Proceedings of the Korean Nuclear Society Spring Meeting
Pohang, Korea, May 1999

Evaluation of Multiple Steam Generator Tube Rupture Events for KNGR

Seok Jeong Park, Cheol Woo Kim, Chul Jin Choi, Sang Yong Lee, and Jong Tae Seo

Korea Power Engineering Company, Inc
150 Duckjingdong Yusungku Taejeon KOREA 305-353

Abstract

The likelihood that steam generator tube ruptures (SGTR) will result in containment bypass is reduced by specific Korean Next Generation Reactor (KNGR) design features. Added features, relative to the Korean Standard Nuclear Power Plant (KSNP) Nuclear Steam Supply System (NSSS) Design, include modifications to assure continuity of steam bypass capability following a safety injection actuation signal. These changes significantly enhance the capability to avoid containment bypass via opening of the Main Steam Safety Valves (MSSVs) during a SGTR relative to the KSNP design. Thermal-hydraulic analyses are performed using RELAP5/MOD3 to evaluate the effectiveness of the added features. The significant result of the analyses is the length of time between event initiation and opening of the MSSVs. With only automatic response of plant systems, this time varies from greater than 4 hours for rupture of one tube to 30 minutes for rupture of five tubes. This paper presents the results of the analyses for 1 tube rupture and 5 tubes rupture cases.

1.0 Introduction

The KNGR is an evolutionary Advanced Light Water Reactor (ALWR). Design features incorporated into the KNGR design provide significant increases in operating margins. Furthermore, the designs provide significantly greater capability in mitigating design basis events. However, the NRC staff [1] suggested that containment bypass of primary coolant following Steam Generator Tube Rupture (SGTR) should be investigated for the evolutionary ALWR design. In particular, it is desired to reduce the reliance on manual operator actions to mitigate the event during the early post-trip period following event initiation.

In SSAR Chapter 15, an analysis of a single SGTR event is presented assuming the rupture of one tube and only with automatic actuation of safety grade systems and components for 30 minutes. After 30 minutes, operator action is assumed to mitigate the event. The offsite radiological dose acceptance criteria are satisfied, but the conservative analysis methods (in SSAR Chapter 15) predict the opening of Main Steam Safety Valves (MSSVs) on the secondary side thereby releasing primary coolant to atmosphere. The conservatively biased calculations required for these safety analyses yield a bounding upper limit on radiological release and are appropriate for their intended purposes.

Reference 2 shows that EPRI and Westinghouse representatives provided comments concerning the staff's position on Multiple SGTR (MSGTR). Westinghouse and EPRI believed that the design basis for the AP600 should remain a single tube rupture, and that multiple tube ruptures should be analyzed on a safety margin basis, using best-estimate techniques.

To mitigate a SGTR for the KNGR design, realistic analyses were performed assuming only automatic actuation of components and systems which include both safety grade and non-safety grade equipment as ABB-CE analyzed in System 80+ [3,4].

A requirement on advanced light water designs is to analyze and show acceptable operator action times for beyond design basis events such as multiple SGTR. A goal is to design the plant such that no operator action is required for 30 minutes following multiple SGTR of up to five tubes consistent with USNRC ALWR policy goals and the ALWR Utility Requirements Document. The length of time from the initiation of the event until the operator must take action to prevent opening of the MSSVs is evaluated for one and five tubes rupture cases. The methods employed for these evaluations are determined in order to calculate the realistic or best estimate responses of the KNGR design. The results of single SGTR and multiple SGTRs with 5 tubes rupture without operator action are compared.

2.0 Design Features for Mitigation of SGTR

The KNGR design provides a number of systems for use in mitigation of a tube rupture event. The design basis of these systems are not typically for the prevention or mitigation of a steam generator tube rupture event, however, their use whether automatic or manual, decreases the probability of a main steam safety valve from lifting. Changes form the KSNP that were incorporated into the KNGR systems and component configurations have resulted in substantial increases in plant safety. The plant changes that increase safety performance are evident in the plant design and the verifications of their safety benefit are evident in the increased operating margins. Interactions among the plant systems and the control and protection systems have been studied and refined to yield the integrated KNGR design.

