Reliability Estimation for Check Valves and Other Components

by

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ABSTRACT

For years the nuclear industry has depended upon component operational reliability information compiled from reliability handbooks and other generic sources as well as private databases generated by recognized experts both within and outside the nuclear industry. Regrettably, these technical bases lacked the benefit of large-scale operational data and comprehensive data verification, and did not take into account the parameters and combinations of parameters that affect the determination of failure rates. This paper briefly examines the historic use of generic component reliability data, its sources, and its limitations. The concept of using a single failure rate for a particular component type is also examined. Particular emphasis is placed on check valves due to the information available on those components. The Appendix presents some of the results of the extensive analyses done by Oak Ridge National Laboratory (ORNL) on check valve performance.

BACKGROUND

Historically, an analyst attempting to quantify component reliability for the purpose of assessing the impact of component failure has been limited to a number of sources of published "classic" generic data, and the analyst's own collection of reference material/data (NUREG/CR-6186, 1994). These "classic" sources, including two NUREGs and an Institute of Electrical and Electronics Engineers (IEEE) Standard, have historically been used (combined with the analyst's own references) as input into those assessments.

NUREG-75/014, "The Reactor Safety Study," had as its goal a risk assessment as opposed to a reliability analysis (NUREG-75/014, 1980). Hence, order of magnitude accuracy was tolerated in its quantified results. The study used a range approach (a standard statistical and probabilistic modeling technique where a range of values is determined such that the "true" value has a high probability of lying within it) because of the inadequacy of the data available at the time. Failure rates were derived from reference handbooks, reports, operating experience, and nuclear power plant experience. Primary shortcomings of the estimates are the poor quality of data, which required broad ranges, and the modest number of total failures contained in the data (only 102 valve failures).

The Licensee Event Report (LER) data summary, NUREG/CR-1363, was based on five years of LERs pertaining to valves (NUREG/CR-1363, 1982). As with the Reactor Safety Study, the report assumed a constant failure rate for the valves over the total number of demand hours. The primary shortcomings of using LER data and associated operating history are that not all valve faults and maintenance/part replacement actions are recorded and available. In general, the effects of maintenance actions, programs, and retrofits are unknown.

The LER data summary report was limited to valves in engineered safety feature systems and primary relief and safety valves, which was the only data found of sufficient quality. Failures and command faults were assumed to be random events, and no common-cause events were considered separately. The operational demands for the valves included those related to testing. The test demands were assumed to be as specified in Section XI of the ASME Boiler and Pressure Vessel Code. Operational demands were added to test demands for safety and relief valves since only one test demand was assumed for them. The number of operational demands used was estimated.

IEEE Standard 500-1984 is another generic source of valve reliability estimates which was prepared to be used by either nuclear systems reliability analysts or design engineers (IEEE, 1983). In addition to design and test planning purposes, the standard was also intended to be used for Probabilistic Risk Assessment (PRA) studies.
Drawing primarily from nuclear and applicable industrial data obtained from NURECs, EPRI reports, and its own surveys, the IEEE Standard 500 represented the most rigorous attempt at the time (the standard was reaffirmed in 1991 with minor changes) to create a comprehensive and defendable source of reliability estimates for a wide variety of nuclear components.

The estimates provided in IEEE Standard 500 were developed using a feedback system for the communication of data and information among a group of experts. This process is an iterative one that uses expert opinion to consider both data and independent expert opinion to produce a product. Valve data was obtained through questionnaires and visits to nine nuclear units. Experts recorded maintenance records pertaining to a five year period. As in the earlier studies, the analysts probably did not have access to accurate operating time data, complete valve cycling data, and information pertaining to various part replacements and other preventive maintenance actions since complete records are generally not kept on these types of data. Assumptions were almost certainly made in these areas, although this detail is not provided in the standard itself.

