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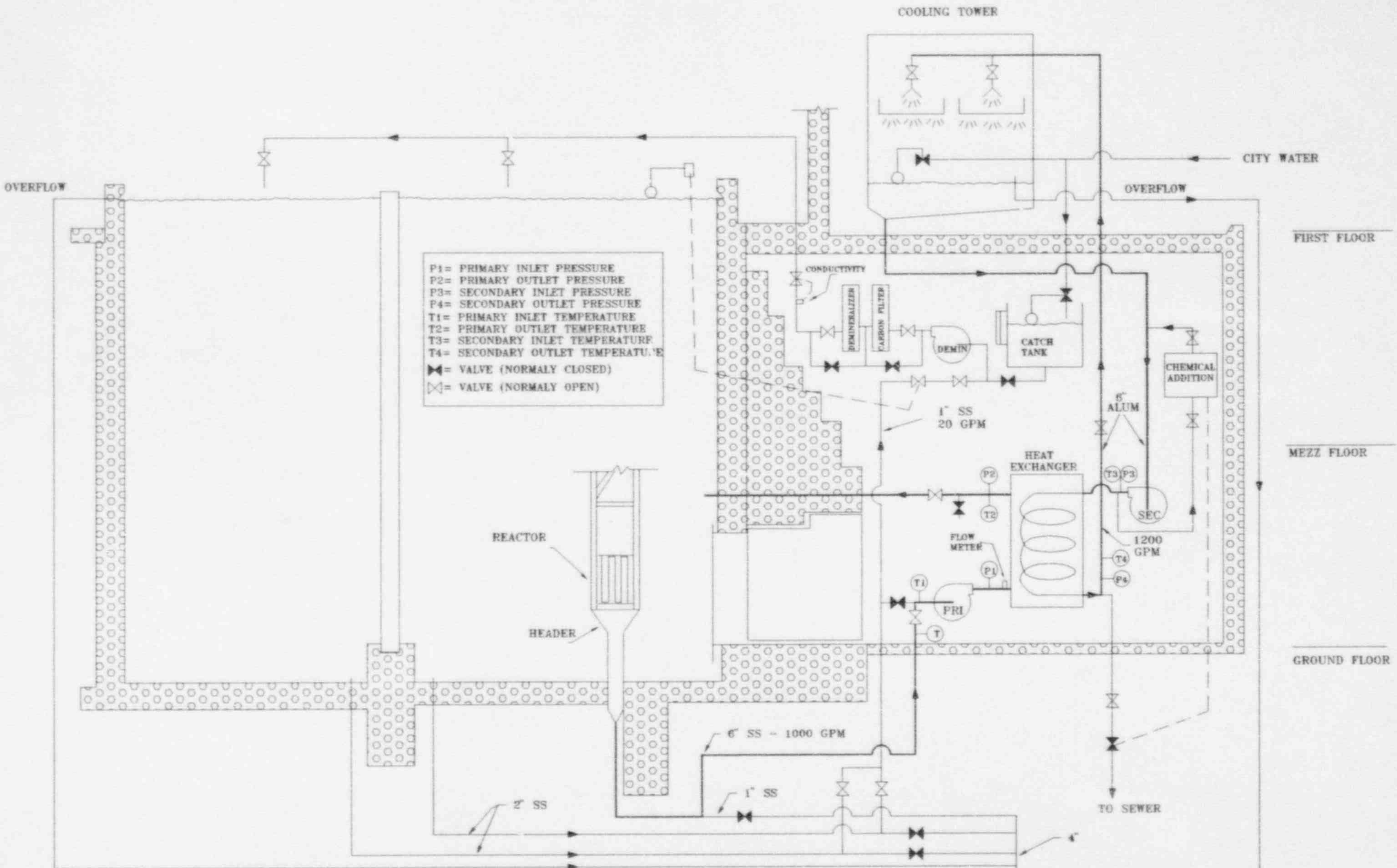


Figure 4.1 Cooling System Flow Diagram
(REVISED 10/95)

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. The data in these tables are updated as components are repaired, modified or replaced.

