Proceedings of the HTR 2014
Weihai, China, October 27-31, 2014
Paper HTR2014-61144

Discussion on Design Transients of Pebble-bed High Temperature Gas-cooled Reactor

Wang Yan，Li Fu, Zheng Yanhua
Institute of Nuclear and New Energy Technology of Tsinghua University, the key laboratory of advanced reactor engineering and safety, Ministry of Education
Beijing, 100084, China
phone: +86-010-62788157,wangyanfcw@tsinghua.edu.cn

Abstract –*In order to assure high quality for the components and their supports in the reactor coolant system, etc., some thermal-hydraulic transient conditions will be selected and researched for equipment design evaluation to satisfy the requirements ASME code, which are based on the conservative estimates of the magnitude and frequency of the temperature and pressure transients resulting from various operating conditions in the plant. In the mature design on pressurized water reactor, five conditions are considered. For the developing advanced pebble-bed high temperature gas-cooled reactor(HTGR), its design and operation has much difference with other reactors, so the transients of the pebble-bed high temperature gas-cooled reactor have distinctive characteristics. In this paper, the possible design transients of the pebble-bed HTGR will be discussed, and the frequency of design transients for equipment fatigue analysis and stress analysis due to cyclic stresses is also studied. The results will provide support for the design and construct of the pebble-bed HTGR.*

I. INTRODUCTION

To ensure a high creditable integrity for the equipment in the reactor coolant system(RCS), the components need to be evaluated under the design transients to satisfy the requirements for Class 1 in ASME Code, Section III. Based on the conservative estimates of the magnitude and frequency of the temperature/pressure/flow transients resulting from various operating thermal-hydraulic conditions in the plant, the transient conditions will be selected for the evaluation of fatigue analysis and stress analysis on equipment, which is considered on the base of engineering judgment and experience in a large extent.

Five kind of transient conditions defined in ASME Code, Section III, are considered in the design of the reactor coolant system Class 1 components, auxiliary Class 1 components, reactor coolant system component supports, and reactor internals, which cover normal operating conditions, anticipated transients, postulated accident conditions expected or postulated to occur during operation, and testing conditions[1,2]:

Level A Service Conditions – (Normal Conditions)

These conditions include any condition in the course of system startup, operation in the design power range, hot standby, and system shutdown other than upset, emergency, faulted, or testing conditions.

Level B Service Conditions – (Upset Conditions)

These conditions include any deviations from normal conditions anticipated to occur often enough that the design includes the capability to withstand the conditions without operational impairment. The service conditions include those transients resulting from any single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system, and transients due to a loss of load or power. Upset conditions also include any abnormal incidents not resulting in a forced outage as well as those that cause for which the corrective action does not include any repair of mechanical damage.

Level C Service Conditions – (Emergency Conditions)

These conditions include those deviations from normal conditions that require shutdown for correction of the conditions or repair of damage. These conditions have a low probability of occurrence but are included to establish that no gross loss of structural integrity will result as a concurrent effect of any damage developed in a system. The total postulated occurrences for such events exceeding 25 strong stress cycles over the plant design lifetime will be evaluated for cyclic fatigue using Level B service limits.

Level D Service Conditions – (Faulted Conditions)

These conditions include those combinations of conditions associated with extremely low-probability postulated events whose consequences are such that the integrity and operability of the nuclear energy system may be impaired to the extent that considerations of public health and safety are involved.

Testing Conditions

Testing conditions are those pressure overload tests that include primary and secondary hydrostatic tests and some individual components tests specified.

The number of occurrences of each transient is determined by the design life of plant. Usually, the design plant lifetime of the current 2nd generation nuclear plants is 40 years, and that of the 3rd generation advanced nuclear plants is 60 years. The design transients and their occurrence numbers for fatigue analysis of components on some reactors are shown and compared in Table 1[1,2,3].

Table 1. Design transients of some reactors

US-EPR (APWR,60 years)	Number	AP1000DCD (APWR,60 years)	Number	QINGSHAN III （CANDU, 40 years）	Number
Level A service condition					
Partial heat up with subsequent shutdown	60	Reactor coolant pump startup and shutdown (cycles of start and stop)	3000	Reactor star-up and shutdown	1000
Partial shutdown with subsequent startup	60	Plant heat-up at 100F per hour	200	Warmup and cooldown	250
Plant startup from cold shutdown to full load	240	Plant cooldown at 100F per hour	200	Power manoeuvering	10000
Plant shutdown from full load to cold shutdown	205	Unit loading between 0 and 15 percent of full power	500		
Power ramp from hot shutdown to full load	3000	Unit unloading between 0 and 15 percent of full power	500		
Daily load follow	42000	Unit loading at 5 percent of full power per minute	19800		
Partial trip to 25 percent full power	560	Unit unloading at 5 percent of full power per minute	19800		
frequency control	150000	Step load increase of 10percent of full power	3000		
Unscheduled power variations	5250	Step unload increase of 10percent of full power	3000		
Unscheduled fluctuations at hot shutdown	4000	large step load decrease with steam dump	200		
		Steady-state fluctuation and load regulation Initial Random Load regulation	 1.5×10^5 4.6×10^6 750000		