SINGLE FAILURE RATE VS. A FAILURE RATE RANGE

Reliability analysts have been accustomed to using generic failure rates available from various sources. It has been generally recognized that failure rates vary according to the type of component. Failure rates have been made available for motor operated valves (MOVs), pumps, check valves, and other components. Occasionally, failure rates have even been broken down by failure mode (e.g., failure to open) and sometimes even by specific valve type, such as for a swing check valve. In general, however, no effort has been made to estimate failure rates for any component type considering such important parameters as service environment/application, failure mode, size, age, specific design features, manufacturer, maintenance practices, in-service testing practices, etc. Recent studies by ORNL (McElhaney, 1995a) have shown that some significant differences exist in relative failure rates for check valves according to, among other factors, their size, system of service, analysis period, and design features (e.g., specific valve type). For example, in 1991 check valves > 10 in. in size were 1.2 times more likely to fail than valves in any other size group. Check valves installed in HPCL, suppression pool support, RCIC, and diesel starting air systems were three to four times more likely to fail significantly than were valves installed in most other systems. This result was also found to be heavily influenced by other factors, such as valve design, operating conditions, and testing practices. For example, most of the valves in the diesel starting air system were small lift check valves that failed due to the presence of foreign material, and most of the failures occurring in the HPCL and suppression pool support systems involved stuck closed vacuum breaker valves. Three of the four systems were exclusive to BWRs, a result that could lead to false conclusions regarding BWR performance until it is recognized during further analysis that BWRs may simply have superior testing practices and are therefore detecting more failures.

The point is, there is no “correct” answer to a question such as “what is the failure rate for check valves?” The answer depends upon the scope of the question: the more specific the question (i.e., the greater the number of parameters specified), the more accurate the answer. A much more accurate estimation can be established, for instance, for the failure rate of “stuck open 10 in. Anchor Darling model 1234 swing check valves in the feedwater systems in Westinghouse pressurized water reactor (PWR) plants during 1990-1995,” than for “check valves in feedwater systems,” assuming that the population of valves in service during the period of interest (corresponding to the given set of parameters) and the number of failures is known. It should be clear, then, that perhaps the best that can be done is to establish a range of failure rate values based upon the nature of the data itself and the set of analysis parameters considered. The analyst may be able to determine a worst case and a best case scenario by reviewing the failure data available, calculating appropriate cross-tabulations with the known parameters, and using the valve population data to calculate failure rates. In some cases this range may be relatively small, while in others it may vary by orders of magnitude. Again, the width of this range would be dependent upon both the data and the set of constraints given. A sensitivity analysis might be performed for the end application to determine just what effect a given failure rate range might have on the analysis.

ORNL DATABASE AND CALCULATION OF ABSOLUTE FAILURE RATES

ORNL has established performance databases for check valves, MOVs, air-operated valves (AOVs), main steam isolation valves (MSIVs), and pumps. These databases were established by downloading, reviewing, screening, and characterizing thousands of nuclear component failure records from the Institute of Nuclear Power Operations’ (INPO’s) Nuclear Plant Reliability Data System (NPRDS) database. For check valves, the review and screening processes eliminated incipient failures, external leakage failures, and failures not involving internal parts degradation. (A “failure” for purposes of this study was defined as a degradation of one or more valve functions.) Failure records remaining for analysis were then characterized according to specific parameters inherent in the NPRDS database, such as system, nuclear steam supply system (NSSS) vendor, valve size, valve age, manufacturer, and plant age. Other parameters not included in NPRDS were also characterized by reviewing the failure narratives. These included failure mode, fail rate area, extent of degradation, general and specific detection methods, and normal system operating status. Many of the ORNL failure characterizations were reviewed and verified by cognizant plant personnel, resulting in an increased level of confidence in the data. In some cases, input from the plants actually resulted in changes to the failure narratives/characterizations. Databases for MOVs, AOVs, MSIVs, and pumps were developed in much the same way as for check valves. These component performance databases also included information on the installed component populations so that relative failure rates could be calculated. The Appendix provides a more detailed review of some of the results of the ORNL check valve studies.

