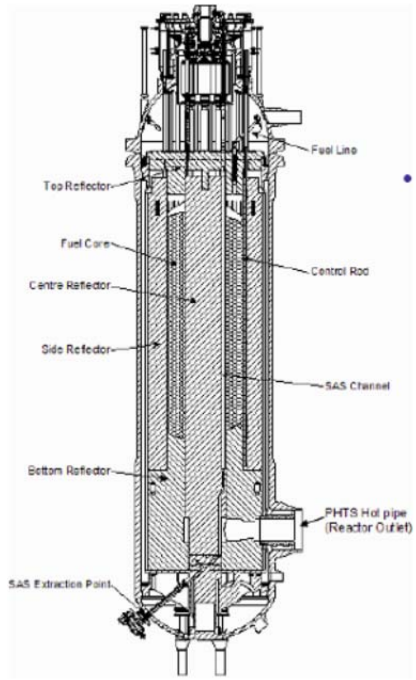


Fig. 6. Original Westinghouse/PBMR (Pty) Ltd. team design.
[Adapted from Ref. 4]

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.

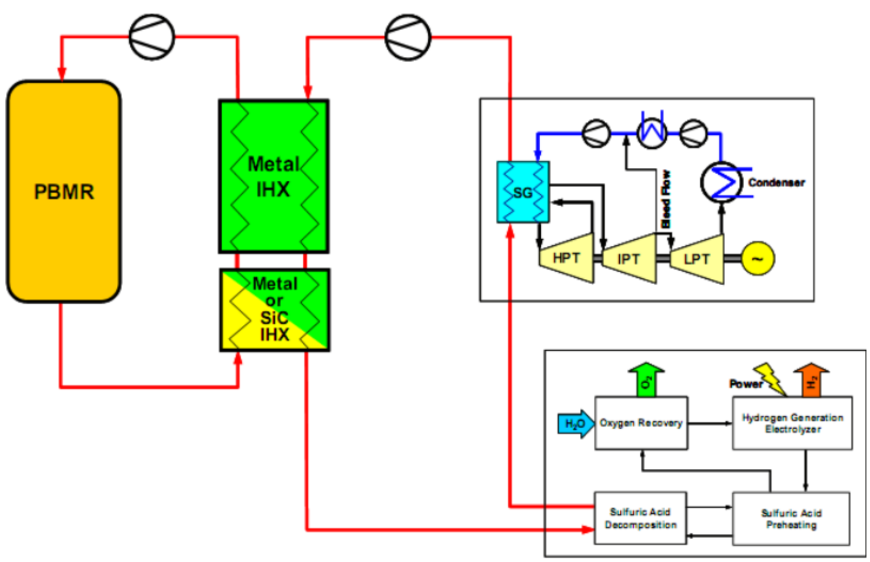
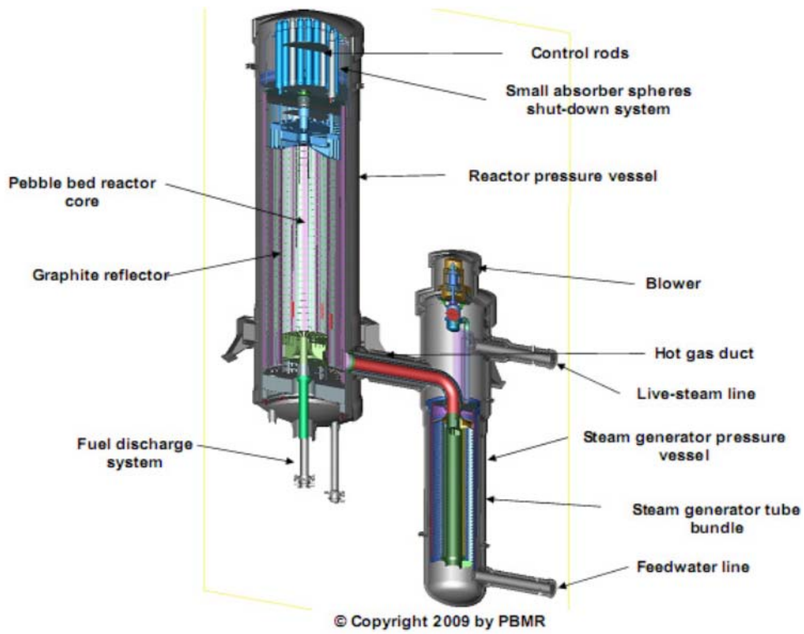


Fig. 7. Flowsheet for the original Westinghouse/PBMR (Pty) Ltd. team design.

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]

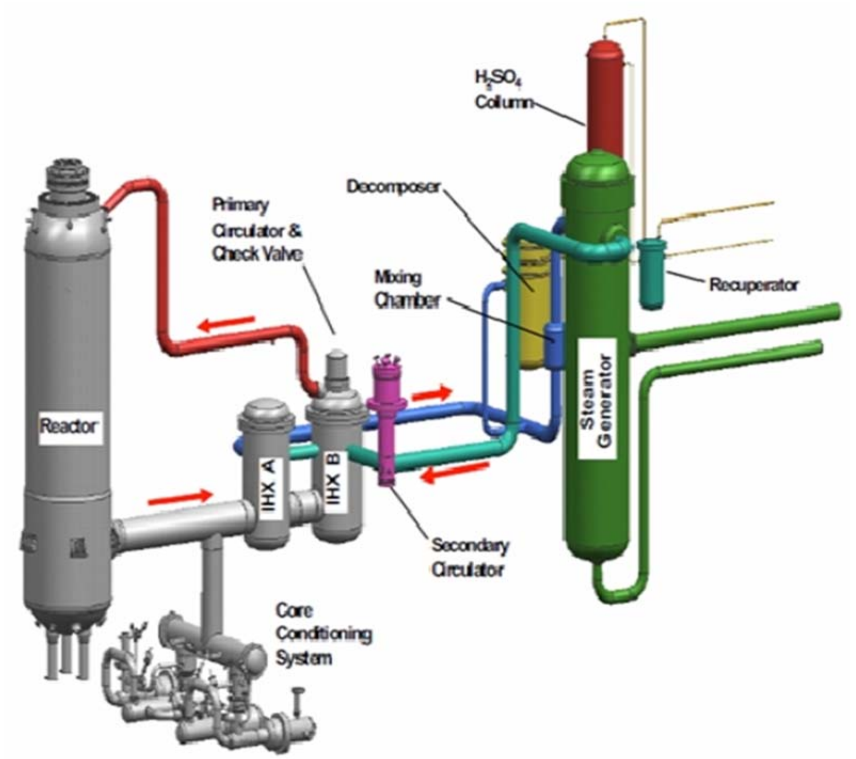


Fig. 9. Westinghouse/PBMR team design with an indirect configuration for electricity and hydrogen production. [Adapted from Ref. 2]

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.

