Analysis of Design Changes and Modifications To Enhance Safety Margins At KANUPP – A PAEC/NSS Team Effort

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– ABSTRACT –

Accident analysis for revision of the Karachi Nuclear Power Plant Final Safety Analysis Report has been completed by a cooperative effort between the Pakistan Atomic Energy Commission and Nuclear Safety Solutions. The revision was created to assess design changes and modifications which had been implemented in the plant to support life extension and enhance safety margins. The enhancements were primarily related to safety system response following loss of coolant accident scenarios. In addition, the revision of the safety report included updates to two dual failure accident sections, loss of coolant coincident with loss of emergency injection water, and loss of reactivity control coincident with loss of protection, which had been a part of the original safety report. The paper provides an overview of the design changes which have been implemented and a summary of the updated safety analysis. The new analysis supports full power operation.

1. INTRODUCTION

The Karachi Nuclear Power Plant (KANUPP) is a single unit CANDU PHWR with a total gross capacity of 137 MW. It is located at Paradise Point on the Arabian Sea coast, about 15 miles to the west of Karachi. It has been in commercial operation since 1972. KANUPP is part of Karachi Nuclear Power Complex and is owned and operated by Pakistan Atomic Energy Commission (PAEC).

Safety analysis supporting plant operation is documented in the KANUPP Final Safety Analysis Report (KFSAR). The KFSAR has undergone two major revisions since the original issue. The first revision, which was completed in 2000, updated the analysis using tools and methods consistent with those employed by the Canadian nuclear industry at the time. The scope of the update included all major accident categories covered in the Canadian safety reports at that time. Topical safety analysis issues at the time, such as increased void reactivity uncertainty, were accounted for in the analysis.

As a result of the work, a number of design changes were recommended to KANUPP. On the basis of these recommendations, the Pakistan Nuclear Regulatory Authority (PNRA) requested that modifications and upgrades be incorporated as part of plant re-licensing and life extension. Substantial improvements have therefore been made to the safety systems as part of current re-licensing and life extension efforts. The improvements primarily affect plant response following Loss of Coolant Accident (LOCA) scenarios.

Two dual failures were excluded from the 2000 update. Assessment of LOCA with coincident loss of emergency water injection was excluded because the Canadian methodologies to address this accident scenario were under developed at the time. Also excluded was the assessment of control system failures coincident with loss of protection.

In 2007, a second revision to the KFSAR was completed. The update included assessment of the new modifications and upgrades which had been recommended in revision 1 of the KFSAR. As previously noted, the modifications were intended to enhance plant safety under LOCA conditions and as such, LOCA analysis was re-assessed. Also included in the revision 2 update were the two dual failure accidents, LOCA coincident with loss of emergency injection water and control system failure coincident with loss of protection. PAEC contracted Nuclear Safety Solutions (NSS) to undertake the revision 2 update in partnership with PAEC staff. Technical support for the project was also provided by Atomic Energy of Canada Limited, and John Luxat of McMaster University.

This paper provides an overview of the assessments that were performed as part of revision 2 of the KFSAR.

2. GENERAL FEATURES OF THE KANUPP DESIGN

The core of the reactor contains 208 channels with an approximate fission power of 456 MW at 100 percent full power. Under normal operation, reactor power is controlled by moderator level. At full power the reactor operates with no significant flux tilt due to the small size of the core and the symmetry of the control devices. The maximum time average channel power at full power operation is 2.8 MW for the central channels. The license limit bundle and channel powers are 495 kW and 3.2 MW.

The primary circuit (Figure 1) is a single figure of eight loop with two reactor inlet headers and two reactor outlet headers. There are 4 heat transport pumps and 3 boilers on each side of the core. Pressure and inventory control is maintained by the charging system.

The single means of shutdown for reactor protection is moderator dump. The calandria dumps into a dump space from where it drains into a moderator drain tank.

The moderator cooling system (Figure 2) draws water from the moderator drain tank and supplies water via the moderator heat exchangers to the spray nozzles at the top of the calandria and dump space. Following normal shutdown by moderator dump, the calandria tubes are cooled by these moderator sprays.

The KANUPP containment is a single unit containment design. Under normal operation, release paths to the atmosphere are through controlled discharge from a stack. Following a high containment pressure signal, containment isolates, and discharge is through building leakage alone. The containment is also equipped with a light water dousing system that activates on high containment pressure and high activity in containment.

Emergency injection water can be provided by two separate systems (Figure 3). One is called the Emergency Injection Water (IJW) system. It uses the moderator cooling pumps to supply water from the moderator drain tank to the reactor headers. It is a relatively low pressure system since the moderator pump head is low and begins injection at a reactor outlet header pressure of 90 psig (0.72 MPa(a)). When the level in the moderator drain tank is low, the system switches to recovery mode and draws water from the reactor building sump, via recovery pumps, and returns it to the moderator drain tank for re-injection.

The second emergency injection water system is new and is called the Forced Emergency Injection Water (FIJW) system. The FIJW pumps are separate and dedicated pumps with a head sufficient to provide full flow (about 30 kg/s) at an reactor outlet header pressure of about 300 psig (2.2 MPa(a)). Injection by FIJW is automatically terminated when the heat transport system pressure has dropped sufficiently to allow IJW injection to commence.

2.1 MODIFICATIONS ASSESSED IN REVISION 2 OF THE KFSAR

The upgrades assessed in revision 2 of the KFSAR were:

- addition of redundant Emergency Injection Water (IJW) system valves to provide improved reliability for the IJW system,
- installation of the separate, redundant and diverse medium pressure Forced Emergency Injection Water (FIJW) system to improve injection reliability and provide injection at higher heat transport system pressures,
- installation of Automatic Boiler Crash Cooling (ABCC) Logic to ensure timely heat transport system depressurization and emergency water injection following LOCA scenarios,

- modifications to the operating logic for the Dousing Spray Water (DSW) Cooling system to allow for activation at a lower reactor building pressure, and

- new instrumentation and control upgrades in KANUPP which shortened the instrument delays associated with reactor trip logic.

