UVAR SAFETY ANALYSIS REPORT

PROPOSED AMENDMENT

SAFETY ANALYSIS REPORT

FOR THE

LOW-ENRICHED URANIUM FUELED

UNIVERSITY OF VIRGINIA REACTOR

("LEU-SAR")

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This report is issued in support of a license amendment to License R-66 for the conversion of the University of Virginia's 2 MW reactor (UVAR) from high enriched uranium (HEU) to low enriched uranium (LEU) fuel. The report supersedes the original SAR for HEU cores and all its revisions and amendments as well as the so-called "UVAR Design and Analysis Handbook", which was the heretofore updated SAR. The present SAR may be referred to as the "LEU-SAR" to distinguish it from the previous SAR.

Table of Contents

Table of Contents (cont.)

Table of Contents (cont.)

List of Figures

List of Figures (cont.)

List of Tables

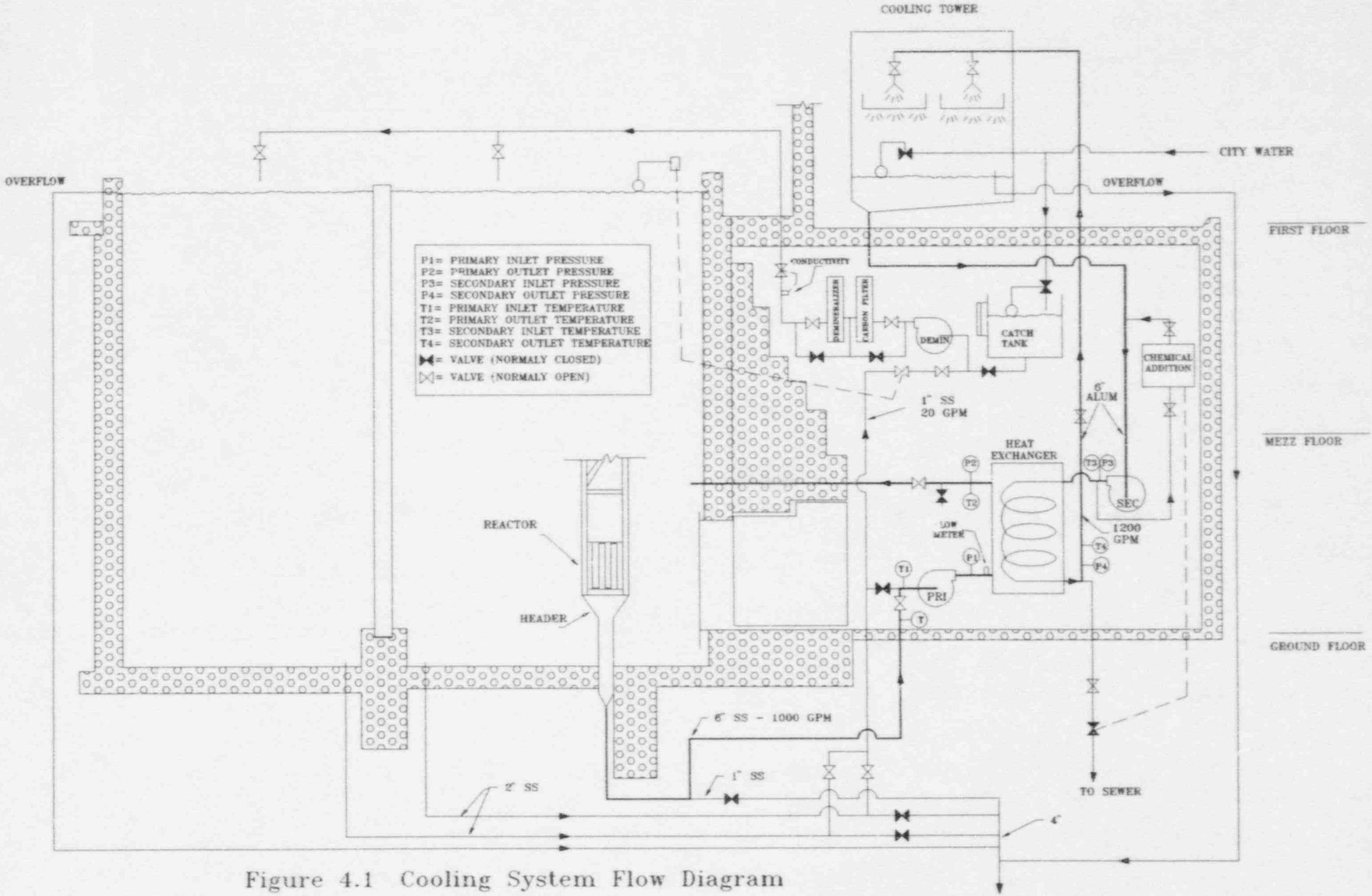


Figure 4.1 Cooling System Flow Diagram
(REVISED 10/95)

4.6. Design Specifications

The design specifications for the Heat Exchanger, Cooling Tower, Secondary Pump and Primary Pump are given in Tables 4.1 through 4.4.

4.6.1 Replacement of Heat Exchanger System Components

When a heat exchanger system component, e.g., primary pump, secondary pump, heat exchanger, secondary tube plug, cooling tower, associated piping and valves, is to be replaced, a Title 10 Code of Federal Regulations 50.59 analysis shall be performed to determine suitability of the replacement equipment. Because only a limited number of system specifications are critical, most components can be replaced with a fairly wide range of substitutes. Consideration must always be given to material compatibility. For example, wetted parts of pumps must be aluminum-compatible. Also, care must be taken to maintain the required primary coolant flow rate. Finally, replacement components should be able to withstand the system pressures, with reasonably large excess margins.

