IAEA-SM-310/ 94

SAFETY RE-ASSESSMENT OF AECL TEST AND RESEARCH REACTORS

D. J. WINFIELD

Chalk River Nuclear Laboratories

ATOMIC ENERGY OF CANADA LIMITED

IAEA-SM-310/94

SAFETY RE-ASSESSMENT OF AECL TEST AND RESEARCH REACTORS

ABSTRACT

Atomic Energy of Canada Limited currently has four operating engineering test/research reactors of various sizes and ages; a new isotope-production reactor MAPLE-X10, under construction at Chalk River Nuclear Laboratories (CRNL), and a heating demonstration/test reactor, SDR, undergoing high-power commissioning at Whiteshell Nuclear Research Establishment (WNRE). The company is also performing design studies of small reactors for hot water and electricity production. The older reactors are ZED-2, PTR, NRX and NRU; these range in age from 42 years (NRX) to 29 years (ZED-2). Since 1984, limited-scope safety re-assessments have been underway on three of these reactors (ZED-2, NRX and NRU). ZED-2 and PTR are operated by the Reactor Physics Branch, all other reactors are operated by the respective site Reactor Operations Branches. For the older reactors the original safety reports produced were entirely deterministic in nature and based on the design-basis accident concept. The limited scope safety re-assessments for these older reactors, carried out over the past 5 years, have comprised both quantitative probabilistic safety-assessment techniques, such as event tree and fault tree analysis, and/or qualitative techniques, such as failure mode and effect analysis. The technique used for an individual assessment was dependent upon the specific scope required. This paper discusses the types of analyses carried out, specific insights/recommendations resulting from the analysis and indicates the plan for future analysis. In addition, during the last 4 years, safety assessments have been carried out on the new isotope-, heat- and electricity-producing reactors, as part of the safety design review, commissioning and licensing activities.

1. INTRODUCTION

The rapid increase in the use of probabilistic safety assessment (PSA) applied to power reactors since the early 1980's has naturally led to increased application of this technique internationally to test and research reactors. Compared with power reactors, the limited resources available in typical establishments operating research reactors, and the smaller number of systems associated with these reactors, generally results in the scope of these PSA studies being of a more limited nature. On the other hand, the diverse nature of the larger research reactors introduces additional features associated with specialized test facilities, which are not relevant to power reactors. It has been clearly recognized that recent changes in the regulatory climate, both internationally and nationally, will result in more stringent future requirements for our older research reactors. Our safety re-assessments to date have however originated from internal safety review requirements, rather than from an external regulatory requirement, although the anticipation of closer externally audit is clearly implicit in the program to date.

The scope of the present program is currently being substantially increased, to provide a more wide ranging and more detailed safety re-assessment, directed towards the major facility, NRU. The main focus will be to satisfy the long-term company objectives for an economical and adequately safe life-extension program. In addition, anticipated future external regulatory demands will be of prime importance in this program.

2. REACTOR COMPARISON AND RE-ANALYSIS SCOPE

Table 1(a), (b) summarizes the manpower spent to date and proposed to be spent, on safety re-analysis of the various reactors. The manpower does not include operational safety assessment provided to the Reactor Operations or the Reactor Physics Branches for compliance monitoring, investigation of abnormal incidents and general safety overview of existing reactors. Safety analysis provided for new experimental facilities, tests or irradiations required to support the ongoing research/test program is also not included. The time estimates also do not include developmental work for staff or analytical tool development. For instance, a user-friendly

MacIntosh computer code, APPLETREE, is under development, to increase efficiency in graphic and quantitative analysis of fault trees.

2.1. NRU

NRU went critical 1957 November and was designed for a 400 MWt rating with natural uranium fuel, but is currently licensed for 135 MWt with a 93% enriched fuel charge. The fuel enrichment will be changed entirely to 20% low enriched uranium (LEU) by 1991. The reactor is D_2O moderated and cooled, by the same main coolant circuit, and has a single, gravity-drop, rod-shutdown system and an H_2O emergency cooling system, operating on either an auto (gravity flow) or manually initiated (pumped flow), once-through circulation mode. NRU is a multi-purpose nuclear research, isotope production and engineering test facility. It also has four independent loop test facilities used for short- and long-term experiments with various materials and severe fuel damage tests. The reactor is on-power refuelled and operated on a continuous basis by the Reactor Operations Branch of the CRNL Major Facilities Division.