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1 I.e., significant in terms of degradation of one or more component functions. Examples of significant failures would include those involving valves that were stuck open, stuck closed, or had loose or broken parts. Other failures which would not have resulted in loss of functionality would include, for example, leak test failures.
The primary benefits of the ORNL-developed component performance databases include their incorporation of many years of actual operational data, consistency of data characterization and development methodology, and reproducibility. Their intended use has been to enable the identification and trend of any significant correlations of failure rates with plant age, component age, or other parameters while considering the effects of the number of components in service during the analysis period (i.e., population effects). NUREGs developed using these databases have contained hundreds of charts and tables showing relationships among the studied parameters while also providing calculated relative failure rates.²

It must be understood, however, that although NPRDS provides reasonably accurate information on, as well as a count of, “failure” incidents, it does not provide data on, for example, valve operating time or number of valve actuations (demands), which are essential for the derivation of failure rates. The concept of component age is also uncertain in light of the type of maintenance performed - often amounting to comprehensive replacement of internal parts. These factors, along with differences in reporting practices among plants/utilities, make it difficult to utilize even the best available operational data to establish absolute failure rates. The concept of extent of degradation to the component must be addressed also; a failure of a valve to pass a leak rate test should not necessarily be given the same weight as a failure involving a detached disc or broken hinge pin in calculating the valve’s “failure rate.”

Again, the analyst is left with no choice except to establish a failure rate range given the best data and largest number of parameters available. Even with its limitations, the data derived from the NPRDS database through review, screening, and characterization is the best representation of years of nuclear operating experience available. A methodology for studying and assessing the effect that such factors as design, size, age, maintenance, and service conditions have on component performance must ideally be based on data such as that characterized from NPRDS by ORNL from nuclear power plant operations. Such a methodology might involve the following steps:

1. Devise a means of sorting component operational data to contrast the factors of interest. This may be a simple sort, or it may involve considerable research, depending on whether the essential data field(s) can be sorted and are complete, reliable, and/or even present in the database.

2. Perform a careful review of all possible factors that may affect performance to determine how many other variables can be eliminated in the data sort being performed. One approach may be to perform additional data sorts of “suspect” factors to see if they are indeed biasing the results. For instance, if one utility is known to maintain the pertinent valves in a significantly different manner, these valves might be excluded from consideration. [It must be kept in mind, however, that the larger the number of parameters reviewed (both for inclusion and elimination), the more complicated and time-consuming the analysis. The analyst must weigh the time and costs involved vs. the confidence level required for the results.]

3. Evaluate the results of the data showing the effects of the factors of interest and establish a failure rate range by identifying best and worst case performance scenarios.

4. Consider the cases of markedly contrasting data carefully. For example, are the differences in performance due to some obvious factor such as a particular design feature or abrasive system fluid? Assess the strength of the factor causing the contrasting data to gain insight into how effective the factor is in causing the observed data.

5. If possible, review the reliability estimates for the component based on historical data and generic sources utilized in past studies.

6. If necessary, modify the historic failure rate estimates based on results of 2 through 4. Make generous use of expert opinion in performing the modification and subsequent peer review.

An example of the calculation of absolute failure rates using NPRDS data and a relatively specific set of analysis parameters can be found in the analysis of accumulator discharge check valve performance (McElhaney, 1995b). This study reviewed the operating history of 231 PWR accumulator discharge check valves during the period 1984-1992. During this period, only 18 failures were recorded in the ORNL check valve performance database. Only 7 of these failures were characterized as significant in terms of degradation to the valve itself. Although NPRDS does not contain specific information on valve operating time, the number of valve-years of service of the valves included in the study could be estimated using the available component in-service data. The number of accumulator discharge check valve failures was then divided by the number of valve-years of service for this component during the study period to arrive at an overall failure rate in terms of valve failures per valve-years.

For example, for accumulator discharge check valves from 1984 through 1992:
Overall failure rate for all ORNL-characterized failures
= 18 failures / 1894 valve-years = 0.0095/year

Overall failure rate for all ORNL-characterized significant failures
= 7 failures / 1894 valve-years = 0.0037/year

Overall failure rate for valves to accomplish their accident mitigation function (failure to open)
= 0 failures / 1894 valve-years = 0/year.

It could then be concluded that even this relatively specific set of parameters for valves with an extremely good performance record represents a range of failure rate values. (The Appendix contains a more detailed discussion of the analysis of this particular check valve application.)