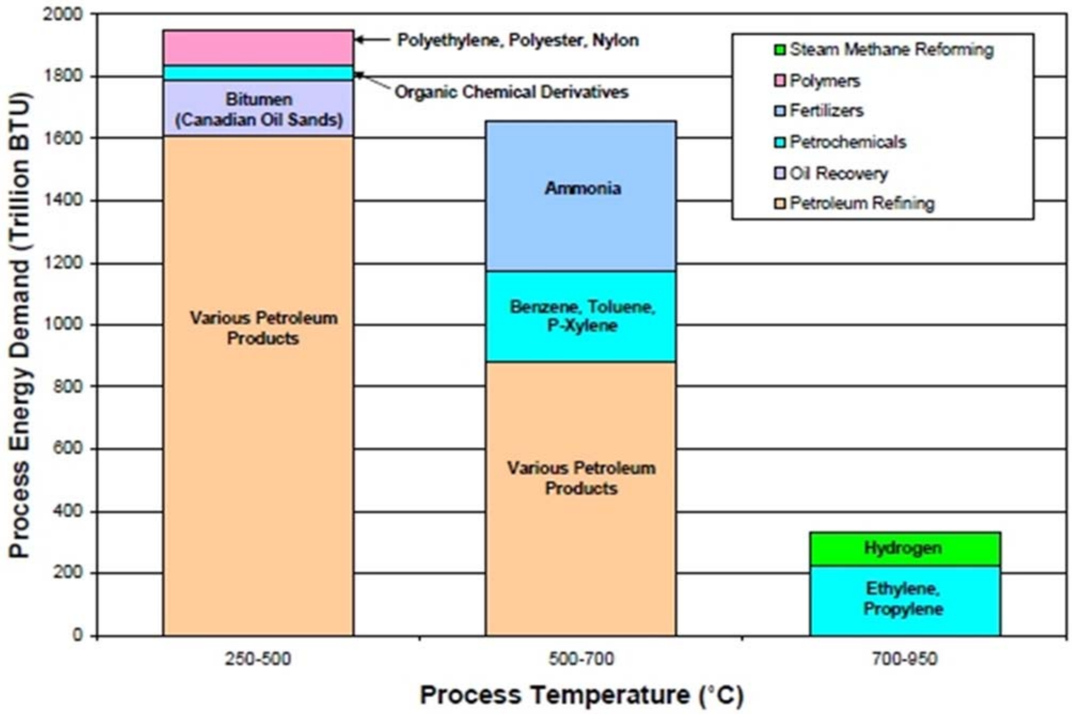


Fig. 10. MPR survey of industrial needs (in the United States) for high-temperature process heat.

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

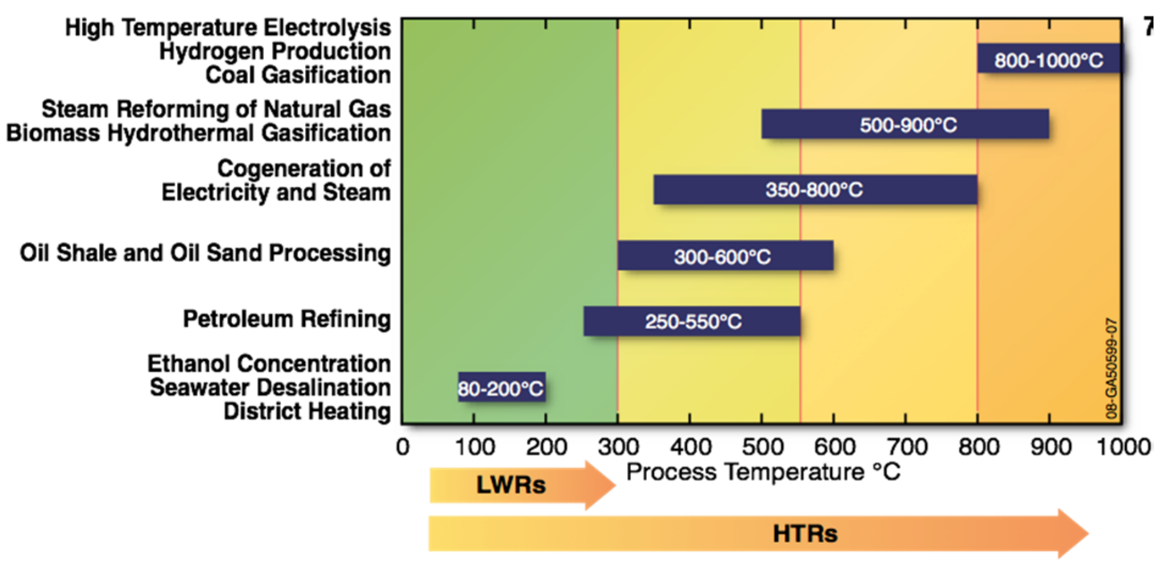


Fig. 11. Summary of temperature requirements for potential end users of HTGRs.

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.

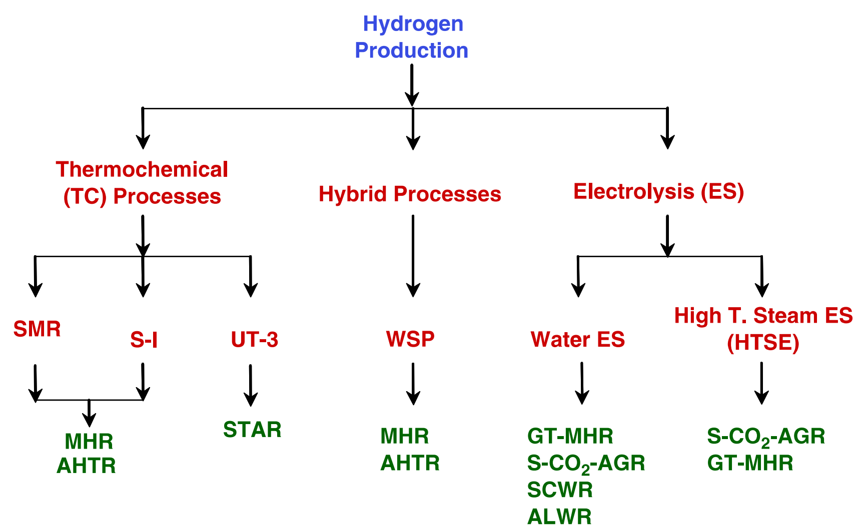


Fig. 12. Technology options for nuclear hydrogen production.

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.

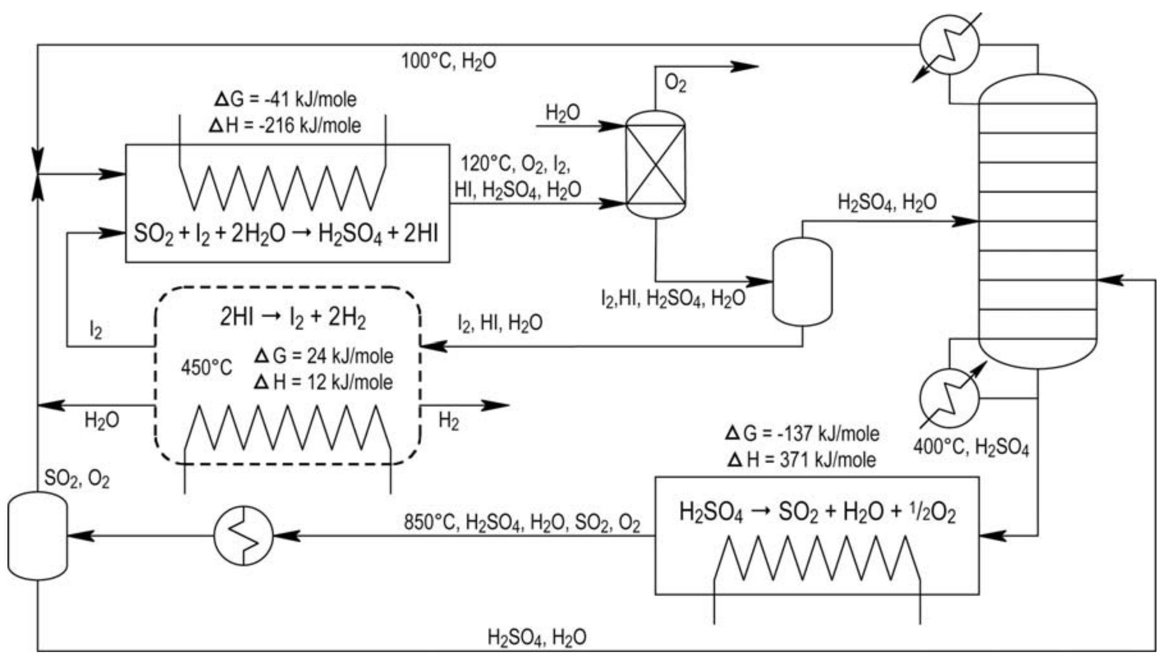


Fig. 13. Simplified schematic for sulfur iodine water splitting flowsheet for hydrogen production.

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.