		Boron concentration equalization	2900		
		feedwater cycling at hot shutdown	18000		
		Core lifetime extension	40		
		Refueling	40		
		Turbine roll test	20		
		Primary Leakage Test	200		
		Secondary Leakage Test	80		
		feedwater heaters out of service	180		
		Core makeup tank high-pressure injection test	5		
		Passive residual heat removal tests	5		
		Reactor coolant system makeup	2820		
Level B service condition					
Turbine trip	60	Loss of load (without reactor trip)	30	Loss of offsite load	500
Loss of feedwater	60	Loss of offsite power	30	Loss of class IV power	50
		RT from reduced power		Reactor overpower/loss of Regulation	50
Reactor trip	90	RT from full power -	120	Reactor trip from 100% power	200
Reactor trip with excessive secondary side heat removal	15	Control rod drop	60	Loss of feedwater supply from 100% full power	100
Spurious RCS depressurization (Faulty Spraying)	15	Cold over-pressure	15	Total loss of feedwater supply from 100% full power(cooldown with HT pumps	10
LOOP with failure to transfer to household Load	30	Inadvertent safeguards actuation	10	Rapid cooldown	15
Unscheduled pressure and temperature Fluctuations between hot and cold shutdown	4010	Partial loss of reactor coolant flow	60	Total loss of feedwater supply, cooldown with SDC pumps	10
Excessive feedwater supply at hot shutdown	15	Inadvertent RCS depressurization -	20	Reactor stepback	500
Pressure difference between the RCPB and the SSPB	15	Excessive feewater flow(hot standby)	30		
Maximum SG pressure with open RCS	30				
Inadvertent closure of one MSIV	15				
Depressurization in the secondary side Leading to maximum	15				
Level C service condition					
Small primary side leak (SB LOCA)	<25	Small LOCA	5	Small loss of coolant	1

Small secondary side leak		Small steam line break	5	Loss of pressure and inventory control during warming	1
Faulty opening of one PZR safety valve		Complete loss of flow(bounded)		System overpressurization	1
RCS pressurization between hot and cold shutdown		Small feedwater line break	5	HT pump shaft seizure	1
SG tube failure (one tube)		Small SG tube rupture	5	Design basis earthquake with loss of class IV power	1
Long-term Turbine trip without TBS station		Inadvertent opening of automatic depressurization system valves	15	SDS1 failure with upset transients	1
Loss of offsite power with natural Circulation cooldown					
Level D service condition					
Primary side break (LB LOCA)	1	Reactor coolant pipe break (large LOCA)	1	Loss of coolant(large break)	1
Main steam line break	1	Large steam line break	1	Steam line breadk	1
MFW line break	1	Large feedwater line break	1	feedwater line failure	1
RCP locked rotor	1	RCP locked rotor	1	SDS1 failure with emergency transients	1
Control rod ejection	1	Control rod ejection	1		
External induced transient	1				
Testing condition					
Hydrostatic test (for each component)	3	Primary side hydrotest	10	Hydrostatic test	10
		Secondary side hydrotest	10	Leak test	10
System hydrostatic test prior to normal operation	3	SG tube leakage test (Second-side pressure psig) 200 400 600 840	 400 200 120 80		
Hydrostatic test following plant operation	4				

II. RCS DESIGN TRANSIENT OF HTGR

The RCS design transients of high temperature gas-cooled reactor need to be focused, because the research on the advanced reactor is still developing. Unlike the developed and standardized general reactor design whose RCS design transients had been regulated, such as EJ383-89 (RCS design transient regulations for 300,000 kilowatt PWR nuclear power plant)[4], the RCS design transients for one kind of HTGR may not be extended and applied for another directly.

The 200MW pebble-bed high temperature gas-cooled research reactor(HTR-PM)[5] designed by Institute of Nuclear and New Energy Technology of Tsinghua University in China is selected for the discussion on the RCS design transient in this paper. The lifetime of HTR-PM is designed as 40 years.

