Nuclear Operating Company

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U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
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Rockville, MD 20852-2738

South Texas Project
Units 3 and 4
Docket Nos. 52-012 and 52-013
Proposed Revision to Environmental Report

Attached is a change to the Combined License Application (COLA) Part 3, Environmental Report. This change provides a new supplemental section; Section 7.5S. This change will be incorporated in the next regular revision of the COLA.

There are no commitments in this letter.

If you have any questions, please feel free to contact me at (361) 972-7136, or Russell W. Kiesling at (361)-972-4716

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 11/10/09

Scott Head
Manager, Regulatory Affairs
South Texas Project, Units 3 & 4

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Attachment: COLA Part 3 Section 7.5S Supplemental Text

D091
NRO

STI: 32572382

cc: w/o attachment except*
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7.5S Design Basis Accident or Severe Accident Impact on Other STP Units

Section 7.1 describes the impacts that a design basis accident would have at one of the ABWR units (STP 3 or 4). Sections 7.2 and 7.3 describe the impacts and costs that a severe accident would have at one of the ABWR units. This section describes (1) the impacts that a design basis accident or severe accident at one of the ABWR units would have on the other three onsite units (the other ABWR unit and STP 1 & 2); and (2) the impacts that a design basis accident or severe accident at either STP 1 or 2 would have on the ABWR units. With a few exceptions, this section does not evaluate the impacts of an accident at STP 1 on STP 2, or vice versa, because such an evaluation is unrelated to STP 3 & 4.

There is no mechanism for fire or explosion by which one unit could affect other units. FSAR Section 2.2S.3 evaluates potential accidents that could impact STP 3 & 4, including those resulting in fires or explosions. That section demonstrates that STP 3 & 4 are located at a safe distance from chemical storage facilities for STP 1 & 2 and therefore are not at risk to impacts from explosions, fires, or release of toxic chemicals from STP 1 & 2. As further discussed in FSAR Section 2.2S.3, the chemicals used at STP 3 & 4 are similar to the chemicals used in STP 1 & 2 and would not be stored any closer than the determined safe distances from explosions, fires, or release of toxic chemicals to STP 1 & 2 and STP 3 & 4. Furthermore, each of the ABWR units is designed to withstand or achieve safe shutdown in the event of fires, explosions, and toxic gases originating at that unit; therefore, each would be able to withstand such events originating at the other ABWR unit. Additionally, design features of the ABWR and STP 1 & 2 would mitigate other types of indirect impacts. For example, in the event of a power disruption caused by the accident at one unit, the emergency diesel generators at the other units can be started to ensure that the other units have sufficient electrical power to provide for and maintain safe shutdown of the other units.

Therefore, this section evaluates a scenario in which airborne radioactivity released from a design basis accident or severe accident at an affected unit (the unit at which the initiating accident occurs) may result in an accident or service disruption at an unaffected unit (a unit other than that at which the initiating accident occurs). A service disruption would entail a delay in returning the unaffected units to service as a result of repair, refurbishment, decontamination, or corrective action. The evaluation considers whether exposures could interrupt safe shutdown of an unaffected unit by either interfering with operator actions or by interfering with or damaging equipment with a safety function. Additionally, this evaluation discusses the environmental impacts and quantifies the potential cost of the temporary loss of use of one of the unaffected units, analogously to that of the affected unit, as described in Section 7.3.

7.5S.1 Background on Impact Mitigation and Prevention

As discussed below, various factors at an unaffected unit mitigate or prevent the impacts from a radiological accident at the affected unit. These include the warning time that the unaffected unit receives of an accident, the ability to place the unaffected unit into a safe shutdown condition, control room habitability, and shielding to personnel and equipment from the plant design.

Plant design and procedures provide protection for operators. Control room habitability systems are designed to protect the control room during an accident and include missile protection, radiation shielding, radiation monitoring, air filtration and ventilation systems, and fire protection. 10CFR50, Appendix A, General Design Criterion (GDC) 19 specifies a 5 rem control room operator dose limit for releases from a design basis accident at that unit. If this dose were to be exceeded within the control room during a severe accident, operators could be protected with additional measures, such as donning a SCBA (self contained breathing apparatus) and limiting exposure time. Additionally, once a plant is shut down, stable, and in long term decay heat removal, operator action is not continually necessary to maintain the plant in a safe shutdown condition. Once a stable cool down rate has been established, operator adjustments are no longer required to maintain the plant in a stable safe shutdown condition. Therefore, at that time, the operators could be evacuated from the control room if necessary.