The re-assessment to date [1] has comprised a limited-scope level 1 PSA (core damage model) and a limited-scope level 3 (off-site consequence model). No level 2 (containment release modelling) analysis was included as it was assumed, for the gaseous and volatile fission products, that a core melt would release conservative fractions of these to the atmosphere, via the reactor ventilation stack. An iodine Emergency Filter System (EFS) had also recently been placed in service. The doses for the consequence model therefore take credit for the availability of this EFS. Fault tree analysis on the EFS has also now been completed [2], in order to assess whether the design for the EFS was capable of achieving the target design availability.

The level 1 analysis calculated the core melt frequency with the reactor in both the operating and shutdown states, utilising fault tree methods. The initiating events were not explicity determined first, as in conventional power reactor PSA, but rather derived from fault tree logic, to the resolution where they are revealed as basic events. This approach reduced the number of fault trees otherwise required utilising the initiating event/event tree/fault tree approach. The relatively low complexity of the system design was such that this type of modelling was practical. The accident sequence cutsets so derived are equivalent to those that would be found utilizing the event tree/fault tree power reactor PSA methods. The scope of the analysis took no credit for the emergency-cooling system operation but full credit was taken (i.e., the analysis excluded the risk of failure) for the secondary cooling system, and those parts of the auxiliary systems capable of being isolated from the main coolant system, and also for successful operation of the shutdown system. Reactivity-initiated events have not been considered. Uncertainty and common mode analysis was included but external event analysis was not. Experimental facilities, including loops, were also not part of the current analysis.

The analysis, providing level 3 PSA information, was based on accident sequences resulting in 100% core melt from full power, occurring within a 10-minute time period, equilibrium fuel burnup conditions, assuming successful operation of the shutdown system. The scope involved on-site and off-site doses resulting from discharge of gaseous/volatile fission products from the reactor ventilation stack. A Gaussian plume atmospheric dispersion model was used for the off-site dose and a box (uniform concentration) model used for the direct dose on site. The objective was to re-assess the on-site and off-site noble gas dose and iodine dose for comparison with the doses calculated in the original reactor safety report for this design-basis accident. The original analyses used conservative dose estimates by not correcting for decay and not using the complete set of noble gas and iodine isotopes released. In addition, the standard plume dose models and nuclear data used have undergone improvements since the original report was issued. The overall uncertainty factor: of typically a factor 5, for the off-site dose still applies, due to inherent uncertainties primarily with the Gausssian plume dispersion modelling assumptions. What the re-assessment provided is a less conservative, more accurate, dose estimate than available in the original safety report.

A proposal to substantially extend the scope of this re- assessment is currently in the planning phase. This proposal is for a full scope safety re-assessment of the reactor using PSA techniques which will extend and update the existing PSA and deterministic analyses. This will be done in conjunction with a technical assessment of NRU. The primary objective of the latter is to determine what will be required to operate NRU until 2010. The scope of the technical assessment will include aspects of NRU reactor operation necessary to satisfy the diverse requirements of the facility's customers. These aspects will include the reactor systems, buildings and structures, associated services, effluent systems and operations/maintenance support requirements.

The proposed safety re-assessment will also address regulatory/licensing issues, opportunities for cost reduction/efficiencies, component and system upgrading necessary to upgrade the asset, calandria vessel replacement, staffing and maintenance issues and provision of new and improved experimental facilities.

2.2. ZED-2

ZED-2 became critical in 1960 September and is the major facility used in support of and operated by the Reactor Physics Branch. It is a versatile D_2O-moderated low-power [<200 watts) lattice-testing research reactor. The reactor is mainly used for reactor physics measurements on different fuel types. The reactor is also used for reactor physics measurements on mockups of reactivity control devices and fuel channels for power reactor lattices. Typically, it is run at powers up to 200 watts for periods usually under an hour a few times per week. The shutdown system consists of triplicated D_2O-moderator dump valves.

The re-assessment analysis scope for this reactor comprised a complete updating of the Safety Analysis Report and a detailed fault tree failure probability analysis of the reactor trip system. The fault tree analysis was initiated by the upgrading to solid state of the control/shutdown system ion chamber electronics. The original 1958 safety analysis report provided no systematic reliability analysis of the trip system. Quantitative and qualitative common mode analysis was an essential part of the new analysis. Assumptions of independent failure would otherwise reflect unrealistically optimistic availability, because of the duplicated and triplicated redundancy features of the system.