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 repaired, modified, or 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 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 and meets applicable 50.59 criteria. With that qualification, heat exchanger system component changes do not pose 10CFR50.59 "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 normally 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 radioactive 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 Heat Exchanger Design 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

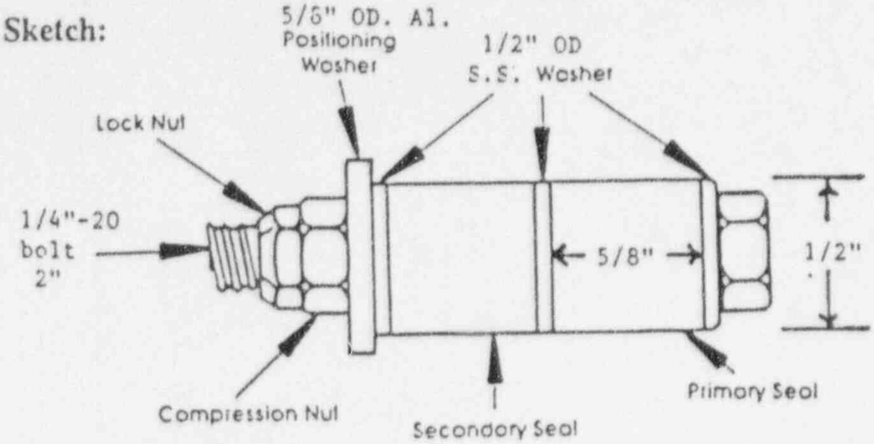
Fabricated in accordance with ASME Code, Section VIII, Division 1.

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 Design Specifications

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, for in-situ visual inspection and installation torque check of all installed 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 the subsections of 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	Equil. Conc. [$\mu Ci/ml_{water}$]	Rad. Decay Const. (λ) [1 / sec]
H-3	4.0 E-4	1.79 E-9
Na-24	2.5 E-3	1.28 E-5
Mg-27	5.0 E-5	1.22 E-3
Cl-38	1.5 E-5	3.10 E-4
Mn-54	1.0 E-5	2.57 E-8
Cr-51	5.0 E-5	2.89 E-7
Sb-122	5.0 E-6	1.22 E-3
W-187	5.0 E-6	8.05 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 end 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],

and 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 A_s^{∞}. 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 (Secondary Water Activity [$\mu Ci/ml_{water}$]) of TABLE 9.20.3. Temporary ratios presented for nuclides in this table are valid for the release duration and are equal to their secondary water activity concentration divided by the water and sewer release limits, respectively.

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 temporary ratios 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 [μCi/ml$_{air}$]	Yearly-Average Airborne Release App.B Table 2 Col. 1 Limits [μCi/ml$_{air}$]	Temporary Ratio [(Air Act. Conc.) / (Air Conc. Limit)]
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-9	1 E-3
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 Temporary 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 concentrations. 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 <u>Water Activity Conc's Compared with Regulatory Limits</u>

Nuclide	Secondary Water Activity [µCi/ml$_{water}$]	Yearly-Average Water Release App. B Table 2 Col. 2 Limits [µCi/ml$_{water}$]	Ratio	Yearly-Average Sewer Release App. B Table 3 Limits [µCi/ml$_{water}$]	Ratio
H-3	5 E-6	1 E-3	0.005	1 E-2	0.0005
Na-24	1.5 E-5	5 E-5	**0.300**	5 E-4	**0.03**
Mg-27	5 E-9	--	--	--	--
Cl-38	6 E-9	--	--	--	--
Mn-54	1 E-7	3 E-5	0.003	3 E-4	0.00033
Cr-51	6 E-7	5 E-4	0.001	5 E-3	0.00012
Sb-122	5 E-8	1 E-5	0.005	1 E-4	0.0005
W-187	4 E-7	3 E-5	0.013	3 E-4	0.0013
Sum of Temporary 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. With no loss of sodium by blowdown, the equilibrium sodium concentration would increase from 1.5 E-5 [µCi/ml$_{water}$] to 2.6 E-5 [µCi/ml$_{water}$]. The temporary ratio for sodium effluent would increase from 0.300 to 0.530, making the corresponding sum of temporary effluent ratios increase from 0.327 to 0.557. The temporary ratio for sodium sewerage would increase from 0.03 to 0.052, making the corresponding sum of temporary sewerage ratios increase from 0.033 to 0.055. 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 observed easily 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 regular sampling of secondary water before release limits are reached. Radionuclides 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.

