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United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Attn: Boyce H. Grier, Regional Director
Region I
631 Park Avenue
King of Prussia, Pennsylvania 19406

Reference: Beaver Valley Power Station, Unit No. 1
Docket No. 50-334, License No. DPR-66
Report of Facility Changes, Tests and Equipment

Dear Mr. Grier:

In accordance with the requirement of 10 CFR 50.59, we are submitting the 1980 report of facility changes, tests and experiments for the Beaver Valley Power Station, Unit No. 1.

Very truly yours,

C. N. Dunn
Vice President, Operations

cc: Director, Office of Inspection and Enforcement (39)
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DUQUESNE LIGHT COMPANY
Beaver Valley Power Station
Docket No. 50-334, License No. DPR-66

Report of Facility Changes, Tests and Experiments

The following is a compilation of facility changes, tests and experiments completed during 1980 which are being reported in accordance with the Code of Federal Regulations, Title 10, Paragraph 50.59, "Changes Tests and Experiments." Several of these changes were previously reported to the NRC as they involved Technical Specification changes.

Design Change DC0094

Design Change DC0094 entitled "Automatic Changeover From the Safety Injection Mode to the Recirculation Mode" involved automatic transfer of safety injection water source from the Refueling Water Storage Tank (RWST) to the containment sump (recirculation mode) on low levels in the RWST. This is accomplished by a 2 of 4 low level signal from the RWST coincident with a Safety Injection signal. The 2 of 4 energize to trip logic system has been designed and installed to meet both separation and redundancy criteria at the Station.

The automatic changeover system performs the following functions:

1) Transfers the suction of the Low Head Safety Injection (LHSI) pumps from the RWST to the containment sump.

2) Closes the LHSI pump recirculation valves.

3) Realigns the suction of the charging pumps from the RWST to the LHSI pumps.

The safety evalution concluded that the change did not constitute an unreviewed safety question. This was based on the Design Change not increasing the possibility of occurrence of an accident or malfunction of equipment to safety as previously evaluated in the FSAR. The automatic changeover system has not created the possibililty of an accident or malfunction of a type that was not previously evaluated in the FSAR. Finally, the margin of safety as designed in the technical specification basis has been maintained during the design, installation and testing of this change. This safety posture is justified by the fact that the automatic changeover system results in the same valve positions and Safety Injection system flow paths as were accomplished by the manual operator transfer method and as approved in the original FSAR. The original FSAR includes a single failure analysis of the manual transfer method and concludes that the system is single failure proof. This single failure analysis is still applicable to the automatic changeover system.

Design Change DC0103

Design Change 0103 entitled Gaseous Waste System (GWS) Modification consisted of the following modifications:

1) Increasing the size of the Degasifier Vent Chiller vapor outlet lines to reduce flow restriction.

2) Increasing the size of the Overhead Gas Compressor suction lines to reduce flow restrictions.

3) Separating the sample lines coming from the Decay Tanks and the Surge Tank to allow for continous sampling of the oxygen concentration in the Surge Tank.

4) Rerouting the Degasifier Vent Chiller and Knockout Pot drain lines to the Primary Drains Transfer Tank, DG-TK-2, to provide for better draining of the condensate.

5) Relocating the oxygen analyzer sample lines from the bottom of the Surge Tank to a different tank outlet line to prevent moisture carryover into the analyzers.

6) Rerouting various gaseous vents, which were piped into the degasifiers, to a point downstream of the degasifiers. This will allow noncondensibles to bypass the degasifiers.

7) Providing an interlock between the oxygen analyzers and the Overhead Gas Compressors which will automatically shutdown the compressors on a high oxygen level in the Surge Tank.

The system modifications summarized above do not create an "unreviewed safety question" as defined in 10 CFR 50.59. The modifications are minor refinements and not major changes in system design and help ensure that the system performance is consistant with the bases established in the FSAR. The modifications were designed consistant with the quality assurance categories and design codes of the original system designs. The accidental release of waste gas from the Primary Drains Transfer Tank combined with the Degasifiers was compared to the accident analysis present in Section 14.2.3 (Accidental Release of Waste Gases) of the FSAR and was determined to be a less severe accident than that addressed in the FSAR. The modifications do not:

1) Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report.

2) Create a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report.

3) Reduce the margin of safety as defined in the basis for any technical specifications.

Design Change DC0142

Design Change Package 0142 entitled "Control Room Annunciation For Diesel Generator Not Available" involved changing the control room annunciator windows that originally read Local Panel Trouble D/G #1 and Local Trouble D/G #2 to D/G #1 Not Available and D/G #2 Not Available. In addition, alarms were added into these windows. Instituting these changes will provide added assurance that the diesel generators are available to more effectively perform their intended function by adding alarm circuitry for generator breaker DC control power failure.