The important KNGR design features to significantly increase the capability to avoid containment bypass during multiple SGTR events are:

a. The Steam Bypass Control System (SBCS) is an automatic system, which provides a path to remove steam from the steam generators. In the event of SGTR, the SBCS will automatically relieve secondary pressure and dump steam to the condenser
b. Two N-16 monitors, one per steam generator, to assist in the diagnosis of the event. The signal would be latched to prevent loss of signal on reactor scram. These monitors would be in addition to the KNGR steam line area radiation monitors and sample and blowdown radiation monitors.
c. The main Feedwater Control System (FWCS) in the present KNGR design automatically terminates main feedwater following a reactor trip with reduced primary coolant temperatures.
d. The KNGR SDVS (Safety Depressurization and Vent System) discharging to the IRWST is actuated by the operator when MSSVs are challenged.
e. The IRWST in the KNGR design is both a large source of safety injection water and a quench tank that confines blowdown fluids within the containment.
f. The large secondary side volume of the KNGR SGs provides extra capacity and therefore extended operator action time before the MSSVs are challenged.
g. The lowered RCS operating coolant temperature decreases the likelihood of a SGTR event.

3.0 SGTR Analysis and Assumption

The SGTR events were analyzed with best estimate analysis methods and assumptions. This best-estimate analysis of multiple SGTR events was conducted by the RELAP5/MOD3 computer program [5]. The terminology "best-estimate" means realistic modeling both the thermal-hydraulic phenomena and systems operation during multiple SGTR. The primary system models include the key nuclear steam supply system components such as the steam generators, reactor vessel and core, hot and cold legs, pressurizer, and reactor coolant pumps. The key safety and control systems are modeled. These systems include the pressurizer sprays, heaters, pilot operated safety relief valves, safety injection pumps, charging and letdown flows, and shutdown rods. The secondary system models include key components from the main feedwater control valve to the turbine stop valves. The steam generators are modeled to receive feedwater from main and auxiliary feedwater systems. The main steam lines contain atmospheric dump, main steam safety, main steam isolation, turbine stop, and turbine bypass valves. The nodalization scheme of RELAP5/MOD3 for KNGR secondary system is shown in Fig. 1. Initial conditions and other plant design parameters relevant to the secondary system models are provided to initialize and initiate transient calculations.

Normal, full power plant conditions are assumed for the analyses (initial power is 102%). The major initial operating parameters for multiple SGTR analysis are provided in Table 1. The following best estimate assumptions are made in the present analyses:

a. Offsite power is available during the transient.
b. All control systems are assumed to be in the automatic mode.
c. No operator actions are included in the analysis.
d. Normal plant protection systems are assumed to be available.
e. Control system actuations during the transient are assumed to be at nominal setpoint values. The condenser is assumed to be available for receiving steam flowing through the turbine bypass valves from the steam generators and steam generator liquid that flows through the blowdown piping.
g. The plant protection systems are assumed to provide automatic protection during the transient.

4.0 Results of Analysis

The significant result of the analyses is the length of time between event initiation and opening of the MSSVs. With the above analysis assumption and only automatic response of plant systems this time varies from greater than 4 hours for rupture of one tube to 30 minutes for rupture of five tubes. Figure 2 shows the relationship between the number of ruptured tubes and the time at which the MSSVs open. The 30 minutes are adequate for the operator to diagnose the SGTR event and to initiate appropriate action to prevent release from the secondary system.

The analyses are aimed at quantifying the system performance under SGTR conditions. Figures 3 through 5 present the results for one tube rupture and Figures 6 through 8 illustrate the results for the five ruptured tubes. The results consist of transient plots for, 1) RCS and SG pressures, 2) break, safety injection, and steam bypass flow rates, 3) steam generator level. The sequences of events occurring during the transients are presented in Tables 2 and 3, respectively. The RCS pressure transients for both cases illustrate that the pressure decreases very rapidly following the rupture(s) resulting in an increase in the charging flow and actuation of the pressurizer heaters. The major difference between two cases is that the parameter change rates for multiple tube ruptured case are more rapid than those of single tube failure case due to higher break discharge flow. As the pressure decreases, a reactor trip on hot leg saturation is obtained at 1209 seconds for the one tube case and at 167 seconds for the five tubes case. A turbine trip occurs immediately afterwards and TBVs are opened within two seconds after reactor trip. The safety injection actuation signal is generated upon reaching a low RCS pressure subsequent to the reactor trip. The safety injection flow eventually causes an increase in RCS pressure and as the break flow is balanced by this flow and charging flow. And the RCS pressure reaches a quasi-steady state value (about 1970 psia (138.505 kg/cm^2A) for one tube rupture and about 1560 psia for 5 tubes ruptured).