The new FIJW and the upgraded IJW systems are shown in Figure 3. For comparison, the original IJW system is shown in Figure 4.

3. SCOPE OF ACCIDENT ANALYSIS

Revision-2 of the KFSAR covered analysis of the following single failure events:

- Small Break Loss of Coolant Accident (SBLOCA)

- Large Break Loss of Coolant Accident (LBLOCA)

The analysis also includes dual failure accident scenarios. Dual failures are accidents for which a failure in a special safety system occurs concurrent with a failure in a process system. The following dual failure scenarios are assessed:

- Loss of Coolant Accident (LOCA) coincident with a Loss of Emergency Coolant Injection (LOECI),

- Loss of Reactivity Control (LORC) coincident with a Loss of Protection (LOP), and

- LOCA coincident with a containment impairment, i.e., failure of containment isolation.

4. ACCEPTANCE CRITERIA

There are two types of safety analysis success criteria: radiological dose limits and high-level safe design criteria. The dose limits are prescribed by the Regulator. The safe design criteria are based on past experience and historical precedent, and are codified into regulatory documents governing the shutdown system, the emergency core cooling system, and containment. (In Canada, these were the R7, R8, R9, and R77 CNSC documents.) To demonstrate compliance with a qualitative safe design criterion, the criterion is transformed into a set of quantitative criteria known as “derived acceptance criteria”. Derived Acceptance Criteria (DAC) generally represent sufficient but not necessary conditions for ensuring that the qualitative criterion is met. The specific

DAC's used depend on how the component under consideration is challenged, e.g., channel integrity following a small LOCA or a large LOCA.

4.1 DOSE LIMITS

The public dose limits applied in this project are consistent with the single/dual failure dose limits applied in Canada for reactors of similar age as KANUPP. The dose limits are shown in Table 1.

Table 1- Public Dose Limits

Dose Criteria	Critical Individual (rem)		Population (Person-rem)	
	Whole Body	Thyroid	Whole Body	Thyroid
Single Failure	0.5	3.0	1.0E+04	1.0E+04
Dual Failure	25.0	250.0	1.0E+06	1.0E+06

4.2 SAFE DESIGN CRITERIA

The extent of meeting the safe design criteria depends on whether single failures or dual failures are considered. The general requirements for single failures are:

- A loss of primary heat transport system integrity shall not result from any fuel failure mechanism (effectiveness of SDS).
- For any accidents other than LBLOCA, single channel events, and dual failures, no fuel failures occur (effectiveness of SDS and ECI).
- All fuel in the reactor and all fuel channels shall be kept in a coolable geometry (effectiveness of ECI).
- Containment remains intact.

For dual failures, the general requirements depend on what failures of the protective system are considered:

- For LOP, ECI remains effective and containment remains intact.
- For LOECI, fuel channel and containment remain intact.

4.3 DERIVED ACCEPTANCE CRITERIA

The shutdown system, emergency coolant injection system, and containment are required to perform adequately under accident conditions, in order to ensure that the various barriers that prevent radioactivity from being released to the public remain intact. These barriers are the fuel matrix, the fuel sheath, the fuel channel, and containment. The criteria used to demonstrate that integrity of these barriers is maintained depend on what the challenges are.

Derived Acceptance Criteria for Maintaining Fuel Integrity[1]

a) Prevention of fuel centreline melting. The criterion is met if the fuel centerline temperature does not exceed 2840°C.

Derived Acceptance Criteria for Maintaining Fuel Sheath Integrity

a) Prevention of sustained fuel sheath dryout. For assessment of trip effectiveness, the duration of dryout is less than 60 s. For ECI effectiveness, the duration of post-dryout is taken into consideration for assessment of fuel sheath integrity.
b) Prevention of fuel sheath melting. Sheath melting is precluded if the sheath temperature remains below the melting point for un-oxidized Zircaloy of 1760°C.

Derived Acceptance Criteria for Maintaining Fuel Channel Integrity

a) Prevention of fuel melting. Meeting this criterion ensures the absence of molten fuel in the fuel channel. The presence of molten fuel is a threat to fuel channel integrity, as molten UO_2 ejected from a fuel element could locally overheat the pressure tube and ultimately cause failure of the fuel channel.
b) Prevention of fuel sheath melting. The presence of molten fuel sheath potentially threatens the integrity of the fuel channel, as contact of molten sheath material with the pressure tube may cause fuel channel failure.
c) Prevention of sustained fuel sheath dryout. Sustained fuel sheath dryout can lead to fuel sheath failures, and possibly to distortion of the fuel elements such that they contact the pressure tube, and cause it to fail if the PHT system is pressurized. This criterion is applied in small LOCA. This criterion is not relevant to fuel in the affected channel for single channel events, or to transition or large LOCA events, for which periods of dryout are limited to very short durations due to an early reactor trip.
d) Constrained axial expansion of the fuel does not occur.

[1] This criterion does not apply to the fuel in the affected channel in a single channel event.

5. COMPUTER CODES

The computer codes applied in the analysis are listed below:

- Thermal hydraulic analysis of the full heat transport system, secondary side, charging system, and injection water systems was performed using the two-fluid code TUF.

- Reactor physics analysis was performed using the WIMS and RFSP codes.

- Detailed analysis of the fuel temperatures and thermal hydraulic conditions in fuel channels was performed using the codes FACTAR_SS and FACTAR (LOCA) for scenarios where fuel bundle slumping and pressure tube sagging were precluded. For scenarios where fuel bundle slumping and pressure tube sagging were expected, the CHAN-II code was applied.