The safety analysis for this Design Change indicated that no changes to any procedure as described in the FSAR had occurred nor did this change result in any unreviewed safety question.

Design Change DC0156

Design Change Package 0156 entitled "Control Room Chlorine Detection" involved automatic closing of the control room ventilation outside air supply and exhaust isolation dampers in the event of a chlorine spill near the Control Room air intake. In addition, an alarm will accuate and an emergency bottled air supply will be activated if any chlorine is detected. Also a redundant bottled air supply capable of supplying approximately eight (8) hours of breathable air has been provided.

As part of the safety review for this Design Change it was determined that the probability of an occurrence of an accident of equipment malfunction had not increased, therefore, no possibility of an accident different from that previously evaluated in the FSAR exists. Furthermore, this Design Change will not reduce the margin of safety within the plant and will not create an unreviewed safety question.

Design Change DC0157

Design Change 0157, "Provide Means of Preventing Breaker Closure on Emergency Diesel Generator (D.G.) on No Field," involved the installation of new electrical equipment to prevent the diesel generator 4160 volt breakers from closing if the generator does not have a functioning field. In addition the existing electro mechanical undervoltage relays (type SV-1) were replaced with a solid state equivalent.

The safety evaluation for this design change stated that implementation will not require changes to procedures as described in the FSAR nor will it result in any unreviewed safety question. It was determined that this design change safely provides the engineering to prevent the D.G. 4160 breakers from closing when the generator does not have a functioning field. Furthermore, replacing the existing electromechanical undervoltage relays (type SV-1) with solid state equivalents will enhance the plant's safety posture by eliminating the present problem of an occasionally sticking relay plunger.

Design Change DC0188

Design Change 0188 entitled "Low Head Safety Injection (LHSI) Pump Modification" involved modifying the pump to eliminate a potential vibration problem and modifying the pump bearings to minimize bearing wear. These problems were encountered at the North Anna Plant whose LHSI pumps are very similar to the Beaver Valley pumps. Specific pump modifications involved installation of a vortex eliminator in the pump volute, the addition of bracing and wedges to stiffen the can, modifying the pump bearings, chrome plating the shaft sleeves and installing a new type of coupling.

During the safety evaluation of this design change it was determined that the probability of an occurrence of an accident or malfunction of equipment important to plant safety as previously evaluated in the FSAR had not increased. This was based on the fact that the pumps are similar to the North Anna pumps where the same modifications were incorporated. These modifications were intended to increase pump reliability by eliminating a potential vibration problem and minimizing wear of the pump bearings. Furthermore, this design change has neither created the possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR nor has it reduced the margin of safety in the basis for any technical specification. Pump performance characteristics will still meet all Tech. Spec. requirements even though they have been modified. This was established in the pump proof testing subsequent to the modifications.

Design Change DC0190

Design Change 0190 entitled "New Steamline Break Protection System" involved a modification to the Solid State Protection System to correct a potential operational problem in which spurious actuation of safety injection system could occur at low power. This actuation could result either from a high steam line differential pressure safety injection actuation signal or by high steam flow in coincidence with low Tave. This change which reduced the amount of equipment used in detecting a steam break should improve the safety of the system by reducing the number of inadvertant plant trips and safety injections. By reducing the number of detectors, the surveillance testing is decreased, thus reducing the time periods the system is susceptible to single failure actuation.

As summarized in Amendment 30 to Facility Operating License No. DPR-66 extensive safety evaluations were performed for the new Steamline Break Protection system by Westinghouse and the NRC.

It was concluded that for normal operation the steamline isolation and safety injection trips would occur within the limits stated in the FSAR and in most cases would occur sooner that in the previous Steamline Break Protection System. During heatup or cooldown, the new system provides a reduced level of protection in that there are no primary trips that actuate Safety Injection for a steamline break. This is considered acceptable because additional protection has been included through a new surveillance requirement to assure adequately borated reactor coolant, a new procedural action to assure adequate charging flow rate, and an analysis that demonstrates that the core will always be covered and RCS remains subcooled with Safety Injection.

Based on the above information and a review of the overall modification, the safety analysis concluded that this modification will not:

1) Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report.

2) Create a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report.

3) Reduce the margin of safety as defined in the basis for any technical specifications.

Design Change DC1091

Design Change 0191 entitled "Deletion of Reactor Trip Following Turbine Trip Below 50% Power" involves a change to the Solid State Protection System that if placed in service would allow the reactor to trip at power levels up to 50% power coincident with a turbine trip.

