May 27, 2004 10 CFR 50.59

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Gentlemen:

In the Matter of	)	Docket Nos. 50-327
Tennessee Valley Authority	)	50-328

SEQUOYAH NUCLEAR PLANT (SQN) - UNITS 1 AND 2 - 10 CFR 50.59, CHANGES, TESTS, AND EXPERIMENTS SUMMARY REPORT

The purpose of this letter is to provide the summary report of the implemented safety evaluations performed in accordance with 10 CFR 50.59(d)(2). The enclosed report covers the period from May 21, 2002 to January 31, 2004.

If you should have any questions, please contact me at (423) 843-7170 or J. D. Smith at (423) 843-6672.

Sincerely,

Original signed by:

Pedro Salas
Licensing and Industry Affairs Manager

Enclosure

ENCLOSURE

SEQUOYAH NUCLEAR PLANT

10 CFR 50.59 SUMMARY REPORT

SEQUOYAH NUCLEAR PLANT

CHANGES IN THE FACILITY FOR AMENDMENT - 18

DCN	DESCRIPTION	SAFETY ANALYSIS
D-20673-A	This change involves modification of the shield building concrete dome, the steel containment vessel (SCV), and the steam generator compartment concrete roofs to support removal of the original steam generators (OSGs) and installation of the replacement steam generators (RSGs) for Unit 1. To remove the OSGs and install the RSGs, openings were cut in the concrete shield building dome, the SCV, and the steam generator compartment roofs. The two openings in the shield building concrete were restored by splicing new reinforcing bar to the existing reinforcing bar using mechanical couplers and pouring new concrete to close the openings. The two openings in the SCV were restored by welding the cut steel pieces back in place. The four openings in the steam generator compartment roofs were restored by reinstalling the cut sections of the roof in their respective holes and using through-bolted connection frames to hold the concrete sections in place.	The modifications do not increase the frequency or likelihood of accidents or malfunctions, increase the consequences of an accident or malfunction or create a new type of accident or malfunction. The SCV and shield building are fission product barriers which have been restored to meet their design bases such that no design limits were altered or exceeded. The use of reinforcing bar mechanical couplers to restore the shield building is different from the reinforcing bar splicing methods previously approved and was subject to NRC approval prior to entering Mode 4. NRC approval for use of the mechanical coupler reinforcing bar splices was provided via a letter dated March 13, 2003. Amendment 283 to the Unit 1 Operating License approving the methodology submitted in Topical Report 24370-TR-C-001 was issued by the NRC on April 24, 2003. The use of through-bolted connection frames to restore the roof of the steam generator compartments was shown to be analytically acceptable. However, the methodology used deviates from that described in the UFSAR and therefore required NRC approval prior to entering Mode 4. NRC approval of the methodology used to analyze the through-bolted connection frames was provided via a letter dated April 18, 2003. Amendment 284 to the Unit 1 Operating License approving the methodology submitted in Topical Report 24370-TR-C-003 was issued by the NRC on April 25, 2003.

SEQUOYAH NUCLEAR PLANT

CHANGES IN THE FACILITY FOR AMENDMENT - 18

DCN	DESCRIPTION	SAFETY ANALYSIS
E-20841-A	This change authorizes an increase in the maximum decay heat load which can be applied to the spent fuel pool (SFP). The increase in heat load was off-set by imposing lower component cooling water system (CCS) water temperature and lower SFP heat exchanger fouling factor requirements. Additionally, this change authorizes operation of the CCS heat exchangers at higher fouling rates provided that reduced limits on Essential Raw Cooling Water (ERCW) temperature and increased limits on ERCW flow are met to off-set the increased fouling.	The existing SFP cooling analysis of record uses design basis values and regulatory specified fuel discharge scenarios for predicting maximum SFP temperatures. The CCS and ERCW systems provide cooling for the SFP such that any increase in SFP decay heat will be rejected to the CCS and ERCW. The CCS and ERCW system cooling analyses were revised as part of this change. The results of these analyses indicate that the increased SFP decay heat load remained within the design basis heat removal capability of the systems. Based on this result, the design basis maximum temperature for the SFP remains unchanged and is bounding for the increased decay heat loads. The design basis accidents and events associated with the proposed change include the primary system loss-of-coolant accident (LOCA) and the design basis (external) flooding event (due to the potential impact on critical decay heat removal systems). Credible failure modes associated with the proposed change include loss-of-offsite power (LOOP) and loss of single cooling train (which is the design basis for all safety related cooling systems). The change does not add any new components, nor does it alter the operation or function of existing components or systems. As such, the change will not increase the frequency or likelihood of accidents or malfunctions, increase the consequences of an accident or malfunction or create a new type of accident or malfunction. A license amendment request for SFP decay heat analysis methodology changes for Watts Bar Nuclear Plant was previously submitted to NRC for review and was approved by an NRC letter dated February 21, 2002 (Amendment 37 to the Watts Bar Operating License). The changes to the Sequoyah analysis methodology are identical to those approved for Watts Bar.