An important factor in mitigating and preventing major impacts at an unaffected unit is the warning time that an operator of that unit receives of an accident at the affected unit. With sufficient warning time prior to radioactive releases from the affected unit, the unaffected units can be put into a safe shutdown mode. Additionally, in the event of a design basis accident or severe accident, non-essential site personnel could be evacuated in keeping with site emergency procedures.

A unit can be put into a hot shutdown condition, in which the reactor is completely shutdown, within minutes. Cooling operations can be commenced shortly after hot shutdown. After about 3 hours, the reactor will be in a stable long-term decay heat removal condition. Once long term decay heat removal is established the operations staff will adjust the cooling systems to establish a stable cool down rate. After that time, operator action is not necessary to maintain the plant in a safe shutdown condition. If the time increment between the onset of the accident and the airborne radioactivity release at the affected unit is longer than the time it takes to place the reactor into a stable long-term decay heat removal condition (approximately 3 hours), then there would be no impact on safe shutdown of an unaffected unit. Once the unaffected units are in safe shutdown, a release from the affected unit would not adversely impact maintenance of that safe shutdown condition because, as described below, the equipment can withstand the doses associated with the release without loss of safety function.

Equipment can inherently withstand large radiation doses from an accident, and plant design features, such as shielding, provide additional protection. For example, the concrete of the containment structure provides substantial shielding and the containment is sealed, thus preventing the intrusion of airborne radioactivity to equipment within containment. The concrete in other concrete buildings also would provide some shielding to equipment. Equipment can also withstand the release of radioactive material from accidents. For example, FSAR Section 2.2S.3.1.7 explains that safety-related structures, systems, and components for the ABWR are designed to withstand the effects of radiological events and the consequential releases from design basis accidents.

Section 7.5S.4 below discusses the expected doses at the exterior of the unaffected units resulting from the evaluated accidents, and demonstrates that the structures and equipment in the unaffected units would still be able to perform their safety function given such doses.

7.5S.2 Evaluation of Impacts of Design Basis Accidents on Safe Shutdown of Other Units

Design Basis Accidents Originating in STP 3 or 4

Section 7.1 provides STP 3 & 4 Exclusion Area Boundary doses that would result from design basis accidents at either STP 3 or 4. FSAR Section 15.6 demonstrates that doses in the control room would be within the regulatory limit of 5 rem during a design basis accident at that same unit. The dose at the control room of the unaffected ABWR would also be within the regulatory limit of 5 rem during a design basis accident at the affected ABWR unit because the design and protection of the control rooms are identical. The control room dose at STP 1 & 2 would be similar in magnitude to the doses experienced at the unaffected ABWR unit control room because of the protection of the control room habitability systems at STP 1 & 2, which are required to satisfy the requirements of GDC 19 similar to STP 3 & 4. In fact, the doses would likely be less due to the larger distance between the affected unit and STP 1 & 2 than the affected unit and the other ABWR. Therefore, the doses experienced at the control rooms of STP 1 & 2 or the unaffected ABWR unit from a design basis accident at either STP 3 or 4 would not prevent the operators from completing safe shutdown of the unaffected units.

As discussed below in Section 7.5S.4, equipment can withstand severe accident doses without loss of function. Therefore, a design basis accident at STP 3 or 4 would not have an impact on the safe shutdown of STP 1 & 2 and the other ABWR unit.

Design Basis Accidents Originating in STP 1 or 2

As described in FSAR Section 2.2S.3.1.7, radiological releases from a design basis accident at STP 1 & 2 would not threaten the safety of STP 3 & 4. As noted previously, control room habitability systems can detect and protect control room personnel from airborne radioactivity. Therefore, a design basis accident at STP 1 or 2 would not cause the operators at STP 3 & 4 to exceed the 5 rem limit in GDC 19 and would not prevent the operators from completing safe shutdown of the ABWR units.

FSAR Section 2.2S.3.1.7 also states that safety related structures, systems, and components for the ABWR have been designed to withstand the effects of airborne releases from a design basis accident at the ABWR that would bound the airborne release from either STP 1 or 2.

Therefore, a design basis accident at STP 1 or 2 would not have an impact on the safe shutdown of STP 3 & 4.

In summary, a design basis accident at any affected unit would not impact the safe shutdown of an evaluated unaffected unit.