It is unlikely that exact replacements of system components several decades old can be located. In the cases of the heat exchanger and cooling tower, the replacement component is likely to be of a different style, design, or type from the original. Such differences are acceptable once a "10 CFR 50.59 analysis" finds that the replacement is capable of performing its intended function. With that qualification, heat exchanger system component changes do not pose "unreviewed safety questions". Changes of these components are permissible once a 10 CFR 50.59 analysis performed by the Reactor Staff is reviewed and approved by the Reactor Safety Committee.

4.7 Water Purification

The pool water purity is maintained by circulating it at a rate of 20 gallons per minute through a carbon filter and a mixed-bed ion exchange demineralizer. The water is normall maintained at a pH of 6.0 to 7.0 with a conductivity of less than 5 micromhos / cm.

4.8 Liquid Waste Disposal System

The Reactor Facility can collect radioactive liquid waste in two underground retention tanks of 5000 gallons each located outside of the Reactor Facility building, but within the site area. The waste is circulated and filtered, as well as given decay time before it is either discharged into the pond or discharged along with the pond as normal procedure. Other storage tanks within the Reactor Facility may also be used to temporarily store liquid waste. The option for sanitary sewer releases exists. All readioactive releases are made in conformance with applicable regulations. Two additional tanks of 250 gallons receive all waste from the Hot Cell. These tanks were installed as underground retention tanks in the

TABLE 4.1 Current Heat Exchanger Specifications

Heat Transfer Rate: 6.83 E6 BTU/h (2 MW)

Materials: Aluminum 6061 Alloy. All materials must be compatible with aluminum. For this reason, no copper-containing alloys can be used.

Maximum length: 18 feet.