The delection of a reactor trip system utilizes a 2 of 4 logic system from the Nuclear Instrumentation System (NIS) P-9 trip to determine if the reactor power is above 50%. The P9 trip is used to block or unblock a reactor trip following a turbine trip. The 2 of 4 logic signal is sent to both trains of the Solid State Protection System (SSPS) which, in turn, coincident with a turbine trip signal either trips the reactor above 50% power or blocks a reactor trip below 50% power and operates steam by-pass valves to accommodate the load rejection. The 2 of 4 logic system and the SSPS inputs and outputs have been designed and installed to meet the separation and redundancy criteria at the plant. Operability of the system is verified by periodic testing of:

1) The P-9 setpoint

2) The 2 of 4 SSPS logic.

3) The SSPS output to reactor trip breakers

Based on the discussion above and a Westinghouse analysis, the DCP has not increased the probability of occurrence of an accident or malfunction of equipment important to safety as previously described in the FSAR. The delection of reactor trip system has not created the possibility of an accident or malfunction of a type that was not previously evaluated in the FSAR. The margin of safety as defined in the technical specifications basis has been maintained during the design, installation and testing of the DCP.

Although the physical hardware has been installed to facilitate this change, the Tech. Spec. for deletion of a reactor trip following a turbine trip below 50% power has not been approved. Currently the P9 setpoint is set at 10% Reactor Power the same as the P10 setpoint to cause a Reactor Trip above 10% Reactor Power with turbine trip as required by present technical specifications pending favorable review by NRC Licensing.

Design Change DC0201/202

Design Changes 0201/202 entitled "Ventilation Modifications to the Primary Plant System" were installed so that in the event of a fuel handling accident in the fuel building the resulting radiation exposure to the public will be within the limits of 10 CFR 100. The Fuel Building Emergency Exhaust System has been eliminated from the Supplementary Leak Collection and Release System (SLCRS). These changes establish that the existing SLCRS system of the fuel building will be utilized during all modes of fuel building ventilation system exhausting. The following design requirements are better met as a result of this change:

1) Adequate ventilation of the fuel building.

2) Negative pressure is maintained in the fuel building.

3) Radiation exposure to the public will be kept within the permissible limits of 10 CFR 100.

4) Fuel building exhaust will be diverted directly to the SLCRS filters in the event of a fuel handling accident.

5) The site boundary dose limitations in accordance with the FSAR and Regulatory Guide 1.25.

The safety review for this modification determined that no unreviewed safety question exists as a result of this change. In addition, the following comments were provided as part of this review:

1) All exhaust from containment during refueling will be run through the main filter banks. This change reduced the possibility of potential exposure to the public. Previously, the system relied on series and parallel redundant dampers to place the main filter banks in service prior to upstream hi detected radioactivity reaching the changeover dampers.

2) This change eliminated the possibility of failure to isolate containment during a seismic event. The original design was not required to meet seismic criteria.

3) This design provides a seal in circuit to prevent damper repositioning on safety signal resets. Previously, the damper would reset to pre-accident position. The new design now requires operator action to reposition the damper.

4) A re-flash system was installed. Originally, if a high radiation alarm sounded, it was not possible to determine if an additional alarm had actuated. Now with one channel in alarm and a second channel alarms, the control room annunciator will reflash.

5) Air Ejector Discharge to Containment Valve [TV100A] is normally closed. A high-high radiation alarm and no Containment Isolation B Phase (CIB) will cause this valve to open to containment. If a CIB occurs, this valve will not open and operators cannot cycle this valve manually. After the CIB clears, the valve can be opened manually or on an air ejector discharge high-high radiation alarm. Since this valve will now stay closed after CIB, an additinal margin of safety to the public exists that did not exist before (i.e., this valve cannot be opened on a CIB and will not return to the open position until the CIB clears).

6) This design change eliminated fuel building discharge to containment for dilution for a fuel handling accident within the fuel building. This change has eliminated the potential of additional exposure to radiation workers in containment.

Design Change DC0205

Design Change 0205 entitled "Solid State Protection System Protection for Reactor Coolant Pump (RCP) Trip on False Undervoltage" involved a modification to the SSPS which will eliminate spurious reactor trips which can occur due to a power failure to the pump breaker open position sensing circuit. This design change eliminates the reactor trip on 1/3 RCP breaker open position (approximately 30% power) above P-8. The reactor trip on 2/3 breaker open position above P-7 and P-8 is unaffected.

The safety evaluation for this design change was pertinent only to the reactor trip logic above P-8 power levels (since no changes have been made to the logic below P-8) and concluded that no unreviewed safety questions were involved.