For the one tube ruptured case, the steam generator pressure (Figure 3) increases rapidly after the turbine trip and the pressure remains at about opening setpoint of the bypass valves (1100 psia) subsequently reaching bypass valve modulation mode. The bypass system remains open for the one tube rupture case during most of the remainder of the transient up to 10,000 seconds (167 minutes). The affected steam generator level (Figure 5) is controlled prior to the reactor trip on steam generator level control and decreases rapidly subsequent to the reactor trip (due to termination of main feedwater on low T_{avg} and level collapse due to increase in the steam generator pressure). Despite the break flow (Figure 4) into the secondary side of the affected steam generator, the level continues decrease for the one tube rupture case, since the TBV remains open removing inventory from the generator. The unaffected steam generator level also decreases up to the time the auxiliary feedwater is initiated on low steam generator level. Subsequently, the level in this steam generator continues to increase until a high steam generator level signal shuts off auxiliary feedwater flow. As a result, the steam generator level begins to decrease.

For the five tubes ruptured case, the reactor trip and turbine trip occurs very early in the transient due to the large amount of break flow (about five times as much as the one tube rupture case). Subsequent to reactor trip the safety injection flow (Figure 7) causes the RCS pressure (Figure 6) to reach the quasi-steady state value of about 1560 psia. The steam generator pressure (Figure 6) increases rapidly following the reactor trip and reaches the TBV opening setpoint of 1100 psia. The affected steam generator level (Figure 8) builds up very rapidly prior to the reactor trip since the steam generator level control system cannot keep up with the large amount of break flow. The level decreases immediately after reactor trip due to the collapse of the two phase steam generator level on high SG pressure, termination of the main feedwater on low T_{avg}, and opening of the turbine bypass valves. The level continues to increase and a Main Steam Isolation Signal (MSIS) is generated on high level at 1726 seconds. The MSIS terminates the steam bypass flow and results in an increase in the affected SG pressure. As this pressure reaches 1195 psia at 1920 seconds, the MSSVs of the affected SG lifts relieving the pressure. The MSSVs close following the pressure relief. Subsequently the MSSVs repeatedly open and close to remove mass and energy from the affected steam generator.

Thus, for the one tube rupture case, the affected steam generator MSSVs remain unchallenged for longer than 10,000 seconds (167 minutes) due to the SBCS (no MSIS signal) and the relatively smaller break flow. However, for the five tubes rupture case, the affected steam generator MSSVs are challenged as an MSIS is generated on a high steam generator level. After opening of MSSVs, these valves cycle open and close to relieve steam generator mass and energy. The steam generator level is expected to increase even after the MSSV opening since the break flow rate is larger than the MSSV release rate to maintain the steam generator pressure around 1195 psia.

5.0 Conclusion

The likelihood that multiple Steam Generator Tube Ruptures (MSGTR) will result in containment bypass is reduced by specific Korean Next Generation Reactor (KNGR) design features. The KNGR design provides a number of systems for use in mitigation of a tube rupture event. These features significantly enhance the capability to avoid containment bypass via opening of the Main Steam Safety Valves (MSSVs) from multiple SGTR. The significant result for the rupture of one to five of steam generator tubes is the length of time between event initiation and opening of the MSSVs. The detailed analyses of one and five tubes ruptured cases are presented. The results show that the MSSVs opening time varies greater than 4 hours for rupture of one tube to about 30 minutes for rupture of five tubes with only automatic response of plant systems and without operator actions. Therefore, the KNGR standard design has a proper mitigation function on the multiple SGTR up to five tubes.

