THE RELEVANCE OF LOAD FOLLOWING CAPABILITY OF NUCLEAR POWER PLANT FOR THE SOUTH AFRICAN GRID

By:

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ABSTRACT

The main focus of this study is to evaluate whether load following operation can be performed using the existing South African nuclear power plant. Of which, knowing that will assist in determining whether the addition of load following capabilities on the planned nuclear fleet is justifiable or not. In this report the relevance of Koeberg Nuclear Power Plant to adapt to the demand is examined and the effects on plant operation simulated. The report analyses the operation of the existing nuclear power plant (Koeberg units) in South Africa and describes the regulations that govern safe operation of the plant. The Koeberg plant is analyzed based on the current design i.e. operating as a base-load station. This allows a prediction of the Koeberg plant response to big load variations. The simulation results of the load variation are analyzed and the results used to make the conclusion that the Koeberg units are not capable of load following safely. Modifying the Koeberg units from being base load station to load following will require changing the Safety Analysis Report and therefore affect the Koeberg license NIL-01.

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NOMENCLATURE

ARCC	– Advanced Rod Cluster Control
BOL	– Beginning of Life
CRDM	– Control Rod Drive Mechanism
DNBR	– Departure from Nucleate Boiling Ratio
DoE	– Department of Energy
ECCS	– Emergency Core Cooling Systems
EDF	– Electricité De France
EOL	– End of Life
FO	– Flexible Plant Operation
IAEA	– International Atomic Energy Agency
IRP	– Integrated Resource Plan
KSAR	– Koeberg Safety Analysis Report
LOCA	– Loss of Coolant Accident
LPD	– Linear Power Density
NNR	– National Nuclear Regulator
NPPs	– Nuclear Power Plants
NSSS	– Nuclear Steam Supply System
OTS	– Operating Technical Specification
PCI	– Pellet Clad Interaction
PWR	– Pressurized Water Reactor
QPTR	– Quadrant Power Tilt Ratio
RCCS	– Rod Cluster Control System
RCS	– Reactor Coolant System
RIL	– Rods Insertion Limits
SA	– South Africa
SAR	– Safety Analysis Report
SCRAM	– Safety Control Rod Axe Man
SDM	– Shutdown Margin

Glossary of Nuclear Power Plant Terms

This section defines some of the terms that are used frequently in this report and are specific to the nuclear power generation industry.

Burnup: The energy produced by the fuel expressed in megawatt days per metric ton of uranium (MWD/MTU).

End of Life (EOL): Period during nuclear fission when there is not enough excess reactivity in the core to maintain the reactor at full power and nominal temperature.

Extended Low Power Operation (ELPO): Continuous operation below 87%Pn for a period of greater than 12 hours per 24 hours of power operation.

House Loading: The action whereby the Nuclear Power Plant automatically disconnects its turbine from the national electrical grid and only supplies its own auxiliaries.

Linear Power Density: Power generation per unit length of fuel element, measured in W/cm.

Nuclear Steam Supply System (NSSS): The reactor and the reactor coolant pumps, the steam generators and associated piping in a nuclear power plant used to generate the steam needed to drive the turbine generator unit.

Pellet Clad Interaction (PCI): The mechanical contact between the fuel pellets and the Zirconium-alloy cladding.

Safety Control Rod Axe Man (SCRAM): Manual or automatic reactor trip

Shutdown Margin (SDM): The amount of reactivity by which the reactor is subcritical from a given state.

CHAPTER 1: Introduction

1.1 Background

In South Africa, nuclear power plants (NPPs) were introduced into the electrical production sector in 1984 & 1985 as base load sources of electricity (Eskom, 2014). With the change in electricity demand and usage patterns, there is a growing need for NPPs to follow the load demand. The capability of the NPPs to follow load is more prevalent in countries that rely on nuclear power and in countries which are increasing the renewable energy share in their energy mix.

In this report, the term “Load following” and “Flexible Operation” will be used interchangeably to cover operations such as:

- Abrupt Load Change Operation – Load rejection at any power level, turbine trip without reactor trip, turbine generator runback to house load.
- Daily Load Following Operation – a typical daily cycle is 100-50-100 (percent power) in a 14-2-6-2(hr) pattern.

- Frequency Control Operation:
    - About 2.5% power changes without control rod movement, for local frequency control.
    - About 5% power changes with control rod movement for grid frequency changes.

- Extended Low Power Operation – operating the reactor at reduced power due to low electricity demand i.e. operating the reactor below 87% power for a period of greater than 12 hours per 24 hours of power operation. This could be caused by change of season or a change in the region’s energy mix i.e. increased input from renewables or reduced demand.

The existing rules and regulation in SA do not support load following since such an option was omitted during the designing, specification and licensing of the Koeberg Nuclear Power Plant (KNPP). At the time, it was not envisaged that KNPP would require load following capabilities because its purpose was to provide grid stability in the Western Cape since the fossil-fuel power stations were too expensive with too many challenges to be viable in Western Cape. At the time, the costs to include load following capabilities could not be justified. Therefore the daily load cycling of NPPs was not permitted by the National Nuclear Regular (NNR), although the Koeberg design can accommodate infrequent load transients caused by disturbances in the national grid.

1.2 Current condition of the South African energy sector

According to IRP2010, South Africa is planning to increase the nuclear share in the energy mix to 23% and also introduce a significant amount (43%) of renewable energy into the energy mix (DoE, 2010). This is perfectly in line with the world's developments in the energy sector, as indicated in the International Energy Agency in Figure 1 below (IAE, 2010).

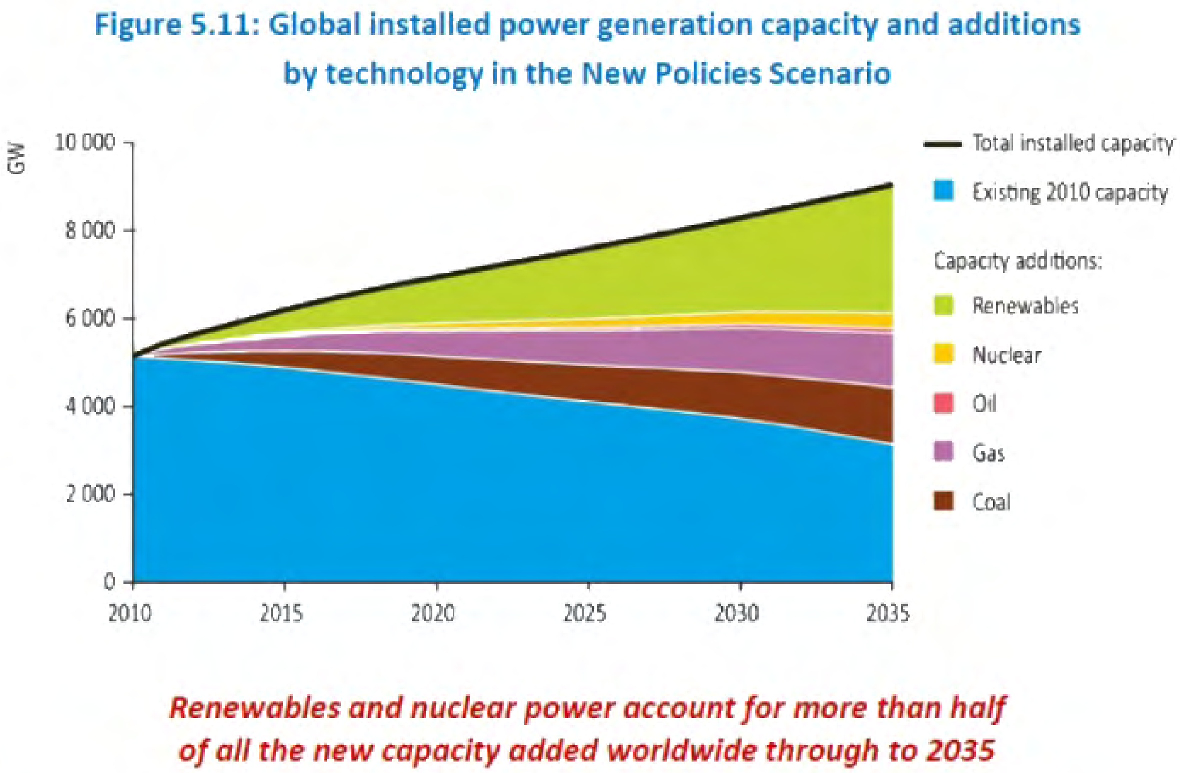

Figure 1: Global installed power generation capacity and additions by technology in the New Policies Scenario

Although coal remains dominant as a source of electricity generation, a national and international policy shift towards low carbon technologies is evident. The global projections are that the shift to nuclear, renewables and other low-carbon technologies will reduce the amount of CO_2 emissions considerably. The South African government committed to reduce the country's emissions by 34% by 2020 and 42% by 2050 (IAE, 2010). The adoption of the 2015 Paris Agreement, under the United Nations Framework Convention on Climate Change, is another platform to encourage the use of low-carbon technologies by using legally binding agreements. The purpose of the agreement is to enhance action prior to 2020 by implementing possible mitigation efforts as stipulated in the Kyoto Protocol and the Doha Amendment to the Kyoto Protocol[1] (COP21, 2015).

The Department of Energy is in a process of publishing the updated Integrated Resource Plan which will create a policy certainty in the energy space. This will clarify South African government's position on nuclear fleet expansion.

1.3 South Africa's Nuclear Build Programme

South Africa is preparing to undertake a possible nuclear plant building project to add 6900 MW of nuclear energy into the energy mix (Government, 2016). With the overnight capital cost in 2015 estimated between \$2021/kW_e in Korea and \$6215/kW_e in Hungary, a large capital investment will be required (WNA, 2016). This initiative comes at a time where neither the SA government nor Eskom utility is in a sound financial position to fund such a project. Cost minimisation therefore should be one of the main priorities of the South African nuclear build programme.

At the moment, in SA, there are two Pressurised Water Reactors (PWRs) operating as base load stations i.e. the two Koeberg operating units, each producing ~970MWe[2]. These reactors were not designed to respond to changes in load demand, their design purpose was to stabilise the grid and therefore required as base load units[3]. The decision for the Koeberg units to operate as base load stations was based on the fact that their total output capacity in the grid was a small percentage (<10%) of the total generating capacity, and significantly less than the grid's minimum demand. This meant that the Koeberg units could be operated at full power while the National Grid Controller uses the coal fired power stations to balance the generation with demand and maintain grid frequency.

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[1]The **Kyoto Protocol** is an international treaty, which extends the 1992 United Nations Framework Convention on Climate Change (UNFCCC) that commits State Parties to reduce greenhouse gases emissions, based on the premise that (a) global warming exists and (b) man-made CO_2 emissions have caused it.
[2]Each unit uses ~40 MW_e for its own power supplies i.e. ~ 930 MW_e is sent out to the grid.
[3]Base load units constantly operating at full capacity.

With SA planning to build more Nuclear Power Plants while increasing the renewables in the energy mix, the question of load following arises. It becomes necessary to interrogate the capability of the NPPs to respond to load changes whether daily, seasonally or to respond to unforeseen grid disturbances. The load changes could be due to a loss of a power generating unit or units, which could change the grid by anything from 100MW – 1000MW. Certain European countries, like France and Belgium, use the "10% Rule"[4] to reduce the output of large reactors at times of low demand (OECD, 2011).

1.4 Load variation challenge to South African NPPs

At the moment, when a big loss (>5%) in generation capacity occurs in the South African grid, the national electric grid controller must correct the nominal 50 Hz grid frequency before it decreases to less than 47.5 Hz, otherwise both Koeberg reactors will house load[5]. This is undesirable because each of the two Koeberg units generates about 970 MW. This means that the turbine has to automatically runback from 970 MW to 40 MW at a rate of about 500 MW/minute. This requires the control rods to insert into the core at maximum speed (72 steps per minute) from ~220 rod steps in order to suppress the nuclear reaction and therefore reduce the heat input into the Reactor Coolant System (RCS). The rapid insertion of control rods occurs simultaneously with the turbine bypass system operating to remove heat from the RCS, therefore cooling the core. This is undesirable because it reduces the shutdown margin and puts strain on major equipment like control rods drive mechanism (CRDM).

There are also morning and evening electricity demand peaks that could occur over a few hours and the generating units need to respond to that demand. One of the main challenges to NPPs manoeuvrability is the xenon production and removal from the core. An increase or decrease in reactor power has a direct effect on xenon production and removal, and xenon oscillation can take several hours before reaching equilibrium. Xenon oscillation therefore directly challenges reactor safety by changing the reactivity and the power density of the core. The reactivity change can be positive or negative, depending on the direction of the power change and core life.

[4] This philosophy is based on the fact that it is difficult to control the fall in frequency after a reactor trip if it generates about 10% or more of system demand at the time.
[5] House Loading – is when the NPP automatically disconnects its turbine from the national grid and only supplies its own auxiliaries.

The focus of this dissertation is on technical aspects to consider when a NPP is operated in load following mode, paying special attention to effects on the integrity of the RCS boundary and fuel performance i.e. RCS pressure & temperature spikes, reactor coolant levels, axial offset, shutdown margin and xenon effects.

1.5 Key questions to be answered in this dissertation

What is load following and why is it necessary i.e. benefits?
What are the disadvantages of load following using nuclear plants?
What challenges would the current Gen II NPPs encounter if they were to operate in load following mode?

1.6 Objective of this dissertation

To examine whether load following is possible for the existing South African NPPs, and whether it is necessary for the future NPPs to load follow.
Note: The generation III reactor designs (EPR, VVER, AP1000 & APR) include advanced rod control system in their designs, and the manufacturers guarantee that the reactors are capable of step changes of up to 20% of nominal power. This is achievable because their control rod system design has been modified to include an Advanced Rod Cluster Control assembly (ARCC) which is more robust and consists of both black and grey rods. Black rods are made of material with a good absorption cross section for thermal neutrons and do not rapidly saturate e.g. 80% silver, 15% cadmium, 5% indium. Therefore black rods absorb all incident thermal neutrons while grey rods only absorb a portion of thermal neutrons. The ARCC is designed to manage xenon oscillation during load following without the use of soluble chemical neutron absorber (boric acid/chemical shim) in the reactor coolant and therefore maintain an evenly distributed flux.

This report will also:

- ❖ Examine the technical difficulty, complexity of operation and safety of NPPs during significant load variations.
- ❖ Determine the necessity for the SA's NPPs to follow load.
- ❖ Obtain an understanding of nuclear plant load following technique.
- ❖ Evaluate how load following affects nuclear safety.

- ❖ Simulate the behaviour of a NPP when responding to load changes and determine possible effects on the core. (Subject to Koeberg simulator availability).
- ❖ Formulate an opinion on whether SA should modify the current NPPs to follow load and or include the load following function in its future nuclear plant designs.

1.7 Research methodology and approach

The research will comprise of five main activities namely:

- ❖ Literature review on nuclear power plant's load following, including defects and equipment failures associated with load following.
- ❖ Demonstration of Koeberg units' load change capability using the Koeberg plant simulator i.e. simulate a house loading event or an islanding event.
- ❖ Analysis of the data collected during simulation or use test results from other researches if Koeberg simulator is not available.
- ❖ Conclusions and possible further work required.