For a complete loss of flow (Station blackout) reactor trip is provided by the undervoltage and underfrequency relays on the reactor coolant pump power supply busses. The loop low flow reactor trip (2/3 per loop) serves as a backup to the undervoltage relay trip for this accident. The breaker position reactor trip was not considered a backup for this type of accident. Therefore, removal of the trip does not affect safety for the actual loss of flow.

For a partial loss of flow, i.e., loss of one reactor coolant pump, the primary reactor trip is the loop low flow. Previous to this design change, the trip on breaker open position provided an anticipatory reactor trip for the case where loss of flow resulted from tripping of the reactor coolant pump breaker. The trip on breaker position was considered as a backup and no credit was taken for it in the FSAR accident analysis. Therefore, deletion of this trip does not increase the probability of an accident as previously evaluated in the FSAR.

The margin of safety is slightly reduced in the sense that one backup system was eliminated. However, the primary reactor trip system on loss of flow, which is the 2/3 per loop low flow trip, meets all required redundancy and separation criteria. Overall, it is expected that reactor safety will be enhanced by preventing spurious reactor trips which are challenges to the plant safety system.

Design Change DC0222

Design Change 0222 entitled "Replace Control Room Air Conditioning Compressor" involved replacing one of the control room air conditioning compressors which failed in service and modifying the controls on both compressors to prevent additional compressor failure.

The probability of occurrence of an accident previously evaluated in the FSAR is not increased by this design change because these compressors are not addressed as the cause of any accident previously evaluated.

As stated in the response to AEC Question 9.2 in the FSAR, the control room air conditioning system is essential to maintain temperature and humidity for the proper operation of the computer and instrumentation and for the comfort and efficiency of the operating personnel during normal operation and accident conditions. It is for this reason that the air conditioning system equipment is important to plant safety. Since the new compressor is identical to the failed compressor with the exception of the addition of an oil cooler, and since the new compressor and controls were procured and installed in accordance with Q.A. Category I requirements, it is believed that the probability of occurrence of a malfunction of this equipment is not increased. In fact, since the controls have been modified to prevent another failure, the reliability of these compressors is improved. Modifications made to the 4B compressor stand (necessitated by the new model) have been seismically qualified by the compressor vendor, thus there is no degradation of seismic integrity.

The consequences of an accident and the consequences of a malfunction of this equipment is not increased for the same reasons discussed in the previous paragraph.

No accident or malfunction of a different type than previously evaluated in the FSAR can be foreseen as a result of this design change.

The only technical specifications directly related to the control room air conditioner compressors is 3/4.7.7, Control Room Emergency Habitability System. Because the reliability of the control room air conditioner compressors is not decreased, and in fact is expected to increase as a result of this design change, the margin of safety as defined in the basis for this technical specification is not reduced.

Based on the above, it has been determined that no unreviewed safety question will occur as a result of this design change.

Design Change DC0232

Design Change 0232 entitled "Charging System Relief Valve," involved the installation of an overpressure protection relief valve on the plant regenerative heat exchanger. This valve was installed to provide a new discharge path from the regenerative heat exchanger as a means of overpressure protection. The original discharge path via a spring loaded check valve could result in a partial loss of Safety Injection flow and affect the flow performance of the Emergency Core Cooling System in the event the outside containment isolation valve failed to close.

The safety analysis for this change determined that since the probability of occurrence of an accident or equipment is not increased, that there is no possibility of an accident different from those previously evaluated in the FSAR, and that there is no reduction of safety margins and no unreviewed safety questions. Furthermore, this change should enhance the operation of the safety injection system.

Design Change DC0237

Design Change 0237 entitled "Removal of the Part Length (P/L) Control Rods," involved the removal of the five (5) P/L control rods and retiring in place their drive mechanisms. The P/L control rods were removed from the reactor core and replaced with thimble plugs. In addition, the P/L control rod drive motors and lead screws were retired in place and the lead screws pinned to prevent rotation and subsequent detachment from the Control Rod Drive Mechanism rotor into the upper head of the vessel. This change was made because the NRC now prohibits the use of part length rods. Therefore, fewer refueling operations with attendant exposure reduction will be realized during future refuelings.

The safety evaluation for this change concluded that the change did not result in any unreviewed safety questions.

