Nuclear Operating Company
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U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

South Texas Project
Units 1 & 2
Docket Nos. STN 50-498, STN 50-499
Response to Request for Additional Information
Regarding the License Amendment Request for
Revision to the South Texas Project Fire Protection Program
Related to the Alternative Shutdown Capability (TAC Nos. ME6346 and ME6347)

Reference:
1. STPNOC letter dated June 2, 2011 from G. T. Powell to the NRC Document Control Desk, "License Amendment for Approval of a Revision to the South Texas Project Fire Protection Program Related to the Alternate Shutdown Capability," dated June 2, 2011(NOC-AE-11002643) (ML11161A143)
2. STPNOC letter dated August 1, 2011 from Charles T. Bowman to the NRC Document Control Desk, "Supplement to the License Amendment Request for Approval of a Revision to the South Texas Project Fire Protection Program Related to the Alternative Shutdown Capability (TAC Nos. ME6346 and ME6347)" (NOC-AE-11002703) (ML11221A330)
3. NRC Document dated February 9, 2012, "South Texas Project, Units 1 and 2 – Request for Additional Information Email, License Amendment Request to Approve Revision to Fire Protection Program in Fire Hazards Analysis Report Related to Alternate Shutdown Capability (TAC Nos. ME6346 and ME6347)" (ML120400126)

In reference 1, STP Nuclear Operating Company (STPNOC) submitted a licensee amendment request (LAR) for approval of a revision to the South Texas Project (STP) Fire Protection Program (FPP) related to the Alternative Shutdown Capability. Reference 2 provided supplementary information in support of the LAR. Per Reference 3, the Nuclear Regulatory

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Commission (NRC) requests additional information to support their review of the LAR. The STPNOC response to Reference 3 is provided in the Enclosure to this letter.

There are no regulatory commitments in this letter.

If there are any questions regarding this amendment request, please contact Ken Taplett at (361) 972-8416 or me at (361) 972-7566.

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

Executed on 3-8-2012
Date

G T Powell

G. T. Powell
Vice President,
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Enclosure: Response to Request for Additional Information

cc:

(paper copy)

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Response to Request for Additional Information

REQUEST FOR ADDITIONAL INFORMATION (RAI)
LICENSE AMENDMENT REQUEST
REVISION TO THE SOUTH TEXAS PROJECT (STP), UNITS 1 AND 2
FIRE PROTECTION PROGRAM RELATED TO THE
ALTERNATE SHUTDOWN VCAPABILITY

NRC RAI 1

In its submittal dated June 2, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML11161A143), STP Nuclear Operating Company (STPNOC, the licensee) stated that a fire induced spurious opening of the pressurizer PORVs would result in an engineered safety features safety injection (SI) signal in approximately 71 seconds. In submittal dated August 1, 2011 (ADAMS Accession No. ML11221A230), however, the time is identified as 61 seconds following reactor trip. Please clarify which value is correct.

STPNOC Response

In the submittal dated June 2, 2011, the 71 seconds reported that would result in an engineered safety features safety injection signal was based on the analysis results which included 10 seconds of steady state operations before the initiation of a reactor trip and pressurizer PORV failing open. In the submittal dated August 1, 2011, this same time was reported as 61 seconds, which did not include the 10 seconds of steady state operations. The value of 61 seconds should be used for the purposes of determining when a safety injection signal can be expected after a reactor trip and pressurizer PORV failing open.

NRC RAI 2

In its request dated June 2, 2011, the licensee states that operators can close the power operated relief valve (PORV) block valves within sixty seconds. Analytic results indicate that a SI signal would be received in either 61 or 71 seconds following spurious PORV opening on reactor trip, assuming the reactor is in a nominal condition prior to reactor trip. Please quantify the uncertainty in SI signal receipt time attributable to the pre-trip power level, reactor coolant system (RCS) average temperature, pressurizer pressure and level, and RCS flow uncertainties and demonstrate that this uncertainty, when accounted for, leaves enough time for operator intervention to close the PORV block valves post-trip.

STPNOC Response

The South Texas Project fire hazards analysis indicates that a SI signal would be received in 61 seconds following spurious PORV opening coincident with reactor trip, assuming the reactor plant is at nominal conditions prior to reactor trip. The analysis assumes the operators can close the PORV block valves within 60 seconds of reactor trip.

