T. Clarification of Prior NRC Approvals

13 Pages Attached

Introduction

The elements of the Callaway Plant current fire protection licensing basis for which specific NRC previous approval is unclear are identified in the following sections. Also provided in the following sections is sufficient detail to demonstrate how those elements of the current fire protection licensing basis meet the requirements in 10 CFR 50.48(c) (per RG 1.205, Revision 1, Regulatory Position 2.2.1).

Prior Approval Clarification Request 1

Current Licensing Basis:

The Callaway Plant licensing basis relative to the Auxiliary Shutdown Panel (ASP) design was approved in SER Supplement 3. However, SER Supplement 4 clarified that the approval in Supplement 3 was based on the assumption that no fuses would need to be replaced after the transfer switches were placed in the isolated position. Approval for the ASP and strategy were granted based on the procedures in place at the time, as well as several planned modifications.

Background/Basis:

A request for approval of the design for the ASP, and for the overall Alternate Shutdown Strategy was submitted via SNUPPS letters to the NRC dated 11/15/1982 and 8/23/1984, to establish compliance with the requirements of 10CFR50 Appendix R, Section III.G.3.

In NUREG-0830, Supplement 3, dated 05/1984, the NRC states:

"The design of the remote shutdown system complies with the performance goals outlined in Section C.5.c of BTP CMEB 9 5-1. Reactivity control is accomplished by manual scram before the operator leaves the control room and by boron addition via the chemical and volume control system using the refueling water storage tank (RWST) and the charging pumps. The reactor coolant makeup function is also performed by the charging pumps and RWST. Reactor coolant inventory is ensured by maintaining reactor coolant pump seal cooling and seal injection and by isolating all possible paths of inventory loss such as power-operated relief valves, RHR suction lines, normal and excess letdown lines, and the reactor vessel head vent. All these operations, including reactor scram, can be accomplished from outside the control room. Reactor decay heat removal to hot shutdown is accomplished by the AFW system through the steam generators and atmospheric dump valves. Decay heat removal to cold shutdown is achieved by the residual heat removal system. The following instruments on the alternate shutdown panel will be used to monitor process variables.

- *pressurizer level*
- *steam generator level (wide range)*
- *AFW flow*
- *reactor coolant system pressure (wide range)*
- *reactor coolant cold leg temperature (TC)*
- *reactor coolant hot leg temperature (TH)*
- *source range nuclear instrument*

The above instrumentation will be isolated from the control room on the train B auxiliary shutdown panel. Isolated valve position indications for the AFW system, the letdown isolation valve, and the atmospheric dump valves are also located on the train B panel.

The staff has reviewed actions required by the procedures for achieving and maintaining safe plant shutdown following a fire. For hot standby, the immediate actions are mainly precautionary measures to ensure that no spurious operations occur as a result of a control room fire. Some operations require cutting a control power cable at the equipment to ensure that a fault in the control room does not prevent certain equipment operation. Such actions may be required for the fuel oil transfer pumps, fuel pool cooling system, and some ventilation dampers that are not *immediately necessary for or detrimental to maintaining hot standby conditions. These actions will be described in the procedures. For achieving and maintaining cold shutdown, local operation of RHR isolation valves, letdown valves, and certain component cooling water system valves may be required and will be in the cold shutdown procedures. The staff has reviewed the proposed actions and staffing requirements and concludes they are in accordance with Section C.5.c of BTP CMEB 9 5-1, because they can be accomplished without fire brigade members and are straight-forward and uncomplicated so that cold shutdown can be achieved within 72 hours.*

On the basis of its review, the staff concludes that the alternative shutdown capability for the control room meets the requirements of BTP CMEB 9 5-1 and GDC 3 and is therefore acceptable."

<u>In NUREG-0830, Supplement 4, dated 10/1984, the NRC states:</u>

"However, staff conclusions reached in SER Supplement 3 were based on the understanding that it would not be necessary to replace fuses after the transfer switches were placed in the isolated position, regardless of the time frame assumed for fire damage to the control room circuits. Following the inspections, the staff recognized that the present SNUPPS design in combination with the alternate shutdown procedures did not meet staff requirements for alternative shutdown capability in the event of a control room fire assumed for fire damage to the control room circuits.

