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SUBJECT: Confirms discussions at meeting held on 941115 at NRR headquarters re rev to procedures for control room fires.

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January 25, 1995
GO2-95-013

Docket No. 50-397

US Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Gentlemen:

Subject: **WNP-2, OPERATING LICENSE NPF-21**
REVISION TO PROCEDURES FOR CONTROL ROOM FIRES

References:

1) Letter G02-94-163, dated July 15, 1994, JV Parrish (SS) to NRC, "WNP-2, Operating License NPF-21, Post Fire Safe Shutdown Procedures"

2) Letter dated May 25, 1989, RB Samworth (NRC) to GC Sorenson (SS), "Issuance of Amendment No. 67 To Facility Operating License No. NPF-21 - WPPSS Nuclear Project No. 2 (TAC No. 64655)"

3) Letter GI2-94-149, dated June 2, 1994, JW Clifford, (NRC) to JV Parrish (SS), "Closeout of Generic Letter 89-19, Request for Action Related to Resolution of Unresolved Safety Issue A-47 Safety Implication of Control Systems in LWR Nuclear Power Plants Pursuant to 10 CFR 50.54(f), for WNP-2 (TAC No. M75019)"

4) Letter GI2-86-0089, dated December 4, 1986, RM Bernero (NRC) to Dennis Kirsch (NRC), "Evaluation of WNP-2 Fire Protection Analysis"

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5) Letter dated June 1, 1984, CO Thomas (NRC) to JF Quirk (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-23785, Revision 1, Volume III (P), The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident"

6) Letter GI2-94-324, dated November 9, 1994, TP Gwynn (NRC) to JV Parrish (SS), "NRC Inspection Report 50-397/94-28 (Notice of Violation)"

This letter confirms the discussions between the Supply System and staff at a meeting held on November 15, 1994 at NRR headquarters and is being submitted to:

- Request approval for a second credited Control Room operator action prior to evacuation due to a fire, and to
- Provide information defining final resolution of safe shutdown issues previously discussed in Reference 1.

Section 3.8.4 of Generic Letter 86-10 indicates that Control Room operator actions beyond reactor trip would be acceptable provided that such actions can be performed prior to Control Room evacuation in a fire situation and could not be negated by subsequent fire-induced spurious equipment actuations. Consistent with that guidance, WNP-2 is requesting approval for one additional operator action as outlined in Attachment 1. Your review and approval of the second credited Control Room operator action is requested by April 15, 1995 to support our Spring 1995 refueling outage schedule. If approval is not received by that date, interim operator actions described in Reference 1 will be maintained, pending final resolution.

A tabulated comparison of the current and future Control Room procedural response is presented in Attachment 2 to provide an overall summary of the procedural changes to be made to resolve the interim actions discussed in Reference 1. Pre-staging operators at remote locations is currently part of the response to a Control Room fire. WNP-2 has reviewed plant design and determined that pre-staging can be eliminated, as discussed in Attachments 3 and 4. WNP-2 is adopting a revised reactor shutdown analysis as the basis for action in response to a Control Room fire. The analysis is summarized in Attachment 5 together with a discussion of procedural implications. No staff action is requested with respect to information presented in Attachments 2, 3, 4, and 5.

Implementation of the actions outlined in this submittal will be based on the fire protection licensing basis applicable to WNP-2. Specifically, the fire protection program at WNP-2 is based on Operating License Condition 2.C (14), as amended by Reference (2).

Should you have any questions or desire additional information regarding this matter, please call me or D.A. Swank, Licensing Manager, at (509) 377-4563.

Sincerely,

J. V. Parrish (Mail Drop 1023)
Vice-President, Nuclear Operations

CJF/ml
Attachments

cc: LJ Callan - NRC RIV
KE Perkins, Jr. - NRC RIV, Walnut Creek Field Office
NS Reynolds - Winston & Strawn
JW Clifford - NRC
DL Williams - BPA/399
NRC Sr. Resident Inspector - 927N

NEW ADDITIONAL OPERATOR ACTION

We are requesting approval to credit one additional immediate operator action to be taken in response to a Control Room fire: manual closure of the Main Steam Isolation Valves (MSIVs). This action is to be taken immediately after the reactor is manually scrammed. It is currently included in the procedure only as an action to be taken if the situation permits.