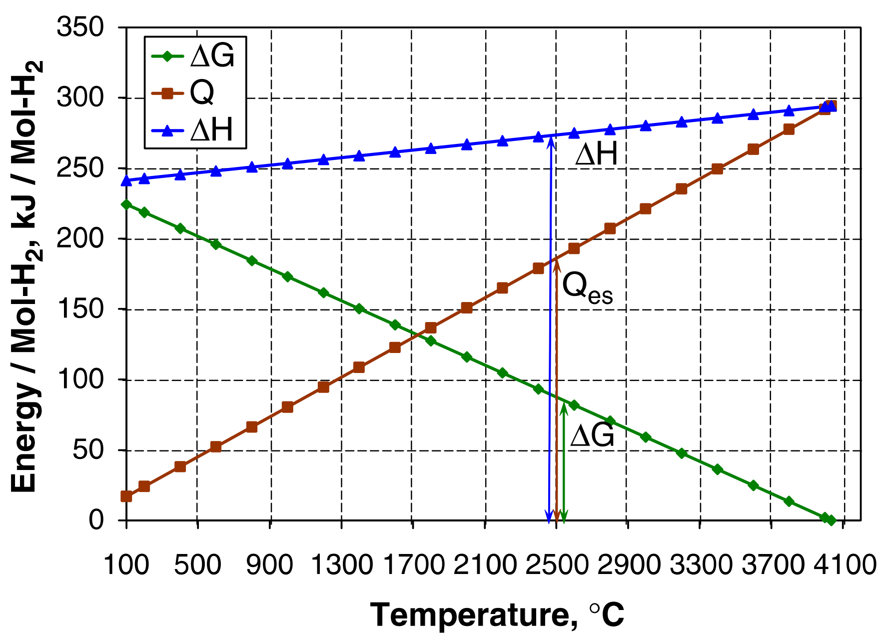


Fig. 14. Energy required for steam electrolysis.

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.

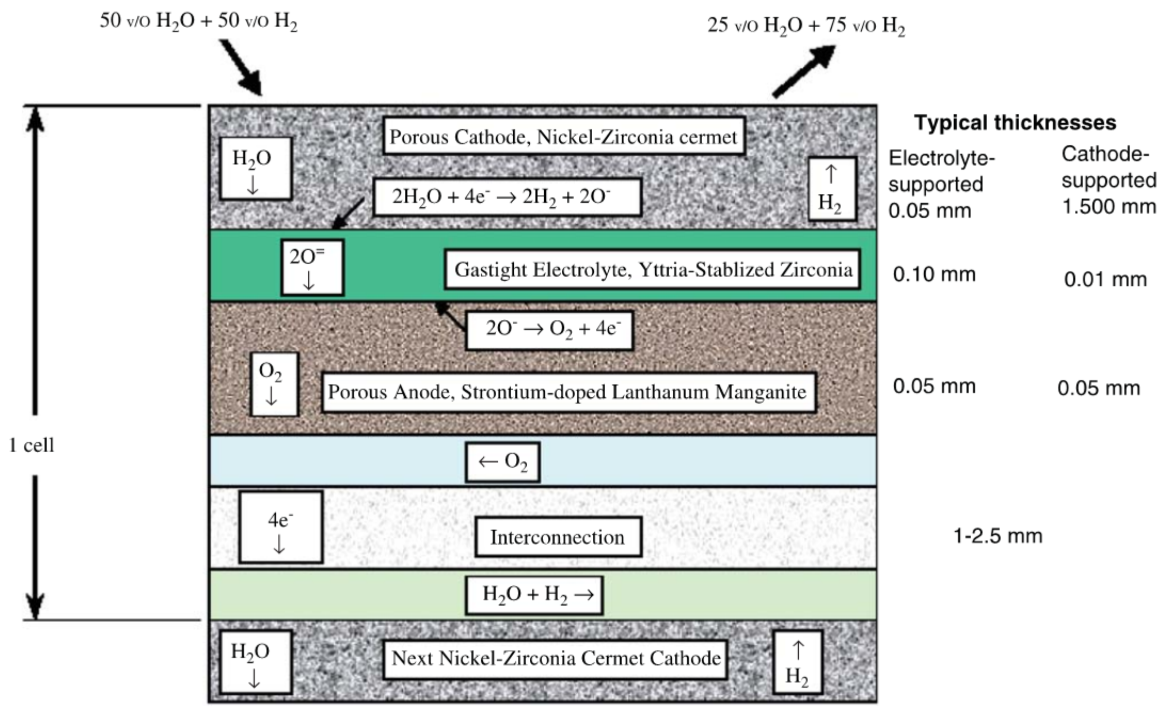


Fig. 15. Schematic for representative solid oxide electrolysis cell.

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

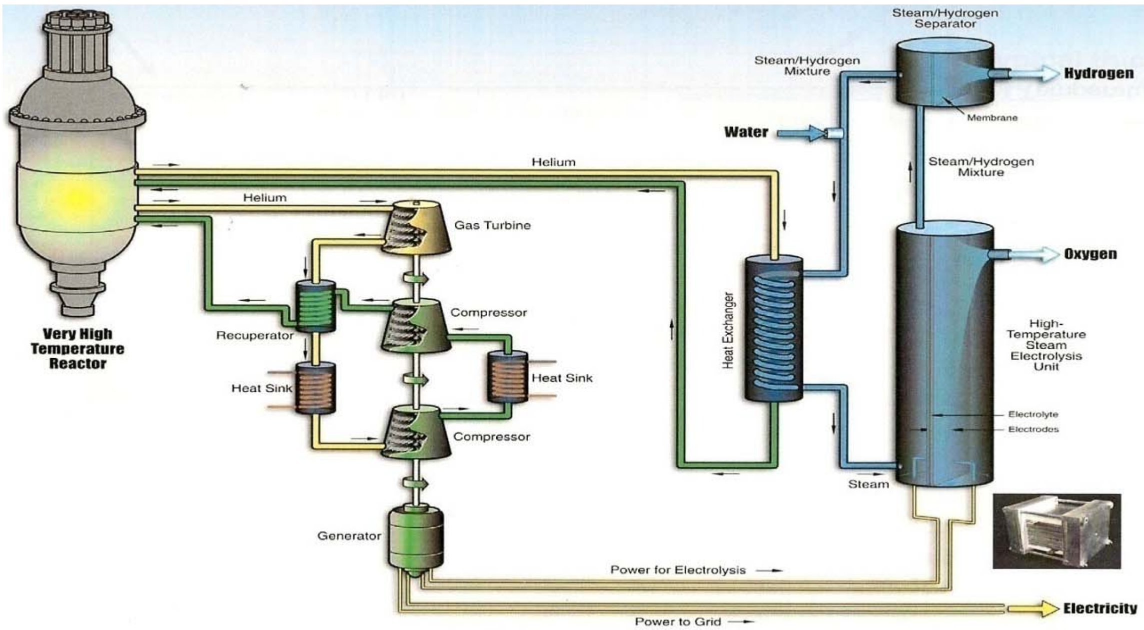


Fig. 16. Schematic of the high-temperature steam electrolysis (HTSE) plant.

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.