- The radiological source term from the fuel was determined using the ORIGEN and SOURCE codes.

- Transport of radionuclides within containment was modeled using the SMART code, with the containment pressure and hydrogen concentration predicted using PRESCON2.

- Public dose was assessed using PEAR.

- For dual failure scenarios resulting in channel failure, conditions in the calandria were assessed using the CANDUFR code.

The major computer codes (TUF, WIMS, RFSP) used during the project have been verified and validated in accordance with the requirements in Generic Action Item 98 G02, Regulatory Guide G-149 and CSA standard N286.7. Other codes that are used in the safety analysis (FACTAR_SS, FACTAR (LOCA), ORIGEN, SOURCE, PRESCON2, SMART, PEAR, CHAN-II (MOD 7) and CANDUFR) have been verified and validated against experiments.

These codes have been used by major owners of CANDU reactors in Canada.

6. TOPICAL ISSUES

The following issues are new elements of safety analysis that were not considered in the original KANUPP analysis, or have evolved since the last update. In general, conservative assumptions are made to compensate for any uncertainties.

1. Void reactivity bias – a bias value of -1.3 is applied to the void reactivity calculated by the WIMS code based on the results of validation exercises. Prior to applying the bias correction, the reference void reactivity is 11.04 mk. The value used in the analysis, after applying the bias, is 9.74 mk. The reference value is based on the moderator and PHT coolant isotopic purity of 99.40 and 98.0 wt% respectively.

2. Fuel string relocation reactivity – KANUPP is fuelled against the flow. Therefore, following a large break upstream of the core, the fuel string will be driven towards the inlet end. Fuel containing a higher concentration of fissile material is shifted towards the centre of the core, thereby increasing core reactivity. The maximum amount of reactivity added is 1.002 mk, over a period of 83 ms following a guillotine break of the reactor north inlet header. This effect is accounted for in the analysis of large break LOCA.

3. Constrained fuel expansion – The analysis determines the extent of axial fuel string expansion due to overheating, and compares the expanded fuel string length with the minimum available axial gap. The issue is that if the fuel string expands to take up the full gap, fuel and fuel channel integrity may be threatened due to the increased potential for PT overheating due to fuel element contact. The analysis shows that there is a gap remaining even for the most limiting temperature transient.

4. Reverse flow impact – The analysis accounts for reverse flow impact of the fuel string following a large break upstream of the core. It is shown that the impact velocity is sufficiently low that no damage to the fuel or end fitting is expected.

5. Presence of fission products in the PHT system – Fuel may become defective during normal operation. When defective fuel is present, a spike of iodine is released into the PHT system following a reactor trip. The analysis accounts for the potential presence of defective fuel and the additional release of activity following reactor trip.

6. Critical heat flux correlation – KANUPP fuel is of the 19-element design. The analysis uses the critical heat flux correlation that is currently used by Ontario Power Generation for 28-element Pickering fuel, and includes a penalty of 0.21% to allow for uncertainties. The validity of this assumption has been confirmed via an assessment of experimental CHF data for 19-element fuel. The assessment shows that use of the 28-element correlation is conservative for prediction of fuel sheath dryout temperatures.

7. Long term hydrogen production: The analysis accounts for hydrogen production from metal-water reaction. Hydrogen is generated in this manner only during the period immediately following the accident when the fuel temperatures are high. The metal water reaction is terminated as the fuel temperatures fall in the longer term. In the longer term, hydrogen may be produced from other sources such as radiolysis of recovery sump water and calandria water, and corrosion of galvanized piping and paint. Hydrogen generation from these longer-term sources occurs at a much slower

rate than the short-term metal water reaction. The greatest challenge to containment due to hydrogen production in the short term is therefore from the metal water reaction. Longer-term post accident management of the hydrogen concentration in containment is not included in KFSAR-R2.

7. SUMMARY OF RESULTS

This section summarizes the key results of the analysis. The focus is on predicted doses and comparison with the limits, and on the extent to which the derived acceptance criteria are met. Margins to the dose limits and derived acceptance criteria are also discussed. Unless indicated otherwise, the results discussed below are for a total thermal power of 100% FP (97% indicated power + 3% uncertainty).

7.1 SHUTDOWN SYSTEM EFFECTIVENESS

The moderator dump shutdown system, together with the set of trip parameters available to detect the need for a reactor trip, is an effective shutdown system. The event that poses the greatest challenge to the shutdown system is a large break LOCA. The analysis shows that for the limiting large break LOCA that imposes the greatest demand on the shutdown system (a 100 % inlet header break), the maximum fuel centre line temperature is less than 2200°C which is well below the UO_2 melting point of 2840°C. Expressing the results in terms of fuel enthalpy, the maximum fuel enthalpy at 5 seconds following the break is 651 kJ/kg of UO_2, which is well below the 960 kJ/kg of UO_2 criterion applied in Revision 1 of the KFSAR. The fuel enthalpy is below that reported in Revision 1 of the KFSAR. One of the factors which has contributed to the reduction in fuel enthalpy is the instrumentation improvements discussed below.

The new instrumentation for reactor shutdown reduces the delays associated with the reactor trip instrumentation from 933 ms to 807 ms. This improvement is relevant only to fast reactivity transients such as those predicted following a large break LOCA.

For small LOCAs, both the primary and backup trips are initiated prior to onset of fuel sheath dryout. The saturation pressure of the PHT system is below the PHTS pressure low trip setpoint. Therefore, an early primary outlet header pressure low trip occurs for all LOCA events. An early reactor building high pressure trip also occurs for all except very small LOCA's. Hence, there are two very early process trips for small LOCA events.

