A SAFETY REVIEW OF THE NRU EFFLUENT HEAT RECOVERY PROJECT

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ABSTRACT

The NRU effluent heat recovery project (EHRP) diverts heated effluent water from the NRU process effluent weir and distributes the water for various heating applications in both the inner and active area at Chalk River Nuclear Laboratories (CRNL). The dominant hazard of the system operation is from leakage of tritiated heavy water from the reactor heavy water system into the light water system and the subsequent contamination of the steam system. Protective features include continuous leakage monitoring and automatic isolation of the recovery system. Modelling of the worst case accident, predicts a dose equivalent from tritium in steam humidification of about 26 mrem (260 μSv). The operation of the heat recovery project does not present an unacceptable risk to CRNL personnel.

INTRODUCTION

A CRNL proposal to recover waste heat from the process cooling water of the NRU research reactor process effluent weir was accepted by the Ministry of Energy, Mines and Resources for partial funding under a retrofit program. The project nominees also received an award (Discovery Award 1984) for proposing a new concept for using low grade heat from the NRU reactor effluent. The heat recovery scheme involves diverting up to 471 kg/s (6280 Igpm) of heated effluent water from the NRU process effluent weir and distributing it for various heating applications in both the inner and active areas at CRNL.

This paper concentrates on the safety and potential operational hazards of the system, outlines the features that will reduce the hazards, and examines the pathway for radioactive exposures that are modelled for both accident and chronic releases to the system. A brief description of the reactor cooling and the heat recovery systems precede these safety discussions.

NRU REACTOR DESIGN

The NRU reactor is a 135 MW (thermal), heavy water moderated and cooled research reactor located at Chalk River. NRU is used for fundamental research, radioisotope production and engineering experiments for power reactor development, including Loss of Coolant Accidents tests. The reactor is cooled by circulation of heavy water from the bottom of eight heat exchangers into a bottom header by eight pumps as shown in Figure 1. The header distributes the water via a tube plate to the rod cups and forces water up through the fuel rods. Outflow is mainly through the orifice section above the fuel, with a small flow where the fuel rods sit in the cups. The heated heavy water then flows into eight volutes at the top of the reactor and flows down through the eight heat exchangers and returns to the pump intakes. Process water is pumped from the Ottawa River, through the secondary side of the eight heat exchangers and returned to the river via the process weir.

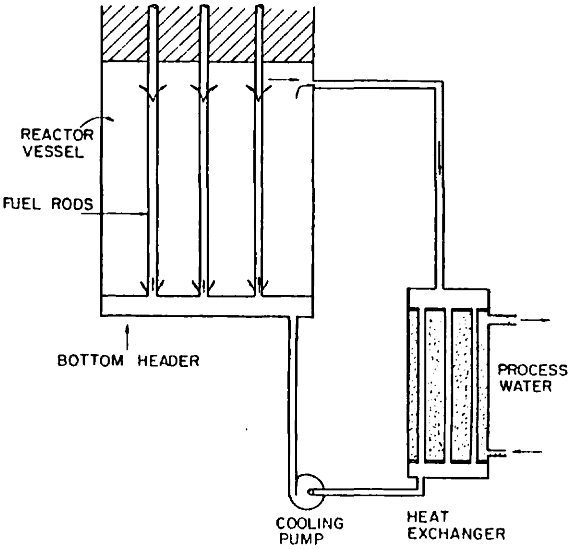

FIGURE 1: NRU COOLING GENERAL ARRANGEMENT.

The process weir is on the discharge side of the heat exchanger process water cooling lines with the spill level well above the top of the heat exchangers (Figure 2). The weir provides a constant back

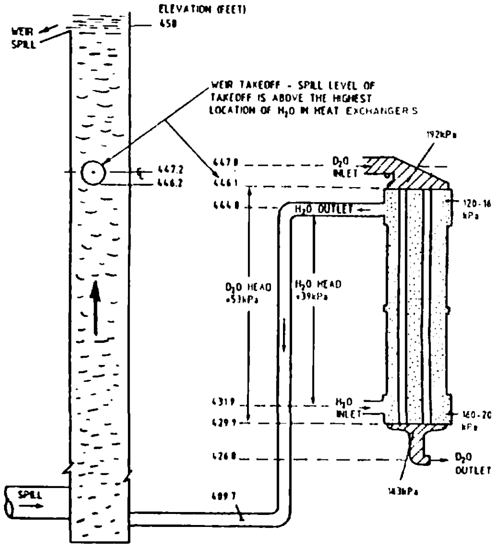

FIGURE 2: PROCESS WATER SYSTEM ARRANGEMENT.

pressure on the process cooling lines, ensuring that the heat exchanger shells remain full and ensures that the heat exchangers cannot be siphon drained. The weir also provides a means for thoroughly mixing the outlet water before temperature measurements are made for power calculation. Typical winter time process cooling water flows are 60 000 L/min while summer time flows peak at 90 000 L/min.