9.20.8 Double-Ended-Heat-Exchanger-Tube-Break Leak Analysis

9.20.8.1 Introduction

This primary-to-secondary leak analysis is predicated upon the occurrence of a double-ended-guillotine-tube-break (DEGTB). Such a leak-initiating event is an applicable, conservative assumption for power plant safety analyses because high flow rates, pressures, temperatures and possibly flow-induced vibration occur in power-plant steam generators. In contrast, operating conditions in the UVAR heat exchanger are quite gentle. In Sections 9.20 to 9.20.7.4, the leak analysis is based upon credible assumptions, conditions, and industry experience (regarding leaks in heat exchangers at low temperatures, pressures, and flow-rates). The analysis in this section, predicated upon a totally hypothetical model, is used to fathom the maximum impact from heat exchanger leaks on public health and safety.

In the UVAR heat exchanger, primary water flows through the shell. If a large amount of primary water were to enter the UVAR secondary system, potential cooling tower water overflow would be directed to a 750,000-gallon on-site effluent-hold-up pond. The pond provides dilution and decay time for radionuclides, making unplanned release to unrestricted areas in excess of regulatory limits extremely unlikely. It is emphasized that a primary-to-secondary UVAR leak would only be possible when the primary pump is on. Since pump status is administratively controlled, the primary pump only would be on when staff is on-site.

9.20.8.2 Conditions Prevailing at Start of Scenario

The most challenging conditions would exist with staff on-site, and the UVAR just shutdown following an operating time sufficiently long for equilibrium radionuclide activity concentrations to exist in pool water. These concentrations, based upon August 1995 reactor operations data, and corresponding radioactive decay constants, are given in TABLE 9.20.1. With the UVAR shutdown, the primary pump, secondary pump, and cooling tower fan could be left, and for the analysis are left, operating to cool down the pool. Primary pool level is assumed at 19'6" above the core, at its highest level short of overflowing. With the UVAR shutdown, the low-pool-level alarm is actuated by pool level dropping to 19'2-1/2", which is lower than the 19' 3-1/2" actuation level of the other low-pool-level alarm (and scram) enabled during reactor operation.

9.20.8.3 Initiating Scenario Event

It is posited that a DEGTB occurs in the heat exchanger. The leak rate must be estimated without having some necessary data in hand, for example, the driving pressure across the break, flow friction factors and the exact primary flow path around baffles. It is sufficient to make conservative assumptions. In analogy with the North Carolina State University (NCSU) DEGTB analysis, it is assumed that the primary flow is through the heat exchanger tubes, rather than the shell. Accordingly, the leak rate is calculated at 5.8 gpm and conservatively rounded up to 10 gpm (600 gph). This leak rate is 600 times larger than the corrosion-initiated leak rate postulated above. It is shown below that, with a large leak, the actual leak rate value is not a critical parameter. Rather, the bounding parameter is the total primary water radionuclide activity released, which can be fairly accurately predicted.

9.20.8.4 Primary-to-Secondary Leak Progression and Mitigation

Since a reactor staff member is required to be present at the facility when the primary pump is operating, the 600 gph leak could continue until primary pool level drops from 19'-6" to the 19'2-1/2" level, at which point the low-pool-level alarm would be activated. There are 200 gallons of primary water per inch of pool water level. Therefore, it is assumed the water level lowers by 3-1/2" (from 19'6" to 19'2-1/2"), and this results in a transfer of 700 gallons of primary water to the secondary side. It is very likely, however, that the actual amount of primary water leaked would be less than 700 gallons because reactor pool level does not routinely approach 19'6".

Discovery of the assumed 10 gpm leak will occur after no more than about 1.2 hours. At that time the low-pool-level alarm would sound in the reactor control room and the facility hallway. Reactor operator(s) would respond by stopping the primary pump and cooling tower fan. These actions stop the primary-to-secondary leak and cooling tower exhaust release immediately. The primary system may also be isolated by closing the pool isolation valves. Contaminated secondary water in the cooling tower would then be disposed of in an appropriate manner.

9.20.8.5 Scenario Consequences

Radionuclide concentrations in cooling tower exhaust (air) and secondary system water are calculated in the manner presented in the subsections of Section 9.20.7. Again, sodium-24 data are used in illustrative calculations. Results pertaining to this and the other radionuclides listed in TABLE 9.20.1 are obtained in the same two ways and have been listed in TABLES 9.20.4 & 5. Equilibrium primary water radionuclide concentrations and decay constants for the calculations may be obtained from TABLE 9.20.1. The same two bounding cases discussed in Section 9.20.7 are applicable, namely:

> 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).