The revision 2 results indicate that the shutdown system is effective following LOCA transients. The absence of in-core devices for controlling the spatial flux distribution[2] means that variations in initial flux shape will be dominated by effects such as fuelling and xenon. The effects of these phenomena are covered by assuming that the maximum channel power is at the licensing limit.

7.2 IJW AND FIJW EFFECTIVENESS

Following a large break LOCA, depressurization of the heat transport system is relatively fast, resulting in early IJW injection with, depending on the scenario, little or no contribution from the FIJW system. The effectiveness of IJW is therefore discussed primarily with reference to large break LOCA analysis.

For small LOCA scenarios, the depressurization is slower, generally resulting in operation of the higher pressure FIJW system for an extended time prior to IJW operation. The effectiveness of FIJW operation is therefore discussed primarily with reference to small LOCA analysis.

Large Break LOCA

Analysis of the critical large break LOCA showed that the fuel sheath temperatures are highest immediately following the break as a result of degraded flow. The maximum temperature reached is relatively low (below about 1100°C). Thereafter the sheath temperatures fall as stored heat is removed from the channel. At low system pressure IJW injection begins ensuring continued effective cooling of the fuel.

During the transient the PT temperatures do not exceed 700ºC and the maximum PT strain remains below 1%. At the time the PT reaches the maximum temperature of 685°C the heat transport system pressure is relatively low, less than 2.5 MPa, so PT ballooning into contact with the CT does not occur.

The IJW system is effective in providing long-term cooling to the core, and the fuel remains in a coolable geometry with no additional fuel failures. Different impairments in IJW system valves are also analyzed at 100%FP. Neither increased fuel sheath failures nor PT ballooning is predicated in any of the degraded conditions.

[2] KANUPP has no liquid zone controllers or adjusters. Boosters are removed from the core. The four absorber rods are inserted uniformly in the core, so there are minimal flux distortions associated with their use.

Small Break LOCA

Analysis of the base cases of small break LOCAs showed that the FIJW system along with ABCC (Automatic Boiler Crash Cooling) is effective in arresting fuel sheath temperature to less than 600°C for all break discharges in the range of approximately 40 kg/s to 500 kg/s (maximum sheath temperature of 552°C for the case of 500 kg/s initial discharge). For breaks below approximately 40 or 50 kg/s, automatic FIJW operation is blocked due to the slow system de-pressurization, i.e., the 800 psig initiating signal required for operation is reached after the 95 second window allowed by the logic. For these small break sizes, the fuel sheath temperatures remain below 600°C until the initiation of the low pressure IJW system, which brings down the fuel temperature further.

Various impairments and degraded conditions in the IJW/FIJW systems are analyzed at 100%FP. Analysis showed that in the absence of the FIJW system the fuel sheath temperature exceeds 600°C for breaks in excess of 100 kg/s (maximum 780°C for 100 kg/s and 873°C for 500 kg/s). Similarly, if ABCC is unavailable, the fuel sheath temperatures exceed 600°C (maximum 800°C) for the case of 100 kg/s initial discharge[3]. Thus, the maximum fuel sheath temperature is about 800°C in both impairment cases. However, detailed analysis using FACTAR shows that there is no fuel failure for these impairment cases.

The IJW system is therefore capable of providing the necessary core cooling following crash cooling of the boilers and termination of the FIJW system (if operated).

The design of the FIJW and IJW systems therefore meet the safety design objectives for small LOCAs, as the fuel is maintained in a coolable geometry with no fuel failures[4], and fuel channel integrity is assured. The new FIJW system limits fuel heatup for breaks at the intermediate range of small LOCAs, and therefore improves the overall effectiveness of emergency core cooling at KANUPP.

7.3 CONTAINMENT INTEGRITY

Maintaining containment integrity is part of the safe design requirements. The analysis shows that containment integrity is not jeopardized either due to peak pressure or hydrogen combustion. This is mainly due to the fact that KANUPP containment is a large single volume having a well-mixed environment and relatively high design pressure. The peak containment conditions for different LOCA scenarios are shown in Table 2. The pressures are all well below the containment design pressure of 27 psig.

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[3] Manual crash cooling was credited after 15 minutes.
[4] Excluding the affected channel for single channel events.

Table 2- Containment Conditions after Various Accidents

Break Type	Containment Status	Peak Pressure (psig)	Peak Temperature (°F)	Time to Peak Pressure (seconds)
LOCA (100% NIH)	Intact	15.8	207	33.3
LOCA (24 % NIH)	Intact	12.3	195	73.9
LOCA (24% NIH)	Isolation Fail	11.5	193	65.4

7.4 LOCA COINCIDENT WITH LOECI

The emergency injection water failure modes considered in the LOECI assessment result in one of two possible outcomes: either only FIJW is available, or only calandria spray cooling is available to remove heat from the core.

Only FIJW Available

For the critical large break LOCA, analysis showed that the FIJW system alone is capable of cooling the fuel and limiting fuel sheath failures to levels similar to those predicted for the single failure critical break large LOCA. After the initial power transient the maximum fuel sheath temperature remains around 700°C for about 300 seconds. Thereafter, continuous FIJW flow reduces the temperature to below 400°C after about 900 seconds. The consequences for scenarios with only FIJW available are therefore similar to those for the single failure LOCA cases.

For in-core small break LOCAs, consideration was given to the effect of filling the calandria with FIJW coolant. The analysis demonstrated that there is no concern with re-criticality. Similarly, with respect to flooding of the boiler room, continuous operation of the FIJW system, in the absence of IJW cooling and re-circulation, is of no concern as long as containment remains intact and the required amount of cooling water is available at the FIJW pump suction.