As a result of meetings with the SNUPPS utilities on August 10, 14, 15, and 22, 1984, the staff determined that new procedures could take care of many of the concerns identified by the inspection, since breakers or valves could still be operated locally. In other cases it was determined that the replacement of fuses was acceptable, since the components in question did not have an immediate effect on hot shutdown and ample time was available to replace fuses. However, there were four instances in which the licensee identified isolation switches that required modifications and five instances in which new isolation switches would have to be added. The new and modified isolation switches will have redundant fuses so that when placed in the isolation position new fuses would be switched into circuitry and the equipment would be isolated and immediately available.

By submittal dated August 23, 1984, the licensee provided a detailed outline of new alternate shutdown procedures and identified where the new and modified switches were required. The proposed new procedures consist of five phases, A through F, which will be performed by four operators. The new procedures assume that the control room is evacuated when the fire starts and operations outside the control room systematically bring all hot shutdown systems on the line and compensate for or prevent spurious operations that could affect achieving or maintaining hot shutdown. …

…On the basis of the staff review of the phased procedural approach outlined with the August 23, 1984 submittal, and the interim procedures identified for use until the installation of the five new isolation switches and the modifications to four of the existing switches, the staff concludes that the SNUPPS alternative shutdown capability is acceptable pending the following conditions:

(1) Because of the time needed to design, procure, install and test the isolation switches, the staff has decided that the Callaway licensee does not have to install the isolation switches before a full-power license is issued. The basis for this deferral is staff judgment that the interim procedures provide a level of safety comparable to the design with the modified and new isolation switches for the time period of the first operating cycle.

(2) Before exceeding 5% of rated power, the licensee will revise his procedures for responding to a fire in the control room in accordance with the licensee's submittal of August 23, 1984 and will train operators to the revised procedures, including the interim procedures.

(3) In addition, the staff will condition the license to require the licensee to install the five new isolation switches and modify the four existing isolation switches that were identified in the August 23, 1984 submittal:

> *(a) Before startup following the first extended outage of known duration (greater than two weeks) occurring after February 15, 1985, or*
>
> *(b) Before startup following the first refueling outage.*

If the full-power license is not issued before March 1, 1985, the staff will require that the new isolation switches be installed and existing isolation switches be modified before exceeding 5% of rated power."

Activation of the ASP involves the transfer of control from the Main Control Room to the ASP (RP118B) through an operator action to manually position three isolation transfer switches and five control switches which are located on RP118B. Following activation of the ASP, the plant operator is provided with the capability to control and monitor secondary side Decay Heat Removal capability utilizing the Auxiliary Feedwater System, the capability to control Reactor Coolant System pressure, and the capability to monitor critical Reactor Coolant System process parameters which are necessary to verify that natural circulation has been established in the RCS and that it is being successfully maintained thereafter.

The NRC approved design for Auxiliary Shutdown Panel RP118B includes the following specific components and features:

- Steam Generator B (2) pressure indication (ABPIC0002B)
- Steam Generator B (2) wide range level indication (AELI0502A)
- Steam Generator B (2) AFW flow indication (ALFI0003B)
- Open control for Steam Supply valve from Steam Generator B (2) to TDAFP (ABHV0005)
- Open and close control for Steam Generator B (2) Atmospheric Steam Dump Valve (ABPV0002)
- Open and close control for Steam Generator B (2) AFW flow control valve from TDAFP (ALHV0010)
- Open and close control for Essential Service Water to suction of MDAFW Pump B (ALHV0030)
- Open and close control for Condensate Storage Tank to suction of MDAFW Pump B (ALHV0034)
- MDAFW Pump B suction pressure indication (ALPI0024B)
- Trip and close control for MDAFW Pump B breaker (NB0205)
- Steam Generator D (4) pressure indication (ABPIC0004B)
- Steam Generator D (4) wide range level indication (AELI0504A)
- Steam Generator D (4) AFW flow indication from MDAFW Pump B (ALFI0001B)