Design Change DC0239

Design Change 0239 entitled "Provide Shielding for Fuel Transfer Tube" involved a modification to the plant areas adjacent to the fuel transfer tube to eliminate potential high radiation areas. Inside containment, two (2) areas were determined to require additional shielding, i.e., the 3/8" gap between the reactor cavity concrete and fuel transfer tube shielding and the seismic separation between the fuel tranfer tube shielding and the containment liner. In addition, one (1) area outside containment was determined to require additional shielding, i.e., the area around the labyrinth shield in the fuel pool leak detection room. The steam generator drain lines were repositioned to allow installation of this shielding.

The safety review for this design change determined that this modification should safely provide the shielding adequate to reduce the radiation levels at the shake space inside and outside of the Reactor Containment. This was later verified during refueling when the first fuel assembly was transferred and extensive radiological surveys were performed both inside and outside containment. It was further determined during the safety review of this modification, that no unreviewed safety question would result from incorporation of the subject change.

Design Change DC0242

Design Change 0242 entitled "Changing Feedwater and Main Steam Root Valves" involved replacing thirty-four (34) Main Steam and Feedwater Instrument Root Valves with new 1500 lb. packless nuclear valves. The original 600 lb. valves were a constant maintenance problem to correct packing and bonnet leaks.

The safety evaluation for this modification stated that the upgrading of these valves will enhance the safety posture of the Main Steam System and the Steam Generator Feedwater System and should improve plant reliability by reducing the number of shutdowns previously required to repair root valve leakage. The safety evaluation concluded that no unreviewed safety question will result from this change.

Design Change DC0249

Design Change 0249 entitled "Add Undervoltage Relays to Emergency Busses" involved installation of undervoltage relays to the emergency busses. The specific modifications resulting from this Design Change will:

1) Provide protection for class IE electrical equipment against a sustained undervoltage (90%) condition.

2) Allow emergency bus undervoltage motor trip relays to operate if the 4160V emergency bus is supplied from the offsite source.

3) Prevent emergency bus undervoltage motor trip relays from operating when the 4160V emergency bus is supplied from the onsite source.

4) Allow emergency bus undervoltage motor trip relays to operate if the 4160V emergency bus is supplied from both the onsite and the offsite sources simultaneously (exercise mode).

The safety analysis for this modification noted that additional protection will be afforded by adding undervoltage relays to emergency 4160V and 480V switchgear busses. This change will also provide additional system reliability. Finally, the safety evaluation concluded that no unreviewed safety questions will result from this change.

Design Change DC0256

Design Change 0256 entitled "Replacement of Cables For Cable Seaparation" involved replacing and/or rerouting electrical cables that were found to be incorrectly installed. Specifically this change involved:

1) Determining why cable anomalies exist so as to prevent their reoccurrence.

2) Replacing and/or rerouting cables as necessary.

3) Reviewing various installed cable routing to determine if any new problems exist.

4) Determining if installed cable routing problems are significant enough to warrant a further cable routing verification program.

The safety evaluation for this design change determined that no unreviewed safety questions will result from this modification. In addition, it was indicated that the following safety features will result from this change:

1) Enhance the safety posture of the plant by eliminating problems with meeting separation criteria.

2) Improving plant reliability by establishing and maintaining separation criteria for safety-related cables. Establishment of separation criteria will also provide more assurance of safe shutdown capability.

Design Change DC0257

Design Change 0257 entitled "Modifications Required by IE Bulletin 79-02" involved modification to Seismic Category I pipe hanger baseplates which use concrete anchors and do not meet the installation inspection criteria established to comply with NRC IE Bulletin 79-02. Specific modifications involved addition of material to the hangers and/or baseplates for the purpose of stiffening a baseplate, reinforcing a hanger or otherwise increasing the capacity of the hanger. Also included were the addition of new hangers to lines as well as the addition of replacement anchors and/or gussets to the baseplates. In addition, existing material has been changed out with heavier material for the purposes of meeting plate flexibility considerations.

The safety evaluation for this change indicated that testing and replacing defective bolt anchors, which failed during testing, will provide an additional margin of safety in that no defective anchor bolts are being utilized for base plate installation. Furthermore, this change will decrease the probability of an accident occurring since the base plates are now installed per the original design. Finally, the safety evaluation concluded that no unreviewed safety question will result from this change.

Design Change DC0267

Design Change 0267 entitled "River Water Pump Modifications" involved a change to the 1B River Water Pump necessary to establish original impeller clearance dimensions. Specific changes involved installation of an oversized bottom bearing and machining of the suction bell to achieve proper fit-up.

The safety evaluation for this change concluded that no unreviewed safety questions will arise as a result of this change.

Design Change DC0276

Design Changes 0276 entitled "Relocation of Four Narrow Range Steam Generator Level Transmitters" involved physically moving these transmitters to a higher elevation which would not be submerged during a post LOCA condition. In addition, a Reactor Coolant Pressure Transmitter was also relocated to a higher elevation for the same reason. It was necessary to relocate these transmitters as a result of the IE Bulletin 79-01B requirements.