HEAT RECOVERY SYSTEM

The heat recovery project involves diverting 28 200 L/min (6280 Igpm) of the effluent water at approximately 30°C from the process effluent weir and distributing that water for various heating applications in both the inner and active areas at CRNL. The potential energy recovery is about 25 MW. The project is estimated to cost about $2.5 M (1985). With an estimated 70 percent availability, the payback period is 6.4 years with the system complete as it stands now (no south loop) or reduced to 4.7 years if the south loop is completed. Partial winter 1986 operation indicates that normal availability may be closer to 75% but can vary widely due to scheduled reactor shutdowns and unscheduled reactor trips. The five distribution circuits and their respective flows are shown in Table 1. The heated water will have both indirect and direct uses. Indirect uses include preheating the main plant service water supply and the inlet air to a number of buildings. Direct uses include main steam plant boiler makeup, fire water makeup and NRX process water makeup. The latter two were included in the design so that the likelihood of freezing the water supplies in storage tower, in cold winter temperatures, would be reduced.

TABLE I : HEAT RECOVERY SYSTEM DISTRIBUTION

CIRCUIT NUMBER	LOCATION	APPLICATION	MAX WNTER FLOW	
			Igpm	l / min
I	NRU Plenum and Fan 7	Space Heating	650	2900
2	Bldg 440	NRX Process Water	1000	4500
3	CRNL-North West Loop	Space Heating	1830	8200
4	CRNL- South, East-Loop	Space Heating	1670	7500
Gravity	Bldg 420 Power House	Boiler Feed	130	580
		Service Water Ht. Ex. Fire Water *	1000 AT 23°	4500
Total			6200	≃28200

* The 30°C water flows through the service water heat exchanger and then drains to the fire pump well

The 60 cm diameter weir take-off design was the product of a number of considerations and restrictions. The take-off location must not jeopardize the primary function of the weir as described earlier. It should avoid the possibility that a system pipe failure could drain the weir. This would also preclude the possibility of connecting with existing piping prior to entering the weir. The location of the weir take-off should be at a location that is well known with regard to the location of reinforcing bar and weir structure soundness. The location chosen should not create a major hazard due to water leakage. The elevation chosen is shown in Figure 2, with Figure 3 showing the multiple barrier design used to ensure leak tightness. Since the take-off is into an isolated room inside the building, leak detectability is provided. The main drawback to this design is that should a leak occur, that leakage will be into the building. Great care was taken to design a multibarrier seal and to demonstrate the grouting effectiveness before actual installation was undertaken.

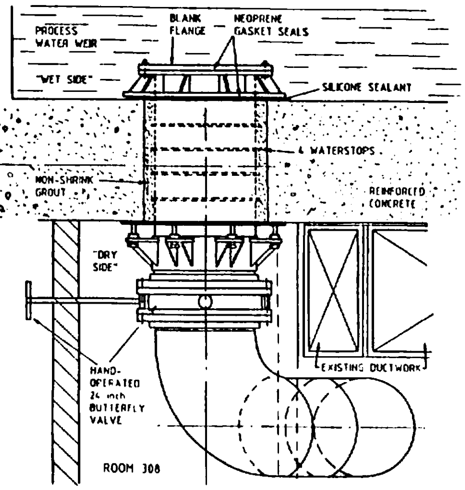

FIGURE 3: WEIR TAKE-OFF PIPING DETAILS.

A manual butterfly valve is installed, as an integral part of the take-off, to provide a means of isolation, should a leak develop in the take-off header piping. A blank flange can be installed on the wet side of the take-off as shown in Figure 3. This was used during completion of construction of the heat recovery system.

All five circuits have isolation valves which are closed automatically when the heat recovery system is tripped or shut down.