CONCLUSIONS

The generic failure rate estimates given for nuclear systems components available from certain classic generic sources should be reviewed carefully in consideration of additional, more specific data since they may not necessarily represent realistic industry operating experience. Most estimates do not consider such important parameters as component design features or operating environment. The resulting single generic values given, for example for “check valve failure rates” can therefore be inaccurate and/or misleading.

A more realistic approach to component failure rate estimation should involve the utilization of as much actual operating data as possible. More accurate values can be obtained by utilizing industry performance databases such as those developed at ORNL and by providing as specific an analysis scope as possible. In other words, the greater the number of component analysis parameters specified, the more accurate the failure rate estimation. Even then, the accuracy of the estimation depends upon the accuracy and completeness of the performance data available and the methodology used to estimate the component reliability. In most cases, the best estimate of component performance can probably only be established by a bounding range of values for the best and worst case performance scenarios. In the case of utilization of these estimates in current estimates of failure impacts, a sensitivity analysis might be in order to determine the relative contribution of component failure rate vs. risk importance as factors in determining the impacts. This tends to take on more meaning today as the nuclear industry heads toward “risk-based” inservice testing intervals. If the failure rates used to develop risk change as the testing intervals increase, then more data will be needed so that new risk values can be calculated. A feedback loop will become necessary to ensure overall confidence in the analyses.

REFERENCES


FAILURE ANALYSIS AND RESULTS

Original ORNL Analysis - 1984–1990 Data

The original analysis performed by ORNL consisted of a review of nearly 5000 check valve failure records from the Institute of Nuclear Power Operations' (INPO's) Nuclear Plant Reliability Data System (NPRDS) database entered through May 1990 (Casada and Todd, 1993). Preliminary manual review of the failure narratives resulted in elimination of those failures occurring prior to 1984 (when significant improvements in NPRDS reporting requirements were implemented), as well as the exclusion of those events not involving valve internal parts damage (e.g., external leakage), valves incorrectly coded as check valves, nonfailures, and minor seat leakage. The resulting analysis database consisted of 1227 failure records occurring during the time period 1984 through 1990.

Failure records remaining for analysis were then characterized according to specific parameters inherent in the NPRDS database, such as system, nuclear steam supply system (NSSS) vendor, valve size, valve age, manufacturer, and plant age. Other parameters not included in the database were also characterized by reviewing the failure narratives. These included failure mode, failure area, extent of degradation, general and specific detection methods, and normal system operating status. The ORNL analysis focused on failures involving degradation of valve internal parts. A "failure" for purposes of this study was defined as a degradation of one or more valve functions (e.g., failure to open, failure to close, loose/broken parts).

After the failure records were manually reviewed, characterized, and recorded in database format, exhaustive data manipulations and cross-correlations were performed. Calculations were performed using data normalized according to the population of check valves existing during the analysis period to enable the establishment of relative failure rates. The Nuclear Industry Check Valve group (NIC) also provided a forum for utility review of much of the characterized data through actual site visits and interviews, further substantiating the validity of the analysis.

Analysis Updates

Desiring to gather additional information and to identify any performance trends, the NRC requested ORNL to update the original failure analysis for two additional study periods, 1991 and 1992. Analysis for 1991 has been completed, while that for 1992 is currently in progress. A somewhat augmented approach was undertaken for the 1991 study, building from experience gained from the 1984–1990 analysis. An important difference in the two studies was the incorporation of specific valve type data for 1991 (e.g., swing, lift/piston, tilting-disc check valves), since this parameter was recognized as being perhaps the most significant in terms of failure correlation (McElhaney, 1995a).

Another notable difference in the analyses for 1984–1990 and for 1991 was the filtering process used to exclude valves from further analysis. While both studies excluded failure reports involving no internals degradation, nonfailures, and noncheck valves, the 1991 (and subsequent) analysis included most failures involving seat leakage. As a result, more "moderate" type failures were included in the 1991 study. Although this approach somewhat invalidates direct comparison between the two periods, it was felt that a better representation was gained of actual check valve performance. Comparison of data based on "significant" failures for the two periods is considered to be valid.