The safety evaluation for this change concluded that no unreviewed safety questions will result from the incorporation of this change.

Design Change DC0277

Design Change 0277 entitled "Insultate Steam Generator Reference Legs" involved insulation of the reference water legs of the level transmitters on the three (3) steam generators. The insulation was installed to comply with IE Bulletin 79-21 and was intended to minimize level measurement errors in the event of a high energy line break.

The safety evaluation concluded that the probability of an occurrence of an accident or malfunction of equipment important to plant safety as previously evaluated in the FSAR had not increased because insulating the reference legs decreases the probability of instrument error due to reference leg heat-up. Furthermore, the possibility created for an accident or malfunction of a different type than previously evaluated had not increased nor had the margin o. safety as defined in the basis for any technical specification been re ced. All safety margins are the same as originally analyzed in the F . The Steam Generator Low-Low trip setpoint was raised from 10 to to compensate for a 2% error resulting from the reference leg heat-up inally, it was determined that no unreviewed safety questions will o r as a result of this change.

Design Change DC0278

Design Change 0278 entitled "Change Out Sol oid Operated Valves (SOV's) per NRC IE Bulletin 79-01" involved the r lacement of existing solenoid operated valves with new environmentally alified valves. Replacement valves were purchased to the requiremen of IEEE 323 and 344. In addition, where replacement SOV's were instal in series a check valve and bypass line were added to ensure full flow to the SOV furthest away from the valve diaphragm in the event it is deenergized and exhausting.

The safety evaluation for this change concluded that the probability of an occurrence or the consequences of an accident or malfunction would not be increased because direct replacement of existing SOV's does not affect their function. Furthermore, the consequences of a Loss of Coolant Accident (LOCA) would be expected to less because the replacement valves are qualified for post-LOCA environment. Addition of the bypass line and check valves does not change the normal function of the solenoid valves, and could prevent possible malfunction. Single failure criteria are satisfied in that no single component failure will prevent closure of associated trip valves in the event of a trip signal. The safety evaluation further stated that the possibility for an accident or malfunction of a different type than any previously evaluated in the SAR had not been created. Also the margin of safety as defined in the basis for any Technical Specification has not been reduced. The valve function and operating times are not affected by this design change, therefore, no change to technical specifications are required. Finally, the safety evaluation concluded that no unreviewed safety questions will result from this change.

Design Change DC0292

Design Change 0292 entitled "Pressurizer Safety and Relief Valve Position Indication," a NUREG 0578 item, involved installation of acoustic pickup monitors to provide plant operations personnel with an indication of the pressurizer relief and safety valve position. The intent is to ensure the integrity of the RCS by monitoring the PORV's and safety valves for leakage. The acoustic monitor pickups were installed with an annunciation alarm that will actuate on the Main Control Board in the Control Room.

The safety evaluation for this change concluded that no unreviewed safety questions were involved. This change will safely provide qualified equipment for positive Control Room indication of RCS PORV and safety valve position derived from a reliable valve position detection device or a reliable indication of flow in the discharge piping. Also the probability of an occurrence or consequences of an accident or malfunction of equipment important to safety as previously evaluated in the SAR has not been increased. The safety evaluation further indicated that the possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR had not been created nor had the margin of safety as defined in the basis for any technical specification been reduced. In fact the installation of the acoustic monitors will provide an additional margin of safety in that it establishes another control for monitoring potential PORV and safety valve leakage.

Design Change DC0293

Design Change 0293 entitled "Subcooled Margin Meter" involved installation of monitoring equipment to provide indication of inadequate core cooling. This meter has been installed in accordance with the NUREG-0578 requirements and uses existing pressure and temperature instruments and incore thermocouples to determine the saturation pressure for any incore or loop temperature. The subcooling meter monitor panel and annunciator are installed in the Control Room.

The safety evaluation for this change determined that no unreviewed safety questions exist and that the installed instrumentation will effectively monitor the RCS subcooled margin to saturation as required by NUREG-0578. Furthermore, the evaluation stated plant operators will now have a more reliable indication of core cooling capabilities during accident recovery. Finally, the safety evaluation concluded that this change will not:

1) Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report.

2) Create a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report.

3) Reduce the margin of safety as defined in the basis for any technical specification.