Number of Secondary Tubes: 712

Secondary Tube Dimensions: 5/8" O.D. with 18 Ga. wall thickness

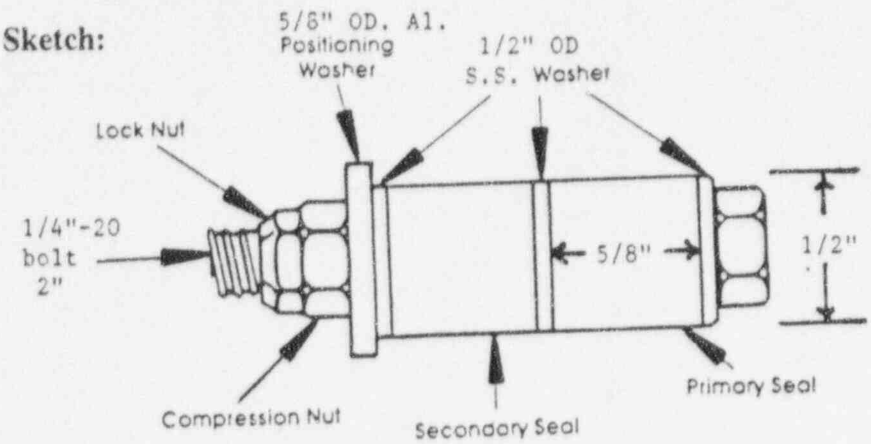
Fabricated in accordance with ASME Code, Section VIII, Division 1.

Inspected, certified, and stamped with the Code U-Symbol.

Hydraulic Specifications:	**Shell (Primary) Side**	**Tube (Secondary) Side**
Fluid circulated	High-purity water	Cooling Tower Water
Nominal Fluid flow rate	1100 gpm	1200 gpm
Nominal Inlet Temperature	110.2 Deg. F.	82.0 Deg. F
Nominal Outlet Temperature	95.0 Deg. F	93.4 Deg. F
Pressure Drop	pump-dependent	pump-dependent
Design Pressure	50 psi	50 psi
Test Pressure	75 psi	75 psi
Design Temperature	150 Deg. F.	150 Deg. F
Inlet and Outlet Pipe Dia.	8 inch	8 inch

TABLE 4.1.A Heat Exchanger Secondary Tube Plug Specification

Plug Sketch:



Screw, nuts, and interior washers may be stainless steel or aluminum.

Retaining washer must be aluminum. No dissimilar metal shall be in contact with the heat exchanger tubes.

Expandable/Compressible Tubing Material: Norprene or equivalent rubber.

Shaft (Screw): 1 pcs., 1/4"-20 by 2".

Nuts: 2 pcs., 1/4"-20, 1 locking, 1 normal.

Washer 1: 3 pcs., 1/2 O.D., stainless steel between seals.

Washer 2: 1 pcs., 5/8" O.D., aluminum only, plug positioning washer

Installation: against tube sheet support only.

Maximum Number to be Installed: To be determined, based upon allowable heat exchanger secondary-side working pressure, and secondary pump flow.

Testing and final installation torque: 12 inch-pounds.
Test Pressure: 150 psi (checked in a bench-rig).

Surveillance interval: annual, with removal and inspection of one plug from longest-installed group of plugs.

9.19 Heat Exchanger Secondary Tube Plugging Analysis

When heat exchanger secondary tubes are found to leak, they may be plugged with the type of replaceable expansion plug specified in TABLE 4.1.A. Tubes may be plugged up to a maximum number at which the heat exchanger secondary maximum working pressure is reached, as specified in TABLE 4.1. These pressures shall be monitored with heat exchanger inlet and outlet pressure gauges shown in Figure 4.1, following plug insertion. Secondary pump operating specifications, in TABLE 4.3, should also be consulted to set a reasonable lower limit on minimum steady-state flow through this pump.

Secondary tube plug installation does not affect primary coolant flow. This is because primary flow in the heat exchanger is through the shell side. The heat transfer rate is not of safety consideration. Reduction of cooling capacity resulting from tube plugging may affect the length of time the UVAR is operated on hot summer days until maximum pool water temperature (scram set-point at 105 Deg. F.) is reached.

9.19.1 Risk Associated with Plug failure

The risk associated with tube plug failure is equal to the product of the following: [the probability of failure of either of the two plugs in a tube] and [the number of plugged tubes] and [the consequences of a failure]. Qualitative evaluation of these quantities is discussed in the following two sections. No significant mode of interaction between failed plugs and intact plugs is recognized. As explained below, the probability of single or multiple plug failures is negligibly small.