6.0 References

1. Letter, D. M. Crutchfield (NRC) to R. A. Matzie (ABB), "CE System 80+ Protection Against Containment Bypass During a Steam Generator Tube Rupture (SGTR)," August 12, 1993.
2 SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs, " April 2, 1993.
3. M. X. Franovich (NRC), "Public Meeting of September 16, 1993, To Discuss CE System 80+ Protection Against Containment Bypass During a Steam Generator Tube Rupture (SGTR)," Minutes of Meeting, September 23, 1993.
4. "The System 80+ Standard Plant Design Control Document , Appendix 5F," 1997.
5. NUREG/CR-5535, "RELAP5/MOD 3 Code Manual (Draft)," June 1990.

Table 1

Parameter Values for Analyses

Initial Conditions & Setpoints		
1.	Initial RCS Pressure, psia (kg/cm^2A)	2250 (158.191)
2.	Initial SG Pressure (100% Power), psia (kg/cm^2A)	1000 (70.3071)
3.	Initial Power Level	102%
4.	Initial Core Inlet Temperature, EF (EC)	555 (290.56)
5.	Initial RCS Flow Rate, % of Design Flow	100
6.	Initial Steam Generator Level, % Wide Range	76.4
7.	Backup Heaters Actuation Setpoint Pressure, psia (kg/cm^2A)	2200 (154.676)
8.	SIAS Setpoint Pressure, psia (kg/cm^2A)	1825 (128.310)
9.	Safety Injection Pump Shutoff Pressure, psia (kg/cm^2A)	1825 (128.310)
10.	SBCS Setpoint Pressure, psia (kg/cm^2A)	1100 (77.3378)
11.	AFW Actuation Setpoint, % Wide Range	23.4
12.	AFW Termination Setpoint, % Wide Range	50
13.	MSIS Setpoint - SG High Level, % Narrow Range	90.8
14.	MSSV Setpoint Pressure, psia (kg/cm^2A)	1195 (84.017)
System/Component Capacities		
1.	SGTR Flow, lbm/sec (kg/sec) @ 1800 psia 1 Tube @ 1600 psia 5 Tubes	38 (17.2365) 173 (78.4716)
2.	Safety Injection Flow Per Pump, lbm/sec (kg/sec) @ 1800 psia @ 1600 psia	 25 (11.3398) 36.64 (16.6196)

Table 2

Sequence of Events for One Steam Generator Tube Rupture

Time (Sec)	Event	Setpoint
0.0	Tube Rupture Occurs	--
90	Pressurizer Backup Heaters Actuated on Low Pressurizer Pressure, psia (kg/cm^2A)	2200 (154.676)
1209	Reactor Trips on Hot Leg Saturation Trip Signal	--
1210	Turbine Trips	--
1211	Steam Bypass Control System Actuated	--
1213	RCS Pressure Reaches Safety Injection Actuation Signal (SIAS) Setpoint, psia (kg/cm^2A)	1825 (128.310)
1370	Main Feedwater Terminated on Low T_{avg}	--
2649	Auxiliary Feedwater to Unaffected Steam Generator Actuated on Low Level, % Wide Range Level	23.4
4145	Auxiliary Feedwater to Unaffected Steam Generator Terminated on High Level, % Wide Range Level	50.0
[1]	Main Steam Isolation Signal (MSIS) would have been Generated on High Level in the Affected Steam Generator, % Wide Range Level	94.68
[1]	Main Steam Safety Valves (MSSVs) Open on Affected Steam Generator on High Steam Generator Pressure, psia (kg/cm^2A)	1195 (84.017)

[1] Event does not occur during the 10000 seconds of transient simulation

Table 3

Sequence of Events for Five Steam Generator Tubes Rupture

	Event	Setpoint
0.0	Tube Rupture Occurs	--
13	Pressurizer Backup Heaters Actuated on Low Pressurizer Pressure, psia (kg/cm^2A)	2200 (154.676)
167	Reactor Trips on Hot Leg Saturation Trip Signal	--
168	Turbine Trips	--
171	Steam Bypass Control System Actuated	--
172	RCS Pressure Reaches Safety Injection Actuation Signal (SIAS) Setpoint, psia (kg/cm^2A)	1825 (128.310)
330	Main Feedwater Terminated on Low T_{avg}	--
1726	Main Steam Isolation Signal (MSIS) is Generated on High Level in the Affected Steam Generator, % Wide Range Level	94.68
1920	Main Steam Safety Valves (MSSVs) Open on High Steam Generator Pressure of Affected Steam Generator, psia (kg/cm^2A)	1195 (84.017)