Only Calandria Spray Cooling Available

This scenario assumes the complete loss of injection to the core. The core heats up and the PTs eventually sag into contact with the CTs. Under these conditions, hydrogen production and numerous sheath failures are expected, but channel integrity is maintained due to effective calandria spray cooling as discussed below.

Calandria spray cooling provides effective heat removal from the core because there is no significant boil off of the spray, sustained CT dryout does not occur, and the calandria tubes remain sufficiently wetted to allow for effective removal of decay heat.

For the most limiting conditions with respect to hydrogen production, the quantity of hydrogen produced is not significant, resulting in a concentration of hydrogen in containment of less than 0.01% volume. Containment integrity is therefore not challenged by hydrogen combustion.

7.5 LORC COINCIDENT WITH LOP

The most limiting event analyzed for LORC events coincident with LOP was an uncontrolled moderator pump up during startup. The uncontrolled pump up of moderator during startup results in a power excursion which is terminated when fuel channels start to fail resulting in discharge of primary heat transport system coolant into the calandria, which then displaces the moderator to initiate reactor shutdown. The mode of pressure tube and calandria tube failure is rapid local strain due to molten fuel relocation onto the pressure tube. Molten fuel relocation is assumed to occur at a radially averaged fuel enthalpy of about 1300 kJ/kg of UO_2. This occurs at about 32 seconds. Four lead channels are assumed to fail simultaneously in the assessment of moderator response leading to reactor shutdown. Later failures of some channels due to overheating and deformation may also occur following the power excursion. In addition to the four lead failure channels, 35 channels were assessed to have high probability of having molten fuel, and 39 more channels were assessed to be the upper-bound of any channels that may have molten fuel. Therefore, a total of 78 channels are included in the calculation of hydrogen generation and fission product releases under molten fuel conditions.

A consideration of the analysis is the loading on the calandria due to the pressure pulse created by the channel failures. The discharge of high enthalpy fluid and molten fuel into the moderator creates a large two-phase region which results in pressurizing the calandria. For the base case scenario, less than four bundles in each failed lead channel have molten fuel. It was assumed that 20% of the fuel in the four bundles which have some degree of melting was discharged as molten material into the moderator at the time of the lead channel failures. Due to the large energy deposition into the moderator, the calandria pressurized up to 2.66 MPa with an impulse loading of 0.2 MPa-s. For comparison, the calandria collapse pressure, and the limiting value of the impulse loading are 3.5 MPa and 0.688 MPa-s respectively. Therefore, the calandria does not fail due to the initial pulse.

Sensitivity cases were performed to determine the effect on calandria integrity of the assumed number of lead channel failures, and the assumed amount of molten fuel. The calandria collapse pressure was exceeded if the number of channel failures increased to 6 with the molten fuel discharge fixed at 20%. With respect to molten fuel, holding the number of lead channel failures at 4, the molten fuel discharge was increased to 30% prior to exceeding the collapse pressure of the calandria.

Hydrogen production was assumed in 78 channels at a rate equal to the production rate in the highest power channel. Hydrogen concentration in containment was then predicted by PRESCON2 to be about 2.33% by volume. The hydrogen concentration is therefore well within the lower hydrogen combustion limit of 4% by volume.

7.6 PUBLIC DOSES

For single failure events, fuel failures were predicted only for large break LOCA scenarios. Single channel small LOCA events were not considered in the new analysis since the source term for these events are essentially unaffected by the design modifications evaluated in the revision 2 KFSAR update.

The single failure large LOCA results have improved from that predicted in revision 1 of the KFSAR (Table 3). This is in part due to a reduction in the source term. The revision 1 analysis conservatively predicted fuel failures in all elements in 25 channels whereas the revision 2 analysis conservatively predicted fuel failures in all elements in 18 channels. The reduction in source term is primarily due to improved analysis tools and methodology. However, a more significant contributing factor to the reduction in doses is the improved operation of the Dousing Spray Water (DSW). The DSW setpoint has been lowered from 17 psig (used in KFSAR Revision 1) to 11 psig. In addition, a fixed delay has been removed from the control logic, allowing for automatic operation of DSW immediately after reaching the setpoint.

The results for the dual failure containment impairment case are shown in Table 3 for the limiting containment impairment. As shown in Table 3, there are large margins between the predicted doses and the dose limits.

For the other dual failure events, fuel failures were assessed as follows:

- In LOCA coincident with LOECI, all sheaths in the equivalent of 127 fuel channels failed.
- In LORC due to uncontrolled pump up of the moderator during startup coincident with LOP, all sheaths in 152 fuel channels initially submerged in moderator fluid failed; in addition, 78 channels have molten fuel.

The resulting doses are summarized in Table 3.

Table 3 - Predicted Doses with Reactor Thermal Power at 100% FP

Event	Dose Limit	Critical Individual (% Dose Limit)		Population (% Dose Limit)	
		Whole Body	Thyroid	Whole Body	Thyroid
Large LOCA	Single Failure	3.84 (11.9)	6.30 (11.3)	1.11 (11.0)	8.14 (39.4)
Large LOCA with Containment Impaired	Dual Failure	0.06 (1.0)	0.06 (1.2)	4.64 (3.4)	57.72 (43.7)
LOCA plus LOECI	Dual Failure	1.44	0.99	0.29	1.30
LORC plus LOP	Dual Failure	39.89	38.40	3.53	25.30

Note: Bracketed numbers have been extracted from the revision 1 update for comparison. Dual failures were not included in the revision 1 update and are therefore not shown. For the LOCA results, an increase in population between the revision 1 and revision 2 predictions has resulted in higher population doses for the impaired containment case in which dose is dominated by release from the stack.