9.19.2 Probability of Plug Failure

Failure of a plug will occur when the plug fails to maintain separation of primary and secondary water at its installed location. The type of plug used in the UVAR heat exchanger is of design and materials that are the same as or comparable to those used by the National Institute of Standards and Technology (NIST) to repair both their aluminum and stainless steel tube and shell heat exchangers (for the NBS Reactor at NIST). Failure of an installed plug is posited via a number of mechanisms:

1) degradation of rubber components because of N-16 gamma-ray interactions;

2) loosening of torqued nut-bolt connections because of flow-induced vibrations;

3) galvanic corrosion of SS plug components, in association with contacting materials;

4) removal of the plug from installed location by pressure gradient.

Any of the above mechanisms could result in the plug coming loose and possibly being ejected from its tube. It is shown in this section that the mechanisms potentially leading to plug failure are all very unlikely.

Degradation of Norprene rubber components, due to irradiation by N-16 gamma-rays which originate in the primary side of the heat water exchanger, occurs at a very slow rate. NIST has operated such plugs for a period of four or five years without failure by this mechanism. A service life of this length can be expected in the UVAR heat exchanger. To test the durability of Norprene, the UVAR staff subjected a sample of the material to a Co-60 gamma-ray dose of 0.15 Mrad, which was calculated to be equivalent to the dose that would be received by the material in an installed plug during 1 year of continuous operation of the UVAR at 2 MW. No deterioration of material flexibility or other properties was observed as a result. Therefore, the probability of plug failure during a surveillance interval of a year by this mechanism is essentially zero.

Loosening of torqued nut and bolt connections, due to vibration and stress-relaxation cannot be ruled out completely. However, NIST has never experienced a plug failure due to this mechanism. Precaution is taken against extreme loosening by use of SS threaded components, lock-nuts, and lock-washers. Regular yearly surveillance over tightness should be adequate to ensure that the probability of failure by this mechanism is essentially zero.

Galvanic corrosion, arising from direct contact of stainless steel plug components and the aluminum heat exchanger would occur at a negligible rate. This is because the corrosion potential between aluminum and stainless steel is small. However, plug design ensures that dissimilar metals will not come into contact with the tubes. Periodic surveillance is adequate to ensure that the probability of mechanical failure due to corrosion is negligible.

Removal of a plug from its installed location by a pressure gradient during reactor operations can be ruled out easily by considering the maximum service pressure differential from primary-to-secondary. This pressure gradient should be about 15 psi., whereas plugs are tested before installation at 150 psi. This ten-fold margin, coupled with the design-feature plug positioning washer, is ample to ensure that the probability of plugs being removed from their installed locations by normal water pressure is essentially zero.

9.19.3 Conclusion Regarding Risk of Plug Failure

The consequence of a tube plug failure is the slow recreation of a pre-existing leak, a possibility which is included in the safety analysis envelope in Section 9.20. Since it is shown above why there is essentially a zero probability that a plug will fail, it is concluded that the risk associated with plug failure is also approximately zero.

9.20 Heat Exchanger Primary-to-Secondary Leak Analysis

9.20.1 Introduction

The worst-credible primary-to-secondary heat exchanger tube leak rate that could develop before discovery is 1 ml/sec (about 1 gph) and would be caused by pitting corrosion. This type of leak starts small, grows slowly, and resultant secondary water activity is detectable before a 1 ml/sec leak rate is reached. The following scenario begins with conservative assumptions being made about reactor system configuration and operation. Progress of the scenario is followed to determine the significance, if any, of environmental releases of diluted pool water. The basis for this leak is presented first. Finally, it is shown clearly that the worst credible primary-to-secondary leak rate will not violate air or water effluent release limits.

9.20.2 Basis for Primary-to-Secondary Leak Rate

In August of 1995, a UVAR heat exchanger leak was observed by the reactor staff following cleaning of secondary tubes. The magnitude of that leak was determined using the activity of sodium-24 measured in primary and secondary water samples. Applying corrections to these data for the operating history of the reactor and secondary water blowdown (water draining from the secondary to the sewers) rate, a leak rate of 1 ml/sec was calculated. This leak rate resulted in sodium-24 activity in the secondary water equal to a factor of eight times the minimum detectable activity in cooling tower water. Thus, a smaller leak could be detected.