SAFETY CONSIDERATIONS

Operational Hazards

The main potential hazard from operating the heat recovery system is from the transfer of radioactivity to other buildings by way of the plumbing. An internal leak in one of the main heat exchangers could allow tritiated heavy water to contaminate the process water. With this water being distributed by the heat recovery system around the plant, three potential pathways for exposure to tritium exist. (The hazards from both fission products and activation products were analyzed in the original safety report [1], but are not considered here, since the tritium dose is the dominant hazard).

The pathway of most concern is coincidental leakage in the service water heat exchanger in the gravity feed circuit with a main heavy water heat exchanger leak because the preheated water is used wherever domestic water is used. Of primary concern was the fact that service water is also used for drinking water. To reduce the possibility of contamination of the service water, a double tube heat exchanger was installed in the powerhouse. This design provides three barriers that require coincidental failure and is a very low probability event. The vented annulus between tubes will indicate a leak from either side of the double tubes.

A second pathway for transfer of activity is the leakage in any system piping system with coincidental failure of a main heat exchanger tube. The random leakage in the piping would likely result in a low flow detection in that circuit and cause automatic isolation of the leaking piping or visible detection of water. This pathway has two barriers with the coincidental failure of the main heat exchanger considered a low probability event.

The third pathway, which has only one barrier is via the gravity feed line to the powerhouse. Effluent water is used as boiler makeup and would result in tritiated water contaminating the steam. Steam is still used in some areas for humidification of dry winter air. This third pathway has only one safety barrier, consisting of the main heavy water heat exchanger and is the pathway that the remainder of this paper will address.

Leak Detection and Activity Monitoring

The majority of the process water that discharges to the effluent weir has been used as secondary cooling for the reactor. Process water makes a single pass through the secondary side of the heat exchangers and is then returned to the river via the weir. If a leak should develop in one of these main heat exchangers, transfer of tritiated heavy water to the process water would be possible. The operating pressures in the heat exchangers (Figure 2) show that the leakage could be into or out of the heavy water system, depending on the leakage elevation.

Existing activity monitors, on the effluent lines, are used as the continuous or real time leak detection monitors. These monitors are located, in shielded boxes, next to each discharge pipe and are very sensitive to activity in the piping. These monitors will detect any activity in the piping, but during reactor operation, the isotope N-16 is the principal activity in the heavy water. N-16 is an activation product in the heavy water and is produced in quantity only when the reactor is operating. With a half life for N-16 of 7.5 seconds, activity detection in the heavy water becomes more difficult when the reactor is shut down. Travel time to the monitor also becomes important since levels of activity decay very quickly. This was a major factor in the decision to use existing monitors which are located approximately 15 seconds travel time from the outlet heat exchanger, the more likely leakage zone considering relative system pressures. Based on tests done when these monitors were originally installed, out leakages of the order of 2 cc/min of heavy water into light water will be detected. These monitors have a small radioactive source to keep them off the zero point and will alarm on low level to give protection against head failure. Thus these monitors give alarm indication of both high and low activity levels.

As a backup to these monitors, manual sampling on a once per eight hour shift basis is also done. Reliable detection of tritium in water samples in the active area lab can be made at a level of 0.5 μCi/L*. In the last few months levels have been measured at less than 0.1 μCi/L. A detection level of 0.5 μCi/L corresponds to a heavy water leakage rate of 1.5 cc/min with 20 Ci/L tritium content in the heavy water.

The heated water diverted from the process effluent is generally returned to the process drain after use in NRU and NRX or is collected in a dedicated drain. Both effluents are continuously sampled for releases, in compliance with regulatory release levels. The fire water circuits are normally collected in surface drains which are routinely sampled for activity levels. The dedicated drain should provide representative samples for release estimates for any unmonitored drainings.

Shutdown and Isolation

To reduce the chance of employee exposure to tritium contamination via EHRP, it is essential that a system shutdown and circuit isolation take place as soon as possible following the leakage initiation. Once the heat recovery system has been started up, it will trip off line should the effluent activity monitors actuate high - indicating a leakage, or low - indicating a failed monitor, or should a reactor trip or shut down take place (recall that the activity monitor of N-16 is greatly reduced when the reactor is not operating).