Design Change DC0306

Design Change 0306 entitled "Upgrade RCS and Steam Generator Level Transmitters" involved the replacement of pressure and differential pressure transmitters with limited environmental and seismic qualifications with seismically and environmentally qualifed transmitters meeting IEEE criteria 1974. The replacement pressure transmitters are static modified Barton Model 393 Electronic Pressure Transmitters and directly replaced the RCS pressure transmitters and the pressurizer pressure transmitters. The replacement differential pressure transmitters are a combination of Barton Model 386 electronics and the Barton Model 752 differential pressure sensor. These transmitters were installed as replacements for the pressurizer level transmitters and steam generator narrow range level transmitters.

The safety evaluation for this change stated that the installation of the qualified Barton transmitters will provide more plant reliability since they possess both seismic and environmental qualifications. Also the new differential pressure cells will now be capable of operating after a LOCA or seismic event which means the station has better monitoring equipment for post-accident conditions. Finally, the safety evaluation concluded that no unreviewed safety questions will result from this change.

Design Change DC0312

Design Change 0312 entitled "Add Vent Lines to Charging Pumps" involved the reinstallation of vent valves on the suction of each of the three charging pumps. These valves were installed to allow air to be vented from the suction line of each Charging Pump prior to pump start-up. Potential pumping problems could result if this free air is not vented from the system. During pump operation, the vent valves are placed in the closed position.

The safety evaluation for this change stated that this change did not create an unreviewed safety question in that it was a rework of a previous installation. It was noted that due to inconsistencies existing between the original as-installed vent valves and applicable codes and standards, that the original vent valves should be removed and new valves installed.

Design Change DC0319

Design Change 0319 entitled "Bergan Patterson Snubber Repair" involved repair of twenty-five (25) of the thirty-six (36) Bergan Patterson large bore snubbers currently installed in the plant. Specific modifications involved machining several of the snubber glands to remove gouged areas and installing a bronze alloy sleeve. In addition, some snubber rams were also machined to remove surface gouges and metal sprayed to attain proper fit-up. Also any damaged or loosened structural bolts were replaced as required.

The safety evaluation for this change indicated that no unreviewed safety questions have resulted and no change to the FSAR had occurred.

Design Change DC0335

Design Change 0335 entitled "Overstressed Floor Beam in the Cable Spreading Area" involved the modification of existing floor beams in the plant Service Building at Elevation 725'6". Specifically this design change ensured the structural adequacy of overstressed structural steel beams supporting the floor, conduits and cable trays below the floor during a seismic event.

The safety evaluation for this change determined that the installed design will safely provide the engineering to ensure the structural adequacy of the overstressed structural steel beams supporting the floor, conduits and cable trays below the floor (Elevation 725'6" of the Cable Spreading Area of the Service Building) during a seismic event. The safety evaluation also concluded that this change does not involve an unreviewed safety question.

DUQUESNE LIGHT COMPANY
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Design Change DC0343

Design Change 0343 entitled "Modify Recirculation Spray Pumps to Block Flush Lines" involved cutting and capping the flush line on the first stage of the inside recirculation spray pumps. This flush line could cause excessive wear on the lower first stage bushing of each pump and was capped at the direction of the pump vendor.

The safety evaluation stated that this change will improve the wear pattern of the first stage bushing and provide more even wear on the remaining bushings. Furthermore, it was concluded that no unreviewed safety questions will result from this change.

Design Change DC0355

Design Change 0355 entitled "Charging Pump Mini-flow By-Pass Valve Elimination" involved the removal of one (1) 3/4" manual isolation valve on the bypass line around each of the charging pump mini-flow orifices. This modification consisted of removal of three (3) valves (one (1) per pump) as the valves were a constant maintenance problem due to leak through and subsequent excessive charging pump mini-flow. These three (3) manual valves are not required for operation by either the NSSS supplier, the A/E or station operating personnel and were installed as part of an earlier generation nuclear station design.

The safety evaluation for this change indicated that no unreviewed safety questions will result from this change. Furthermore, the evaluation concluded that the change will not:

1) Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report.

2) Create a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report.

3) Reduce the margin of safety as defined in the basis for any technical specification.

The intent of these valves was to provide an orifice bypass thereby creating additional flow if sufficient flow was not passed by the orifice. As determined by testing and current Westinghouse design, the orifice itself currently provides sufficient charging pump mini-flow.

Steam Generator Tube Inspection

Technical Specification Paragraph 4.4.5.5b requires that the complete results of any steam generator tube in-service inspection be included in the Annual Operating Report for the period in which this inspection was completed. Since an Annual Operating Report is no longer submitted, it was decided to submit the S/G eddy current inspection results in this report.