SEQUOYAH NUCLEAR PLANT

CHANGES IN THE FACILITY FOR AMENDMENT - 18

DCN	DESCRIPTION	SAFETY ANALYSIS
D-20887-A	This change adds the Original Steam Generator Storage Facility (OSGSF) to the site and establishes the Original Steam Generator (OSGs) configuration required for storage. The OSGSF is a non-safety related, non-seismic, reinforced concrete structure that provides an interim storage area for the OSGs and any attached components removed from the Unit 1 reactor building during the Unit 1 steam generator replacement. The OSGSF is located outside the protected area, but within the exclusion area and site boundary. It is a stand-alone facility and does not interface with any plant structures or systems during any mode of plant operation. Further, design analyses have determined that the impact of natural phenomena events on the OSGSF will not adversely impact plant operations.	The approach used in assessing the effectiveness of the OSGSF shielding design is an alternative method to that described in the UFSAR and was evaluated relative to the 10CFR50.59 criteria. The OSGSF shielding evaluation used the SHIELD-SG and MORSE computer programs. SHIELD-SG is a point-kernel program used to calculate direct doses at various distances from the source. MORSE uses the Monte Carlo methodology to calculate direct and skyshine doses. The MORSE portion of the evaluation incorporated an analysis performed for the Indian Point 3 steam generator storage facility which was normalized for application to Sequoyah. The evaluation determined that the OSGSF shielding analysis is not a departure from a method of evaluation described in the UFSAR. This conclusion was based on 1) the results from the SHIELD-SG computer program used in the Sequoyah calculation are essentially the same as those generated by the methods described in the UFSAR and 2) the combination of the SHIELD-SG and MORSE results are conservative over all evaluated distances from the OSGSF, compared to those generated by just the SHIELD-SG computer program.

DCN	DESCRIPTION	SAFETY ANALYSIS
E-21582-A	To limit the amount of reactor coolant system flow which bypasses the reactor core, thimble plugs are installed on the top nozzle of fuel assemblies which do not contain reactivity control components (i.e., control rods, source rods, or burnable absorber rods). The thimble plug assemblies consist of a flat base plate with short rods suspended from the bottom surface. The rods, called thimble plugs, project into the upper ends of the fuel assembly rod cluster control guide tube to block the flow through the guide tubes. The thimble plug assemblies in current use at Sequoyah are approaching the end of their qualified life. As an alternative to assembly replacement, this change will allow the thimble plugs to be removed from the core for future core designs.	The direct impact of thimble plug removal is an increase in core bypass flow through the empty fuel assembly guide tubes. The consequential effects of increased core bypass flow involve departure from nucleate boiling (DNB) performance margins, core hydraulics and fuel assembly qualification. Removal of the thimble plug assemblies resulted in an increase in core bypass flow. A total bypass flow assumption which bounds the calculated bypass increase was used to assess the impact on core design DNB margins, core hydraulic effects and fuel assembly qualification. Sufficient retained margin was available to accommodate the DNB penalty associated with the increased bypass flow. Additional analysis confirmed the beneficial effects of thimble plug removal for core pressure differential, hydraulic lift loads, and inter-assembly crossflow velocities. The modified core conditions were also confirmed to be well within the equipment qualification limits of the current reload fuel assembly design. Given these results, removal of the thimble plug assemblies was concluded to be acceptable.