1.8 Motivation and implication of research

There is a need for academics, engineers, scientists and specialists to get involved in South Africa's nuclear program to advise, support or educate the stakeholders involved in this project including the decision makers. The nuclear deal should be specific to the South African energy mix requirements and unnecessary features in the design should be eliminated to minimise costs while maintaining nuclear safety as one of the high priorities. The findings of this research could be used when choosing the appropriate technology suitable for South African grid conditions and could also be used to highlight areas where capital costs could be reduced.

1.9 Research outputs

- ❖ The research will produce a detailed reasoning of whether load following should be considered or not in SA.
- ❖ The pros and cons of load following with NPPs.
- ❖ The current design's capability to accommodate big load changes.

1.10 Koeberg Simulator model

Model used: Pressurized Water Reactor (PWR) of inverted U-tube steam generators, 3 loop Framatome plant (MST, 2006).

The simulator model is run on Orchird® software suite created by L3-MAPPS. It is capable of simulating a variety of accident and transient conditions that could occur in a nuclear power plant. The simulation ranges from sensor failures and equipment failures to grid disturbances and cold leg double-ended guillotine break. The software operates in the Windows XP environment at a speed faster than real-time. The speed of simulation can be adjusted to 16 times faster than real-time. The status of important parameters and equipment is displayed in a high-resolution colour mimic and colour coded control panels. The software allows simulation of operator actions by means of interactive controls. The Koeberg model consists of the main process areas found in a NPP, including all control & safety systems. Therefore the simulator reflects the behavior of a real NPP.

The software package consists of a set of initial conditions that represent different plant conditions e.g. time of life and power level. Auto / manual modes of plant operation are selectable using the mouse. All the possible incidents and accidents analysed in the Koeberg Safety Analysis Report (KSAR) can be simulated using the software. This is normally performed for training, investigation or design purposes. Incidents and accidents that can be simulated include:

- Full power plant operation
- Start-up, shutdown and power changes
- Turbine trip with or without bypass available
- Reactor trip with all control rods inserted or Anticipated transient without scram (ATWS)
- Steam generator tube rupture (SGTR)
- Loss of coolant accident (LOCA) or steam line break accident
- Dilution or boration transients
- House load or islanding
- Loss of off-site power

- Feed water transients
- Steam line break
- Any combination of the above

Initial conditions can be selected from a variety of set initial conditions corresponding to various power levels and time of life conditions of the plant. This version is also used by the International Atomic Energy Agency (IAEA) as a training platform for its simulation workshops. It is a practical tool used in licensed operator training, instructor training, emergency exercises, investigation & analysis, and probabilistic safety assessment.

CHAPTER 2: Literature Review

2.1 Basic layout of a Pressurized Water Reactor (PWR) plant

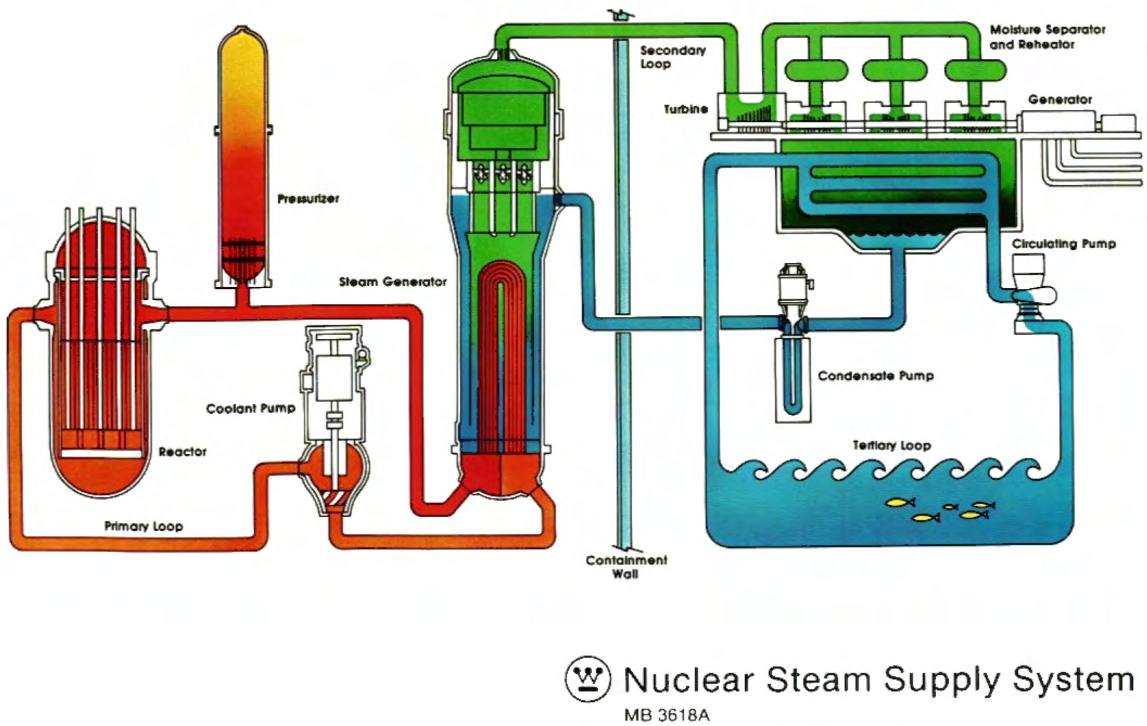

Figure 2: Illustrates the basic layout of a PWR, showing only the major components
. (Eskom, 2014)

2.1.1 Reactor

The reactor vessel volume is designed taking into account the equipment contained: core, control rod clusters, internal core support structures and other parts directly associated with the core. The core is the heart of the nuclear power plant, where the nuclear fission takes place. Heat is generated in the core through the fission process in the fuel rods and the energy produced is removed by the reactor coolant i.e. in a light water reactor, moderation is by elastic scattering of neutrons by protons. The reactor coolant enters the reactor vessel through the vessel inlet nozzles, flows down into the annulus formed by the vessel and the core barrel, rises into the core and the outlet plenum and leaves the vessel through the vessel outlet nozzles.

In PWRs, the coolant also acts as a neutron moderator. The coolant is force-circulated by the reactor coolant pumps and is maintained sub-cooled by the reactor coolant system pressure. The system operates at 15.5 MPa pressure and the average reactor coolant system temperature is maintained at 296 °C i.e. ~20 °C sub-cooling margin (Eskom, 2007). The coolant consists of borated water for excess reactivity control.

2.1.2 Pressurizer (PRZ)

The pressuriser is designed to maintain the RCS pressure in a range compatible with the safety and availability requirements of the reactor. It maintains the RCS pressure at 15.5 MPa in order to maintain sub-cooling[6] in the primary loop. The purpose of the pressurizer is to:

- Act as a surge tank for the part of the main reactor coolant system which is constituted by the vessel, the reactor coolant pump casings, the tubes and plenums of the steam generators and the reactor coolant piping that connects them.
- Control the pressure owing to the presence of saturated liquid in equilibrium with its steam and of control devices such as heaters, spray in the steam phase and steam relief valves.
- Protects the plant against overpressures by providing a sufficient quality of steam phase and steam relief valves.

In steady state, the pressuriser volume includes saturated liquid and saturated steam.

2.1.3 Steam generator

The steam generators transfer the core thermal power towards the plant turbo-alternator. They use the reactor coolant heat to produce the saturated secondary steam which drives both the main turbine and the steam driven feed pumps. In order to supply a steam quality compatible with the efficient functioning of the turbine, the steam generators are equipped with moisture separators and driers to ensure that moisture carry-over will be less than 0.25% by weight on the steam outlet nozzle. They also act as a radiation barrier between the primary and secondary loops. After steam has done work in the turbine it is then condensed by being cooled by the tertiary loop and pumped back to the steam generators. The tertiary loop could either be sea (as in the case of Koeberg), lake or river water and is pumped back into the natural reservoir.

[6] Sub-cooling refers to a liquid existing below its normal boiling point.

2.2 Reasons for baseload operation

Baseload operation is the steady full load operation at all times, with few exceptions like coast down operation, stretch out operation, power reduction due to incidents or component failure and during Operating Technical Specification (OTS) fallbacks. The various reasons for this mode of operation are:

- When NPPs contribute a small percentage of the grid's total generating capacity; generation and demand can be balanced, and frequency controlled without requiring the nuclear units to be part of flexible operation.
- Commercially, nuclear plants are preferred to operate at full power; this is due to the high capital cost with relatively low fuel costs. The priority is to maximise revenue in order to repay the loan taken for the NPP building. This would reduce the interest incurred and result in reaching the "break-even point"[7] relatively early.
- The design, licensing and operation of a nuclear plant operating at constant load are simpler.
- Baseload operation is the least challenging for plant operation and maintenance.
- Base-load operation uses nuclear fuel efficiently and outages can be planned accurately because fuel depletion estimations are accurate.

2.3 Load following justification

One of the major challenges the energy industry faces is that there is no direct way to store electrical energy in bulk. It therefore becomes an important requirement for the electrical systems to be able to continuously adjust the generating capacity to match the demand at different times of the day and during various seasons of the year. In order to compensate for electricity generation losses or gains or even changes in electrical energy demand, it may be necessary to change the output of certain generating units.

When the renewable energy input increases in a grid that consists of a fair portion of nuclear power, the flexible operation of NPPs becomes significant. The significance of flexible operation becomes evident when the electrical energy input from the renewable energy power sources changes, due to changes in weather patterns, the NPPs have to increase or decrease load depending on load balancing requirements. The change in generating output of NPPs could be

[7] B*reak-even point* - defined as a point where total costs (expenses) and total sales (revenue) are equal.

planned or unplanned, automatic or manual and the load adjustment could be required within a short period of time. Therefore a flexible operating NPP must be able to stabilise power at any value, ramp power up/down at a defined ramp rate and participate in grid frequency control.

Electricité De France (EDF) invested more than 15 years researching Flexible Power Operation and France is the most nuclear power dependant country in the world, therefore most operating experience with regards to load following comes from EDF (OECD, 2011). The majority of EDF's nuclear fleet is designed for flexible power operation and mostly operate in load following mode.

2.3.1 Reasons why SA should consider load following using NPPs

The research carried out by a joint IAEA and EDF technical team concluded that the reasons for load following were one, or the combination, of the following factors (ZV, 2014):

1. <u>Large percentage of the nuclear generating capacity;</u>

   To minimise the overall operational costs in a grid that consists of a mixture of generating technologies, the generated output from the units with the high marginal costs are operated in a flexible mode. This is done to maximise the generation output from the units with low marginal costs. This is a possible scenario for South Africa in the event of introduction of 17% (9.6 GW) of nuclear energy and 42% (18.2 GW) renewable energy (DoE, 2010).

2. <u>Rapid growth in renewable generation;</u>

   This means that the total output from the units with low marginal costs increases. The South African government plans to increase the amount of renewable energy input into the energy mix, therefore flexible load operation will become necessary. According to IRP2019, the South African government is focusing on adding more renewable energy sources to supply the national electricity grid.

3. <u>Deregulation of the public electricity supply system;</u>

   In Europe, many states have deregulated their electricity industry therefore making it lucrative for NPPs to follow load in order to benefit financially from the higher tariffs charged for reduced power operation. In a deregulated market, there is a financial incentive to generators when they follow load at a request by the grid controller. South African government have resisted the trend to deregulate the electricity market, but the calls to consider the deregulation of the electricity supply are still persistent.

4. Transmission constraints;

   Under certain conditions the capacity to transmit power from the NPP can be limited, forcing the NPP to operate at reduced load for an extended period of time. If the capacity of nuclear generation exceeds the minimum demand, it poses a challenge in controlling frequency, balancing generation and demand becomes impossible. Then it becomes imperative for NPPs to operate flexibly.

5. Changes to the electricity market during the long operating lifetime of the NPP;

   The lifespan of the modified Generation II NPPs can be extended to operate for up to 60 years and the country's energy mix can change drastically over such a long period. A significant growth in renewables, like wind and solar, would also necessitate the traditionally base load stations to operate flexibly. Germany is a typical example of what effects a change in energy mix can have. The German laws were changed to prioritise energy generation from renewables. Since the renewables have limited controllability compared to the base load stations and also have varying output, the NPPs had to be able to operate flexibly (OECD, 2011). South Africa could face similar challenges.

2.4 Limitations on current Koeberg design

All fatigue related transients are identified and accounted for and confirmed to be within design analysis criteria for the life cycle of the Nuclear Steam Supply System (NSSS) and auxiliary systems. All fluid system pressure, temperature and flow transients that have been considered when designing the RCS components have a maximum allowable number of cycles/occurrences for the life of the station. Each transient is classified under a pressure vessel code as Category I, II, III and IV. Any normal operating change in the plant's operating status (e.g. RCS heat-up or cool down, power raising, turbine trip etc.) may become an "Accountable Transient" after transposition to the design transient list (Table: 1) even though they may not have exceeded a threshold or set point.

Load following operating mode introduces transients that have not been accounted for in the original design of the core and the RCS. During low power operation, following a rapid load reduction, the lower RCS average temperature (T_{avg}) affects the pressuriser spray nozzle integrity and the pressuriser surge line temperature. This is caused by the decrease in cold leg temperatures introducing thermal shock to the system.

Therefore in order to ensure plant integrity and plant operation within the boot curve[8], the surge line temperature limits must be respected. Flexible power operation might challenge these temperature limits and the integrity of the primary system might not be guaranteed during normal plant operations. The National Nuclear Regulator (NNR) will require assurance that the safety limits will not be threatened during normal plant operation since it has the mandate to ensure compliance with 10 CFR 50. According to the NNR the reactor coolant system (RCS) must meet the requirements of criteria 10 and 15 of 10 CFR 50 – Appendix A (NRC, 2014).

Criteria 10 (Reactor design) states that "The reactor core and associated coolant, control and protection systems shall be designed with appropriate margin to ensure that specified acceptable fuel designed limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences". Criteria 15 (Reactor coolant system design) states that "The reactor coolant system and associated auxiliary control and protection system shall be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during any normal operation condition including anticipated operational occurrences".

The RCS design of the Koeberg units is based on maintaining the surge line temperature between 50 °C & 110 °C and the pressuriser spray nozzle temperature less than 177 °C (Eskom, 2007). Load following is not one of the transients that were considered and analysed in Koeberg Safety Analysis Report (KSAR). This can be seen in Framatome's design transient list which is the basis for "Accountable Transient" programme at Koeberg (KAA-652, n.d.).