Primary water gamma-ray spectroscopy data provided a basis for calculating the equilibrium activity concentrations of isotopes normally expected in pool water as a consequence of operation. These concentrations are given in TABLE 9.20.1

TABLE 9.20.1 Calculated Primary Water Equilibrium Activity Concentrations

Nuclide Activity	Equilibrium Concentration [$\mu Ci/ml_{water}$]
H-3	4.0 E-4
Na-24	2.5 E-3
Mg-27	5.0 E-5
Cl-38	1.5 E-5
Mn-54	1.0 E-5
Cr-51	5.0 E-5
Sb-122	5.0 E-6
W-122	5.0 E-6

9.20.3 Conditions Prevailing at Start of Scenario

The UVAR is assumed to have been in operation at 2MW for a long time, compared to the radionuclide half-lives of interest. The primary and secondary pumps and cooling tower fan are operating normally. Pool water contains saturation levels of typical radionuclides.

9.20.4 Initiating Scenario Event

A 1 ml/sec primary-to-secondary leak is assumed to begin instantaneously in the heat exchanger through a pin-hole leak in a secondary tube. Assumption of sudden leak initiation is conservative. For simplicity and conservatism, it is also assumed that equilibrium cooling tower water activity concentrations are reached instantaneously upon initiation of the leak. Thus, steady-state releases of radioactive material begin from the cooling tower at a constant rate.

9.20.5 Primary-to-Secondary Leak Progression

It is assumed that the reactor staff does not notice the leak from observations of pool level, although a leak of this magnitude (about 24 gal/day) is detectable by that method. Therefore, the leak is assumed to continue until discovered when the next cooling tower water sample is analyzed and sodium or other nuclides are detected. A leak will be considered identified if the concentration of sodium-24 in cooling tower water exceeds 1 E-6 μCi/ml. Leak detection could occur as long as ten days after leak inception, since cooling tower water samples are analyzed "weekly".

9.20.6 Mitigation of Heat Exchanger Leak Consequences

Soon after a leak is identified, the cooling tower fan would be stopped. The UVAR would be shutdown, and primary isolation valves would be closed. Cooling tower water blowdown would be disabled. These actions would effectively stop further addition of radionuclides to the secondary water and further release of radioactive material to the air and sanitary sewer. Water remaining in the secondary system could then be disposed of appropriately.

9.20.7 Calculated Release Rates

Equilibrium activities for normally observed radionuclides (TABLE 9.20.1) were determined from primary water analysis and corresponding operating history. These activities, along with the observed leak rate from August 1995, were used as the source term for possible airborne, water, and sewer release activity concentration calculations and comparison to regulatory limits for each release mode.

Two bounding cases are considered:

> Case 1: All activity in the primary water leaking into the secondary is postulated to become airborne instantaneously in the cooling tower exhaust.
>
> Case 2: All activity in the primary water leaking into the secondary becomes concentrated in the secondary water through evaporation of some secondary water in the cooling tower. This water is then assumed to be released to the sanitary sewer (the normal blowdown path), or to the environment (see below).

Since the source term is fixed, the release from each of the two paths will have lower radionuclide concentrations than the releases calculated by assuming only one pathway at a time. An actual release will normally be a combination of air and sewer releases.

When the secondary pump stops, water in the upper basins flows down to the lower basin, which does not have sufficient capacity to contain about 1000 gallons of excess water. Thus, a direct release of cooling tower water to the roof of the building, and from there to the pond, is possible.

"Concentration Ratio", Cr [unitless], is defined as a measure of the degree of accumulation of a given element in cooling tower water due to cooling tower operation and blowdown. The ratio is calculated by dividing the concentration of an element in cooling tower water by the concentration of the element in make-up water. This is typically determined by using the concentrations of calcium, since this element is ubiquitous.

Equation 9.20.1. $Cr[unitless] = C_s / C_m$

where C_s = Element concentration in secondary water [atoms/ml],

C_m = Element concentration in make-up water [atoms/ml]

When the make-up rate M [gph] is known, then the blowdown rate B [gph] required to obtain a given concentration ratio can be calculated from the following relationship:

Equation 9.20.2 $Cr[unitless] = M / B$

where M = Secondary water make-up rate [gph],

and B = Secondary water blowdown rate [gph].