A rapid shut down is available via an emergency shutdown switch located remotely in the NRU control room (for use by the Reactor Shift Engineer should he decide a rapid shutdown is required) or via a local switch at the pump/system control panel. Normal shut downs are made by a microprocessor controller which performs a sequence shut down of the five distribution circuits on demand.

For operational convenience, the system will also trip off line should the effluent water temperature drop too low and present the possibility of air intake coils freezing. Individual circuits will also trip off line if low flows are detected in that circuit.

When the isolation control signal is actuated, the pumps automatically shut down and pump discharge valves shut. The gravity feed line has isolating valves at NRU and the powerhouse. The powerhouse valve has a limit switch in place to inform operating personnel of the status of the isolating valve. This procedure will isolate the five circuits and prevent the spread of radioactive contamination, should that be the cause of the EHRP automatic trip.

Accident Dose Estimate

A guillotine failure of a single tube in the upper regions of a main heat exchanger is believed to be the most serious, credible accident associated with the operation of the EHRP. If we assume that the continuous monitor fails to detect the leakage, the tritium activity transferred to the steam used for live steam humidification would result in an estimated whole body dose equivalent of 25.8 mrem (0.26 mSv) for an 8 hour exposure.

The leakage rate from the failed tube is estimated to be 9 L/min and continue until a total leakage of 500 L will occur. When 500 L has been removed from the heavy water system (after 55 minutes), the receiver expansion tank (a reactor heavy water system tank) level will drop to the point where reactor D_2O level control pumps will gas lock, and the heavy water level in the reactor will drop below the automatic trip level. At this point the reactor trip will cause isolation of the heat recovery system and an end to the further contamination of the steam system. It is interesting to note that a catastrophic failure of all tubes in the heat exchanger would result in an out

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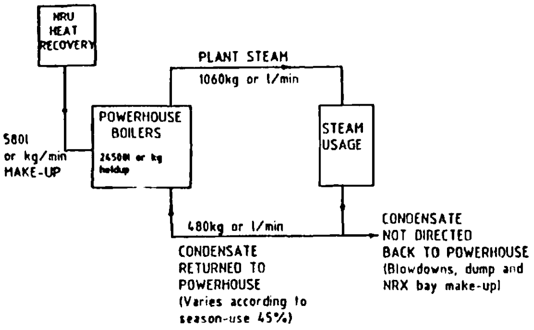
* 1 Ci = 37 GBq

leakage of more than 15 000 L/min and result in a reactor trip in about 2 seconds, and a subsequent system isolation long before the contamination front would reach the isolating valves. It takes about 70 seconds for water flowing into the bottom of the weir to reach the distribution pumps.

To make an estimate of the dose due to tritium in steam used for humidification, one must determine the tritium contamination in the steam. A model of the boiler steam system was used to estimate the tritium concentrations in steam over the eight hour exposure period. A simplified model of the boiler/steam system is shown in Figure 4. Keep in mind that the total flow of process water to the effluent system is 59 940 L/min with 580 L/min maximum diverted to the boiler feed makeup. If a total of 500 L of 20 Ci/L water leaks into the effluent over the 55 minutes, then 10 000 Ci will have contaminated the total flow during the release period. This will result in tritium entering the boiler at the rate of 1.82 Ci/min during that initial 55 minutes. At the end of that period, system isolation will take place and tritium in the steam/condensate system will be diluted, based on the estimate that there is a 45% steam condensate return and contamination free makeup from the river will be used.



PEAK HEATING SEASON - FLOW CONDITIONS

(NOTE: 1 l OF WATER PRODUCES 1 kg OF STEAM)

FIGURE 4: STEAM SYSTEM FLOWS.

The concentration of tritium in the boiler holdup inventory represents the tritium contamination that will be produced in the steam used for humidification. The contamination level is dynamic and will be analyzed for condition during three periods of time:

1. From t = 0 to 30 minutes. Contaminated makeup and clean condensate is entering the boiler and contaminated steam is leaving the boiler.
2. From t = 30 to t = 55 minutes. Contaminated makeup and condensate is entering the boilers and contaminated steam leaving the boiler (condensate contamination is conservatively estimated at current contamination levels).
3. From t = 55 minutes to t = 480 minutes (8 hours). Clean makeup and contaminated condensate is entering the boiler and contaminated steam is leaving the boilers.

To model the activity in the boilers, the amount of tritium in the boiler holdup water is defined as "A". From Figure 5, the rate of change of that activity can be defined and the differential equation solved for each period under consideration.