MSIV closure limits inventory loss to the condenser, terminates steam flow to the feedwater pump drive turbines, and eliminates high-low pressure interfaces in piping systems external to the containment. MSIV closure will be effected at Control Room Panel H13-P601 by depressing the four Nuclear Steam Supply Shutoff System (NSSSS) Isolation Logic push buttons. This will interrupt power to the NSSSS logic circuits, allowing venting of compressed air from the actuators of all eight MSIVs, and resulting in automatic closure of the MSIVs. A closure of the MSIVs cannot be reversed by a Control Room fire, as discussed in Attachment 3. Control Panel H13-P603 is approximately 20 feet from Panel H13-P601 where the first action is performed. Consequently, the incremental time between initiating SCRAM and closing the MSIVs is quite small, perhaps 30 seconds, assuming that one operator performs the entire sequence.

It should be noted that WNP-2 Technical Specification 6.2.2 requires that three licensed operators be in the Control Room on a continuous basis during power operation: the Control Room Supervisor and two licensed Reactor Operators. Therefore, considering all actions as being performed by a single operator is very conservative. Generic letter 86-10 allows additional operator actions if such can realistically be performed.

Arizona Public Service Palo Verde plants 1, 2, and 3 and Duke Power McGuire plants 1 and 2 have received approval for more than one operator action in event of a Control Room fire. These plants close the MSIVs, trip the feedwater/reactor coolant pumps, and trip the main turbine in addition to scramming the reactor. Consequently, our request for one additional operator action is consistent with multiple actions previously approved by the staff.

COMPARISON OF OPERATOR RESPONSE ACTIONS PRESENTED FOR NRC STAFF INFORMATION

Currently, there are three plant procedures governing response to fires: PPM 4.12.1.1, "Control Room Evacuation and Remote Cooldown"; PPM 4.12.4.1, "Fires"; and PPM 5.7.1, "Appendix R Fires Outside the Main Control Room". After changes outlined in this letter are implemented, PPMs 4.12.1.1 and 4.12.4.1 will be changed as indicated below. PPM 5.7.1 will then be eliminated because the changes will make it unnecessary.

CURRENT SEQUENCE FOR CONTROL ROOM FIRE	COMPARISON	NEW SEQUENCE FOR CONTROL ROOM FIRE
Immediate Action		**Immediate Action**
Per PPM 4.12.4.1, prestage a Control Room Operator (CRO) to the remote shutdown rooms to transfer control of RHR valves 24B, 27B, and FCV-64B to remote location. CRO then to proceed to B RPS M-G set room to await instructions to trip RPS EPA circuit breakers.	The new sequence will not include prestaging to transfer control of RHR valves because design changes discussed in Attachment 4 eliminate the potential for spurious actuation of those valves in the period between evacuation of the Control Room due to a fire and acquisition of plant control from remote panels. The new sequence will not include tripping the EPA breakers per discussion in Attachment 3.	No equivalent action.
Per 4.12.4.1, prestage the Radwaste CRO to feedwater pump control panels in T-G building to await instructions to trip both feedwater pumps.	The new sequence will not include feedwater pump tripping per the discussion in Attachment 3.	No equivalent action.

CURRENT SEQUENCE FOR CONTROL ROOM FIRE	COMPARISON	NEW SEQUENCE FOR CONTROL ROOM FIRE
Immediate Action (Cont'd)		Immediate Action (Cont'd)
Per PPM 4.12.1.1, SCRAM the Reactor by depressing all 4 manual SCRAM push buttons, and locking the Mode Switch in SHUTDOWN position. Verify control rods insert and APRM indications go downscale.	New sequence action is same as currently outlined in PPM4.12.1.1.	No change.
	This is the additional immediate operator action for which approval is being requested. It is currently included in PPM 4.12.1.1 as a subsequent action. This action will automatically close the eight MSIVs.	Manually close MSIVs by arming and depressing the four MSIV isolation logic push buttons on Panel P601.
Per PPM 4.12.1.1, request security doors in Radwaste, Diesel Generator, Reactor, and both SW pumphouse buildings be unlocked.	New sequence action is same as currently outlined in PPM 4.12.1.1.	No change.