Typically, concentration ratios for stable elements are controlled to be in the range of from 6 to 7 by a blowdown system that automatically sends secondary water to the sanitary sewer when secondary water conductivity reaches a pre-set level. It is assumed that the concentration ratio that exists for stable elements is also applicable to radionuclides. To be conservative, the isotope concentration ratio is assumed to be equal to 10 in the Case 2 Calculations.

9.20.7.1 Calculation of August 1995 Heat Exchanger Leak Rate

The fractional primary-to-secondary leak rate in August 1995 is calculated from Equation 9.20.3, below, using the predominant radioisotope in primary water, sodium-24. First, secondary water blowdown rate is assumed to have been 300 gph. Given that the secondary system contains 2000 gallons, this blowdown rate value leads to a fractional blowdown rate, B_r, 4.2 E-5 [1/sec], as noted below.

Equation 9.20.3 $\mathbf{L = (\lambda + B_r)} * A_s^{\infty} / A_p^{\infty}$ [1/sec],

where:

L = Primary leak rate divided by secondary volume **[1/sec]**;

λ = Decay constant of sodium-24 **[1/sec]**;

$\mathbf{B_r}$ = Fractional blowdown rate from secondary to sewer per unit time **[1/sec]**;

Note: $\mathbf{B_r}$ **[1/sec] = B [gph] /** $\mathbf{V_s}$ **[gal] / (3600 sec / h)**,

where B is assumed = 300 [gph] (typical),

and V_s = 2000 gallons of secondary water volume.

$\mathbf{A_s^{\infty}}$ = Calculated equilibrium sodium-24 activity concentration in secondary water **[μCi/ml$_{water}$]** , based on August 1995 secondary water samples;

$\mathbf{A_p^{\infty}}$ = Calculated equilibrium sodium-24 activity concentration in primary water **[μCi/ml$_{water}$]**, based on August 1995 primary water samples.

Using appropriate values:

L = 1.28 E-5 [1/sec];

B_r = 4.2 E-5 [1/sec];

$\mathbf{A_p^{\infty}}$ = 2.5 E-3 [μCi/ml$_{water}$];

$\mathbf{A_s^{\infty}}$ = 6.1 E-6 [μCi/ml$_{water}$], the fractional leak rate, **L**, is found to be 1.34 E-7 [1/sec]. Finally, multiplying this rate by the secondary volume, 2000 gal, the volumetric leak rate is found to be 1 ml/sec (about 1 gph).

Once the fractional leak rate is known, Equation 9.20.3 can be arranged to solve for $\mathbf{A_s^{\infty}}$. Steady-state activity concentrations in cooling tower water for the other nuclides with source terms listed in Table 9.20.1 are found in this way. The resultant concentrations are listed in Column 2 (Water Activity [$\mu Ci/ml_{water}$]) of TABLE 9.20.3.

9.20.7.2 Airborne Release Calculation (Case 1)

Airborne release of radionuclides from secondary water in the cooling tower will be generated and diluted by the forced air flow from the cooling tower fan. The flow rate of this fan is 5 E+7 ml/s. For this analysis, it is assumed that all activity leaking from the primary to the secondary becomes instantly airborne. Table 9.20.2 gives the airborne concentrations based upon this scenario, assuming the TABLE 9.20.1 activity concentrations for activation radionuclides normally present in the primary water as the source term. A ratio equal to the derived airborne concentration divided by the applicable limit given in TABLE 9.20.2 is then calculated. The sum of these is calculated and found to be much less than 1, assuring that regulatory release limits will not be violated by the postulated air release.