The transient occurs at the initiation of a manual reactor trip. The fire hazards analysis conservatively assumes that a fire-induced spurious actuation of a PORV occurs immediately following the reactor trip. The SI signal occurs when pressurizer pressure is reduced to the SI actuation setpoint. The pre-trip parameters whose uncertainty would have a significant impact on the SI receipt time are the initial pressurizer pressure signal and low pressurizer pressure SI setpoint. The pressurizer pressure instrument uncertainty is 19.7 psi and the low pressurizer pressure SI setpoint uncertainty is 117.68 psi. The analysis assumed the nominal initial pressurizer pressure of 2235 psig and a nominal low pressurizer pressure SI setpoint of 1857 psig. The depressurization rate for this event is approximately 6.2 psi per second. Subtracting the pressurizer pressure instrument uncertainty from the initial pressurizer pressure and adding the low pressurizer pressure SI setpoint uncertainty to the low pressurizer pressure SI setpoint minimizes the time to the low pressurizer pressure SI signal. By taking the difference between these two pressures when considering uncertainties and dividing by the depressurization rate, the low pressurizer pressure SI setpoint would be reached in approximately 38.8 seconds.

However, the SI signal has a minimal impact on the plant response. The shutoff-off head for the High Head Safety Injection pumps is approximately 1700 psig. When the block valve to the spuriously opened pressurizer PORV is closed, the depressurization will stop and the reactor coolant system (RCS) pressure will quickly return to the High Head Safety Injection pump shut-off head pressure. Bounding analysis presented in UFSAR Section 15.6.1, "Inadvertent Opening of a Pressurizer Safety or Relief Valve," shows that core damage will not occur.

In addition, the operator has sufficient time to ensure emergency core cooling system (ECCS) injection does not occur in the event of a low pressurizer pressure SI signal. Assuming the initial pressurizer pressure is 2215 psig and a depressurization rate of 6.2 psi per second, the operator would have approximately 83 seconds to secure the pressurizer block valve before ECCS injection occurs when RCS pressure decreases to 1700 psig.

During validation of operator action times in response to the off-normal procedure that requires evacuation of the control room due to a fire, operators have demonstrated the capability to close the PORV block valves in less than 30 seconds following initiation of a manual reactor trip which is the start of the transient. For this condition, control room operators would readily diagnose the initiation of a fire. Progression of the fire and the need for evacuation should be apparent so that time should be available to anticipate the need to perform the actions. Once a decision to evacuate the control room is made, the control room operator uses a procedure to direct the actions to be taken. The mitigating action to block the PORV is taken early in the sequence of actions performed in the control room prior to evacuation.

In summary, a SI may occur conservatively assuming (a) pressurizer pressure instrument and the low pressurizer pressure trip setpoint uncertainties from nominal conditions, (b) the operator takes up to 60 seconds to close the PORV block valves, and (c) the spurious actuation resulting in an open PORV occurs immediately upon initiation of the manual reactor trip. No injection from the ECCS into the RCS occurs. The initiation of the SI does not prevent the operators from intervening and closing the PORV block valves post-trip which terminates the impact of the SI. SI can be secured from outside the control room so that the capability to shutdown the plant and achieve fire safe shutdown conditions are not adversely impacted.

NRC RAI 3

Letter dated June 2, 2011 states that operators can place centrifugal charging pumps in pull to lock within 120 seconds, and a SI actuation would occur at 146 seconds, if the pressurizer auxiliary spray valve opens on reactor trip. Please demonstrate that the 26 second margin between stated operator capability and the analyzed SI injection signal receipt time is sufficient to account for reactor pre-trip statepoint uncertainties.

STPNOC Response

The South Texas Project fire hazards analysis indicates that a SI signal would be received in 146 seconds following spurious auxiliary spray valve opening coincident with reactor trip with the centrifugal charging pumps running, assuming the reactor plant is at nominal conditions prior to reactor trip. The analysis assumes the operators can secure the centrifugal charging pumps within 120 seconds of reactor trip.

The analysis assumes the transient occurs at the initiation of a manual reactor trip and that a fire-induced spurious actuation of an auxiliary spray valve occurs immediately following the reactor trip. If a SI occurs, the consequences are the same as that discussed in the response to RAI 2. As discussed in the response to RAI 2, the pre-trip parameters whose uncertainty would have a significant impact on the SI receipt time are the initial pressurizer pressure value and low pressurizer pressure SI setpoint. The depressurization rate for this event is approximately 2.6 psi per second. By subtracting the low pressurizer pressure setpoint from the initial pressurizer pressure and considering uncertainties as discussed in the response to RAI 2, the low pressurizer pressure SI setpoint would be reached in approximately 92.5 seconds.

However, as discussed in the response to RAI 2, the impact of a SI signal has a minimal impact on the plant response. The shutoff-off head for the High Head Safety Injection pumps is approximately 1700 psig. When the centrifugal charging pump is secured, the depressurization will stop and the RCS pressure will quickly return to the High Head Safety Injection pump shut-off head pressure. Bounding analysis presented in UFSAR Section 15.6.1, "Inadvertent Opening of a Pressurizer Safety or Relief Valve," shows that core damage will not occur.