The first data available relative to specific valve type for 1991 resulted in a failure distribution of 35% lift/piston check, 33% swing check, 12% unknown valve type, and the remainder composed of tilting-disc, duo/double disc, stop, in-line, and other. (It was not possible, however, to establish relative failure rates, since information does not yet exist regarding the population distribution by specific valve type.) Insight gained from this data, however, includes correlations such as valve type vs. failure mode and failure cause.

ACCUMULATOR DISCHARGE CHECK VALVES

A detailed analysis was also undertaken by ORNL to evaluate check valves in an application that had been reported to have experienced good performance and reliability (McElhaney, 1995b). The purpose of the analysis was to examine the potential for relaxation of current requirements for in-service testing for check valves in this particular application based on industry performance history. Accordingly, 231 PWR safety injection accumulator tank discharge check valves were identified and their performance history reviewed from 1984 through 1992. During this time period, only 18 failures involving these valves were recorded in the ORNL-characterized database derived from NPRDS. Of these, only seven failures were characterized as being significant in terms of extent of degradation (to the valve), and no failures were reported that would have affected any valve’s ability to perform its forward flow function in the event of its actuation during an accident.

Methodology

In an effort to more accurately assess the performance of check valves in this specific application, a characterization of information...
available from NPRDS was performed. Since the engineering record associated with each component included in the NPRDS database does not list the component’s specific application, it was first necessary to review plant Final Safety Analysis Reports (FSARs) to determine which valves (according to location and component identifier) were actually accumulator discharge check valves. FSARs from 39 randomly chosen PWR units were reviewed, resulting in the identification of 231 check valves serving an accumulator discharge function. Of the 231 valves identified, 142 were at Westinghouse plants, 61 were at CE plants, and 28 were at B&W plants.

After identification of the 231 accumulator discharge check valves, a review of their failure history was initiated. Using the ORNL check valve database, 18 failures of accumulator discharge valves were identified. The criteria for failure in this evaluation was the same as that employed in the previous studies; i.e., a “failure” was defined as a degradation of one or more valve functions. Failures were again evaluated in terms of their effects on the valves themselves, not in terms of any resulting plant or system effects.

![Overall failure rates](image)

Overall failure rates
The overall failure rates for accumulator discharge check valves are presented in Fig. A1. Overall failure rates are presented for all failures and for significant failures only (i.e., failures characterized as being significant in terms of extent of degradation of the valve). Failure rates are presented for all check valve failures and for accumulator discharge check valves only. These rates were calculated as follows, using failure data from NPRDS as filtered and characterized by ORNL (Note that these failure rates were derived for comparison purposes only. Their magnitude is affected by many factors, including NPRDS reporting practices, ORNL filtering methodology, and sample size for the valves included in the accumulator discharge study. They should not be considered for use in probabilistic or risk-based applications without further evaluation):

Overall failure rate for all ORNL-characterized check valve failures (1984-1992) = No. failures / No. valve-years
= 2065 failures / 155,498 valve-years = 0.0133/year

Overall failure rate for all ORNL-characterized significant failures
= 949 failures / 155,498 valve-years = 0.0061/year

Similarly, for accumulator discharge valves in the study,

Overall failure rate for all ORNL-characterized failures
= 18 failures / 1894 valve-years = 0.0095/year

Figure A1 indicates that the valves in the accumulator discharge check valve study exhibit failure rates lower than both the ORNL-characterized check valve failure population for all failures (both moderate and significant in terms of extent of degradation) and for significant failures only.
Conclusions

Of the 231 accumulator discharge check valves identified, only 18 failures were recorded in the ORNL database from 1984-1992, inclusive. Of these, only seven failures were characterized as being significant in terms of their extent of degradation (to the component itself). Also, of the seven significant failures, three were attributable at least in part to human error. The other four significant failures involved either broken or missing parts. None of the reported failures should have had any effect on the forward flow functionality of the valves, had they been called upon in the event of an accident. This result, in addition to the absence of any stuck closed failures, indicates that in terms of ability to perform their required accident functions, accumulator discharge check valves have been extremely reliable.

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