- Open and close control for Steam Generator D (4) Atmospheric Steam Dump Valve (ABPV0004)
- Open and close control for Steam Generator D (4) AFW flow control valve from MDAFW Pump B (ALHV0005)
- Open and close control for Essential Service Water to suction of TDAFP (ALHV0033)
- TDAFP suction pressure indication (ALPI0026B)
- Open and close control for TDAFP Governor Control valve (FCFV0313)
- Open and close control for TDAFP Trip and Throttle valve (FCHV0312)
- Pressurizer level indication (BBLI0460B)
- Reactor Coolant System pressure indication (BBPI0406X)
- Reactor Coolant System Loop 2 cold leg temperature indication (BBTI0423X)
- Reactor Coolant System Loop 4 hot leg temperature indication (BBTI0443A)
- Intermediate and source range neutron monitoring indication (SENI0061X and SENI0061Y)
- Trip and close control for Pressurizer Backup Group B breaker (PG2201)

Request

As part of this LAR submittal and transition to NFPA 805, it is requested that the NRC formally document as “prior approval” the physical design and capabilities for Auxiliary Shutdown Panel RP118B, including the specific components and features cited above.

The “phased procedural approach” that is discussed in the SER Supplement 4 approval has been revised as part of the NFPA 805 transition. Ameren Missouri seeks only to maintain the approval of the original design of the ASP and its physical capabilities. The NSCA has been performed under the transition to NFPA 805 and will be submitted separately for NRC approval.

Prior Approval Clarification Request 2

<u>Current Licensing Basis:</u>

A deviation was granted for non-fire rated penetrations in the reactor containment walls. The deviation did not specifically identify approval of all three containment hatches.

<u>Background/Basis:</u>

A request for deviation from Section C.5.b of Appendix A to BTP ASB 9.5-1 for Containment Penetrations was submitted with a specific analysis of the Personnel Hatch penetrating the Reactor Building into Fire Area A-20.

<u>In "SNUPPS Fire Hazard Analysis," Rev. 14, transmitted to NRC 3/14/1984. Callaway Plant states:</u>

"9.5.1.2.2.3 Fire Barriers

REACTOR BUILDING PENETRATIONS- The reactor building shell is 4 feet thick, is lined with a continuous ¼-inch thick liner plate, and is designed to be airtight at a design pressure of 60 psig. All penetrations through the containment shell are designed to ASME Section III. Since the penetrations are an integral part of the reactor building boundary and do not incorporate independent fire barrier seals, no testing was performed to establish a rating of the penetrations. The following discussions describe the reactor building penetrations into adjacent buildings…

PERSONNEL HATCH- The personnel hatch, shown on Figure 3.8-45, penetrates the reactor building into Fire Area A-20. The hatch has two bulk head doors on either side of the reactor building wall which are secured by multiple pin latches. The gap between the door and the bulk heads is sealed by double-o-ring gaskets. When closed, the gap between the door and the bulk head is less than 5 mills. The bulk heads and hatch doors are in series and provide redundant fire barrier protection."

<u>In NUREG-0830, Supplement 3, dated 05/1984, the NRC states:</u>

"Containment Penetrations

The reactor containment walls are penetrated by numerous mechanical and electrical penetrations, as well as by personnel hatch and a fuel transfer tube.

The containment wall is 4-foot-thick reinforced concrete with a continuous 1/4-inch-thick steel liner. The construction is capable of withstanding a 60-psig overpressure without failure.

Because of the construction of the containment wall and the special nuclear safety-related purposes these penetrations serve, the staff considers that they provide a level of safety equivalent to the technical requirements of Section C.5.b of BTP CMEB 9.5-1."

Although the personnel hatch to Fire Area A-20 was specifically called out in the approval, the containment emergency personnel and equipment hatchways to the yard, Fire Area YD-1, were not called out in the SER although these hatches should have been considered to be approved under this deviation. The emergency personnel hatchway is of identical construction to the personnel hatch to Fire Area A-20. The equipment hatch, while not identical, is equally robustly constructed, consisting of a welded steel assembly with a double gasketed, flanged, and bolted cover and provided with a moveable missile shield on the outside of the Reactor Building. The hatches are located as described below:

1. The Emergency Personnel Hatch is provided for evacuation purposes at El. 2013'. The hatch opens to an enclosed stairwell (Room 2202) leading to the outside grade elevation that is separated by a 3-hour barrier from the Reactor Building and contains no equipment.
2. The Equipment Hatch opens to the Yard fire area outdoors and is located on the refueling floor elevation 2047'.

Request

As part of this LAR submittal and approval it is requested that the NRC formally document as “prior approval” that the Emergency Personnel Hatch and the Equipment Hatch in the Reactor Building/Containment walls are acceptable as installed based on the general text of SER Supplement 3 regarding containment penetrations.