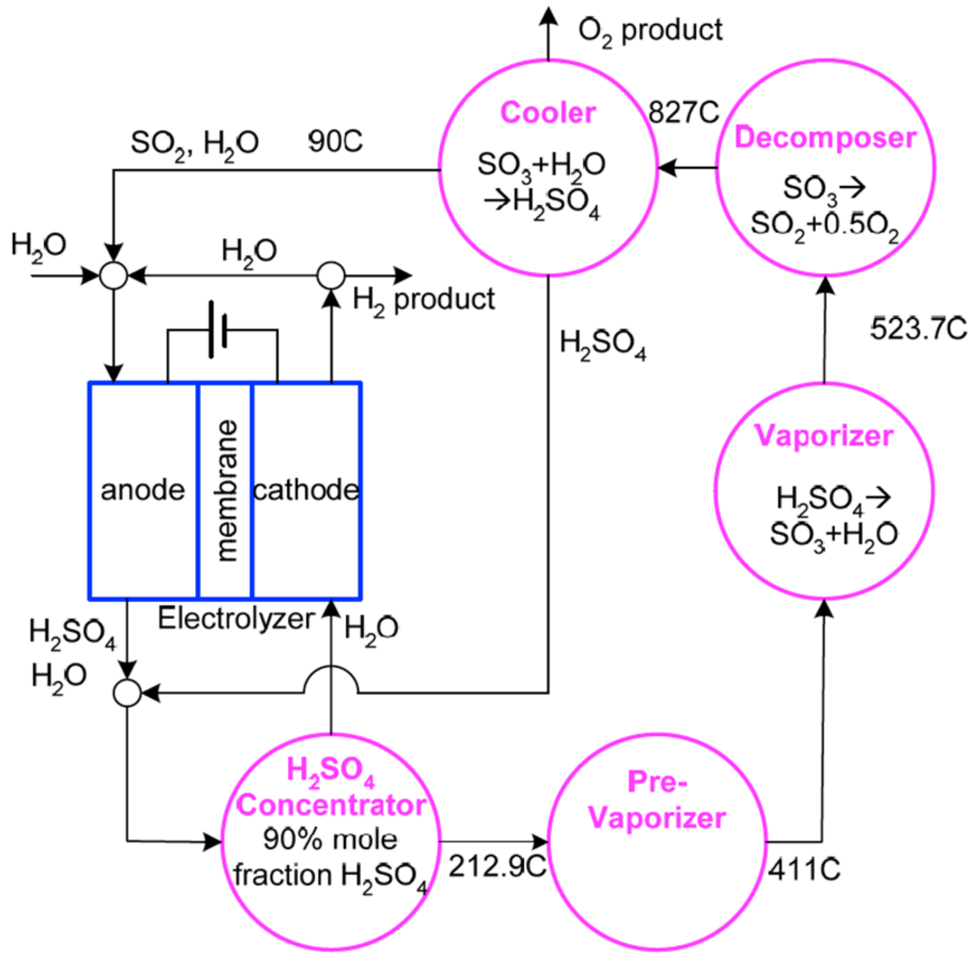


Fig. 17. Schematic diagram for the simplified hybrid sulfur model.

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.

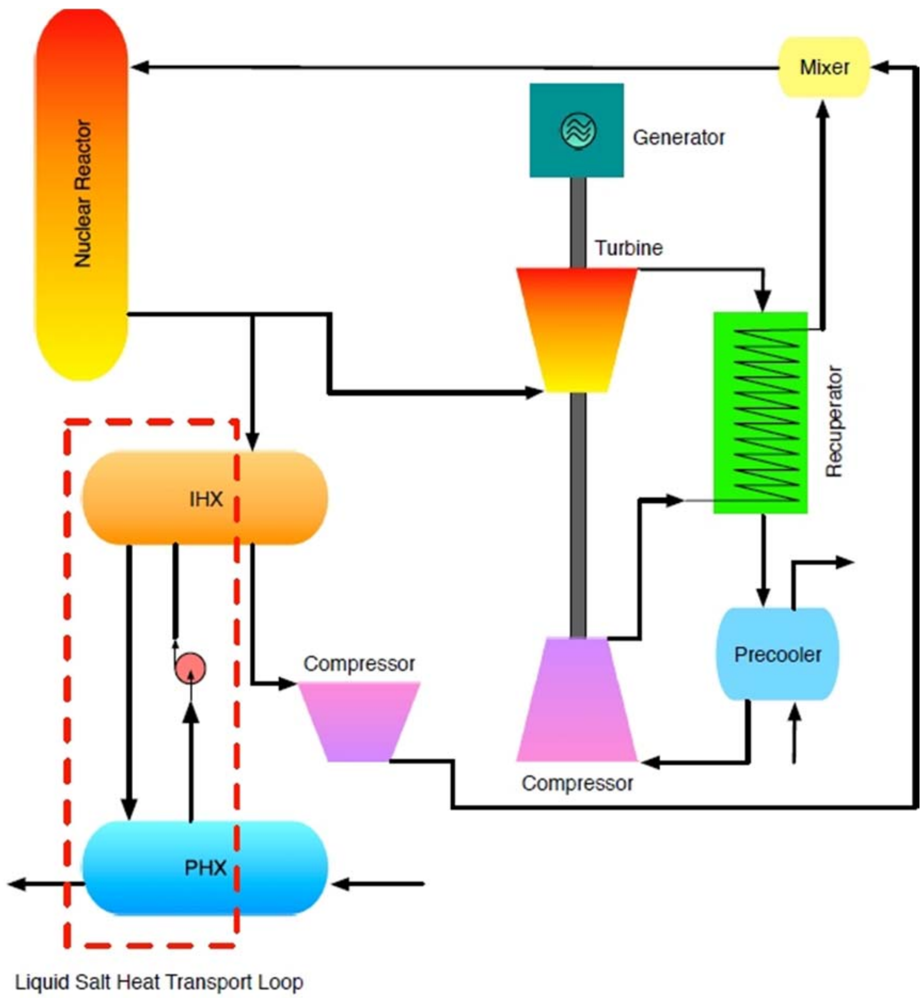


Fig. 18. Possible configuration option for a liquid-salt, heat-transport loop—direct electrical cycle and a parallel IHX.

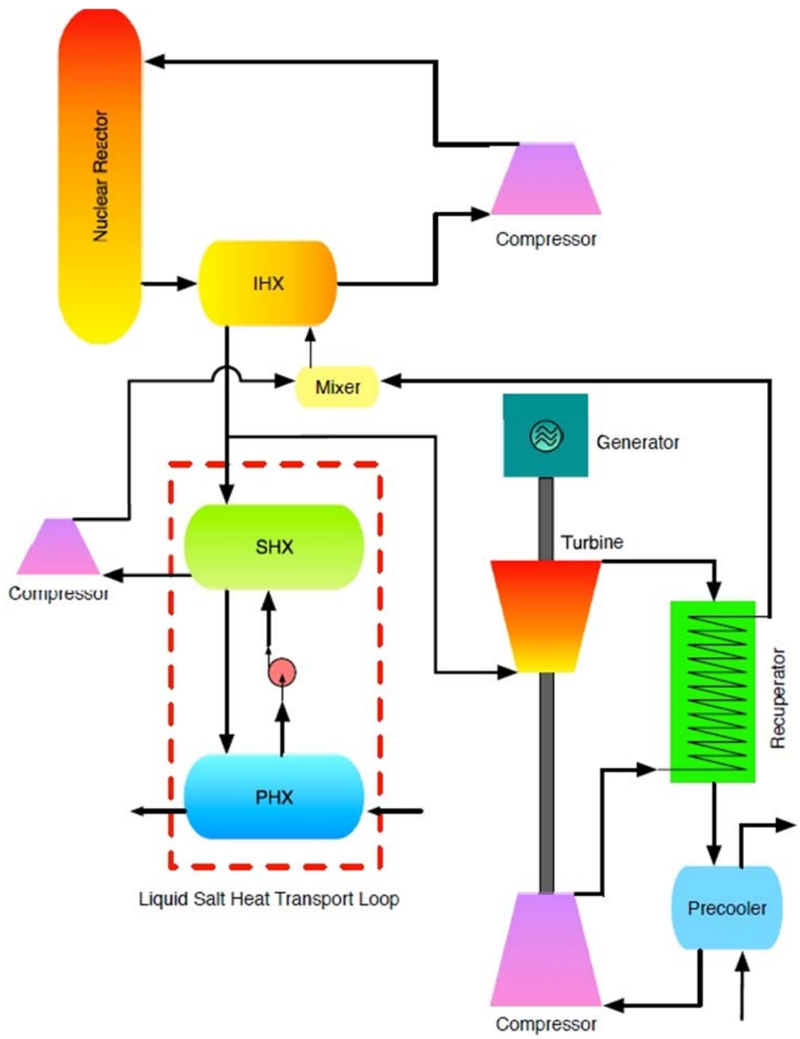


Fig. 19. Possible configuration option for a liquid-salt, heat-transfer loop—indirect electrical cycle and a parallel SHX.