In addition, the operator has sufficient time to ensure ECCS injection does not occur in the event of a low pressurizer pressure SI signal. Assuming the initial pressurizer pressure is 2215 psig and a depressurization rate of 2.6 psi per second, the operator would have approximately 198 seconds to secure the charging pumps before ECCS injection would occur when RCS pressure decreases to 1700 psig.

During validation of operator actions times in response to the off-normal procedure that required evacuation of the control room due to a fire, operators have demonstrated that the charging pumps can be placed in pull to lock in approximately one minute following initiation of a manual reactor trip which is the start of the transient. As stated in the response to RAI 2, the control room operators would readily diagnose the initiation of a fire and have time for performing the actions.

In summary, a SI may occur conservatively assuming (a) pressurizer pressure instrument and the low pressurizer pressure trip setpoint uncertainties from nominal conditions, (b) the operator takes up to 120 seconds to secure the centrifugal charging pumps, and (c) the spurious actuation resulting in an open auxiliary spray valve occurs immediately upon initiation of the manual reactor trip. No injection from the ECCS into the RCS occurs. The initiation of the SI does not prevent the operators from intervening and securing the centrifugal charging pumps post-trip which terminates the impact of the SI. SI can be secured from outside the control room so that the capability to shutdown the plant and achieve fire safe shutdown conditions are not adversely impacted.

NRC RAI 4

Letter dated August 1, 2011, states that steam generator water level will go off-scale high in 130 seconds following a reactor trip with a spurious main feedwater isolation valve opening. The letter dated June 2, 2011 states that operators have demonstrated the capability to secure startup feed pumps within 120 seconds. Please demonstrate that the 10-second difference between stated operator capability and analytic results is sufficient to account for pre-trip reactor statepoint uncertainty.

STPNOC Response

The South Texas Project fire hazards analysis indicates that steam generator water level will go off-scale in 130 seconds following spurious feedwater isolation valve opening on reactor trip with the startup feedwater pump running, assuming the reactor plant is at nominal conditions prior to reactor trip. The analysis assumes the operators can secure the startup feedwater pump within 120 seconds of reactor trip.

The startup feedwater pump will automatically start due to a main feedwater pump turbine trip when the main steam isolation valves are shut and steam pressure is lost to the turbine-driven main feedwater pumps. Isolation of main feedwater is one of the actions performed in the control

room prior to evacuation in the event of a fire. The analysis assumes the transient occurs after the initiation of a manual reactor trip and a fire-induced spurious actuation of a main feedwater isolation valve occurs after main feedwater isolation. Main feedwater isolation is one of the actions performed prior to evacuating the control room.

The pre-trip parameter that would have a significant impact on the indicated steam generator water level to go off-scale high is the initial steam generator water level. A higher initial steam generator water level would result in less time before the indicated steam generator water goes off-scale high. The analysis assumed an initial steam generator water level of 70.7%. The maximum water level instrument uncertainty is 4.6%. Therefore, the maximum initial water level assuming the steam generator water level controller uncertainty is 75.3%. The results of the analysis show that on a reactor trip, the steam generator water level will decrease to approximately 20%, when not considering uncertainties. Assuming 4.6% uncertainty on the initial steam generator water level, the resulting steam generator water level after a reactor trip would be approximately 24.6%. The results also show that the indicated steam generator water level increases at a rate of 1% per second on the steam generator with the spuriously opened main feedwater valve after the initiation of the startup feedwater pump, which is 50 seconds into the event. When considering a post reactor trip steam generator water level of 24.6% and an indicated steam generator water level fill rate of 1% per second, the steam generator with the spuriously opened main feedwater isolation valve would go off-scale high in approximately 125.4 seconds, which is greater than the 120 seconds assumed for the operator to secure the startup feedwater pump.

In summary, the analysis demonstrates that the Appendix R requirements are still met with a spurious actuation of a main feedwater isolation valve when assuming uncertainty for the initial steam generator water level.

NRC RAI 5

Letter dated June 2, 2011 appears to consider each of the credited post-trip operator actions independently. Please clarify whether operators are assessed in their capability to follow all actions in aggregate following a reactor trip due to fire, or whether the actions are assessed separately. If the actions are assessed separately, explain how the independent assessments lead to a conclusion that all operator actions can be performed within their stated time frames, when performed together.

STPNOC Response

The capability to perform the post-trip operator actions following a reactor trip is assessed in the aggregate. The validation of the operator actions are timed in sequence such that the time to perform any individual action is dependent on successfully performing the prior required actions.