8. CONCLUSIONS

Revision 2 of the KFSAR was successfully completed by a joint effort from PAEC and NSS staff. The following conclusions were reached:

- The safety analysis supports operation up to 100 % thermal power (97 % indicated power plus 3 % uncertainty allowance) with a maximum channel power of 3.2 MW and a maximum bundle power of 495 kW.

- Public dose limits are met for all design basis events, including dual failures involving impairments of safety systems (the shutdown system, the emergency coolant injection system, and containment).

- Safe design criteria are met by demonstrating that derived acceptance criteria to maintain integrity of fuel, fuel sheath, fuel channel, and containment are met with large margins.

- The moderator dump shutdown system is effective in shutting down the reactor for the LOCA that imposes the greatest demand on the shutdown system among all design basis events. The assessment has conservatively accounted for fuel string relocation and other conservative assumptions such as using the lower moderator and heat transport isotopic limits, and the presence of boron in the moderator.

- Short-term hydrogen concentration in all analyzed scenarios is well below the lower limit of flammability of 4 % by volume.

In summary, the KFSAR has been revised using methods consistent with those currently applied in Canada. The revision was created to assess design changes that had been implemented in the plant to support life extension and enhance safety margins. The enhancements were primarily related to safety system response following loss of coolant accident scenarios. In addition, revision 2 of the safety report included updates to two dual failure accident sections, loss of coolant coincident with loss of emergency injection water, and loss of reactivity control coincident with loss of protection, which had been a part of the original safety report. The new analysis supports full power operation.

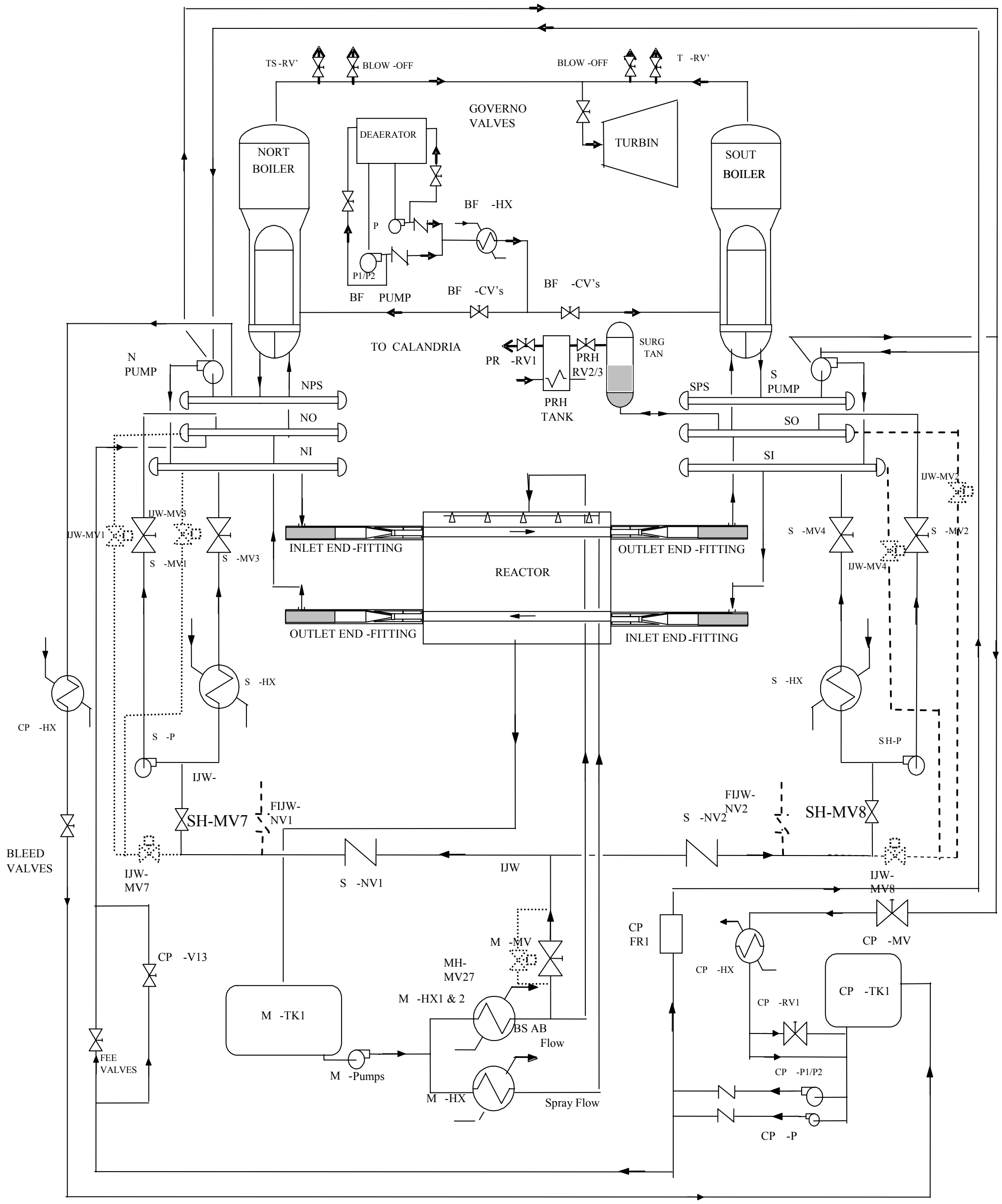

Figure 1. Schematic of Primary Heat Transport System, Secondary Side Systems and Charging System