TABLE 9.20.2 Air Activity Concentrations Compared with Regulatory Limits

Nuclide	Air Activity [$\mu Ci/ml_{air}$]	Airborne Release App.B T2 Col. 1 Limits [$\mu Ci/ml_{air}$]	(Air Act. Conc.) / (Air Conc. Limit) Ratio
H-3	8 E-12	1 E-7	8 E-5
Na-24	5 E-11	7 E-9	7 E-3
Mg-27	1 E-12	(1 E-7)	1 E-5
Cl-38	3 E-13	6 E-8	5 E-6
Mn-54	2 E-13	1 E-9	2 E-4
Cr-51	1 E-12	3 E-8	3 E-5
Sb-122	1 E-13	3 E-9	3 E-5
W-187	1 E-12	1 E-8	1 E-4
Sum of Ratios			**0.008**

9.20.7.3 Water-Borne Release Calculations (Case 2)

Radionuclides in the cooling tower basin will be concentrated as water is evaporated. Normally, the concentration ratio for elements is controlled by blowdown to be between 6 and 7. To be conservative, a concentration ratio of 10 is chosen for these calculations. Radioactive decay of the shorter-lived radionuclides limits the maximum activity that can be obtained, independent of the concentration ratio in cooling tower water. The formula used to determine secondary water activity concentrations is Equation 9.20.3, rearranged to solve for those. As explained above, with a concentration ratio of 10 and a normal average make-up of 800 gph, B_r becomes 1.1 E-5 [1/sec].

TABLE 9.20.3 Water Activity Conc's Compared with Regulatory Limits

Nuclide	Water Activity [μCi/ml$_{water}$]	Effluent Limit App. B.T2 Col. 2. [μCi/ml$_{water}$]	Ratio	Sewer Limit. App B. T3 [μCi/ml$_{water}$]	Ratio
H-3	5 E-6	1 E-3	0.005	1 E-2	5.0 E-4
Na-24	1.5 E-5	5 E-5	**0.300**	5 E-4	**3.0 E-2**
Mg-27	5 E-9	--	--	--	--
Cl-38	6 E-9	--	--	--	--
Mn-54	1 E-7	3 E-5	0.003	3 E-4	3.3 E-4
Cr-51	6 E-7	5 E-4	0.001	5 E-3	1.2 E-4
Sb-122	5 E-8	1 E-5	0.005	1 E-4	5.0 E-4
W-187	4 E-7	3 E-5	0.013	3 E-4	1.3 E-3
Sum of Ratios			**0.327**		**0.033**

Because sodium-24 is the dominant and most limiting radionuclide in the primary water, an additional calculation was done by assuming no loss of sodium-24 by blowdown to show that the concentration ratio (effectively infinite in this case) is not a critical parameter in these calculations. This results in an equilibrium sodium-24 activity concentration of 2.6 E-5 μCi/ml$_{water}$ and only increases the sums of ratios in TABLE 9.20.3 to 0.77 for water effluent and to 0.053 for sewerage. The result remains that the postulated water releases will not violate regulatory release limits.

9.20.7.4 Release Analyses Conclusions

The primary-to-secondary leak was readily identified in August 1995 by means of radio-analysis of the secondary water at an activity level of about 1 E-6 μCi/ml. Any future leak should be easily observed before the leak grows to the size of the observed leak of August 1995. Sodium-24 is the most observable and most limiting radionuclide in the secondary water and can be detected through normal sampling of the secondary water before any release limits are reached. Radio-nuclides not on the list of those analyzed (TABLES 9.20.1,2 & 3) will be present at such low levels compared to the sodium-24 that they do not need to be individually considered.

9.21 Heat Exchanger Secondary-to-Primary Leak Analysis

At times when the primary pump is off, secondary water may pass through a leaking heat exchanger tube into the primary water system. This is possible because of the elevation difference between the water surface in the cooling tower basin and the reactor pool surface. If the secondary pump is running, an enhanced secondary-to-primary leak rate occurs due to increased secondary-side pressure.

Abundant industry operating experience with leaks caused by corrosion of tube and shell heat exchangers shows that they start out at a small flow rate and then increase gradually with time. Therefore, secondary water leaking into the primary water will degrade pool water quality gradually. Overall water quality with the leak rate basis assumed in Section 9.20 (i.e., 1 ml/sec) can be kept easily within technical specification limits. There is a demineralizer system in continuous operation which mitigates the effect of secondary-to-primary leakage. It is concluded that pool water quality is monitored on a frequency (at a minimum, weekly) that is sufficient to identify leaks well before they pose water quality concerns.

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