During the 1980 refueling outage the "C" Steam Generator was in-service inspected using an eddy-current examination technique. A total of one hundred and twenty-two (122) tubes in the upper four (4) wedge areas were inspected around the U-bend and down through the first support on the outlet side while an additional one hundred and ninety-five (195) tubes were inspected from the inlet through the U-bend. Since no reportable indications were found in any of the examined tubes, it was not necessary to institute any tube plugging.

Special Liquid Waste Temporary Operating Experiment

A temporary operating procedure completed during 1980 involved the processing of liquid waste utilizing an auxiliary liquid waste demineralizer and associated piping. As a result of the Solid Waste System being out of service and the Liquid Waste Evaporator and associated steam system not in operation, due to a plant modification to comply with the ASME Boiler and Pressure Vessel Code requirements, it became necessary to utilize an auxiliary demineralizer to process liquid waste. Since this procedure was only temporary in nature it was considered an experiment rather than a change and is being reported in the 10 CFR 50.59 report since the procedure is not described in the FSAR. This experiment was successful in that the plant was able to process sufficient liquid waste and reduce the existing volume without resulting in any plant problems.

The safety evaluation for this experiment stated that it did not involve a change to the plant technical specifications or constitute an unreviewed safety question. Reviews of 10 CFR 20 "Standards For Protection Against Radiation" and 10 CFR 190 "Environmental Radiation Protection Standards" concluded that compliance with these two standards had been achieved. This is based on the temporary demineralizer and piping being contained within the existing decontamination building and facilities, and all discharges resulting from its use will be accomplished using the normal Liquid Waste System.

Next to be reviewed as part of the safety evaluation were Regulatory Guide 1.21 and Standard Review Plan Section 11.5 "Process and Effluent Radiological Monitoring and Sample System." It was determined that this temporary Liquid Waste Clean-up evolution would not alter any existing monitoring and/or sampling systems.

The safety evaluation reviewed the system design and operation with respect to postulated accidents and the radiological consequences of unexpected and uncontrolled releases of radioactivity stored or transferred in a water system. It was concluded that since the demineralizer and piping are contained within the existing decontamination building facilities and all discharges will be conducted using the normal Liquid Waste System, no problems of this type would occur. Finally, the safety evaluation concluded that the probability of occurrence or the consequences of an accident or malfunction of equipment is not created or increased since a line break will be contained within the decontamination building facilities.

FSAR, Table 6.2.1

FSAR, Table 6.2.1 (attached) entitled "Piping Inspection" was modified to delete the random inspection of Class III (Q-3) piping. Random as previously identified in this table pertained to 20% inspection of all welds from an entire population of welds in a specific category. In instances where random inspections were deleted, 100% magnetic particle (MT) or 100% liquid penetrant (PT) inspections were substituted.

The safety evaluation for this change concluded that the safety of the plant is being maintained by virtue of the fact that 20% random radiography of girth butt welds and 20% MT or PT of fillets, sockets and welded branch connections is now being replaced by 100% MT or PT of all Q-3 welds. This is considered to be an upgrading of the NDE requirements and is consistant with the ASME Section III requirements implemented on BVPS-Unit No. 2.

TABLE 6.2-1

PIPING INSPECTION

	Class I (Q1)	Class II (Q2)	Class III (Q3)
Girth Butt Welds			
a. Radiography	100% required	100% required	Not required
b. Magnetic Particle	100% required	Not required	100% required
c. Liquid Penetrant	of (b) or (c)	Not required	of (b) or (c)
Longitudinal Butt Welds			
a. Radiography	100% required	100% required	100% required
b. Magnetic Particle	100% required	Not required	Not required
c. Liquid Penetrant	of (b) or (c)	Not required	Not required
Fillet and Socket Welds			
a. Radiography	Not required	Not required	Not required
b. Magnetic Particle	100% required	100% required	100% required
c. Liquid Penetrant	of (b) or (c)	of (b) or (c)	of (b) or (c)
Seal Welds			
a. Radiography	Not required	Not required	Not required
b. Magnetic Particle	100% required	100% required	Not required
c. Liquid Penetrant	of (b) or (c)	of (b) or (c)	Not required
Welded Branch Connections			
a. Radiography	100% required over 4 in. pipe size	100% required over 4 in. pipe size	Not required
b. Magnetic Particle	100% required	100% required on 4 in. and under	100% required
c. Liquid Penetrant	of (b) or (c)	of (b) or (c)	of (b) or (c)
Attachment Welds - Supports, Legs, Anchors and Guides			
a. Radiography	Not required	Not required	Not required
b. Magnetic Particle	100% required	100% required	Not required
c. Liquid Penetrant	of (b) or (c)	of (b) or (c)	Not required