Although the control rods movement generate fast core reactivity change, they also cause power distribution changes that lead to changes in flux distribution, and they also add

[8] Boot curve- Design operating envelop for normal plant operation

mechanical load on components like RCCS. An alternative method to control reactivity during load changes is to change the boron concentration in the RCS. Varying boron concentration has little effect on flux distribution but its effects are time delayed (5-10 minutes) and its effectiveness depends heavily on core life. At the beginning of life (BOL) the boron concentration in the primary system is high (~2000 ppm) and low at end of life (~10 ppm). Once the primary system is borated at EOL, it is almost impossible to dilute the system for RCS temperature change. With RCS boron concentration < 40 ppm, a temperature change of 0.1 ˚C requires an addition of ~6000L of demineralised water. For example, if 500L of borated water is added into the RCS at EOL, ~ 15m^3 of demineralised water would be required in order to dilute the cooling water system and therefore stabilise reactor power.
NB: Thumb-Rule for boron/dilution relationship is that for every 1ppm boron change, 30L of water is required i.e. in order to increase reactor power by reducing the boron concentration by 1 mg/kg, 30L of demineralised water would need to be added into the RCS.

Therefore, changing boron concentration to compensate for load following will not be feasible because of the time delayed effects and the volume of water required to dilute the reactor coolant system at EOL. Any unplanned reactivity change increases the probability of a nuclear incident due to possible sudden power change and therefore unplanned reactivity changes are undesirable.

There are also morning and evening electricity demands that could occur over a few hours and requires intervention by the National grid controller. The Koeberg units respond to any grid frequency increase of more than 4% (2 Hz) by decreasing turbine load. Under no circumstances will the units respond by increasing turbine load because they are base-load stations. They operate at full power, so any further load increase will challenge the thermal power safety limits which would be a violation of the Koeberg licence. A further turbine load increase would result in thermal limits being exceeded and the reactor operating outside the analysed zone for design base accidents, which would be a violation of the KSAR. Such event would have dire consequences for Koeberg operation since thermal limits are not to be exceeded under any circumstances. The mandatory requirement, to maintain reactor power below thermal limits, is in accordance with the Koeberg License Basis Manual. The purpose of the mandatory requirement is to ensure compliance with standards and processes of the license conditions and regulations (NNR, 2008).

List of design transients that will require close monitoring and continuous evaluation should the Koeberg units be required to load follow:

Table 1: Design Transient List for Koeberg Unit 1&2 (KAA-652, n.d.)

Description of Design Transient	**Maximum Occurrences Allowed during Reactor lifetime**
Heat up of RCS	180
Depressurising and cooling down of the RCS	200
Load increase at 5% per minute between 15% and 100%	2400
Load increase at 5% per minute during stretch-out	80
Sudden load increase from 90% to 100% full power	2000
Sudden load decrease from 100% to 90%	2000
House load – Large load loss from 100% (including turbine trip without scram) -- During stretch-out	120 ---------------------------- 40
Steady state fluctuations at power -- During stretch-out	1 700 000 ---------------------------- 300 000
Increasing the load between 0% and 15% full power	2200
Decreasing the load between 15% and 0% power	2200
Turbine trip from 100% load with partial condenser availability resulting in reactor scram	80
Loss of external electrical 400kV supplies	8
Reactor trip from full power without inadvertent cool down (including spurious control rod drop) --- During stretch-out	170 ---------------------------- 80
Reactor trip from full power with cool down and no safety injection	60
Reactor trip from full power with cool down and safety injection	10
Depressurisation of RCS resulting in reactor trip and safety injection	20
Inadvertent safety injection actuation	16
Large amplitude transient ($\Delta T = 50$ °C)	200
Opening of relief valve	150

Opening of safety valve	20
Ejection of control rod cluster	1

Assuming the worst case scenario of daily power changes of more than 10%, it can be seen from the table above (sudden load decrease/increase) that the reactor vessel would need to be evaluated within 6 years, instead of the 10 yearly inspections.

2.5 Set points to be monitored

ΔP Set Points:

1. Primary Circuit = 1.0 MPa / 3 hour
2. Steam Generator = 0.5 MPa / 3 hour

ΔT Set Points:

1. Primary Loops

   ≥ 280 °C = 5 °C / 3 hour (T_{avg})

   ≤ 280 °C = 5 °C / 3 hour (T_{hot} or T_{cold})

2. Pressuriser liquid phase = 6 °C / 3 hour
3. RCV charging line = 20 °C / hour

ΔP Set Points:

Reactor load variation ≥ 20 %

Table 1 above, indicates that the design of the current Koeberg units did not consider frequent load changes. Plant life would be significantly reduced by transients like:

Heat up of RCS, depressurizing and cooling down of the RCS, Load increase at 5% per minute between 15% and 100%, sudden load increase from 90% to 100% full power, sudden load decrease from 100% to 90% and House load or large load loss from 100%. All these transients are associated with load following and they would occur daily, with the exception of the house load event. Such daily occurrences would challenge the reactor protection design system and would require modification of the reactor protection logics. For example, a sudden change in RCS temperature changes the density of the coolant and therefore affects fast neutron thermalisation. A change in neutron moderation would directly affect neutron flux and reactor protection trip signal on flux runaway might be generated. Note: A 5% flux change over 2 seconds whether positive or negative, triggers a reactor trip.

2.6 Load following modifications

2.6.1 Some of the systems that would be affected

Many plant systems will be directly affected by load following and will have to be modified. The following systems will be greatly affected:

REA – Reactor Boron and Water Make-up System

RGL – Rod Control System

GRE – Turbine Governing System

GSE – Turbine Protection System

RPR – Reactor Control and Protection System

TEP – Boron Recovery and Recycle System

TEU – Liquid Waste Treatment System

TES – Solid Waste Treatment System

TEG – Gaseous Waste System

REN – Nuclear Sampling System

ARE – Feed water Flow Control System

DVN – Nuclear Island Ventilation System

VVP – Main Steam System

RCV – Chemical and Volume Control System

RCP – Reactor Coolant System

GPV – Turbine Valves and Drain System

GCT – Turbine Bypass-Steam Dump System

RPN – Nuclear Flux Instrumentation System

Impact on each of the listed systems will have to be evaluated.

2.6.2 Procedure changes and periodic testing

Most of the current Koeberg operating procedures will have to be reviewed and changed to accommodate load following as normal operation. Some of the procedures that would be greatly affected are:

- KWB-I-RGL (Rod Control System Malfunction),
- KWB-I-5 (Loss of Main Off-Site Power),
- KWB-G2 (Power Increase from Hot Standby and Reactor Control During Power Operation),
- KWB-GS3 (Run-up, Synchronization and Power raising of the Turbine / Generator) and
- KWB-GS4 (Programmed Shutdown of the Turbine – Generator Set).

The periodical tests and the frequency of safety related and essential systems would also need to be reviewed.

All the plant control circuits would need to be modified. Pressurizer level control, RCP pressure control logics, RGL control circuit, T_{avg} computer program (both reference and average temperatures), SG level program, reactor trip logics, controls, permissive and set points would also need to be reviewed. This would require NNR approval (NNR, 2008).

2.6.3 Training of personnel

According to NNR requirements for operator license holders at Koeberg, all the current Licensed and Non-licensed operators will have to be retrained and relicensed (NNR, 2008). Maintenance philosophy will also require to be changed and the maintenance regime updated. Training personnel will have to learn about NPPs load following operation, incidents and accidents associated with flexible operation. This would include creating new incident and accident scenarios to train and test operator competencies in dealing with nuclear accidents. New operating experience, related to plant degradation as a result of load following operations, would have to be sourced from international bodies like INPO, WANO and IAEA. The System Approach to

Training (SAT) would have to be used to analyze training requirements, design and develop a training program, implement and evaluate the effectiveness of such an intervention. This could necessitate sending training instructors abroad to acquire the necessary knowledge and skills, it could also be necessary to hire training instructors from NPPs that operate in load following mode. The local tertiary institutions would also need to run academic programs to assist the nuclear industry in this regard.

The Koeberg simulator is a full-scope control room simulator specific to the Koeberg units and will require improved models and computers in order to adequately simulate flexible plant operation. It will then be used to familiarize operators with the new operating procedures, develop operators' diagnostic skills, train operators on how to operate the plant equipment safely and conservatively during load changes, incidents and accidents, and will also be used to troubleshoot the real plant problems. NNR has a requirement for training, examination and requalification of Licensed Operators to be performed on a plant specific simulator in order to eliminate the potential for negative training. NNR is guided by 10 CFR part 55 (Operator's Licenses) and NUREG 1021 (Operator Licensing Examination Standards for Power Reactors), therefore Koeberg License conditions will have to be reviewed (NRC, 2014).

2.6.4 Fuel Design modifications

In 1999, KEPCO & Westinghouse collaborated on a project to design fuel for load following NPPs (KEPCO, 2012). This design has been approved in Europe for use in most of the Gen III NPPs. The fuel designers boast about an improved structural integrity of fuel and the fact that it meets the IAEA's four criteria for fuel integrity. The criteria give assurance that:

- Fuel damage is guaranteed not be severe enough to prevent the rods from inserting into the core when required.
- Fuel will stay intact during normal plant operation and during anticipated operational occurrences e.g. house load, islanding or turbine trip events.
- The number of fuel rod failures determined conservatively i.e. it is not underestimated for postulated accidents.
- The ability to cool fuel will always be maintained.

This is a very important criterion to ensure compliance with ECCS criteria i.e. Emergency Core Cooling System response to a large break Loss of Coolant Accident (LOCA). The nuclear plant vendors will have to demonstrate to the nuclear regulators that their designs meet the ECCS acceptance criteria including the effects that a higher burnup will have on cladding performance. The new improved fuel design was compared with the conventional fuel and confirmed to have an improved fuel performance and advanced mechanical integrity. According to KEPCO the advanced design features of the PLUS7 fuel are:

- High mechanical strength i.e. mid grid buckling strength improved by 45%.
- High thermal performance – overpower margin increased by more than 10%.
- High burnup performance – batch average burnup increased from 45000 to 55000 MWD/MTU.
- Improved fretting wear resistance by increased grid to rod fretting wear resistance with conformal grid springs and dimples.
- Enhanced fuel production by standardising the design and manufacturing processes.
- Improved economy by enhancing uranium utilization and neutron economy.

These improvements may increase vendor workload significantly and the licensing cost would also increase.

2.7 Load following capability of GEN III NPPs (PWR)

The 3rd Generation (GEN III) PWRs are designed with the capability to follow load repetitively in the range of 15-100% of the rated power. Although this capability is said to be available throughout the core life, there are certain limitations to the load following operation towards the end of life (EOL). According to the GEN III plant designers, the PWR can make step load changes of 20% of rated power and ramp changes of 5% of rated power per minute. A daily load cycle of 12-2-8-2 can be followed easily using these NPPs. This consists of a full power operation for 12 hours, load reduction to 50% power over a 2 hour period, operate at 50% for 8 hours and then increase load back to full load over 2 hours (Bruynooghe C, 2010).

They are designed with a 50% load rejection capability, from full power, without a reactor trip. Load rejection could be as a result of a turbine runback signal or loss of main off-site power supply. Turbine runbacks can be triggered by a variety of incidents e.g. a loss of steam feed pump, high stator conductivity or failed islanding. A full load rejection is also within the GEN III PWR's design capabilities e.g. a turbine trip from 100% power without a reactor trip. The Gen II NPPs, like Koeberg, are also designed with the capability to accept a full load rejection from full power without a scram. This design capability is made possible by the ability of the secondary system to divert up to 85% of steam produced by the NSSS away from the turbines.

2.8 Challenges and limitations to load following using Gen II reactors

Changes to licensing basis, plant design and Safety Analysis Report are the overall challenges specific to changing a baseload NPP to a flexible operating plant. Technical challenges and limitations specific to load following are: Reactivity management, load following effects on control rods, integrity of the RCS boundary, increased radioactive effluent and change in generation and outage planning.

2.8.1 Reactivity management effects

- Nuclear poison[9]: neutron flux distortion
- Use of soluble boron
- Xenon transients, Uneven flux distribution

2.8.2 Effects on control rods

a) There will be a need for additional equipment and outage changes due to:
- Increased wear on control rods and drive mechanism
- Decreasing control rod worth due to fuel burnup

b) There will be a change in preventative maintenance of:
- Control rod drive mechanism
- Control rods
- RCC guide tubes
- Mechanical structures of the coolant pressure boundary

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[9] Any fission product with large neutron absorption capacity in the core, either than fissionable material.

c) Possible control Rod Failures:

- Accidental withdrawal
- Control mechanism failure (“uncontrolled withdrawal”)
- Mechanical failure (ejection)
- Misalignment
- System Malfunction
- Operator Error

d) Modifications required:

- Use of standard/black control rods (full neutron absorption strength)
- Use of grey control rods (part neutron absorption strength)
- Hot channel factors monitoring

2.8.3 Physical integrity of components exposed to pressure and temperature transients

- Pellet-Clad-Interaction (figure 4.2 in the next chapter): The PCI phenomenon is the main cause for clad failures in PWRs. The thermal expansion of the pellets, caused by gaseous and volatile fission products inside the pellets that cause swelling, reduces the gas volume inside the pellet and compresses the helium gas.
- Most pump motors operate at constant speed with control valve controlling the flow. Control valves could operate outside their optimum opening position and therefore accelerate wear and tear of the valve internals. This is prevalent in steam systems and could increase corrosion and erosion of valves and pipe bends.
- The secondary system (feed train) could experience water hammers.

The National Energy Technology Laboratory of the United State of America found out that:

- Pipe thermal stress and fatigue cracking were some of the most significant problems with load following.
- Condenser tube grooving at the support plates can occur due to poor water chemistry.
- Feed water heater tube grooving at the support plates can occur due to poor water chemistry.
- Most fan motors are constant speed, requiring more oscillation of the dampers during load following. This causes wear and tear on dampers and motors (NETL, 2012).

2.8.4 Operator's responsibility with regards to power distribution

- Verify correct rod bank sequence and overlap
- Maintain flux deviation within the acceptable band
- Verify QPTR less than the OTS limit
- Maintain rods above Rod Insertion limit
- Keep each rod cluster within 24 steps of control bank demand

2.8.5 Increased radioactive effluent

For load decrease while maintaining uniform flux around the core, boric acid solution must be added. The reactor coolant system is a constant volume system therefore for every cubic meter of boric acid solution added into the system, the same volume of reactor coolant must be removed and sent to the waste treatment system. This would be the case for load increase also; the amount of demineralized water added into the primary system will result in the same volume of borated water being diverted to the waste treatment system. This could result in chemistry upsets and increase in general plant radiation levels. Therefore large amounts of borated and processed water can be used wastefully. With South Africa being a water scarce country this would necessitate a desalination plant to be added to the design. The current water processing unit will have to be modified to cope with the erratic water usage.

CHAPTER 3: Factors Affecting Electrical Grid

A case for including load following ability into design for SA nuclear fleet

In order to accommodate the proposed energy mix, in accordance with the IRP2010, there are several considerations that need to be explored: Load following using NPPs, deregulation of the South African electricity markets, transmission constraints and possible changes to the electricity markets. This dissertation focuses on NPP load following.