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

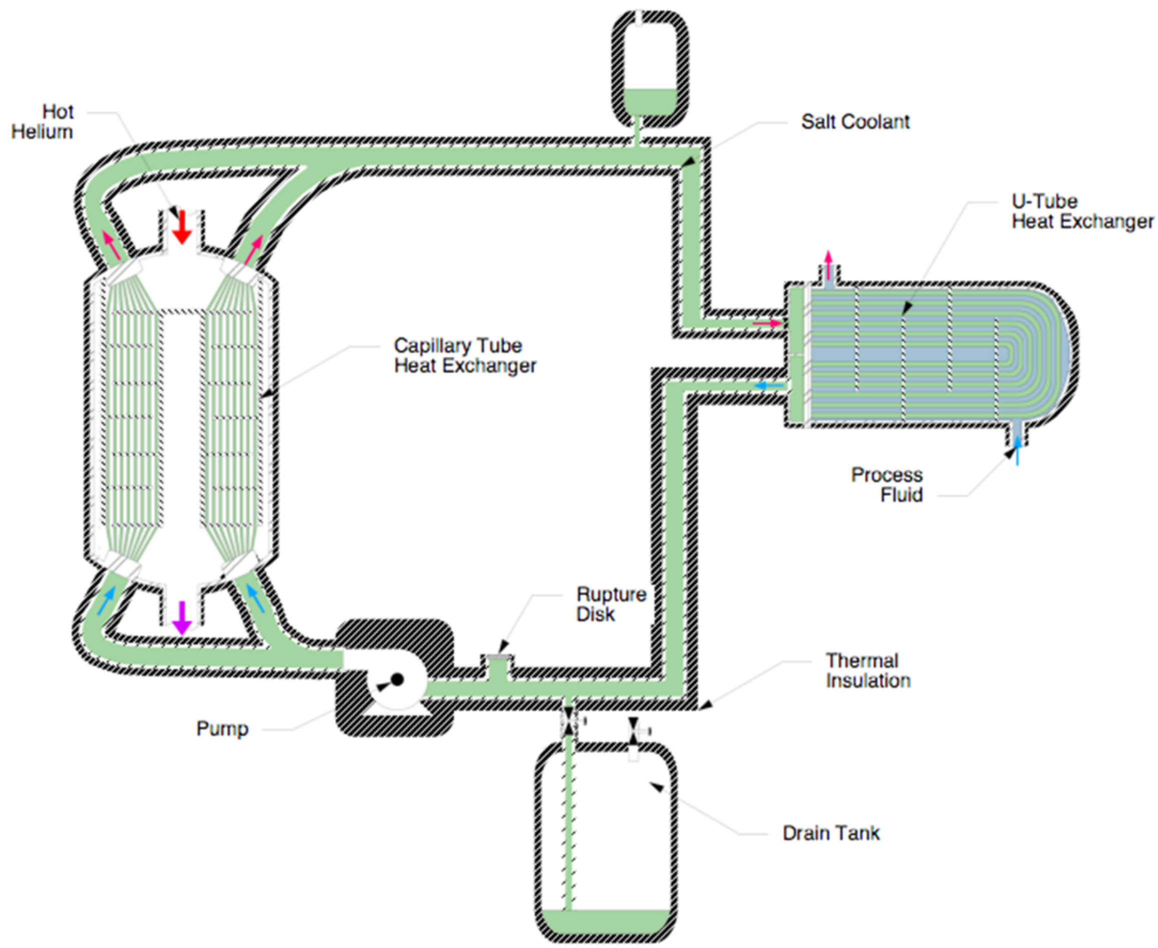


Fig. 20. Principal elements of an NGNP relevant heat transfer loop.

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^{3}K, 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

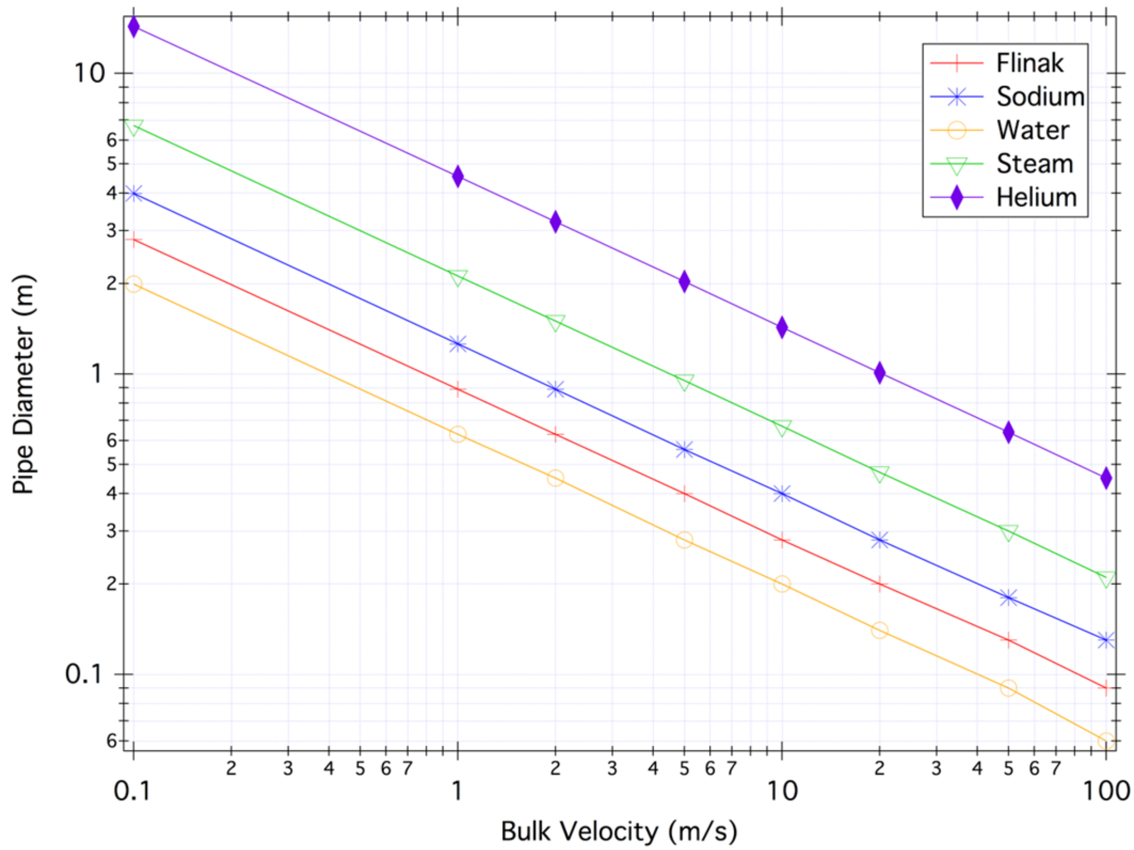


Fig. 21. Variation of pipe diameter as a function of bulk fluid velocity.

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.