Prior Approval Clarification Request 3

Current Licensing Basis:

Callaway Plant credits the use of an NRC approved manually charged fire suppression system within the Reactor Building as equivalent to an automatic system.

Background/Basis:

Approval was granted for the Reactor Building fire suppression system based on the submitted analysis. Since approval, some containment fire protection features have been modified via the standard license condition allowance.

In "SNUPPS Fire Hazard Analysis," Rev. 4, transmitted to NRC 6/29/1981. Callaway Plant states:

"A fixed, manually charged, closed head sprinkler system is provided over the cable trays in Zones RB-3 and RB-4. The design density for the system is 0.3 gpm/square feet of floor area. The pressure and flow rate are based on all heads open in the most remote 1,000 square feet. A manual system is installed to prevent an inadvertent actuation of the sprinklers during normal plant operation."

In NUREG-0830, dated 10/1981, the NRC states:

"The reactor building is separated from adjacent buildings by 3-hr fire barriers.

There are no physical boundaries enclosing localized fire hazards within the reactor building.

An automatic detector is installed above each reactor coolant pump. Line type thermal detectors are also installed in all areas where cable trays are concentrated and ionization type detectors are installed in the containment cooler ducts. The automatic detection system alarms are zoned in the control room.

Portable extinguishers and manual hose stations are permanently installed inside containment. At the request of the staff, the applicant agreed to install two additional hose station so that every hose station will be spaced no more than 100 ft. from an adjacent hose station.

*A fixed, manually charged closed head sprinkler system is provided over the two cable tray penetration areas. The design density for the system is 0.3 gpm/ft**2. The pressure and flow rate are based on all heads open in the most remote 1000 ft**2.*

Based on its review, the staff concludes that the fire protection for the containment meets the guidelines of Appendix A to BTP ASB 9.5-1 and is, therefore, acceptable."

Since approval of the manual fire protection system in lieu of an automatic system in the Reactor Building Engineering Equivalency Evaluations RFR 201103067, RFR 3510C and RFR 201007840 have made minor revisions to the configuration. These changes are summarized by EEE below:

- RFR 201103067: Fire Hose is not installed at each hose outlet in the Reactor Building during power operations. Hoses are staged at the Reactor Building personnel hatch for fire brigade use as needed. This deviation was found acceptable based on the fact that the Reactor Building is not normally occupied during these operational periods and that the hoses are accessible for use by the fire brigade on entry to the Reactor Building if needed in a fire event.

- RFR 201007840: Portable fire extinguishers are not installed inside the Reactor Building during power operations. The EEEE was approved to eliminate any seismic concerns associated with their mounting. Five extinguishers have been provided just outside the personnel hatch and are available to take into the Reactor Building should a fire occur during normal operation. During refueling and maintenance outages, a full complement of fire extinguishers is staged throughout the Reactor Building to assist in fighting incipient stage fires. The extinguishers are installed as soon as practical following entering MODE 5, when descending in power, and removed as late as practical in MODE 5, when ascending in power. The configuration was evaluated and judged to provide adequate protection and safety measures. The Reactor Building is not occupied during normal operations; therefore, removal of extinguishers during normal operations does not affect the availability of extinguishers to plant personnel in the event of a fire in that area. Extinguishers provided outside the personnel hatch provide an equivalent level of protection. NFPA 805 Section 3.7 specifically states that “extinguishers shall be permitted to be positioned outside of fire areas due to radiological conditions.” Therefore, the removal of extinguishers from the Reactor Building during power operations is allowed under NFPA 805.
- RFR 3510C: This evaluation approved locking open the emergency release valves to fire protection deluge valves KCXV0261 and KCXV0262. The evaluation determined that locking open the valves would remove the risk that an electrical failure would render the system inoperable. Since the system is a manual pre-action system activated via motor-operated containment isolation valve KCHV0253 at the standpipe, the open emergency release valves will not affect the system function.

Request

As part of this LAR submittal and transition to NFPA 805, it is requested that the NRC formally document as a “prior approval” the manual fire protection system in the Reactor Building in lieu of an automatic system to meet the deterministic requirements of NFPA 805 Section 4.2.3.4 (c). The changes to the Reactor Building fire protection features were completed within the guidelines of the fire protection license condition and do not adversely affect the overall level of fire protection in the Reactor Building.