3.1 Load Following

The South African government planned to increase the amount of clean energy input into the energy mix, by increasing the current nuclear energy input to 17% and renewable energy input to 40% (DoE, 2011). This was according to IRP2010. The latest plan focuses on increasing energy input from renewable sources, this is according to IRP2019. A rapid growth in renewable generation together with a large percentage of the nuclear generating capacity in the energy mix, requires NPPs to operate in load following mode. This change in operating philosophy is driven by economic factors and efficient power generation. Load following operation of the traditionally baseload power generating units enables generation output from the units with low marginal costs to be maximized. Such operation results in low cost electric power production and maximum use of green technologies, since coal fired power plants are also part of the baseload power generating units. This would be favourable to consumers and since SA is one of the countries that adopted the United Nations Framework on Climate Change it has a mandate to stabilise, limit and reduce atmospheric concentrations of greenhouse gases. Therefore the use of renewable energy, whenever it is available, will play a role in keeping the global warming below 2°C as agreed upon in 2015 Paris Climate Conference (COP21, 2015).

3.2 Deregulation of the South African electricity markets

Arguments in favour for deregulation of the local electricity markets have been ongoing for several years in S.A, with some experts arguing that S.A. is lagging behind other 3rd world countries which already deregulated their electricity markets (Convention, 2014). The call for rigorous transformation of the electricity market was repeated at the 64th AMEU[10] Convention. At the moment South Africa has not followed the global trend for deregulation of electricity markets, although even many African countries have. The South African government's philosophy leans towards central planning and state control. Should the government's position change in favour of deregulation of electricity market, it will be more lucrative for NPPs to follow load since they will benefit from higher tariffs charged for reduced power operation. This is a topic that needs an independent research of its own.

3.3 Transmission constraints

Under certain conditions, the capacity to transmit power from the NPP can be limited therefore forcing the NPP to operate at reduced load for an extended period of time. Such operation results in Xenon oscillations which then cause uneven flux distribution. Therefore, designing NPPs with load following capabilities is an advantage. Changes to the structure of electricity supply system can also lead to transmission constraints. As the capacity of nuclear generation increases in the energy mix it poses a challenge with regards to grid frequency control. A sudden loss of a big load (>5% of the total load on the grid) in the grid would require the baseload units to adjust load by reducing output power in order to maintain frequency at 50 Hz. Since Thyspunt is the approved site for the new NPPs, this means that the Eastern and Western Cape Provinces would rely on nuclear power for baseload energy supply. The load adjustments would have to be performed by NPPs. This might be the scenario that South Africa will be faced with after adding 9.6 GW of nuclear power into the grid.

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[10] Association of Municipal Electricity Undertakings is an association of municipal electricity distributors as well as national, parastatal, commercial, academic and other organizations that have direct interest in the electricity supply industry in Southern Africa.

SA also has an operating experience when it comes to transmission constraints. Veld fires under the 400 kV lines once rendered one of the transmissions lines outside Koeberg Power Station inoperable. This event occurred on 16th November 2005 and required the nuclear reactors to be operated at reduced power for a prolonged period of time (>72 hours), one of the two units had to be placed in Hot Shutdown state (HSD) in order to ensure reactor safety. The Western Cape suffered several blackouts as a result of this incident. If the Koeberg units were equipped with load following capability, they could have both reduced load to 50% and the shutdown would have been avoided. Reactor start-up from HSD to full power can take up to a week, therefore an unnecessary shutdown could have been avoided.

3.4 Changes to the electricity market during the long operating lifetime of the NPP

The life span of the modified Generation II plants can be extended to operate for up to 60 years and the country's energy mix can change drastically over such a long period. A significant growth in renewables, like wind and solar, might also necessitate the traditionally base load stations to operate flexibly. Germany is a typical example of what effects a change in energy mix can have (OECD, 2011). The German laws were changed to prioritize energy generation from renewables. Since the renewables have limited controllability compared to the base load stations and also have varying output, the NPPs had to be able to operate in load following mode.

CHAPTER 4: SIMULATION

This chapter will interrogate the response of the Koeberg units to one of the biggest load variations possible, the house load[11] event. Big load variations will be more likely to occur frequently when the current energy mix changes to that planned in IRP2010-30. A house load event is also one of the design events that poses danger to reactor fuel should there be no Operator intervention, as illustrated in this research.

IRP2010 proposed an increase of the nuclear share in the energy mix to 23% and also to introduce a significant amount (43%) of renewable energy into the energy mix (DoE, 2010). Even with the subsequent elimination of nuclear power as a baseload in IRP2019, the threats caused by electrical grid disturbances would still exist. An increase in modular power sources, which are intermittent by nature, would result in more frequency and voltage variations and that increases the likelihood of the Koeberg units automatically disconnecting from the national grid i.e. house load.

Note: IRP 2010 was used for reference purposes since it specified the amounts of energy addition from various technologies. IRP 2019 is not that specific about the amounts of planned nuclear capacity addition. It is bias towards adding extra capacity in increments and categorizes nuclear power as inflexible capacity. This is true for Koeberg reactors due to the old design that did not foresee the need for flexible operation of a nuclear plant in South Africa. This further highlights the need for load following capabilities to be considered for future nuclear plants in South Africa in order to improve the competitiveness of nuclear technology when compared with other technologies. Note: The validity of this study is not based on government's timeframes for building nuclear

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[11] The action whereby the Nuclear Power Plant automatically disconnects its turbine from the national electrical grid and only supplies its own auxiliaries. NB: House load is triggered by voltage and or frequency deviations.

plants, the argument is valid irrespective of the implementation time i.e. the load following capability argument would still be applicable to the 2500MW nuclear build programme mentioned in Decision 8 of IRP2019.

The Koeberg simulator was used to perform all the scenarios and to collect the data used for illustration. The data was collected and compared between Beginning of Core Life (BOL) and End of Core Life (EOL) scenarios. The EOL scenarios were used in order to illustrate the worst case scenario that the load variations would have on nuclear fuel.

Notes:

- The scenarios were limited to 1 hour runs over four weeks due to the limited availability of the simulator.
- It is also important to note that for the following simulations there was no operator intervention, so there was no credit taken for the highly trained Reactor Operators who would have been in the control room. Should Operator actions be taken into consideration, this event would be an acceptable incident i.e. it would require Koeberg Power Station to report it to NNR and would be subject to the limitation (120 house load events for the lifetime of the power station) stipulated in Table 1.

4.1 Justification for EOL demonstration as the worst case scenario for load following

Fuel pellets for the PWRs (Gen II & III) are made of sintered UO_2. During manufacture only 95% of the theoretical density of UO_2 is achieved (KEPCO, 2012). Under the high operating temperature of the reactor, their density may increase. This would cause the whole stack of pellets to shorten. A stuck pellet would cause a gap. In the region of the gap, the moderator/fuel ratio increases, and since the reactor is under moderated, the reactivity and thermal flux will peak at that location. The adjacent pellets produce more power as a result and could lead to a local over power situation, especially at EOL. Load following exacerbates the localized over power effect, especially on Gen II reactors.

Load changes in a nuclear plant are more challenging at EOL mostly because of two reasons:

1. Copious amounts of water are needed to dilute the RCS at low boron concentration. There are restrictions on dilution flow rate, the demineralised water reservoirs are finite and the rate of waste water treatment is low.

2. At EOL, xenon oscillations are more likely to start and very difficult to control. These oscillations lead to core over power and uneven fuel burnup.

Therefore the cost of water processing and the difficulty to control axial flux distribution are the main challenges. The axial flux distribution is a reactivity management issue therefore a nuclear safety concern.

One of the main core safety limits is the maximum fuel temperature limit of 2590 °C. The basis for this limit is to prevent fuel centreline melt which occurs at 2804 °C. The fuel centreline melt is based on new UO_2 fuel pellets. The melt point temperature decreases with exposure to radiation, approximately 32°C for every 10,000 MWD/MTU (RFE, 2015). The melt point temperature limit is therefore conservative for fuel assemblies up to 67,000 MWD/MTU burnup, which corresponds to EOL, since irradiated fuel melts at lower temperature. At EOL the fuel centreline temperature would be significantly higher than at BOL because of the crud layer that coats the fuel cladding (Figure 4). This layer of crud affects heat transfer from the fuel pellets to the reactor coolant. Since PWRs are designed to operate at constant average reactor coolant temperature, the fuel temperature would increase in order to maintain a constant RCS temperature. The pellet clad temperature (approximately 350 °C) depends on the heat transfer coefficients in fuel, helium gap, cladding, convection film and LPD[12].

NB: The fuel cladding temperature is limited to 1200 °C to prevent Zirconium-water reaction.

All these parameters are affected by neutron bombardment during core life. The fuel clad temperature is limited by limiting the linear power density. In order to illustrate the operating point of the Koeberg units on Figure 5, the average LPD for the entire core at full power was determined.

Core data is:

Each fuel rod is 4m (active fuel height = 3.65m)

157 fuel assemblies in the core

264 fuel rods per assembly (*FA*) of which 42 are guide thimbles (*GT*)

Core thermal power is 2775 MW

<u>Solution:</u>

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[12] Hot spot of the fuel centreline temperature at full power is approximately 1830 °C.

$$\text{Average linear power density} = \frac{total\ thermal\ power}{total\ fuel\ rod\ length}$$

$$= \frac{Q\ [kW]}{(Total\ Number\ of\ Fuel\ Assemblies\ in\ Core)\ (FA{-}GT){\times}L\ [m]}$$

$$= \frac{2.775 \times 10^{6}\mathrm{kW}}{3.65\ (222)\ (157)\ m}$$

$$= 21.8\ \mathrm{kW/m} \equiv 218\ \mathrm{W/cm}$$

Fuel centreline temperature at full power is ~ 1000 °C (RFE, 2015). Figure 3 indicates that the Koeberg reactors operate at a higher than average linear power density. This demonstrates that as the fuel centreline temperature increases toward EOL the safety margin decreases. Therefore the hot spot of the fuel centreline temperature could approach the fuel centreline temperature limit especially if the core experiences axial flux variation due to xenon oscillation.

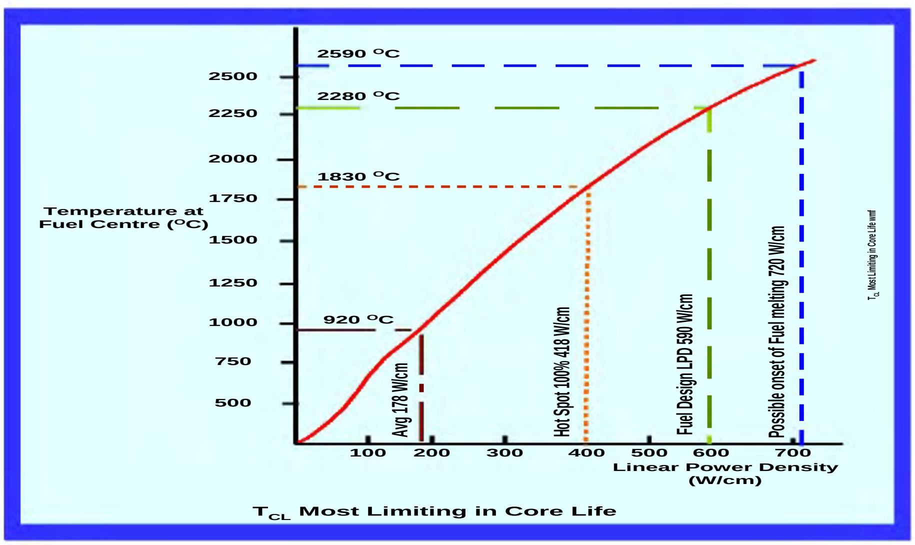


Figure 3: Fuel Centreline Temperature vs Linear Power Density (NRC, 2007)

At EOL, the risk of reaching the maximum LPD is high as a result of fuel changes during reactor operation. The helium gap between clad and fuel is reduced over core life. This is caused by the swelling of the fuel pellet, narrowing the fuel cladding gap. This phenomenon is known as the Pellet Clad Interaction (PCI) and is undesirable because it could lead to fuel damage. Load following then creates high temperature (> 295 °C) and high pressure (> 15.4 MPa) conditions inside the primary circuit, which causes the clad to be pushed onto the fuel pellets resulting in *clad creep*[13].

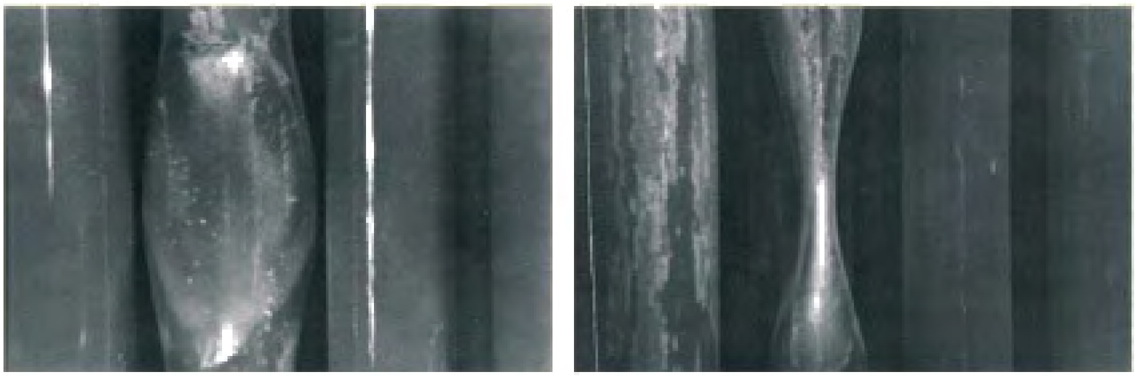


Figure 4: Picture showing fuel cladding collapse resulting in fuel perforation (David G. Franklin, 1988)

[13] Clad Creep is defined as the slow irreversible deformation of fuel cladding influenced by a variety of stresses, thermal, mechanical or irradiation.