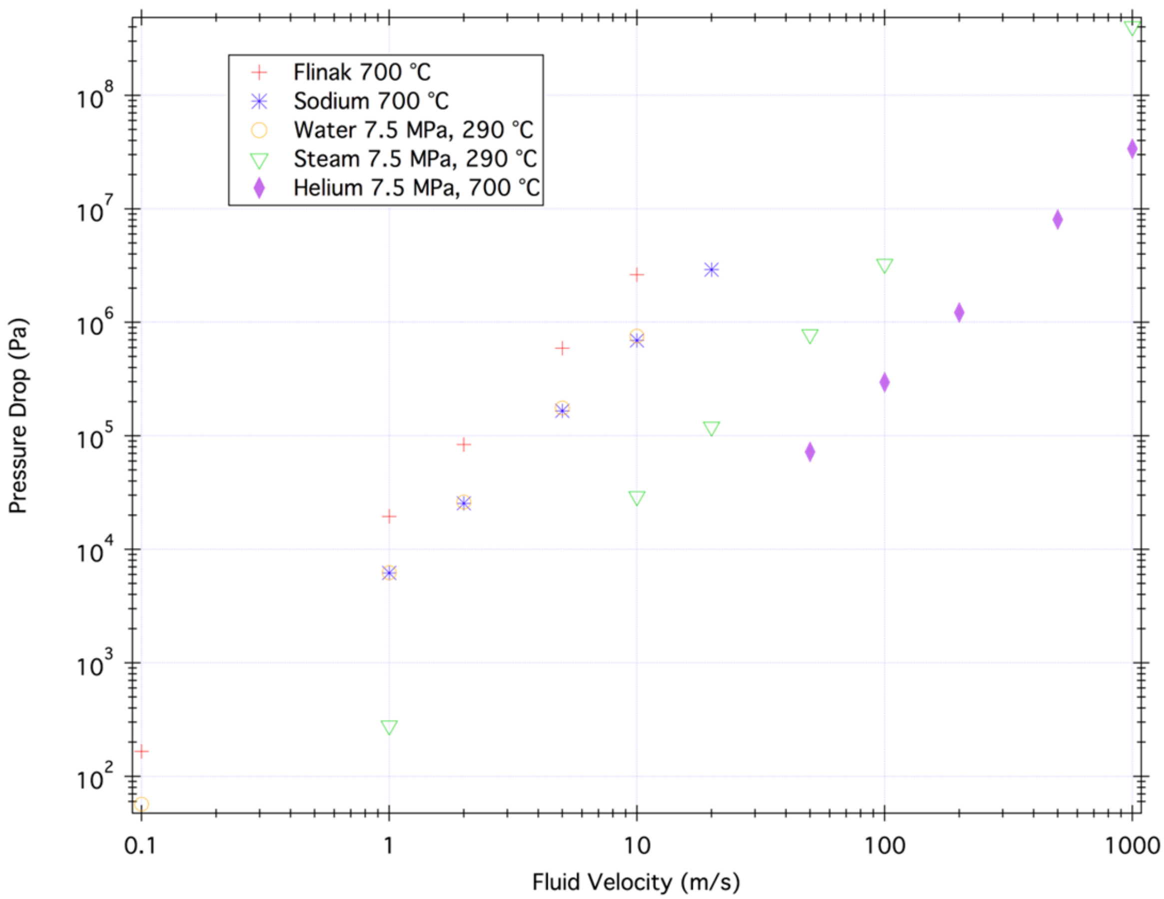


Fig. 22. Variation of total pressure drop with respect to bulk fluid velocity.

5.3.3 Pumping power

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

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

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)	Δp_{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.

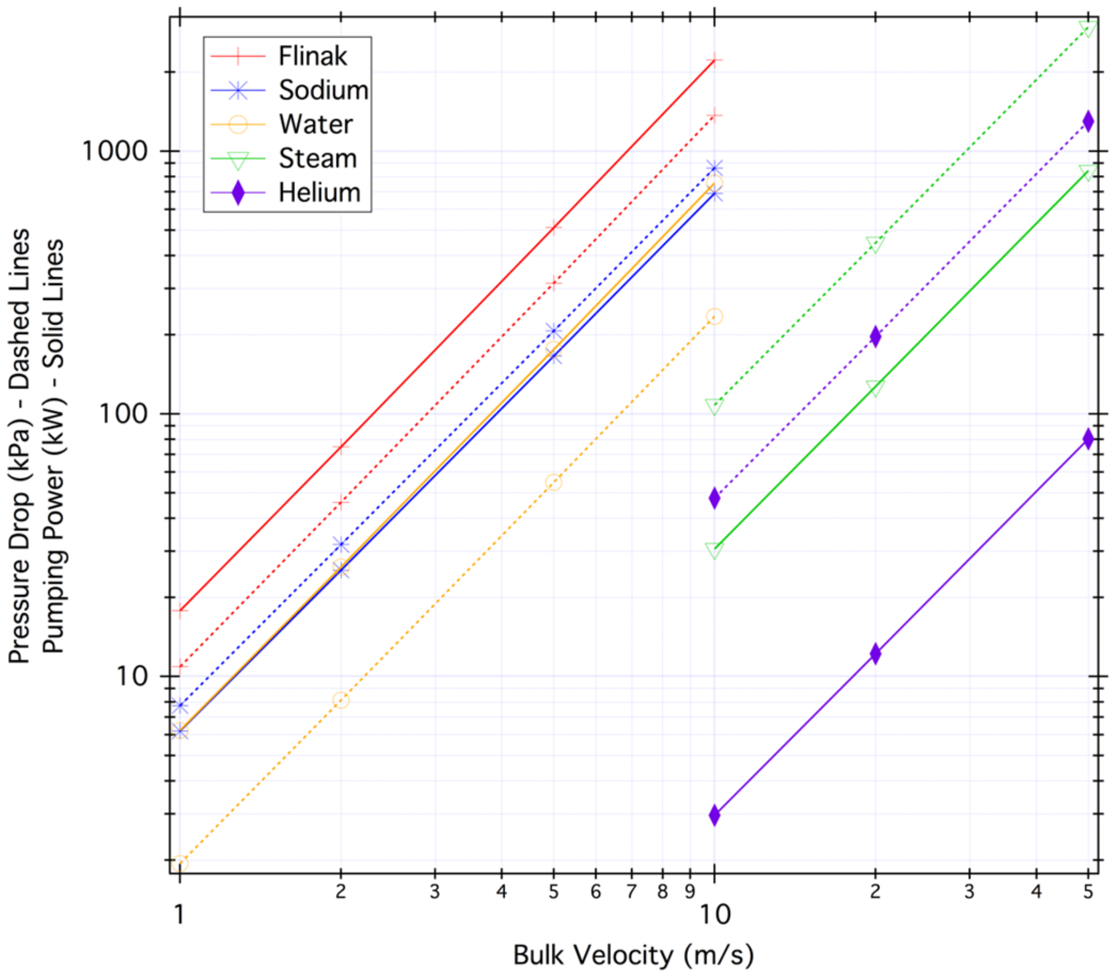


Fig. 23. Required pumping power for selected fluids as a function of fluid velocity.

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.