II.A. Description of RCS Design Transient

The RCS design transient classification of the existing reactor design descripted as the above could be a practical reference for that of HTR-PM. However, the transients in detail should be considered and selected carefully due to the great difference of the reactor design, and the occurrence numbers in each transient will be determined with the actual operation condition.

1. Level A services condition

Opening of RCS

In contrast to the current PWR design, continuous process of fuel loading and unloading is a key feature of HTR-PM. However, it is important to consider the RCS openings, scheduled as well as unscheduled. For the staff access to the RCS, the primary pressure should decrease to the atmosphere pressure and the temperature of RCS should be less than 50℃ under the transient.

Closing of RCS

A sequence of actions will be performed after closing of the RCS, including the helium/air switch in the pressure vessel, the primary-side leakage test. The occurrence of this transient during the design life of HTR-PM is considered to be the same as the 'opening of the RCS' transient.

Cold shutdown to low load operation

The plant will heat up after the scheduled or unscheduled cold shutdown. During the transient, the temperature increase of helium coolant in the reactor resulting from the helium blower should be limited under a certain rate conservatively. When the superheated steam from the SG in the secondary side satisfies the design value, the steam will output to turbine and the reactor power increases from no-load to lower load less than 30% full power. The coolant heat-up begins from ambient temperature and ends at the design load temperature.

Hot shutdown to low load operation

If the reactor malfunctions clear in short time and the reactor system satisfies the requirement of re-starup after emergency shutdown, the reactor should star up from hot shutdown. The systems will act to make the reactor renew to low load operation condition as soon as possible, in order to reduce the accumulation of xenon, but the heating-up rate of coolant will be controlled under the limit.

Load operation to cold shutdown

Due to periodic maintenance or inadvertent emergency shutdown for some reasons (e.g. component malfunction), the operating reactor will switch into cold shutdown.

Feedwater cycling

The decay after cold shutdown or emergency shutdown will result in a low steam generation rate and the SG water level will decrease slightly. To compensate the decrease of water level, the SGs are designed to be supplied feedwater with low flow rate periodically, and the main feedwater cycling acts on the SG intermittently under no-load condition.

Cooldown with helium purification facility (residual heat removal system out of service）

When the main residual heat removal system is out of service after cold shutdown, the helium purification facility also functions as the decay heat removal.

Ramp load increase and decrease between 30 and 50 percent of full power

The ramp load increase and decrease transients between the 30% and 50% load are represented by continuous and uniform ramp steam load changes with the feedwater supply changes. During load increase, the reactor coolant temperatures are increased from the designed values at 30% load to that at 50% load. During load decrease, the reactor coolant temperature change is reversed.

Ramp load change between 50% and 100% of full power change under automatic control

This load change is based on the daily load follow operation which varies between the 50% and 100% power. The uniform ramp power changes is limited with a certain rate under automatic reactor control. Especially, if the power decreases too fast during the unload transient, the excessive negative reactivity from xenon accumulation will result in reactivity compensation from the control rod being insufficient to keep the reactor in critical level. Therefore, the decrease rate of power should be limited to ensure the reactor in the normal operation.

Small step load increase and decrease

Due to disturbances of the electrical net, a step load change may generate. After a step load decrease, the secondary-side steam pressure and temperature begin to increase and the RCS average temperature and pressurizer pressure also increase. The automatic control system inserts the control rods to reduce the core power to keep power match of the reactor. Therefore, the reactor coolant temperature and pressure decreases again. The step load increase transient is reversed with the step load decrease transient.

The RCS design is required not to trip emergency shutdown with small step load change (less than 10% full power) when the reactor operates between 50% and 100% full power under automatic control.

Large step load increase and decrease

The transient is assumed as a large reactor step load change (larger than 10% full power) during the operation condition. The step load change will result in a mismatch between reactor and turbine, and a steam bypass system will be initiated to avoid emergency shutdown trip.

Steady-state fluctuations and load regulation

Reactor coolant pressure and temperature will have fluctuation in normal steady operation from dead zone of control rod. Another fluctuations come from load regulation operation for grid frequency control.

Secondary-side leakage test

A secondary side leakage test is performed to check closures after the opening of the secondary system. To avoid the main steam safety valve opening inadvertently, the test pressure should be limited under its design pressure value. Meanwhile, the primary side should be pressurized to control the pressure difference between the primary side and the secondary side.

2. Level B service conditions

For the moderate frequency accidents of level B service conditions, there are 4 typical accidents to be considered for the pebble-bed HTGR.

Loss of load

This transient occurs when there is a step decrease in the turbine load from full power without an immediate reactor trip, and it is the most severe pressure transient on the RCS in level B service conditions. However, the reactor will be tripped to shut down as a sequence of the trip of the reactor protection system.