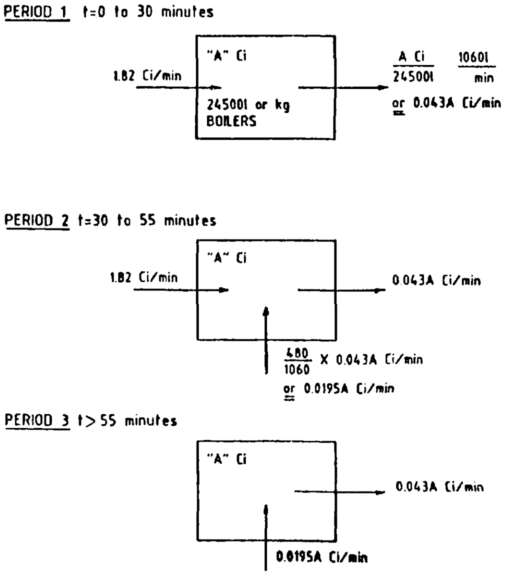


FIGURE 5: MODELS FOR BOILER OPERATION.

Period 1 (t = 0 to 30 minutes)

The rate of change of the activity in the boiler is

$$dA/dt = 1.82 - 0.043A \text{ Ci/min}$$

At t = 0, A = 0 and the activity in the boiler during period 1 is:

$$A = 42.3 - 23.3e^{0.6-0.043t} \text{ Ci} \qquad (1)$$

Period 2 (t = 30 to 55 minutes)

The rate of change of the activity in the boiler is

$$dA/dt = 1.82 - 0.043A + 0.0195A \text{ Ci/min}$$

At t = 30 min, A = 30.6 Ci and the activity in the boiler during period 2 is:

$$A = 77.4 - 42.5e^{0.801-0.0235t} \text{ Ci} \qquad (2)$$

Period 3 (t = 55 to 480 minutes)

The rate of change of the activity in the boiler is

$$dA/dt = -0.043A + 0.0195A \text{ Ci/min}$$

At t = 55 min, A = 51.4 Ci and the activity in the boiler during period 3 is:

$$A = e^{5.2-0.0235t} \text{ Ci} \qquad (3)$$

At t = 480, A is essentially zero.

To simplify the calculations of the estimated dose of a person who spends 8 hours working in a building humidified by the steam carrying tritium from the boilers, an average contamination in the steam during each of the three periods will be assumed based on a straight line relationship using the values at the limits of each period. The steam contamination at any time is A(avg) Ci/24500 kg. Weighting the contamination over the 8 hour period, the average contamination of the steam is 1.05×10^{-3} Ci/kg. With a steam in air ratio 1 to 60, the tritium in air contamination is calculated to be 2.27×10^{-5} Ci/m^3. The estimated committed dose from an 8 hour exposure using a dose conversion factor from N288.1 [2], a breathing rate of 16 L/min and a factor of 2 for skin absorption would be

$$\text{Dose} = \text{Intake} \times \text{dose conversion factor} \times 2$$

$$= 2.27 \times 10^{-5} \frac{\text{Ci}}{\text{m}^3} \times \frac{16\ \text{L}}{\text{min}} \times \frac{60\ \text{min}}{\text{h}} \times$$

$$8\ \text{h} \times \frac{1\ \text{m}^3}{1000\ \text{L}} \times 0.74 \times 10^5 \frac{\text{mrem}^*}{\text{Ci}} \times 2$$

$$= 25.8\ \text{mrem}$$

Chronic Dose Estimate

The on-line monitors will detect leakage rates of the order of 2 cc/min while the manual sampling can detect leakage rate reliability below 1.5 cc/min (0.5 μCi/L sample). As outlined in the monitoring section, tritium levels of the order of 0.1 μCi/L and less are common analysis results. If it is assumed that the worst case non-detectable chronic release rate of 1.5 cc/min exists, one can model the boiler inventory of tritium as for the accident case and make an estimate of the chronic dose due to steam humidification.

The simplified model for the boiler with the maximum chronic tritium leakage is shown in Figure 6.

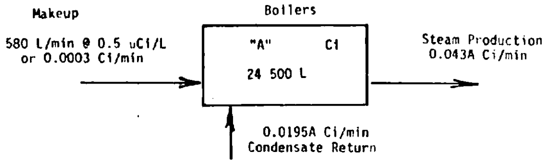


FIGURE 6: CHRONIC TRITIUM RELEASE BOILER MODEL.