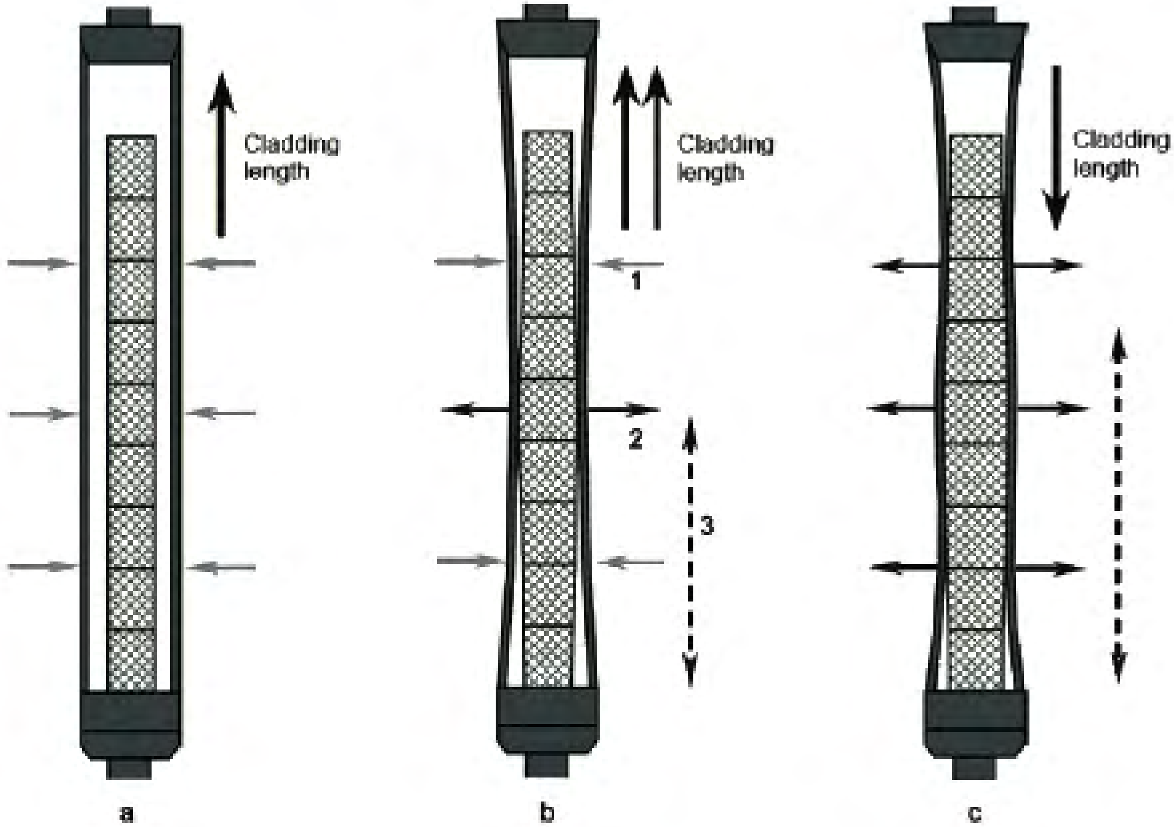


Figure 5: above: Picture showing fuel cladding collapse resulting in fuel perforation: a)

Before fuel-cladding interaction, b) fuel-cladding interaction, c) fuel-cladding interaction took place at most part of the fuel column, 1) coolant effect, 2) diametric effect of fuel column, 3) axial effect of the fuel column part (David G. Franklin, 1988).

Although fuel cladding damage can occur during normal plant operation, it is highly probable during major load variations like house loading or load following, especially at end of core life. This is mainly due to the elevated fuel temperature, more negative moderator temperature coefficient, and the reduced soluble boron content in the RCS. Any system temperature and pressure changes outside normal operating parameters tend to be exaggerated.

4.2 Simulator Set-up

Table 4.1: Initial Conditions (Plant operating at full power at EOL)

Main Parameters	
Reactor Power	2747 MW_{th}
Generator Power/output	962 MW_e
Boron Concentration	12 ppm
Rods Position	220 steps
Average RCS Temperature	296 °C
RCS Pressure	15.3 MPa
Pressuriser Level	-2.1 m
Core Burn-up	EOL
Xenon	-2827 pcm
Samarium	-974 pcm
Flux Deviation (ΔI)	+0.07

4.3 House load event simulation

A house load is an automatic plant response to a loss of 400 kV supply, triggered by the opening of the 400 kV breaker supply to each of the Koeberg units. There is a list of signals that could initiate the opening of the 400 kV breaker with the unit at power e.g. manual opening of the breaker, high voltage yard voltage fault, generator under or over-frequency etc. For this simulation a manual breaker opening was initiated at time [t] = 100s. The 100s time delay was to allow the plant parameters to stabilize. When the 400 kV breaker opens, turbine rejects load from 970 MW to 40 MW instantly. The turbine bypass system then activates and creates an alternative steam release path to remove the steam generated.

Note: The house load event was simulated until all parameters stabilized (~2000s) and then the turbine load was increased to full power at 50 MW/min.

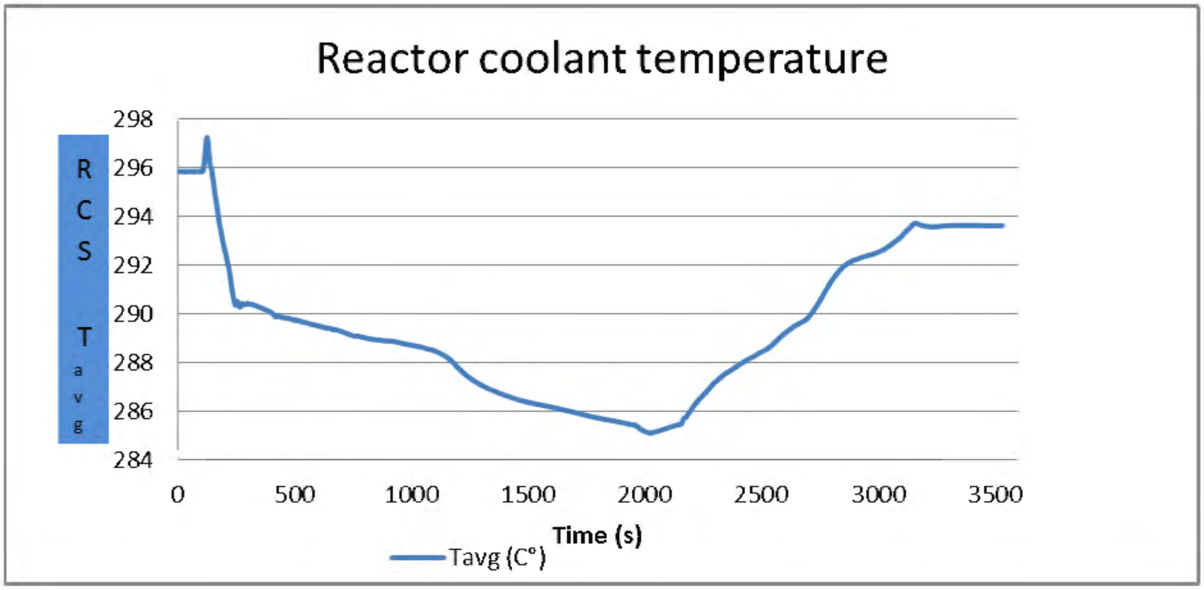


Figure 6: Reactor coolant temperature transient

The house load event reduced the amount of steam extracted from the steam generators and results in less heat removal from the reactor coolant. As the average reactor coolant temperature (T_{avg}) increased, the control rods automatically inserted into the core to stop T_{avg} by reducing reactor power. The insertion of control rods, as illustrated in Figure 7, was triggered by the RCS temperature deviation of > 0.56 °C between T_{avg} and the reference temperature of the reactor coolant (T_{ref}). When the house loading signal was inserted, a T_{avg} spike was observed (Figure 6) and it was terminated by the control rods auto-insertion into the core. The motion of the control rods together with the actuation of the turbine bypass system (Figure 8) resulted in a continuous decrease of T_{avg} to a new reference temperature below 286 °C.

NB: T_{avg} should have stabilized at a no-load value of 286 °C.

The temperature control systems halted the temperature drop at 286 °C but Xenon build-up was still adding negative reactivity into the core. T_{avg} continued to decrease to ~285 °C. The effect of neutron poisons (Xe-135 & Sm-149) is analyzed later in Figure 11.

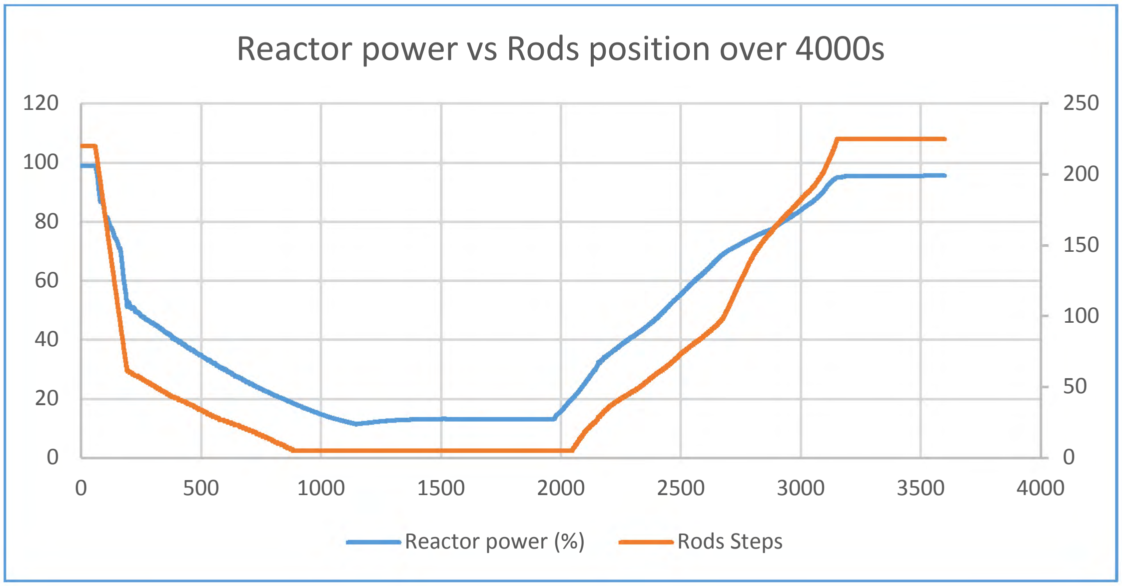


Figure 7: Control Rods movement (in simulation described in text)

As soon as T_{avg} / T_{ref} deviation exceeded 2.8 °C, the control rods were observed inserting into the core at maximum insertion speed, 72 rod steps/minute, from their fully withdrawn position of 220 steps. This resulted in control rods inserting past the control rods insertion limits and such operation evokes a Limiting Condition of Operation (LCO). Control rods should not be allowed to insert below the insertion limits because of three reasons:

- The worst case scenario for postulated rod ejection accident, according to the KSAR, was when the control rods were low into the core i.e. below control rods insertion limits.
- According to KSAR, accident analysis assumed that the flux deviation (delta-I) would be < 5% whenever reactor power > 87% (Eskom, 2007). Delta-I is a critical safety parameter which is monitored and control by the Reactor Operators, maintained < 5% at during high reactor power operation. In this event, the flux deviation was > 20% due to control rods inserting too deep into the core (see Figure 7).
- Rods should always be maintained at the rod bite position in order to be effective immediately. This ensures that should a load rejection incident occur, the differential rod worth available at rod bite position is adequate for control systems to respond as designed. Rod bite position also ensures a stable flux.

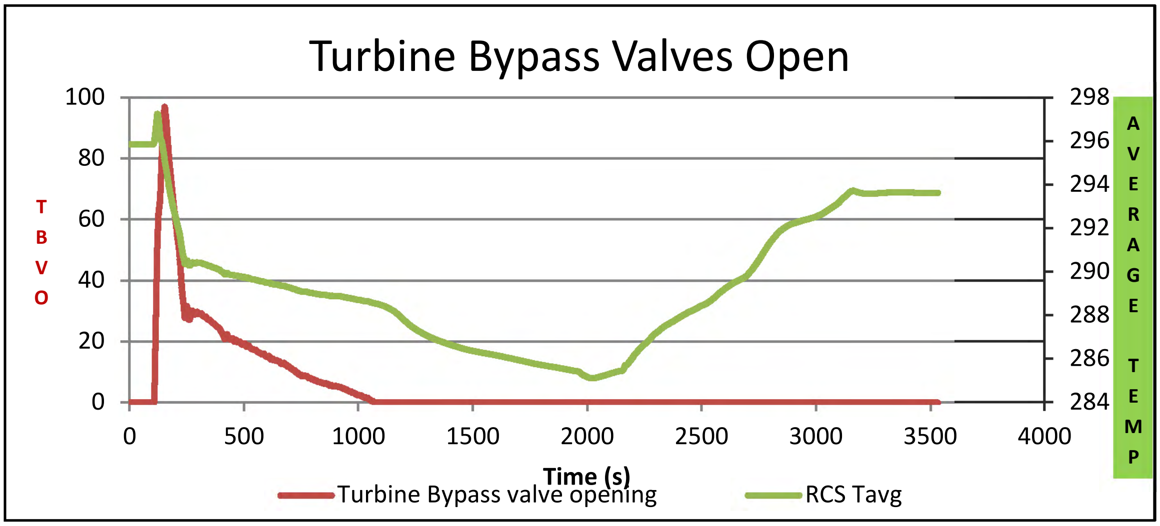


Figure 8: Actuation of the Turbine Bypass-Steam Dump System

The turbine bypass system is designed to remove the energy produced by the reactor if the energy produced by the reactor exceeds the energy removed by the turbine. Turbine bypass valves were observed opening fully in order to remove excess energy from the primary system. The opening of turbine bypass valves is triggered by a temperature deviation of > 2.8 °C between T_{avg} and T_{ref}. In Figure 8 above, the turbine bypass valves were observed fully closing at 288 °C in an attempt to maintain a balance between energy used and energy produced. This occurred in about 1150s into the simulation. Xenon continued to insert negative reactivity into the core and therefore reducing T_{avg} below the no load value of 286 °C. In order to maintain the reactor online (for simulation purposes), turbine load was increased (at time 2000s) to full power level. This action prevented reactor shutdown by xenon induced negative reactivity.

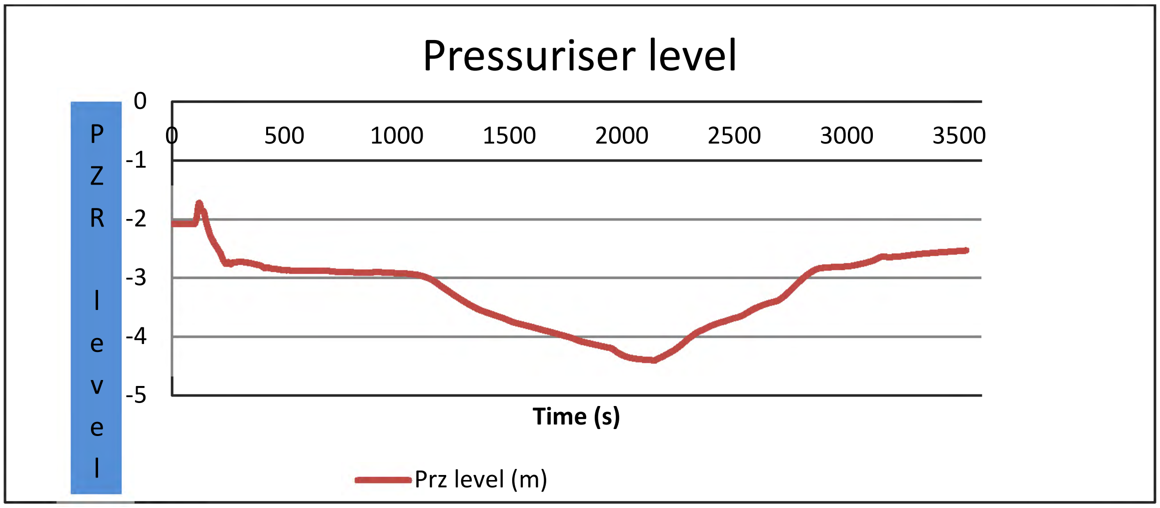


Figure 9: Pressuriser level transient

The pressuriser level transient followed the same trend as the T_{avg} transient observed in Figure 6 i.e. initial increased, followed by a continuous decrease as a result of rods insertion and steam extraction via turbine bypass system. This phenomenon is caused by the expansion of the spaces between the water molecules with increasing temperature, followed by shrinking as the T_{avg} temperature decreases. Note: This is the only indication available for RCS inventory in most Gen II reactors. During normal operations (0-100% reactor power), the level indication must always be > -3.7m. Loss of pressurizer level indication is not acceptable because it is the only indication of reactor coolant inventory. From the illustration above (Figure 9), it is demonstrated that the automatic level control system cannot respond adequately to this event.