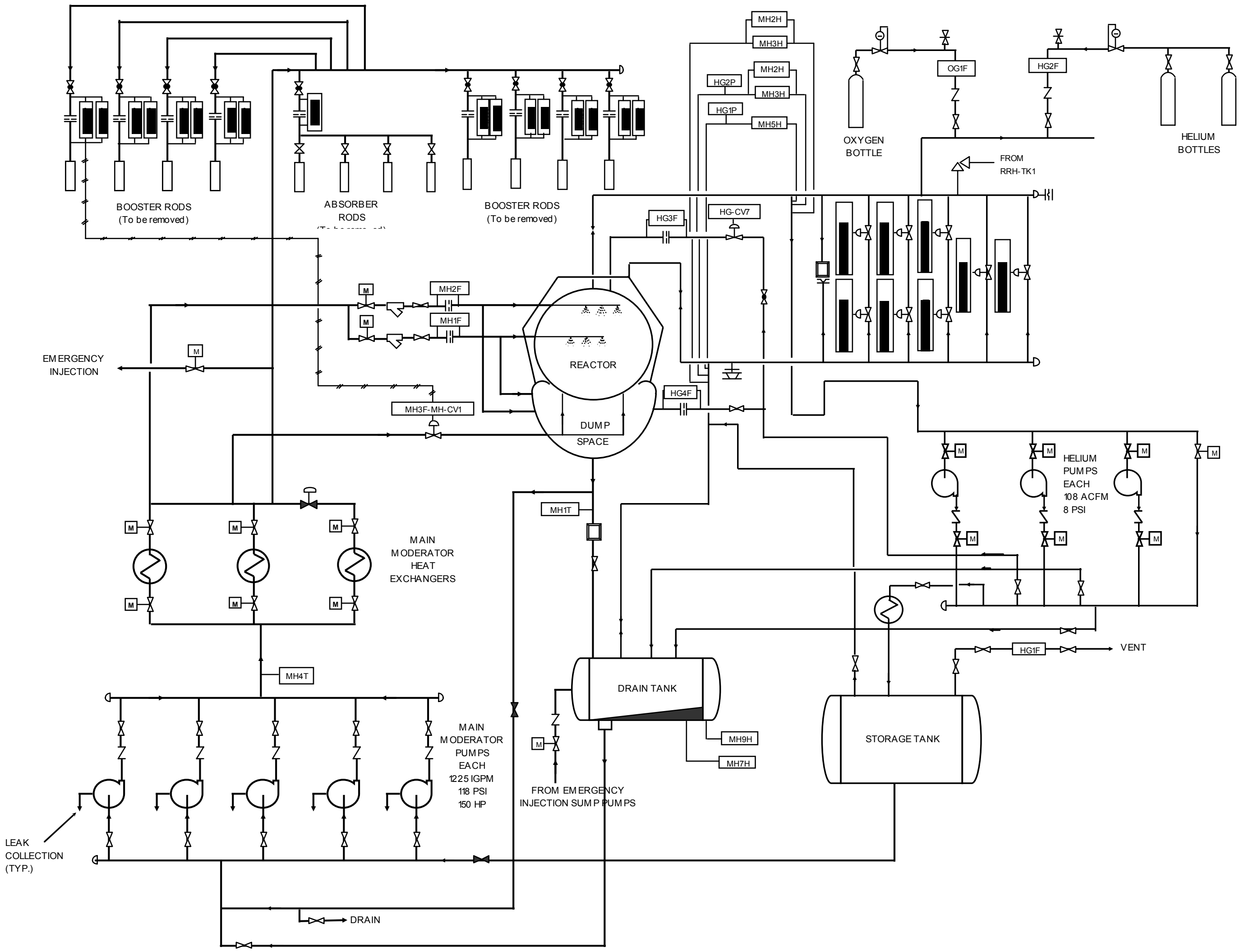

Figure 2. Moderator and Helium System

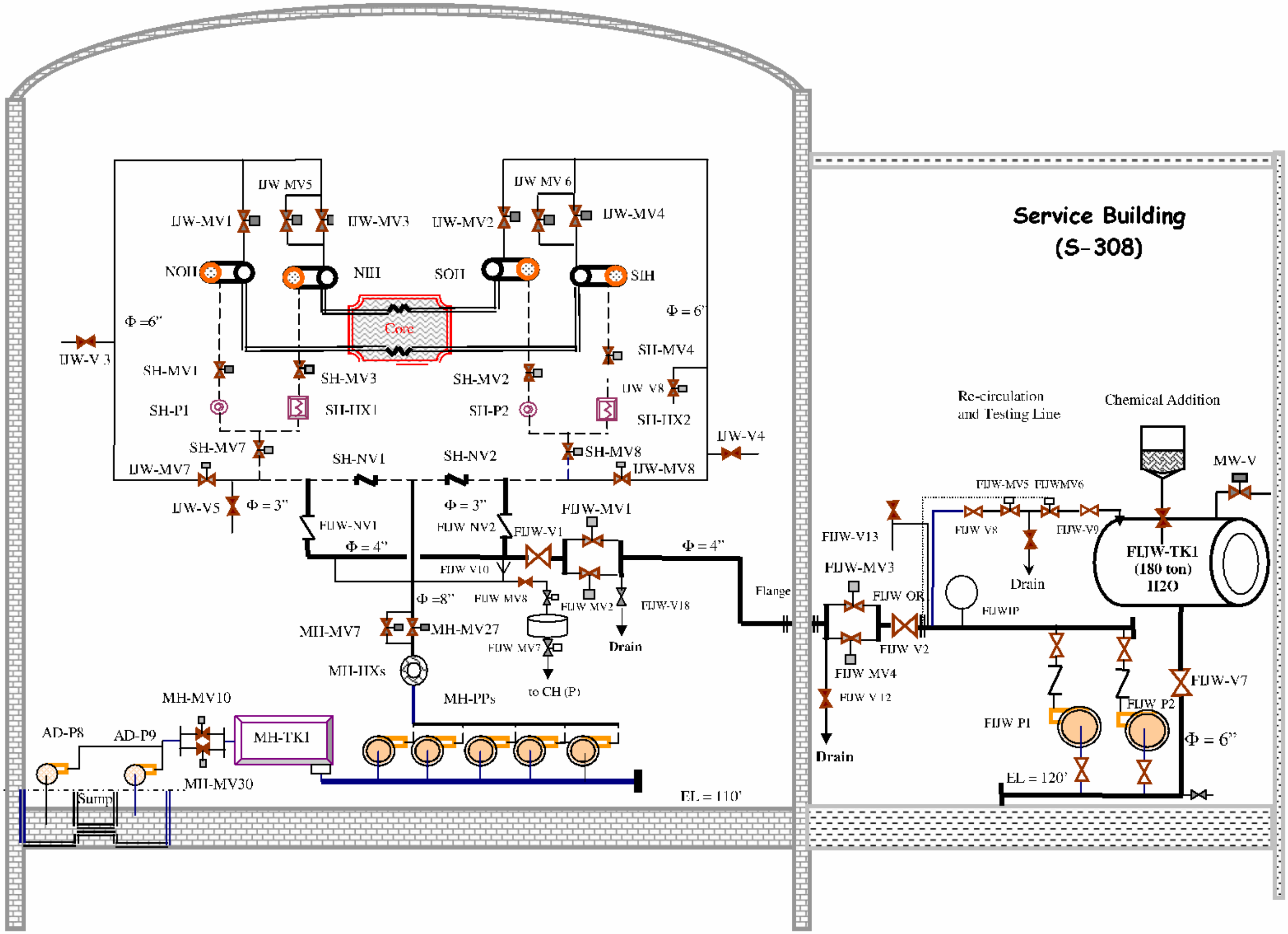