Loss of offsite power

Due to loss offsite power, the main helium blower and the feedwater pumps will be tripped which results that the heat removal capacity of the secondary side decreases. The RCS temperature and pressure will increase which cause negative reactivity to decrease the reactor power. In this condition, the control rods drop by gravity because of the loss of offsite power, and the reactor is tripped to ensure the reactor safety.

Reactor trip from low power

The transient assumes that the reactor is tripped before the power of HTR-PM reaches 30% full power when the reactor is in load operation. The temperature and pressure change in this transient is similar to that with reactor trip from full power, but its magnitude is smaller.

Reactor trip from full power

Reactor trip from full power is caused by many reasons. The RCS temperature and pressure decrease after the emergency shutdown initiated by the trip of reactor protection system, the conditions of the SG secondary side also change accordingly due to heat transfer between the SG primary side and the SG secondary side.

3. Level C service conditions

The following transients are suggested in level C service condition.

Small LOCA with isolation

The leak or small break on the components connected with the pressure vessel may result small LOCA. When the rate of change of the primary pressure reaches a limit, the reactor protection system will close isolation valves on the tubes connected with the primary loop, in order to prevent the helium coolant loss.

In HTR-PM, the break of a DN10mm instrumentation tube connected with the primary loop is suggested to be considered in this transient.

Steam generate tube rupture(SGTR)

Water ingress into the reactor is a most particular and important transient for high temperature gas-cooled reactor during the postulated SGTR accident, which is different from other kind of reactors. Steam ingress to the primary circuit of reactor will cause a positive reactivity introduction with the increase of steam density in reactor core to enhance neutron slowing-down, as well as the chemical corrosion of graphite fuel elements and reflector structure material. Besides, increase of the primary pressure may result in the opening of the safety valves, consequently leading the release of radioactive isotopes and flammable water gas.

Small main steam line break

A small main steam line break is considered as a break equivalent to a main steam safety valve opening and remaining open.

Inadvertent opening of safety valve

The transient is assumed that the safety valve of the primary coolant loop opens inadvertently, which will result in rapid RCS depressurization.

4. Level D service conditions

The following typical level D service condition transients are considered for HTR-PM.

Large main steam line break

A double-ended rupture of a main steam line is assumed to occur in this transient. The rupture of main steam line will decrease the secondary side pressure rapidly, which cause the change of the pressure difference on both sides of the SG heat exchange tubes.

Main feedwater line break

This transient applies to a break in the piping between the SG and the main feedwater isolation valve under normal operation. The main feedwater line break will result in the feedwater flow through the break partly or completely in the affected loop. Main feedwater loss causes the decrease of SG water level and the increase of the RCS pressure and temperature. The reactor protection system will be actuated to trip emergency shutdown, and the decay heat from reactor will be removed by residual removal system.

ATWS

Anticipated transient without scram(ATWS) is an anticipated operational occurrence followed by failure of the reactor trip portion of the protection system. In this transient, the flow of helium coolant in the primary loop is blocked due to shutdown of main helium blower and close of blower flaps, the decrease of removal capacity for generated heat from reactor will results that the RCS pressure and temperature increase. In the accidents analysis of the HTR-PM, the typical ATWS transients, such as false withdrawal of control rod with ATWS, loss of offsite power with ATWS and loss of feedwater with ATWS, are considered.

Pipe break between pressure vessel and isolation valves

This transient is assumed that the location of a pipe break is between pressure vessel and isolation valves. In the transient, the helium coolant will leak from the break and cannot be isolated to prevent its loss, until a balance is reached between the RCS pressure and the environment pressure. The pipes between pressure vessel and isolation valves are produced under the highest strict standards, so the transient is an extremely low-probability postulated event.

A D65mm pipe break with isolation

The transient is assumed that a double-ended guillotine break on a D65mm pipe connected to pressure vessel occurs, which results that the primary system pressure decreases rapidly. Because the break location is downstream of isolation valves, the loss of helium coolant could be prevent by close of the isolation valves actuated by reactor protection system trip.

5. Testing conditions

Testing conditions is independent of other transients in the fatigue analysis, including the tests on the primary side and the secondary side. The test pressure is 1.25 times design pressure, and the temperature of test fluid is required to compatible with reactor material ductility. During the primary-side or secondary-side pressure tests, the pressure is maintained in the other side of SG tubes to avoid overstress on the SG tubes. The pressure tests is performed before plant star-up or after major repair.