2.3. NRX

NRX went critical 1947 July, and is currently licensed for 42 MWt operation. It has 93% enriched fuel, heavy water moderation and light water once-through cooling system. It has rod drop and heavy water dump shutdown systems and an H_2O emergency cooling system, operating on either an auto (gravity flow) or manually initiated (pumped flow), once-through circulation mode. It is currently operated on a periodic basis (hot standby) for 8 h per week, to produce irradiated silicon, for neutron radiography work and also for neutron spectroscopy basic research, as a back up to NRU.

The scope of the safety re-analysis has been a failure mode effect and criticality analysis (FMECA) to identify key production and safety related potential problems and also an aging study to identify specific components that may limit the remaining scheduled reactor lifetime (5 years).

2.4. PTR

PTR went critical 1957 November, and is a light water pool-type reactor, with 93% enriched MTR-type fuel and a rod-drop shutdown system licensed for100 watts. It is currently used by the Reactor Physics Branch for self-powered flux detector calibrations and reactivity measurements on samples of materials used in components for NRU and NRX. The reactor is used infrequently and no re-assessment has been done, or is scheduled.

2.5. SDR

SDR went critical 1987 July, and reached 1 MWt (50% of nominal full power) 1989 May during commissioning tests. SDR is a LEU pool-type reactor, using primary system natural circulation, and is a progenitor of a higher-power 10 MW Slowpoke Energy System (SES) district heating reactor that AECL is currently designing. SDR is being used for demonstration of the low-temperature building heating concept and prototype testing of concepts and systems in support of the SES program.

As part of the design, commissioning and licensing activities, safety assessments utilising event tree, fault tree and failure mode, effect and criticality analysis (FMECA) have been used as an integral part of the initial safety analysis. As the reactor is of a novel design and has not yet completed high-power commissioning tests, there is no need for re-assessment. However, as a result of novel design features, on-going PSA is an integral part of the safety case presentations. Particular attention is being focussed on the potential for common-mode failure.

2.6. MAPLE-X10

MAPLE-X10 is a 10 MWt pool-type reactor utilizing LEU, an H_2O moderator/coolant and a D_2O reflector currently under construction at CRNL. It will be used as a dedicated isotope-production facility to supplement the isotopes produced by NRU.

The safety analysis to date [3] has followed a similar path to that of the other new reactor, SDR, and has involved a substantial component of PSA analysis to supplement deterministic analysis, as part of the design, commissioning and licensing activities.

3. RESULTS OF ANALYSES

3.1. NRU

The dominant core melt sequences and associated probabilities, accounting for recovery actions where relevant, identified from the limited-scope PSA analysis (section 2.1), are:

Reactor at power :

1. Loss of off-site power coincident with the unavailability of the emergency DC motor control circuits. (1.4×10^{-4}/a)

2. Loss of off-site power coincident with the unavailability of the Class 1 (battery) power supply. (3×10^{-4}/a)

3. Core melt due to large LOCA from severance failures in the large diameter (30 cm) non-isolable piping. (3×10^{-4}/a)

4. Core melt due to small LOCA's from severance failures of intermediate size (3-7 cm diameter) non-isolable piping. (1×10^{-4}/a)

5. Core melt due to small LOCA's from non-severance failures of large diameter (30 cm) non-isolable piping. (4×10^{-3}/a)

The LOCA's from the latter two events are expected to be mitigated by the emergency coooling system, although the current analysis does not credit its operation.

Reactor Shutdown:

1. Lowering of vessel level below vessel bottom volute. (1×10^{-5}/a)

2. Loss of off-site power and vessel D_2O level lowered for dump test and emergency DC motors unavailable. (1×10^{-5}/a)

Specific recommendations identified from the PSA were:

- implement routine test of battery status monitor alarm.
- separation of emergency dump panel dump valve pushbutton/position indicating light bulb.
- provision of an additional emergency DC motor control centre.
- eliminate diesel generator potential common-mode failure in fuel tanks' common vent line.
- in-service inspection of non-isolable expansion bellows on main pump discharge lines.
- in-service inspection of non-isolable large piping on main pump discharge lines.
- deterministic thermalhydraulic analysis of the effect of non-closure of emergency pump discharge check valve in the event of emergency pump failure.
- eliminate semi-annual test procedure to dump heavy water.
- provide improved cooling system for battery room.

Specific recommendations identified from the PSA conducted on the EFS filter building are discussed in detail in [2] and are summarized here as:

- increase the redundancy level of the high-radiation level sensors.
- modify the pneumatic circuitry of the automatic action ventilation isolating valves.