CURRENT SEQUENCE FOR CONTROL ROOM FIRE	COMPARISON	NEW SEQUENCE FOR CONTROL ROOM FIRE
Immediate Action (Cont'd)		Immediate Action (Cont'd)
Per PPM 4.12.1.1, use the PA and radio systems to announce SCRAM and Control Room evacuation, request RPS EPA breakers be tripped, request local trip of both Feedwater Pumps, and request local start of DG-2.	In the new sequence the action is the same as currently outlined in PPM 4.12.1.1 except feedwater pumps and EPA circuit breakers will not be tripped per the discussion in Attachment 3.	Use the PA and radio systems to announce SCRAM and Control Room evacuation, and request local start of DG-2.
Per PPM 4.12.1.1, evacuate Control Room (if situation requires it).	New sequence action is the same as currently outlined in PPM 4.12.1.1.	No changes.

CURRENT SEQUENCE FOR CONTROL ROOM FIRE	COMPARISON	NEW SEQUENCE FOR CONTROL ROOM FIRE
Subsequent Control Room Operator Action if Situation Permits		**Subsequent Control Room Operation Action if Situation Permits**
Per PPM 4.12.1.1, manually close MSIVs by arming and depressing all four MSIV Isolation Logic push buttons and placing the control switches in CLOSED position.	In new sequence, MSIV closure is performed as an Immediate Operator Action (see above).	Performed earlier in sequence.
Per PPM 4.12.1.1, verify valves MS-V-16 AND MS-V-19 are closed.	The new sequence action is the same as currently outlined in PPM 4.12.1.1.	No change.
Per PPM 4.12.1.1, stop the condensate and condensate booster pumps.	The new sequence does not include stopping these pumps per the discussion in Attachment 3.	No equivalent action.
Per PPM 4.12.1.1, manually initiate RCIC.	The new sequence action is the same as currently outlined in PPM 4.12.1.1.	No change.
Per PPM 4.12.1.1, start both Service Water pumps.	The new sequence action is the same as currently outlined in PPM 4.12.1.1.	No change.
Per PPM 4.12.1.1, stop RWCU blowdown by closing valve RWCU-FCV-33.	The new sequence action is the same as currently outlined in PPM 4.12.1.1.	No change.
Per PPM 4.12.1.1, trip the Main Generator.	The new sequence action is the same as currently outlined in PPM 4.12.1.1.	No change.
Per PPM 4.12.1.1, transfer divisional buses SM-7 and SM-8 to Backup Transformer TR-B.	The new sequence action is the same as currently outlined in PPM 4.12.1.1.	No change.
Per PPM 4.12.1.1, classify the emergency as an ALERT.	The new sequence action is the same as currently outlined in PPM 4.12.1.1.	No change.
Per PPM 4.12.1.1, station guards at entry doors to vital areas and Control Room.	The new sequence action is the same as currently outlined in PPM 4.12.1.1.	No change.
Per PPM 4.12.1.1, evacuate the Control Room (if situation requires it).	The new sequence action is the same as currently outlined in PPM 4.12.1.1.	No change.

ELIMINATION OF OPERATOR PRE-STAGING

This attachment is presented for NRC staff information; no staff action is being requested.

The initial response to Control Room fires will not require pre-staging operators at remote locations. Pre-staging will no longer be necessary for RHR system control after the design changes described in Attachment 4 are implemented, thus eliminating vulnerability of some control room circuits to spurious actuations. Additionally, operator prestaging will not be necessary to trip the RPS EPA circuit breakers to prevent possible spurious opening of MSIVs after closure, or to trip feedwater pumps from a remote location.

