TABLE 3.6-1 (Continued)

SECONDARY CONTAINMENT BYPASS LEAKAGE PATHS

PENETRATION		RELEASE LOCATION
X-84A	Pressurizer Relief Tank Gas Sample	Auxiliary Area
X-85A	Excess Letdown Heat Exchanger	Auxiliary Area
X-90	Control Air	Auxiliary Area
X-93	Accumulator Sample	Auxiliary Area
X-94ABC	Radiation Sample	Auxiliary Area
X-95ABC	Radiation Sample	Auxiliary Area
X-96C	Hot Leg Sample	Auxiliary Area
X-98	ILRT	Auxiliary Area
~~X-107~~	~~RHR~~	~~Auxiliary Area~~
X-110	UHI	Auxiliary Area
X-114	Ice Condenser	Auxiliary Area
X-115	Ice Condenser	Auxiliary Area
X-40D	Hydrogen Purge	Auxiliary Area

TABLE 3.6-1 (Continued)

SECONDARY CONTAINMENT BYPASS LEAKAGE PATHS

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X-95ABC	Radiation Sample	Auxiliary Area
X-96C	Hot Leg Sample	Auxiliary Area
X-98	ILRT	Auxiliary Area
~~X-107~~	~~RHR~~	~~Auxiliary Area~~
X-110	UHI	Auxiliary Area
X-114	Ice Condenser	Auxiliary Area
X-115	Ice Condenser	Auxiliary Area
X-40D	Hydrogen Purge	Auxiliary Area

ENCLOSURE 2
SEQUOYAH NUCLEAR PLANT

Deletion of Appendix J Requirements on RHR Supply Line Isolation Valve

At Sequoyah Nuclear Plant, the residual heat removal (RHR) supply line transports primary coolant from the No. 4 hotleg to the RHR pumps and penetrates primary containment at penetration X-107. Two inline valves are located immediately inboard of the penetration. These valves, 74-1 and 74-2, act as pressure isolation boundaries between the primary system and the RHR system. Additionally, the valve closest to containment, 74-2, acts as the inner containment isolation barrier for this penetration.

This valve should not have been placed under the requirements of Appendix J because it is not a potential leakage path during an accident situation. The requirements for valves to be type C tested in Appendix J do not apply to this valve.

Both of these valves are leakrate tested with water at least once every nine months when coming back from cold shutdown as required in 10 CFR 50.50A and the ASME Boiler and Pressure Vessel Code Section XI. In this test, the leakrate is determined by measuring the primary water leakage past each valve when the primary system is at an elevated temperature and pressure.

We believe that this valve was mistakenly added to Table 3.6-1 before issuance of the technical specifications. Upon preparation of procedures for the Appendix J test of 74-2, numerous factors were noted which make this particular test both infeasible and unneccessary. The reasons for the infeasibility of the air test are as follows:

1. During refueling, cold shutdown, and hot shutdown, reactor operation modes 6, 5, and 4 respectively, the RHR supply line is required for reactor cooling, thus airtesting 74-2 is not possible in these modes.

2. During normal operation (reactor operation modes 1 and 2) airtesting is not possible due to the high radiation dose and temperatures to which test personnel would be exposed.

3. At modes other than those listed above, the temperatures and pressures in the primary system are at elevated levels. At the temperature and pressure conditions in these modes, primary fluid leakage past the acting block valve (74-1) into the test volume would flash to steam. This would make draining the test volume, hooking up, and unhooking the test equipment dangerous to the test personnel. The larger the leakage past the block valve, the more of a problem this would present.

Airtesting the containment isolation valve in the RHR supply line is also unnecessary when the post-LOCA line use is considered.

In the event of a nonisolable LOCA, two RHR use modes exist:

1. RHR miniflow

2. RHR injection

In the RHR miniflow mode, the RHR pumps are operating in closed loop recirculation and no fluid is being injected into containment. Any leakage past the isolation valve flows into the RHR system and is recirculated in the miniflow. Leakage from the RHR system to the auxiliary building has previously been analyzed and accounted for in offsite dose analyses.

During the RHR injection mode, any leakge past valve 74-2 would be pumped back into containment. As in the previous mode, subsequent leakage out of the RHR is of no consequence.

For accidents which can result in an isolable LOCA, the RHR may be used by recirculating water through the RHR supply line and back into the reactor vessel. Valve 74-2 is open in this case and valve leakage is of no concern.

Therefore, since: (1) the requirements of Appendix J do not apply to this valve, (2) valve 74-2 is required to be within certain leakage limits as determined by periodic water tests, (3) moderate leakage through this valve will not affect the analyzed dosage due to leakage from the RHR system, (4) an additional airtest involves risk to test personnel, and (5) valve 74-2 has a backup valve (74-1) which is also monitored for leaktightness, it is requested that valve 74-2 be deleted from table 3.6-2.