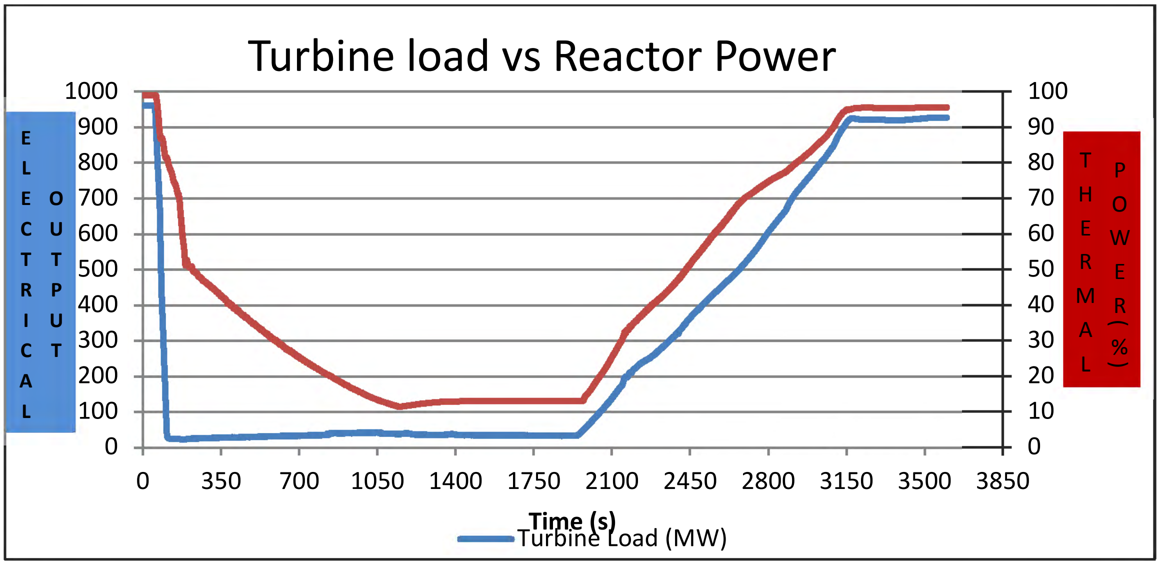


Figure 10: Illustration of fission reaction as indicated by thermal power

The fission reaction rate (indicated by % thermal power) decreased due to the negative moderator temperature co-efficient (Figure 10). As the coolant temperature increased, as illustrated in figure 6, its density decreased. This resulted in reduction of thermal neutrons available to cause fission and that caused a negative change of reactivity. This is an inherent safety feature designed to shut the reactor down when cooling is lost or reduced. The rate of neutron flux reduction is also dependent on the differential rod worth, which is why the minimum rod bite is important (refer to initial rods position in Table 4.1). The minimum rod bite provides for a minimum reactivity insertion of about 2.5 pcm/step. This insertion rate is sufficient to prevent a reactor trip from over temperature condition caused by temperature deviation (OTΔT).

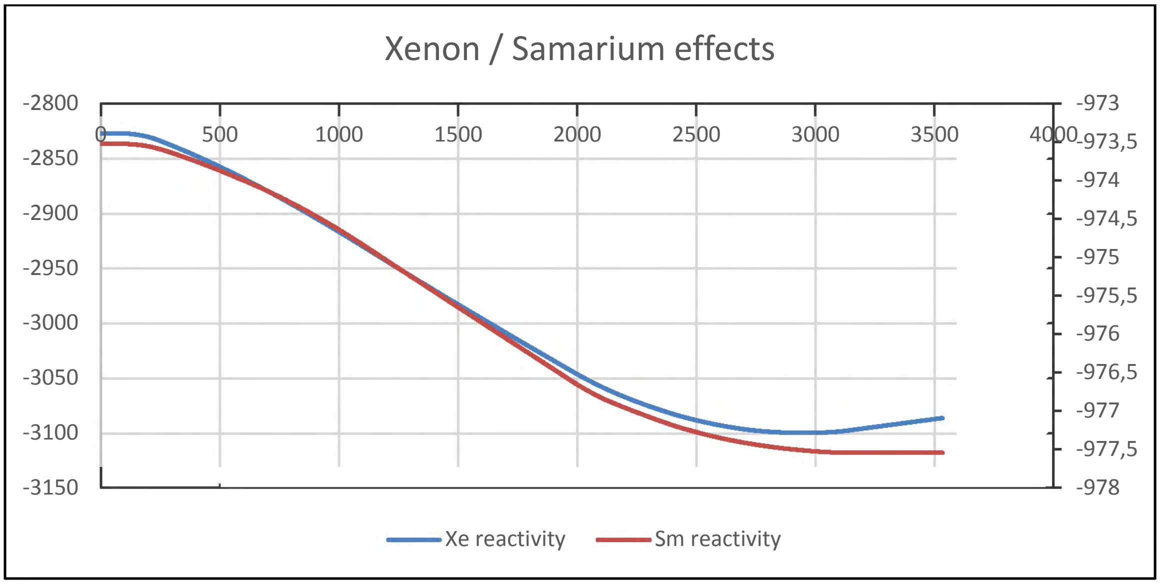


Figure 11: Reactivity added by the main fission product poisons Xe-135 & Sm-149

NB: Prior to the house load event the reactor has been operating at full power for sufficiently long duration therefore Xe & Sm were assumed to be at equilibrium.

The amount of negative reactivity added by Xenon was observed to be more than 3 times the amount added by Samarium. From the beginning of the transient till the end, Xe changed by ~10% while Sm change was negligible ~0.4%.

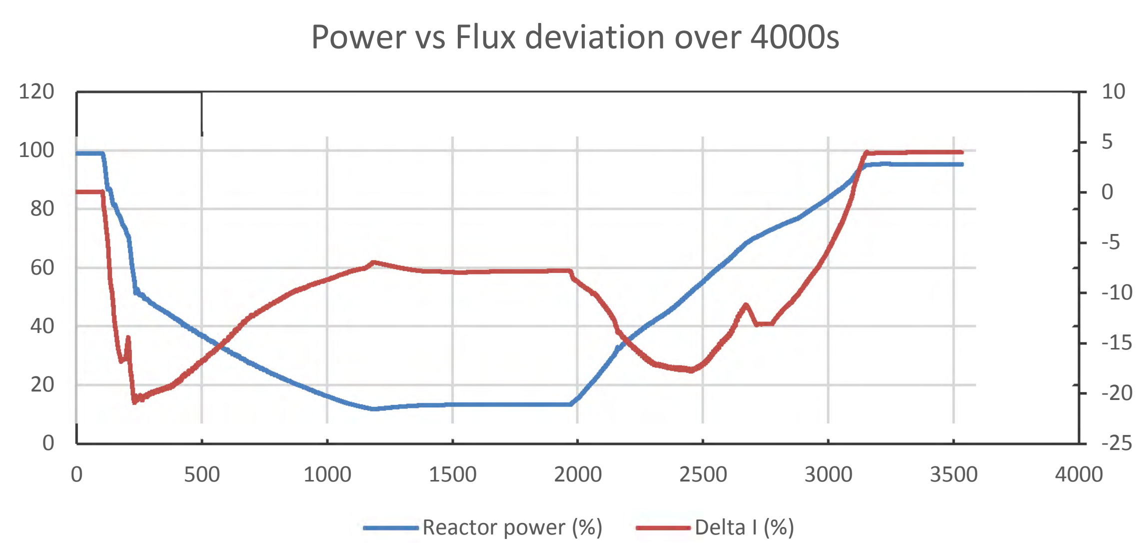


Figure 12: Flux deviation from reference (Delta I)

Delta I is the neutron flux difference between the upper and lower section of the core i.e. axial flux difference. As control rods inserted into the core, they suppressed the neutron flux at the upper section of the core and resulted in negative delta I i.e. more neutron flux at the bottom section of the core. This meant that more power was produced at the bottom section of the core than on the upper section. That poses an over-power threat to the reactor which could result in core damage. Shutdown margin, as indicated by various rod insertion alarms, was also inadequate since the control rod position was below the low insertion limit (Figure 7). Shutdown margin is an OTS requirement, and is achieved by borating the reactor coolant in accordance with the Physics Data Book[14] requirements. This operation was outside the permissible neutron flux difference operating band of (+/-5%). As illustrated in Figure 12 above, delta I was > 20%.

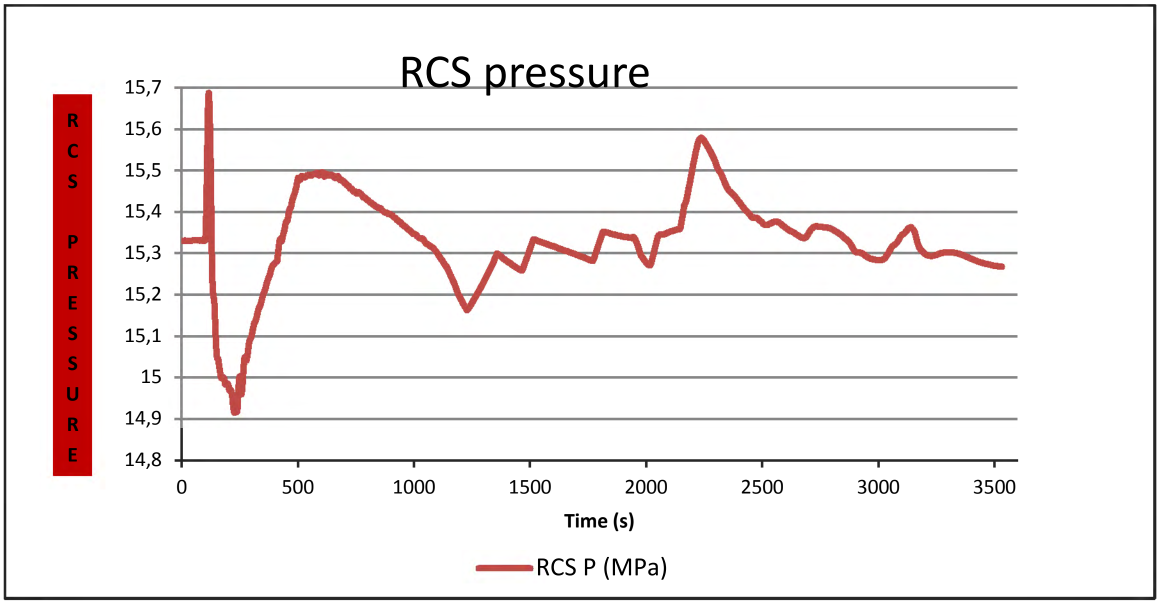


Figure 13: Pressuriser pressure transient

As the pressuriser level increased, with an increase in T_{avg}, the steam bubble was adiabatically compressed and pressure rapidly increased. The pressuriser pressure control circuit is fitted with a Proportional Integral Derivative (PID) unit, which reacted to the rate of change of the error signal. Although the spray valves opened immediately to reduce RCS pressure, the pressure

[14] Physics Data Book is a compilation of the reactor core data illustrated in figures and tables and it is specific to a certain nuclear fuel core. PDB is compiled by Reactor Fuel Engineers and used by the Reactor Operators to control fission reaction in the core.

continued to increase above the normal operation pressure set point of 15.5 MPa. The Power Operated Relief Valve (PORV) with the low set-point (15.9 MPa) was observed opening to stop the RCS pressure increase, therefore preventing the operation of safety relief valves and thus ensuring RCS integrity. The pressure increase was therefore terminated by the automatic opening of the PORV. NB: Figure 13 indicates that the PORV opened before the set point of 15.9 MPa and stopped the pressure spike at 15.69 MPa. This is because the PORV is rate sensitive.

It was observed that the pressure increase was terminated before the pressuriser level stabilised at normal working level (Figure 14). When the pressuriser level decreased towards the new set point, the RCS pressure decreased rapidly. The pressure drop was caused by the pressuriser spray valves & PORV being open and was exacerbated by the decrease in reactor coolant temperature illustrated in Figure 6.

The Pressuriser pressure control system automatically actuated the fixed heaters to increase RCS pressure while sending another signal to close the pressuriser spray valves. The pressuriser level and pressure control circuit together with the average reactor coolant temperature control circuits responded differently to the load rejection transient and resulted in an extended transient time. This can be observed in Figures 14 & 15.

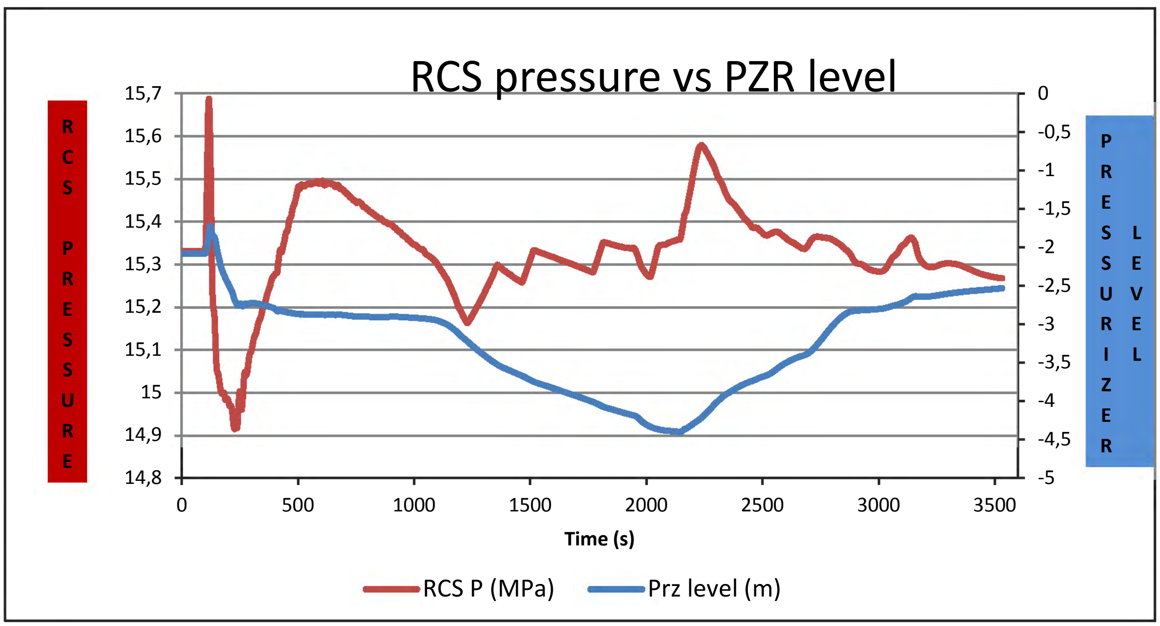


Figure 14: RCS pressure vs level transient

In this house load event, a maximum of 2.5% pressure spike during the RCS temperature swing was observed. It was triggered by reactor coolant temperature increase caused by the load reject. The power operated relief valve opened and caused a 5.1% pressure drop. The pressure transient was so big that the normal pressure control system could not stabilise pressure within normal operating limits. The automatic operation of the PORVs stopped the pressure increase and prevented the opening of safety valves. The operation of safety valves is undesirable because they are the ultimate protection of the primary system integrity. By design, safety valves are reliable in opening but they might not re-seat properly. A partially opened safety valve would result in a continuous system pressure drop due to inventory loss.