PROCEDURE	SUMMARY OF DESCRIPTION	SAFETY ANALYSIS
0-SO-30-10, R22 and 0-SO-30-3, R25	The removal and replacement of the Unit 1 steam generators required construction openings to be cut in the Unit 1 shield building and steel containment vessel as detailed in DCN D-20673-A. Revisions to Procedure Nos. 0-SO-30-10, "Auxiliary Building Ventilation Systems" and 0-SO-30-3, "Containment Purge System Operation," were made to minimize the flow of unfiltered air from containment to the auxiliary building or the outside environment while the openings were in place. The revision to Procedure No. 0-SO-30-10 allowed the differential pressure controller for the normal auxiliary building ventilation system to be reset such that the system operated at a 0.0" w.g. differential pressure relative to the outside atmosphere, instead of the normal 0.25" w.g. This minimized the flow of air from containment into the auxiliary building during the steam generator replacement. Also, to minimize air flow from containment to the auxiliary building (and to minimize air flow out the openings in the shield building), only the containment purge exhaust fans were operated while the containment building was breached. The revision to Procedure No. 0-SO-30-3 allowed the containment purge supply fans to be deactivated and the dampers closed, such that the exhaust fans operated without the supply fans. This resulted in an airflow path in through 1) the shield building openings and 2) the equipment hatch (from the auxiliary building) and out through the purge exhaust HEPA filters, charcoal beds, and fan.	The revisions to Procedure Nos. 0-SO-30-3 & 0-SO-30-10 do not increase the frequency or likelihood of accidents or malfunctions, increase the consequences of an accident or malfunction, or create a new type of accident or malfunction. The UFSAR assumes that all reactivity entering the auxiliary building is released to the environment for the first five minutes of the bounding accident. In this analysis, the auxiliary building secondary containment enclosure (ABSCE) is assumed to isolate within the first four minutes following an auxiliary building isolation (ABI) signal and the auxiliary building gas treatment system (ABGTS) was assumed to draw down the auxiliary building differential pressure to -0.25" w.g. within one minute. Analysis has determined that the ABGTS can accomplish the reduction of pressure within one minute starting at 0.0" w.g. The ability of the normal ventilation to isolate, and the ABGTS system to reduce the auxiliary building to -0.25" w.g. is not adversely affected by this activity. The ABI initiation instrumentation and signals are not affected by this change. The containment purge exhaust radiation monitors which initiate isolation of the purge system on high radiation will not be affected by this change. No design bases limits were altered or exceeded by this change.
STI-160, R0	The purpose of Special Test Instruction (STI) No. STI-160 was to determine the effect of operating the normal auxiliary building ventilation system under diverse pressure differentials while simulating the openings in the ABSCE boundary which may exist during the Unit 1 steam generator replacement outage. This special test involved resetting the differential pressure controller for the normal auxiliary building ventilation system such that the system operates at a 0" w.g. differential pressure relative to the outside atmosphere. The controller will then be reset to obtain data on ventilation flow and building pressure. After each controller change, the impact of operation of the ventilation system in this configuration will be assessed by reviewing various flow rates through the openings, effects on doors, and fan performance.	The performance of STI-160 does not increase the frequency or likelihood of accidents or malfunctions, increase the consequences of an accident or malfunction, or create a new type of accident or malfunction. The ABSCE is a barrier to the release of radioactivity and the railroad bay door and/or hatches will be closed to restore the ABSCE within the required drawdown time if the ABGTS actuates during testing. No design basis limits will be altered or exceeded.

PROCEDURE	SUMMARY OF DESCRIPTION	SAFETY ANALYSIS
0-VI-SGR-000-004, R2	This procedure implements the scope of Engineering Package (EP) 24370-EP-004 involving temporary changes associated with the steam generator replacement. Specifically, it addresses the equipment and installations required to 1) rig and transport the original steam generators (OSGs), replacement steam generators (RSGs), shield building and steam generator enclosure concrete sections and steel containment vessel sections, 2) assemble/disassemble and load test the outside lift system (OLS), 3) perform load testing of the haul route, and 4) protect safety-related structures, systems and components (SSCs) from the effects of a postulated-load drop. These activities do not implement any permanent changes to existing permanent plant SSCs subject to design configuration control management. Consistent with the requirements of NUREG-0612 and Nuclear Regulatory Commission (NRC) Bulletin 96-02, rigging activities associated with movement of the steam generators will not be performed over spent fuel or while fuel is in the reactor core. However, movement of heavy loads near or over safety-related SSCs will be required to support steam generator replacement. These safety-related SSCs include the Unit 1 containment building, auxiliary building, essential raw cooling water (ERCW) system, the Unit 1 refueling water storage tank (RWST) and primary water storage tank (PWST), and Unit 1 main steam and feedwater system piping.	Topical Report 24370-TR-C-002 discusses the heavy load lifts over the safety-related SSCs. As detailed in the topical report, a postulated load drop could affect the ERCW system for Unit 2. Compensatory measures and a one-time change to the Unit 2 Operating License were proposed and approved to allow load handling over the ERCW piping and to mitigate the consequences of the postulated load drop. NRC approval of the one-time Operating License change and use of the compensatory measures was obtained prior to use of the OLS (refer to Amendment 273 of the Unit 2 Operating License approved by NRC in a letter dated March 26, 2003). .