The dose calculations for a 100 % core melt, assuming 100 % noble gas release and 12.5 % radioiodine core release, with credit for the EFS, indicated off-site (6 km) boundary doses of 2 rem for noble gas and 5 rem for radioidines.

The current status of the proposed NRU re-assessment, discussed in section 2.1, is that planning to conduct the work has commenced and will be submitted for executive approval, September 1989. Following approval and identification of the implementation team, the assessment is anticipated to start without delay. The duration expected is from 2 to 3 years, with expected manpower commitments as indicated in Table 1(b).

3.2. ZED-2

The dominant trip system failure events and associated unavailabilities (failures per demand), identified from the PSA analysis (section 2.1), were:

1. Analogue trip parameter instrument failures:

   - linear power (2×10^{-3})

   - log rate comparator (7×10^{-4})

   - log power (4×10^{-4})

2. Failure of various combinations of critical trip relay contacts to open upon demand. (10^{-5})

3. Any two out of three dump valves failing to mechanically open when required. (6×10^{-5})

Specific recommendations and features arising from the analysis were to:

- implement routine testing of the low-air-pressure alarm on moderator dump valve pneumatic circuits.

- implement routine testing of dump valves' opening timer.

- ensure no coincident systematic error in operational ion chamber sensitivity is occurring, by using periodic power calibration using foil activation analysis calibration.

3.3. NRX

In the case of NRX, the qualitative FMECA, to identify any production and safety-relevant problems and the aging study resulted in the following activities:

1. Replacement of the triplicated vacuum-tube control system by a solid-state system. The replacement is being done in stages, one channel at a time. As part of this activity, the vacuum-tube amplifier response-time algorithm was modelled deterministically and the newly installed solid-state control response checked against this when installed.

2. In-service inspection on the 42-year-old process piping is being performed periodically.

3. A forced circulating system was added to the fuel bay water cooling, which was formerly a natural circulation system.

No further re-analysis on NRX is anticipated.

3.4. MAPLE-X10 and SDR

Although safety re-assessments are not yet required for these new reactors, for completeness the recommendations to date that have resulted from the use of PSA analysis are summarized. These recommendations are thus typical of a re-assessment analysis, even though the analysis was part of the design/commissioning stages of development.

For MAPLE-X10, the recommendations, based on both safety and production reliability considerations, were:

- safety system redundancy scheme changes.
- various control system interlock changes.
- control system redundancy scheme changes.

Similarly for the SDR, recommendations were:

- logic changes in shutdown system final trip relays.
- changes in trip reset pushbutton circuity.
- installation of a remote trip pushbutton facility.
- changes to operating and test procedures for mechanically driven reactivity control devices.

4. REVIEW AND REGULATORY PROCESS

The review and regulatory process used for AECL's research facitities that has been applied for the safety assessments above and future changes anticipated in the process, is discussed at length elsewhere in these proceedings, [4].

TABLE 1

(a) Safety Re-assessment Manpower to date for AECL Test/Research Reactors, 1984-1989			
(b) Safety Re-assessment Manpower proposed for AECL Test/Research Reactors, 1990-1992			
reactor	**reactor power**	**man-months (a)**	**man- months (b)**
NRU	135 MWt	20	100-200
NRX	42 MWt	13	0
ZED-2	200 watts	16	0
PTR	100 watts	0	0

REFERENCES

[1] WINFIELD, D.J. A probabilistic risk assessment of the loss of fuel rod cooling in the NRU 135 MW research reactor, AECL-8808, (Presented at the sixth annual conference of the Canadian Nuclear Society, Ottawa, June 2-5,1985).

[2] HICKEY, D.M. LOUNSBURY, R.L. Safety aspects of the emergency filtration system for NRU reactor, (Paper SM-310-95P, IAEA International Symposium on Research Reactor Safety, Operations and Modifications, Chalk River, October 23-27, 1989).

[3] COTNAM, K.D. Safety assessment of the MAPLE-X10 reactor, (Paper SM-310-96, IAEA International Symposium on Research Reactor Safety, Operations and Modifications, Chalk River, October 23-27, 1989).

[4] AXFORD, D.J. Nuclear safety review process at the Chalk River Nuclear Laboratories, (Paper SM-310-97, IAEA International Symposium on Research Reactor Safety, Operations and Modifications, Chalk River, October 23-27, 1989).