- MSIV/EPA Breakers

Plant design embodies a normally energized one-out-of-two taken twice logic to control the position of the MSIVs. De-energization of the four logic channels releases compressed air holding the MSIVs open against the MSIV actuator spring forces. Tripping the EPA circuit breakers would have the effect of de-energizing these logic channels. However, closure of the MSIVs from Panel P601 by depressing the isolation logic push buttons also results in de-energization of the four logic channels. The design is such that spurious energization of one of the logic channels would not cause the MSIVs to reopen. Reset of multiple logic channels by way of hot shorts is extremely unlikely particularly since the circuits are routed in grounded raceways. Consequently, spurious opening of the MSIVs after closure by the method outlined above is not credible, making it unnecessary to trip the RPS EPA circuit breakers.

- Feedwater Pumps

Reference 1 indicates that operators will be pre-staged at the feedwater pumps to trip the pumps locally because of the concern that fire damage to control circuitry could result in a reactor overfill condition. However, closure of the MSIVs, as discussed in Attachment 1, will interrupt steam to the feedwater pump drive turbines. Consequently, the pumps will be coasting down after MSIV closure as energy stored in the system external to Primary Containment is exhausted. Feedwater pump operation will effectively cease a few minutes after MSIV isolation minimizing the potential of reactor overfill.

In the current design configuration, actuation of the overfill protection logic involves de-energization of the overfill protection logic circuits, automatically de-pressurizing the feedwater turbine control oil systems and thereby tripping the feedwater turbines, causing the pumps to stop. Reactor overfill protection is provided through a normally energized 2-out-of-3 logic. Each of the three logic channels includes a separate reactor water level sensor located outside of the Control Room. Should an actual high water level be reached in the reactor, these level sensors will provide trip signals to the corresponding logic channel in the Control Room. One set of logic channels is located in Control Room Panel H13-P613. The other two sets are located in Control Panel H13-P612. These logic channels are powered by either direct or alternating current, and are fused so that a short to ground or power supply negative would cause a feedwater trip. Loss of power to any logic channel will trip that particular channel. A high impedance short to any logic channel will not disable overfill protection because the

other two logic channels would be unaffected, maintaining the 2-out-of-3 logic underlying overfill protection. Actuation of the logic results in energization of a two conductor cable routed through raceways under the Control Room floor connecting panels H13-P613 and H13-P612 from which the circuit exits the Control Room. The total length of this cable is approximately 50 feet routed through different raceways. All raceways under the Control Room floor are equipped with ionization and thermal type fire detectors which, if triggered, will cause Halon to be released to the affected under-floor raceway. Since the under-floor cabling is short and is routed through protected raceways, it is very unlikely that a Control Room fire could defeat reactor overfill protection.

This assessment is consistent with the conclusions of Reference (3), which stated that it is highly unlikely that a loss of power event or a fire affecting the feedwater control circuitry could defeat reactor overfill protection. NRC staff findings are that upgrading BWRs with existing automatic reactor vessel overfill protection to the separation and independence criteria identified in GL 89-19 is not warranted on a cost/safety-benefit basis, provided that (1) automatic overfill provisions are periodically tested, and (2) operators are trained to be able to mitigate overfill events that may occur via condensate booster pumps during reduced system pressure operation. WNP-2 meets these conditions. Consequently, our procedure will no longer require prestaging an operator to locally trip the feedwater pumps for Control Room fires. Subsequent operator action to trip the condensate and condensate booster pumps from the Control Room will also be eliminated.

PLANT HARDWARE CHANGES

This attachment is presented for NRC staff information; no staff action is being requested.

Residual Heat Removal (RHR) System

As discussed previously in Reference 1, evaluation of plant design established that a Control Room fire could cause spurious opening of either of two RHR system valves: one intended to be used to control flow to the suppression pool spray header, and the other used to control RHR flow for suppression pool cooling and RHR system operability testing. Plant design will be changed during the next refueling outage, currently scheduled for April 1995, to eliminate this potential for spurious response of these RHR system valves. After the change is implemented, it will no longer be necessary to pre-stage operators at the Remote Shutdown Panel for immediate transfer of RHR B loop control from the Control Room. This will be a change from the actions outlined in Reference 1.