The temperature control systems (turbine bypass and control rods) then responded and restored the average T_{avg} to within normal operating parameters (see Figures 7 & 8) and the pressuriser pressure control system started controlling pressure (~1200s into the scenario).

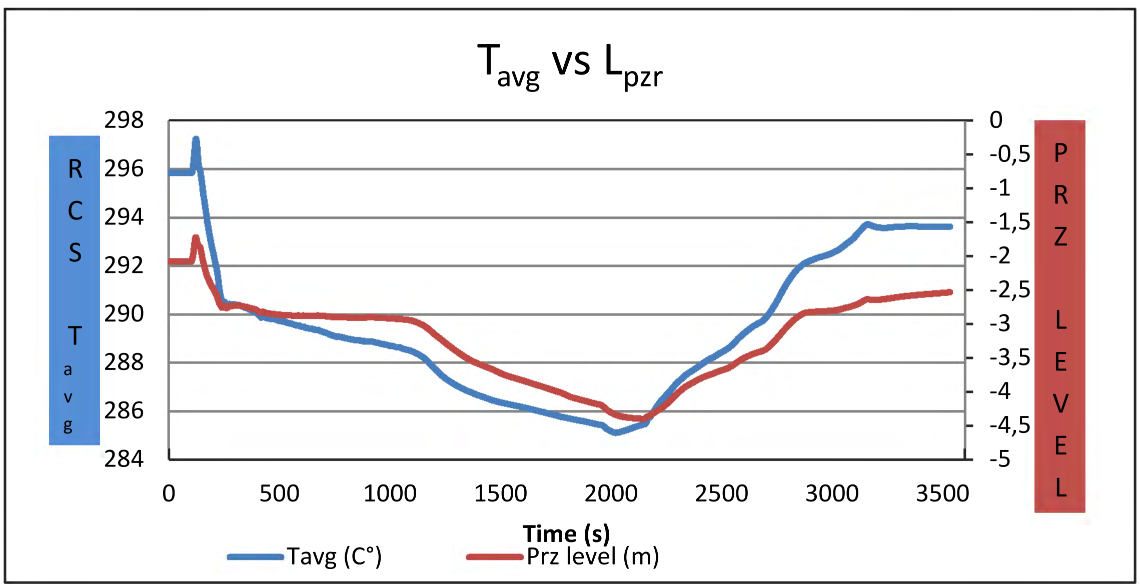


Figure 15: RCS temperature vs level transient

The relationship between the average reactor coolant system (T_{avg}) and the pressuriser level (L_{pzr}) demonstrate the effects that T_{avg} has on L_{pzr}. All the postulated accidents used in KSAR assumed T_{avg} > 284 C as a baseline, therefore T_{avg} < 284 C would mean that the plant operated outside the analysed region and that necessitates a plant shutdown for an engineering analysis. From Figure

15, it can be seen that without the load increase (@time 2000s) at maximum rate, T_{avg} would have decreased < 284 °C.

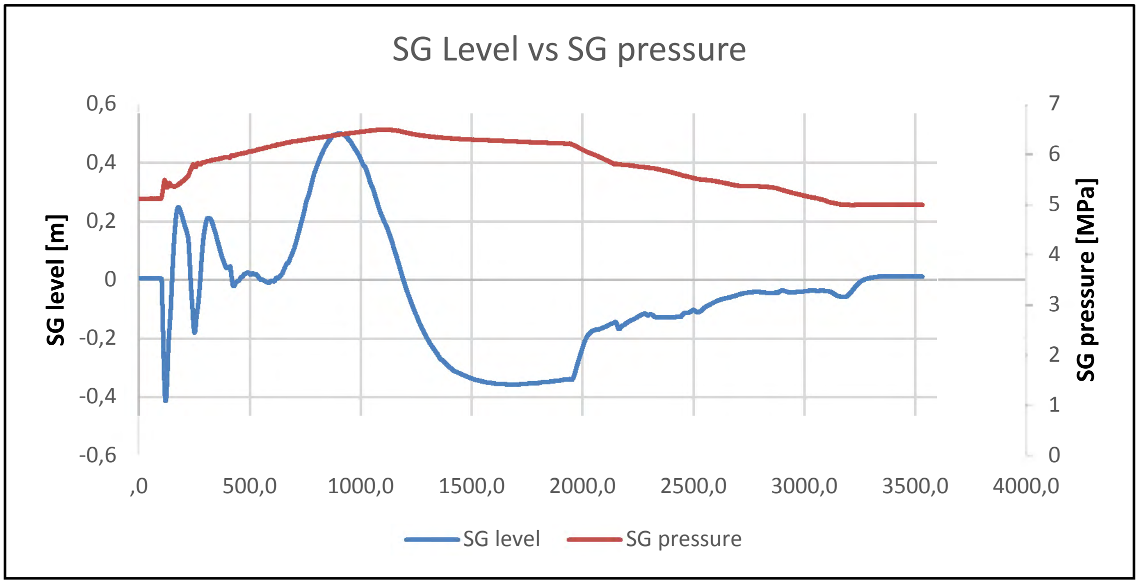
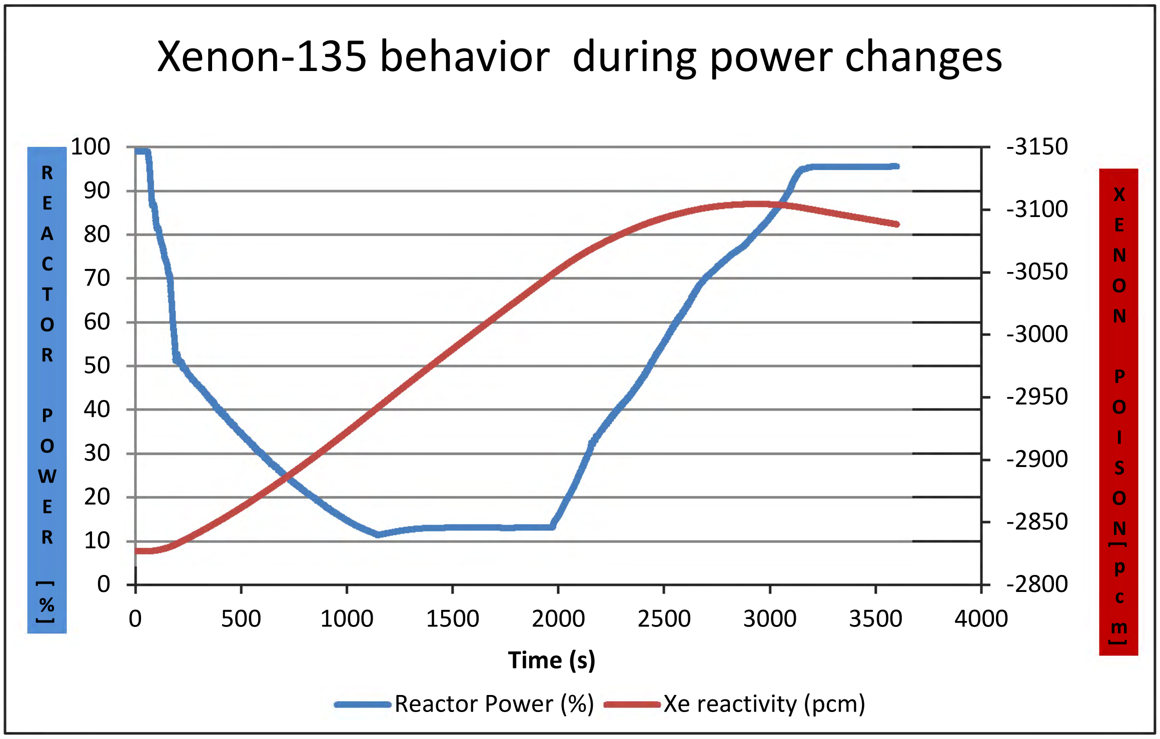


Figure 16: Steam generator shrink and swell transient

Steam generator (SG) level initially decreased as a result of the increase in steam generator pressure which then collapsed the steam bubbles and returned them to liquid form. This resulted in decreased steam generator level, referred to as a shrink effect in the nuclear industry. When the steam dump valves opened, the steam generator pressure decreased causing more steam bubbles to be formed. The SG levels then increased due to the swell effect. The level control systems then responded and controlled the level at the required level set point in accordance with the Steam Generator level program. It is important to note that the SG level variations (Figure 16) are an indication of the shrink and swell phenomenon, not an increased mass in the steam generator.



Reactivity addition by Xe-135 (Figure 19):

This scenario was initiated by triggering a house load event with reactor stable at full power and xenon at equilibrium (-2827 pcm). Xenon-135 is a fission product and the most important neutron poison in the core. It is produced in two ways in the core; directly as a fission product and by the decay of iodine-135 during reactor operation.

Nuclide chain:Te-135	⟶ I-135	⟶ Xe-135	⟶	Cs-135	⟶ Ba-135
Half-life:	19 sec	6.6 hr	9.1 hr		$2.6X10^6$ yr stable

NB: The decay of I-135 to Xe-135 constitutes 95% of all the Xe-135 production in the core, the rest being produced directly from fission.

As reactor power decreased from 100% to 10%, xenon concentration initially increased as a result of iodine decaying to produce Xe-135 and Xe-135 production directly from fission. This was caused by the reduced Xe burn-out which was a result of reactor power reduction. To illustrate the most possible burnout of xenon, at 2000s the reactor power was increased to full power while xenon was approaching the maximum peak conditions. NB: The only reason why such an operation was possible is because the xenon dead time[15] is not simulated. This is because the reactor has low thermal flux levels (about $5x10^{12}$ neutrons/cm^2-sec), most xenon is removed by decay as opposed to neutron absorption (NRC, 2007)[16]. It was observed than Xe build up added about 11% [2800 to 3100 pcm] of negative reactivity into the core.

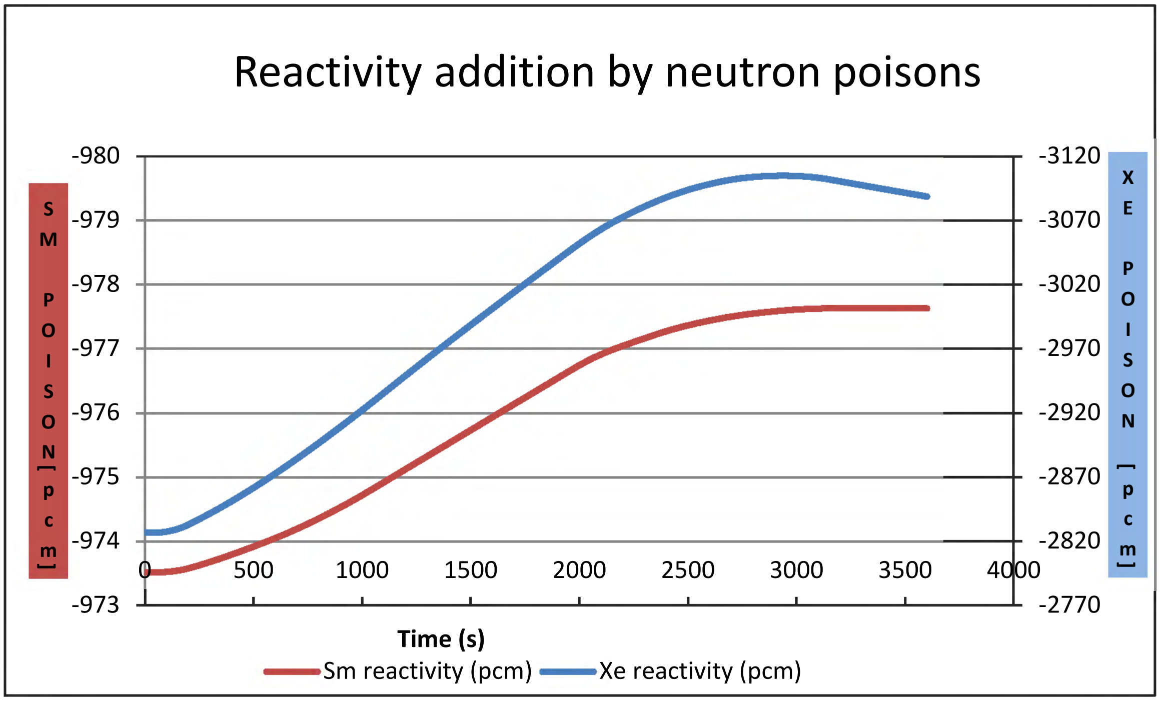


Figure 20: Reactivity added by the main fission product poisons Xe & Sm

Another neutron poison that is worth mentioning is Samarium-149 (Sm-149), because it has the second most significant effects on reactor operation due to its high thermal neutron absorption cross section. The behaviour of Sm-149 is different from Xe-135 because its nuclear properties are also different. The difference between Xenon and Samarium is that Sm-149 is not radioactive and does not decay. Sm-149 always adds negative reactivity because it builds up from loss of burn out. It can be seen (Figure 20 above) during reactor power reduction that Sm-149 builds-up as a result of the decay of promethium-149 and the reduction of the burn up factor, but once the

[15] Xenon dead time is the period of time where the reactor is unable to override the effects of xenon.
[16] PWR Generic Fundamentals - Reactor theory (Nuclear Parameters) DOE-HDBK-1019/2-93

equilibrium is reached, its concentration remains constant for the rest of full power operation. The Sm-149 build-up was stopped by the reactor power increase (back to full power) and that was due to Sm-149 burn-up. It can be seen from Figure 4.9 that Sm-149 poisoning is minor compared to Xe-135 poisoning, less by a factor of 3.

Control rods (Figure 7): The insertion of control rods was made worse by the absence of operator intervention i.e. no boron addition was performed. Control rods should not be allowed to insert below the low insertion limits. The low insertion limits are designed to:

- Minimise the consequences of rod ejection accident. That is why rods must be kept above the active core.
- Maintain acceptable in-core flux distribution. As rods insert into the core, they disturb flux distribution. The natural flux profile disturbance caused by the rod insertion is illustrated in Figure 12.
- Ensure adequate shutdown margin assuming that the rod with the highest worth is stuck out of the core.

It can be seen that without the addition of boron into the RCS, following a house load, the nuclear safety design limits will be challenged.