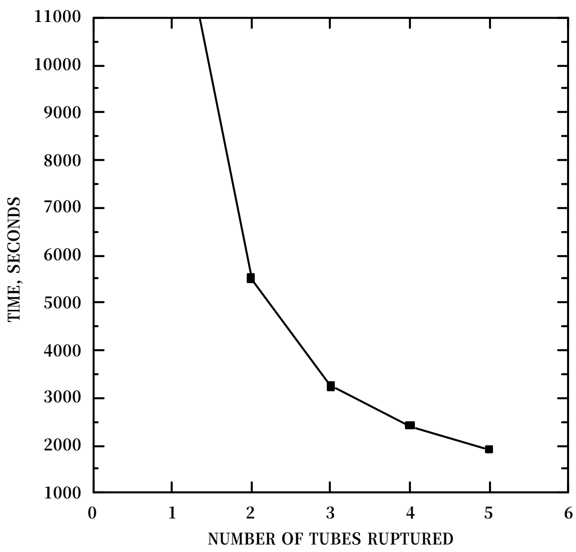

Figure 2. MSSV Lift Time vs. No. of Tubes Ruptured

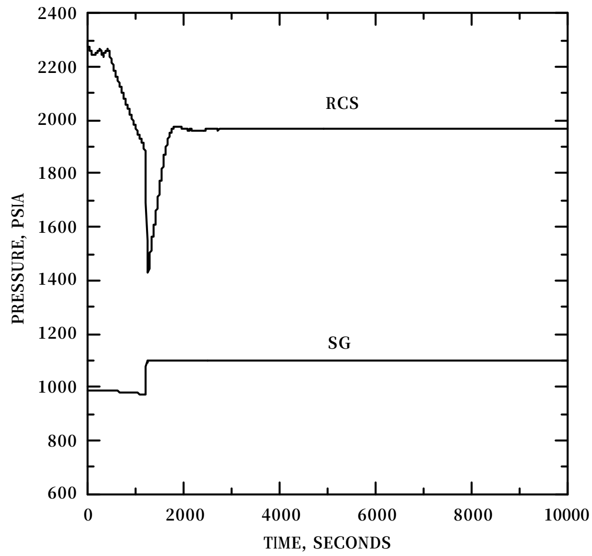

Figure 3. RCS and SG Pressures vs. Time for One Tube Ruptured

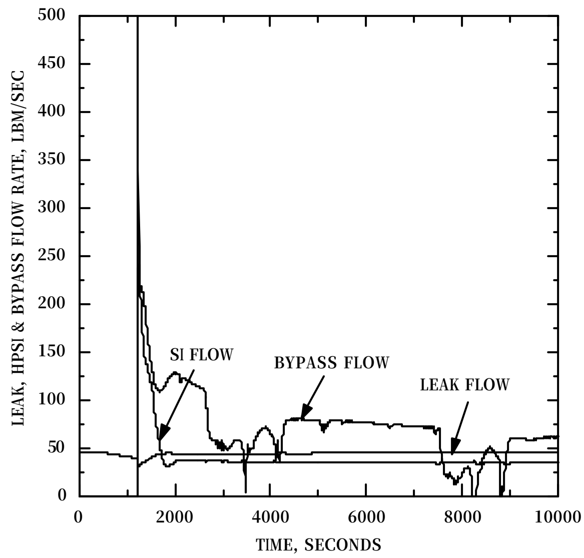

Figure 4. Leak, SI and Bypass Flow Rates vs. Time for One Tube Ruptured