PROCEDURE	SUMMARY OF DESCRIPTION	SAFETY ANALYSIS
0-VI-SGR-000-009, R0	This procedure implements the scope of Engineering Package (EP) 24370-EP-009 involving temporary changes associated with the steam generator replacement. Specifically, it provides access to the auxiliary building from the containment access facility (CAF) through the Unit 2 fan room air intakes on Elevation 714. Personnel will proceed from the CAF through an enclosed walkway to the auxiliary building intake louver, enter the access enclosure at the two filter bank area sheet metal single doors, pass through the heating and ventilation equipment room, and exit the access enclosure into the auxiliary building at interlocked Door Nos. A132 and A133. This interlock will be defeated to allow Door Nos. A132 and A133 to be opened at the same time by depressing the emergency override button. Operating the normal auxiliary building ventilation system with Door Nos. A132 and A133 open has the potential to affect the ABSCE boundary during the Unit 1 steam generator replacement outage.	The performance of this procedure does not increase the frequency or likelihood of accidents or malfunctions, increase the consequences of an accident or malfunction, or create a new type of accident or malfunction. The ABSCE is a barrier to the release of radioactivity and Door No. A132 will be closed within the required four minute timeframe allowed for establishment of the ABSCE if an ABI initiation occurs. No design basis limits will be altered or exceeded.

SEQUOYAH NUCLEAR PLANT

CHANGES IN THE FACILITY FOR AMENDMENT - 18

TACF	SUMMARY OF DESCRIPTION	SAFETY ANALYSIS
1-03-035-068	This temporary change involves closing valve No. 1-VLV-068-576 which is shown on FSAR Figure 5.1-1 as a normally open valve. The valve is being closed to isolate a flow control valve (Valve No. 1-FCV-43-001) which is leaking through the valve stem and cannot be safely isolated for repair with the reactor coolant system at normal operating temperature and pressure. Closure of valve No. 1-VLV-068-576 will isolate the pressurizer gas sample line.	The pressurizer gas sample line is an infrequently used operator aid which is not credited in the plant safety analysis or post accident monitoring system. Temporary isolation of the sample line to prevent primary system leakage will not affect any system functional requirements described in the UFSAR. It will not increase the probability or consequences of any event evaluated in the UFSAR.
1-03-036-241	This temporary change removes sudden pressure relay (SPR) protection from the Unit 1 main bank transformers. This change will disable the generator trip and turbine trip initiated by the SPRs. Removal is accomplished through a trip cut out switch. Differential protection and fire deluge protection will not be affected by this change. Before this change, the SPRs were susceptible to false actuation due to mechanical agitation of the relay hose assembly. This false actuation could result in an unnecessary generator trip, turbine trip, and fire protection deluge actuation. This change will reduce the risk to plant systems due to unnecessary actuation of the SPRs until a modification to prevent false actuation due to mechanical agitation can be made.	Disabling the SPRs does not increase the frequency of occurrence of UFSAR accident. The SPRs provide protection to the transformer from sustained damage. They do not prevent loss of load or turbine trip transients. Diverse protection from the transmission system transients is provided by differential relays. The only equipment important to safety in the vicinity of these transformers is the 6.9KV unit boards located inside the turbine building. These unit boards are separated from the main transformers by a concrete wall which prevents damage to the boards due to catastrophic failure of the transformer. As such, the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR are not increased. Failure of the main bank transformer is bounded by the loss of off-site power transient evaluated in the UFSAR. No Technical Specification (TS) equipment is affected by this change such that the margin of safety as defined in the basis for any TS is not reduced.
2-03-038-061	This temporary change involves a reroute of the ice condenser air handling unit condensate drain line in the reactor building upper containment. The reroute involves disassembling a vertical 2 inch carbon steel drain pipe near the refueling floor, installing temporary copper tubing with a temporary hanger to the drain and directing condensate through the tubing to an existing refuel floor drain. The temporary alteration is required to properly drain the condensate from the ice condenser. Blockage in the existing drain line has caused the condensate to back up inside the ice condenser and freeze on an intermediate deck door. Currently, manual removal of condensate from the clogged drain line is performed daily to prevent freezing of intermediate deck doors and to maintain the average ice bed temperature.	The temporary alteration will redirect effluent resulting from a potential overfill of the system expansion tank (i.e., ice condenser air handling unit condensate and potentially glycol) from the reactor building floor and equipment sump to the auxiliary reactor building floor and equipment sump (pocket sump). The design of the auxiliary reactor building floor and equipment sump (pocket sump) complies with Regulatory Guide 1.45, Revision 0. The pocket sump has level instrumentation to detect the rate of rise. The sensitivity is such that one gallon per minute inflow can be detected in approximately one hour to promptly detect a potential reactor coolant system pressure boundary break. This functional capability will not be affected by the temporary alteration.

