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U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Subject: Beaver Valley Power Station, Unit No. 2
Docket No. 50-412, License No. NPF-73
Commitment Changes and
Report of Facility Changes, Tests and Experiments

In accordance with 10 CFR 50.59(d)(2), the Report of Facility Changes, Tests, and Experiments for the Beaver Valley Power Station Unit No. 2 is provided as Enclosure 1. This report provides a brief description of facility and procedure changes which required a 50.59 evaluation and a summary of each evaluation. The report covers the period of October 12, 2003 through April 11, 2005, which corresponds to a period ending with Fuel Cycle 11.

Four commitment changes are described in Enclosure 2, and are forwarded as part of this submittal in accordance with the NRC endorsed guidance of the Nuclear Energy Institute (NEI) related to the commitment change process (Reference: NEI 99-04). There are no new regulatory commitments contained in this letter or Enclosure 1.

If you have any questions regarding this report, please contact Mr. Henry L. Hegrat, Fleet Licensing Supervisor at 330-315-6944.

Sincerely,

L. W. Pearce

L. William Pearce

Enclosures:
1. Beaver Valley Power Station Unit 2, Report of Facility Changes, Tests, and Experiments for the period October 12, 2003 through April 11, 2005.
2. Beaver Valley Power Station Unit 2, Commitment Changes

Reference: NEI 99-04, "Guidelines for Managing NRC Commitments"

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c: Mr. T. G. Colburn, NRR Senior Project Manager
Mr. P. C. Cataldo, NRC Senior Resident Inspector
Mr. S. J. Collins, NRC Region I Administrator
Mr. D. A. Allard, Director BRP/DEP
Mr. L. E. Ryan (BRP/DEP)

Change Title

Small Break Loss of Coolant Accident Peak Clad Temperature Re-Analysis

Change

The small break Loss of Coolant Accident (LOCA) was re-analyzed. The vendor calculation of small break LOCA Peak Clad Temperature (PCT), PCT rack-up sheet, and Updated Final Safety Analysis Report markups for the small break LOCA peak clad temperature re-analysis were evaluated per 10 CFR 50.59. The 10 CFR 50.59 evaluation was required because the analysis PCT results increased. The analysis determined that the PCT was less than the acceptance criterion of 2200 degrees Fahrenheit. (Reference Evaluation Number 03-04128.)

Change Title

ECP 04-0527, Construction of New Access Openings in the Intake Structure

Change

This design change installed two (2) new manway openings and enlarged six (6) existing manway openings in the 705 foot elevation floor slab of the Intake Structure. The new and enlarged openings provide additional access to the intake bays (1-INTS-BAY-A, -B, -C, -D). The existing manways will continue to be used for normal and emergency personnel access and egress and the new and enlarged openings will be used for equipment access and egress. Checkered plate covers were installed over the new and enlarged openings. The covers will either slide or can be lifted to gain access to the manways. (Reference Evaluation Number 04-04434.)

The structural and seismic integrity of the intake structure was maintained with the installation of the new access openings. The new access openings do not penetrate the service and river water cubicle walls that provide flood, fire and tornado protection. Accident and component malfunction analyses, as currently described in the Updated Final Safety Analysis Report, remain bounding for all cases where the function of the service water and river water pumps is required.

Change Title

Temporarily Remove and Reinstall Roof Plug in the Unit 2 Safeguards Building While in Mode 1

Change

A concrete roof plug was removed from the Safeguards building and placed on the building roof adjacent to the opening to allow for installation of a personnel contamination monitor seismic restraint. The concrete roof plug was then reinstalled on the Safeguards building.

Evaluation of the floor plug opening as a potential post accident release point for radioactive material, and temporary degradation of the safeguards structure missile protection were evaluated. Although radioactive material that reaches the safeguards building atmosphere as a result of a design basis accident would normally be collected and filtered by the building's ventilation system (including HEPA and charcoal filters) prior to release to the environment, the accident analysis does not take credit for holdup or these treatment functions. Because of the short duration of the activity, and precautions taken to avoid performing the activity during severe weather, the probability of occurrence for missile damage during a design basis accident coincident with severe weather was found to be minimal. Damage to equipment in this area cannot initiate a design basis accident. (Reference Evaluation Number 04-02259.)

Change Title

Change to Administrative Requirement to Reduce Power During AMSAC Inoperability

Change

The October 10, 1996 response to a Notice of Violation (EA 96-244) for Beaver Valley Power Station (BVPS) Unit No. 1 committed to a voluntary administrative limitation for BVPS Unit Nos. 1 and 2 that imposed a time limit for inoperability of the Anticipated Transient Without Scram Mitigating System Actuation Circuit (AMSAC) system, after which plant power level was to be reduced to less than 40 percent. The administrative requirement to reduce plant power level has been replaced by a requirement to assess the safety significance of the condition per Generic Letter 91-18 within the next seven days.