Switches located in Control Room panel H13-P601 providing the capability for control of motor operated valves RHR-V-24B and RHR-V-27B will be protected such that a panel fire could not damage either switch prior to transfer to control at the remote shutdown panels. Additionally, conductors from these switches will be re-routed to eliminate the potential for fire-induced spurious actuation of the valve(s).

A switch located in Control Room panel H13-P601, controlling minimum flow valve RHR-FCV-64B, will also be protected such that it could not be damaged by a local fire prior to transfer to control at the remote shutdown panel. Logic relays which provide the automatic control function for valve RHR-FCV-64B will be relocated from Control Room Panel H13-P618 to the Remote Shutdown Panel. This will eliminate the possibility of a Control Room fire causing either spurious closure or failure-to-open of valve RHR-FCV-64B and potential RHR pump damage.

Main Steam Relief Valve (MSRV) Control Change

WNP-2 design includes a total of 18 MSRVs, designed to open automatically in relief mode when reactor pressure reaches the relief setpoint level, or in safety mode if reactor pressure reached the somewhat higher safety setpoint level. Seven of the 18 MSRVs are included in the Automatic Depressurization System (ADS), and are designed to automatically depressurize the reactor as an Emergency Core Cooling System response to a small-break LOCA. Switches on Control Room panel H13-P601 allow operators to open one or more of the 18 MSRVs at any reactor pressure. The seven ADS MSRVs may also be opened by operators using switches located on Control Room panels H13-P628 and H13-P631. Three of the ADS MSRVs and three of the non-ADS MSRVs may also be opened by switches located on the remote shutdown panels after transfer switches also located on the panels are actuated.

Each of the 18 MSRVs is equipped with a 1.3 cubic foot pneumatic accumulator located inside of primary containment, supplied with nitrogen from a supply external to primary containment through valve CIA-V-20. Actuation of the MSRVs in either relief or manual mode involves use of the compressed nitrogen stored in the 1.3 cubic foot accumulators.

The seven ADS MSRVs each have an additional 4.3 cubic foot accumulator located inside primary containment, serviced by nitrogen from a supply external to primary containment. Actuation in ADS mode will utilize gas only from the 4.3 cubic foot accumulator, because valving inside of containment separates the relief and ADS nitrogen supplies of the seven MSRVs.

Evaluation revealed that a Control Room fire could cause a spurious closure of CIA-V-20, thereby isolating the 1.3 cubic foot non-ADS accumulators inside primary containment from the compressed nitrogen supply external to primary containment. Cyclic actuation of the MSRVs in relief mode, which may occur during a Control Room fire scenario, would quickly deplete the isolated accumulators, resulting in temporary closure of the MSRVs. Subsequently, reactor pressure would rise to reopen the MSRVs in safety mode as assumed in the reactor shutdown analysis outlined in Attachment 5. While reactor overpressure would thereby be prevented, the lack of available compressed nitrogen in the 1.3 cubic foot accumulators would prevent manual opening of the three non-ADS MSRVs from the Alternate Remote Shutdown Panel, as was originally planned as part of the safe shutdown process for a Control Room fire. Reactor depressurization could be effected from the Remote Shutdown Panel with the three ADS MSRVs, but the resulting peak fuel cladding temperature would be higher than previously reviewed and approved for Control Room fires, although no actual fuel damage would occur.

To avoid this potential challenge to the fuel, plant design will be changed during the 1995 refueling outage to alter the control circuit arrangement in the Alternate Remote Shutdown Panel such that the existing switches will control three ADS MSRVs instead of three non-ADS MSRVs. When implemented, it will be possible to open a total of six ADS MSRVs from outside the Main Control Room. This will eliminate the problem outlined above.

A Control Room fire could cause spurious closure of either of the containment isolation valves in the supply lines between the compressed nitrogen supply external to primary containment and the 4.3 cubic foot accumulators for the ADS MSRVs. However, that would not prevent use of the ADS MSRVs, including the six controllable from the remote shutdown system panels. The larger accumulators are not used during pressure relief operation, and each 4.3 cubic foot accumulator has sufficient capacity to open and hold open the corresponding ADS MSRV for the duration of the postulated fire conditions.