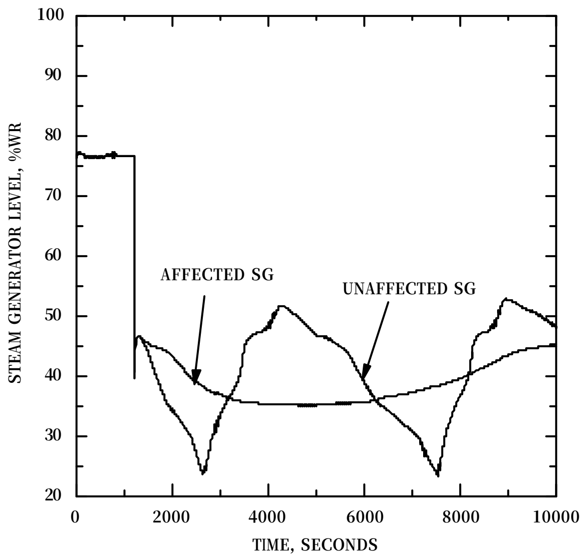

Figure 5. Steam Generator Level vs. Time for One Tube Ruptured

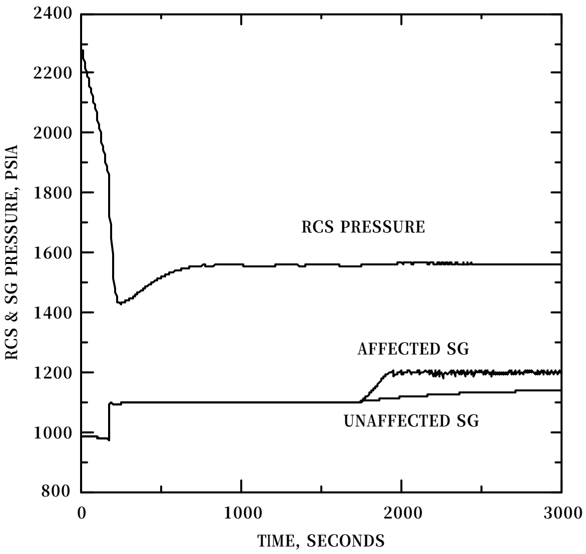

Figure 6. RCS and SG Pressures vs. Time. for Five Tubes Ruptured

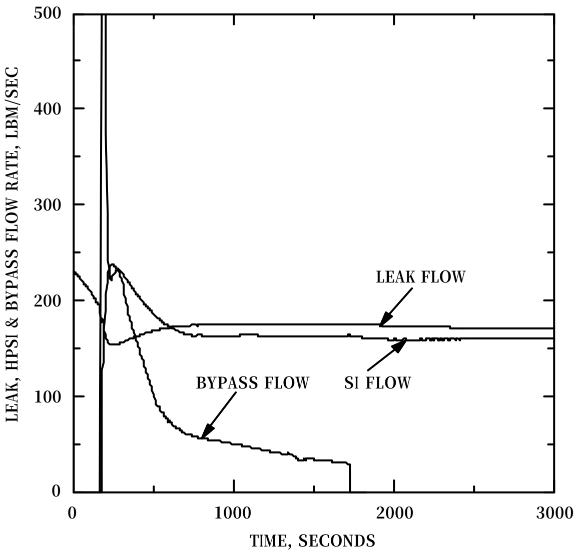