AMSAC is a backup system for auxiliary feedwater initiation required by 10 CFR 50.62. There is no regulatory requirement to establish an allowable outage time for AMSAC and the system is not used to mitigate a design basis transient analyzed in the Updated Final Safety Analysis Report.

The NRC Significance Determination Process was considered in evaluating the safety significance of eliminating the power reduction and extending the outage time by seven days while the Generic Letter 91-18 evaluation is being performed. Since the event duration category (3-30 days) remains the same, the degree of risk is unchanged. It has also been determined that the coincident probability of an event requiring AMSAC and a total loss of feedwater (i. e. conditions where AMSAC function is needed) is very small. (Reference Corrective Actions 04-01036-08 & -13.)

Change Title

Replace Procedural Requirement to Place a Failed Feedwater Flow Channel in the Tripped Condition

Change

This change replaced a procedural requirement to place a failed feedwater flow channel in the tripped condition, with a requirement to declare the Anticipated Transient Without Scram Mitigating System Actuation Circuitry (AMSAC) system inoperable and enter the administrative allowable outage time. The procedural requirement to place a failed feedwater flow channel in the tripped condition was communicated to the NRC in response to a November 19, 1987 conference call. The response was documented in a December 2, 1987 letter to the NRC that stated: "Should a feedwater flow channel fail and defeat the AMSAC output, we will require the failed AMSAC flow channel to be placed in a tripped condition within 48 hours which will remove the auto block feature."

The requirement to place a failed feedwater flow channel in the tripped condition is being eliminated because of the potential for an inadvertent AMSAC actuation due to failure of one of the remaining channels. It has also been determined that the coincident probability of an event requiring AMSAC and a total loss of feedwater (i. e. conditions where AMSAC function is needed) is very small. Based on the degree of risk, declaring AMSAC inoperable and entering the allowable outage time are aceptable when compared to the risks associated with an inadvertent actuation when a failed feedwater channel is maintained in the tripped condition. (Reference Corrective Actions 04-01036-13 and -9.)

Change Title

Revised Commitment to Test Service Water System Valves 2SWM-MOV562, 2SWM-MOV563, 2SWM-MOV564, and 2SWM-MOV565 as Part of the Generic Letter 96-05 MOV Periodic Verification Program.

Change

A letter to the NRC dated March 17, 1997 stated that the Generic Letter 96-05 program periodically verifies safety related Motor Operated Valves (MOVs) within the scope of the Generic Letter 89-10 Program continue to be capable of performing their safety functions within their respective current licensing basis. A letter to the NRC dated April 29, 1994 responded to Generic Letter 89-10, Supplement 6, and specifically identified valves 2SWM-MOV562, 2SWM-MOV563, 2SWM-MOV564, and 2SWM-MOV565 among the Generic Letter 89-10 MOVs that would be tested during the fifth refueling outage. The NRC Safety Evaluation Report (SER) for Generic Letter 96-05, forwarded to Beaver Valley Power Station by a letter dated February 22, 2000, stated that the NRC staff review found that the MOV program scope was consistent with Generic Letter 89-10 and its supplements.

Based on the foregoing docketed correspondence, the NRC was informed that 1) the MOVs in question were in the Generic Letter 89-10 population, and 2) the valves in the Generic Letter 89-10 population are in the Generic Letter 96-05 MOV Periodic Verification Program valve population.

The Normal System Arrangement (NSA) position of valves 2SWM-MOV562, 2SWM-MOV563, 2SWM-MOV564, and 2SWM-MOV565 was formerly OPEN, and AUTO-CLOSED on a safety injection system signal. The Chemical Injection System has been retired; therefore, these valves are no longer required to perform any emergency or safety function. The current NSA position of the valves is sealed shut. These valves have been de-energized closed, and are functionally isolated.

Unit 2 valves 2SWM-MOV562, 2SWM-MOV563, 2SWM-MOV564, and 2SWM-MOV565 will no longer be included in the Generic Letter 89-10 MOV population, and will no longer be subject to periodic testing under the Generic Letter 96-05 program. (Reference Corrective Action CA 04-02618-04.)

Change Title

Change to Method for Verifying Heat Transfer Capability of Safety Related Heat Exchangers

Change

In response to Generic Letter (GL) 89-13, the means for verifying heat transfer capability for specific safety related heat exchangers was provided in a letter to the NRC dated January 29, 1990. The response identified the specific means for verification (that is, testing and/or maintenance) for each particular heat exchanger. The revised commitment instead allows the use of either testing or regular maintenance for any of the heat exchangers. This change is based on experience that testing has normally been a less effective method of monitoring than periodic maintenance. Generic Letter 89-13 describes frequent regular maintenance as an acceptable alternative to testing for degraded performance. (Reference Corrective Action CA 04-07471-03.)