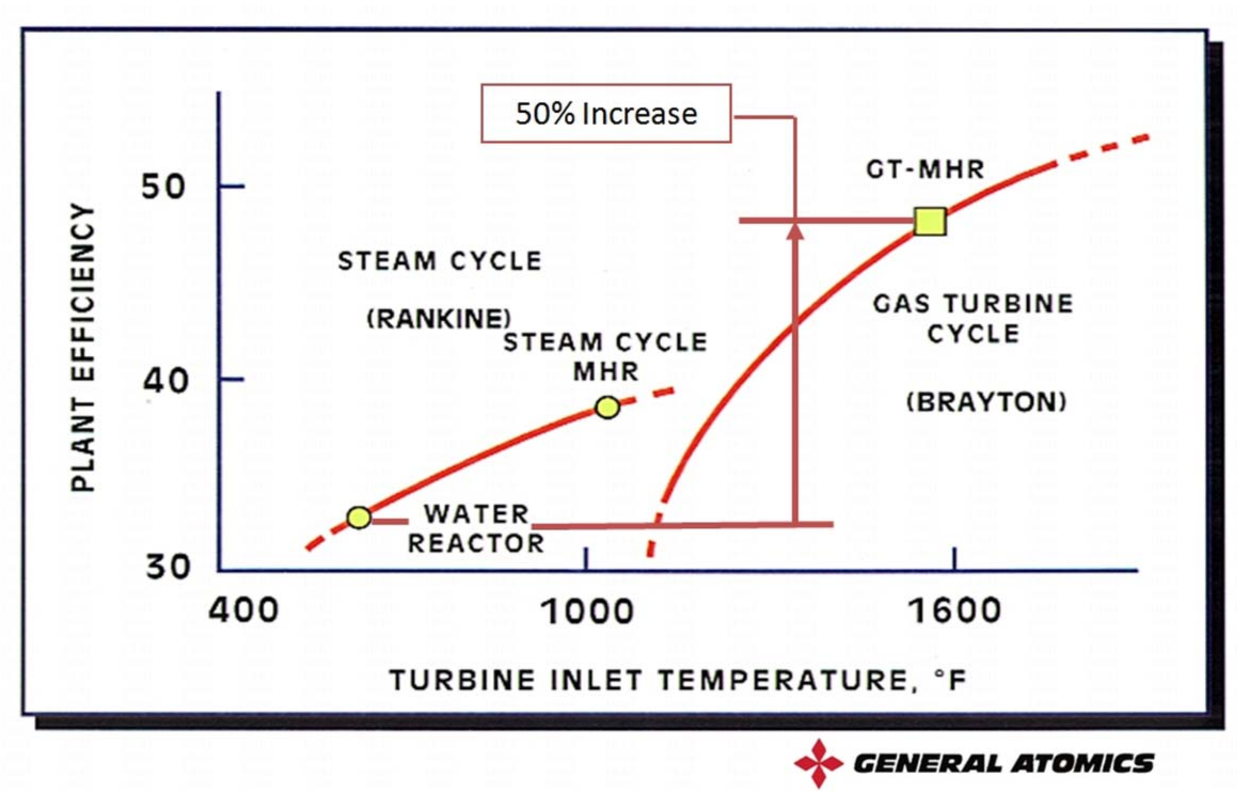


Fig. 24. Electrical production efficiency as a function of turbine inlet temperature.

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.

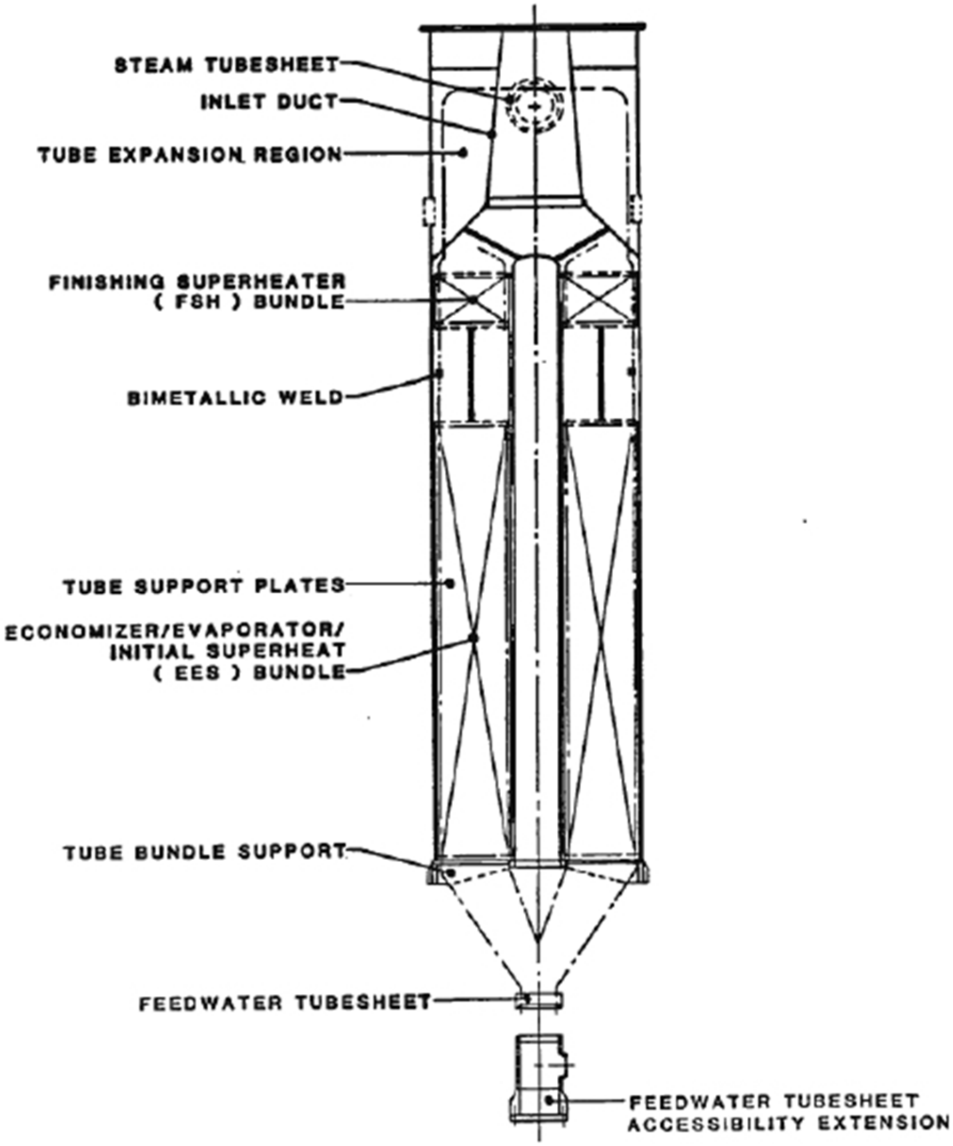


Fig. 25. Typical once-through steam generator employing a finishing superheater section.

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.

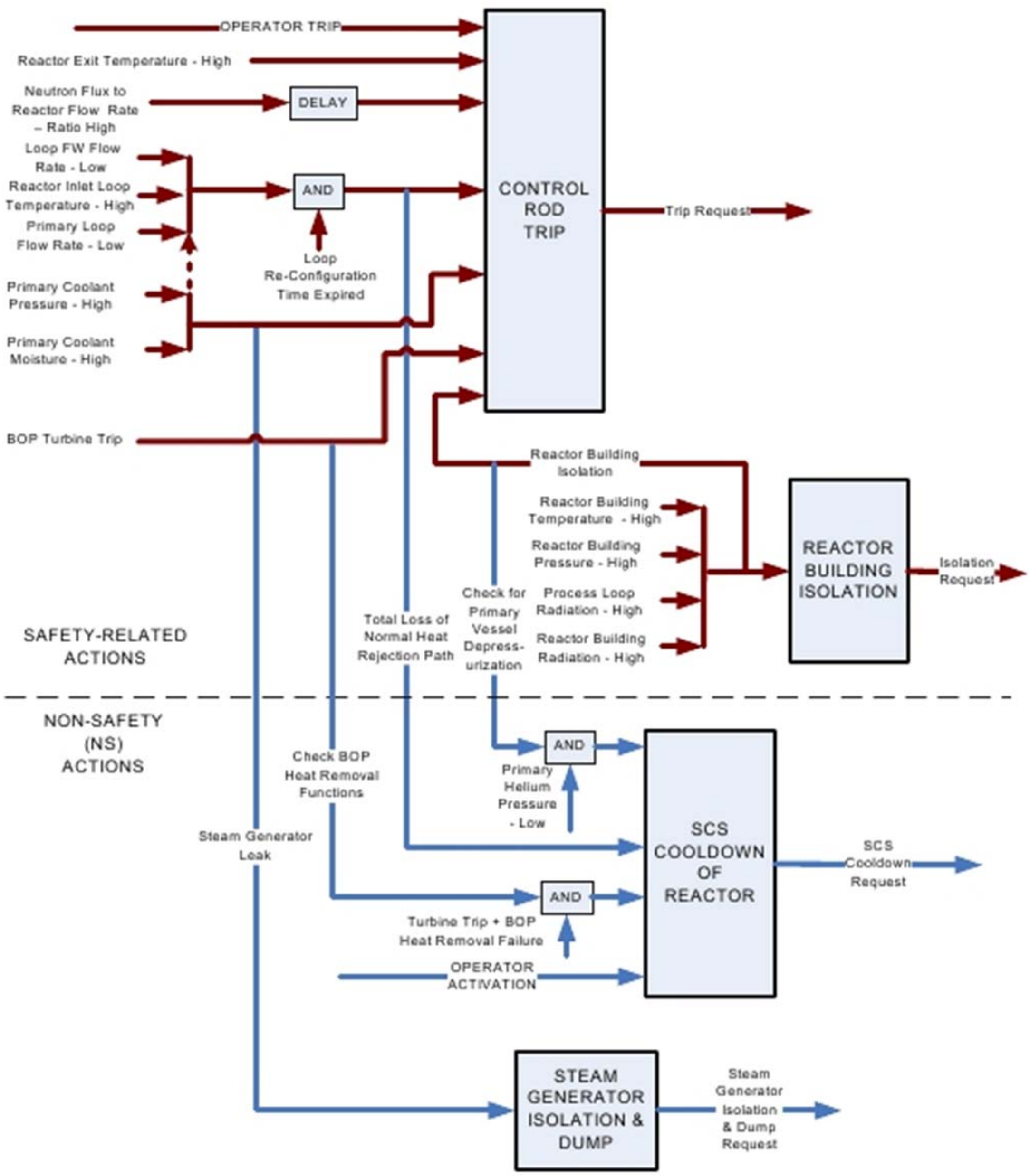


Fig. 26. Integration of steam related protection systems—steam generator in primary loop.