SEQUOYAH NUCLEAR PLANT

CHANGES IN THE FACILITY FOR AMENDMENT - 18

WORK ORDER	SUMMARY OF DESCRIPTION	SAFETY ANALYSIS
WO 01-005035-000	This work order will change the position of the component cooling system (CCS) flow throttle valve to the Unit 2, Train A residual heat removal (RHR) heat exchanger (Valve No. 2-FCV-70-156). During normal operation, this valve is closed (or throttled to support CCS pump minimum flow requirements). The valve is positioned in the full open position when the RHR heat exchanger is placed in service. Due to a problem with the valve actuator, opening or throttling the valve cannot be performed reliably. To ensure the operability of the valve, this work order will place the valve in the full open position and leave the valve in this position for normal and accident operation.	A functional review of Valve No. 2-FCV-70-156 in the full open position confirmed 1) the valve is in the correct alignment required for shutdown cooling or for accident mitigation, 2) other CCS served components receive adequate flow and total CCS pump flow is within analysis limits, 3) alternate valves in the CCS are available to isolate a CCS line break, 4) CCS flow is monitored and when conditions warrant a second CCS pump can be started to maintain CCS required flow and 5) alternate valves are available to control RCS cooldown during shutdown operations such that there is no increase in the probability of an accident or the malfunction of equipment. Since all safety-related functions and operational functions can be met with Valve No. 2-FCV-70-156 aligned and maintained in the full open position, there can be no more than a minimal increase in the probability of an accident or a malfunction of equipment
WO 02-006283-008	The essential raw cooling water (ERCW) return line from upper compartment cooler (UCC) Nos. 1B and 1D is to be cut and capped upstream of Valve No. 1-VLV-67-590B to support abandonment of the coolers during the Unit 1, Cycle 12 refueling outage. Because the ERCW system is in service and Valve No. 1-VLV-67-590B is not sealing, a freeze plug will be placed in the pipe between Valve No. 1-VLV-67-590B and the 20" diameter ERCW discharge header to allow the pipe upstream of Valve No. 1-VLV-67-590B to be drained, cut and capped.	A failure of the freeze plug or a brittle fracture of the ERCW pipe would result in a maximum 3" diameter pipe break in Room A6-690. There is no Unit 2 (i.e., the operating unit) electrical equipment required for safe shutdown or accident mitigation that would be affected by the water spray from the break. This room and the remainder of the auxiliary building are analyzed for a moderate energy line break (MELB) flood originating in A6-690 which is equal to this 3" ERCW pipe break. The UCC No. 1B and 1D supply line has been cut and capped so it does not feed the break. Only backflow from the main discharge header is involved. The MELB mitigation strategy is to detect and isolate a MELB within 60 minutes of its initiation. This allows 60 minutes for the shutdown of Unit 2 and isolation of ERCW discharge header B. During the shutdown of Unit 2, an unintentional or deliberate trip of the reactor may occur. Because the freeze plug is a one-time activity needed to implement a Unit 1 modification during a planned outage, there is only a minimal increase in the probability of a reactor trip on Unit 2.

WORK ORDER	SUMMARY OF DESCRIPTION	SAFETY ANALYSIS
		Isolating ERCW discharge header B disables Train B of all engineered safeguard features (ESF) equipment. Because the limiting condition for operation (LCO) for T/S 3.7.4 has been entered for ERCW Train 2B, no single failure needs to be postulated for any Train 2A SSC. As a result, Train 2A of all ESF equipment is available and is sufficient to mitigate all design basis accidents on Unit 2. Since isolation of the Train B discharge header results in isolation of the 1A1 & 1A2 component cooling heat exchangers (thereby impacting spent fuel pool cooling), both units must be maintained at Mode 5, 6, or no mode until the header is back in service. A total loss of spent fuel pool cooling is evaluated in the UFSAR with 30 hours available to restore cooling or initiate makeup to the pool.