7.5S.3 Evaluation of Impacts of Severe Accidents on Safe Shutdown of Other Units

Severe Accidents Originating in STP 3 or 4

Section 7.2 describes the offsite dose and cost risks that could accompany a severe accident at either STP 3 or 4. A number of accident sequences, each of which represents a broader family of accidents, are analyzed for such an accident class. For the ABWR, ten accident sequences are analyzed for internally initiated events. The sum of the frequencies of occurrence for each of the ten accident sequences, which are shown in Table 7.2-1, is the core damage frequency (CDF). The CDF of an ABWR for internal events is 1.6 $\times 10^{-7}$ per year.

However, not all core damage events result in a large release (i.e., the containment is able to prevent the release of significant amounts of radioactivity to the environment). Absent a large release, there would be no impact on the safe shutdown of the unaffected units. In general, the large release frequency (LRF) for the ABWR for internal events is about 2.2 $\times 10^{-8}$ per year. Furthermore, most of the LRF consists of a release which has been scrubbed by the suppression pool water and passes through the Containment Overpressure Protection System. The LRF for a release that has not been scrubbed is 1.0 $\times 10^{-9}$ per year.

Externally initiated events, and their associated small contribution to risk, are described in FSAR Section 19.4 and 19.6, which in turn incorporate by reference the associated sections of the ABWR DCD. As stated in the Final Safety Evaluation Report for the ABWR (NUREG-1503):

> Although direct comparison of external-event results to [the Commission's safety goals] is not possible, the ABWR design has significant margins above the design bases for seismic, fire, and internal flood-initiating events and, where computed, has low estimated core damage frequencies from these bounding analyses. The staff believes that the ABWR design meets the Commission's safety goals.

As discussed previously, operators with sufficient warning of an accident at an affected unit can safely shutdown an unaffected unit. The time increment from general emergency warning time until the first release of radioactivity to the environment for all ten accident sequences is greater than the time required to put an unaffected unit into a stable long-term decay heat removal condition. Therefore, any doses experienced at the control rooms of STP 1 & 2 or the unaffected ABWR unit from a severe accident at either STP 3 or 4 would not prevent the operators from completing safe shutdown of the unaffected units. Additionally, as discussed below in Section 7.5S.4, equipment can withstand the bounding radiation dose from a severe accident without loss of function.

Severe Accidents Originating in STP 1 or 2

A similar analysis of representative accident sequences was performed for STP 1 & 2. Nine representative sequences were analyzed, with a total CDF of 1.0×10^{-5} per year and a Large Early Release Frequency (LERF) of 6.1×10^{-7} per year for internal and external events. The time increment from general emergency warning time until the first release of radioactivity to the environment for all nine representative sequences is greater than the time required to put an unaffected unit into a stable long-term decay heat removal condition. Therefore, any doses experienced at the control rooms of STP 3 or 4 from a severe accident at either STP 1 or 2 would not prevent the operators from completing safe shutdown of the unaffected ABWR units. Additionally, as discussed below in Section 7.5S.4, equipment can withstand the bounding radiation dose from a severe accident without loss of function.

In summary, a severe accident at any affected unit would not impact the safe shutdown of an evaluated unaffected unit.

7.5S.4 Evaluation of Impacts of Design Basis Accidents and Severe Accidents on Equipment Function

Equipment at the unaffected units would continue to function under the accident scenario considered above in Sections 7.5S.2 and 7.5S.3 that results in the highest radiation dose. The accident with the highest radiation dose is an Induced Steam Generator Tube Rupture (ISGTR) at STP 1 or 2, which is characterized by a low CDF of 6.1×10^{-7} per year.

The ISGTR is initiated by a loss of offsite power (LOOP) that is not recoverable prior to fuel damage in the affected unit. All emergency diesel generators are assumed to also fail and are not able to be recovered. The turbine driven auxiliary feedwater pump likewise is assumed to fail and is also not able to be recovered.

Exposures at an unaffected ABWR unit resulting from severe accident releases from this accident sequence were calculated using the MACCS2 code. That code is described in Section 7.2, and was used there to calculate offsite dose and cost risks from severe accidents. Only the code's early release phase was exercised for this analysis, i.e., short-term doses to personnel and equipment during the release plume passage were calculated. This is the period of concern relative to safe shutdown of the unaffected units and onsite contamination.

The STP 1 & 2 ISGTR accident sequence results in an estimated worst case dose at STP 3 & 4 of up to 2,500 rad to air. This calculated exposure is without any shielding, and is cumulative over the entire duration of the early airborne releases. The meteorology is also conservatively assumed to be at the 95% level, meaning that 95% of the time actual weather conditions would lead to less exposure at that location.