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.

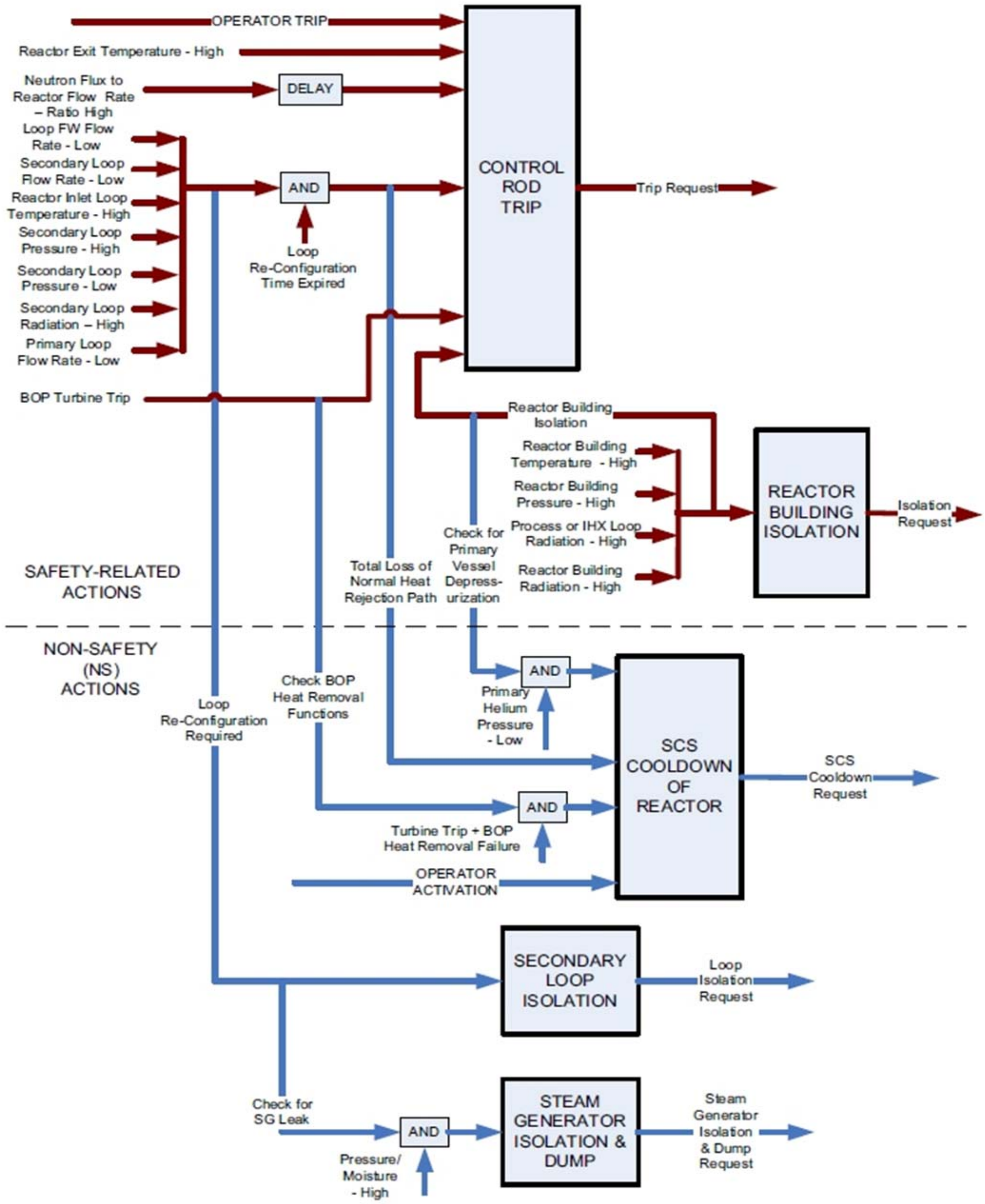


Fig. 27. Steam-related protection system logic for steam generator in secondary loop.

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):

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

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).

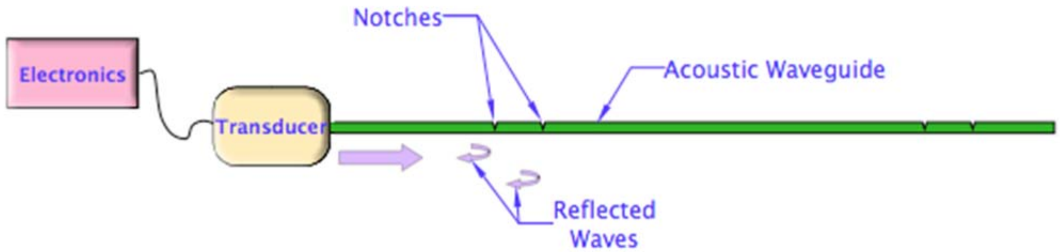


Fig. 28. An ultrasonic thermometry system including a notched waveguide.

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]

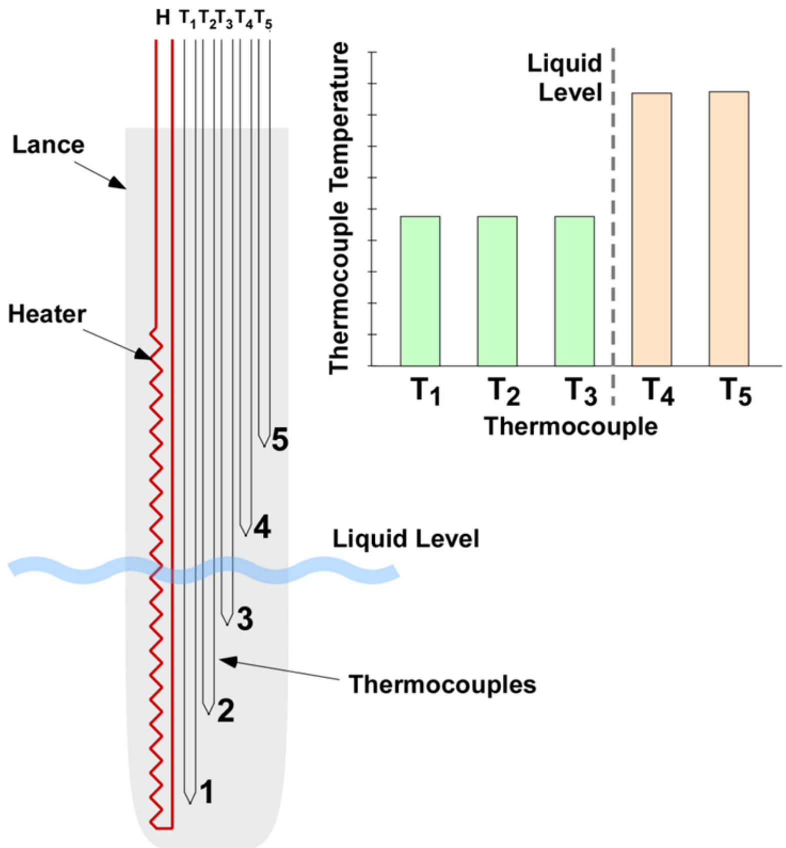


Fig. 29. Illustration of functioning of heated lance type level measurement system.

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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