Prior Approval Clarification Request 4

Current Licensing Basis:

The Callaway Plant licensing basis relative to Section C.7.i of Appendix A to BTP ASB 9.5-1 for Emergency Diesel Generator Day Tank Dike Configuration was approved in SER Supplement 3.

Background/Basis:

A request for deviation was submitted per 2/1/1984 SNUPPS letter to the NRC SLNRC 84-0014, justifying the diesel fuel oil day tank containment dikes.

The 2/1/84 SNUPPS submittal letter states:

"The day tanks are located above a diked area with a free volume of greater than 110% of the tank volume. The diked area is provided with a floor drain which drains to a sump within the room."

The analysis also cited the following bases for acceptability of the overall system design:

- *"The area adjacent to the day tank contains no hot surfaces or ignition sources. Any fuel oil on the general floor area will enter the floor drain system and be routed to the sump. Duplex sump pumps are provided to evacuate the sump. The nearest floor drain is approximately 10' outside of the dike."*
- *"The day tank and all piping associated with the pressure boundary is Seismic Category I and not postulated to fail due to an earthquake. Also, following an accident, no passive piping failures are postulated in accordance with current licensing practices. The day tanks are unpressurized tanks vented to the outdoors via piping equipped with flame arrestors. The NRC previously questioned (Q430.12) the design of the fuel oil piping from the day tank to the diesel engine and accepted the response provided…"*
- *"The day tank is also provided with Class 1E level indication which alarms in the control room when the 4" stand pipe volume/level decreases by less than 3 gallons when the diesel is not operating; therefore, any loss in fuel oil would be readily detected during normal plant operating."*
- *"Also, security personnel make tours of all safety related areas during each shift."*
- *"Diesel generator testing is conducted from the control panel within the diesel room. Any leakage occurring during diesel operation would be detected by test personnel."*
- *"The potential for missile generation by an operating diesel was similarly questioned (Q430.8). The response provided by the diesel manufacturer was accepted by the NRC staff. Refer to FSAR page 430.8-1, attached. In addition, the NRC staff has previously requested additional information regarding the design of the day tank dike and has indicated that the design is adequate."*

In NUREG-0830, Supplement 3, dated 05/1984, the NRC states:

"The diesel fuel oil day tanks are located in each diesel generator room. The SER states that a containment dike would be provided beneath each day tank to contain 100% of the fuel oil, however, during its visit, the staff noted that the top of the dike is beneath the tank. The staff was concerned that not all leaks would be contained by this configuration and that the applicant should modify the dike to provide more positive collection ability (such as by completely surrounding the day tank) in accordance with Section C.7.i of BTP CMEB 9.5-1.

By letter dated February 1, 1984, the applicant indicated that the existing fuel tank and all piping are seismic Category I. The fuel oil system is a gravity-feed-type system, therefore, no pressurized sprays will occur as a result of a leak. The floor area adjacent to the dike has floor drains. The day tank is provided with level indication that alarms in the control room if there are more than 3 gallons of leakage.

The applicant considers that the current design of the tank is adequate and, on the basis of the information provided, the staff agrees. If any leaks should occur, they would be promptly detected, and the floor drains would collect the majority of the leakage.

On the basis of its review, the staff concludes that the diesel fuel day tank and dike assembly meets the guidelines in Section C.7.i of BTP CMEB 9.5-1, and is, therefore, acceptable."

Subsequent to the NRC approval it was determined that the actual capacity of the diesel day tank dike was found to be less than the 110% cited in the analysis and the 100% cited in the SER.

Request

As part of this LAR submittal and transition to NFPA 805, it is requested that the NRC formally document as "prior approval" the current design configuration of the two diesel generator day tanks.

The original NRC approval was granted based on the overall design of the fire protection features in the rooms and did not specifically rely on the dike capacity. Therefore, although the day tank dike capacity is less than originally stated, the gravity fed system design; drainage and level indication are still applicable. The reduction in dike capacity is not considered to affect the overall performance of the configuration in the event of a leak.

Prior Approval Clarification Request 5

<u>Current Licensing Basis:</u>

The plant-wide manual standpipe and hose system was approved by the NRC. The approval did not specifically identify that the hose stations in the ESW Pumphouse are supplied with water from the ESW system, not the fire protection water system.