III. OCCURRENCES OF RCS DESIGN TRANSIENTS

The occurrence numbers of the RCS transients during the plant lifetime are obtained on the base of the conservative estimates from operating conditions in plant and engineering experience, and their magnitude should be calculated conservatively from the thermal-hydraulic analysis on operation transients and accidents. According to the current research, the occurrences of the above listed RCS transients during the 40-year design lifetime of the HTR-PM are suggested as shown in Table 2.

Table 2 Occurrences of RCS design transient for HTR-PM（40-year design lifetime）

Transient Description	**Number of occurrences**
Level A service conditions	
Opening of RCS	5
Closing of RCS	5
Cold shutdown to low load	90
Hot shutdown to low load	40
Load operation to cold shutdown	50
Feedwater cycling	900
Cooldown with helium purification facility（residual heat removal system out of service）	10
Ramp load increase and decrease between 30 and 50 percent of full power	130
Ramp load change between 50% and 100% of full power change under automatic control	13200（90% plant availability）
Small step load increase and decrease（without RT）	400
Large step load increase and decrease（without RT）	40
Steady-state fluctuations and load regulation Steady-state fluctuations Load regulation	 3×10^6 5×10^5
Secondary-side leakage test	4
Level B service conditions	
Loss of load	20
Loss of offsite power	20
Reactor trip from low power	10
Reactor trip from full power	40
Level C service conditions	
Small LOCA with isolation	3
Steam generate tube rupture	3

Small main steam line break	3
Inadvertent opening of safety valve	10
Level D service conditions	
Large main steam line break	1
Main feedwater line break	1
ATWS	1
Pipe break between pressure vessel and isolation valves	1
A D65mm pipe break with isolation	1
Testing conditions	
Primary side test	1
Secondary side test	1

IV. DISCUSSION ON HTR-PM RCS DESIGN TRANSIENTS

The RCS design transients for the HTR-PM and their occurrence numbers during the reactor lifetime are discussed in the above section. Due to the large difference on the design between the pebble-bed high temperature gas-cooled reactor and other reactors, some RCS design transients are entirely different. For example, the primary-side pressure of HTR-PM with helium coolant is much lower than the secondary-side pressure, but it is reversed in the general PWR design. Therefore, the thermal-hydraulic phenomena and process during the ‘steam generator tube rupture’ transients is distinct for the HTR-PM. The water in the secondary-side of steam generator will spout into the primary side when a tube break occurs, which will cause a positive reactivity introduction, as well as the chemical corrosion of graphite fuel elements and reflector structure material. In the consideration on the occurrence numbers of transients, it is also different due to the design difference between the HTR-PM and other kind of reactors. For instance, the key feature of continuous loading and reloading fuel without reactor shutdown in the HTR-PM will result in the occurrences of the transients (‘opening of RCS’, ‘closing of RCS’, ‘Cold shutdown to low load’, ‘Load operation to cold shutdown’, etc.) differ from that of PWR which needs a periodic reloading with reactor shutdown.

Therefore, the design transients of HTR-PM should be considered and determined carefully, not a simple copy from that of other reactors. And with the increasing experience and feedback in the HTR-PM design, the further research on the RCS design transient of the HTR-PM is necessary.

V. CONCLUSIONS

In order to assure high quality for the components and their supports in the reactor coolant system, etc., some thermal-hydraulic transient conditions will be selected for the evaluation of fatigue analysis and stress analysis on equipment to satisfy the requirements ASME code, which are based on the conservative estimates of the magnitude and frequency of the temperature and pressure transients resulting from various operating conditions in the plant.

Due to the large difference on the design between the pebble-bed high temperature gas-cooled reactor and other reactors, the RCS transients for the pebble-bed HTGR need to be considered separately. The main RCS transients of the HTR-PM are studied and their occurrence numbers are discussed in this paper, but the results on the magnitudes of the transients obtained by the thermal-hydraulic analysis are not listed in detail.

The results on the frequency and magnitude of the temperature/pressure/flow transients under the above HTR-PM RCS design transients will provide support for the fatigue analysis and stress analysis on the RCS components design of the pebble-bed HTGR.

REFERENCE

[1] AP1000 Design Control Document, Revision 15. Westinghouse Electric Company. 2005.
[2] U.S. EPR Final Safety Analysis Report. AREVA. 2007.
[3] Qinshan Phase-III CANDU Nuclear Power Station FSAR.
[4] Nuclear industry standard of the people's Republic of China EJ383-89. 1989.
[5] Zhang, Z., Sun, Y. Economic potential of modular reactor nuclear power plants based on the Chinese HTR-PM project. Nucl. Eng. Des. v237, p.2265–2274, 2007