9.20.8.6 Airborne Release Calculation (Case 1)

The rate of sodium activity in-flow to secondary water is

(10 gpm) * (3800 ml/gal) * (1 min / 60 sec) * (2.5 E-3 $\mu Ci/ml_{water}$)

= 1.6 [μCi/s].

This activity in-flow is diluted by cooling tower air-flow at a rate of 5 E+7 ml/s, resulting in a sodium activity concentration in air of (1.6 μCi/s) / (5 E+7 ml/s)

= 3.2 E-8 [$\mu Ci/ml_{air}$].

For all other equilibrium radionuclides listed, it follows that the appropriate multiplicative factor for conversion of primary water concentration to secondary air concentration is (10 gpm) * (3800 ml/gal) * (1 min / 60 sec) / (5 E+7 ml/s)

= 1.3 E-5 [$\mu Ci/ml_{air}$ / $\mu Ci/ml_{water}$].

TABLE 9.20.4 **Air Activity Concentrations Compared with Regulatory Limits**

Nuclide	Air Activity [µCi/ml$_{air}$]	Yearly Average Airborne Release App.B Table 2 Col. 1 Limits [µCi/ml$_{air}$]	Temporary Ratio [(Air Act. Conc.) / (Air Conc. Limit)]
H-3	5.1 E-9	1 E-7	0.051
Na-24	3.2 E-8	7 E-9	**4.57**
Mg-27	6.3 E-10	1 E -9	0.600
Cl-38	1.9 E-10	6 E-8	0.003
Mn-54	1.3 E-10	1 E-9	0.130
Cr-51	6.3 E-10	3 E-8	0.021
Sb-122	6.3 E-11	3 E-9	0.021
W-187	6.3 E-11	1 E-8	0.006
Sum of Temporary Ratios			**5.4**

The sum of ratios in TABLE 9.20.4 indicates that radionuclide concentrations in cooling tower exhaust <u>during the postulated 1.2-hour atmospheric release</u> are <u>temporarily</u> above the regulatory limit of 1, which is linked with <u>a continuous year-long release</u>. This sum of temporary ratios is a valid at the cooling tower, which is about 25 meters from the site boundary. For the above calculation, no credit was taken for dilution of air release from the cooling tower to the site boundary. Dilution of the air release into the atmosphere begins immediately, with the cooling tower exhaust expelled forcefully upward. Diffusion of radionuclide content occurs in all directions as the plume travels to the site boundary. Intuitively, even during the release time, there will be a dilution factor of at least 10 to bring the sum of ratios to a number less than 1 at the site boundary, about 25 meters away from the cooling tower. If this dilution is ignored, it would take only about 6 additional hours of cooling tower exhaust with no measurable radionuclide content to ensure that the calendar-year-averaged sum of ratios is reduced to less than 1. Hence, the postulated Case 1 analysis concludes that the regulatory limit pertaining to air releases will not be exceeded.

9.20.8.7 Water-Borne Release Calculations (Case 2)

For a DEGTB, radionuclide concentrations in secondary water are calculated using Equation 9.20.5 to evaluate $\mathbf{A_s^\infty}$. This equation is valid in conjunction with the conservative assumption that steady-state secondary water radionuclide concentrations are reached instantaneously.

Equation 9.20.5 $$\mathbf{A_s^\infty = L * A_p^\infty / (\lambda + B_r)\ [\mu Ci / ml_{water}]},$$

where, following the definitions of terms given in Section 9.20.7.1, conservative variable values are:

$\mathbf{L}$ = (10 gpm) * (1 min / 60 sec) / (2000 gal) = [8.3 E-5 /sec];

$\mathbf{A_p^\infty}$ = 2.5 E-3 [μCi/ml_{water}] (TABLE 9.20.1);

λ = Decay constant of sodium-24 = 1.28 E-5 [1/sec] (TABLE 9.20.1);

$\mathbf{B_r}$ = 4.2 E-5 [1/sec] (typical value, again assuming a concentration factor of 10).