Equipment at the unaffected units can withstand this bounding radiation dose and continue to properly function. As discussed in the STP 1 & 2 Design Criteria for Equipment Qualification Program (Reference 7.5S-1), an environment with exposures of less than $1x10^5$ rad is considered to be a mild environment and does not require any special qualification requirements to ensure equipment function. Thus, equipment installed at the STP units can withstand $1x10^5$ rad of radiation exposure without any impact on equipment functionality. As discussed above, the bounding radiation dose for the evaluated accidents is 2,500 rad, which is significantly less than the radiation level that would impact equipment functionality. Therefore, all equipment necessary to complete safe shutdown of the unaffected units would be able to operate as designed without any degradation to its functional capabilities for the exposure levels associated with the airborne release from the accidents evaluated above.

7.5S.5 Economic Impacts of a Temporary Shutdown of the Unaffected Units

The potential economic impacts from an accident at an affected unit on an unaffected unit are quantified by monetizing the onsite exposure and cleanup costs at the unaffected units together with replacement power costs from an outage at the unaffected units. The calculations are analogous to those in Section 7.3 (using the same methodology as described there unless otherwise noted) for an accident and attendant impacts from a single affected unit.

The principal inputs to the analysis are the severe accident CDF, the outage period of the other site units, and the economic discount rate (7 percent and 3 percent are NRC precedents established by NUREG/BR-0184). With these inputs, monetized impacts per unaffected unit are presented in Tables 7.5S-1 and 7.5S-2 for an event at one of the existing units (STP 1 & 2) and at one of the ABWR units (STP 3 & 4), respectively. A design basis accident would have much lower releases associated with the accident compared to a severe accident, resulting in much lower contamination levels that would be bounded by the evaluation for a severe accident. Therefore, only the severe accident CDFs were considered to produce a long term outage period from the associated cleanup and refurbishment of equipment.

Unlike in Section 7.3, where an accident at one unit could result in offsite impacts due to radiation releases from that unit, no releases or offsite impacts would result from the unaffected units. In order to determine the economic impact of a temporary shutdown in the unaffected units, the methodology used for the affected unit is conservatively applied here to all unaffected units.

The onsite cleanup cost includes cleanup and decontamination of the unaffected units. The cleanup of these units, unlike for the affected unit (as analyzed in Section 7.3), is based on recovering the units for restart. Recoverable cleanup costs have been estimated as 30% of the non-recoverable cleanup costs (as included in the Section 7.3 initiating unit costs) for BWR units (STP 3 & 4) and 26% for PWR units (STP 1 & 2) (Reference 7.5S-2); 30% is conservatively used here for all units. Those costs are based on cleanup of a small LOCA (loss of cooling accident) which results in a moderately contaminated containment building; applying these costs to units which would not have internal releases but instead can be contaminated by external releases from the initiating unit is conservative.

It is expected that the unaffected units could be restarted within months. However, it took 6 years to restart Three Mile Island Unit 1 (TMI-1) after the accident at TMI-2. It should be noted that cleanup and refurbishment were not the driving actions for the restart delay. Instead, the restart awaited application and approval of lessons learned from the TMI-2 accident. This analysis assumes that the unaffected unit with the same design of the affected unit would be shutdown for 6 years for cleanup, refurbishment, and application and approval of lessons learned. The analysis assumes that the other two unaffected units would be shutdown for 2 years for cleanup and refurbishment.

The cost of repairs and refurbishment at the unaffected units was estimated at $1,400 per hour of outage duration, which is $1,000 per hour escalated to the calculation basis year (Reference 7.5S-2), and is included in the onsite cleanup cost. If Tables 7.5S-1 and 2 were based on 6 months of outage rather than 6 years, the total costs would be 40-45% (depending on discount rate) of those indicated in the tables. Almost all of that decrease would be due to the decrease in replacement power cost.

NRC suggests that a typical short-term replacement power cost (i.e., power plant that will be restarted) for a 910 MWe power plant is $310,000 per day (Reference 7.5S-2). That value, scaled to the STP power levels, replaced the present value in the replacement power cost calculation of Section 7.3 for this analysis.