The rate of change of the activity in the boiler is:

$$dA/dt = 0.0003 - 0.043A + 0.0195A\ \text{Ci/min}.$$

At t = 0, A = 0 and the activity in the boiler is

$$A = 0.01277 + 0.0129e^{-0.0235t}\ \text{Ci}.$$

When t is large, the tritium inventory in the boiler is:

$$A = 0.01277\ \text{Ci}$$

The concentration in the steam is 0.52 Ci/kg, which translates to tritium in air contamination of 1.12×10^{-8} Ci/m^3.

The estimated committed dose for an 8 hour exposure would be

$$\text{Dose} = \text{Intake} \times \text{dose conversion factor} \times 2$$

$$= 1.12 \times 10^{-8} \frac{\text{Ci}}{\text{m}^3} \times 16 \frac{\text{L}}{\text{min}} \times 60 \frac{\text{min}}{\text{h}} \times$$

$$8\ \text{h} \times \frac{1\ \text{m}^3}{1000\ \text{L}} \times 0.74 \times 10^5 \frac{\text{mrem}}{\text{Ci}} \times 2$$

$$= 0.013\ \text{mrem}$$

A 200 working day exposure would result in an estimated committed dose of 2.5 mrem.

Other Safety Concerns

The dose due to fission products in the heavy water has been analyzed, based on 1986 samples and is small relative to the tritium hazard [1]. The tritium hazard will reduce when the Tritium Extraction Plant is placed in service and tritium concentrations in NRU heavy water are reduced. The dose due to fission products could increase if the heavy water becomes particularly contaminated due to a fuel failure. If this should occur, then an administrative decision would have to be made to take the heat recovery system off-line.

The long term buildup of activity in the non-active inner area buildings was a concern for some people in regards to activity measurements or effects on experiments. To monitor this, a system of lithium fluoride dosimeters (i.e. TLD's) is being used to measure the relative buildup in a selected steam condensate tank. The TLD's will be checked annually against a TLD monitoring background levels.

Of a more serious concern, and one which is being evaluated during the construction stage, is evaluating the effects of building ventilation shutdowns caused by a system trip. To protect the water heating coils from freezing, the intake fans of each building connected to the system shut down when flow is lost to the coils. The fan will remain down until an increase in powerhouse steam production can make steam available to supply existing heating coils. This down time will likely be 30 minutes and may be up to 90 minutes depending on the steam loading, the fan start sequence and the rate of firing up of boiler at the powerhouse. This interruption in building ventilation supply air could have a detrimental effect on the normal movement of air in that building. This interruption could be particularly serious in buildings containing fume hoods where radioactivity or hazardous chemicals are handled. Face velocities could be reduced below acceptable working standards, or in fact result in reversal of fume hood exhaust.

To ensure that no dangerous situations are created by extended supply fan shutdown, simulated tests will be conducted. The tests will involve closing windows and doors (as would be the case during winter heating periods), shutting down the designated supply fans and measuring the face velocities of fume hoods and active cells to ensure positive flows within recommended standards.

Problems encountered with loss of building supply fans are corrected before space heating is placed in service.

CONCLUSION

The NRU Heat Recovery Project weir take-off is sufficiently high in the weir that the main heavy water heat exchangers cannot be siphoned and the weir function is not at risk. The take-off piping is robust and the seal design multilayered such that leakage is unlikely to occur. Adequate effluent piping isolation is possible should a leak occur. A monitoring system will detect any significant leakages of heavy water into the process water while routine samples will detect extra low level leaks. Multiple barriers provide adequate isolation of critical pathways where possible. Simple modelling of boiler/

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* 1 mrem = 10 Sv

steam system provides a means of estimating the dose by the most credible pathway. Based on dose estimates, the operation of the NRU Heat Recovery Project should not present an unacceptable risk to plant personnel while, at the same time, providing a large annual saving in CRNL energy costs.

REFERENCES

[1] Ballantyne, P.R., "A Description and Safety Review of the NRU Effluent Heat Recovery Project", Atomic Energy of Canada Report, AECL-MISC-300-S-2, 1985 (Draft).

[2] CSA Standard N288.1, "Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities, Sixth Draft, 1984 May.