Substitution of the above values results in secondary water sodium-24 activity concentration of $\mathbf{A_s^\infty = 3.8\ E\text{-}3\ [\mu Ci/ml_{water}]}$. Note that this activity concentration is greater than $\mathbf{A_p^\infty}$ because of concentration of sodium in the cooling secondary system. There is no universal multiplicative factor for conversion of $\mathbf{A_p^\infty}$ to $\mathbf{A_s^\infty}$ for the other listed radionuclides because the decay constant for each isotope must be incorporated from TABLE 9.20.1 in the calculation of $\mathbf{A_s^\infty}$. The calculation is straight-forward, however, and the resulting A_s^{∞} for the other listed radionuclides are presented here in TABLE 9.20.5. Results of sum-of-ratios calculations for the water- and sewer-release sub-cases are also shown in TABLE 9.20.5.

TABLE 9.20.5 Secondary Water Activity Conc's Compared with Regulatory Limits

Nuclide	Secondary Water Activity [$\mu Ci/ml_{water}$]	Yearly-Average Effluent Release App. B Table 2 Col. 2 Limits [$\mu Ci/ml_{water}$]	Ratio	Yearly-Average Sewer Release App. B Table 3 Limits [$\mu Ci/ml_{water}$]	Ratio
H-3	2.0 E-4	1 E-3	0.20	1 E-2	0.02
Na-24	3.8 E-3	5 E-5	**76**	5 E-4	**7.6**
Mg-27	3.3 E-6	--	--	--	--
Cl-38	3.6 E-6	--	--	--	--
Mn-54	2.0 E-5	3 E-5	0.67	3 E-4	0.07
Cr-51	1.0 E-4	5 E-4	0.20	5 E-3	0.02
Sb-122	3.3 E-7	1 E-5	0.03	1 E-4	0.003
W-187	8.3 E-6	3 E-5	0.28	3 E-4	0.028
Sum of Temporary Ratios			**77**		**7.7**

The sum of temporary ratios, in the case of the water-only effluent release pathway is significantly larger than 1. However, there is no water effluent released to the environment because it is mostly retained in the secondary coolant [Note: Secondary make-up shuts down automatically due to the fast leak rate which tends to improve (lower) the conductivity of water in the cooling tower.].

At worst, some cooling tower water could overflow to the on-site reactor pond, also used for liquid waste dilution. The pond dilution factor is about 750,000 gallons / 2000 gallons, or 375-to-1. Only a fraction of the entire 2000 gallons of secondary water could overflow to the pond. Natural radionuclide concentration in the pond is negligible.

With pond-water dilution, the sum of temporary ratios for water effluent is reduced to 77 / 375 = 0.21, well below the allowed calendar-year-average limit of 1. It is concluded that with a DEGTB and a water-only effluent pathway, the calendar-year-average water effluent limit would not be exceeded. Pond water is only released periodically, subsequent to radioanalysis. In the case of this event it would most likely be possible to delay a pond release long enough to achieve significant decay of the 15-hour sodium-24.

The sum of temporary ratios for the sanitary sewer release is marginally greater than 1 during a short period of time (about 1 hour). A great volume of cooling tower water with no measurable radionuclide content is released to the sewer by blowdown during any calendar year. This will ensure that the calendar-year-average sum of ratios will be less than 1. Thus, it is concluded for this case that sewer release regulations will not be exceeded.

9.20.8.8 <u>Conclusions</u>

The primary-to-secondary heat exchanger leak scenario involving the DEGTB analyzed above is extremely conservative. The posited 600 gph leak rate and staff discovery time of 1.2 hours or less represents a hypothetical situation postulated for examination of the potential maximum consequences to the public. Calculated radionuclide concentrations for limiting air-only and water-only release cases are sufficiently low and dilution sufficiently high that calendar-year-average release limits are not challenged. Therefore, it is concluded that even an extremely severe (hypothetical) heat exchanger tube rupture poses no significant threat to public health and safety.

The analysis just performed reveals that the leak rate, per se, is not a critical parameter. While radionuclide concentrations in a release depend strongly upon the primary-to-secondary leak rate, this is a temporary situation. On a calendar-year-averaged basis, it is the total activity of each isotope released, i.e., the volume of primary water lost to the environment, that is significant.

9.21 Heat Exchanger Secondary-to-Primary Leak Analysis

During 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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