The monetized impacts for an accident at STP 3 or 4 affecting another unit are very low, as shown in Table 7.5S-2. The cost at a 7 percent discount rate at the other ABWR unit would be approximately $3,000 and the cost at STP 1 or 2 would be approximately $1,800 per unit. Even at a 3 percent discount rate, the cost at the other ABWR unit would be approximately $4,500 and the cost at STP 1 or 2 would be approximately $3,000 per unit. These costs are less than half of the costs of an accident at the affected unit. The Section 7.3 conclusion that there is no cost-effective ABWR operation design change holds for the mitigation of impacts at other site units.

The monetized risk-based impacts from an accident at STP 1 or 2 on the ABWR units are larger than the impacts from an ABWR initiated accident case, due to the larger CDF of the existing units. The monetized impact cost to an ABWR unit from a large severe accident release at one of the existing units is shown in Table 7.5S-1. The cost at a 7 percent discount rate at the ABWR units would be approximately $110,000 per unit, and at a 3 percent discount rate would be approximately $170,000 per unit. None of the severe accident mitigation design alternatives considered for the ABWR would be cost effective and mitigate the potential impacts (contamination and down time) from a large release severe accident at the existing units (Reference 7.5S-3).

7.5S.6 Conclusions

As demonstrated above, a design basis accident or a severe accident at the affected unit would not prevent the unaffected units from safely shutting down. Additionally, all equipment necessary to complete safe shutdown of the unaffected units would be able to operate as designed without any degradation to its functional capabilities for the exposure levels associated with the airborne release from the accidents evaluated. Therefore, the accident scenarios would not result in any incremental environmental impacts attributable to the unaffected units beyond those evaluated in Sections 7.1 and 7.2.

Furthermore, even if it is arbitrarily postulated that a severe accident in the affected unit could cause a simultaneous severe accident in each of the unaffected units, the cumulative environmental impacts would still be SMALL. In such a scenario, the releases of radioactivity from all four units would be approximately four times the release from an individual unit. However, even if the environmental impacts (risks) discussed in Section 7.2.4 for an accident originating in one of the ABWR units were to be multiplied by a factor of four, the environmental risks would still be insignificant. For example, the cumulative risk from all four units would be about 0.017 person-rem/year (i.e., 4 x 0.0043 person-rem per reactor year), which is more than a factor of ten less than the cumulative dose risk from normal operation (about 0.5 person-rem per reactor year). Furthermore, the risk of cancer from such an accident scenario would be about 0.0044% of the background risk (i.e., four times 0.0011% of the background risk). This value is well below the 0.1% value specified in the Commission's Safety Goal Policy Statement.

As discussed in Section 7.5S.3, the LERF for Units 1 and 2 is approximately 30 times greater than the LRF for Units 3 and 4. However, even if the risk-based values in the previous paragraph were to be multiplied by a factor of 30, the resulting dose risk would be equivalent to the cumulative dose risk from normal operation and the resulting cancer risk would be equivalent to the Commission's Safety Goal. Therefore, the environmental impact from such a scenario would be SMALL.

7.5S.7 References

7.5S-1 Design Criteria for Equipment Qualification Program 4EQ19NQ1009 for South Texas Nuclear Operating Company (Units 1&2), January 26, 2000.

7.5S-2 NRC (U.S. Nuclear Regulatory Commission). 1997. Regulatory Analysis Technical Evaluation Handbook, NUREG/BR-0184. Office of Nuclear Reactor Regulation. Washington, D.C. January.

7.5S-3 GE (General Electric Company). 1995. Technical Support Document for the ABWR, Revision 1, MPL Number A90-3230, 25A5680, January.

Table 7.5S-1 Monetized Impacts for an Accident at STP 1 or 2 per Unaffected Unit.

	STP 3 or 4 Impacts	
	7 Percent Discount Rate	3 Percent Discount Rate
Onsite exposure cost	$5,569	$10,355
Onsite cleanup cost	$61,900	$108,526
Replacement power cost	$42,727	$55,194
Total	$110,196	$174,075

Table 7.5S-2 Monetized Impacts for an Accident at STP 3 or 4 per Unaffected Unit.

	STP 3 or 4 Impacts		STP 1 or 2 Impacts	
	7 Percent Discount Rate	3 Percent Discount Rate	7 Percent Discount Rate	3 Percent Discount Rate
Onsite exposure cost	$76	$152	$137	$219
Onsite cleanup cost	$949	$1,764	$934	$1,676
Replacement power cost	$1,980	$2,557	$688	$1,153
Total	$3,005	$4,473	$1,759	$3,049