Figure 7 illustrates the challenge to the power distribution limits, which should not be exceeded during normal and abnormal plant operation. OTS specifies that rod insertion limits should always be maintained above their insertion limits regardless of power level. Due to the importance of this requirement, there are alarms in the control room to warn the operator when the insertion limit has been reached or is going to be reached and instruct the operator to take action.

Temperature transient (Figure 6): Due to the inherent design of the PWRs, the T_{avg} increase results in decrease in reactor power due to Doppler Effect. The T_{avg} channel is slower than the power mismatch channel because T_{ref} passes through a lag unit in the rod control system, before it is compared to T_{avg}. This lag is used to optimise the rods control system response to the other design transients of the rod control system e.g. 10% step load changes (positive or negative) and 5% / min ramp load changes (positive or negative).

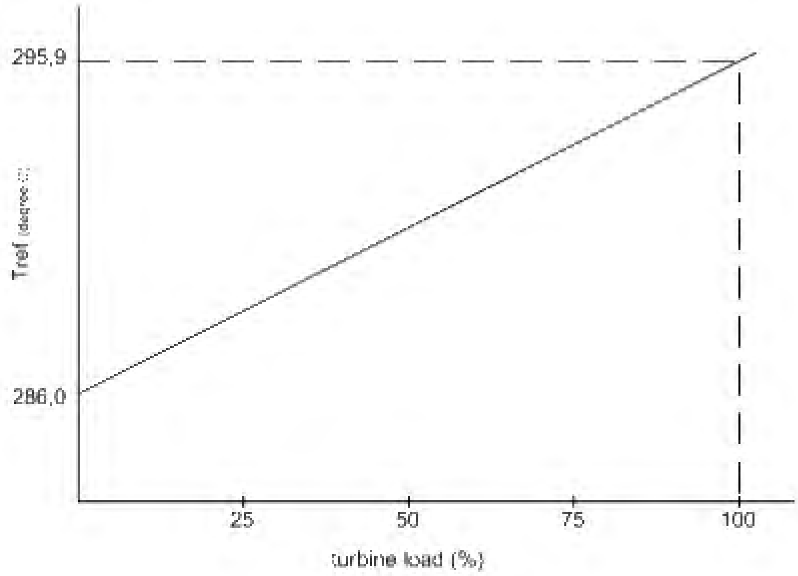


Figure 17: Tref Program (Eskom, 2007)

T_{avg} decreasing below 286 °C during a house load event is not an acceptable occurrence. In this case, a carefully controlled dilution would have been used to offset the xenon effect instead of increasing load. Increasing reactor power from 10%, while xenon is adding maximum negative reactivity at EOL, is not possible in a reactor as big as the Koeberg reactors. It would take about 10 hours for xenon to reach its peak, after which a load increase could be initiated.

Level transient (Figure 10): The pressuriser level transient threatened the isolation of letdown flow which occurs automatically when level drops below -4.6m. A letdown flow isolation would have posed an indirect threat to RCS integrity since the pressuriser would have filled up until 2.6m and the reactor would have automatically tripped.

Pressure transient (Figure 13): The pressure transient observed during this house load scenario could lead to clad creep and also threaten the sub-cooling margin in a PWR design. The pressure increase causes a reduction in the gap between the fuel pellet and the clad as illustrated in Figure 4 & 5. The amount of heat transferred from the fuel to the coolant gets affected i.e. the rate of heat transfer decreases. This results in an increased operating fuel temperature which then leads to high LPD and this threatens the maximum fuel temperature limit. To protect the RCS integrity and prevent DNB[17], there are two automatic reactor trips associated with pressure excursions. The reactor is designed to trip automatically if the RCS pressure increases to 17 MPa or decreases to 13 MPa. During the house load simulation scenario, the RCS pressure peak was less than 16 MPa therefore the integrity of the primary system was not challenged.

<u>Axial offset (Figure 12):</u> Delta I is used for reactor protection, it provides overpower protection and places operational restrictions in case of a skewed flux i.e. uneven power distribution. In order to maintain the heat flux hot channel factor within the acceptable limits, the axial flux difference must be maintained less than 5%.

During normal operation, the LPD must be less than 418 W/cm; this value ensures that if a LOCA were to occur, the maximum cladding temperature would not exceed 1 200°C. Therefore, maintaining ΔI within the trapezium (Figure 18 - design limits) would ensure that core melt would not occur. The ΔI operating mode is determined by the following conditions:

- Reactivity is controlled by adjusting the reactor coolant system boron concentration.
- Axial offset is controlled by control rod motion.

Authorized Band for Axial Flux Difference is governed by OTS Limiting Condition of Operation. Maintaining the axial flux difference [ΔI] for a power above 15%Pn, within ± 5% of reference value [ΔI ref] which varies with power and burnup, will result in a constant axial offset.

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[17] Departure from Nucleate Boiling is defined as the point at which the heat transfer from the fuel rod rapidly decreases due to the insulating effect of a steam blanket that forms on the control rod surface when the temperature continues to increase (NRC, 2014).

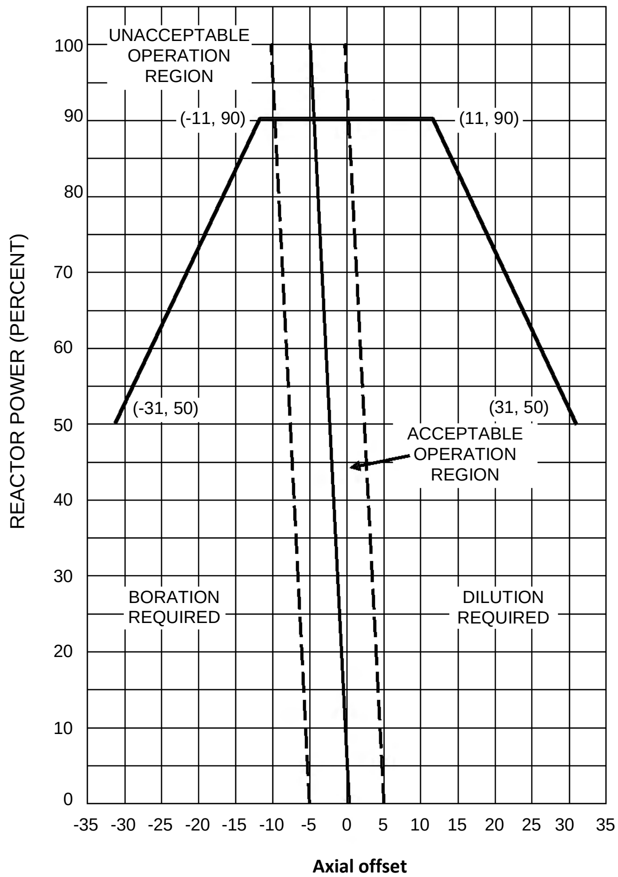

Figure 18: Axial flux difference limits as a function of rated thermal power (RFE, 2015)

CHAPTER 5: CONCLUSION

5.1 NNR would not allow the Koeberg units to follow load

The nuclear regulator lists nine major safety functions that must be maintained to ensure safety of PWR nuclear power plant operation (NNR, 2008):

- <u>Heat removal from reactor core</u> – main turbine generator system, steam generators, steam dump system and turbine bypass control.
- <u>Reactivity control</u> – control rods drive system, chemical and volume control system.
- <u>RCS inventory control</u> – pressuriser level control system and chemical and volume control system.
- <u>Reactor pressure control</u> – pressuriser pressure control.
- <u>Containment integrity</u> – pressuriser relief tank, containment cooling system, containment spray system.
- <u>Radioactive release</u> – liquid waste system, waste gas disposal system.
- <u>Electrical power supply</u> – emergency diesel generators, A.C. and D.C. Electrical distribution.
- <u>Instrumentation</u> – reactor protection system, nuclear instrumentation system, process radiation system.
- <u>Plant service systems</u> – component cooling water system, spent fuel pool cooling system, fire protection system.

From the chapter 4 analysis and discussions it was evident that 5 out of these 9 major safety functions were challenged and therefore nuclear power plant safety was compromised.

<u>The heat removal from the reactor core</u> was excessive resulting in moderator temperature decreasing below the no load temperature value of 286˚C (Figure 4.3). This affects *<u>reactivity control</u>* by adding positive reactivity to the core because of the negative moderator temperature coefficient (-MTC)[18]. *The light water reactors are designed with a -MTC which is a safety function to shut down the reactor in case of any incident that causes a rapid temperature*

[18] -MTC means that there is less than optimum amount of moderator between fuel rods, therefore an increase in RCS temperature and voids decreases K_{eff} of the system and inserts negative reactivity and vice versa.

increase. The physics behind this phenomenon can be explained in terms of the six factor formula. In this event, the low insertion limits of the control rod banks were exceeded (Figure 4.4) and that is not allowed in accordance with Operating Technical Specification (OTS). The decrease in *reactor coolant system inventory* below -4m level in the pressuriser (Figure 4.6), as a direct result of reactor coolant temperature-drop, increases the demand from the reactor boron and water make-up system. Letdown flow automatically isolated to reduce the effects of a loss of coolant, and pressuriser heaters automatically de-energised for heaters burn-out protection. The *reactor pressure control* was therefore unstable as can be observed in Figure 4.11. This transient resulted in a significant increase in nuclear waste water production and later yielded more nuclear waste water that needed to be treated. This puts a huge burden on the liquid radioactive waste processing plant. This has a direct effect on *radioactive release* which is one of the major safety functions that must be maintained at minimum in a NPP. The loss of shutdown margin indicated in Figure 4.9 (Delta I) is a non-compliance to OTS, therefore not allowed during normal plant operation.

Accurately measuring the reactor water level in a NPP is critical to safe operation of the plant. Since the Generation II NPPs, like Koeberg, do not have reactor water level indication, load transients that result in decreasing pressuriser level (as indicated in Figure 4.6) pose a threat of core uncovering.

Based on this technical finding, the Koeberg units cannot be operated safely under load following conditions. NNR would therefore not issue the Koeberg plant with a licence to operate as a load following station.

5.2 The current plant design does not allow load following

Flexible operation of the Koeberg Operating Units is technically possible but it will require plant modifications, procedure reviews & changes and the FSAR will have to be revised. Plant modifications could take up to 5 years, then procedural changes could take up to 2 years and the SAR review would take about 5 years. Such changes would be outside the scope of running Koeberg as stipulated by NIL-01, therefore extra funds and resources would be required. An Environmental Impact Assessment would also need to be performed.

In order to meet the IAEA requirements and guidelines, a technical review and relicensing of Koeberg would be necessary. In a developing country like South Africa, these processes can be long, tedious and costly. The time and cost of plant modification, loss of revenue, training of operating and maintenance personnel and relicensing the NPP and Reactor Operators cannot be justified.

In addition, the Koeberg units only contribute about 4.5% supply to the South African grid, therefore they achieve the highest efficiency and economic benefits when operating as baseload stations i.e. at full power. The grid input from nuclear plants is not big enough to require load following by the existing plants and since electricity price is regulated in SA, there is no incentive for the NPPs to operate at reduced load.

The stipulated Emergency Core Cooling System (ECCS) criteria requires the NPP manufacturers to design emergency core cooling systems that can accomplish a cladding temperature of < 1204 °C following a LOCA. As illustrated with RCS temperature spike, pressure spike and flux deviation (Figure 4.3, Figure 4.9 & Figure 4.10 respectively), the ECCS criteria would be challenged. The 1204 °C limit meant to prevent the Zirc-Water reaction, which would result in high hydrogen production inside containment during a LOCA. That would be a direct threat to the 3rd radiation release barrier (the containment) because of the increased risk of hydrogen explosion.

All Generation III power plants are designed with load following capabilities and it is highly likely that the next fleet of nuclear plants to be built in SA will be Generation III. NNR will have to deal with challenges of limited resources and experience in order to regulate load following NPPs and deal with technical issues that could be technology specific. NNR will therefore have to invest in staff training and hiring specialists in order to understand the operating limits and conditions, the design (fuel & reactor control), inspection and maintenance programs, supporting research & development programs, periodic testing requirements and plant commissioning.

5.3 Extended Reduced Power Operation not recommended

As illustrated in chapter 4 (Figures 4 & 5), when the core is operated at reduced power for significant amounts of time, the clad may creep as a result of the differential pressure between RCS and the inside of the rod. This will be a direct threat to core safety limits because it may cause the clad to come into contact with the fuel pellets and therefore result in fuel failures.

The rapid temperature increase (> 28 °C) observed with the power increase (Figure 4.3) is not acceptable because it indicates a rapid rise in fuel temperature. Such an operation would violate OTS and challenge the design basis of the plant.

During the rapid power increase from low power operation, the temperature of the fuel will rise rapidly and the pellets will expand. Although the zirconium alloy cladding is selected because of its good ductility, the expansion of the pellet can result in pellet-clad interaction which stresses cladding causing the clad to crack or fail. Once the fuel expands and gets into contact with cladding they can react chemically and the chemical reaction produces a brittle layer that thins the cladding wall (Chichester, 2012).

In order to ensure safe plant operation, Koeberg OTS imposes a limit of 3% per hour on the rate of power increase. Therefore the current Koeberg units cannot be operated safely in load following mode, since the load ramps could be up to 5% per hour (depending on grid demand).

5.4 Load Following should be considered for future NPPs in South Africa

Considering the South African Government's energy mix plan (IRP2010), to increase nuclear power input into the grid, NPP manoeuvring capabilities should be a mandatory requirement for the planned nuclear power plants. The future of electricity production in SA dictates that nuclear power plants should be capable of load following. This will require Gen III reactors. Therefore the manufacturers of major equipment like control rods, turbines, and steam governor valves will have to enhance the equipment design. More robust equipment, that would withstand the additional equipment stresses added by flexible load operation, would be necessary.

Load following should be considered for the new plants in order to maintain grid stability. The current grid stability would be compromised by an addition of a big nuclear plant generating capacity without frequency control capability. This will make the grid more unstable. Since NPPs operate for a long time (about 60 years), smart grid combined with renewable energy input could change the current energy management solutions and it might then be mandatory for NPPs to load follow. For the envisaged South African energy mix, during low demand it would be cheaper to maintain NPPs at low power than to shut them down;

- Fewer resources would be required compared to the case where the plant would be shut down.
- Start-up time would be shorter.

- Less borated water would be used and therefore reducing waste water production.

For these reasons it will be advantageous for the future NPPs to be capable of load following.

5.5 Recommendations for Future Work

1. Impact cost analysis of Gen II NPP when considering life extension. This might negatively affect the Koeberg Operating Units considering that major equipment spares might be obsolete due to lack of demand or the power utility (Eskom) might have to bear storage costs.

2. In order to understand the cost of cycling a 970 MW NPP, a study similar to the one done by Aptech Engineering in 1997 would have to be conducted with specific focus on South African conditions (NETL, 2012).

3. Financial impact of modifying the Koeberg Gen II units for load following. Benefits of load following capability will have to be weighed against the costs of modifying the plant.

4. A more in-depth study is necessary to fully understand how load following would impact;
   i. Plant reliability
   ii. Change in plant radiation
   iii. Nuclear waste production

5. The change in Probabilistic Risk Assessment for postulated accident during load following, with more focus on rod ejection accident.

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