Figure 7. Leak, SI and Bypass Flow Rates vs. Time for Five Tubes Ruptured

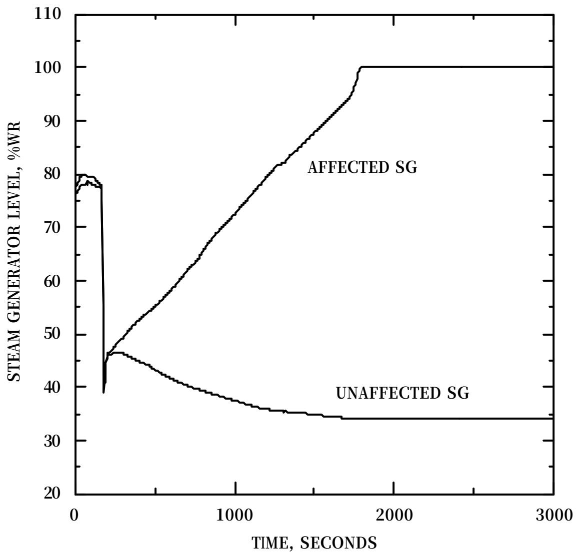

Figure 8. Steam Generator Level vs. Time for Five Tubes Ruptured

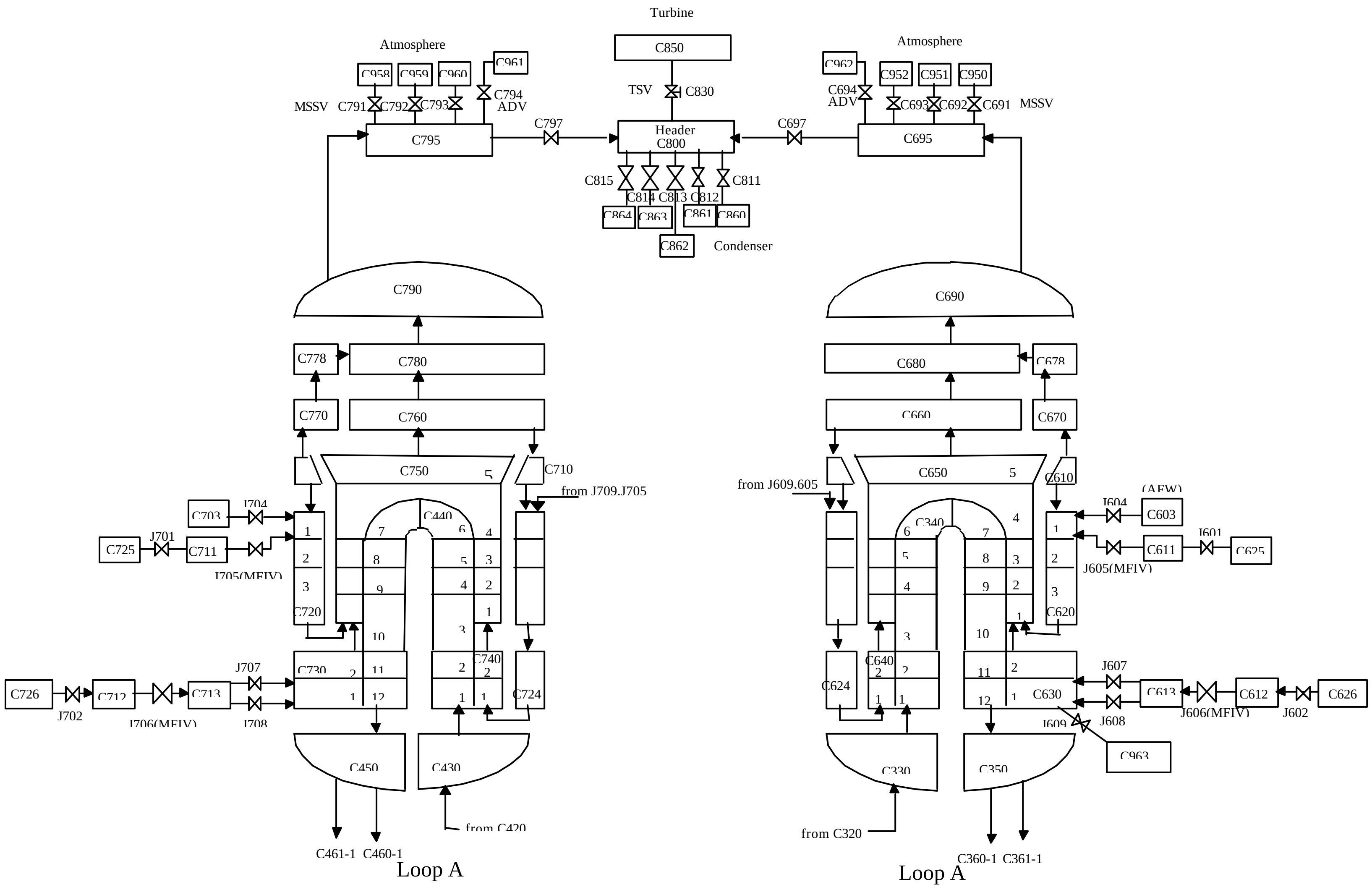

Figure 1. RELAP5/MOD3 Nodalization for KNGR Secondary System