REVISED REACTOR SHUTDOWN ANALYSIS

This attachment is presented for NRC staff information; no staff action is being requested.

A new analysis, performed by General Electric, examines reactor water levels and fuel cladding temperatures after a reactor SCRAM initiated as a result of a Control Room fire at WNP-2. The analysis assumes:

- SCRAM from full power at uprated conditions;
- Immediate closure of the Main Steam Isolation Valves (MSIVs);
- Immediate termination of feedwater flow;
- Unavailability of makeup from the High Pressure Core Spray (HPCS) and Reactor Core Isolation Cooling (RCIC) Systems;
- Initiation of reactor depressurization by operators when the reactor water level has declined to a level equivalent to the top of active fuel (TAF), -161 inches, approximately 23 minutes after SCRAM.

The analysis shows that the reactor pressure rises immediately after SCRAM and is stabilized near 1200 psig when Main Steam Relief Valves (MSRVs) open cyclically in the safety mode of operation. Reactor water level declines during the initial stages of the event because inventory is being lost through MSRV cycling, resulting in a degree of reactor core uncovery. Water level begins to recover after reactor depressurization due to reflooding from the Residual Heat Removal (RHR) system operating in Low Pressure Core Injection mode.

This new analysis was performed by GE using their SAFER code which has been approved by the NRC by Reference 5. This code is a more accurate version of the earlier SAFE code which was used in the 1987 studies which were based on initiation of depressurization when water level had declined to -82 inches, approximately 10 minutes after SCRAM. While Reference 6 confirmed that the 10-minute interval is within the bounds of operator response, the new analysis shows that reactor fuel parameters are within previously reviewed and approved limits, based on manual depressurization at TAF (-161 inches), 23 minutes after SCRAM. This significantly increases the time available for operators to provide an orderly plant shutdown.

Three cases were analyzed involving use of two, five, or six MSRVs to depressurize the reactor. As can be seen from the summary table below, use of 5 MSRVs to depressurize after a fire-induced control room evacuation is acceptable from a technical viewpoint, with peak cladding temperatures (PCT) below the 762°F level reviewed and approved by the NRC in Reference 4.

DEPRESSURIZATION CASE	APPROXIMATE DEGREE OF TRANSIENT CORE UNCOVERY	TIME DURATION WATER LEVEL IN FUEL CHANNELS IS BELOW TAF (MINUTES)	PEAK CLAD TEMPERATURE EXCURSION, °F
6 MSRVs used	50%	2.6.	605
5 MSRVs used	60%	3.7	740
2 MSRVs used	100%	12.5	1785

NOTE: Reactor fuel cladding begins to expand at approximately 1500°F, with a likelihood of perforation at approximately 1700°F, as outlined in GE report NEDO-20566, "Analytical Model for Loss of Coolant Analysis in Accordance with 10 CFR 50 Appendix K".

Procedural Implications

Section F.4.3 of Appendix F to the WNP-2 FSAR currently mandates use of 6 MSRVs and a 10-minute time limit for operator actions. Based on the new analysis, the FSAR will be revised to specify a minimum of 5 MSRVs and to extend the time limit for initiation of reactor depressurization to approximately 23 minutes.

With 5 MSRVs being used to depressurize the reactor, the RHR system is capable of providing sufficient water to provide makeup for inventory lost by steam through open MSRVs and to restore the RPV water level while maintaining peak clad temperatures within previously reviewed and approved limits. Use of 6 or more MSRVs is also acceptable from this viewpoint. However, Control Room operators are trained to respond to emergency conditions using the actions outlined in the Emergency Operating Procedures (EOPs), wherein depressurization of the reactor is initiated at TAF using a minimum of 5 MSRVs for all emergencies. In the case of Control Room fires, current procedures require depressurization to be initiated when water level reaches -82 inches using a minimum of 6 MSRVs. This inconsistency increases the possibility for operator error. We plan to eliminate the inconsistency by revising plant procedure 4.12.1.1, "Control Room Evacuation and Remote Cooldown" to reflect the FSAR change indicated above.