<u>Background/Basis:</u>

During its initial review of the SNUPPS FP design the NRC submitted comments on the design via an NRC to SNUPPS letter dated 10-18-1979. This letter from Olin Parr and the responses became FSAR Appendix 9.5D. Item 6 of this letter identifies that the sole use of fire extinguishers as a backup means of manual suppression is unacceptable and that hose stations should be added to the areas where extinguishers are the sole back up. Item 12 of the letter specifically identified that the ESW areas needed to have hose stations added to the design.

<u>In NRC Letter dated October 18, 1979, from Olan Parr to Mr. J. Bryan (Union Electric), et al, the NRC states:</u>

"6. Page 9.5-6a (Safety Evaluation Two)

You state that in most areas of high fire loading, a backup system will be available in case of failure of the primary suppression system in a given area. However, the backup system is a portable extinguisher or a hose station. It is our position, as stated in Section E.3(d) of Appendix A, that portable extinguishers, due to their limited capacity and effectiveness, are not considered as secondary protection. Hose stations should be provided so that all areas of the plant can be properly protected. Revise your design accordingly…

12. Page 9.5A-41

It is our position, as stated in Section F11 of Appendix A, that both early warning fire detection with alarm and annunciation and hose stations be provided for protection of the essential service water pumphouse in addition to the proposed fire extinguishers. Revise your design accordingly."

Callaway developed a point by point response to the Olin Parr letter which was placed in FSAR Appendix 9.5D and FSAR Addendum Appendix 9.5D during conduct of the ongoing NRC review of the Fire Protection program. FSAR Appendix 9.5D was included in SNUPPS FSAR Rev 1 issued to the NRC via SLNRC 80-42 dated 9/17/80. The site specific FSAR Addendum, Appendix 9.5D was included in FSAR Site Addendum Rev. 1 issued to the NRC via ULNRC-00383 dated 9/18/80. Both of these submittals predate the original Callaway SER dated 10/81 which indicates this information was included within the scope of the original Fire Protection Program being reviewed.

In Item 6 of Appendix 9.5-D of Revision 1 of the FSAR Site Addendum, in response to the Olan Parr Letter dated October 18, 1979, and submitted to the NRC in Letter ULNRC-00383 from John Bryan (UE) to Denton (NRC) dated September 18, 1980, Union Electric states:

"The only safety-related building which was not covered by hose stations was the Essential Service Water (ESW) System pumphouse. We have modified our design by tapping into the ESW system to supply hose stations at the ESW pumphouse.

The ESW lines are normally pressurized by the plant Service Water System and under emergency conditions by the ESW pumps. The use of these hose stations does not impair the ability of the ESW system to perform its intended function.

The combination of automatic fire detection and annunciation to the control room, hose stations, and portable fire extinguishers provides adequate fire protection for the ESW pumphouse."

The original site SER was issued October 1981 and it approved the standpipe system as installed, Section 9.5.1 of the SER states that the SER is based on NRC review of applicant letters dated April 15, 1977 and the revised Fire Hazards Analysis dated June 29, 1981. As noted above the FHA dated June 29, 1981 contained the description of the ESW hose stations.

In Section 9.5.1.1 of NUREG 0830, NRC SER dated October 1981, the NRC states:

"Manual hose stations are located throughout the plant to ensure that an effective hose stream can be directed to any safety-related area in the plant. The standpipes are consistent with the requirements of NFPA 14, 'Standard for the Installation of Standpipe and Hose Systems.' Standpipes are 4- and 2-1/2-in. diameter pipe for multiple and single hose station supplies, respectively, Based on this evaluation, the staff concludes that the sprinkler and standpipe systems are adequate, meet the guidelines of Appendix A, Sections C.3.a and C.3.d, and are, therefore acceptable."

Request

As part of this LAR submittal and transition to NFPA 805, it is requested that the NRC formally document as "prior approval" that the hose stations in the ESW Pump house are supplied with water from the ESW system, not the fire protection water system. The design of the ESW system was identified as a deficiency by the NRC during its initial review and the Callaway Plant response contained documented in FSAR Section 9.5D identified that a hose station fed from the ESW water supply was provided. Subsequent to that the NRC issued the original SER approving the overall design of the Callaway Plant hose stations as consistent with NFPA 14.