UFSAR REVISION	SUMMARY OF DESCRIPTION	SAFETY ANALYSIS
Section 3.1.2 and 8.2	This change involves a revision to the Updated Final Safety Analysis Report (UFSAR) to change the capabilities of the off-site power system. The current description of off-site power capabilities indicates there are 2 immediate sources of off-site power. Recent grid calculations indicate there will be times (due to system load, individual plant status, etc.) that the off-site power system will not be capable of providing 2 qualified power sources without offsite dispatcher action. One immediate source will be available, but an alternate source which requires manual actions to establish does not qualify as immediately available. This change will revise the UFSAR to indicate that there is one immediate and one delayed source of off-site power available. In addition, this change will remove excessive/extraneous details from various sections of the UFSAR that relate to offsite power. These include the name and distances to various connected plants, specific load cases for evaluation of offsite power availability and other details.	The existing UFSAR description indicating that there are 2 immediate sources of off-site power available exceeds the requirements of General Design Criteria (GDC) No. 17 and Regulatory Guide (RG) 1.32. The revised capabilities will meet the requirements of both GDC No. 17 and RG 1.32. This change does not affect the ability of the off-site power system to supply normal power for continued operation and does not affect the on-site emergency power system.
Section 15.2.1	This change involves a revision to Section 15.2.1 of the Updated Final Safety Analysis Report (UFSAR) to incorporate a revised analysis of the uncontrolled rod cluster control assembly (RCCA) withdrawal from subcritical conditions transient. The revised analysis demonstrates that the reactor coolant system temperature-pressure response to the transient is well within the system design limits and confirms that the minimum calculated departure from nucleate boiling (DNB) ratio is above the established safety limit. The results meet all of the previously established transient acceptance criteria. The revised analysis differs from the previous analysis in that it uses an NRC approved evaluation methodology developed by Framatome Advanced Nuclear Power (FANP). The FANP methodology establishes core power and reactor coolant system temperature and pressure conditions using the RELAP5/MOD2-B&W computer code and the resultant DNB conditions using the LYNXT computer code. The previous analysis uses a methodology developed by Westinghouse Electric Company which establishes the core power transient using the TWINKLE computer code, the system heat addition using the FACTRAN	The revised analysis uses methodology and computer codes approved by NRC and is in compliance with the conditions and restrictions placed on their application. It does not represent a departure from an approved method of evaluation. As a result, the revised analysis was adopted as the analysis of record under the current licensing basis.

UFSAR REVISION	SUMMARY OF DESCRIPTION	SAFETY ANALYSIS
	code and the resultant DNB conditions using the THINC computer code. The reanalysis was performed to make the methodology used for the uncontrolled rod withdrawal from subcritical conditions analysis consistent with other transient analyses which use methodologies developed by FANP. FANP is the current supplier of reload fuel and reload analysis services for Sequoyah.	
Section 15.5	This change involves a revision to Section 15.5 of the Updated Final Safety Analysis Report (UFSAR) to incorporate results of revised dose calculations for all design basis accidents. The dose calculations for the steamline break (SLB), steamline break with alternate tube repair criteria (SLB-ATRC) and steam generator tube rupture (SGTR) transients were revised to address the iodine spiking issue outlined in Westinghouse Nuclear Safety Advisory Letter NSAL-00-004. In addition to addressing the iodine spiking issue, the revised calculations were based on ICRP 30 methodology which uses methods and parameters outlined in Federal Guidance Report Number 11 and Federal Guidance Report Number 12. The balance of the analyses (loss of off-site power (LOOP), waste gas decay tank rupture (WGDTR), loss-of-coolant accident (LOCA) and the fuel handling accident (FHA) were also revised to use the decontamination factors from ICRP-30.	The increase in dose or consequence for these design basis accidents is minimal if the increase is 1) less than or equal to 10 percent of the difference between the current calculated dose value and the regulatory guideline value [10CFR100 or GDC-19, as applicable] and 2) the increased dose does not exceed the current standard review plan guideline value for the particular design basis event. While the revised doses are greater than the existing doses for the loss of offsite power LOOP, steamline break SLB, SLB-ATRC, SGTR and fuel handling accident inside containment (FHA-IC) design basis accidents, the results demonstrate that the increase in dose is minimal except for the thyroid dose at the exclusion area boundary (EAB) for the SGTR. The consequence analysis for the SGTR has been previously described to the NRC and was approved in accordance with a safety evaluation report issued for Technical Specification Change No. 00-06 by an NRC letter dated September 30, 2002. Therefore, the UFSAR changes may be made without additional NRC review/approval.