Figure 3. Upgraded IJW and FIJW systems

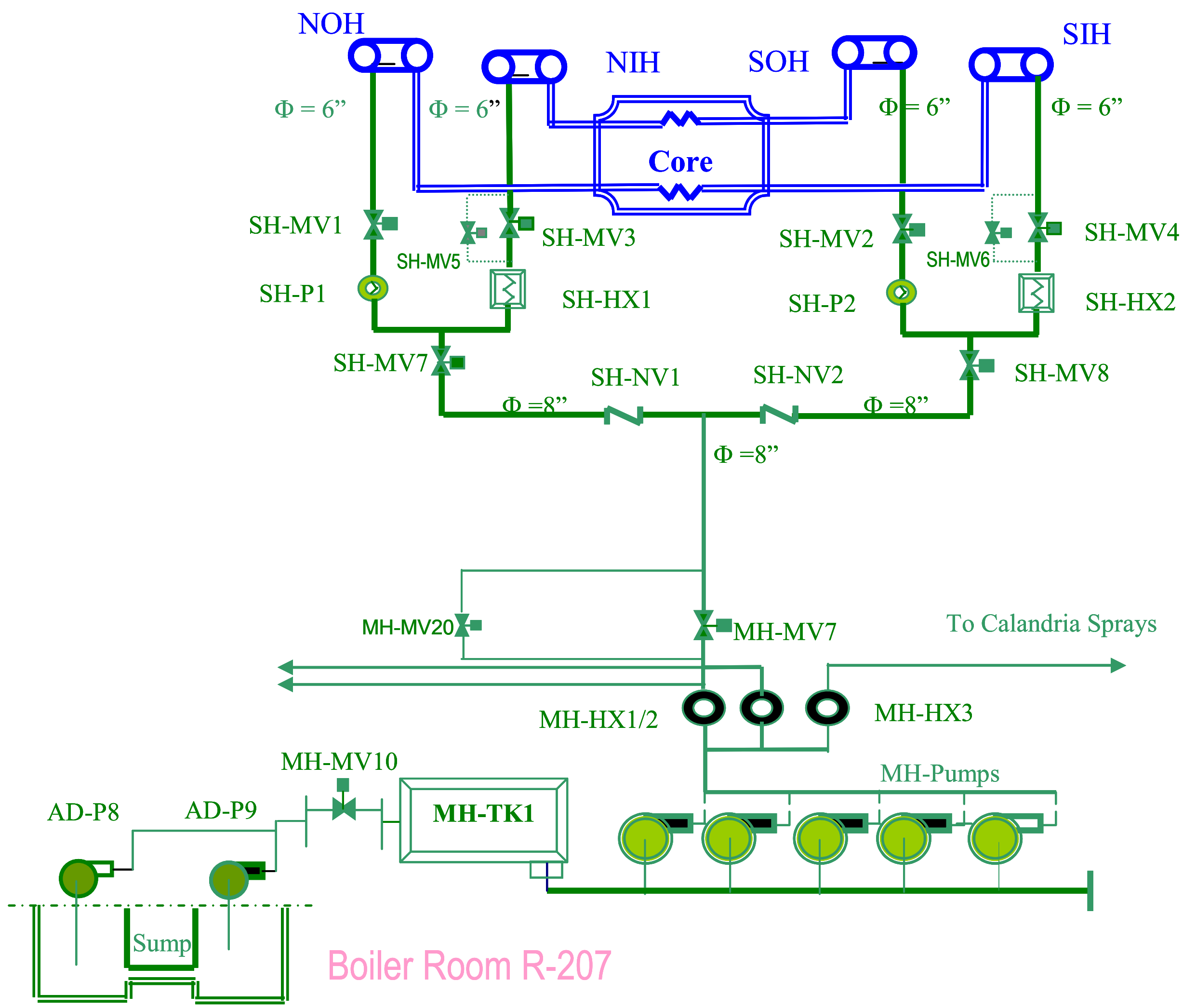

Figure 4. Original IJW Design